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

Appendix I
(August 30, 1990)

Application methods of detailed conditions and criteria that should be referred for the specific events to be evaluated in accordance with II.3 and III.3 of the Guide and in the analyses and evaluation of these events are shown in the following. And also, this appendix should be supplemented, as necessary, based on design improvements, buildup of experiences etc.

I. Safety Design Evaluation

1. Specific events to be evaluated

Specific events for the "anticipated operational occurrences" and "accidents" to be evaluated shall be as provided in the following. But, depending on the design, they are not limited to the following, and adding appropriate events in light of the objectives of the evaluation, they should be evaluated, as necessary. Moreover, depending on the design, when it has been demonstrated that a certain event is sufficiently included in the evaluation of another event, the analysis of the event is may be omitted.

1.1 Anticipated operational occurrences

1.1.1 Abnormal change in reactivity or power distribution in the core

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

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

(3) Drop and inconsistency of control rods (PWR)

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

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

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

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

(3) Loss of offsite power (PWR, BWR)

(4) Loss of main feed water flow (PWR)

(5) Abnormal increase in steam load (PWR)

(6) Abnormal depressurization of a secondary cooling system (PWR)

(7) Excessive water supply to a steam generator (PWR)

(8) Loss of feed water heater (BWR)

(9) Malfunction of a reactor coolant flow control system (BWR)

1.1.3 Abnormal change in reactor coolant pressure or reactor coolant inventory

(1) Loss of load (PWR, BWR)

(2) Abnormal depressurization of the reactor coolant system (PWR)

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

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

(5) Failure of the feed water control system (BWR)

(6) Failure of the reactor pressure control system (BWR)

(7) Complete loss of feed water flow (BWR)

1.2 Accidents

1.2.1 Loss of reactor coolant or considerable change in core cooling

(1) Loss of the reactor coolant (PWR, BWR)

(2) Loss of reactor coolant flow (PWR, BWR)

(3) Stuck of a reactor coolant pump rotor (PWR, BWR)

(4) Main feed water pipe break (PWR)

(5) Main steam line break (PWR)

1.2.2 Abnormal reactivity insertion or rapid change in reactor power

(1) Control rod ejection (PWR)

(2) Control rod drop (BWR)

1.2.3 Abnormal release of radioactive materials to the environment

(1) Failure of a radioactive gaseous waste processing facility (PWR, BWR)

(2) Main steam line break (BWR)

(3) Steam generator tube break (PWR)

(4) Drop of a fuel assembly (PWR, BWR)

(5) Loss of the reactor coolant (PWR, BWR)

(6) Control rod ejection (PWR)

(7) Control rod drop (BWR)

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

(1) Loss of the reactor coolant (PWR, BWR)

(2) Generation of flammable gas (PWR, BWR)

(3) Occurrence of dynamic load (BWR)

2. Analysis of anticipated operational occurrences

Specific conditions and application methods of criteria, which should be referred to when analyzing each event of "anticipated operational occurrences" listed in the above 1.1., are shown in the following.

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)

(1) An event of continuous control rod withdrawal that results in reactor power rise due to a failure of the control rod drive system, inadvertent operation or other causes during reactor startup is assumed.

(2) The reactor is assumed to be in cold-shutdown or hot standby state, and be critical or very near to criticality.

(3) For PWR, two control-rod-cluster banks is assumed to be withdrawn with the combination and at the maximum velocity which are allowed by the design, so as to result in the severe consequences in light of the criteria .

(4) For BWR, one control rod or one control-rod group with the maximum reactivity worth allowed by the control rod worth minimizer is assumed to be withdrawn with the maximum velocity allowed from the design.

(5) When it can be demonstrated that the interlocks etc. for the control rod withdrawal are in operation before the occurrence of the event and during it, and their reliability is adequately high, its operation may be taken into consideration.

(6) Other conditions for the analyses shall satisfy the requirements for the "Evaluation Guide for Reactivity Insertion Events."

(7) The criteria specified in (3) and (4) of the Guide II.4.1 (hereinafter referred to as "4.1") and the "Evaluation Guide for Reactivity Insertion Events" are applied as

criteria.

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

(1) An event of continuous control rod withdrawal that results in reactor power rise due to a failure of the control rod drive system, inadvertent operation or other causes during reactor power operation is assumed.

(2) The reactor is assumed to be in operation at the steady-state condition. In addition, the initial reactor power level is chosen so as to result in the severe consequences in light of the criteria .

(3) For PWR, two control-rod-cluster banks are assumed to be withdrawn in a design allowable combination at a design allowable withdrawal velocity so as to result in the severe consequences in light of the criteria .

(4) For BWR, one control rod or one control-rod group near the fuel assembly at the thermal limit condition within the reactor core are assumed to be withdrawn at a design allowable velocity so as to result in the severe consequences in light of the criteria .

(5) When it can be demonstrated that the interlocks etc. for control rod withdrawal are in operation before the occurrence of the event and during it, and their reliability is adequately high, its operation may be taken into consideration.

(6) (1), (2) and (4) of 4.1 are applied as criteria.

2.1.3 Drop and inconsistency of control rods (PWR)

(1) An event of power distribution change in a reactor core caused by an anomaly in alignment of the control rods inserted in the reactor core due to a failure etc. of the control rod drive system during reactor power operation is assumed.

(2) It is assumed that the one most reactive control rod cluster is assumed to drop from the fully withdrawn position to the fully inserted position while the reactor is in operation at a power a little lower than the rated power.

(3) As another event, the inconsistency is assumed that the control-rod-cluster bank inserted in the reactor core is at the insertion limit position, and that one control rod cluster of the bank is at the fully withdrawn position during reactor operation at a power little lower than the rated power.

(4) When the automatic reactor power control system is provided, both cases of automatic control and manual control shall be taken into consideration.

(5) (1), (2) and (4) of 4.1 are applied as criteria.

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

(1) An event that boron concentration decreases in the primary coolant by injection of pure water into the primary coolant due to a failure, inadvertent operation etc. of the chemical and volume control system, resulting in reactivity insertion during reactor startup or reactor power operation is assumed.

(2) It is assumed that the reactor is in a cold-shutdown state for startup and the boron concentration in the primary coolant is at the maximum plausible concentration during normal operation. It is assumed that the reactor has been operating at a power level with a margin taken into account to the rated power level during power operation and the boron concentration in the primary coolant is the one corresponding to that power level.

(3) The pure water is assumed to be injected into the primary coolant with the maximal flow rate allowed by the design.

