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

Appendix II

(March 29, 2001)

The matters, which should be noted in the dose evaluations for "accidents" in the case of safety design evaluation and for "major accidents" and "hypothetical accidents" in the case of siting evaluation, are shown in the following.

1. Evaluation of dose equivalent under the "accidents"

1.1 Evaluation of effective dose due to external exposure

1.1.1 Effective dose due to radioactive materials released to the atmosphere

The effective dose from the gamma ray from radioactive clouds due to radioactive materials released to the atmosphere shall be evaluated based on the relative dose using the air kerma caused by radioactive materials in accordance with the "Meteorological Guide." The conversion factor from the air kerma to effective dose shall be 1 Sv/Gy.

Moreover, when it is required to take into account the exposure due to the steam clouds containing radioactive materials because of the process in which the radioactive materials are released to the atmosphere with the high-temperature and high-pressure reactor coolant is plausible, the formation and moving velocity of the steam clouds shall be conservatively evaluated.

In addition, the effective dose due to the external exposure from the beta ray is not subject to this evaluation since it is not a significant value compared with that due to the gamma ray.

1.1.2 Effective dose due to radioactive materials in the nuclear reactor facility buildings.

The effective dose due to the direct gamma ray and skyshine gamma ray caused by radioactive materials released in the facility buildings shall be evaluated appropriately taking account of locations of the facilities, shielding structures, geographical conditions, etc. The conversion factor from the air kerma to effective dose shall be 1 Sv/Gy.

When it is apparent that the direct dose and skyshine dose do not make a significant contribution to the effective dose due to the accident concerned, their evaluations may be omitted.

1.2 Evaluation of effective dose due to internal exposure

The effective dose due to intake of iodine released to the atmosphere by inhalation shall be evaluated with the following formula based on the relative concentration of iodine in the air of the ground surface and the amount of I-131 equivalent in accordance with the "Meteorological Guide." In addition, the parameters etc. to be used for the calculation shall be the value for infants (1 year old) shown in Table 1.

Effective dose = K_{He} x M x Q_e ·x (χ / Q)
K_{He}: effective dose coefficient of infants by intake of I-131 by inhalation
M: respiratory coefficient of infants
$Q_{e:}$ amount of iodine released (amount of I-131 equivalent)
(χ / Q): relative concentration

The respiratory coefficient shall be selected according to the situation and duration of iodine release.

The I-131 equivalent Qe in this case means the summation of the ratio of the effective dose coefficient of each iodine isotope to the effective dose coefficient of I-131 isotopes multiplied by the amount of each iodine isotope and it shall be calculated with the following formula;

$Q_e = \Sigma_i (K_{Hi} / K_{He}) \times Q_i$
K_{Hi}: effective dose coefficient of infants by intake of nuclide i by inhalation
Q_i: amount of nuclide i released

2. Dose evaluation during the "major accidents" and "hypothetical accidents"

2.1 Evaluation of dose to the whole body

2.1.1 Dose due to radioactive materials released to the atmosphere

The dose to the whole body due to the gamma ray from radioactive clouds caused by radioactive materials released to the atmosphere shall be evaluated based on the relative dose using the air kerma due to radioactive materials in accordance with the "Meteorological Guide." The conversion factor from the air kerma to the dose to the whole body shall be 1 Sv/Gy.

Moreover, when it is required to take into account the exposure due to the steam clouds containing radioactive materials because of the process in which the radioactive materials are released to the atmosphere with the high-temperature and high-pressure reactor coolant is plausible, the formation and moving velocity of the steam clouds shall be conservatively evaluated.

2.1.2 Dose due to radioactive materials in the nuclear reactor facility buildings

The dose to the whole body due to the direct gamma ray and skyshine gamma ray caused by radioactive materials released in the facility buildings shall be evaluated appropriately taking account of locations of the facilities, shield structures, geographical conditions, etc. The conversion factor from the air kerma to the dose to the whole body shall be 1 Sv/Gy.

When it is apparent that the direct dose and skyshine dose do not make a significant contribution to the dose to the whole body due to the accident concerned, their evaluations may be omitted.

2.2 Evaluation of dose to the thyroid gland

The dose to the thyroid gland due to intake of iodine released to the atmosphere by inhalation shall be evaluated with the following formula based on the relative concentration of iodine in the air of the ground surface and the amount of I-131 equivalent released in accordance with the "Meteorological Guide." In addition, the parameters etc. used for calculation shall be the value for infants (1 year old) in case of the "major accidents" and for adults in case of the "hypothetical accidents" shown in Table 2.

