Safety Analysis
in Application for Reactor Establishment Permit

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Contents

1. Requirements

1.1 Basic Requirement

Structures, systems and components important to safety in nuclear reactor facilities are required to perform specified functions not only during normal operation, but also in abnormal conditions to ensure safety. In confirming the adequacy of the basic policy for the safety design of nuclear reactor facilities, it is essential to perform analyses and evaluations concerning any abnormal conditions, i.e. "anticipated operational occurrences" and "accidents". Described below are the events to be postulated, criteria for judgment and matters to be taken into consideration for the safety design evaluation.

1.2 Specific events to be postulated

In order to confine radioactive materials, measures should be taken not to release radioactive materials from fuel rods. Excessive temperature rise of fuel pellets and fuel claddings of fuel rods would cause their failures, leading to a loss of the first containment barrier function. The cladding temperature rises due to an increase in heat generation rate and/or a decrease in heat removal rate. The heat removal rate goes down with a change of boiling modes or a lowering of cooling water level in the core. The boiling modes vary with the coolant flow rate and/or a decrease in the coolant pressure, and the cooling water level goes down with a reduction in the coolant inventory.

Moreover, assumed damage to the coolant pressure boundary would result in a loss of the coolant leading to a decrease in the coolant pressure and inventory. Further assumption of a simultaneous loss of off-site power would lead to a decrease in coolant flow rate or lowering of the reactor water level, resulting in cladding-tube temperature increase. Thus, changes in conditions surrounding fuel rods in the core would progress in a direction to impair the first confinement barrier of radioactive-materials.

By repeating the above-mentioned analyses, all the events assumed to be generated by single failure of structures, systems, or components consisting of light water reactors or by a single erroneous operator action are investigated. And the event which makes it the most difficult to maintain the integrity of multi-layer barriers such as fuel rods is selected out of events with the same event progress. When the event will progress in a different way, it shall be selected as a different event.

The events to be postulated in the safety evaluation and analysis are provided in the "Review Guide for Safety Assessment of Light Water Nuclear Power Reactor Facilities" issued by the Nuclear Safety Commission of Japan. The review guides serve as criteria to verify that the contents of an application for establishment permit conform to those guides. That is, when the contents conform to the review guides, the basic principles for safety design of a nuclear reactor facility are judged adequate ("safety design assessment".)

(1) Abnormal operational transients

Events during reactor operation which may lead to abnormal states caused by single component failures, single component malfunctions or single erroneous operations that are anticipated to occur during the facility lifetime and by disturbances with a similar probability of occurrence.

Representative events shall be selected for evaluations from among events which may potentially lead to excessive damage to the core or to the reactor coolant pressure boundary if the nuclear reactor facility is left uncontrolled. For these selected events, it is required to confirmed the adequacy of the design of structures, systems and components belonging to mitigation systems for abnormal conditions, or simply referred to as "mitigation systems (MSs)," such as the safety protection system and the reactor shutdown system.

Specifically, the events are those that may cause the following abnormal conditions:

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

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

The criteria are as follows;

1) The minimum critical heat flux ratio or the minimum critical power ratio shall be larger than the acceptable limit.
2) Fuel cladding shall not be mechanically damaged.
3) Fuel enthalpy (**) shall not exceed the acceptable limit.
4) Pressure on the reactor coolant pressure boundary shall not exceed 110% of the maximum allowable working pressure.

The events to be postulated are shown in Table-1.

(*) Minimum critical power ratio (MCPR); ratio between the maximum heat that can be removed by the water flowing around the surface of fuel rods and the heat generated by the fuel rods. Therefore, fuel rods with larger MCPR can be cooled by a larger margin, indicating that they are on the safer side. The allowable limit of the MCPR, safety criterion, is set at 1.07 with account taken of a margin for boiling water reactors, judging from various experimental results. The safety criterion for pressurized water reactors is 1.17, and this difference comes from the difference in reactor types. The minimum critical heat flux ratio is defined as the minimum ratio between limiting heat flux (heat flux to cause a departure from nucleate boiling) and actual heat flux in the core.

(**) Fuel enthalpy; thermal energy stored into a fuel, which serves as a measure in evaluating the fuel integrity during reactivity accidents. The large fuel enthalpy could lead to excessive thermal expansion or melting of fuel pellets, resulting in fuel-cladding damage.

Table-1 Abnormal Operational Transients

Events to be evaluated		Type of reactor	Description of events
Abnormal change in reactivity or power distribution in the core			
1	Abnormal withdrawal of control rods during reactor startup	PWR	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	Abnormal withdrawal of control rods during power operation	PWR&BWR	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.
3	Drop and inconsistency of control rods	PWR	An event of power distribution change in a reactor core caused by an anomaly in arrangement 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.
4	Abnormal dilution of boron in the reactor coolant	PWR	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.
Abnormal change in heat generation or heat removal in the core			
1	Partial loss of reactor coolant flow	PWR&BWR	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	Inadvertent start up of the inactive loop of the reactor coolant system	PWR&BWR	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.
3	Loss of offsite power	PWR&BWR	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.
4	Loss of main feed water flow	PWR	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.
5	Abnormal increase of steam load	PWR	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.

Events to be evaluated		Type of reactor	Description of events
6	Abnormal depressurization of the secondary cooling system	PWR	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.
7	Excessive water supply to a steam generator	PWR	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.
8	Loss of feed water heater	BWR	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.
9	Malfunction of the reactor coolant flow control system	BWR	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.
Abnormal change in reactor coolant pressure or reactor coolant inventory			
1	Loss of load	PWR&BWR	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	Abnormal depressurization of the reactor coolant system	PWR	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.
3	Inadvertent startup of the emergency core cooling system during power operation	PWR	An event of inadvertent ECCS startup during reactor power operation is assumed.
4	Inadvertent closure of the main steam isolation valves	BWR	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.
5	Failure of the feed water control system	BWR	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.
6	Failure of the reactor pressure control system	BWR	An event of main-steam flow change due to a failure of the pressure control system, etc. during reactor power operation is assumed.
7	Complete loss of feed water flow	BWR	An event of feed water flow decrease by a failure of the feed water control system, etc. during reactor power operation is assumed.

(2) Accidents

Representative events shall be selected for evaluation from among the events in which radioactive materials released from the nuclear reactor facility may potentially affect the surrounding area of the site, from the standpoint of confirming the adequacy of the design of structures, systems and components mainly belonging to MSs such as engineered safety features. The events for evaluation shall address the following abnormal states:

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

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

(1) The core shall not be considerably damaged, and can be sufficiently cooled.
(2) Fuel enthalpy shall not exceed the specified limit.
(3) Pressure on the reactor coolant pressure boundary shall not exceed 120% of the maximum allowable working pressure.
(4) Pressure on the reactor containment boundary shall not exceed the maximum allowable working pressure.
(5) There shall be no significant radiological risk to the public

The events to be postulated are shown in Table-2.

