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Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities

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Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities

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I. Introduction

This guide, Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities (hereafter referred to as “Safety Assessment Guide”) defines the bases on which the adequacy of the safety assessment proposed in the applications for permission to install (or modify) light water nuclear power reactors, which will be referred to as light water reactors (hereafter referred to as “LWRs”) is judged in the process of licensing review. The licensing review of LWRs necessitates a verification that the safety assessment in the applications fully meets the requirements specified in this guide.

The safety assessment shall cover (a) the assessment related to safety design policy of the nuclear reactor facility, which will be referred to as safety design assessment, and (b) the assessment related to isolation of the reactor from the public as siting condition, which will be referred to as siting assessment.

Safety Assessment Guide was originally instituted in September 1978 by the then Atomic Energy Commission for licensing review against the applications for permission to install LWRs. More than 10 years have passed since then, and in the meantime considerable progress in LWR technology has been made and useful experience and findings have been accumulated. This has made it necessary to update "Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities" (hereinafter referred to as “Safety Design Guide”). At the same time, "Regulatory Guide for Reviewing Classification of Importance of Safety Function for Light Water Nuclear Power Reactor Facilities" (hereinafter referred to as “Importance Classification Guide”) has been newly instituted. Accordingly, Safety Assessment Guide has been subjected to thorough review and revised to coordinate with new Safety Design Guide and Importance Classification Guide as well as to render its contents more specific and systematic.

While this guide is primarily intended to cover LWRs under present employment, it is believed that the fundamental concept in this guide will be helpful as well for licensing review of other types of nuclear reactor facilities.

It should be recognized that partial incongruities of the contents of an application to this guide do not necessarily constitute disapproval of the application if the deviation is based on justifiable reasons. This guide should be subject to revision as required in the light of future experience and design improvement.

Ⅱ. Safety Design Assessment

1. Objective of Safety Design Assessment

The adequacy of the basic safety design concept proposed for the nuclear power reactor facility is reviewed in accordance with Safety Design Guide. It is required in Safety Design Guide that certain structures, systems and components in the nuclear reactor facility achieve their necessary functions from safety point of view, not only during normal operation but also in abnormal conditions.

In proving the validity of the basic safety design policy for the nuclear power reactor facility, it is therefore essential to perform analytical assessment concerning abnormal conditions, i.e., "anticipated operational occurrences" and "accidents". Described below are the events to be postulated, criteria for judgment and matters to be taken into consideration in analysis for safety design assessment.

2. Scope of Assessment

2.1. Anticipated Operational Occurrences

Assessment shall be performed as to the events during reactor operation which may lead to such conditions as deviate from normal operation and are expected to occur once or several times during the operating life of the nuclear reactor facility by single component failures, single component malfunctions or single misoperations or by disturbances with a similar probability of occurrence.

2.2. Accidents

Assessment shall be performed as to the events beyond anticipated operational occurrences which have quite small probabilities of occurrence and yet may potentially lead to the release of radioactive materials from the nuclear reactor facility and thus have to be postulated in the light of the necessity of confirming the safety of the nuclear reactor facility.

3. Selection of Events for Assessment

Based on the objective and scope of safety design assessment described above, the events for assessment shall be adequately determined with respect to anticipated operational occurrences and accidents in the nuclear reactor facility.

3.1 Anticipated Operational Occurrences

Based on the requirement in section 2.1., representative events shall be selected for assessment from among the events, which may potentially lead to excessive damage to the core or the reactor coolant pressure boundary if the nuclear reactor facility is left uncontrolled, from the viewpoint of confirming the adequacy of the designed functions of structures, systems and components belonging in general to abnormality mitigation systems, or simply referred to as mitigation systems (MSs), such as the safety protection system and the reactor shutdown system.

The events for assessment shall cover the following abnormal states. If there may be two in general to abnormality mitigation systems, or simply referred to as mitigation systems (MSs), such as the safety protection system and the reactor shutdown system. The events for assessment shall cover the following abnormal states. If there may be two or more similar events, the severest event in the light of the criteria specified in section 4 can be selected as a representative event of them.

(1) Abnormal change in reactivity or power distribution in the core
(2) Abnormal change in heat generation or heat removal in the core
(3) Abnormal change in reactor coolant pressure or reactor coolant inventory
(4) Other events necessary for assessment depending on the design of the nuclear reactor Facility

3.2. Accidents

Based on the requirement in section 2.2, representative events shall be selected for assessment from among the events, which may potentially lead to undue exposure of the off-site public by the radioactive materials released from the nuclear reactor facility, from the viewpoint of confirming the adequacy of the designed functions of structures, systems and components belonging in general to MSs such as the engineered safety features. The events for assessment shall cover the following abnormal states. If there are two or more similar events, the severest event in the light of the criteria specified in section 4 can be selected as a representative event of them.

(1) Loss of reactor coolant or considerable change in core cooling
(2) Abnormal reactivity insertion or rapid change in reactor power
(3) Abnormal release of radioactive materials to the environment
(4) Abnormal change in pressure, atmosphere, etc. in the reactor containment
(5) Other events necessary for assessment depending on the design of the nuclear reactor facility

4. Criteria

4.1 Anticipated Operational Occurrences

It shall be verified that the nuclear reactor facility is designed such that a postulated event does not result in damage to the core and that the event can be put under control with the condition which allows the resumption of normal operation.

The criteria for this verification are as 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.

4.2. Accidents

It shall be verified that the nuclear reactor facility is designed such that a postulated event does not lead to melting or considerable damage of the core, that the event does not cause, in its process, a secondary damage which would lead to another abnormal condition, and that the function of the barriers against the release of radioactive materials in the event is adequate. The criteria for this verification are as specified below.

(1) The core shall not be damaged considerably, and adequate coolable state 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 radiological risk to the off-site public shall be acceptably low.

4.3 Principles for Application of Criterion

In case multiple criteria are applied to one event, the parameters for analyses shall in general be specified such that each set of parameters will bring the severest result with respect to each criterion.

A representative analysis, however, can be accepted for such an event with the parameters corresponding to the severest criterion, if it is evident that the results of the analyses are not significantly affected by the variation in parameters or the rest of the criteria are satisfied by the representative analysis.

5. Considerations in Analysis

5.1 Scope for Analysis

The initial conditions for the analysis of a postulated event shall be specified such that the severest result can be obtained with respect to the applied criteria, taking into consideration the whole ranges of normal operation and operating period of the nuclear reactor facility including long term physical change with burnup in the core during various cycles and with refueling and expected change in operational modes. The analysis shall in general cover the time range up to the point where the event converges and it can be reasonably inferred that the reactor would reach a cold shutdown state in due course.

5.2 Assumptions on Safety Functions

(1) Of safety functions designed to cope with postulated events, those which are allowed to be taken into account in the analysis shall in general be limited to safety functions to be performed by structures, systems and components belonging to MS-1 and MS-2 specified in Importance Classification Guide. Safety functions of structures, systems and components belonging to MS-3 can be taken into account in the analysis only if the expectation on those functions is proved to be justifiable.

(2) In the analysis, a single failure of a component or within a system which is designed to cope with an accident shall be assumed in addition to postulating an initiating event for assessment. A single failure shall be selected such that it leads to the severest consequence with respect to each of the fundamental safety functions of reactor shutdown, core cooling and radioactivity confinement. A single failure shall be assumed of an active component during a short term, and of an active component or a passive component during a long term after the event initiation. The failure of a component to be operated continuously before and after the event initiation may not need to be assumed in general. The failure of a passive component may not need to be assumed in the analysis if the system including the said component with failure is designed to be capable of fulfilling its required safety functions, the failure can be removed or repaired within a time not impairing the safety, or the probability of the occurrence of the failure is sufficiently small.

(3) The analysis shall take into account appropriate amount of time for manual operations if such actions by operators are expected in order to cope with the postulated event.

(4) If functions of the safety protection system are expected in the analysis, the kinds of signals to actuate it and the timings at which those signals are to be generated shall be defined. The same requirement shall also be applied to other systems if their expected performance affects the result of the analysis.

(5) The analysis of an accident shall take into account unavailability of off-site power if functions of the engineered safety features are expected.

(6) If the effect of a reactor scram is expected in the analysis, the kinds of signals to initiate the scram shall be defined, and appropriate delay times for effective scram initiation shall be considered. In addition, the shutdown effect shall be evaluated on the assumption that a control rod (or a group of control rods connected to a common drive mechanism) with the maximum reactivity worth in the postulated condition is held at the fully withdrawn position.

5.3 Calculation Programs, Models and Parameters Used for Analysis

The calculation programs, etc. used for the analysis of a postulated event shall be verified with respect to their applicability. The models and parameters for the analysis shall be specified such that they give a severe result to a reasonable extent in view of the objective of the analysis. If there can be uncertain factors in specifying the parameters, appropriate safety margins shall be taken into account.

Ⅲ. Siting Assessment

1. Objective of Siting Assessment

The suitability of the reactor siting conditions is reviewed in accordance with "Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and Application Criteria" (hereinafter referred to as “Reactor Siting Guide”). It is required in Reactor Siting Guide that an exclusion area, a low population zone and distances to densely populated regions surrounding the reactor be adequately established so that the evaluated radiation doses to the public by a postulated "major accident" or "hypothetical accident" may be lower than the specified criteria. In proving the suitability of the reactor siting conditions, it is therefore essential to perform assessment concerning major accidents and hypothetical accidents. Described below are the events to be postulated, criteria for judgment and matters to be taken into consideration in analysis for siting assessment.

2. Scope of Assessment

Assessment shall be performed as to the events which have to be postulated in the light of necessity of confirming the suitability of the reactor siting conditions based on Reactor Siting Guide.

3. Selection of Events for Assessment

3.1 Major Accidents

Based on the requirement in section 2 above, such events as may have the potential to enlarge the release of radioactive materials shall be selected out of the accidents treated in section 3.2 of chapter II "Safety Design Assessment", and the largest amounts of radioactive materials to be released technically possible shall be assumed. The events to be postulated shall represent both types of accidents , one where radioactive materials are released into the reactor containment and the other where radioactive materials are directly released at the outside of the reactor containment.

3.2 Hypothetical Accidents

Based on the requirement in section 2 above, the events identical with the major accidents but with the assumption of larger amounts of radioactive materials to be released shall be postulated.

4. Criteria

The results of the assessment shall satisfy the criteria specified in Reactor Siting Guide.

5. Considerations in Analysis

The analyses of major accidents and hypothetical accidents shall be performed in the light of the intent of Reactor Siting Guide.

Commentary

I. Purpose in Revision of This Guide

Safety Assessment Guide was originally instituted on September 29, 1978 by the then Atomic Energy Commission for licensing review against the applications for permission to install LWRs, and had since been applied in the process of licensing review to judge the adequacy of the assessment related to safety design policy of the nuclear reactor facility and the assessment related to isolation of the reactor from the public as siting condition in the light of Safety Design Guide and Reactor Siting Guide. Later, the guide was modified in part on March 27, 1989 along with other licensing review guides to reflect the recommendations of the International Commission on Radiological Protection (ICRP) issued in 1977. (This modified version will be referred to as the former guide hereinafter.)

The requirements in licensing review guides are continually reexamined based on technological progress and accumulated experience and findings. Safety Design Guide is thus being revised of late, and Importance Classification Guide, a new guide concerning the classification of safety function importance in nuclear reactor facilities, is being established. Safety Assessment Guide has already been in use for more than 10 years since its first institution, and it has been recognized that the guide should be revised to reflect the technological progress and experience gained in the meantime and to coordinate with the new version of Safety Design Guide and newly established Importance Classification Guide.

In the present revision of Safety Assessment Guide, the composition of the entire text is rearranged with separate sections pertaining to safety design assessment and siting assessment. Besides, Appendix I describing typical events to be postulated for assessment along with respective analytical conditions to be taken into consideration and Appendix II describing recommended methods for evaluating radiation doses are newly added.

Furthermore, Commentary on Appendix describing the matters to be taken into consideration for the analysis of postulated events shown in Appendix I is newly provided.

This revised version of Safety Assessment Guide reflects what is believed to be the standard design concepts for current LWRs in Japan. It is expected that the improvement in the design of LWRs will continue along with the accumulation of experience and the progress in safety research and analytical techniques.

It should therefore be recognized that partial incongruities of the contents of an application for permission to install an LWR with new design to this guide do not necessarily lead to disapproval of the application and need careful examination if the deviation from the guide is based on technological progress and new findings.

This guide itself should be subject to revision as required in the light of future experience and technological progress. The appendices and the commentary on appendix of this guide are edited with the date of resolution to facilitate supplementation or revision by adopting new findings as required.

II. Safety Design Assessment

1. Safety Design Assessment

Section II "safety Design Assessment" in the text of this guide defines the essential requirements necessary for the assessment related to safety design policy of the nuclear reactor facility, i.e. safety design assessment, in the light of Safety Design Guide.

Safety Design Guide specifies various requirements for structures, systems and components in nuclear reactor facilities from the viewpoint of ensuring safety, and part of the requirements are intended for ensuring necessary safety functions during abnormal conditions, i.e. anticipated operational occurrences and accidents.

In accordance with those requirements, Section II "Safety Design Assessment" specifies the events to be postulated, criteria for judgment and matters to be taken into consideration in analysis necessary for safety design assessment.

2. Scope of Assessment and Selection of Events for Assessment

The "scope of assessment" in the safety design assessment shall cover anticipated operational occurrences and accidents. To encompass those conditions by the analysis of a limited number of events, proper selection of the events for assessment shall be made. The anticipated operational occurrences and accidents discussed herein are limited to internal events whose causes arise within nuclear reactor facilities. As to natural phenomena or external man-induced events, the adequacy of design considerations against them is reviewed separately based on Safety Design Guide and others. While internal events take various forms, most of them are caused by failures, damages or misoperations of systems or components belonging to abnormality prevention systems (PS s) defined in Importance Classification Guide. Out of them, the events extracted for consideration in safety design of nuclear reactor facilities and in its assessment are referred to as design basis events (DBEs).

When the occurrence of a certain DBE is postulated, the process of its evolution varies with the operating status of individual structures, systems and components in the nuclear reactor facility involved. A combination of one DBE with the related operating status of systems and components mainly belonging to MSs, the status of power supply, etc. is the "event for assessment" in the safety design assessment. Of the events for assessment, those categorized as anticipated operational occurrences shall cover such events as are expected to occur once or several times during the operating life of the nuclear reactor facility (their main causes may be attributed to loss of off-site power, a single failure or malfunction of an active component, or a single misoperation). The "single failure", etc. referred to here (including single malfunction and single misoperation) include multiple secondary failures, etc. due to a single cause. Those categorized as accidents shall cover such events as may potentially lead to severer consequences on the nuclear reactor facility and the public, though the probability of occurrence may be smaller, and have to be postulated from the viewpoint of examining the adequacy of the safety design. In this guide, the events for assessment categorized as anticipated operational occurrences and accidents are classified into three kinds and four by the nature of abnormal states, respectively, and typical events to be postulated representing each abnormal state are indicated in Appendix I along with their basic analytical conditions. There can be additional appropriate events for assessment depending on specific design of a nuclear reactor facility, and the selection of such events shall be carefully examined.