The dose to the thyroid gland = $K_{Te} \times M \times Q_e \times (\chi / Q)$
K_{Te}: dose coefficient pertaining to the equivalent dose to the thyroid gland due to intake of I-131 by inhalation (for infants: 3.2×10^{-6} Sv/Bq and for adults: 3.9×10^{-7} Sv/Bq)
M: respiratory coefficient
Q_e: amount of iodine released (amount of I-131 equivalent)
(χ / Q): relative concentration

In addition, the respiratory coefficient shall be selected according to the situation and duration of iodine release.

Moreover, the I-131 equivalent Qe in this case means the summation of the ratio of the effective dose coefficient of each iodine isotope to the effective dose coefficient of I-131 isotopes multiplied with the amount of each iodine isotope and it shall be calculated with the following formula;

$Q_e = \Sigma_i (K_{Ti} / K_{Te}) \times Q_i$
K_{Ti}: dose coefficient pertaining to the equivalent dose to the thyroid gland due to intake of nuclide i by inhalation
Qi: amount of nuclide i released

2.3 Evaluation of cumulative value of dose to the whole body

The population cumulative value of the whole-body dose shall be evaluated in the range with the angle of 30 degrees in the horizontal direction around the nuclear reactor facility.

The range with the angle of 30 degrees in the horizontal direction around the nuclear reactor facility shall be selected so that the population cumulative value of the whole-body dose becomes the maximum.

Table 1 Parameters etc. to be used to evaluate the effective dose due to iodine

Parameter etc.	Symbol	Unit	Value
Effective dose coefficient of infants by intake of nuclide i by inhalation	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}
Respiratory coefficient of infants	M	M^3/h	0.31 (when active)
		M^3/d	5.16 (day average)

Table 2 Parameters etc. to be used to evaluate the dose to the thyroid gland due to iodine

Parameter etc.	Symbol	Unit	Value	
Dose coefficient pertaining to the equivalent dose to the thyroid gland by intake of nuclide i by inhalation	K_{Ti}	mSv/Bq	Infant	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}
Respiratory coefficient	M	M^3/h	0.31 (for infants when active)	
			1.2 (for adults when active)	
		M^3/d	5.16 (day average for infants)	
			22.2 (day average for adults)	

Appendix Commentary

(August 30, 1990)

The specific events to be evaluated in the safety design evaluation and the application methods of the specific conditions and criteria that should be taken into account when analyzing these events are shown in Appendix I. When actually analyzing the events according to those, the matters that are considered useful for appropriate interpretation of the purpose to assume the specific events to be evaluated and the requirements for the analysis are shown in the following.

1. General matters

(1) Interpretations of the term commonly used in Appendix I shall be as follows.

1) The "power a little lower than the rated power" means the power level in consideration for instrumentation error etc. added to the licensed maximum reactor thermal power. This margin is subject to change depending on the design of instrumentation systems etc., but it shall not be less than at least 2% of the maximum thermal reactor power level.

2) "Be in operation for a sufficiently extended period of time" means a state in which various kinds of state quantities that would affect the analysis, such as the power distribution, buildup status of fission products and temperature distribution in a reactor, are nearly in equilibrium conditions with taking into account its operating cycle, etc.

3) "Cold" means a state in which the temperature of the reactor coolant is below 100 degrees Celsius.

4) "During power operation" means a state of operation with a power level that can continuously supply steam to a turbine.

(2) In accordance with the intent of Assumption (1) on safety functions in II 4.2 of the Commentary, credit for the functions to mitigate abnormal consequences which belong to the following MS-3 can be taken in the analyses of "anticipated operational occurrences."

1) Taking credit for the functions is admitted in the analyses of the "accidents."

2) The following ones in which the reliability is considered sufficient from the experiences etc. up to now.

PWR: turbine trip function, closing function of block valves of main steam relief valves

BWR: turbine trip function, turbine bypass valve function, recirculation pump trip function, relief valve function of safety-relief valves

3) Taking credit for the functions is admitted in Appendix I and the Commentary.

2. Analysis of anticipated operational occurrences

2.1 Abnormal change in reactivity or power distribution in the core

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

This event is assumed to confirm that the reactivity worth, drive speed, etc. of control rods are designed appropriately in relation to the reactor characteristics and functions of safety protection system, etc. In view of the intent of this assumption, when the interlock etc. for control rod operation can be considered as so called "on-duty" equipment before and after the occurrence of the event, the functions of its system and equipment can be expected. However, when the control rods are so designed to be operable with the system and equipment separated, it shall be justified to assume that the functions can be expected in assuming the event. Parts (3) and (4) of 2.1.1 in Appendix I are provided for the current typical design, and when the design is changed, the control rods to be withdrawn and the reactivity worth, etc. shall be appropriately selected according to the design.