Table-2 Accident events

Events to be evaluated		Type of reactor	Description of events
Loss of reactor coolant or considerable change in core cooling			
1	Loss of the reactor coolant	PWR&BWR	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	Loss of reactor coolant flow	PWR&BWR	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.
3	Locked-rotor of a reactor coolant pump	PWR&BWR	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.
4	Main feed water pipe break	PWR	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.
5	Main steam line break	PWR	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.
Abnormal reactivity insertion or rapid change in reactor power			
1	Control rod ejection	PWR	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	Control rod drop	BWR	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.
Abnormal release of radioactive materials to the environment			
1	Failure of a radioactive gaseous waste processing facility	PWR&BWR	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	Main steam line break	BWR	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.
3	Steam generator tube break	PWR	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

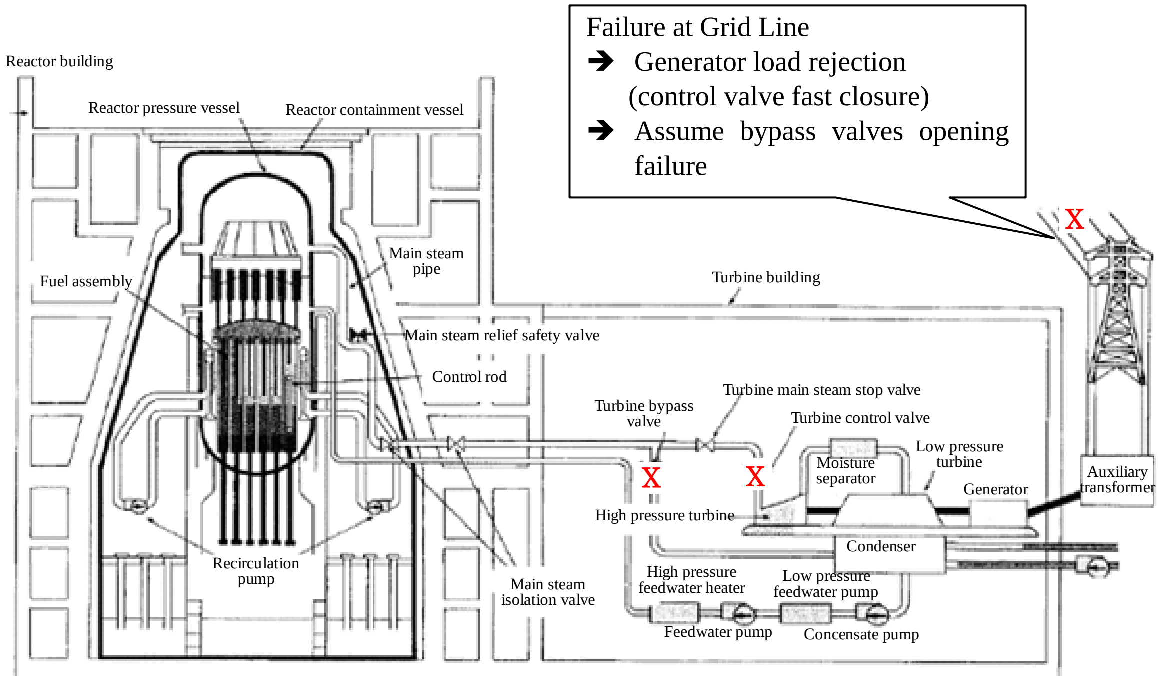
Events to be evaluated		Type of reactor	Description of events
			the steam generator during reactor power operation.
4	Drop of a fuel assembly	PWR&BWR	An event is assumed that a fuel assembly drops for some reason and fails during reactor refueling which in turn result in the release of radioactive material to the environment.
5	Loss of the reactor coolant	PWR&BWR	An event is assumed that radioactive material is released to the environment during a loss of reactor coolant.
6	Control rod ejection	PWR	An event is assumed that radioactive material is released to the environment at the occurrence of the control rod ejection.
7	Control rod drop	BWR	An event is assumed that radioactive material release to the environment at the control rod drop.
Abnormal change in pressure, atmosphere, etc. in the reactor containment			
1	Loss of the reactor coolant	PWR&BWR	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	Generation of combustible gas	PWR&BWR	An event is assumed that combustible gas is generated during a loss of the reactor coolant.
3	Generation of dynamic load	BWR	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. Examples of analysis

2.1 Abnormal Operational Transients

(1) For BWR

"Load rejection" is selected as an example to explain about "abnormal operational transients." The load rejection is one of the abnormal transient events, resulting in reactor power rise due to increase of reactor coolant pressure. The load rejection is an event that increases the coolant pressure due to a rapid reduction in steam flow to a turbine by a rapid turbine-control-valve closure activated by trouble of a power transmission line or failure of a generator during power operation of a nuclear reactor as shown in Figure 2-1. When lightning hits or a grounding fault occurs on a power transmission line, its protection circuit opens the main breaker to shut down generating power supplies connected to the line and the turbine control valves by a rapid and simultaneous close. Rapid close of the turbine control valves is to prevent the turbine from overspeed by load rejection of the generator. It is necessary to control the overspeed to the minimum.



The safety analysis is performed assuming the load rejection with a simultaneous failure of the turbine bypass valves. The turbine bypass valves are not components important to safety, and they are controlled by a pressure control system of the nuclear reactor with a function to control a reactor pressure rise by opening the turbine bypass valves when the reactor pressure increases as in the case of load rejection, etc. The turbine bypass valve system consists of two or more valves, and its total capacity is 25% of the rated steam flow at many nuclear power plants. In order to conservatively evaluate the pressure rise in the analysis, all turbine bypass valves failure is assumed.

The event sequence during load rejection is shown in Figure 2-2.

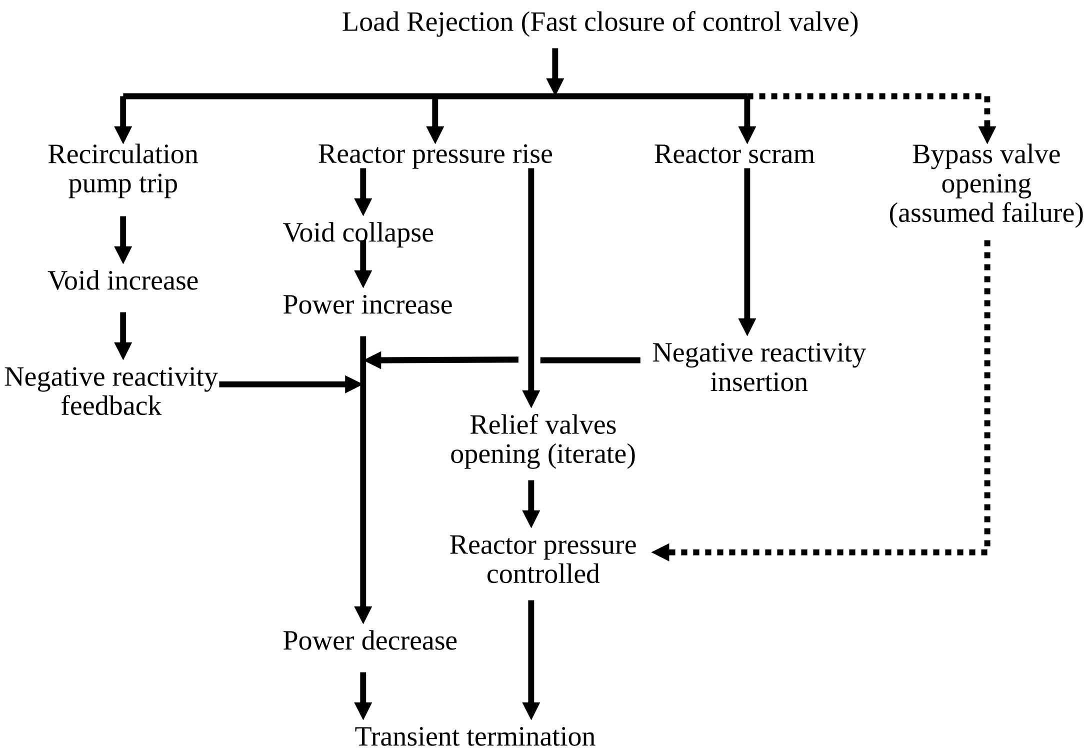


Figure 2-2 Event sequence during load rejection (BWR)

(1) Sensing a rapid-closure signal of turbine-control valves, nuclear reactor scrams.