Some postulated events may each be related to two or more kinds of abnormal states classified as above. In that case, analytical conditions shall be adequately specified with due consideration to the purpose of each assessment. If there may be two or more similar events for an abnormal state, they can be represented by one event that gives the severest result.

3. Criteria

The "criteria" to be followed in the safety design assessment are the standards by which the adequacy of the safety design of the nuclear reactor facility against anticipated operational occurrences and accidents is judged from the results of the analyses of the "events for assessment" mentioned above. The following provides the basic idea behind these criteria.

The criteria for anticipated operational occurrences basically reflect the requirement that the nuclear reactor facility is designed such that normal operation can be resumed without any significant repairs but the restoration of the failed parts that have formed the cause of the event, and those are basically the same as the criteria under the former guide. As to criterion (3), detailed requirements are specified in "Regulatory Guide for Evaluating Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities" (hereinafter referred to as “Reactivity Insertion Events Evaluation Guide”), The criteria for accidents basically reflect the requirement that the nuclear reactor facility is designed such that the event does not lead to melting or considerable damage of the core and that the release of radioactive materials to the environment can be limited as low as acceptable. The requirement that the event does not form a cause of other secondary abnormal conditions before it is put under control is also taken into consideration.

Criteria (1) through (5) are based on this basic concept. The requirement in criterion (1) saying that adequate coolable state of the core shall be maintained implies that the core shall keep such a geometry as allows quantitative, or at least semi-quantitative, assessment of the heat removal from the core, i.e. "coolable geometry". The practical determination of conformance to this criterion shall in general be subject to the following requirements specified in "Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Power Reactors" (hereinafter referred to as ECCS Performance Evaluation Guide)

(a) The calculated maximum fuel cladding temperature shall not exceed l,200°C.

(b) The calculated stoichiometric amount of oxidation of the fuel cladding shall not exceed 15% of the cladding thickness before significant oxidation.

It should be recognized that other appropriate requirements can be substituted for the above, depending on the reactor coolant pressure, duration of high temperature of the fuel and other factors, if the conformance to such requirements evidently ensures the prevention of considerable core damage and the maintenance of adequate core cooling.

As to criterion (2), detailed requirements are specified in Reactivity Insertion Events Evaluation Guide. The radiological risk mentioned in criterion (5) should be determined based on the dose equivalent resulting from an accident and the probability of occurrence of the accident. According to the ICRP recommendations, the principal limit of annual effective dose equivalent for a member of the public is 1 mSv, but the use of a subsidiary dose limit of 5 mSv in a year for some years is also accepted unless the average annual effective dose equivalent over a lifetime exceeds 1 mSv. This is a concept concerning normal radiation exposure. This concept is applied to the case of an accident in this guide, and the risk can be judged as acceptably low if the evaluated effective dose equivalent for a member of the public as a result of an accident does not exceed 5 mSv. As for an accident with an extremely low probability of occurrence, the risk may be considered to be acceptably small even if the evaluated effective dose equivalent exceeds 5 mSv to a certain extent.

A technical review was made about radionuclides that should be taken into account in the assessment of dose equivalents to the public. As a result it was found that, except for iodine and rare gases, no other radionuclides released into open air added significant contributions. It is therefore required that the sum of the effective dose equivalent incurred from intake of iodine and the effective dose equivalent resulting from external exposure by rare gases, in general, be calculated in evaluating the effective dose equivalent induced by the radionuclides released into open air. In addition, the effective dose equivalent resulting from external exposure by direct gamma rays and skyshine gamma rays from radionuclides contained in buildings within the nuclear reactor facility shall be properly evaluated.

4. Considerations in Analysis

4.1 Scope for Analysis

The analyses of anticipated operational occurrences and accidents in the safety design assessment are to demonstrate the adequacy of the safety design policy and shall cover all abnormal events that can take place through the whole ranges of normal operation and operating period.

The parameters for analyses shall be specified to meet this purpose. In addition, analytical results should indicate that the postulated event can converge to a safe state satisfying acceptance criteria as a bounding case that represents other similar events. The analyses should therefore be performed up to the point in time where the event converges and it can be reasonably inferred that the reactor would reach a cold shutdown state in due course. There can be, however, exceptions depending on the nature of events; in the case of a "loss of reactor coolant", for example, criterion (4) specified in ECCS Performance Evaluation Guide should be applied.

4.2 Assumptions on Safety Functions

(1) The structures, systems and components with safety functions are divided into three classes by relative importance of their safety function as specified in Importance Classification Guide, and accordingly, those having abnormality mitigation functions are divided into MS-1, MS-2 and MS-3. The functions to put abnormal conditions, when they occurred, under control and mitigate their consequences shall have reliability as high as required according to their importance. In view of this, it is considered to be necessary that nuclear reactor facilities should in general be capable of coping with accidents without depending on the mitigation functions of MS-3 whose expected reliability is as high as that of ordinary industrial facilities. Therefore, in the text of this guide, mitigation functions of structures, systems and components belonging to MS-1 and MS-2 only are in general accepted as those which deserve consideration in the analyses of accidents. Mitigation functions of those belonging to MS-3, however, can be taken into consideration in the analyses, provided that their reliability is sufficiently high. In other words, mitigation functions of structures, systems and components belonging to MS-3 must have reliability as high as that of MS-1 and MS-2 if they are to be expected in the analyses of accidents.

Likewise, in this guide, mitigation functions of structures, systems and components belonging to MS-1 and MS-2 only are in general accepted as those which deserve consideration in the analyses of anticipated operational occurrences. Mitigation functions of those belonging to MS-3, however, can be taken into consideration in the analyses, provided that their reliability is reasonably high. Examples of practical application are provided in Appendix I and Commentary on Appendix.

(2) It is required in Safety Design Guide that systems with safety functions of especially high importance be capable of fulfilling their safety functions even assuming a single failure of any of the components that comprise the systems. The systems to which this requirement is applied are specified in Importance Classification Guide with an abnormal condition and includes multiple failures due to secondary causes. The idea of single failure shown in those guides is to apply the assumption of a single (sentences missing). The single failure referred to here means the loss of intended safety functions of a component necessary for coping failure to specific systems; but it is distinct from a failure of a component as a cause of an abnormal condition.

In the present revision of this guide, the basic concept of the application of single failure remains unchanged. A single failure shall be assumed to occur in each combination of systems or components that are necessary for coping with an accident as MSs with respect to each of the fundamental safety functions of reactor shutdown, core cooling and radioactivity confinement. In the case of a loss of reactor coolant, for example, the safety function of core cooling is achieved by a proper combination of the emergency core cooling system (ECCS) to inject cooling water, the safety protection system to requires the assumption of a single failure, which brings about the severest analytical actuate the ECCS, electric systems to supply power for the ECCS, systems for cooling associated components and transporting heat to an ultimate heat sink, etc. This guide result, for such a combination of systems or components for performing a safety function as above.

It is thus a basic rule in this guide that the assumption of a single failure shall be applied to every system or component, belonging to any class of MS, which is expected to cope with an accident by performing any of the aforementioned fundamental safety functions. The systems and components to which the assumption of a single failure is applied must include not only competent systems defined in Importance Classification Guide but also the supporting systems that are directly needed by each competent system for fulfilling its safety functions. The failure of a component which would keep performing its functions from before the occurrence of the event and throughout the process of the event, or what is called an "on-duty" component, may not need to be assumed.

(3) It is a basic requirement that systems and components with safety functions should be designed to be capable of performing their necessary functions without depending on operator actions immediately after the onset of an abnormal condition. If operator actions are needed, sufficient time and adequate information shall be available so that the operator may be able to properly judge the situation and take necessary actions with a high degree of confidence. The analysis shall take into account a time allowance of at least 1 0 minutes for the start of necessary operator actions after adequate information becomes available for proper judgment.

(4) If functions of the safety protection system are expected in the analysis, the actuation signals for it shall be characterized including the timings of signal generation. The actuation signals shall be adequately selected considering the whole events that are encompassed by the postulated event. These requirements shall also be applied to other systems if the characterization of their actuation signals affects the result of the analysis. Safety Design Guide and Importance Classification Guide specify the systems which should be supplied with electric power by emergency on-site power systems.

(5) It shall therefore be demonstrated that those systems are designed to be capable of performing their functions even in case of loss of off-site power. Especially, the analysis involving the assessment of the performance of the engineered safety features shall satisfy this requirement. It should be noted, however, that, depending on the nature of the postulated event, the assumption of availability or unavailability of off-site power may affect the severity of the consequence. Consideration shall therefore be given to availability of off-site power as well in determining the parameters for the analysis of an accident so that the severest result may be obtained.

(6) If a reactor scram is expected in the analysis, the shutdown effect shall be evaluated on the assumption that a control rod (or a group of control rods connected to a common drive mechanism) with the maximum reactivity worth is held at the fully withdrawn position; or a "stuck rod margin". This is a design margin for the control rod-dependent shutdown system, and is not to assume a failure or other causes which render the control rod inoperative. Thus, it is different from the assumption of a single failure explained

Ⅲ. Siting Assessment

1. Siting Assessment

Section Ⅲ. "Siting Assessment" in the text of this guide defines the essential requirements necessary for the assessment related to isolation of the reactor from the public as siting condition, i.e. siting assessment, in the light of Reactor Siting Guide. Reactor Siting Guide requires that the reactor be adequately isolated from the public so that the evaluated radiation doses to the public by a postulated major accident or hypothetical accident may be lower than the specified criteria. In accordance with this requirement, SectionⅢ. "Siting Assessment" specifies the events to be postulated, criteria for judgment and matters to be taken into consideration in analysis necessary for siting assessment.

2. Scope of Assessment and Selection of Events for Assessment

The "scope of assessment" in the siting assessment shall cover major accidents and hypothetical accidents. The objective of postulating major accidents and hypothetical accidents is to confirm that the reactor involved is isolated from the public at a proper distance. The minimum distance necessary for isolation depends on the basic configuration, reactor power, safety measures including engineered safety features and other characteristics of the reactor. This point shall be taken into consideration in specifying major accidents and hypothetical accidents above.

In specifying hypothetical accidents, for example, if assumptions are made by neglecting the functions of all multiple barriers against fission products in the core, the distance necessary for isolation would be determined in effect only from the reactor power, and the effects of other important factors would be ignored.

Such assumptions may not be appropriate to the purpose of judging the minimum distance necessary for reactor isolation and thus are not required as essential conditions in Reactor Siting Guide.

In view of the above and in the light of current LWR designs (basic configurations, safety measures and other characteristics), major accidents shall cover two modes of accidents which may have the potential to enlarge the release of radioactive materials: one where radioactive materials are released into the reactor containment, and the other where radioactive materials are directly released at the outside of the reactor containment. With respect to each mode of major accidents, the largest amounts of radioactive materials to be released technically possible shall be assumed from the viewpoint of demonstrating the reactor isolation from the public, taking into consideration the consequences of the accidents treated in section 3.2 of chapter Ⅲ."Safety Design Assessment".

As for hypothetical accidents, the events identical with the major accidents shall be postulated, but larger amounts of radioactive materials to be released shall be assumed from the engineering point of view.

(Appendix 1) (August 30, 1990)

Concrete events to be evaluated based on II. 3 and III. 3 in the guidelines, concrete conditions to be referred to in the analysis and assessment of these events, and judgment criteria are shown below. This appendix should be supplemented as and when necessary in response to improvements in design, accumulation of experiences, etc.

I. Safety design assessment

1. Concrete events to be evaluated

Concrete events to be evaluated concerning abnormal transient changes during operation and accidents are as shown below. Note that it is necessary to add proper events to the following for assessment in light of the objectives of the assessment, depending on the design. If it is clear that an event has been sufficiently included in the assessment of other events, depending on the design, the analysis of this event may be omitted.

1.1 Abnormal transient changes during operation

1.1.1 Abnormal changes in core reactivity or power distribution

(1) Abnormal withdrawal of control rods at the startup of the reactor (PWR, BWR)
(2) Abnormal withdrawal of control rods during output-constant operation (PWR, BWR)
(3) Drop and inconsistency of control rods (PWR)
(4) Abnormal dilution of boron in reactor coolant (PWR)

1.1.2 Abnormal change in heat generation or removal in the core

(1) Partial loss of reactor coolant flow (PWR, BWR)
(2) Accidental startup of the reactor coolant system shutdown loop (PWR, BWR)
(3) Loss of external power supply (PWR, BWR)
(4) Loss of main feedwater flow (PWR)
(5) Abnormal increase in steam load (PWR)
(6) Abnormal depressurization of the secondary coolant system (PWR)
(7) Excess feedwater to the steam generator (PWR)
(8) Feedwater heating loss (BWR)
(9) Malfunction of the reactor coolant flow control system (BWR)

1.1.3 Abnormal change in reactor coolant pressure or stockpile

(1) Loss of load (PWR, BWR)
(2) Abnormal depressurization of the reactor coolant system (PWR)
(3) Accidental startup of the emergency core cooling system during output-constant operation (PWR)
(4) Accidental closing of the main steam isolation valve (BWR)
(5) Failure of the feedwater control system (BWR)
(6) Failure of the reactor pressure control system (BWR)
(7) Total loss of feedwater flow (BWR)

1.2 Accident

1.2.1 Loss of reactor coolant or remarkable change in core cooling

(1) Loss of reactor coolant (PWR, BWR)
(2) Loss of reactor coolant flow (PWR, BWR)
(3) Sticking of the reactor coolant pump shaft (PWR, BWR)
(4) Main feedwater pipe rupture (PWR)
(5) Main steam pipe rupture (PWR)

1.2.2 Abnormal insertion of reactivity or sharp change in reactor power

(1) Rod ejection (PWR)
(2) Rod drop (BWR)

1.2.3 Abnormal release of radioactive substances into the environment

(1) Damage to a radioactive gaseous waste disposal facility (PWR, BWR)
(2) Main steam pipe rupture (BWR)
(3) Steam generator tube rupture (PWR)
(4) Fuel assembly drop (PWR, BWR)
(5) Loss of reactor coolant (PWR, BWR)
(6) Rod ejection (PWR)
(7) Rod drop (BWR)

1.2.4 Abnormal change in pressure, ambient atmosphere, etc. inside the reactor containment vessel

(1) Loss of reactor coolant (PWR, BWR)
(2) Generation of flammable gas (PWR, BWR)
(3) Generation of dynamic load (BWR)

2. Analysis of abnormal transient changes during operation

Concrete conditions and judgment criteria to be referred to in the analysis of each event in the abnormal transient changes during operation described in item 1.1 above are shown below.