2.1.2 Abnormal withdrawal of control rods during power operation (PWR, BWR)

This event is assumed to confirm that the reactor characteristics and design of the safety protection system, etc. are appropriate when the reactivity is inserted due to control rods during power operation. Treatment of the interlocks etc. for control rod operation is the same as in 2.1.1. Moreover, the operational conditions including the power level of the reactor before the occurrence of the event shall be determined so as to envelop all operation modes allowed by the design.

In this event, the case with a larger reactivity insertion rate does not necessarily result in more severe consequences. Therefore, the reactivity insertion rate shall be justified by demonstrating that it results in the most severe consequences within the design. When the consequences become more severe with a smaller reactivity insertion rate, the evaluation method assuming a near static change of the power level is accepted as adequate.

2.1.3 Drop and inconsistency of control rods (PWR)

This event is assumed to evaluate the effect of anomalies in power distribution in a core of PWR. The following two cases are assumed to be representing such a state.

The first case is that one fully withdrawn control rod cluster drops to a reactor core due to a failure etc. of the control rod drive system during a rated power (including an appropriate margin, the same hereinafter) and is fully inserted. When the reactor power is automatically controlled at this time, the reactor power level will return to the original state due to automatic withdrawal of the control rods for power control after the reactor power level has once lowered due to a control rod drop. When the reactor power is manually controlled, the control rods would not be withdrawn. The effect of skewness in the power distribution produced in those states is evaluated.

The second case is to assume an inconsistent state and evaluate the effect of skewness in the power distribution produced in that state in which one control rod cluster of the bank remains in the fully withdrawn position due to a failure etc. of the control rod drive system when all of the control rods in one control rod cluster bank should be in the insertion limit.

2.1.4 Abnormal dilution of boron in the reactor coolant (PWR)

In PWR, boron in the primary coolant has an important role for reactivity control and shutdown of a reactor. This event is assumed to evaluate the effect of reactivity insertion due to a change in the boron concentration. In line with the intent of this assumption, pure water injection into the primary coolant is assumed from the chemical and volume control system due to a failure etc. of a control circuit. At this time, when the reactor power is automatically controlled, control rods are automatically inserted following dilution of the boron concentration in the primary coolant, and the reactivity shutdown margin with control rods decreases. When the reactor power is manually controlled, the power level increases. The effect of each case shall be evaluated. When the development of this event is slow and actions by operating personnel for its termination can be expected with a high reliability during the development of the event, the actions may be taken into consideration. In that case, the requirement of II 5.2(3) of the guide body shall be satisfied.

2.2 Abnormal change in heat generation or heat removal in the core

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

This event is assumed to confirm that the reactor characteristics and design of safety protection system, etc. are appropriate when the flow of reactor coolant decreases abnormally during reactor power operation. In line with the aintent of this assumption, a loss of the driving power supply of one reactor coolant pump is assumed as a typical event, but this is an assumption only in view of the intent to assume such an event, and when an operation of the safety protection system is expected, it should be assumed that its operation is actuated in principle by a decrease in the reactor coolant flow or number of revolutions of the pump.

In addition, for a design to control the reactor coolant flow with a valve, the closure of the valve shall be also included in the analysis.

2.2.2 Inadvertent start-up of a shutdown loop of a reactor coolant system (PWR, BWR)

This event is assumed to confirm that the reactor characteristics and design of the safety protection system, etc. are appropriate when the reactivity is inserted as the low-temperature reactor coolant flows into the reactor core. In line with the intent of this assumption, it is assumed that several reactor coolant pumps are out of service, and are erroneously started by a certain cause during reactor operation.

For the pumps erroneously started, the conditions for the analysis shall be determined so as to result in the most severe consequences within the total number of the pumps that can start simultaneously due a single failure etc. taking account of the pump control system design etc. and in its combination with an initial power level.

2.2.3 Loss of off-site power (PWR, BWR)

This event is assumed to confirm the adequacy of design of the on-site emergency power supply and its system. Therefore, even when the main generator is, for an example, designed so as to be capable of operating alone on site, an assumption of this event shall not be omitted. Moreover, the coast-down power of the main generator after a turbine trip shall not be expected, either. It is natural to take a time margin for the startup of the emergency power supply, but the design adequacy shall be demonstrated especially for bus switching, the connection order of components to the emergency power supply system and its time margin, etc.