(4) When the automatic reactor power control system is provided, both cases of automatic control and manual control shall be taken into consideration. In the case of automatic control, when it is sufficiently expected that the anomaly can be removed by operating personnel due to reliable information is available as there is sufficient-time to loose the reactivity shutdown margin after the control rod reaches the insertion limit, it may be taken into consideration of.

(5) As criteria, (1), (2) and (4) of 4.1 shall be applied.

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

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

(1) An event of decrease in reactor core coolant flow due to a failure of the pump driving reactor coolant (for PWR, the primary coolant, the same hereinafter) etc. during reactor power operation is assumed.

(2) The reactor is assumed to be in operation at a power a little lower than the rated power.

(3) It is assumed that the drive power for one of the reactor coolant pumps (however, all pumps concerned for those that are likely to fail simultaneously by a single failure etc. since they are, for an example, connected to the same bus or the same control device) is lost and thereby the reactor-coolant flow rate of the reactor core decreases.

(4) When the automatic reactor power control system is provided, both cases of automatic control and manual control shall be taken into consideration. The inertia effect of the stopping reactor coolant pump and its drive system may be appropriately taken into consideration.

(5) When the operation of the safety protection system is expected, the actuation signal shall be initiated by a reduction of the reactor coolant flow, a change of the

operating state of the pump, etc.

(6) (1), (2) and (4) of 4.1 are applied as criteria.

2.2.2 Inadvertent start up of the inactive loop of the reactor coolant system (PWR, BWR)

(1) An event is assumed that some of the reactor coolant pump are inactive and the reactor is operated at partial load, and the inactive pumps are forced to start operation due to a failure of pump control system or inadvertent operation, etc. and then the comparatively low-temperature coolant in the loop connected to the pump is injected into the reactor core, resulting in the reactivity insertion and thereby bringing a reactor power rise.

(2) The reactor is assumed to be in operation at the steady-state condition. The initial reactor power level and the initial number of reactor coolant pumps in operation are chosen so as to result in the most severe consequences in light of the criteria.

(3) It is assumed that one inactive reactor coolant pump (however, all pumps that are likely to fail simultaneously by a single failure etc. since, for an example, they are connected to the same bus or the same control device) starts accidentally.

(4) Even if a reactor power automatic control system is provided, it is assumed that the system is not taken credit for operation.

(5) (1), (2) and (4) of 4.1 are applied as criteria.

2.2.3 Loss of offsite power (PWR, BWR)

(1) An event of external power loss by a failure of the transmission system, station main power generation equipment or others during reactor power operation is assumed.

(2) It is assumed that the reactor is operated at a power a little lower than the rated power.

(3) The internal power supply system is assumed to become a no-voltage state.

(4) For startup of an emergency power source, a sufficient-time margin shall be taken into account.

(5) (1), (2) and (4) of 4.1 are applied as criteria.

2.2.4 Loss of main feed water flow (PWR)

(1) An event of a decrease in heat removal capability from a reactor caused by loss of the feed water to all steam generators due to failure of the main feedwater pumps, condensate-pumps, feed water control system, etc. during reactor power operation is

assumed.

(2) It is assumed that the reactor is operated at a power a little lower than the rated power for a sufficiently extended period of time.

(3) All of the main feedwater pumps of the secondary cooling system is assumed to stop simultaneously.

(4) (4) of 4.1 is applied as criterion.

2.2.5 Abnormal increase of steam load (PWR)

(1) An event is assumed that the main-steam flow increases abnormally due to an inadvertent opening of the turbine bypass valve, control valve or main steam relief valve of the secondary cooling system during reactor power operation, resulting in temperature drop of the primary coolant leading to reactivity insertion, and thereby the reactor power rises.

(2) It is assumed that the reactor is operated at a power a little lower than the rated power.

(3) It is assumed that one valve with the maximum steam flow among the turbine bypass valve, control valve or main steam relief valve opens fully.

(4) When the automatic reactor power control system is provided, both automatic control and manual control cases are taken into consideration.

(5) (1), (2) and (4) of 4.1 are applied as criteria.

2.2.6 Abnormal depressurization of the secondary cooling system (PWR)

(1) An event of reactivity insertion by decrease of the primary coolant temperature due to an inadvertent opening of the valve of secondary cooling systems, such as the turbine bypass valve and main steam relief valve during reactor hot shutdown is assumed.

(2) It is assumed that the reactor is in hot shutdown state and control rods are fully inserted. The boron concentration of the primary coolant should be a minimum allowable design concentration.

(3) It is assumed that one valve with the maximum depressurization effect among the secondary cooling system valves opens fully.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

2.2.7 Excessive water supply to a steam generator (PWR)

(1) An event is assumed that the feed water to steam generators become excessive due to a failure of the feed water control system and inadvertent operation, etc.

during reactor power operation, resulting in temperature drop of the primary coolant leading to reactivity insertion, and thereby the reactor power rises.

(2) It is assumed that the reactor is operated at a power a little lower than the rated power.

(3) It is assumed that one feed water control valve (however, all valves concerned for those that are likely to actuate simultaneously by a single failure etc. since they are, for an example, connected to the same control system,) of the secondary cooling system opens fully and the feedwater is supplied to one steam generator at a flow rate equivalent to fully opened control valve.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

2.2.8 Loss of feed water heater (BWR)

(1) An event is assumed that feed water temperature decreases due to a loss of steam flow to a feedwater heater during reactor power operation, resulting in reactor-core inlet subcooling increase and thereby the reactor power rises.

(2) The reactor is assumed to be in operation at the steady-state condition. In addition, the initial reactor power is assumed so as to result in the most severe consequences in light of the criteria.

(3) It is assumed that one stage of feedwater heaters (however, all feed water heaters concerned provided that that are likely to lose their functions simultaneously by a single failure since, for an example, they are connected to the same control system,) loses its heating function and the feed water temperature decreases by the maximum temperature change in that case.

(4) The reactor coolant recirculation system is assumed to be in manual operation mode.

(5) (1), (2) and (4) of 4.1 are applied as criteria.

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

(1) An event is assumed that recirculation flow increases due to a failure etc. of the control system for the reactor coolant recirculation flow during reactor power operation and thereby the reactor power rises.

(2) The reactor is assumed to be in steady state operation for a sufficiently extended period of time. In addition, the initial recirculation flow is at the lower limit of the flow control range and the initial reactor power is within the power range corresponding to the flow and selected so as to result in the most severe consequences in light of criteria.