2.1 Abnormal changes in core reactivity or power distribution

2.1.1 Abnormal withdrawal of control rods at the startup of the reactor (PWR, BWR)

(1) An event is assumed to have occurred when control rods are continually withdrawn one after another owing to a failure, an operation error, etc. of the control rod drive system at the startup of the reactor, resulting in a rise in reactor power.
(2) The reactor in cold shutdown or on hot standby is assumed to be in or very close to a state of criticality.
(3) In a PWR, it is assumed that a combination, allowable in design terms, of two control rod cluster banks are withdrawn at the maximum speed so that the analysis results are the strictest in light of the judgment criteria.
(4) In a BWR, it is assumed that a control rod or a control rod group of the maximum reactivity value allowed by the rod worth minimizer is withdrawn at the maximum allowable speed in design terms.
(5) If an interlock related to the withdrawal of a control rod is in operation before or during the occurrence of an event and its reliability has proved sufficiently high, its operation can be taken into consideration.
(6) Other analysis conditions must meet the requirements of “Guidelines for Reactivity Insertion Event Assessment.”
(7) As judgment criteria, (3) and (4) in Guidelines II. 4.1 (hereafter called “4.1”) and criteria established by the “Guidelines for Reactivity Insertion Event Assessment” apply.

2.1.2 Abnormal withdrawal of control rods during output-constant operation (PWR, BWR)

(1) An event is assumed to have occurred when control rods are withdrawn continually owing to a failure, an operation error, etc. of the control rod drive system during output-constant operation of the reactor, resulting in a rise in reactor power.
(2) The reactor is assumed to have been in normal operation. Note that the initial power of the reactor must be selected so that the analysis results are the strictest inlight of the judgment

criteria.

(3) In a PWR, it is assumed that the allowable combination of two control rod cluster banks in design terms are withdrawn within the allowable speed so that the analysis results are the strictest in light of the judgment criteria.
(4) In a BWR, it is assumed that a control rod or a control rod group near the fuel assembly which is under the condition of a thermal limit value in the core is withdrawn within a speed allowed in design terms so that the analysis results are the strictest, in light of the judgment criteria.
(5) If an interlock related to the withdrawal of a control rod is in operation before or during the occurrence of an event and its reliability has proved sufficiently high, its operation can be taken into consideration.
(6) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.1.3 Drop and inconsistency of control rods (PWR)

(1) An event is assumed to have occurred when a control rod inserted in the core is displaced owing to a failure of the control rod drive system or other factors during the output-constant operation of the reactor, resulting in a change in power distribution in the core.
(2) It is assumed that a control rod cluster of the maximum reactivity value drops from all-withdrawal position to all-insertion position during operation with a margin added to the reactor rated power.
(3) As another event, an inconsistent condition is assumed to have occurred when a control rod cluster bank inserted in the core is at the insertion limit position during operation with a margin added to the reactor rated power and one of the control rod clusters is in the all-withdrawal position.
(4) If an automatic control system for reactor power is available, both automatic and manual control must be taken into consideration.
(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.1.4 Abnormal dilution of boron in reactor coolant (PWR)

(1) An event is assumed to have occurred when demineralized water is injected into the primary coolant owing to a failure, an operation error, etc. of the chemical and volume control system at the startup of the reactor or during output-constant operation, causing the concentration of boron in the primary coolant to decrease, resulting in an increment in reactivity.
(2) The reactor is assumed to be in cold shutdown at the startup and the concentration of boron in the primary coolant is assumed to be the maximum allowable during normal operation. The reactor is assumed to be running with a margin added to the rated power during output-constant operation and the concentration of boron in the primary coolant is assumed to be the allowable maximum during normal operation.
(3) It is assumed that demineralized water is injected into the primary coolant at the maximum allowable flow in design terms.
(4) If an automatic control system for reactor power is available, both automatic and manual control must be taken into consideration. In the case of automatic control, after a control rod reaches the insertion limit, when sufficiently time is allowed until the reactivity shutdown margin is lost and highly reliable information is available, an operator’s troubleshooting can be taken into consideration if it is fully expected.
(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2 Abnormal change in heat generation or removal in the core

2.2.1 Partial loss of reactor coolant flow (PWR, BWR)

(1) An event is assumed to have occurred when there is a decrease in core coolant flow owing

to a failure of the pump which drives (circulates) reactor coolant (The primary coolant in the case of a PWR. The same applies below.) or other factors during output-constant operation of the reactor.

(2) The reactor is assumed to have been in operation with a margin added to the rated power.
(3) It is assumed that the drive power of a reactor coolant pump (all the pumps concerned if the pumps may break down at the same time owing to a single failure or other factors because they are connected to the same bus or the same control device) is lost, resulting in a decrease in core coolant flow.
(4) If an automatic control system for reactor power is available, both automatic and manual control must be taken into consideration. The effects of the inertia of the reactor coolant pump and drive system to be shut down can be properly taken into consideration.
(5) When the operation of the safety protection system is expected, a decrease in reactor coolant flow, a change in the pump operating state, etc. must be detected as its operating signals.
(6) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.2 Accidental startup of the reactor coolant shutdown loop (PWR, BWR)

(1) An event is assumed to have occurred when some of the reactor coolant pumps are out of operation, one of those pumps starts owing to a failure or an operation error of the pump control system or other factors while the reactor is in operation under partial load, and relatively low-temperature coolant in the loop to which the pump is connected is injected into the core, causing reactivity to occur, resulting in a rise in reactor power.
(2) The reactor is assumed to have been in normal operation. Note that the initial power of the reactor and the number of reactor coolant pumps in initial operation must be selected so that the analysis results are the strictest in light of the judgment criteria.
(3) It is assumed that a reactor coolant pump out of operation (all the pumps concerned if the pumps may break down at the same time owing to a single failure or other factors because they are connected to the same bus or the same control device) starts accidentally.
(4) Even if an automatic control system for reactor power is available, its operation is not expected.
(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.3 Loss of external power supply (PWR, BWR)

(1) An event is assumed to have occurred when an external power supply is lost owing to a failure of the transmission system or the main generator set in the station during the output-constant operation of the reactor.
(2) The reactor is assumed to have been in operation with a margin added to the rated power.
(3) The external power supply system in the station is assumed to be in no-voltage state.
(4) Sufficiently time must be allowed for starting the emergency power.
(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.4 Loss of main feedwater flow (PWR)

(1) An event is assumed to have occurred when feedwater to all the steam generators is suspended owing to a failure of the main feedwater pump, condensate pump, or feedwater control system, or other factors during the output-constant operation of the reactor, resulting in a decrease in the capability of removing heat from the reactor.
(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.
(3) It is assumed that all of the main feedwater pumps of the secondary coolant system are suspended at the same time.
(4) As judgment criteria, (4) in 4.1 applies.

2.2.5 Abnormal increase in steam load (PWR)

(1) An event is assumed to have occurred when there is an abnormal increase in main steam flow owing to an accidental opening of the turbine bypass valve, turbine governor valve, or main steam relief valve of the secondary coolant system during the output-constant operation of the reactor, causing the temperature of primary coolant to decrease and reactivity to occur, resulting in a rise in reactor power.
(2) The reactor is assumed to have been in operation with a margin added to the rated power.
(3) It is assumed that a valve with the maximum steam flow, among the turbine bypass, turbine governor, and main steam relief valves, is fully open.
(4) If an automatic control system for reactor power is available, both automatic and manual control must be taken into consideration.
(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.6 Abnormal depressurization in the secondary coolant system (PWR)

(1) An event is assumed to have occurred when the temperature of primary coolant decreases owing to an accidental opening of a secondary coolant system valve such as the turbine bypass valve and main steam relief valve while the reactor is out of operation at a high temperature, resulting in an increment in reactivity.
(2) The reactor is assumed to be in out of operation at a high temperature and all control rods are assumed to have been inserted. The concentration of boron in the primary coolant is assumed to be the allowable minimum in design terms.
(3) It is assumed that a valve with the maximum depressurization effect, among the secondary coolant system valves, is fully open.
(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.7 Excess feedwater to the steam generator (PWR)

(1) An event is assumed to have occurred when water is excessively fed to the steam generator owing to a failure of the feedwater control system, an operation error, or other factors during the output-constant operation of the reactor, causing the temperature of the primary coolant to decrease and reactivity to be occurred, resulting in a rise in reactor power.
(2) The reactor is assumed to have been in operation with a margin added to the rated power.
(3) It is assumed that a feedwater control valve (all of the valves concerned which may operate at the same time owing to a single failure or other factors because they have been installed to the same control system) of the secondary coolant system is fully open and water is fed to a steam generator at a flow rate of the control valve full-open capacity.
(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.8 Feedwater heating loss (BWR)

(1) An event is assumed to have occurred when the steam flow to the feedwater heater is lost during the output-constant operation of the reactor, causing the temperature of the feedwater to decrease and the subcooling at the core inlet to decrease, resulting in a rise in reactor power.
(2) The reactor is assumed to have been in normal operation. Note that the initial power of the reactor must be selected so that the analysis results are the strictest in light of the judgment criteria.
(3) It is assumed that a stage of the feedwater heater (all of the stages of the haters concerned which may lose the functions simultaneously owing to a single failure or other factors because they have been installed to the same control system) loses the heating function and the temperature of the feedwater decreases by the maximum temperature change
(4) The reactor coolant recirculation system is assumed to be in manual operation mode.

(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.2.9 Malfunction of the reactor coolant flow control system (BWR)

(1) An event is assumed to have occurred when the coolant recirculation flow increases owing to a failure of the recirculation flow control system or other factors during the output-constant operation of the reactor, resulting in a rise in reactor power.
(2) The reactor is assumed to have been in normal operation for a sufficient period of time. The initial recirculation flow rate is assumed to be at the lower limit of the flow control range. The reactor initial power must be within the output range corresponding to this flow rate and selected so that the analysis results are the strictest in light of the judgment criteria.
(3) It is assumed that the recirculation flow changes to the maximum allowable rate of the recirculation flow control system in design terms.
(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.3 Abnormal change in reactor coolant pressure or stockpile

2.3.1 Loss of load (PWR, BWR)

(1) An event is assumed to have occurred when the steam flow to the turbine sharply decreases owing to a failure of the external power supply or turbine or other factors during the output-constant operation of the reactor, resulting in a rise in the pressure of the reactor.
(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.
(3) It is assumed that the external load is completely lost instantly.
(4) In a BWR, a case where the turbine bypass valve does not function must be also taken into consideration.
(5) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.3.2 Abnormal depressurization of the reactor coolant system (PWR)

(1) An event is assumed to have occurred when the reactor pressure decreases owing to a failure of the primary coolant pressure control system or other factors.
(2) The reactor is assumed to have been in operation with a margin added to the rated power.
(3) It is assumed that either pressurizer relief or spray valve (all of the valves concerned which may be fully opened at the same time owing to a single failure or other factors because they have been installed to the same control system) with the effect of depressurizing the reactor the most is fully open.
(4) As judgment criteria, (1) in 4.1 applies.

2.3.3 Accidental startup of the emergency core cooling system during output-constant operation (PWR)

(1) An event is assumed to have occurred when the ECCS is accidentally started during the output-constant operation of the reactor.
(2) The reactor is assumed to have been in operation with a margin added to the rated power.
(3) It is assumed that the high-pressure injection system of the ECCS starts, which will cause cooling water to be injected into the primary coolant system. When selecting a coolant flow rate, add a margin to the value determined by the pressure of the primary coolant system and the pump characteristics.
(4) As judgment criteria, (1) and (4) in 4.1 apply.

2.3.4 Accidental closing of the main steam isolation valve (BWR)

(1) An event is assumed to have occurred when the main steam isolation valve is closed owing to a failure or operation error of the isolation valve control system or other factors during

the output-constant operation of the reactor, resulting in a rise in the pressure of the reactor.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.

(3) It is assumed that the main steam isolation valve is closed in the shortest permissible time in design terms.

(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.3.5 Failure of the feedwater control system (BWR)

(1) An event is assumed to have occurred when the feedwater flow increases owing to a failure of the feedwater control system or other factors during the output-constant operation of the reactor, causing subcooling at the core inlet to increase, resulting in a rise in reactor power.

(2) The reactor coolant recirculation system is assumed to be in manual operation. The reactor initial power must be chosen so that the analysis results are the strictest in light of the judgment criteria.

(3) It is assumed that the feedwater flow reaches a permissible maximum in design terms instantly.

(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.3.6 Failure of the reactor pressure control system (BWR)

(1) An event is assumed to have occurred when the main steam flow varies owing to a failure of the pressure control system or other factors during the output-constant operation of the reactor.

(2) The reactor is assumed to have been in normal operation for a sufficient period of time. The reactor initial power, the operating status of the recirculation flow control system, etc. must be selected so that the analysis results are the strictest in light of the judgment criteria.

(3) It is assumed that the maximum output signal is issued from a line of the pressure control system (all of the control systems concerned which may stop functioning simultaneously owing to a single failure or other factors because two or more pressure control systems share a unit).

(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

2.3.7 Total loss of feedwater flow (BWR)

(1) An event is assumed to have occurred when the feedwater decreases owing to a failure of the feedwater control system or other factors during the output-constant operation of the reactor.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.

(3) It is assumed that the feedwater pump trips and the feedwater flow is completely lost within the time when the inertia of the pump is taken into consideration.

(4) As judgment criteria, (1), (2), and (4) in 4.1 apply.

3. Analysis of accidents

Concrete conditions to be referred to in analyzing each event in “Accident” listed in item 1.2 above and judgment criteria are shown below.

3.1 Loss of reactor coolant or remarkable change in core cooling

3.1.1 Loss of reactor coolant (PWR, BWR)

(1) An event is assumed to have occurred when reactor coolant is flowed out of the system owing to rupture in the piping composing the coolant pressure boundary, damage to a device attached to the boundary, or other factors during the output-constant operation of the

reactor, resulting in a decrease in the cooling capability of the core.

(2) For analysis conditions, the ECCS Performance Evaluation Guide must be satisfied. Note that analysis must include the process from when an event occurs to when it is reasonably anticipated that the system can be put back to recirculation mode.