(2) The reactor pressure temporarily goes up because of a steam shutoff due to rapid closure of turbine-control valves and failed turbine-bypass valves, which reduces the void fraction in the reactor core resulting in the reactor power rise.

(3) Reactor recirculation pumps stop because off-site power supply is unavailable due to a failure of power transmission line, and the core flow goes down, resulting in an increase in core void fraction, which increases the negative reactivity to control the reactor power rise.

(4) A temporary rise of the reactor pressure opens the relief-safety valves when it reaches their setpoints, which transfers the steam to the suppression pool of the containment resulting in suppression of the reactor pressure rise.

(5) Through such sequences, reactor power and reactor pressure transients will terminate.

The analysis results for1100 MWe BWR are shown in Figure 2-3-A and Figure 2-3-B. The calculations of nuclear reactor power, pressure etc. are performed using a dynamic-characteristics analytical code. Moreover, the minimum critical power ratio is calculated using parameters, such as reactor power and pressure in Figure 3-3-A with detailed calculation codes of thermodynamics in boiling region around fuel rods.

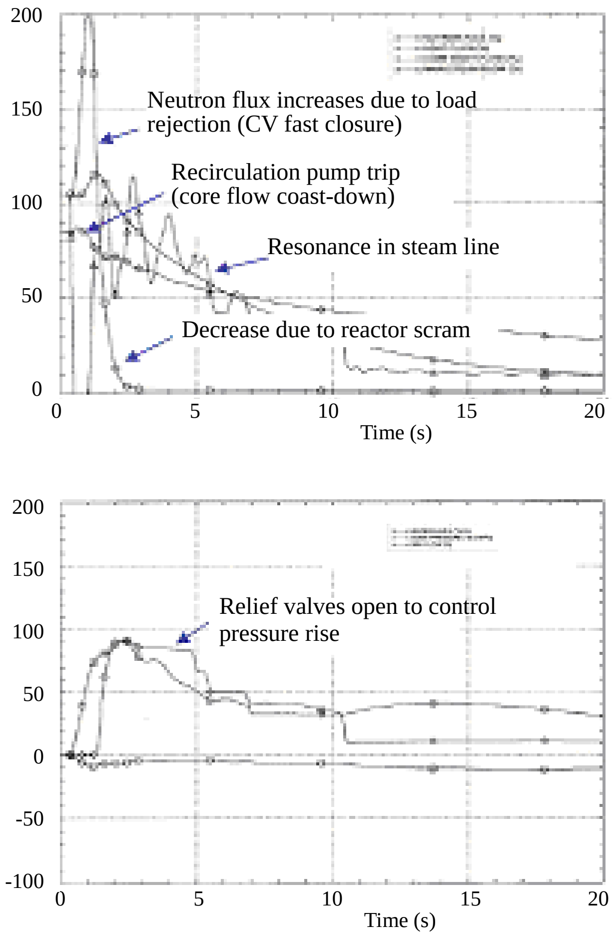


Figure 2-3-A Transient analysis curve for load rejection (BWR)

Description of the analysis

(1) The initial operating power is 105 % of rated power. Initial core inlet flow rate is 85% of the rated flow rate. (For these initial conditions, conservative values are used for thermal conditions)

(2) The neutron flux goes up to a maximum of 200 % of the initial value as the void reactivity increases due to an increase in reactor pressure, and goes down rapidly following the control rods insertion by scram.

(3) The heat flux increases by approx. 10 %. The heat flux is generated several seconds later after the neutron flux increases, and it is a factor to increase fuel temperature.

(4) The reactor pressure temporarily goes up, but an opening of the safety relief valves reduces the pressure.

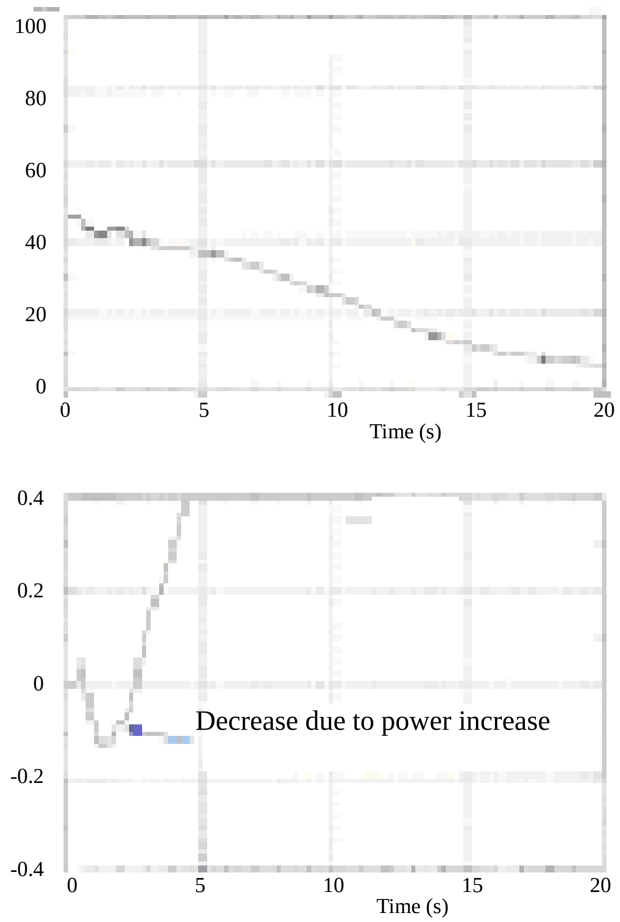


Figure 2-3-B Transient analysis curve of MCPR during load rejection (BWR)

Description of the analysis

(1) MCPR goes up momentarily due to scram, but it goes down because of a heat flux increase and a decrease in heat removal capacity due to the core flow reduction. A MCPR decrease

means a fuel temperature increase.

(2) MCPR recovers with a heat flux reduction.

(3) In the transient curve, it is expressed as an increment from the initial MCPR value. (Δ MCPR)

Judgment

One of the purposes of this safety evaluation is to verify that its results meet the following safety criteria, assuming the conservative initial calculation conditions; high fuel temperature and low heat removal capability resulting from initial reactor power 105 % of the rated power and initial core flow 85 % of the rated flow, and inoperable turbine bypass valves.