(3) The recirculation flow shall be assumed to change to the maximum allowable design flow rate of the recirculation flow control system.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

2.3 Abnormal change in reactor coolant pressure or reactor coolant inventory

2.3.1 Loss of load (PWR, BWR)

(1) An event is assumed that the steam flow to a turbine decreases rapidly due to a failure etc. of the offsite power or turbine during reactor power operation and thereby the reactor pressure rises

(2) It is assumed that the reactor is in operation for a sufficiently extended period of time at a power a little lower than the rated power.

(3) External load is assumed to be instantaneously and completely lost.

(4) For BWR, the case in which turbine bypass valves do not function is taken into account.

(5) (1), (2) and (4) of 4.1 are applied as criteria.

2.3.2 Abnormal depressurization of the reactor coolant system (PWR)

(1) An event is assumed that reactor pressure decreases due to a failure etc. of the pressure control system of the primary cooling system during reactor power operation.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power.

(3) One valve with the largest effect on reducing the reactor pressure (however, all valves concerned for those that are likely to fully open simultaneously by a single failure etc. since they are, for an example, connected to the same control system,) is assumed to open fully.

(4) (1) of 4.1 is applied as criterion.

2.3.3 Inadvertent startup of the emergency core cooling system during power operation (PWR)

(1) An event of inadvertent ECCS startup during reactor power operation is assumed.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power.

(3) It is assumed that the high pressure injection system of the ECCS starts and it injects the cooling water into the primary cooling system. In addition, the value determined by the pressure of the primary cooling system and the pump characteristics with the value of a margin to the cooling water flow rate taken into

account.

(4) (1) and (4) of 4.1 are applied as criteria.

2.3.4 Inadvertent closure of the main steam isolation valves (BWR)

(1) An event of reactor pressure rise due to closure of the main steam isolation valves due to a failure of the isolation-valve control system, inadvertent operation etc. during reactor power operation is assumed.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) A main steam isolation valve is assumed to close in a minimum allowable design time.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

2.3.5 Failure of the feed water control system (BWR)

(1) An event is assumed that the feed water flow increases due to a failure of the feed water control system, etc. during reactor power operation, resulting in the reactor-core inlet subcooling increase and thereby the reactor power rises.

(2) The reactor is assumed to be in operation with the reactor coolant recirculation system in manual control. In addition, the initial reactor power shall be selected so as to result in the most severe consequences in light of the criteria.

(3) The feed water flow is assumed to go up instantaneously to the maximum allowable design flow.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

2.3.6 Failure of the reactor pressure control system (BWR)

(1) An event of main-steam flow change due to a failure of the pressure control system, etc. during reactor power operation is assumed.

(2) It is assumed that the reactor is assumed to be in steady state operation for a sufficiently extended period of time. In addition, the initial reactor power and operation conditions of the recirculation flow control system etc. shall be selected so as to result in the most severe consequences in light of the criteria.

(3) One train of the pressure control system (however, all trains of the system concerned for the pressure control systems that are likely to fail simultaneously by a single failure etc., since plural pressure control systems have, for an example, a common portion) is assumed to send the maximum output signal.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

2.3.7 Complete loss of feed water flow (BWR)

(1) An event of feed water flow decrease by a failure of the feed water control system, etc. during reactor power operation is assumed.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) It is assumed that feed water pumps trip and the feed water flow is completely lost within time which considered the pump inertia.

(4) (1), (2) and (4) of 4.1 are applied as criteria.

3. Analyses of accidents

Specific conditions and application methods of the criteria, which should be referenced to when analyzing each event of "accidents" listed in the above 1.2, are shown in the following.

3.1 Loss of reactor coolant or considerable change in core cooling

3.1.1 Loss of the reactor coolant (PWR, BWR)

(1) An event is assumed that the reactor coolant discharges outside the system due to a break of the pipes constituting the reactor coolant pressure boundary or of the components associated with these pipes during reactor power operation, resulting in a decrease in the reactor core cooling capability.

(2) The analysis conditions etc. shall satisfy the requirements of the "ECCS Performance Evaluation Guide." In addition, the analyses shall include the period from the time of the occurrence of the event until the time that the shift to the recirculation mode without any difficulty can be reasonably estimated.

(3) The criteria specified in (1) of the Guide body II.4.2 (hereinafter referred to as "4.2") and the "ECCS Performance Evaluation Guide" are applied as criteria.

3.1.2 Loss of reactor coolant flow (PWR, BWR)

(1) An event of significant decrease of the reactor coolant flow rate from the rated power flow rate to the natural circulation flow rate during reactor power operation is assumed.

(2) It is assumed that the reactor is assumed to be in operation at a power a little lower than the rated power.

(3) The power supply for all reactor coolant pumps is assumed to be lost simultaneously.

(4) Operating conditions of the automatic control system etc. of the reactor-coolant flow rate is selected so as to result in the most severe consequences in light of the

criteria. The inertia effect of the stopping reactor coolant pumps and drive systems may be appropriately taken into consideration.

(5) When an operation of the safety protection system is to be expected, the actuation signal shall be initiated by a reduction of the reactor-coolant flow rate and a change of the operating state of the pumps, etc.

(6) (1) and (3) of 4.2 are applied as criteria.

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

(1) An event is assumed that a rotor of the reactor coolant driving pump is locked during reactor power operation, resulting in rapid reactor coolant flow decrease.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power.

(3) It is assumed that a rotor of one reactor coolant pump is locked and the pump stops instantaneously.

(4) (1) and (3) of 4.2 are applied as criteria.

3.1.4 Main feed water pipe break (PWR)

(1) An event is assumed that a loss of the secondary coolant occurs due to a break of a feed-water-system pipe during reactor power operation, resulting in reactor cooling capability decrease.

(2) It is assumed that the reactor is in operation a power a little lower than the rated power for a sufficiently extended period of time.

(3) It is assumed that one main feedwater pipe is double-end ruptured instantaneously, the water retained in the steam generator connected to that pipe is lost, and the main feedwater to all steam generators is lost simultaneously with the occurrence of the main feedwater pipe break.

(4) An external power is assumed to be not available.

(5) (1) and (3) of 4.2 are applied as criteria.

3.1.5 Main steam line break (PWR)

(1) An event is assumed that the primary coolant temperature decreases due to a break of the secondary cooling system, etc. during hot reactor shutdown condition, resulting in reactivity insertion.

(2) The reactor is assumed to be in hot shutdown condition. It is assumed that the control rods are fully inserted and the boron concentration of the primary coolant is in the minimum design allowable concentration.

(3) It is assumed that one main steam pipe from a steam generator to a turbine is double-end ruptured instantaneously.

(4) The temperature decrease rate of the secondary side of the steam generator in association with the main steam pipe break shall be strictly evaluated with an appropriate margin considering the structure, etc. of the steam generator. Cases for both with and without offsite power available are assumed to be considered.