(3) As judgment criteria, (1) in Guidelines II. 4.2 (hereafter called “4.2”) and criteria established by the ECCS Performance Evaluation Guide apply.

3.1.2 Loss of reactor coolant flow (PWR, BWR)

(1) An event is assumed to have occurred when the flow of reactor coolant sharply decreases from a flow rate at a rated output to a natural circulation flow rate during the output-constant operation of the reactor.

(2) The reactor is assumed to have been in operation with a margin added to the rated power.

(3) It is assumed that the power for driving all the reactor coolant pumps is lost at the same time.

(4) The operating state of the automatic control system of the reactor coolant flow must be selected so that the analysis results are the strictest in light of the judgment criteria. The effects of the inertia of the reactor coolant pump and drive system to be shut down can be properly taken into consideration.

(5) When the operation of the safety protection system is expected, a decrease in reactor coolant flow, a change in the pump operating state, etc. must be detected from its operating signals.

(6) As judgment criteria, (1) and (3) in 4.2 apply.

3.1.3 Sticking of the reactor coolant pump spindle (PWR, BWR)

(1) An event is assumed to have occurred when the spindle of a coolant-driving pump is stuck during the output-constant operation of the reactor, resulting in a sharp decrease in coolant flow.

(2) The reactor is assumed to have been in operation with a margin added to the rated power.

(3) It is assumed that the spindle of a coolant pump is stuck and suspended instantly.

(4) As judgment criteria, (1) and (3) in 4.2 apply.

3.1.4 Main feedwater pipe rupture (PWR)

(1) An event is assumed to have occurred when a feedwater pipe is ruptured during the output-constant operation of the reactor, causing a loss of secondary coolant, resulting in a decrease in the reactor cooling capacity.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.

(3) It is assumed that the main feedwater pipe is ruptured at both ends instantly, resulting in a loss of water held in the steam generator to which the pipe is connected and also in a loss of main feedwater to all the steam generators simultaneously with the rupture.

(4) The external power supply cannot be used.

(5) As judgment criteria, (1) and (3) in 4.2 apply.

3.1.5 Main steam pipe rupture (PWR)

(1) An event is assumed to have occurred when the temperature of primary coolant decreases owing to a break in the secondary coolant system while the reactor is out of operation at a high temperature, resulting in an increment in reactivity.

(2) The reactor is assumed to be in out of operation at a high temperature. All control rods are assumed to have been inserted. The concentration of boron in the primary coolant is assumed to be the allowable minimum in design terms.

(3) It is assumed that the main steam pipe from the steam generator to the turbine is ruptured at

both ends instantly.

(4) The temperature decrease rate on the secondary side of the steam generator caused by the rupture of the main steam pipe must be estimated on the high side with a proper margin added, with due consideration given to the structure and other factors of the steam generator. Both cases where the external power supply is and is not available must be taken into consideration.

(5) As judgment criteria, (1) and (3) in 4.2 apply.

3.2 Abnormal insertion of reactivity or sharp change in reactor power

3.2.1 Rod ejection (PWR)

(1) An event is assumed to have occurred when there is a drastic increment in reactivity and a change in power distribution owing to damage to the control rod drive mechanism or its housing or other factors when the reactor is in or very close to a state of criticality.

(2) The reactor is assumed to be in or very close to a state of criticality. The initial state of the reactor must be selected so that the analysis results are the strictest in light of the judgment criteria.

(3) It is assumed that the control rod housing is instantly damaged and reactivity equivalent to that when a control rod cluster of the maximum reactivity value is quickly ejected from the core is inserted.

(4) It is assumed that any negative reactivity effect caused by flushing owing to damage to the control rod housing is not taken into consideration.

(5) For analysis conditions and other factors, the requirements in the “Guidelines for Reactivity Insertion Event Assessment” must be satisfied.

(6) As judgment criteria, (2) and (3) in 4.2 and “Guidelines for Reactivity Insertion Event Assessment” apply.

3.2.2 Rod drop (BWR)

(1) An event is assumed to have occurred if a control rod separated from its drive shaft drops from the core when the reactor is in or very close to a state of criticality, resulting in a drastic increment in reactivity and a change in power distribution.

(2) The reactor is assumed to be in or very close to a state of criticality. The initial state of the reactor must be selected so that the analysis results are the strictest in light of the judgment criteria.

(3) It is assumed that reactivity equivalent to that when a control rod of the maximum reactivity value drops from the core is inserted.

(4) For analysis conditions and other factors, the requirements in the “Guidelines for Reactivity Insertion Event Assessment” must be satisfied.

(5) As judgment criteria, (2) and (3) in 4.2 and “Guidelines for Reactivity Insertion Event Assessment” apply.

3.3 Abnormal release of radioactive substances into the environment

3.3.1 Damage to a radioactive gaseous waste disposal facility (PWR, BWR)

(1) An event is assumed to have occurred when part of a radioactive gaseous waste disposal facility is damaged, resulting in the release of gaseous radioactive substances stored in the facility into the environment.

(2) It is assumed that gaseous radioactive substances are stored in the storage tank, holdup tank, or other containers of the radioactive gaseous waste disposal facility in maximum allowable quantities in terms of design of the reactor facility during normal operation of the reactor (including the startup, hot standby, output-constant operation, and shutdown).

(3) It is assumed that part of a radioactive gaseous waste disposal facility is damaged, resulting in the release of gaseous radioactive substances stored in the facility. The damaged part must be selected so that the analysis results are the strictest in light of the judgment criteria,

with due consideration given to the amount of radioactive substances stored, isolation time of the damaged part, etc.

(4) It is assumed that devices connected to the damaged part that may increase the release of gaseous radioactive substances are in operation so that the analysis results are the strictest within the allowable range in design terms. If valves and other devices that can be isolated from the damaged part are available, their functions can be expected with a sufficient margin added to the time required for their operation.

(5) If the damaged part is found indoors, the ventilation system in the auxiliary building or turbine building must be assumed to be operating in such a way that the analysis results are the strictest in light of the judgment criteria.

(6) It is assumed that the diffusion of radioactive substances released into the environment is evaluated in accordance with the "Guidelines for Weather Conditions Related to Safety Analysis of Power Reactor Facility" (hereafter called "Guidelines for Weather Conditions").

(7) As judgment criteria, (5) in 4.2 applies.

3.3.2 Main steam pipe rupture (BWR)

(1) An event is assumed to have occurred when the main steam pipe is ruptured outside the reactor containment vessel during the output-constant operation of the reactor, causing reactor coolant to flow from the break, resulting in the release of radioactive substances into the environment.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.

(3) It is assumed that the main steam pipe is instantly ruptured at both ends outside the reactor containment vessel.

(4) It is assumed that the main steam isolation valves are fully closed in the maximum operation delay time and closing time in design terms.

(5) The functions of the flow controller can be taken into consideration in calculating the flow rate of the reactor coolant. It is assumed that the effect of the main steam isolation valve upon flow restriction is not considered until a critical flow occurs at the isolation valve.

(6) It is assumed that the external power supply is lost as soon as this event occurs.

(7) The concentration of fission products in reactor coolant before the event occurs is assumed to be equal to the maximum allowable concentration of I-131 during operation, and the fission products are assumed to be of a diffusion composition. The concentration of halogen in the steam phase is assumed to be 2% of the liquid-phase concentration.

(8) It is assumed that the additional release of I-131 from the fuel rod caused by a decrease in the pressure of the reactor is a value obtained by adding a proper margin to the mean value of the actual measurements in precursor reactors, the composition of other fission products is determined as an equilibrium composition, and rare gas is released twice as much as iodine. The additional release rate of fission products is assumed to be proportionate to a reactor pressure decrease rate. The time necessary for fission products released from the fuel rod before the main steam isolation valve is closed to reach the isolation valve in the course of the event can be taken into consideration in the assessment.

(9) In the course of the event, it is assumed that organic and inorganic iodine account for 4% and 96%, respectively, of all the iodine released from the fuel rod and that 10% of the organic iodine instantly transforms into the gas phase, and the rest is dissolved. It is postulated that the carry-over rate of 90% dissolved organic iodine, inorganic iodine, and halogen except iodine to the gas phase is 2% and all the rare gas instantly transforms into the gas phase. It is also assumed that 50% of the dissolved organic iodine released into the turbine building, inorganic iodine, and halogen except iodine is deposited on the floor, walls, and other places.

(10) It is assumed that the reactor coolant released before the main steam isolation valve is

closed is completely evaporated and transformed into a steam cloud homogeneously containing the fission products released at the same time. It is also assumed that the fission products released after the isolation valve is closed is diffused into the atmosphere.

(11) It is assumed that one of the main steam isolation valves is not closed. It is assumed that a leak determined by a design leakage rate, temperature, and pressure occurs from the closed valves.

(12) It is assumed that after the main steam isolation valve is closed, the steam equivalent to decay heat moves to the suppression pool through the residual heat removal system, safety relief valve, etc.

(13) It is assumed that after the main steam isolation valve is closed, the reactor pressure linearly drops to the atmospheric pressure in the time required for the reactor core isolation cooling system or other systems to drop the reactor pressure to it or in 24 hours, whichever is longer.

(14) It is assumed that the formation and movement of a steam cloud is evaluated with proper parameters and the diffusion of fission products released into the environment after the isolation valve is closed is evaluated in accordance with the “Guidelines for Weather Conditions.”

(15) As judgment criteria, (5) in 4.2 applies after it is confirmed that no further damage to the fuel rod occurs.

3.3.3 Steam generator tube rupture (PWR)

(1) An event is assumed to have occurred when a heat-transfer pipe of the steam generator is ruptured during the output-constant operation of the reactor, resulting in the release of primary coolant outside the reactor containment vessel through the secondary coolant system.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power and the reactor pressure is assumed to be the maximum during normal operation.

(3) It is assumed that a heat-transfer pipe of the steam generator is instantly ruptured at both ends.

(4) Both cases where the external power supply is and is not lost are taken into consideration. When the ECCS automatically starts, it is assumed that the ECCS operates in such a way that it increases the flow rate of the primary coolant.

(5) The concentration of fission products in primary coolant before the event occurs is assumed to be a value calculated by using a fuel cladding failure rate anticipated in design terms.

(6) It is assumed that rare gas and iodine are additionally released from a gap of the fuel rod with a defect anticipated in design terms, in proportion to the pressure decrease rate of the reactor.

(7) It is assumed that the whole amount of rare gas flowing into the secondary coolant system is released into the atmosphere. It is also assumed that iodine is released into the atmosphere together with steam at a partition coefficient of 100.

(8) If an operator needs to isolate the damaged steam generator, sufficiently time must be allowed for the operation. After the isolation, it is assumed that the reactor pressure linearly drops to the atmospheric pressure in the time required for an operable cooling system to drop the reactor pressure to it or in 24 hours, whichever is longer. It is assumed that a leak determined by a design leakage rate, temperature, and pressure occurs from the valve of the isolated steam generator.

(9) It is assumed that the diffusion of fission products released into the environment is evaluated in accordance with the “Guidelines for Weather Conditions.”

(10) As judgment criteria, (5) in 4.2 applies after it is confirmed that no further damage to the fuel rod occurs.

3.3.4 Fuel assembly drop (PWR, BWR)

(1) An event is assumed to have occurred when a fuel assembly is dropped for some reason or other and damaged during fuel exchange, resulting in the release of radioactive substances into the environment.

(2) In a PWR, it is assumed that a fuel assembly handled in the spent fuel pit is dropped from the top of the operational position.

(3) In a BWR, it is assumed that a fuel assembly handled on the reactor core is dropped from the top of the operational position to the core.

(4) It is assumed that the dropped fuel assembly is the maximum output assembly with which the reactor has been in operation for a sufficient period of time with a margin added to the rated power and the event occurs after a lapse of proper cooling and work time after the reactor is shut down. The attenuation of radioactivity during this period can be properly taken into consideration.

(5) The number of fuel rods damaged from the drop must be the maximum unless there is evidence based on experiment.

(6) It is assumed that fission products are released into water from a gap of the damaged fuel rod. It is assumed that the whole amount of rare gas, among them, transforms into the gas phase. The decontamination factor of iodine in water is assumed to be 500.

(7) It can be expected that the ventilation system in the auxiliary building or reactor building, emergency ventilation system, etc. operate as designed.

(8) It is assumed that the diffusion of fission products released into the environment is evaluated in accordance with the “Guidelines for Weather Conditions.”

(9) As judgment criteria, (5) in 4.2 applies.

3.3.5 Loss of reactor coolant (PWR, BWR)

(1) An event is assumed to have occurred when radioactive substances are released into the environment on the occasion of the loss of reactor coolant postulated in item 3.1.1.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.

(3) The concentration of fission products in the reactor coolant before the event occurs must be postulated in the same way as in item 3.3.2 or 3.3.3.

(4) If this event causes damage to a fuel rod, the reasonable amount of fission products is assumed to have been released according to the condition of the damaged fuel rod. If no fuel rod is damaged, the additional amount of fission products must be evaluated in the same way as in item 3.3.2 or 3.3.3.

(5) It is assumed that this event releases rare gas and iodine from the fuel rod into the reactor containment vessel and that organic and inorganic iodine account for 4% and 96%, respectively. It is also assumed that 50% of the inorganic iodine, which is deposited in the reactor containment vessel, does not contribute to leakage. In addition, the effect of removing some inorganic iodine by spray water in the reactor containment vessel or dissolving it in suppression pool water can be taken into consideration. In this case, the decontamination factor, partition coefficient, etc. must be experimental values or values with a sufficient safety allowance added. The effect of organic iodine and rare gas is ignored.

(6) Leaks from the reactor containment vessel are evaluated based on its design leak rate and the analysis results in item 3.4.1, on the assumption of a leak rate commensurate with the pressure in the reactor containment vessel. In a PWR, it is assumed that 97% of the leak occurs in the annulus and the remaining 3% outside the annulus. The deposition of leaked fission products in the annulus or reactor building is not taken into consideration.

(7) The seizing signal of the emergency ventilation system (including a filter) in the annulus or

reactor building must be made clear and its function can be expected with sufficiently time allowed.

(8) If the ECCS is operated in recirculation mode and the water in the reactor containment vessel is lead outside the vessel, it is assumed that recycled water leaks at a design leak rate outside the reactor containment vessel. It is also assumed that the same amount of iodine as in items (3) and (4) is dissolved in recycled water as inorganic iodine and that the transformation rate of leaked iodine into the gas phase is 5% and the deposition rate in the auxiliary building or reactor building is 50%.

(9) The direct dose by fission products in the reactor containment vessel and skyshine dose should be evaluated, with due consideration given to the location of the fission products in the vessel and the shielding of the vessel.