2.2.4 Loss of main feed water flow (PWR)

This event is assumed to evaluate the effect of the abnormal event caused by the feedwater system out of the abnormal events, which decrease the core cooling capability, due to anomalies in the secondary cooling system in PWR. As a typical event, it is assumed that the heat removal capability of steam generators decrease due to a loss of feedwater to all of them caused by a failure, etc. of the main feedwater pumps, condensate pumps or the feed water control system.

2.2.5 Abnormal increase of steam load (PWR)

2.2.6 Abnormal depressurization of the secondary cooling system (PWR)

2.2.7 Excessive feedwater supply to the steam generator (PWR)

Each of the above three events is assumed to evaluate effects of reactivity insertion due to anomalies in the second cooling system in PWR. Among these, 2.2.5 and 2.2.6 are events that an abnormal increase of the steam flow from the steam generator due to a malfunction of the turbine bypass valves etc. causes the secondary coolant temperature decreases, resulting in the primary coolant temperature decrease and thereby reactivity insertion, and as their initial conditions, the reactor is assumed to be in power operation or hot shutdown. Moreover, 2.2.7 assumes an event of an abnormal feedwater flow increase resulting in the same consequence. For these events, the integrity of the fuel and the reactor coolant pressure boundary is mainly to be evaluated, but in addition to this, 2.2.6 requires to confirm that when the reactor goes critical, it would go back to a sub-critical state without fail.

2.2.8 Loss of feedwater heater (BWR)

This event is assumed to evaluate the effect of reactivity insertion due to anomalies of the feed water system in BWR. In line with the intent of this assumption, it is assumed that the feed water temperature decreases due to a loss of the feedwater heater function, resulting in an increase of sub-cooling of the reactor coolant at the core inlet. It is required to select the initial power level etc. so that the most severe consequence results since the speed of phenomenon might depend on the change in the sub-cooling and reactor coolant flow, etc.

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

Since BWRs are designed to control the power level with the recirculation flow of the reactor coolant, this event is assumed to evaluate the effect of reactivity insertion by an abnormal increase of the recirculation flow due to a malfunction and inadvertent operation, etc. of the flow control system. The recirculation flow change rate shall be the maximum one intended in the design for the analysis. In this case, it is required to select whether the number of revolutions of one recirculation pump changes or that of two or more pumps change depending on the design of the recirculation flow control system. Furthermore, for a design in which flow change is done by the valve opening and is not by the number of revolutions of pumps, the same manner as mentioned above applies.

2.3 Abnormal change in reactor coolant pressure or reactor coolant inventory

2.3.1 Loss of load (PWR, BWR)

This event is assumed to evaluate the effect of rapidly decreasing steam flow to the turbine due to a loss of load, etc., Since the reactivity could be inserted, especially in BWR, to the reactor accompanying a rapid pressure increase due to a decrease of steam flow, in order to appropriately evaluate this issue, the case for turbine bypass valve failure is included in the analysis. In addition, when the function of selected control rods are provided, the function can be expected.

2.3.2 Abnormal depressurization of the reactor coolant system (PWR)

This event is assumed to evaluate the effect of a change in the reactor coolant pressure in PWR. In line with the intent of this assumption, it is assumed that a pressure control valve opens due to its failure when the reactor is in operation at the rated power level. A valve to be failed is assumed to be either a pressurizer bypass valve or a pressurizer spray valve (including other valves with a similar effect when they are provided depending on the design), which has the maximum effect on reduction of the reactor pressure. When a block valve that closes automatically is provided upstream of the valve to be failed, credit is taken for the function of the valve with a sufficient time margin to activate.

2.3.3 Inadvertent start-up of the emergency core cooling system during power operation (PWR)

This event is assumed to evaluate the effect of inadvertent start-up of the ECCS during power operation in PWR. In line with the intent of this assumption, it is assumed that the high pressure injection system of the ECCS starts and injects the low-temperature cooling water to the primary cooling system while the reactor is in operation at the rated power.

2.3.4 Inadvertent closure of a main steam isolation valve (BWR)

This event is assumed to evaluate the effect of reactor pressure increase caused by steam flow decrease due to a failure etc. of the isolation valve control system in BWR. In line with the intent of this assumption, it is assumed that one main steam isolation valve (when there are 2 valves or more that simultaneously close due to a single failure etc., all of the valves concerned) closes in its shortest closing time while the reactor is in operation at the rated power.

2.3.5 Failure of the feed water control system (BWR)

This event is assumed to evaluate the effect of reactivity insertion by an abnormal feed water flow increase due to a failure, etc. of the feed water control system in BWR. For the analysis, the reactor coolant recirculation system is assumed to be in manual operation and it is required to select the reactor power level at the occurrence of the event in the same manner as in 2.2.8 so as to result in the most severe consequences in light of the criteria.