(5) (1) and (3) of 4.2 are applied as criteria.

3.2 Abnormal reactivity insertion or rapid change in reactor power

3.2.1 Control rod ejection (PWR)

(1) An event is assumed that rapid reactivity insertion and power-distribution change due to a break etc. of the control rod drive system or its housing occurs while the reactor is critical or near criticality.

(2) The reactor is assumed to be critical or near criticality. In addition, the initial reactor conditions are assumed so as to result in the most severe consequences in light of the criteria.

(3) It is assumed that the control-rod housing instantaneously breaks and reactivity equivalent to rapid ejection of the one most reactive worth control rod cluster out of the core is inserted.

(4) The negative-reactivity effect due to flashing in association with a break of the control rod housing is not be taken into consideration.

(5) Analysis conditions etc. must satisfy the requirements of the "Evaluation Guide for Reactivity Insertion Events."

(6) (2) and (3) of 4.2 and the criteria specified in the "Evaluation Guide for Reactivity Insertion Events" are applied as criteria.

3.2.2 Control rod drop (BWR)

(1) An event of rapid reactivity insertion and power-distribution change due to a drop of a control rod separated from the control-rod drive shaft out of the reactor core, while a reactor is critical or near criticality is assumed.

(2) The reactor shall be assumed to be critical or near criticality. In addition, the initial reactor conditions are assumed so as to result in the most severe consequences in light of the criteria.

(3) It is assumed that reactivity equivalent to the drop of the one most reactive worth control rod out of the reactor core is inserted.

(4) Analysis conditions etc. must satisfy the requirements of the "Reactivity

Insertion Event Evaluation Guide."

(5) (1) and (3) of 4.2 and the criteria specified in the "Evaluation Guide for Reactivity Insertion Events" are applied as criteria.

3.3 Abnormal release of radioactive materials to the environment

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

(1) An event of the release of gaseous radioactive material stored at a radioactive gaseous waste processing facility to the environment due to damage of part of the facility is assumed.

(2) The maximum amount of gaseous radioactive material to be expected by the design of the nuclear reactor facility is assumed to be stored in the tank, holdup tank etc. of the radioactive gaseous waste processing facility during normal reactor operation (including startup, hot standby, power operation and shutdown).

(3) It is assumed that part of the radioactive gaseous waste processing facility is damaged and gaseous radioactive material stored is released. In addition, the damaged part is assumed so as to result in the most severe consequences in light of the criteria with the amount of the stored radioactive material and isolation time taken into account.

(4) The components etc. connected to the damaged part, which is likely to increase the release of the gaseous radioactive material, is assumed to be in operation so as to result in the most severe consequences within the allowable design range. If damaged portions can be isolated by valves, etc., the credit can be taken for their function with taking account of a sufficient margin to its operation time.

(5) When the damaged part is located indoors, the air ventilation system etc. of the auxiliary-building or turbine building shall be assumed to be in such an operating condition that the most severe consequences result.

(6) Diffusion of the radioactive material released to the environment shall be evaluated in accordance with the "Meteorological Guide for Safety Analysis of Nuclear Power Reactor Facilities" (hereinafter referred to as the "Meteorological Guide.")

(7) (5) of 4.2 is applied as criterion.

3.3.2 Main steam line break (BWR)

(1) An event of the release of radioactive material to the environment due to a main steam pipe break outside a reactor containment vessel and discharge of the reactor coolant from the break opening shall be assumed during reactor power operation.

(2) The reactor is assumed to be in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) An instantaneous double-ended break of one main steam pipe outside of the reactor containment is assumed.

(4) The main steam isolation valve is assumed to close fully with the longest design operation delay time and closing time.

(5) In calculation of the reactor coolant discharge rate, the function of the flow restrictor may be taken into consideration. However, for the main steam isolation valve, the flow limiting effect of the valve shall not be taken into consideration until the critical flow is generated at the part.

(6) It is assumed that an offsite power supply is lost simultaneously with the occurrence of the event.

(7) It is assumed that the concentration of the fission products in the reactor coolant before the occurrence of the event is the one equivalent to the maximum concentration of I-131 allowed for operation and the composition of which is the diffusion composition. The halogen concentration in the steam phase is assumed to be 2% of its concentration in the liquid phase.

(8) The amount of additional release from the fuel rod in association with the reactor pressure decrease shall be the value with an appropriate margin taken into account for the mean value of the actual measurements in the preceding reactors etc. for I-131, the calculated values assuming the equilibrium compositions for other fission products and the value two times the release of iodine for noble gas. The additional release rate of fission products is assumed to be proportional to the reactor pressure-decrease rate. The time required for the fission products released from the fuel rods to reach the main steam isolation valves before their closure during the process of the event may be taken into consideration for the evaluation.

(9) The organic iodine is assumed to be 4% of the released iodine from the fuel rod during the process of the event, and the remaining 96% is assumed as inorganic iodine. It is assumed that 10% of the organic iodine transfers instantaneously to the gas phase and the rest decompose. The fraction of decomposed organic iodine, inorganic iodine and halogens other than iodine carried over to the gas phase is assumed to be 2%. All of the noble gas is assumed to transfer instantaneously to the gas phase. It is assumed that 50% of decomposed organic iodine, inorganic iodine and halogens other than iodine released to the turbine building deposit on the floor, wall etc.

(10) It is assumed that the reactor coolant released before closure of the main steam isolation valves evaporates and becomes the steam cloud uniformly containing the fission products simultaneously released. The fission products released after closure of the isolation valves is assumed to diffuse from the ground to the atmosphere.

(11) One of the main steam isolation valves is assumed not to close. In addition, it is assumed that the closed isolation valves have the leakage determined by the design

leakage rate, temperature and pressure.

(12) After closure of the main steam isolation valves, the steam corresponding to decay heat is assumed to transfer to the pressure suppression pool passing through the residual heat removal system or safety relief valves etc.

(13) After closure of the main steam isolation valves, the reactor pressure is assumed to decrease to the atmospheric pressure linearly within the longer time of either the time to decrease to the atmospheric pressure by the reactor core isolation cooling system etc or 24 hours.

(14) Formation and movement of the steam cloud shall be evaluated using appropriate parameters, and the diffusion of the fission products released to the environment after closure of the isolation valves shall be evaluated according to the "Meteorological Guide."

(15) (5) of 4.2 is applied as criterion, after confirming that any additional fuel rod failure does not occur.