(10) The accident should be evaluated until the internal pressure of the reactor containment vessel decreases to such an extent that leaks from the vessel are negligible.

(11) The diffusion of fission products released into the environment should be evaluated in accordance with the "Guidelines for Weather Conditions."

(12) As judgment criteria, (5) in 4.2 applies.

3.3.6 Rod ejection (PWR)

(1) An event is assumed to have occurred when radioactive substances are released into the environment on the occasion of the control rod ejection postulated in item 3.2.1.

(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.

(3) The effective dose-equivalent is evaluated in accordance with item 3.3.5.

(4) As judgment criteria, (5) in 4.2 applies.

3.3.7 Rod drop (BWR)

(1) An event is assumed to have occurred when radioactive substances are released into the environment on the occasion of the control rod drop postulated in item 3.2.2.

(2) If this event occurs while the reactor is on hot standby or operating at a limited capacity, the reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power until 30 minutes before the event occurs. If this event occurs while the reactor is in cold state, the reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power until 24 hours before the event occurs.

(3) It is assumed that fission products are released into the reactor coolant from the gap of the damaged fuel rod, that organic iodine accounts for 4% and inorganic iodine makes up the remaining 96% of all the iodine released from the fuel rod, and that 10% of the organic iodine instantly transforms into the gas phase and the rest is dissolved. It is also assumed that the carry-over rate of 90% dissolved organic iodine and inorganic iodine to the gas phase is 2% and all the rare gas instantly transforms into the gas phase.

(4) It is assumed that the main steam isolation valve is closed in the maximum operation delay time and closing time in design terms. It is also assumed that 50% of the inorganic iodine, among fission products moved into the condenser, is deposited and that the remaining fission products in the gas phase leak into the turbine building at a leak rate of 0.5%/day in relation to the condenser and turbine free space.

(5) If the ventilation system in the turbine building is in operation, its functions are taken into consideration.

(6) The diffusion of fission products released into the environment is evaluated in accordance with the "Guidelines for Weather Conditions."

(7) As judgment criteria, (5) in 4.2 applies.

3.4 Abnormal change in the pressure, ambient atmosphere, etc. inside the reactor containment vessel

3.4.1 Loss of reactor coolant (PWR, BWR)

(1) An event is assumed to have occurred when reactor coolant flows out of the system owing to damage to a pipe or other devices composing the reactor coolant pressure boundary during the output-constant operation of the reactor, resulting in an abnormal rise in the pressure and temperature inside the reactor containment vessel.
(2) The reactor is assumed to have been in operation for a sufficient period of time with a margin added to the rated power.
(3) A pipe composing the reactor coolant pressure boundary is assumed to be rupturing at both ends instantly. The pipe assumed to be rupturing and its break must be selected so that the pressure inside the reactor containment vessel is maximum.
(4) As soon as the event occurs, it is assumed that the external power supply is disabled.
(5) As judgment criteria, (4) in 4.2 applies after it is confirmed that the temperature inside the reactor containment vessel does not exceed the maximum allowable working temperature.

3.4.2 Generation of flammable gas (PWR, BWR)

(1) An event is assumed to have occurred when flammable gas is generated during the loss of reactor coolant postulated in item 3.4.1.
(2) The amount of hydrogen generated by metal – water reaction is assumed to be five times greater than that calculated in item 3.1.1 or equivalent to a reaction of water with a metal having a thickness of 0.0058 mm from the surface of the cladding tubes of all the fuel rods, whichever is greater.
(3) The radiolysis of water in the vessel must be properly evaluated, on the assumption that 50% of the halogen and 1% of the fission products except rare gas and halogen, among the contents of the fission products in the core, are present in the liquid phase of the water in the reactor containment vessel. In addition, that of water present in the core must be properly evaluated, on the assumption that other fission products except rare gas are all present in the core. The decomposition rate per unit absorbed energy of water must be an experimental value with a proper margin added.
(4) If the reactor is so designed that alkaline substances can be added to the water in the reactor containment vessel, hydrogen generated by chemical reaction with a metal structure in the vessel must be properly evaluated.
(5) If the reactor is so designed that some systems to control the concentration of flammable gas in the hydrogen recombiner can be installed, their functions can be expected within the range of design of these systems.
(6) As judgment criteria, the concentration of either oxygen or hydrogen in a reactor containment vessel atmosphere must be 5% or 4% or less, respectively, for at least 30 days after this event occurs.

3.4.3 Generation of dynamic load (BWR)

(1) A local dynamic load is assumed to have occurred in the pressure suppression reactor containment vessel when reactor coolant is lost or a safety valve is activated.
(2) A dynamic load in the reactor containment vessel must be evaluated in accordance with the “Guidelines for Evaluating Dynamic Load Applied to BWR. MARK I (Containment Vessel) Pressure Suppression System” or “Guidelines for Evaluating Dynamic Load Applied to BWR. MARK II (Containment Vessel) Pressure Suppression System.”
(3) As a result of the assessment, if it is proved that the stress and other factors of each part of the reactor containment vessel meet the applicable standards, requirements, etc. or that the vessel is designed in line with the policy of meeting them, the design or design policy is recognized as adequate.

II. Site Assessment

1. Concrete events in major and hypothetical accidents

Concrete events in major and hypothetical accidents to be evaluated are as shown below.

1.1 Loss of reactor coolant (PWR, BWR)
1.2 Steam generator tube rupture (PWR)
1.3 Main steam pipe rupture (BWR)

2. Assessment of major and hypothetical accidents

How to apply concrete conditions and judgment criteria that will help to evaluate each event in major and hypothetical accidents described in item 1 above is shown below.

2.1 Loss of reactor coolant
2.1.1 Loss of reactor coolant (PWR)

Major accidents

(1) An event is assumed to be occurring when radioactive substances are released into the environment owing to a loss of reactor coolant postulated in item 3.1.1 of “I. Safety Design Assessment” in Appendix I.

(2) The reactor is assumed to have been in operation long enough with a sufficient margin added to the rated power.

(3) After the event occurs, it is assumed that rare gas and iodine, which are fission products, are released into the reactor containment vessel and the proportion of the release of rare gas and iodine to the accumulation in the core is 2% and 1%, respectively.

(4) It is assumed that organic and inorganic iodine account for 10% and 90%, respectively, of the whole amount of iodine released into the vessel.

(5) It is assumed that among iodine released into the containment vessel, 50% of the inorganic iodine, which is deposited in the reactor containment vessel and devices in this vessel, does not contribute to the leakage from the vessel and this effect of organic iodine and rare gas is ignored.

(6) The removal efficiency of inorganic iodine by containment spray water should be an experimental value with a sufficient margin added. For example, if the equivalent half-life evaluated in terms of design is 50 s or less, it is recognized as adequate to estimate the equivalent half-life at 100 s. This effect of organic iodine and rare gas should be ignored.

(7) Leaks of rare gas and iodine from the containment vessel should be taken into consideration. Leaks from the vessel should be evaluated on the basis of its design leak rate and the analysis results in item 3.4.1 of “I. Safety Design Assessment,” on the assumption of a leak rate commensurate with the pressure in the vessel with a sufficient margin added. It is assumed that 97% of the leak from the vessel occurs in the annulus and the remaining 3% occurs outside the annulus.

(8) The seizing signal of the annulus air recirculation facility (including the filter) must be made clear and its functions can be expected with sufficient time allowed. Incidentally, the efficiency of removing iodine with a filter is a design value with an adequate margin added. For example, if the design iodine removal efficiency is 95% or more, it is recognized as adequate to estimate the efficiency at 90%.

(9) If the ECCS is operated in recirculation mode and the water in the reactor containment vessel is lead outside the vessel, it is assumed that a leak of recycled water occurs outside the vessel at a design leak rate with a sufficient margin added. It is also assumed that 1% of the amount of iodine accumulated in the core is dissolved in the recycled water immediately

after the event occurs, the transformation rate of iodine leaking from the ECCS recirculation system to the auxiliary building to the gaseous phase is 5%, and the deposition rate of iodine in the auxiliary building is 50%.

(10) If the ventilation system in the auxiliary building where the ECCS recirculation system is installed is provided with an iodine filter, its removal rate should be a design value with a sufficient margin added. For example, if the design removal value of iodine is 95% or more, it is recognized as adequate to estimate the removal rate at 90%.

(11) The direct dose by fission products in the reactor containment vessel and skyshine dose should be evaluated, with due consideration given to the shielding of the vessel. In doing so, it is assumed that the proportion of the amount of rare gas, halogen, and other elements released into the vessel to the amount of those accumulated in the core is 2%, 1%, and 0.02%, respectively.

(12) The accident should be evaluated until the internal pressure of the reactor containment vessel decreases to such an extent that a leak from the vessel is negligible, but the accident assessment period should not be less than 30 days.

(13) The diffusion of fission products released into the environment should be evaluated in accordance with the “Guidelines for Weather Conditions.”

(14) Judgment criteria should conform to the “Guidelines for Reactor Site Assessment.”

Hypothetical accidents

The same also applies to hypothetical accidents as in the case of major accidents except for the following items.

(3) After the event occurs, it is assumed that rare gas and iodine, which are fission products, are released into the reactor containment vessel and the proportion of the release of rare gas and iodine to the accumulation in the core is 100% and 50%, respectively.

(9) If the ECCS is operated in recirculation mode and the water in the reactor containment vessel is lead outside the vessel, it is assumed that a leak of recycled water occurs outside the vessel at a design leak rate with a sufficient margin added. It is also assumed that 50% of the amount of iodine accumulated in the core is dissolved in the recycled water immediately after the event occurs, the transformation rate of iodine leaking from the ECCS recirculation system to the auxiliary building to the gaseous phase is 5%, and the deposition rate of iodine in the auxiliary building is 50%.

(11) The direct dose by fission products in the reactor containment vessel and skyshine dose should be evaluated, with due consideration given to the shielding of the vessel. In doing so, the proportion of the amount of rare gas, halogen, and other elements released into the vessel to the amount of those accumulated in the core is 100%, 50%, and 1%, respectively.

2.1.2 Loss of reactor coolant (BWR)

Major accidents

(1) An event is assumed to be occurring when radioactive substances are released into the environment owing to a loss of reactor coolant postulated in item 3.1.1 of “I. Safety Design Assessment” in Appendix I.

(2) The reactor is assumed to have been in operation long enough with a sufficient margin added to the rated power.

(3) After the event occurs, it is assumed that rare gas and iodine, which are fission products, are released into the reactor containment vessel and the proportion of the release of rare gas and iodine to the accumulation in the core is 2% and 1%, respectively.

(4) It is assumed that organic and inorganic iodine account for 10% and 90%, respectively, of the whole amount of iodine released into the vessel.

(5) It is assumed that among iodine released into the containment vessel, 50% of the inorganic

iodine, which is deposited in the reactor containment vessel and devices in this vessel, does not contribute to the leakage from the vessel and this effect of organic iodine and rare gas is ignored.

(6) It is assumed that the proportion of the dissolved inorganic iodine to the suppression pool water is 100 in terms of distribution coefficient and this effect of organic iodine and rare gas is ignored.

(7) Leaks of rare gas and iodine from the containment vessel should be taken into consideration. Leaks from the vessel should be evaluated on the basis of its design leak rate and the analysis results in item 3.4.1 of "I. Safety Design Assessment," on the assumption of a leak rate commensurate with the pressure in the vessel with a sufficient margin added.

(8) The seizing signal of the emergency ventilation system (including the filter) and other systems in the reactor building must be made clear and its functions can be expected with sufficient time allowed.

The capacity of the emergency ventilation system should be a design values. Incidentally, the efficiency of removing iodine with a filter is a design value with an adequate margin added. For example, if the design iodine removal efficiency is 99% or more, it is recognized as adequate to estimate the efficiency at 95%. The efficiency of removing fission products by deposition in the reactor building should be ignored and only spontaneous disintegration should be taken into consideration.

(9) If the ECCS is operated in recirculation mode and the water in the reactor containment vessel is lead outside the vessel, it is assumed that a leak of recycled water occurs outside the vessel at a design leak rate with a sufficient margin added. It is also assumed that 1% of the amount of iodine accumulated in the core is dissolved in the recycled water immediately after the event occurs, the transformation rate of iodine leaking from the ECCS recirculation system to the auxiliary building to the gaseous phase is 5%, and the deposition rate of iodine in the auxiliary building is 50%.

(10) After being treated by the standby gas treatment system in the reactor building, it is assumed that fission products leaking from the containment vessel into the reactor building are released into the environment from the stack.

(11) The direct dose by fission products in the reactor containment vessel and skyshine dose should be evaluated, with due consideration given to the shielding of the vessel. In doing so, it is assumed that the proportion of the amount of rare gas, halogen, and other elements released into the vessel to the amount of those accumulated in the core is 2%, 1%, and 0.02%, respectively.

(12) The accident should be evaluated until the internal pressure of the reactor containment vessel decreases to such an extent that a leak from the vessel is negligible, but the accident assessment period should not be less than 30 days.

(13) The diffusion of fission products released into the environment should be evaluated in accordance with the "Guidelines for Weather Conditions."

(14) Judgment criteria should conform to the "Guidelines for Reactor Site Assessment."

<u>Hypothetical accidents</u>

The same also applies to hypothetical accidents as in the case of major accidents except for the following items.

(3) After the event occurs, it is assumed that rare gas and iodine, which are fission products, are released into the reactor containment vessel and the proportion of the release of rare gas and iodine to the accumulation in the core is 100% and 50%, respectively.

(9) If the ECCS is operated in recirculation mode and the water in the reactor containment vessel is lead outside the vessel, it is assumed that a leak of recycled water occurs outside the vessel at a design leak rate with a sufficient margin added. It is also assumed that 50% of the amount of iodine accumulated in the core is dissolved in the recycled water

immediately after the event occurs, the transformation rate of iodine leaking from the ECCS recirculation system to the auxiliary building to the gaseous phase is 5%, and the deposition rate of iodine in the auxiliary building is 50%.

(11) The direct dose by fission products in the reactor containment vessel and skyshine dose should be evaluated, with due consideration given to the shielding of the vessel. In doing so, the proportion of the amount of rare gas, halogen, and other elements released into the vessel to the amount of those accumulated in the core is 100%, 50%, and 1%, respectively.

2.2 Steam generator tube rupture (PWR)

Major accidents

(1) An event is assumed to be occurring when a heat-transfer pipe of the steam generator is ruptured during the output-constant operation of the reactor, resulting in the release of primary coolant outside the reactor containment vessel through the secondary coolant system.

(2) The reactor is assumed to have been in operation long enough with a sufficient margin added to the rated power.

(3) It is assumed that a heat-transfer pipe of the steam generator is instantly ruptured at both ends.