(1) The minimum critical power ratio shall be larger than the acceptable limit.
(2) Fuel cladding shall not be mechanically damaged.
(3) Fuel enthalpy shall not exceed the acceptable limit.
(4) Pressure on the reactor coolant pressure boundary shall not exceed 110% of the maximum allowable working pressure.

(2) For PWR

"Load rejection" is selected as an example to explain about "abnormal operational transients." The load rejection is an event that increases the coolant pressure due to a rapid reduction in steam flow to a turbine by a rapid turbine-control-valve closure activated by trouble of the power transmission line or a failure of a generator during power operation of a nuclear reactor as shown in Figure 2-4. The reason to select the same "load rejection" transient as an example for PWR as in the case for a BWR is to show the difference due to their reactor types.

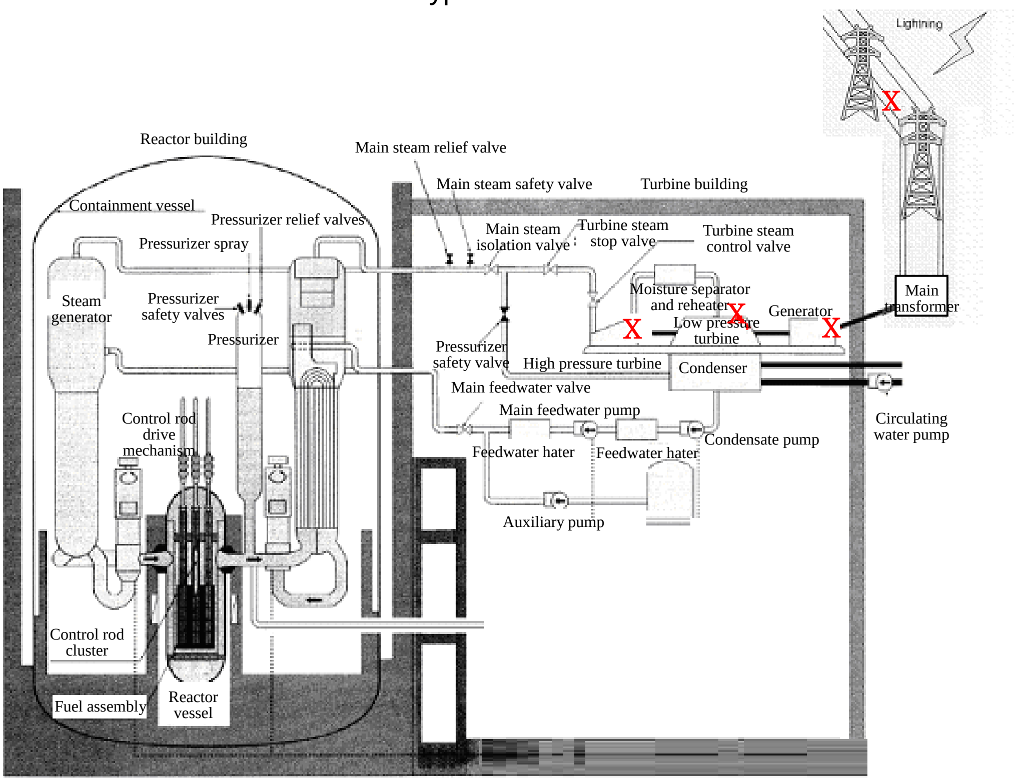


Figure 2-4 Abnormal operational transient, "load rejection"

The event sequence during load rejection is shown in Figure 2-5.

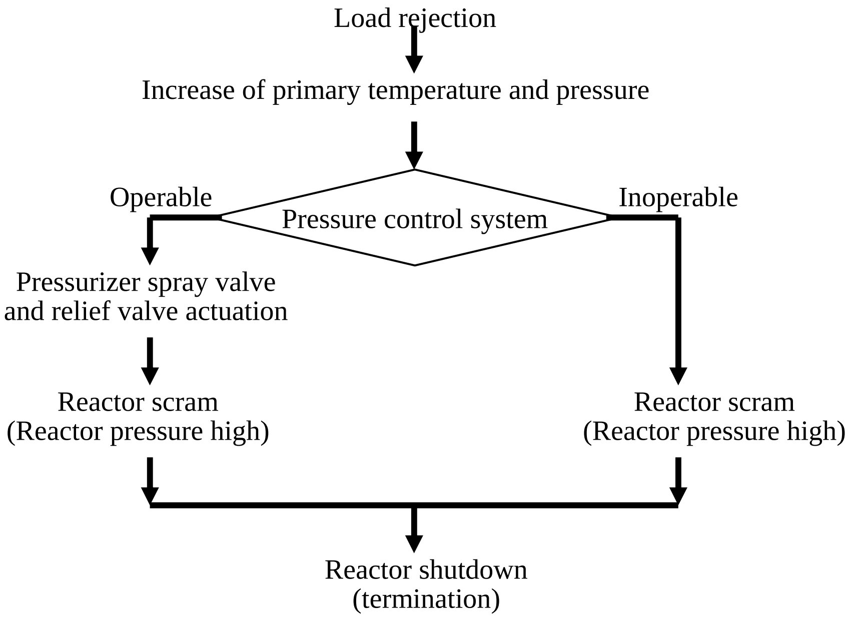


Figure 2-5 Event sequence during load rejection (PWR)

(1) Detecting a pressure rise, pressurizer spray valves and relief valves are opened.
(2) Detecting reactor-pressure high signals, the nuclear reactor scrams.
(3) Through such sequences, reactor power and reactor pressure transients will terminate.

The analysis results of 1100 MWe PWR are shown in Figure 2-6. The calculations of nuclear reactor power, pressure etc. are performed using a dynamic-characteristics analytical code.