3.3.3 Steam generator tube break (PWR)

(1) An event is assumed that the primary coolant is released outside the reactor containment through the secondary cooling system due to damage to heat transfer tubes of the steam generator during reactor power operation.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power for a sufficiently extended period of time and the reactor pressure is the highest one in normal operation.

(3) It is assumed that an instantaneous double-ended break occurs to one heat transfer tube of the steam generator.

(4) Cases for both with and without offsite power available are assumed to be considered. Moreover, when the ECCS starts automatically, a larger discharge of the primary coolant is assumed to be caused by its operation.

(5) The concentration of the fission products in the primary coolant before the occurrence of the event shall be the value calculated using the clad defect rate assumed in the design.

(6) It is assumed that noble gas and iodine is additionally released in proportion to the pressure decrease rate of the reactor from the gap of fuel rods which have defects assumed in the design.

(7) It is assumed that all of the noble gas discharged into the secondary cooling system is released to the atmosphere. Moreover, it is assumed that iodine is released with the steam to the atmosphere with a gas-liquid partition coefficient of 100.

(8) When isolation of the damaged steam generator requires operation by operators,

sufficient time margin is required to be taken into account. After the isolation, it is assumed that the reactor pressure decreases linearly to the atmospheric pressure within a longer time either a time in which the reactor pressure decreases to the atmospheric pressure by the operable cooling system or 24 hours and the isolated valves have the leakage determined by the design leakage rate, temperature and pressure.

(9) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide".

(10) (5) of 4.2 is applied as criterion, after confirming that additional fuel rod failure does not occur.

3.3.4 Drop of a fuel assembly (PWR, BWR)

(1) An event is assumed that a fuel assembly drops for some reason and fails during reactor refueling, which in turn result in the release of radioactive material to the environment.

(2) For PWR, it is assumed that one fuel assembly being handled within the spent fuel pit drops from the operationally highest position.

(3) For BWR, it is assumed that one fuel assembly being handled above the reactor core drops from the operationally highest position into the core.

(4) It is assumed that the dropped fuel assembly has had the maximum power output when the reactor has been operating at a power level with the value of a margin to the rated power level taken into account for a sufficient long period of time, and the event occurs after appropriate cooling time and time period for required work after reactor shutdown. In addition, the radioactivity decay for this time period may be appropriately taken into consideration.

(5) The number of failed fuel rods due to the drop shall be the maximum number of the fuel rods in that assembly as long as there is no experimental basis.

(6) It is assumed that fission products are released from the gap of the failed fuel rods into the water. All of the noble gas is assumed to transfer to the gas phase. The underwater decontamination factor of iodine shall be 500.

(7) The air ventilation system and emergency ventilation system, etc. of the auxiliary building or reactor building may be expected to operate as designed.

(8) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide".

(9) (5) of 4.2 is applied as criterion.

3.3.5 Loss of the reactor coolant (PWR, BWR)

(1) An event is assumed that radioactive material is released to the environment during a loss of reactor coolant assumed in 3.1.1.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) The concentration of the fission products in the reactor coolant prior to the occurrence of the event is assumed in the same manner as the cases of 3.3.2 or 3.3.3.

(4) When it is calculated that fuel rod failures newly occur due to the event, an appropriate amount of fission products released depending on the conditions of the failed fuel rods is assumed. Moreover, when it is calculated that a new fuel rod failure does not occur, the amount of additional fission product release shall be evaluated in the same manner as the case of 3.3.2 or 3.3.3.

(5) It is assumed that noble gas and iodine are released into the reactor containment due to this event. The organic iodine is assumed to be 4% of the iodine released from the fuel rods into the reactor containment, and the remaining 96% is assumed as inorganic iodine. Fifty percent of the inorganic iodine is assumed to deposit on the inside of the reactor containment, and it does not contribute to the leakage. Furthermore, the effect of the inorganic iodine to be removed by the reactor containment spray water or to dissolve in the suppression pool water may be taken into consideration. In this case, the decontamination efficiency, gas-liquid partition coefficient, etc. shall be the values based on experiments or the values including sufficient safety margin. For organic iodine and noble gas, these effects shall be ignored.

(6) The leakage from the reactor containment shall be evaluated assuming the leakage rate corresponding to the pressure in the reactor containment based on the design leakage rate of the reactor containment and the analytical results of 3.4.1. For PWR, it is assumed that 97% of the leakage occurs at the annulus region and the remaining 3% at the regions other than the annulus region. Deposition of the leaked fission products on the inside of the annulus or reactor building shall not be taken into consideration.

(7) The function of the emergency ventilation system etc. (including the filters) of the annulus or reactor building may be expected with a sufficient time margin taken into account after clarifying their actuation signals.

(8) When the ECCS is operated in recirculation mode and the water in the reactor containment is led outside the reactor containment, a leakage of recirculating water with the design leakage rate is assumed to occur outside the reactor containment. It is assumed that the amount of iodine equivalent to (3) and (4) is dissolved in the recirculation water as inorganic iodine, the transfer rate of the leaked iodine to the gas phase is 5% and the deposition rate in the auxiliary building or reactor building is 50%.

(9) The direct dose rate and sky-shine dose rate due to the fission products in the reactor containment is evaluated taking into account the locations of the fission products in the reactor containment and the shielding of the reactor containment etc.

(10) The accident evaluation period shall be the time period that the internal pressure of the reactor containment decreases to the extent that the leakage from the reactor containment can be ignored.

(11) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide".

(12) (5) of 4.2 is applied as criterion.

3.3.6 Control rod ejection (PWR)

(1) An event is assumed that radioactive material is released to the environment at the occurrence of the control rod ejection assumed in 3.2.1.

(2) It is assumed that the reactor is in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) The effective dose rate evaluation shall be in accordance with the case of 3.3.5.

(4) (5) of 4.2 is applied as criteria.

3.3.7 Control rod drop (BWR)

(1) An event is assumed that radioactive material release to the environment at the control rod drop assumed in 3.2.2.

(2) When an event occurs during high temperature standby or during partial power operation, the reactor is assumed to be in operation at a power a little lower than the rated power for a sufficiently extended period of time up until thirty minutes before the occurrence of the event. Moreover, when an event occurs during cold shutdown, the reactor is assumed to be in operation at a power a little lower than the rated power for a sufficiently extended period of time up until 24 hours before the occurrence of the event.

(3) Fission products is assumed to release from the gap of the failed fuel rods into the water. The organic iodine is assumed to be 4% of the released iodine from the fuel rod, and the remaining 96% is assumed as inorganic iodine. It is assumed that 10% of the organic iodine transfers instantaneously to the gas phase and that the rest is decomposed. The rate of carryover to the gas phase of iodine from the decomposed organic iodine and the inorganic iodine shall be 2%. The noble gas is assumed to transfer instantaneously to the gas phase.