(4) Either case where the external power source is lost or not lost is assumed to be occurring, whichever is stricter. When the ECCS automatically starts, it is assumed that it operates so that a large amount of primary coolant flows.

(5) The concentration of fission products in primary coolant before the event occurs is assumed to be a value calculated by using a fuel cladding failure rate anticipated in design terms.

(6) It is assumed that rare gas and iodine are additionally released from a gap of the fuel rod having a defect anticipated in design terms, in proportion to the pressure decrease rate of the reactor.

(7) It is assumed that, among these fission products in the primary coolant, the amount of radioactivity, flowing from the primary coolant system to the secondary one until the steam generator is isolated, depends on the radioactive concentration in the primary coolant.

(8) It is assumed that organic iodine accounts for 1% of the iodine flowing into the secondary coolant system and inorganic iodine makes up the remaining 99% and the whole amount of organic iodine is released into the atmosphere. It is also assumed that inorganic iodine is released together with steam into the atmosphere at a partition coefficient of 100.

It is assumed that the whole amount of rare gas flowing into the secondary coolant system is released into the atmosphere.

(9) It is conceivable that further fission products are not released into the atmosphere after the damaged steam generator is isolated, but it is assumed in terms of assessment that inorganic iodine is released into the atmosphere owing to a leak of steam from the secondary valve. The steam leak rate from the valve is a design value with a sufficient margin added. After the isolation, it is assumed that the reactor pressure linearly drops to the atmospheric pressure in the time required for an operable cooling system to drop the reactor pressure to it or in 24 hours, whichever is longer. It is also assumed that, in response to this pressure, steam leaks from the valve at a design leak rate.

(10) It is assumed that the diffusion of fission products released into the environment is evaluated in accordance with the "Guidelines for Weather Conditions."

(11) Judgment criteria should conform to the "Guidelines for Reactor Site Assessment."

Hypothetical accidents

The same also applies to hypothetical accidents as in the case of major accidents except for the following items.

(6) It is assumed that rare gas and iodine are additionally released into the primary coolant system from a gap of the fuel rod having a defect anticipated in design terms immediately after the accident occurs.

(7) It is assumed that, among the fission products in the primary coolant, the proportion of the amount of radioactivity flowing from the primary coolant system to the secondary one until the steam generator is isolated to the whole amount of radioactivity is the same as that of the amount of primary coolant flowing out at that moment to the entire holding water quantity.

(9) After the damaged steam generator is isolated, it is assumed that inorganic iodine is released into the atmosphere owing to a leak of steam from the secondary valve. It is also assumed that steam leaks from the valve last for 30 days and its rate is a design value with a sufficient margin added.

2.3 Main steam pipe rupture (BWR)

Major accidents

(1) An event is assumed to be occurring when a main steam pipe is ruptured outside the reactor containment vessel during the output-constant operation of the reactor, resulting in the release of radioactive substances into the environment.

(2) The reactor is assumed to have been in operation long enough with a sufficient margin added to the rated power.

(3) It is assumed that a main steam pipe is instantly ruptured at both ends outside the reactor containment vessel.

(4) It is assumed that the main steam isolation valve is fully closed in the maximum operation delay time and closing time in design terms.

(5) The functions of the flow controller can be taken into consideration in calculating the flow of reactor coolant. Note that the effect of a main steam isolation valve upon flow restriction will not be taken into consideration until a critical flow occurs at the isolation valve.

(6) It is assumed that the external power source is lost as soon as this event occurs.

(7) The concentration of fission products in reactor coolant before the event occurs is assumed equal to the maximum allowable concentration during operation in I-131, and the fission products are assumed to be of a diffusion composition. The concentration of halogen in the steam phase is assumed to be 1/50 of the liquid-phase concentration.

(8) It is assumed that the amount of I-131 further released from the fuel rod caused by a decrease in the pressure of the reactor is the actual measurements obtained from precursor reactors with a safety margin allowed, the composition of other fission products is determined as an equilibrium composition, and rare gas is released twice as much as iodine.

(9) The additional fission product release rate from the fuel rod before the main steam isolation valve is closed is assumed to be proportionate to the decreasing rate in the pressure of the reactor before the valve is closed. It is assumed that 1% of the amount of fission products additionally released is emitted from the break.

(10) It is assumed that the additional fission product release rate from the fuel rod after the main steam isolation valve is closed is gradually released into reactor coolant in proportion to the decreasing rate in the pressure of the reactor.

(11) In the course of the event, it is assumed that organic and inorganic iodine account for 10% and 90%, respectively, of all the iodine released from the fuel rod and that 10% of the organic iodine instantly transforms into the gas phase. It is also assumed that the carry-over rate of the remaining iodine and other halogens to the gas phase is 2% and all the rare gas instantly transforms into the gas phase.

(12) It is assumed that the reactor coolant released before the main steam isolation valve is closed is completely evaporated and transformed into a steam cloud homogeneously

containing the fission products released at the same time. It is also assumed that the fission products released after the isolation valve is closed are diffused into the atmosphere.

(13) It is assumed that one of the main steam isolation valves is not closed and steam leaks from the closed isolation valve. It is also assumed that the leak rate of the closed main steam isolation valve is a design value with a sufficient margin added and its value varies, depending on the temperature and pressure.

(14) It is assumed that after the main steam isolation valve is closed, the steam equivalent to decay heat moves to the suppression pool through the residual heat removal system, safety relief valve, etc.

(15) It is assumed that after the main steam isolation valve is closed, the reactor pressure linearly drops to the atmospheric pressure in the time required for the reactor core isolation cooling system or other systems to drop the reactor pressure to it or in 24 hours, whichever is longer.

(16) It is assumed that the formation and movement of a steam cloud is evaluated with proper parameters and the diffusion of fission products released into the environment after the isolation valve is closed is evaluated in accordance with the “Guidelines for Weather Conditions.”

(17) Judgment criteria should conform to the “Guidelines for Reactor Site Assessment.”

Hypothetical accidents

The same also applies to hypothetical accidents as in the case of major accidents except for the following items.

(10) It is assumed that all the fission products from the fuel rod are additionally released into reactor coolant in an instant, immediately after the main steam isolation valve is closed.

(13) It is assumed that one of the main steam isolation valves is not closed and steam leaks from the closed isolation valve. It is also assumed that the leak rate of the closed main steam isolation valve is a design value with a sufficient margin added and this leak rate is constant.

(15) After the accident occurs, it is assumed that the reactor pressure is maintained at a set pressure of the relief safety valve for a long time and leaks from the main steam system last for an indefinite time.

(Appendix II) (March 29, 2001)

The following matters will help evaluate the dose in accidents as a safety design assessment and in major accidents and hypothetical accidents as a site assessment.

1. Assessment of the dose in accidents

1.1 Assessment of the effective dose by external exposure

1.1.1 Effective dose attributable to radioactive substances released into the atmosphere

The effective dose attributable to gamma rays from radioactive clouds caused by radioactive substances released into the atmosphere should be evaluated on the basis of the relative dose using the air kerma attributable to radioactive substances, in accordance with the "Guidelines for Weather Conditions." The conversion coefficient from the air kerma to the effective dose should be 1 Sv/Gy.

In addition, if a process where radioactive substances are released into the atmosphere together with high-temperature and high-pressure reactor coolant is assumed to be occurring and exposure to steam clouds containing radioactive substances needs to be considered, the formation and moving speed of the steam clouds should be evaluated on the safe side.

The effective dose by external exposure to beta rays is excluded from the assessment because it is not significant compared with that by external exposure to gamma rays.

1.1.2 Effective dose by radioactive substances in the building of a reactor facility

The effective dose attributable to direct gamma rays and skyshine gamma rays attributable to radioactive substances released into the building of a reactor facility should be evaluated with due consideration given to the location, shielding structure, and geographical features of the facility and other factors. The conversion coefficient from the air kerma to the effective dose should be 1 Sv/Gy.

If it is obvious that neither direct nor skyshine dose significantly contributes to the effective dose attributable to a relevant accident, their assessment can be omitted.

1.2 Assessment of effective dose by internal exposure

The effective dose by inhalation of iodine released into the atmosphere should be evaluated from the following equation on the basis of the relative concentration of iodine in the surface air and iodine 131 equivalent amount, in accordance with "Guidelines for Weather Conditions." Note that the values of a one-year old child shown in Table 1 are used as parameters for the calculation.

Effective dose = K_{He} ・ M ・ Q_e ・ (x/Q)

K_{He}: Effective dose coefficient of a child by inhalation of I-131
M: Child's breathing rate
Q_e: Amount of released iodine (I-131 equivalent amount)
(x/Q): Relative concentration

Note that the breathing rate should be selected according to how and how long iodine is released.

I-131 equivalent amount Q_e in this case refers to a sum of values obtained by multiplying the amount of each isotope by the ratio of the effective dose coefficient of each isotope of iodine to the I-131 effective dose coefficient. It is calculated from the following equation.

$$Q_e = \sum_i \left(K_{Hi} / K_{He}\right) \bullet Q_i$$

K_{HI}: Child's effective dose coefficient by inhalation of radionuclide i

Q_i: Amount of radionuclide i released

2. Assessment of the dose in major accidents and hypothetical accidents

2.1 Assessment of the dose for the whole body

2.1.1 Dose attributable to radioactive substances released into the atmosphere

The dose for the whole body attributable to gamma rays from radioactive clouds caused by radioactive substances released into the atmosphere should be evaluated on the basis of the relative dose using the air kerma attributable to radioactive substances, in accordance with the “Guidelines for Weather Conditions.” The conversion coefficient from the air kerma to the dose for the whole body should be 1 Sv/Gy.

If a process where radioactive substances are released into the atmosphere together with high-temperature and high-pressure reactor coolant is assumed to be occurring and exposure to steam clouds containing radioactive substances needs to be considered, the formation and moving speed of the steam clouds should be evaluated on the safe side.

2.1.2 Dose by radioactive substances in the building of a reactor facility

The dose for the whole body attributable to direct gamma rays and skyshine gamma rays attributable to radioactive substances released into the building of a reactor facility should be evaluated with due consideration given to the location, shielding structure, and geographical features of the facility and other factors. The conversion coefficient from the air kerma to the dose for the whole body should be 1 Sv/Gy.

If it is obvious that neither direct nor skyshine dose significantly contributes to the dose for the whole body attributable to a relevant accident, their assessment can be omitted.

2.2 Assessment of thyroid dose

The thyroid dose by inhalation of iodine released into the atmosphere should be evaluated from the following equation on the basis of the relative concentration of iodine in the surface air and I-131 equivalent release amount, in accordance with the “Guidelines for Weather Conditions.” Note that as parameters for the calculation the values of a one-year old child and an adult shown in Table 2 are used for a major accident and a hypothetical accident, respectively.

Thyroid dose = $K_{Te} \cdot M \cdot Q_e \cdot (x/Q)$

K_{Te}: Dose coefficient related to the thyroid equivalent dose by inhalation of I-131 (3.2×10^{-6}Sv/Bq for a child and 3.9×10^{-7}Sv/Bq for an adult)

M: Breathing rate

Q_e: Amount of released iodine (I-131 equivalent amount)

(x/Q): Relative concentration

Note that the breathing rate should be selected according to how and how long iodine is released.

I-131 equivalent amount Q_e in this case refers to a sum of values obtained by multiplying the amount of each isotope by the ratio of the dose coefficient of each isotope of iodine to the I-131 thyroid equivalent dose coefficient. It is calculated from the following equation.

$$Q_e = \sum_i \left(K_{Ti} / K_{Te}\right) \bullet Q_i$$

K_{TI}: Thyroid equivalent dose coefficient by inhalation of radionuclide i

Q_i: Amount of radionuclide i released

2.3 Assessment of an integrated value of the whole-body dose

The population (demographic) integrated value of the whole-body dose should be evaluated within a range at an angle of 30 degrees in a horizontal direction around a reactor facility. It is necessary to select a range at an angle of 30 degrees around a reactor facility so that the population (demographic) integrated value is the maximum.

Table 1 Parameter and others used to evaluate the effective dose by iodine

Parameter and others	Symbol	Unit	Value
Child's effective dose coefficient 1) by inhalation of radionuclide1)	K_{Hi}	MSv/Bq	I-131: 1.6×10^{-4} I-132: 2.3×10^{-6} I-133: 4.1×10^{-5} I-134: 6.9×10^{-7} I-135: 8.5×10^{-6}
Child's breathing rate1)	M	m^3/h	0.31 (Active)
		m^3/d	5.16 (Daily mean)

Table 2 Parameter and others used to evaluate the thyroid dose by iodine

Parameter and others	Symbol	Unit		Value
Dose coefficient 1) related to the thyroid equivalent dose by inhalation of radionuclide 1)	K_{Ti}	MSv/Bq	Child	I-131: 3.2×10^{-6} I-132: 3.8×10^{-8} I-133: 8.0×10^{-7} I-134: 7.3×10^{-9} I-135: 1.6×10^{-7}
			Adult	I-131: 3.9×10^{-7} I-132: 3.6×10^{-9} I-133: 7.6×10^{-8} I-134: 7.0×10^{-10} I-135: 1.5×10^{-8}
Breathing rate 1)	M	m^3/h	0.31 (When a child is active) 1.2 (When an adult is active)	
		m^3/d	5.16 (Daily mean for a child) 22.2 (Daily mean for an adult)	

Reference

1) Age-dependent Doses to Members of the Public from Intake of Radionuclides: Part 4 Inhalation Dose Coefficients, ICRP Publication 71 (1995)

Comments on Appendix (August 30, 1990)

Appendix 1 shows concrete events to be evaluated in a safety design assessment, concrete conditions to be referred to in analyzing these events, and how to apply judgment criteria. On the basis of Appendix 1, referential matters to properly interpret the requirements for analysis, as well as the purport of postulated concrete events to be evaluated are shown below.

1. General matters

(1) The terms used in Appendix I are defined as follows:

1) "Rated output with a sufficient margin added" refers to a maximum thermal output of a reactor applied for with an appropriate margin allowed for measuring errors and other factors. This margin may vary, depending on the design of the measuring system and other factors, and must not be less than at least 2% of the maximum thermal output.

2) “To have been in operation long enough” refers to when various states that affect the analysis of power distribution in the reactor, such accumulation of fission products, temperature distribution, etc. have reached approximate equilibrium, with due consideration given to the operating cycle and other factors.
3) “Cold” refers to the state where the temperature of the reactor coolant is less than 100℃.
4) “To be in output-constant operation” refers to the state where the reactor is in operation at an output at which steam can be continuously supplied to the turbine.