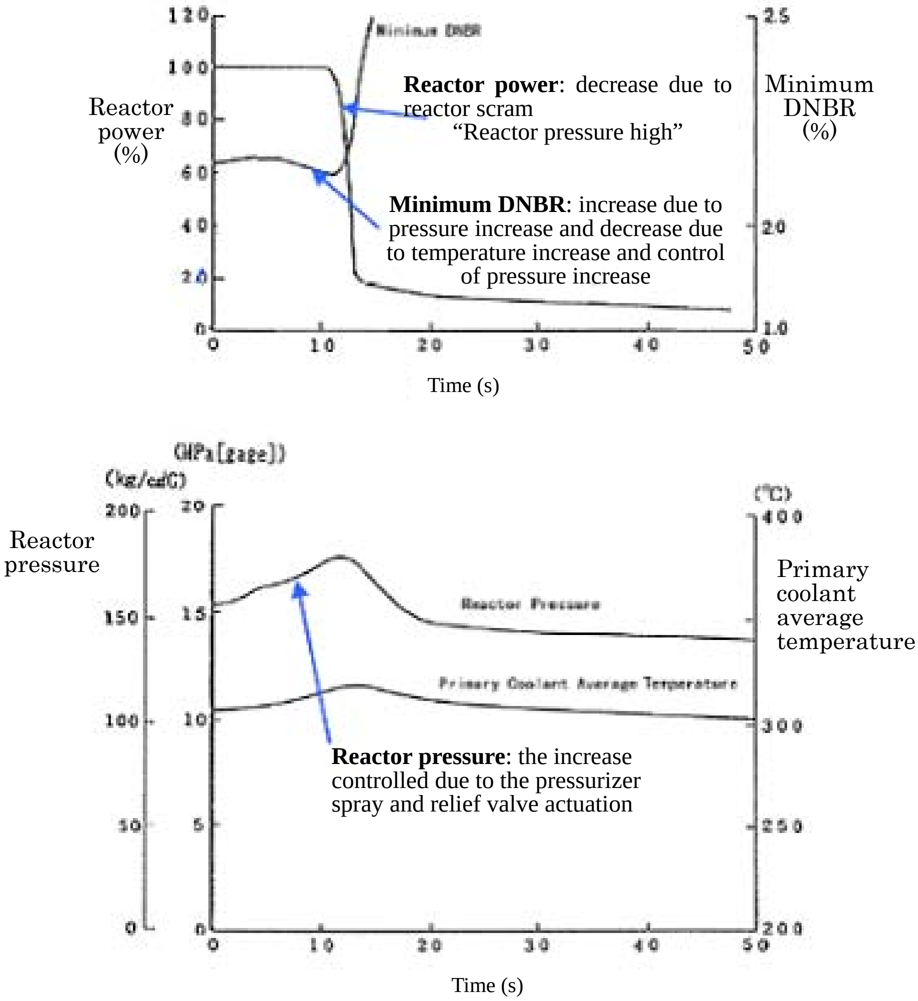


Figure 2-6 Transient analysis curve for load rejection (PWR)

Description of analysis

(1) The reactor pressure goes up at first, but its pressure rise is mitigated by pressurizer spray valves and relief valves.

(2) The reactor power decreases due to reactor scram.

(3) MCPR tends to decrease at first due to coolant temperature increase, but it increases due to the scram and mitigation of the reactor pressure rise.

Judgment

One of the purposes of this safety evaluation is to verify that its results meet the following safety criteria, regardless of the pressure mitigation functions of pressurizer spray valves and relief valves. (As an example, the case with the pressure mitigation functions is shown)

(1) The minimum critical heat flux ratio shall be larger than the acceptable limit.

(2) Fuel cladding shall not be mechanically damaged.

(3) Fuel enthalpy shall not exceed the acceptable limit.

(4) Pressure on the reactor coolant pressure boundary shall not exceed 110% of the maximum allowable working pressure.

2.2 Accident

(1) For BWR

The "loss of coolant" is selected as an example for explanation of "accidents". A loss of coolant is an accident that recirculation piping ruptures during operation and the coolant flows out from a reactor pressure vessel, as shown in Figure 2-7. The functions of the emergency core cooling system (ECCS), integrity of the reactor containment, and the functions to confine radioactive materials against this accident are verified by analysis. The loss of coolant accident of BWR is shown in Figure 2-7.

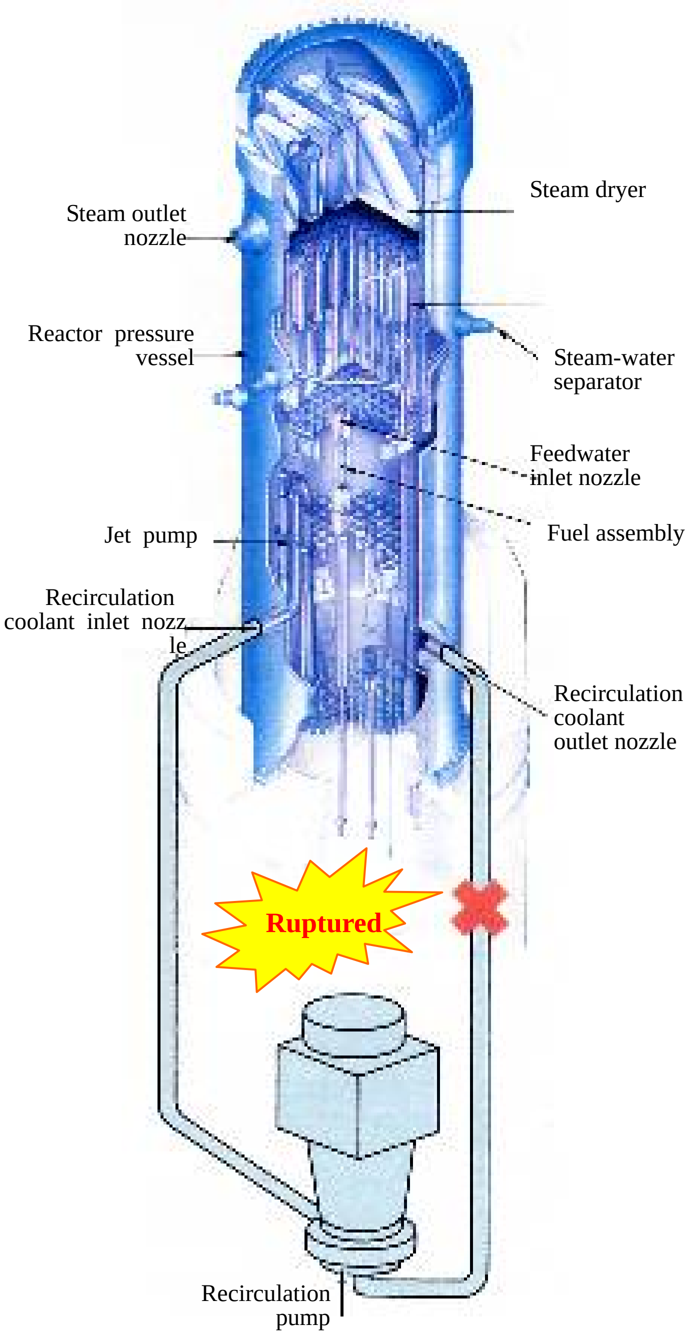


Figure-2-7 Loss of coolant accident (BWR)

The following conditions are assumed to perform a conservative evaluation.

(1) Loss of coolant at a rate of 200% of the rated coolant flow due to a double-ended rupture of one recirculation loop piping

(2) Loss of offsite power

(3) Failure of one ECCS (inoperable low pressure core spray system)

The event sequence during LOCA is shown in Figure 2-8.