(4) The main steam isolation valve is assumed to close with a longest design operation delay time and closing time. It is assumed that 50% of inorganic iodine of the fission products transferred to the condenser deposits and that the remaining

fission products in the gas phase leakage into the turbine building at the 0.5% / day of leakage rate of the free space of condenser and turbine.

(5) The function of the air ventilation system in the turbine building etc. shall be taken into account, when it is working.

(6) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide".

(7) (5) of 4.2 is applied as criterion.

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

3.4.1 Loss of the reactor coolant (PWR, BWR)

(1) An event is assumed that the reactor coolant discharges outside the system due to damage to piping etc. constituting the reactor coolant pressure boundary, resulting in abnormal pressure and temperature rise in the reactor containment during reactor power operation.

(2) It is assumed to be in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) It is assumed that an instantaneous double-ended break occurs to one pipe constituting the reactor coolant pressure boundary. In addition, the pipe assumed to break and its break location shall be selected so as to result in the maximum reactor containment pressure.

(4) The offsite power is assumed not available simultaneously with the occurrence of the event.

(5) (4) of 4.2 is applied as criterion after it is confirmed that the temperature in the reactor containment does not exceed the maximum operating temperature.

3.4.2 Generation of flammable gas (PWR, BWR)

(1) An event is assumed that combustible gas is generated during a loss of the reactor coolant assumed in 3.4.1.

(2) The amount of hydrogen generated by metal-water reaction shall be the larger value of either five times the amount generated by metal-water reaction that is calculated in 3.1.1 or the amount generated when the metal of 0.0058mm thickness from the surface of the cladding tubes of all fuel rods reacts with water.

(3) Assuming that 50% of halogen and 1% of the fission products excluding noble gas and halogen out of the fission products inventory in the reactor core exist in the liquid phase of the water in the reactor containment, the radiolytic decomposition of the water in the reactor containment shall be appropriately evaluated. Furthermore, assuming that all other fission products excluding noble gas exist in the reactor core,

the radiolytic decomposition of the water in the reactor core shall be appropriately evaluated. The decomposition rate of the water per unit energy absorbed shall be the value confirmed by experiments with an appropriate margin taken into account.

(4) For a design that adds materials such as alkali in the reactor containment water, the hydrogen generated by chemical reaction with metal structures in the reactor containment shall be appropriately evaluated.

(5) For a design that provides a system to control the concentration of combustible gases, such as a hydrogen recombiner, the function may be expected within the design range of these systems.

(6) As criteria, the concentration of either oxygen or hydrogen in the reactor containment atmosphere shall be 5% or 4% or less, respectively, for at least 30 days after the occurrence of the event.

3.4.3 Generation of dynamic load (BWR)

(1) An event is assumed that local dynamic load is generated in the pressure-suppression type reactor containment during a loss of the reactor coolant, safety valve actuation etc.

(2) The dynamic load in the reactor containment is evaluated in accordance with the "Evaluation Guide for Dynamic Load Added to the Mark-I Containment Pressure Suppression System of Boiling Water Reactors" or the "Evaluation Guide for Dynamic Load Added to the Mark-II Containment Pressure Suppression System of Boiling Water Reactors."

(3) When it is demonstrated that the stresses etc. of each part of the reactor containment satisfy the provisions of codes and standards etc. to be based on or the design is intended to satisfy them as a result of the evaluation, the design or design method is accepted as adequate.

II. Siting Evaluation

1. Specific events of the major accidents and hypothetical accident

The specific events of the major accidents and hypothetical accident to be evaluated are as follows.

1.1 Loss of the Reactor Coolant (PWR, BWR)

1.2 Steam Generator Tube Break (PWR)

1.3 Main Steam Line Break (BWR)

2. Evaluation of the major accidents and hypothetical accident

Specific conditions and application methods of criteria, which should be referred to when analyzing each event of the major accidents and the hypothetical accident listed in the above 1, are shown in the following.

2.1 Loss of the reactor coolant

2.1.1 Loss of the reactor coolant (PWR)

For the major accident

(1) The event is assumed that radioactive material is released to the environment during a loss of the reactor coolant assumed in 3.1.1 in "I. Safety Design Evaluation" of Appendix I.

(2) The reactor is assumed to be in operation at a power a little lower than the rated power for a sufficiently extended period of time.

(3) The amount of fission product release into the reactor containment after the occurrence of the event shall be 2% of noble gas and 1% of iodine out of their inventories in the reactor core.

(4) The organic iodine is assumed to be 10% of the released iodine to the reactor containment, and the remaining 96% is assumed to be inorganic iodine.

(5) For the iodine released into the reactor containment, 50% of the inorganic iodine is assumed to deposit on the inside of the reactor containment and on components in the same containment, which does not contribute to the leakage from the containment. For organic iodine and noble gas, this effect shall be ignored.

(6) The removal efficiency of inorganic iodine by the reactor containment spray water shall be the value evaluated based on experiments by taking account of a margin. For an example, when the equivalent half-life evaluated by the design is 50 seconds or less, it is accepted as adequate to treat the equivalent half-life as 100 seconds. For organic iodine and noble gas, this effect shall be ignored.

(7) The leakage of noble gas and iodine from the reactor containment shall be taken into consideration. The leakage from the reactor containment is evaluated by assuming the leakage rate which corresponds to the reactor containment pressure with margin obtained from the design leakage rate and the analytical results of 3.4.1 in "I. Safety Design Evaluation." It is assumed that 97 % of the leakage from reactor containment occurs at the annulus and the remaining 3% at the parts other than the annulus.

(8) The credit can be taken for the function of annulus air recirculation system (including filters) by taking account of a sufficient time margin after clarifying the actuation signal. In addition, the iodine removal efficiency of the filter is the design value with a margin taken into account. For an example, when the design iodine removal efficiency is 95% or more, the iodine removal efficiency of 90% is considered to be adequate.

(9) When the ECCS is operated in the recirculation mode and the water in the reactor containment is led outside the reactor containment, a leakage of recirculating water with the design leakage rate with a margin taken into account is assumed to exist outside the reactor containment. It is assumed that 1% of iodine of the core inventory dissolves into the recirculation water immediately after the occurrence of the event, the transfer rate to the gas phase of iodine leakage from the ECCS recirculation system into the auxiliary building is 5% and the deposition rate of the iodine in the auxiliary building is 50%.