(2) On the basis of the purport of assumption (1) for the safety functions in II.4.2 in the Comments, the following mitigation functions belonging to MS-3 can be expected in analyzing an abnormal transient change during operation.

1) Mitigation functions of which the function can be expected in the analysis of an accident.
2) The following mitigation functions have been shown to be sufficiently reliable from past experience.
   PWR: Turbine trip function and main steam relief valve closing function
   BWR: Turbine trip function, turbine bypass valve function, recirculation pump trip function, and relief valve function of a relief safety valve
3) Other mitigation functions of which the functions can be expected in Appendix 1 and Comments on Appendix.

2. Analysis of abnormal transient changes during operation

2.1 Abnormal change in reactivity in the core or power distribution

2.1.1 Abnormal withdrawal of a control rod when the reactor is started (PWR, BWR)

This event is assumed to be occurring in order to check that the reactivity worth and driving speed of a control rod are properly designed in relation to the characteristics of the reactor and the functions of the safety protection system and other systems. On the basis of the purport of this assumption, if the interlock related to the operation of the control rod can be regarded as an “on-duty” device throughout the occurrence of the event, the functions of the system and device can be expected. Note that in a design in which the reactor can be operated with the system and device detached, the validity of the assumption that the functions can be expected must be proved when the event is assumed to be occurring. (3) and (4) in I. 2.1.1 in Appendix I apply to a current typical design. If the design is changed, the control rod to be withdrawn, reactivity worth, etc. need to be properly selected according to the design.

2.1.2 Abnormal withdrawal of a control rod during output-constant operation (PWR, BWR)

This event is assumed to be occurring in order to check that the characteristics of the reactor and the functions of the safety protection system and other systems are properly designed when reactivity is added owing to an abnormal withdrawal of a control rod during output-constant operation. The interlock related to the operation of the control rod is handled in the same way as described in 2.1.1. Also, the operational status including the reactor power before this event occurs must be decided so that all operational modes allowable in design terms can be included.

In this event, a strict analysis is not necessarily made if the reactivity insertion rate is great. Therefore, it must be proved that the reactivity insertion rate is a value that makes the analysis the strictest within the scope of the design value. If the smaller the reactivity insertion rate is, the stricter the analysis is made, the assessment method on the assumption that the output varies quasi-statically is adequate.

2.1.3 Drop or inconsistency of a control rod (PWR)

This event is assumed to be occurring in a PWR in order to evaluate the effect of an abnormal power distribution in the core. It is assumed that the following two cases represent this condition.

In the first case, a control rod cluster in the “all-withdrawal position” drops onto the core when the reactor is in operation at a rated output (including a proper margin. The same applies below.) owing to a failure of the control rod drive system, resulting in the “all-insertion position.” At this moment, if the reactor power has been automatically controlled, the drop of the control rod will temporarily decrease the power and then the control rod will be automatically withdrawn, causing the power to return to its normal state. Also, if the power is manually controlled, the control rod will not be withdrawn. The effect when the power distribution is distorted in this condition is evaluated.

In the second case, when all of the control rods in a control rod cluster bank should be at the insertion limit, it is assumed that one of the control rod clusters remains in the “all-withdrawal position” owing to a failure of the control rod drive system. The effect of the distortion of power distribution, which occurs in this inconsistent condition, is evaluated.

2.1.4 Abnormal dilution of boron in reactor coolant (PWR)

Boron in the primary coolant plays an important role in a PWR for controlling and stopping the reactor’s reactivity. This event is assumed to be occurring in order to evaluate the effect of the insertion of reactivity attributable to a change in the concentration of boron. On the basis of the purport of this assumption, it is assumed that demineralized water is injected from the chemical and volume control system into primary coolant owing to a failure of the control circuit. At this moment, if the reactor power is automatically controlled, a control rod will be automatically inserted and the reactivity stop margin by the control rod will decrease as boron in the primary coolant is diluted. On the other hand, if the reactor power is manually controlled, the power will rise. The effect in both cases should be evaluated. If this event progresses slowly and the operator’s correction of the event can be expected with a high reliability while the event is in progress, it can be taken into consideration. In this case, requirement (3) as stated in II.5.2 in the guidelines must be satisfied.

2.2 Abnormal change in generation and removal of heat in the core

2.2.1 Partial loss of flow of reactor coolant (PWR, BWR)

This event is assumed to be occurring in order to check that the characteristics of the reactor and the design of the safety protection system are appropriate if there is an abnormal decrease in the flow of reactor coolant during the output-constant operation. On the basis of the purport of this assumption, it is hypothesized as a typical event that the driving power of a reactor coolant pump is lost. This is only a hypothesis in light of the purport of the assumption of the event. If the operation of the safety protection system is expected, it must be assumed that the system is started by a decrease in the flow of reactor coolant or in the number of revolutions of the pump motor.

If the reactor is so designed that the flow of reactor coolant can be controlled by a valve, closing this valve should be included in the analysis.

2.2.2 Accidental activation of the shutdown loop of the reactor coolant system (PWR, BWR)

This event is assumed to be occurring in order to check that the characteristics of the reactor and the design of the safety protection system are appropriate if a flow of low-temperature reactor coolant into the core inserts reactivity. On the basis of the purport of this assumption, it is postulated that some reactor coolant pumps that are out of operation while the reactor is in operation are accidentally started for some reason or other.

It is assumed that all of the pumps which may be activated at the same time owing to a single failure are accidentally started, with due consideration given to the design of the pump control system, and the analysis conditions must be established so that the analysis is made the strictest in combination with an initial output.

2.2.3 Loss of the external power source (PWR, BWR)

This event is assumed to be occurring in order to check the validity of the design of the emergency power and emergency power system in the station. For this reason, even if the generator in the station is so designed that its individual operation is possible, the assumption of this event must not be omitted. Coast down power of the main generator after a turbine trip must not be expected. It is quite natural to allow time to activate the emergency power. In particular, it must be proved that the design is adequate concerning the bus transfer, the sequence of connecting devices to the emergency power system, and the allowance of time for the connection, etc.

2.2.4 Loss of the flow of main feedwater (PWR)

The core cooling capacity in a PWR decreases owing to a failure of the secondary coolant system. Among these abnormal events, this event is assumed to be occurring in order to evaluate the effect of the decrease in the cooling capacity attributable to the feedwater system. It is assumed as a typical event that feedwater to all the steam generators is stopped owing to a failure of the main feedwater pump, condensate pump, or feedwater control system, resulting in a decrease in the heat removal capacity of the steam generators.

2.2.5 Abnormal increase in steam load (PWR)

2.2.6 Abnormal depressurization of the secondary coolant system (PWR)

2.2.7 Excess feedwater to the steam generator (PWR)

The three events above are assumed to be occurring in a PWR in order to evaluate the effect of the insertion of reactivity owing to a failure of the secondary coolant system. Among them, the events described in 2.2.5 and 2.2.6 are events in which the temperature of the secondary coolant decreases owing to an abnormal increase in the amount of steam flow from the steam generator, causing the temperature of the primary coolant to decrease, resulting in the insertion of reactivity. As an initial condition of these events, the reactor should be assumed to be in output-constant operation or out of operation at a high temperature. Also, the event described in 2.2.7 is assumed to be occurring if the amount of feedwater flow abnormally increases, resulting in a similar condition. In these events, fuel and the soundness of the reactor coolant pressure boundary are mainly evaluated. In addition to them, it must be confirmed as described in 2.2.6 that the reactor can be returned to a state of subcriticality without fail if it reaches a state of criticality.

2.2.8 Feedwater heating loss (BWR)

This event is assumed to be occurring in a BWR in order to evaluate the effect of the insertion of reactivity caused by an abnormal condition of the feedwater system. On the basis of the purport of this assumption, it is postulated that the functions of the feedwater heater are lost, causing the feedwater temperature to decrease, resulting in an increase in the subcooling at the reactor coolant core inlet. Since the phenomenon occurrence speed may vary, depending on the change in the subcooling, reactor coolant flow rate, etc., the initial output needs to be selected so that the analysis results are made the strictest.

2.2.9 Malfunction of the reactor coolant flow control system (BWR)

ABWR is designed so that its output is controlled by the recirculation flow of the reactor coolant. This event is assumed to be occurring in order to evaluate the effect of an abnormal increase in recirculation flow and an insertion of reactivity caused by a malfunction, operation error, etc. of this flow control system. In analyzing the event, the recirculation flow change rate is assumed to be the maximum allowable value in terms of design. In this case, it is necessary to select whether the number of revolutions of one or more than one recirculation pump motor changes, depending on the design of the recirculation flow control system. The same applies to designs in which the flow is changed not by the number of revolutions of the pump motor(s) but by the opening of the valve.

2.3 Abnormal change in reactor coolant pressure or stockpile

2.3.1 Loss of load (PWR, BWR)

This event is assumed to be occurring in order to evaluate the effect of a sudden decrease in steam flow to the turbine owing to a loss of load. In particular, reactivity is likely to be inserted into a BWR as the pressure rises owing to a sudden decrease in steam flow. In order to properly evaluate this point, a malfunction of the turbine bypass valve is also included in the analysis. If the reactor is provided with a selective control rod function, its functions can be expected.

2.3.2 Abnormal depressurization in the reactor coolant system (PWR)

This event is assumed to be occurring in order to evaluate the effect of fluctuations in the pressure of PWR coolant. On the basis of the purport of this assumption, it is postulated that the valve for controlling the pressure opens owing to a failure while the reactor is in operation at a rated output. Either pressurizer relief or spray valve (if other similar valves have been installed, depending on the design, these valves are included), which decreases the reactor pressure the most is assumed to be defective. If a main valve that is automatically closed is installed on the upstream side of the valve assumed to be defective, its functions can be expected with sufficient time allowed for its operation.

2.3.3 Accidental activation of the emergency core cooling system during output-constant operation (PWR)

This event is assumed to be occurring in order to evaluate the effect of an accidental activation of the ECCS while a PWR is in output-constant operation. On the basis of the purport of this assumption, it is postulated that the high-pressure injection system in the ECCS is activated, causing low-temperature cooling water to be injected into the primary coolant system while the reactor is in operation at a rated output.

2.3.4 Accidental closing of the main steam isolation valve (BWR)

This event is assumed to be occurring in order to evaluate the effect when the steam flow decreases owing to a failure of the isolation control valve, resulting in a rise in the reactor pressure. On the basis of the purport of this assumption, it is postulated that the main steam isolation valve (all the relevant valves if there are two or more valves that are simultaneously closed owing to a single failure) is closed in the shortest time while the reactor is in operation at a rated output.

2.3.5 Failure of the feedwater control system (BWR)

This event is assumed to be occurring in BWR in order to evaluate the effect when there is an abnormal increase in the flow of feedwater owing to a failure of the feedwater control system, resulting in an increment in reactivity. In analyzing the event, it is assumed that the reactor coolant recirculation system is in manual operation as in the case described in 2.2.8 and the reactor output at the occurrence of the event is selected so that the analysis results will be the strictest in light of the judgment criteria.

2.3.6 Failure of the reactor pressure control system (BWR)

In a BWR, it is important to control the reactor pressure because a change in the void fraction in the core greatly affects reactivity. Considering this characteristic of a BWR, this event is assumed to be occurring in order to evaluate the effect when there is a sudden change in the reactor pressure owing to a failure of the pressure control system. On the basis of the purport of this assumption, it is postulated that a wrong signal is issued owing to a failure of the pressure control system, resulting in a sudden change in the main steam flow. It is assumed that the reactor output, the operational status of the recirculation flow control system, etc. are selected so that the analysis results will be the strictest in light of the judgment criteria.

2.3.7 Total loss of the feedwater flow (BWR)

This event is assumed to be occurring in a BWR in order to evaluate the effect when there is a total loss of feedwater flow owing to a failure of the feedwater control system. On the basis of the purport of this assumption, it is postulated that the feedwater pump trips and the feedwater flow is totally lost within the time with the pump's inertia allowed.

3. Analysis of accidents

3.1 Loss of reactor coolant or remarkable change in core cooling

3.1.1 Loss of reactor coolant (PWR, BWR)

This event is assumed to be occurring in order to check that the reactor is designed so that core cooling is ensured when the reactor coolant flows out of the system owing to damage to the pipes of the reactor coolant system. Incidentally, when reactor coolant is lost, radioactive substances may be released into the environment or the internal pressure of the reactor containment vessel may rise. These two events are evaluated in 3.3.5 and 3.4.1. The analysis conditions and judgment criteria for this event should conform to the provisions in the ECCS Performance Evaluation Guide.

3.1.2 Loss of the flow of reactor coolant (PWR, BWR)

This event is assumed to be occurring in order to evaluate the effect when there is a significant decrease in the flow of reactor coolant. On the basis of the purport of this assumption, it is postulated that all of the pumps that circulate reactor coolant lose driving power. Since this event is selected as a typical event that significantly decreases the flow of reactor coolant, when the operation of the safety protection system is expected, it must be postulated that it is activated, in principle, by a decrease in the flow of reactor coolant or a decrease in the RPM of the pump's motor. Note that if the reactor is designed properly so that events that significantly decrease the flow of reactor coolant, except a loss of pump driving power, hardly occur, it would be acceptable that a loss of pump driving power stands for issuing a signal to activate the safety protection system.

Incidentally, if a loss of the driving power of all the reactor coolant pumps is taken into consideration in 2.2.1, this event does not need to be evaluated.

3.1.3 Sticking of the shaft of a reactor coolant pump (PWR, BWR)

This event is assumed to be occurring in order to evaluate the effect when there is a sudden decrease in the flow of reactor coolant. On the basis of the purport of this assumption, it is postulated that the pivot of the reactor coolant pump is stuck while the reactor is in operation at a rated output, resulting in an instantaneous shutdown.

3.1.4 Main feedwater pipe rupture (PWR)

This event is assumed to be occurring in a PWR in order to evaluate the effect when the feedwater system pipe is ruptured, resulting in a loss of secondary coolant. On the basis of the purport of this assumption, it is postulated that a main feedwater pipe is ruptured at both ends in an instant while the reactor is in operation at a rated output.

3.1.5 Main steam pipe rupture (PWR)

In a PWR, there are no radioactive substances, in principle, in the secondary coolant system; therefore, should the main steam pipe be ruptured, radioactive substances would not be released into the environment immediately. It is possible to cool down the core without any problems by using a sound steam generator and main steam pipe. However, this event is assumed to be occurring in order to evaluate the effect when steam flows out owing to a break of the secondary coolant system, causing the temperature of primary coolant to decrease, resulting in an increment in reactivity. Its contents are the same as those described in 2.2.6.