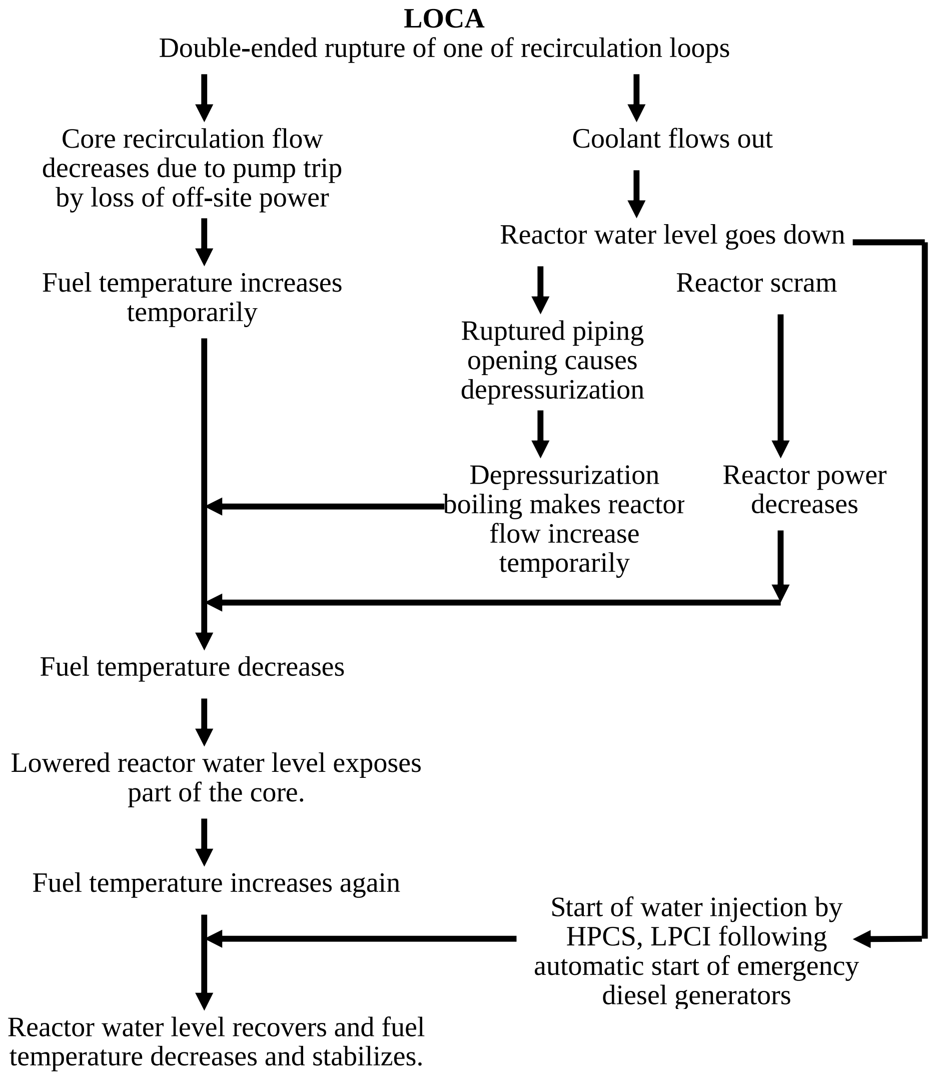


Figure 2-8 Event sequence in a loss of coolant accident (BWR)

An example of analysis results for a typical 800 MWe BWR is shown in Figure 2-9.

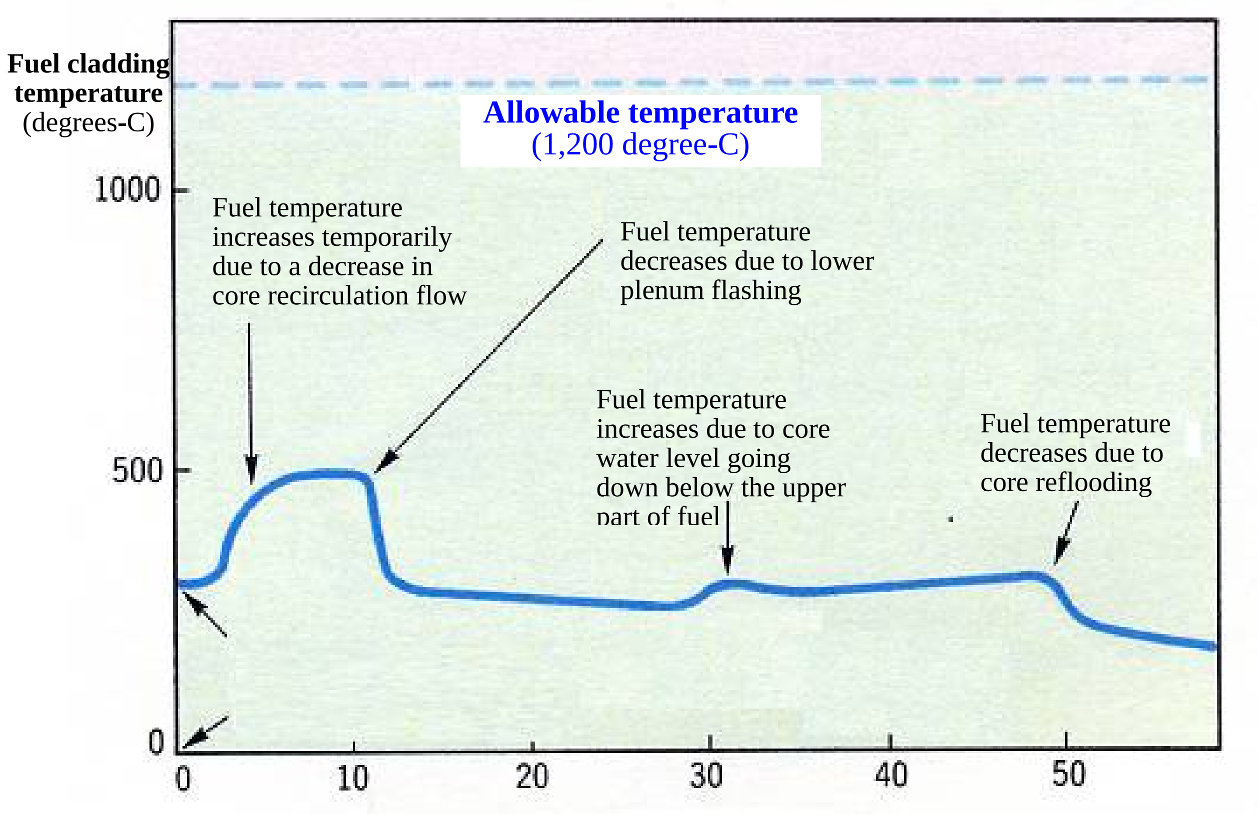


Figure 2-9 Analysis of a loss of coolant accident (typical 800 MWe BWR)

Description of analysis and judgment

(1) The fuel-cladding-tube temperature changes with the reactor water level. The maximum temperature is about 520 degrees C, which is below the allowable limit (1200 degrees C), specific criterion.

(2) The thickness of fuel cladding oxidation is approx. 0.002% of the fuel-cladding thickness, which is blow the allowable limit (15%), specific criterion.

(3) The thickness of fuel cladding oxidation is approx. 0.002%, which is below the allowable limit (1% or less of the total amount of fuel cladding in a core), specific criterion.

(4) The decay heat removal over a long period of time is ensured by operation of any one pump of ECCS.

The above-mentioned (1), (2), and (4) show that "the core would not seriously be damaged, and can be adequately cooled." The above (3) shows that "the amount of hydrogen generated by fuel cladding and structural material reaction with water in a core is sufficiently small not to cause hydrogen combustion, which can ensure the containment integrity."

(2) For PWR

The "loss of coolant" is selected as an example for explanation of "accidents".

A loss of coolant is an accident that coolant flows out from a reactor vessel assuming that the pipe with the maximum amount of coolant flow (one from the primary-coolant-pump outlet to the inlet nozzle of the reactor vessel) ruptures during reactor operation as shown in Figure 2-10. The function of the emergency core cooling system (ECCS), integrity of the reactor containment, and the function to confine radioactive materials are verified by analysis.