2.3.6 Failure of the reactor pressure control system (BWR)

Control of the reactor pressure is important in BWR since a change of void fraction in a reactor core affects the reactivity significantly. This event is, in consideration of such BWR characteristics, assumed to evaluate the effect of rapid change of the reactor pressure due to a failure etc. of the pressure control system. In line with the intent of this assumption, it is assumed that the main steam flow changes rapidly due to an spurious signal generated by a failure of the pressure control system. The reactor power level and operating conditions of the recirculation flow control system, etc. shall be selected so that the analysis results in the most severe consequences.

2.3.7 Complete loss of feedwater flow (BWR)

This event is assumed to evaluate the effect of a loss of feed water flow due to a failure etc. of the feed water control system in BWR. In line with the intent of this assumption, it is assumed that the feed water flow is completely lost within a time considering the pump inertia due to a feed water pump trip while the reactor is in operation at the rated power.

3. Analysis of accidents

3.1 Loss of reactor coolant or significant change in core cooling state

3.1.1 Loss of the reactor coolant (PWR, BWR)

This event is assumed to confirm that the design ensures appropriate cooling of the reactor core when the reactor coolant discharges outside the reactor coolant system due to a pipe failure etc. of the system. Furthermore, radioactive materials could be released to the environment or the internal pressure of the reactor containment could increase with a loss of the reactor coolant, but these are to be evaluated in 3.3.5 and 3.4.1, respectively. The analysis condition and criteria of this event shall be as provided in the "Evaluation Guide for Emergency Core Cooling System Performance of Light Water Nuclear Power Reactors".

3.1.2 Loss of reactor coolant flow (PWR, BWR)

This event is assumed to evaluate the effect of a significant decrease of the reactor coolant flow. In line with the intent of this assumption, it is assumed that all pumps that circulate the reactor coolant loose the driving power supply. Since this event is selected as a typical event that causes a significant decrease of the reactor coolant flow, when the operation of the safety protection system is expected, it should be assumed that the system is actuated in principle by a decrease of the reactor coolant flow or a decrease in the number of revolutions of the pumps. However, when the design is appropriately performed so that the event to cause a significant reduction of the reactor coolant flow is very unlikely to be caused by other than a loss of pump power supply, it is acceptable that a loss of the pump driving power supply is used as an actuation signal of the safety protection system.

When a loss of power supply for all reactor coolant pumps is considered in 2.2.1, it is not required to evaluate this event.

3.1.3 Locked rotor of a reactor coolant pump (PWR, BWR)

This event is assumed to evaluate the effect of rapidly decreasing the reactor coolant flow. In line with the intent of this assumption, it is assumed that a rotor of the reactor coolant pump seizes and the pump instantaneously stops while the reactor is in operation at the rated power.

3.1.4 Main feed water pipe break (PWR)

This event is assumed to evaluate the effect of a loss of the secondary coolant due to a pipe break of the feed water system in PWR. In line with the intent of this assumption, it is assumed that an instantaneous double-ended rupture occurs to one main feedwater pipe when the reactor is operating at the rated power level.

3.1.5 Main steam line break (PWR)

Radioactive materials do not exist in principle in the secondary cooling system in PWR and therefore, radioactive materials would not be immediately released to the environment even with a main steam line break. Moreover, the reactor core can be cooled without any problem using intact steam generators and main steam lines. However, this event is assumed to evaluate the effect of reactivity insertion due to a primary-coolant temperature decrease as a result of steam discharge due to a secondary cooling system break etc., which is the same purpose as in 2.2.6.

When it is demonstrated that the design has been made in such a manner that the reactor does not go critical even with the reactivity insertion due to a main steam line break, the analysis of this event can be omitted. When the reactor goes critical with assumption of this event, the criteria provided in (1) and (2) of 4.2 of the Guide body II shall be applied, after the reactor is confirmed to go back to subcritical. Furthermore, as a guide to confirm that this event does not progress when the critical state continues, (1), (2) and (4) of 4.1 of the Guide body II shall be satisfied in this case. Moreover, it is required to be able to easily estimate that other abnormal conditions do not occur until the reactor returns to the subcritical state.

3.2 Abnormal reactivity insertion or rapid change of reactor power

3.2.1 Control rod ejection (PWR)

This event is assumed to evaluate the effect of rapid reactivity insertion due to a failure etc. of the control rod drive system or control rod housing. In line with the intent of this assumption, it is assumed that one control rod cluster with the maximum reactivity worth rapidly ejects from the fully inserted position or control rod insertion limit position due to a break of control rod housing. The reactor is assumed to be critical or near critical and the initial conditions shall be selected so that the most severe consequences result. When the reactivity inserted by this assumption becomes one dollar or more, specific analysis conditions and criteria shall be as provided in the "Evaluation Guide for Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities." Besides, a pressure of the primary coolant decreases as a result of a failure of the control rod housing, resulting in flashing and thereby the negative reactivity effect, but this effect shall not be taken into account for the analysis purpose.