If it is proved that the reactor is designed so that it does not reach a state of criticality even if the main steam pipe rupture increments reactivity, the analysis of this event can be omitted. As judgment criteria, II.4.2 (1) and (3) should apply after it is confirmed that the reactor can be put back to a state of subcriticality without fail if it goes critical on the assumption that this event occurs. In this case, II.4.1 (1), (2), and (4) in the guidelines must be satisfied as the criterion for confirming that this event will not progress while the state of criticality continues. Also, it needs to be presumable with ease that other abnormal conditions will not occur until the reactor is put back to a state of subcriticality.

3.2 Abnormal insertion of reactivity or a sudden change in reactor output

3.2.1 Control rod ejection (PWR)

This event is assumed to be occurring in order to evaluate the effect when there is a sudden insertion of reactivity owing to damage to the control rod drive system or control rod housing. On the basis of the purport of this assumption, it is postulated that the control rod housing is damaged, causing a control rod cluster with a maximum reactivity worth to be quickly ejected from the all-insertion position or control rod insertion limit position. The reactor is assumed to be in or close to a state of criticality, and the initial condition must be selected so that the analysis results will be made the strictest. When the reactivity inserted based on this assumption is 1 dollar or more, concrete analysis conditions and judgment criteria should conform to the provisions in the “Guidelines for Reactivity Insertion Event Assessment.” Note that the primary coolant pressure will decrease owing to damage to the control rod housing, causing a flushing, resulting in a negative reactivity, but this effect will not be taken into consideration in analyzing this event.

3.2.2 Control rod drop (BWR)

This event is the same as that described in 3.2.1. Since a current BWR is designed so that the control rod is driven from under the core, the course of the event is considered to be different from that described in 3.2.1. A control rod with the maximum reactivity worth is assumed to be dropping. In this case, if it is proved that the rod worth minimizer continues to function without fail before the event occurs, its functions can be expected. Concrete analysis conditions and judgment criteria should conform to the provisions in the “Guidelines for Reactivity Insertion Event Assessment.”

3.3 Abnormal release of radioactive substances into the environment

3.3.1 Damage to a gaseous radioactive waste disposal and treatment facility (PWR, BWR)

There are various radioactive substances in a reactor facility, and gaseous radioactive waste, among others, is very likely to be released into the environment. In other words, gaseous radioactive substances that are being accumulated in or are in process by a gaseous radioactive waste disposal and treatment facility may be released into the environment owing to damage to the facility. From this point of view, this event is assumed to be occurring in order to evaluate the effect when part of a gaseous radioactive waste disposal and treatment facility is damaged. Analysis conditions should be selected so that the analysis results will be made the strictest, with due consideration given to the amount of gaseous radioactive waste accumulated in terms of design in each place of the facility, temperature, pressure, isolation time, etc.

3.3.2 Main steam pipe rupture (BWR)

In a BWR, part of the reactor coolant is directly transformed into steam, which is led to the turbine outside the reactor containment vessel through the main steam pipe. If steam is released outside the vessel owing to damage to this pipe, radioactive substances in the steam will be directly released outside the vessel. This event is assumed to be occurring in order to evaluate the effect when this rupture occurs.

In this event, it is postulated that the main steam pipe is instantly ruptured at both ends outside the reactor containment vessel. Firstly steam and then a two-phase flow is released from the break of the pipe. The functions of devices such as the flow controller, a main steam isolation valve, etc. can be expected, with the proper safety margin allowed, in calculating the flow rate of the reactor coolant. In the early stage of this event, critical flow is likely to occur in the flow controller and then another critical flow may occur at the main steam isolation valve as it closes. In this case, the flow control effect by the main steam isolation valve should be ignored until the critical flow occurs at the valve.

The amount of radioactive substances released into the environment should be calculated in the following way.

Before the event occurs, the reactor coolant contains radioactive substances at a concentration equivalent to a maximum allowable concentration in terms of design. The radioactive substances will be released into the environment as the reactor coolant flows out owing to the occurrence of this event. Also, with a decrease in the pressure of the reactor coolant, additional radioactive substances will be released from the fuel rods into the reactor coolant. A further release of radioactive substances from the fuel rods assumed to have already leaked is taken into consideration. It is recognized as adequate to determine the amount on the basis of the operation experience and other factors. It is assumed that rare gas is instantly transformed into the gas phase including bubbles in the reactor coolant and does not dissolve in the liquid phase.

Some of the main steam isolation valves are closed to prevent steam from being released. It is assumed that one isolation valve is not closed and there is a leak determined by the temperature and pressure of the reactor coolant as well as by a design leak from the closed isolation valves. Radioactive substances released owing to this leak should also be taken into consideration.

In view of the fact that this event is characterized by a direct release of radioactive substances from the reactor cooling system to the outside of the reactor containment vessel, it must be proved that another fuel rod is not damaged by this event.

3.3.3 Steam generator tube rupture (PWR)

In a PWR, the primary coolant is likely to flow into the secondary coolant system owing to damage to the heat-transfer pipe of the steam generator and to be released outside the reactor containment vessel through the main steam relief valve of the secondary coolant system, causing radioactive substances in the primary coolant to be released into the environment. This event is assumed to be occurring in order to evaluate the effect in such a case.

The concentration of the radioactive substances in the primary coolant before the event occurs is assumed to be the value obtained by using the failure rate of the fuel cladding postulated in terms of design. It is adequate to assume that rare gas and iodine are additionally released from the gap of a fuel rod with a defect postulated in terms of design in proportion to the depressurization rate of the primary coolant.

If an operator’s intervention is required to isolate the steam generator with the damaged heat-transfer pipe, enough time must be allowed for the operation.

In view of the fact that this event is characterized by the direct release of radioactive substances from the reactor cooling system to the outside of the reactor containment vessel, it must be proved that another fuel rod is not damaged by this event.

3.3.4 Drop of fuel assembly (PWR, BWR)

One of the places that have radioactive substances in a significant quantity, other than the reactor core in a reactor facility, is a spent fuel pit. From this point of view, this event is assumed to be occurring in order to evaluate the effect when radioactive substances are released from a fuel rod, particularly, while fuel is being handled. In analyzing this event, it is assumed that a fuel assembly which is being handled drops from the highest position in terms of operation owing to a failure of or damage to the fuel handling equipment, causing radioactive substances to be released from the gap of the fuel rod damaged from a drop impact.

The amount of rare gas and iodine released from the fuel rod is evaluated in two ways as shown in 3.3.5 and either of them are adequate.

It is required to assume that a fuel assembly drops inside the spent fuel pit of a PWR and an assembly drops from the core of a BWR because a case where the number of damaged fuel rods of the dropped fuel assembly becomes a maximum is selected, with due consideration given to a current typical design.

3.3.5 Loss of reactor coolant (PWR, BWR)

A loss of reactor coolant owing to damage to the reactor cooling system is not only a typical accident of a light water reactor from the viewpoint of a remarkable change in core cooling but also a critical event in light of the release of radioactive substances into the environment.

The same assumption applies to radioactive substances released into the reactor containment vessel, which have been present in the reactor coolant before this event occurs, as in the cases presented in 3.3.2 and 3.3.3. The same assumption also applies to the case where it is proved that additional fuel rods are not damaged as in the cases presented in 3.3.2 and 3.3.3. If it is proved that other fuel rods are damaged owing to this event, the amount of radioactive substances released must be properly estimated, depending on how they are damaged, on the assumption that the core cooling condition meets judgment criteria stated in 3.1.1.

Either of the following two assessment methods of the amount of rare gas and iodine released from a fuel rod are adequate. In the first method, the release of radioactive substances from the gap of the damaged fuel rod is evaluated with due consideration given to the power density, burn up, etc. In the second method, the amount that is not below the proportional value obtained by calculating 1% and 0.5%, respectively, of the total amount of rare gas and iodine from the power density ratio of a fuel rod, the rate of the number of damaged fuel rods to the total number of fuel rods in the core, and other factors is considered to be equal to the amount of radioactive substances released from the gap of the damage fuel rod.

The formation (composition) rate, deposition, and removal effect by spray water in the reactor containment vessel of organic iodine, among radioactive substances, are as described in Appendix 1.

After being treated by the annulus or the emergency ventilation system in the reactor building, gas including radioactive substances leaking from the reactor containment vessel is released from the stack. In this case, the seizing signal of the emergency ventilation system must be clarified, and at the same time it must be proved that the diffusion of radioactive substances until it starts to function is evaluated on the safe side with enough time allowed for achieving a negative pressure. For filter efficiency, the design value of the emergency ventilation system can be used from the time when it starts to function. If a PWR is designed so that the annulus can cover the entire external area of the reactor containment vessel, it can be assumed that the total amount of leak from the vessel is attributable to the annulus.

If the ECCS is operated in recirculation mode and water in the reactor containment vessel is led outside the vessel, it is assumed that recirculation water leaks from devices along the channel at a leak rate postulated in terms of design, and the contribution of this leak to the release of radioactive substances into the environment should be evaluated. Note that if it is not obvious that the leak of recirculation water significantly contributes to the effective dose caused by this event, the assessment can be omitted.

If it is obvious that neither direct dose nor skyshine dose significantly contributes to the effective dose caused by this event, the assessment can be omitted.

3.3.6 Ejection of control rod (PWR)

The ejection of a control rod is evaluated as a sudden reactivity insertion event as described in 3.2.1. In this case, it is assumed that the effects of flushing, depressurization, etc. of primary coolant caused by damage to the control rod housing are ignored. This accident, however, may cause damage to a fuel rod and primary coolant to flow, resulting in the release of radioactive substances into the environment. The effect of this release should be evaluated. The possibility that a fuel rod is damaged is evaluated on the basis of the analysis results presented in 3.2.1. Also, the assessment of an effective dose caused by the outflow of primary coolant should conform to the case presented in 3.3.5.

3.3.7 Dropping of control rod (BWR)

Fuel rods may be damaged from the dropping of the control rod as postulated in 3.2.2. This event is assumed to be occurring in order to evaluate the effect of radioactive substances upon the environment when they are released from the damaged fuel rods. AS stated in 3.2.2, it is required to select the reactor power and other factors when the event occurs so that the fuel enthalpy becomes the maximum. In this event, it is necessary to select the initial condition of the reactor when the event occurs so that the amount of damaged fuel rods becomes the maximum because the amount of radioactive substances contained in the damaged fuel rods is an issue. For this reason, the reactor operation histories by condition when the event occurs are specified (recorded).

3.4 Abnormal changes in the internal pressure, ambient, etc. of a reactor containment vessel

A reactor containment vessel is designed on the assumption that various events occur. The “Guidelines for Safety Design Review” requires a reactor containment vessel to withstand a load caused by those events and have the functions to decrease the concentration of radioactive substances released into the environment and to control the concentration of hydrogen or oxygen present in the vessel. For this reason, concrete contents of the events postulated to design the vessel and requirements in analyzing these events have been defined in 3.4.

The reactor containment vessel forms the last barrier to prevent the release of radioactive substances into the environment. In light of its importance, requirements for analyzing the events postulated to design the vessel are based on the considerations different from those in the case of other structures, systems, and devices.

Subsections 3.4.1, 3.4.2, and 3.4.3 define a rise in pressure and temperature, the generation of flammable gas, and the occurrence of dynamic loads, respectively, as the events postulated to design the vessel. The concentration of radioactive substances released into the environment is omitted in 3.4 because it is evaluated in the site assessment shown in II of Appendix I as well as in 3.3.5.

3.4.1 Loss of reactor coolant (PWR, BWR)

The loss of reactor coolant attributable to damage to the reactor cooling system is a typical cause of the occurrence of abnormal loads in the reactor containment vessel. The analysis conditions must be selected so that the pressure and temperature inside the vessel becomes the highest. In other words, it is necessary to set various conditions to select and analyze a pipe that is assumed to be rupturing and its break so that the energy flow rate into the vessel is the maximum. In concrete terms, the rupture is assumed to be occurring in the intake pipe of the reactor coolant pump in a PWR and a BWR with an external recirculation loop.

The functions of the cooling systems for the reactor containment vessel including a spray system can be expected when an accident occurs. In this case, a single failure is properly assumed to be occurring. Also, it must be assumed that the external power source cannot be used.

In the course of the event, water that has cooled the core once flows into the vessel. The contribution of energy generated in the core during the occurrence of the event as well as energy of reactor coolant flowing out owing to a pipe rupture to the pressure and temperature inside the vessel needs to be properly considered.

3.4.2 Generation of flammable gas (PWR, BWR)

In the 1979 TMI accident, hydrogen generated by a metal–water reaction was released into the reactor containment vessel and the mixture of the hydrogen with air ignites, resulting in a sudden combustion. Before the TMI accident, it had been widely known that a metal–water reaction generates hydrogen and radiolysis of water generates oxygen and hydrogen. For this reason, the “Guidelines for Safety Design Review” have established requirements for the flammability control system in terms of design. With all things including analysis techniques based on these requirements considered, the design validity assessment method of the flammability control system has been summarized in 3.4.2. In particular, for the generation of oxygen and hydrogen by radiolysis of water, radiolysis by radioactive substances that is more serious than the case presented in 3.1.1 should be taken into consideration. The reason why an assessment based on such a postulation is made is that the above-mentioned importance of the reactor containment vessel is taken into consideration.

3.4.3 Generation of dynamic loads (BWR)

When the reactor coolant is lost or a safety relief valve is activated, a dynamic load may occur in the suppression pool of the pressure suppression container used in a current BWR. Guidelines for the assessment method of such a dynamic load have been separately established. An assessment based on the guidelines must be made. It is necessary to confirm that stress caused by the dynamic load does not exceed allowable stress based on proper standards and criteria in judging the analysis results. Note that the design of detailed structures in the vessel may have not been completed at the stage of application for installation permission. In this case, if the basic policy of making a detailed design based on the above guidelines, criteria, etc. is confirmed, the design policy is adequate.

(Reference) Written decision dated August 30, 1990 by the Nuclear Safety Commission

Guidelines for Safety Assessment Review of Light Water Reactor Power Stations

The Nuclear Safety Commission examined the contents of the report on the issue above submitted by the Special Committee on Nuclear Safety Standards on July 24, 1990. As a result, the commission established the “Guidelines for Safety Assessment Review of Light Water Reactor Power Stations” as shown in the attachment.

The Nuclear Safety Commission has used the “Guidelines for Safety Assessment Review of Light Water Reactor Power Stations” decided by the commission on September 29, 1978 (partially revised by the Nuclear Safety Commission on March 27, 1989) in order to conduct safety reviews of light water reactor power stations. The commission decided to replace it with new “Guidelines for Safety Assessment Review of Light Water Reactor Power Stations,” which are attached to this document.

In response to new findings and experience, the guidelines will be reexamined as and when necessary.