The loss of coolant accident of PWR is shown in Figure 2-10

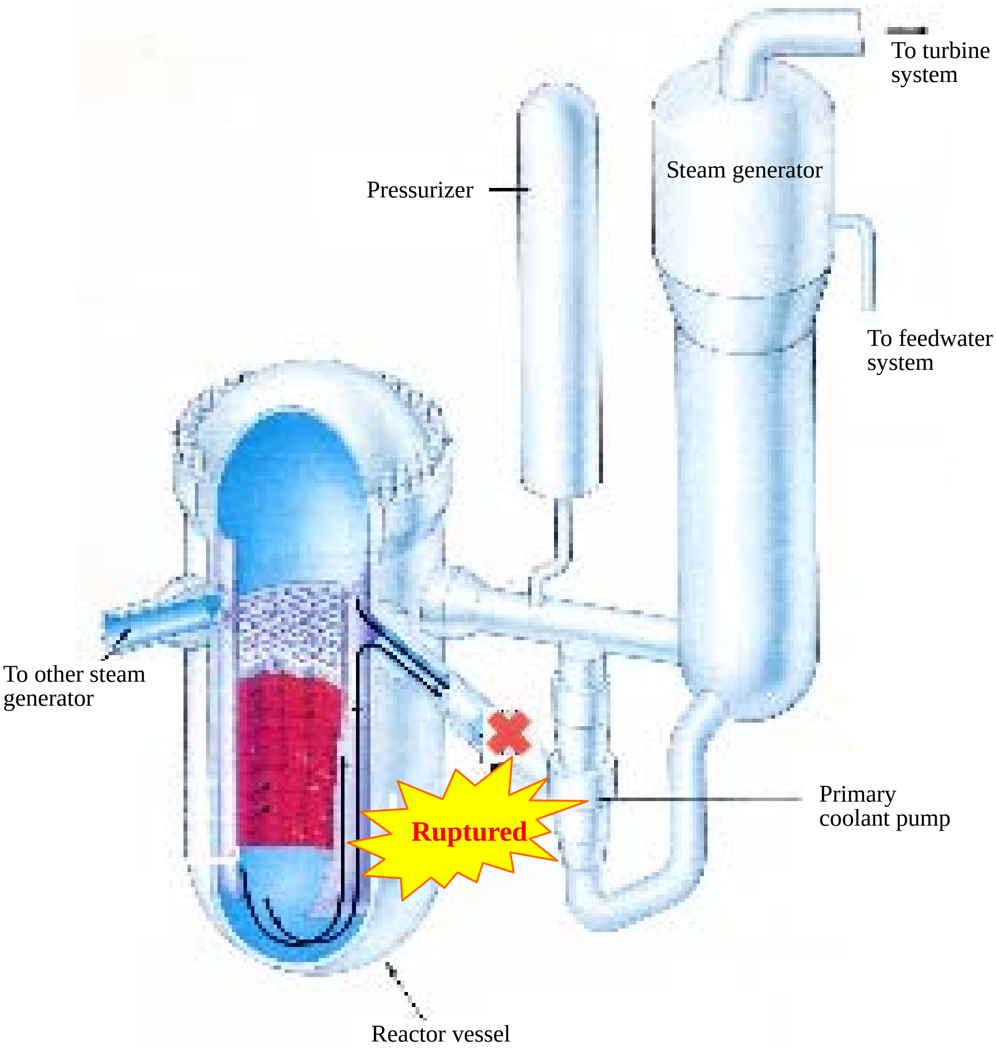


Figure 2-10 Loss of coolant accident (PWR)

The following conditions are assumed to perform a conservative evaluation.

- A double-ended rupture is assumed to a pipe with the maximum amount of coolant flow
- Loss of offsite power
- Failure of one ECCS (one inoperable train out of two low pressure core spray system trains)

The event sequence during LOCA is shown in Figure 2-11.

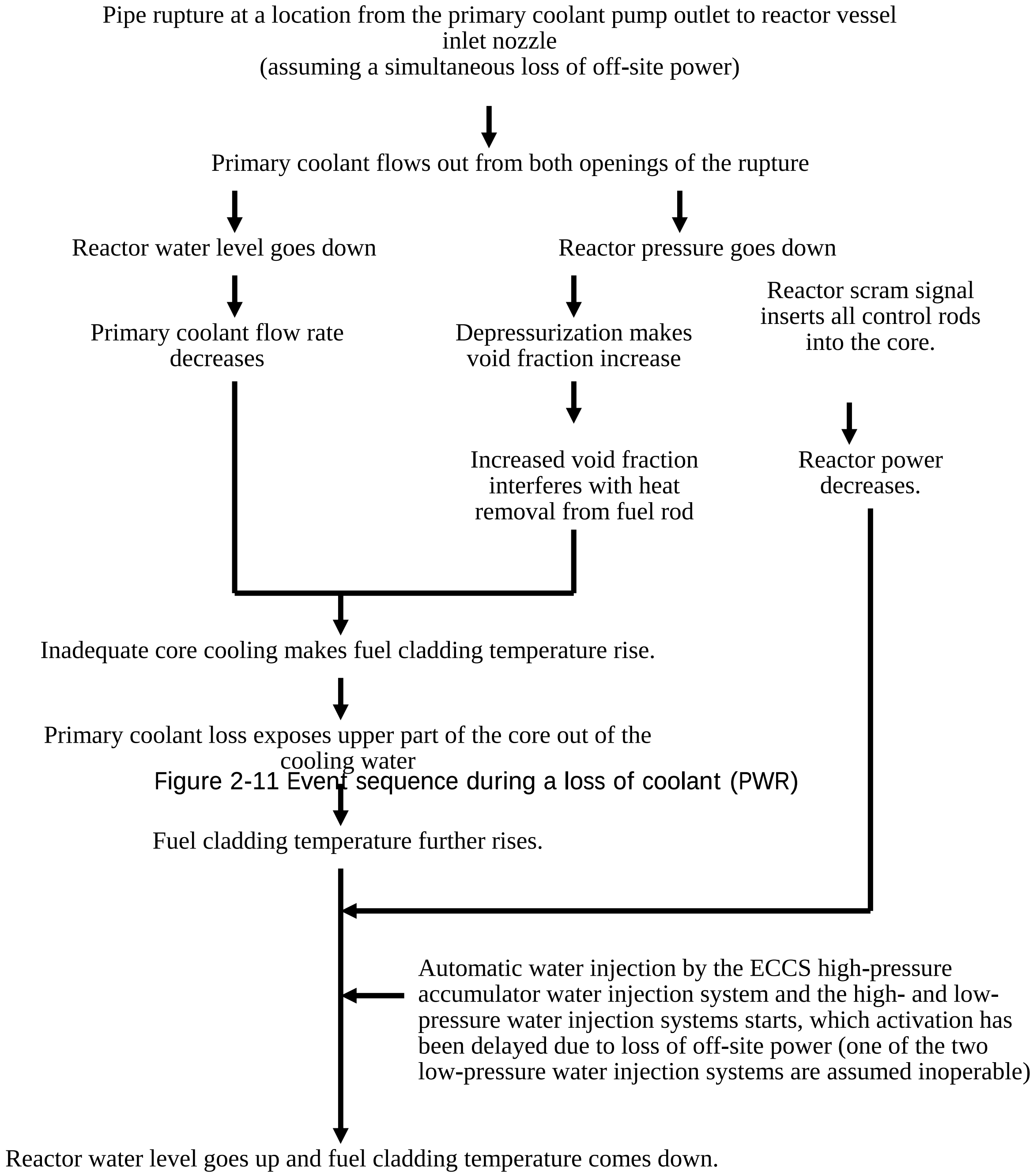


Figure 2-11 Event sequence in a loss of coolant accident (PWR)

An example of analysis results of a typical 1,100 MWe PWR is shown in Figure 2-12.