3.2.2 Control rod drop (BWR)

This event has the same purpose as in 3.2.1, but since the current BWRs are designed to drive control rods from below the reactor core, the process of the occurrence of the event is different from that of 3.2.1. The control rod assumed to drop is one control rod with the maximum reactivity worth, but in this case, when it can be justified that the rod worth minimizer has reliably continued to operate before the occurrence of the event, its function can be expected. Specific analysis conditions and the criteria shall be as provided in the "Evaluation Guide for Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities."

3.3 Abnormal release of radioactive materials to the environment

3.3.1 Failure of the gaseous radioactive waste processing facility (PWR, BWR)

There are various kinds of radioactive materials in a nuclear reactor facility, but gaseous radioactive waste has a comparatively high possibility of release to the environment among them. Namely, depending on damage etc. to a radioactive gaseous waste processing facility, the gaseous radioactive materials in storage or in process there could be released to the environment. This event is assumed from such a point of view to evaluate the effect of damage to part of the gaseous radioactive waste processing facility. In the analysis, taking account of the design storage capacity, temperature, pressure, isolation time etc. of each part of the gaseous radioactive waste processing facility, these conditions shall be selected so as to result in the most severe consequences.

3.3.2 Main steam line break (BWR)

In BWR, part of the reactor coolant changes to steam, and then the steam flows to the turbine outside the reactor containment through main steam lines. When the steam is released due to a break etc. of the main steam line outside the reactor containment, the radioactive materials in the steam will be directly released with the steam outside the reactor containment. This event is assumed to evaluate the effect of such a case.

It is assumed in this event that an instantaneous double-ended break occurs to one main steam line outside the reactor containment. The steam discharges at first from the break opening of the main steam line, and the two-phase flow is released later. In the calculation of the discharge flow rate of the reactor coolant, functions of the flow restrictor and main steam isolation valve, etc. can be expected with an appropriate safety margin taken into account. At an early stage of the event, the critical flow is generated at the flow restrictor, and as the main steam isolation valves close, the critical flow could occur at the valves. In this case, the flow limiting effect of the main steam isolation valves shall be ignored until the critical flow occurs at the valves.

The quantity of radioactive materials released to the environment shall be calculated as follows.

Namely, it is assumed that radioactive materials equivalent to the maximum concentrations allowed by the design are contained in the reactor coolant prior to the occurrence of an event. These radioactive materials will be released to the environment with discharge of the reactor coolant due to this event. Moreover, as the reactor coolant pressure decreases, additional release of radioactive materials will occur from fuel rods to the reactor coolant. The additional release due to this event from the fuel rods assumed to have been leaking prior to the occurrence of the event shall be taken into account, and it is accepted as adequate to determine the quantity of the release based on operating experiences etc. Besides, the noble gas is assumed to transfer instantaneously to the gaseous phase of the reactor coolant including bubbles and not to dissolve into the liquid phase.

Main steam isolation valves close in order to stop a release of steam, but it is assumed that one of the isolation valves does not close and the closed isolation valves have a leakage of which leakage rate is determined by the design leakage rate and the temperature and pressure of the reactor coolant, and the radioactive materials released with the leakage is taken into account.

In the light that this event has characteristics of the direct release of radioactive materials from the reactor cooling system to the outside of the reactor containment, it shall be demonstrated that this event does not cause any additional fuel rod failure.

3.3.3 Steam generator tube break (PWR)

In PWR, when heat transfer tubes of a steam generator fail, the primary coolant flows into the secondary cooling system and is released to the outside of the reactor containment through the main steam relief valves of the secondary cooling system etc., and the radioactive materials in the primary coolant could be released due to this event to the environment. This event is assumed to evaluate the effect of such a case.

The concentrations of radioactive materials that exist in the primary coolant prior to the occurrence of the event shall be the values calculated using the cladding tube defect rate assumed in the design. It is accepted as adequate to assume that noble gas and iodine are additionally released due to this event, in proportion to the depressurization rate of the primary coolant from the gap of fuel rods with defects assumed in the design.

When it requires operators' action to isolate the steam generator with damaged tubes, a sufficient margin for the time required for the action shall be taken into account.

In the light that this event has characteristics of the direct release of radioactive materials from the reactor cooling system to the outside of the reactor containment, it shall be demonstrated that this event does not cause any additional fuel rod failure.