(10) When the filter for iodine is provided to the air ventilation system in the auxiliary building installed with the ECCS recirculation system, the removal efficiency shall be the design value with a margin taken into account. For an example, when the design iodine removal efficiency is 95% or more, it is accepted as adequate to treat the iodine removal efficiency as 90%.

(11) The amount of direct dose rate and sky-shine dose rate due to the fission products in the reactor containment shall be evaluated taking into account the shielding of the reactor containment etc. In addition, for evaluation of the direct dose rate and sky-shine dose rate, the fraction of the fission products released into the reactor containment to the fission product inventory in the core is assumed to be 2% of noble gas, 1% of halogen and 0.02% of others.

(12) The accident evaluation period shall be the time period that the internal pressure of the reactor containment decreases to such an extent that the leakage from the reactor containment can be ignored, but not less than 30 days.

(13) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide".

(14) The criteria shall be in accordance with the "Review Guide for Reactor Siting."

<u>For the hypothetical accident</u>

The hypothetical accident is evaluated in the same manner as the major accident excluding the following:

(3) The amount of fission product release into the reactor containment after the occurrence of the event shall be 100% of noble gas and 50% of iodine out of their inventories in the reactor core.

(9) When the ECCS is operated in the recirculation mode and the water in the reactor containment is led outside the reactor containment, a leakage of recirculating water with the design leakage rate with a margin taken into account is assumed to exist outside the reactor containment. It is assumed that 50% of iodine of the core inventory dissolves into the recirculation water immediately after the occurrence of the event, the transfer rate to the gas phase of iodine leakage from the ECCS recirculation system into the auxiliary building is 5% and the deposition rate of the iodine in the auxiliary building is 50%.

(11) The amount of direct dose rate and sky-shine dose rate due to the fission products in the reactor containment shall be evaluated taking into account the shielding of the reactor containment etc. In addition, for evaluation of the direct dose rate and sky-shine dose rate, the fraction of the fission products released into the reactor containment to the fission product inventory in the core is assumed to be 100% of noble gas, 50% of halogen and 1% of others.

2.1.2 Loss of the reactor coolant (BWR)

For the major accident

(1) The event is assumed that radioactive material is released to the environment during a loss of the reactor coolant assumed in 3.1.1 in "I. Safety Design Evaluation" of Appendix I.

(2) The reactor is assumed to be in operation a little lower than the rated power for a sufficiently extended period of time.

(3) The amount of fission product release into the reactor containment after the occurrence of the event shall be 2% of noble gas and 1% of iodine out of their inventories in the reactor core.

(4) The organic iodine is assumed to be 10% of the released iodine to the reactor containment, and the remaining 90% is assumed as inorganic iodine.

(5) For the iodine released into the reactor containment, 50% of the inorganic iodine is assumed to deposit on the inside of the reactor containment and on components in the same containment, which does not contribute to the leakage from the containment. For organic iodine and noble gas, this effect shall be ignored.

(6) The dissolution rate of inorganic iodine in the suppression water is assumed to be a partition coefficient 100. For organic iodine and noble gas, this effect is ignored.

(7) The leakage of noble gas and iodine from the reactor containment is taken into consideration. The leakage from the reactor containment is evaluated by assuming the leakage rate which corresponds to the reactor containment pressure with margin obtained from the design leakage rate of the reactor containment and the analytical results of 3.4.1 in "I. Safety Design Evaluation."

(8) Credit is taken for the function of the emergency ventilation system etc. (including filters) of the reactor building by taking account of a sufficient time margin after clarifying the actuation signal. The capacity of the emergency ventilation system shall be the values determined by the design. In addition, the iodine removal efficiency of the filter shall be the design value with a margin taken into account. For an example, when the design iodine removal efficiency is 99% or more, it is accepted as adequate to treat the iodine removal efficiency as 95%. The fission product removal effect due to deposition at the reactor building shall be ignored, and only the spontaneous disintegration shall be considered.

(9) When the ECCS is operated in recirculation mode and the water in the reactor containment is led outside the reactor containment, a leakage of recirculating water with the design leakage rate with a margin taken into account is assumed to exist outside the reactor containment. It is assumed that 1% of iodine of the core inventory dissolves into the recirculation water immediately after the occurrence of the event, the transfer rate to the gas phase of iodine leakage from the ECCS recirculation system into the reactor building is 5% and the deposition rate of the iodine in the reactor building is 50%.

(10) It is assumed that the fission products leaked from the reactor containment into the reactor building is released to the environment from the stack after being processed by the standby gas treatment system in the reactor building.

(11) The amount of direct dose rate and sky-shine dose rate due to the fission products in the reactor containment shall be evaluated taking into account the shielding of the reactor containment etc. In addition, for evaluation of the direct dose rate and sky-shine dose rate, the fraction of the fission products released into the reactor containment to the fission product inventory in the core is assumed to be 2% of noble gas, 1% of halogen and 0.02% of others.

(12) The accident evaluation period shall be the time period that the internal pressure of the reactor containment decreases to such an extent that the leakage from the reactor containment can be ignored, but not less than 30 days.

(13) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide".

(14) The criteria shall be in accordance with the "Review Guide for Reactor Siting."

<u>For the hypothetical accident</u>

The hypothetical accident shall be evaluated in the same manner as the major accident excluding the following:

(3) The amount of fission product release into the reactor containment after the occurrence of the event shall be 100% of noble gas and 50% of iodine out of their inventories in the reactor core.

(9) When the ECCS is operated in the recirculation mode and the water in the reactor containment is led outside the reactor containment, a leakage of recirculating water with the design leakage rate with margin is assumed to exist outside the reactor containment. It is assumed that 50% of iodine of the core inventory dissolves into the recirculation water immediately after the occurrence of the event, the transfer rate to the gas phase of iodine leakage from the ECCS recirculation system into the reactor building is 5% and the deposition rate of the iodine in the reactor building is 50%.

(11) The amount of direct dose rate and sky-shine dose rate due to the fission

products in the reactor containment shall be evaluated taking into account the shielding of the reactor containment etc. In addition, for evaluation of the direct dose rate and sky-shine dose rate, the fraction of the fission products released into the reactor containment to the fission product inventory in the core is assumed to be 100% of noble gas, 50% of halogen and 1% of others.

2.2 Steam generator tube break (PWR)

<u>For the major accident</u>

(1) The event is assumed that the primary coolant is released outside the reactor containment through the secondary cooling system due to damage to heat transfer tubes of the steam generator during reactor power operation.

(2) It is assumed that the reactor has been operating at a power level with the value of a margin to the rated power level taken into account for a sufficient long period of time and the reactor pressure is the highest one of normal operation.