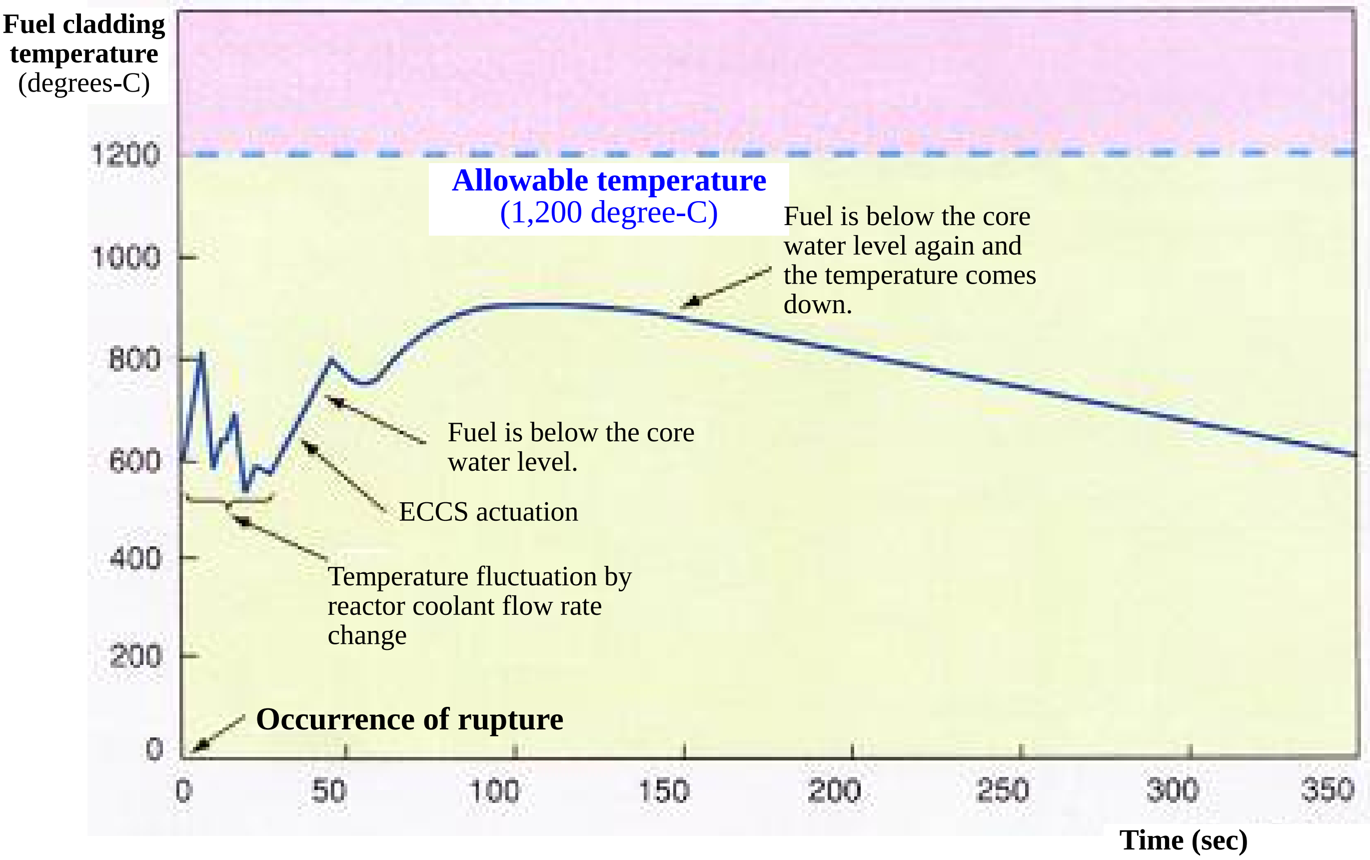


Figure-2-12 Analysis of a loss of coolant accident (typical 1,100MWe PW

Description of analysis and judgment

(1) The fuel-cladding-tube temperature changes with the reactor water level. The maximum temperature is approx. 930 degrees C, which is below the allowable limit (1200 degrees C), specific criterion.

(2) The thickness of fuel cladding oxidation is approx. 1% of the fuel-cladding thickness, which is below the allowable limit (15%), specific criterion.

(3) The thickness of fuel cladding oxidation is approx. 1 % even at its maximum location, which is below the allowable limit (1% or less of the total amount of fuel cladding in a core), specific criterion.

(4) The decay heat removal over a long period of time is ensured by operation of any one pump of ECCS.

The above-mentioned (1), (2), and (4) shows that "the core would not seriously be damaged, and can be adequately cooled." The above (3) shows that "the amount of hydrogen generated by fuel cladding and structural material reaction with water in a core is sufficiently small so as not to cause hydrogen combustion, which can ensure the containment integrity."

3. Conclusions

Examples of the analytical evaluations of "abnormal operational transients" and "accidents" are shown above. In a process to issue an establishment permit, all of the events shown in Table-1 and Table-2 are evaluated, and the adequacy of evaluations submitted by a utility for the application is reviewed. The conditions to be assumed for their analyses and their specific and quantitative criteria are provided for each event. For details, refer to the "Review Guide for Safety Assessment of Light Water Nuclear Power Reactor Facilities (Nuclear Safety Commission of Japan)."

An electric utility who has received the approval for establishment can start construction of a nuclear power plant. After manufacturing structures and components consisting of a light water reactor and completion of its construction, pre-service inspections are performed to ensure that it works as designed before going into its commercial operation. The pre-service inspections are performed one by one of structures, systems, and components consisting of a light water reactor, and comprehensive inspections are conducted at the final stage of these inspections. During the comprehensive inspections, specific tests are performed on some representative events of "abnormal operational transients", to verify the design adequacy and safety of a nuclear power plant. Naturally, as "accident" conditions cannot be created, verifications of accident events are not performed. For example, the "load rejection" explained in this text is performed at each stage of 25%, 50%, 75%, and 100% of the reactor power. Comparing the test results with analytical results, the adequacy of the analyses will be checked. In addition, typical events to be tested are a loss of offsite power, loss of main feed water flow, closure of main steam isolation valves, etc.

The calculation programs, etc. used for the analysis of a postulated event shall be verified with respect to their applicability. The models and parameters for the analysis shall be specified such that they give a severe result. The use of reasonable models and parameters, however, may be allowable within the context of the evaluation purpose. If there can be uncertainty in the parameters, appropriate safety margins shall be taken into account. Although it is impractical to perform tests simulating accidents during the pre-service inspections, the results of various tests with conditions similar to those of "abnormal operational transients" are evaluated to improve accuracies of "computer programs, models, and parameters."

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