3.3.4 Drop of a fuel assembly (PWR, BWR)

One of the locations other than a reactor core in a nuclear reactor facility where a not-negligible amount of radioactive materials exists is a spent fuel handling facility. This event is assumed from this point of view to evaluate the effect of a release of radioactive materials from fuel rods especially during fuel handling. It is assumed in the analysis that one fuel assembly during its handling drops from the highest position of its handling due to a malfunction or failure, etc. of a handling device, and radioactive materials are released from the gap of fuel rods damaged by the dropping impact.

Two kinds of methods provided in 3.3.5 for evaluation of the amount of noble gas and iodine released from fuel rods are accepted as adequate.

It is required to assume that a fuel assembly could drop inside of a spent fuel pit for PWRs and a fuel assembly could drop from above a reactor core for BWRs, and these conditions are selected so as to result in the maximum number of fuel rods expected to fail in the dropping of a fuel assembly while taking the current typical designs into account.

3.3.5 Loss of the reactor coolant (PWR, BWR)

The loss of the reactor coolant due to a failure etc. of the reactor coolant system is a typical accident of a light water reactor from a standpoint of significant change of cooling state of a reactor core, but it is a very important event also from a view point of a release of radioactive materials to the environment.

Among radioactive materials released inside of the reactor containment, those that have existed in the reactor coolant prior to the occurrence of the event and is assumed in the same manner as in 3.3.2 or 3.3.3. Even when it is calculated that a new fuel rod failure does not occur due to the accident, an assumption shall be made in the same manner as in 3.3.2 or 3.3.3. When it is calculated that a new fuel rod failure occurs due to the accident, the amount of radioactive materials to be released shall be appropriately assumed according to the failure conditions, on the premise that the core cooling conditions satisfy the criteria of 3.1.1.

Moreover, the evaluation of the amount of noble gas and iodine released from fuel rods by the following two kinds of methods is accepted as adequate. The first method is to evaluate the release of radioactive materials from the gap of the failed fuel rods with account taken of their power density and burnup conditions etc. The second method is to define the amount of radioactive materials released from the gap of the failed fuel rods as the amount not less than the proportionally calculated values from 1% and 0.5% of the core inventory of noble gas and iodine, respectively, using the power density ratio of the failed fuel rods and the fraction of the number of the failed ed fuel rods to the total number of all fuel rods in the reactor cores, etc.

For iodine among the radioactive materials, the generation ratio of organic iodine, deposition, and removal effect by the reactor containment spray water, etc. shall be as provided in Appendix I.

The gas leaked from the reactor containment containing radioactive materials is released from the stack after being processed by the emergency ventilation system etc. of the annulus or reactor building. In this case, the actuation signals for the emergency ventilation system etc. should be clarified, a sufficient time margin should be taken into account for the time to achieve the negative pressure etc., and it should be demonstrated that the diffusion of radioactive materials until the system functions are achieved is conservatively evaluated. Besides, after the emergency ventilation system etc. started to function, the design value may be used for filter efficiency. In PWRs designed to cover the whole outside of the reactor containment with the annulus, it may be assumed that all leakage from the reactor containment occurs in the annulus.

When ECCS is operated in the recirculation mode and the water in the reactor containment is led outside the reactor containment, it is assumed that the recirculated water leaks from the components etc. installed in the flow path at a leakage rate assumed in the design, and a contribution of the release of radioactive materials due to this to the environment shall be evaluated. However, when it is apparent that the leakage of the recirculated water does not meaningfully contribute to the effective dose due to this event, the evaluation can be omitted.

When it is apparent that direct and skyshine doses do not significantly contribute to the effective dose due to this event, the evaluation can be omitted.

3.3.6 Control rod ejection (PWR)

The control rod ejection is evaluated in 3.2.1 as a rapidly reactivity inserted event. In that case, as a conservative assumption, the effects of boiling and depressurization, etc. of the primary coolant due to a failure of control rod housing are to be ignored. However, in this case, since a failure of fuel rods and discharge of the primary coolant could occur due to this accident, the effects of a release of radioactive materials due to them on the environment are to be evaluated. The possibility of fuel rod failure is based on the analysis results of 3.2.1. Moreover, the effective dose due to the discharge of the primary coolant is evaluated in the same manner as in 3.3.5.

3.3.7 Control rod drop (BWR)

Since a failure of fuel rods could occur at the time of control rod drop assumed in 3.2.2, this event is assumed to evaluate the effect of radioactive materials released from the failed fuel rods to the environment. Section 3.2.2 requires that the reactor power level, etc. at the time of the event is selected so as to result in the highest fuel enthalpy. But for this event, since the amount of radioactive materials contained in the failed fuel rods is a matter of concern, the initial state of the reactor at the time of the event shall be selected so as to result in the maximum number of failed fuel rods. It is for this reason why the operating history is also designated for each reactor state at the time of the event.