(3) It is assumed that an instantaneous double-ended break occurs to one heat transfer tube of the steam generator.

(4) For offsite power supply, cases with and without offsite power shall be taken into consideration. When the ECCS starts automatically, its operation is assumed to result in a larger discharge rate of the primary coolant.

(5) The concentration of the fission products in the primary coolant before the occurrence of the event is the value calculated using the clad defect rate assumed in the design.

(6) It is assumed that noble gas and iodine is additionally released in proportion to the pressure decrease rate of the reactor from the gap of fuel rods which have defects assumed in the design.

(7) Of these fission products in the primary coolant, the amount of radioactivity that is discharged from the primary cooling system into the secondary cooling system until the steam generator is dependent on the concentration in the primary coolant.

(8) The organic iodine is assumed to be 1% of the iodine discharged into the secondary cooling system, and the remaining 99% is assumed to be inorganic iodine. It is assumed that all of the organic iodine is released to the atmosphere. It is assumed that the inorganic iodine is released with the steam to the atmosphere with a gas-liquid partition coefficient of 100. All of the noble gas released into the secondary cooling system is to be released to the atmosphere.

(9) Even though it is considered that there is no release of the fission products to the atmosphere after the failed steam generator is isolated the inorganic iodine is to be released to the atmosphere due to steam leakage from the secondary side valves for the evaluation purpose. The steam leakage rate from the valves shall be the

design value with a margin taken into account. The reactor pressure is assumed to decrease to the atmospheric pressure linearly after the isolation within the longer time of either one to decrease to the atmospheric pressure with the operable cooling system or 24 hours, and the steam is assumed to leak out from the valves at the design leakage rate corresponding to this pressure.

(10) The diffusion of the fission products released to the environment shall be evaluated according to the "Meteorological Guide."

(11) The criteria shall be in accordance with the "Reactor Siting Review Guide."

For the hypothetical accident

The hypothetical accident shall be evaluated in the same manner as the major accident excluding the following:

(6) It is assumed that noble gas and iodine is additionally released from the gap of fuel rods which have defects assumed in the design.

(7) Of these fission products in the primary coolant, the fraction of radioactivity that is discharged from the primary cooling system into the secondary cooling system until the steam generator is isolated is assumed to be the same fraction of the amount of the primary coolant that discharged to the total water inventory.

(9) After the failed steam generator is isolated, it is assumed that the inorganic iodine is released to the atmosphere due to steam leaking from the secondary side valves. The rate of steam leaking from the valve shall be the design value with a margin taken into account, and the leakage is assumed to continue for 30 days.

2.3 Main steam line break (BWR)

For the major accident

(1) The event of the release of radioactive material to the environment due to a main steam pipe break outside a reactor containment vessel and discharge of the reactor coolant from the break opening is assumed during reactor power operation.

(2) The reactor is assumed to have been operating at a power level with the value of a margin to the rated power level taken into account for a sufficient long period of time.

(3) An instantaneous double-ended break of one main steam pipe outside of the reactor containment is assumed.

(4) The main steam isolation valve is assumed to close fully with the longest design operation delay time and closing time.

(5) In calculation of the reactor coolant discharge rate, the function of the flow restrictor may be taken into consideration. However, for the main steam isolation

valve, the flow limiting effect of the valve is not be taken into consideration until the critical flow is generated at the main steam isolation valve.

(6) It is assumed that external power supply is lost simultaneously with the occurrence of the event.

(7) It is assumed that the concentration of the fission products in the reactor coolant before the occurrence of the event is the one equivalent to the maximum concentration of I-131 allowed for operation and its composition is the diffusion composition. The halogen concentration in the steam phase is assumed to be 2% of its concentration in the liquid phase.

(8) The amount of additional release from the fuel rod accompanying the reactor pressure decrease is the value based on the actual measurements in the preceding reactors etc. taking account of an appropriate margin for I-131, the calculated values assuming the equilibrium compositions for other fission products and the value two times the release of iodine for noble gas.

(9) It is assumed that a fraction of additional fission product release from fuel rods before closure of the main steam isolation valves is proportional to the reactor pressure decrease rate before closure of the main steam isolation valves and 1% of the additionally released fission product is released from the break opening.

(10) It is assumed that the fraction of the additional fission product release from fuel rods after closure of the main steam isolation valves is gradually released into the reactor coolant accompanying the reactor pressure decrease rate.

(11) The organic iodine is assumed to be 1% of the released iodine from fuel rods during the event process, and the remaining 99% is assumed to be inorganic iodine. It is assumed that 10% of the organic iodine transfers instantaneously to the gas phase. The fraction of the remaining iodine and other halogens carried over to the gas phase shall be 2%. All of the noble gas is assumed to transfer instantaneously to the gas phase.

(12) It is assumed that all of the reactor coolant released before closure of the main steam isolation valves completely vaporizes and becomes the steam cloud uniformly containing the radioactive material released simultaneously. The radioactive material released after closure of the isolation valves is assumed to diffuse from the ground to the atmosphere.

(13) One of the main steam isolation valves is assumed not to close. In addition, the steam is assumed to leak from the closed isolation valves. The leakage rate of the closed main steam isolation valves is assumed to be the design value with a margin taken into account, and the value is assumed to change depending on the temperature and pressure.

(14) After closure of the main steam isolation valves, the steam corresponding to decay heat is assumed to transfer to the pressure suppression pool through the

residual heat removal system or the safety-relief valves etc.

(15) After closure of the main steam isolation valves, the reactor pressure is assumed to decrease to the atmospheric pressure linearly within the longer time of either the one to decrease to the atmospheric pressure by the reactor core isolation cooling system etc. or 24 hours

(16) Formation and movement of the steam cloud is evaluated using appropriate parameters, and the diffusion of the fission products released to the environment after closure of the isolation valves is be evaluated according to the "Meteorological Guide."

(17) The criteria shall be in accordance with the "Review Guide for Reactor Siting."

For the hypothetical accident

The hypothetical accident shall be evaluated in the same manner as the major accident excluding the following:

(10) Concerning additional release of the fission products from fuel rods after closure of the main steam isolation valves, it is assumed that all of these fission products is released into the reactor coolant immediately after closure of the main steam isolation valves.

(13) One of the main steam isolation valves is assumed not to close. In addition, the steam is assumed to leak from the closed isolation valves. The leakage rate of the closed main steam isolation valves is the design value with margin, and this leakage rate is constant.

(15) After the occurrence of the accident, it is assumed that the reactor pressure is kept at the set-point pressure of the safety-relief valves for a long period of time, and the leak from the main steam system is assumed to continue for an infinite period time.