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

In the "Safety Design Review Guide", the reactor containment is required for the postulated events used for the reactor containment design to withstand the load due to the events, to have a function to decrease the concentrations of radioactive materials released to the environment, and to control the concentrations of hydrogen or oxygen in the reactor containment. Following these requirements, section 3.4 specifies the specific contents of the postulated events for the reactor containment design and requirements for the analyses.

Since the reactor containment constitutes the ultimate barrier to prevent a release of radioactive materials to the environment, the requirements for the analyses of the postulated events used for the reactor containment design are based, in view of its importance, on different considerations from those for other structures, systems and components. As the postulated events used for the reactor containment design, 3.4 provides 3.4.1 for an increase of the pressure and temperature, 3.4.2 for a generation of flammable gas, and 3.4.3 for an occurrence of dynamic load. In addition, since the concentrations of radioactive materials released to the environment are evaluated in 3.3.5, and also in the siting evaluation provided in II of Appendix I, the evaluation is omitted in 3.4.

3.4.1 Loss of the reactor coolant (PWR, BWR)

As a cause of abnormal load etc. on the reactor containment, a loss of the reactor coolant due to a failure etc. of the reactor coolant system is the typical one. The analysis conditions shall be selected so as to result in the highest pressure and temperature in the reactor containment. Namely, the pipe assumed to break and its break location and various conditions for the analyses is selected so as to result in the maximum energy discharge to the reactor containment. Specifically, the pipe break location shall be the suction side of a reactor coolant pump for PWRs and for BWRs with external recirculation loops.

The cooling system of the reactor containment at the time of the accident, such as the reactor containment spray system, can be expected to perform its function, but in this case, a case without offsite power supply as well as an appropriate single failure is assumed.

Besides, during the process of the event, the cooling water that has cooled the reactor core once discharge in the reactor containment, and not only the energy of the reactor coolant discharging out due to the pipe break, but also the energy generated in the reactor core during the event shall be appropriately taken into account for their contributions to the internal pressure and temperature of the reactor containment.

3.4.2 Generation of flammable gas (PWR, BWR)

In the TMI accident in 1979, the hydrogen generated by a metal-water reaction was released in the reactor containment and mixed with air, resulting in ignition and rapid combustion. It had been well known before the TMI accident that hydrogen is generated by a metal-water reaction and oxygen and hydrogen are generated by radiolysis of water. Therefore, the "Safety Design Review Guide" provides the design requirements for a combustible gas control system. Integrating the analysis methods etc. for these requirements up to now, the design adequacy evaluation methods for combustible gas concentration control are summarized in 3.4.2. In this section, particularly for generation of oxygen and hydrogen by radiolysis of water, the radiolysis by radioactive materials exceeding the case of 3.1.1 is to be taken into consideration. The evaluation with such an assumption is for consideration of the importance of the reactor containment as mentioned above.

3.4.3 Occurrence of dynamic load (BWR)

Dynamic load is likely to occur in the pressure suppression pool during a loss of the reactor coolant and safety-relief valve actuation, etc. for the pressure-suppression type containment employed by current BWRs. The guide is separately provided on the evaluation methods of such a dynamic load, and the evaluation shall be made following the guide. In judgment of the evaluation results, it shall be confirmed that the stress by a dynamic load does not exceed the allowable stress that is based on appropriate codes and standards. However, at the stage of an application for approval of establishment, the design could not be completed for the details of the structures of the reactor containment. In such a case, if the basic design policy to carry out the detailed design in accordance with the above-mentioned guide, standards etc. is confirmed, the design policy may be accepted as adequate.

(For information)

The statement authorized by the Nuclear Safety Commission of Japan as of August 30, 1990

Regarding the Evaluation Guide for Emergency Core Cooling System Performance of Light Water Nuclear Power Reactors

The Committee, as a result of studying contents of the report regarding the subject guide submitted from the Special Committee on Safety Standards of Reactors, establishes "Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities" as in the separate attachment as of July 24, 1990.

Previously, the Committee, when reviewing the safety of a light water nuclear power reactor facility, had used the "Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities" established (partly modified on March 27, 1989 by the Nuclear Safety Commission) on September 29, 1978 by the Atomic Energy Commission, but decided to use, instead of this, the "Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities" in a separate attachment from now on.

In addition, this guide is to be reviewed, as appropriate, according to future knowledge and experience.