R5: April 27, 2010

Dose Evaluation
in Application for Reactor Establishment Permit

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Incorporated administration agency
Japan Nuclear Energy Safety Organization
JNES

Contents

1. Summary

When intending to establish a nuclear power plant, Article 23 of the Act for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (hereinafter referred to as the "Reactoor Regulation Act) stipulates that "a person who intend to establish a nuclear power reactor facility shall submit an application for establishment permit to the competent Minister and receive a permit from the Minister."

As one of the conditions to receive permit, there is a provision that the location, structure and equipment of a reactor facility shall not imapir prevention of disasters caused by nuclear fuel materials or materials contaminated by nuclear fuel materials (including fission products) or by reactors (iv of Article 24 of the Reactor Regulation Act."

From this reason, an electric utility who intends to establish a nuclear power plant submits an application for the reactor establishment permit, and the National Government performs a safety review to ensure safety.

The safety review includes reviews of siting conditions and safety design and reviews of safety evaluations (evaluation of exposure dose during normal operation, safety evaluation and siting evaluation). Each review is performed pursuant to the "**Review Guide for Nuclear Reactor Siting**" (Review Guide for Nuclear Reactor Siting and Reference Criteria Concerning its Application), the "Review Guide for Safety Design" (Review Guide for Safety Design of Light Water Nuclear Power Reactor Facilities) and the "Review Guide for Safety Evaluation" (Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities), respectively.

The "Review Guide for Safety Evaluation" requires that the contents of the application for reactor establishment permit be examined to conform to this guide in the safety review of a light water reactor. When the contents conform to this guide, evaluation of the principles on the safety design of the nuclear reactor facility is considered to be adequate, and evaluation of the distance of the reactor from the public in the vicinity, which is a nuclear reactor siting condition, is judged to be adequate.

Validity of the basic principles on the safety design of the nuclear reactor facility requires that some of the structures, systems and components of the facility should exercise required functions during normal conditions and abnormal conditions that exceed the normal conditions. Therefore, it is necessary to analyze and evaluate abnormal conditions, i.e. anticipated operational occurrences and accidents. As for the dose evaluation, the judgment that there is no significant risk of radiation exposure to the surrounding public is required in the accident analysis during the safety evaluation.

Judgment on the suitability of the siting conditions of a nuclear reactor, i.e. siting evaluation, is reviewed pursuant to the "Review Guide for Nuclear Reactor Siting." Whenjudging the suitability of the siting conditions, it is required to evaluate "major accidents" and "hypothetical accidents" and ensure the appropriate distance between the reactor and the surrounding public so that the public doses should lower the reference criteria.

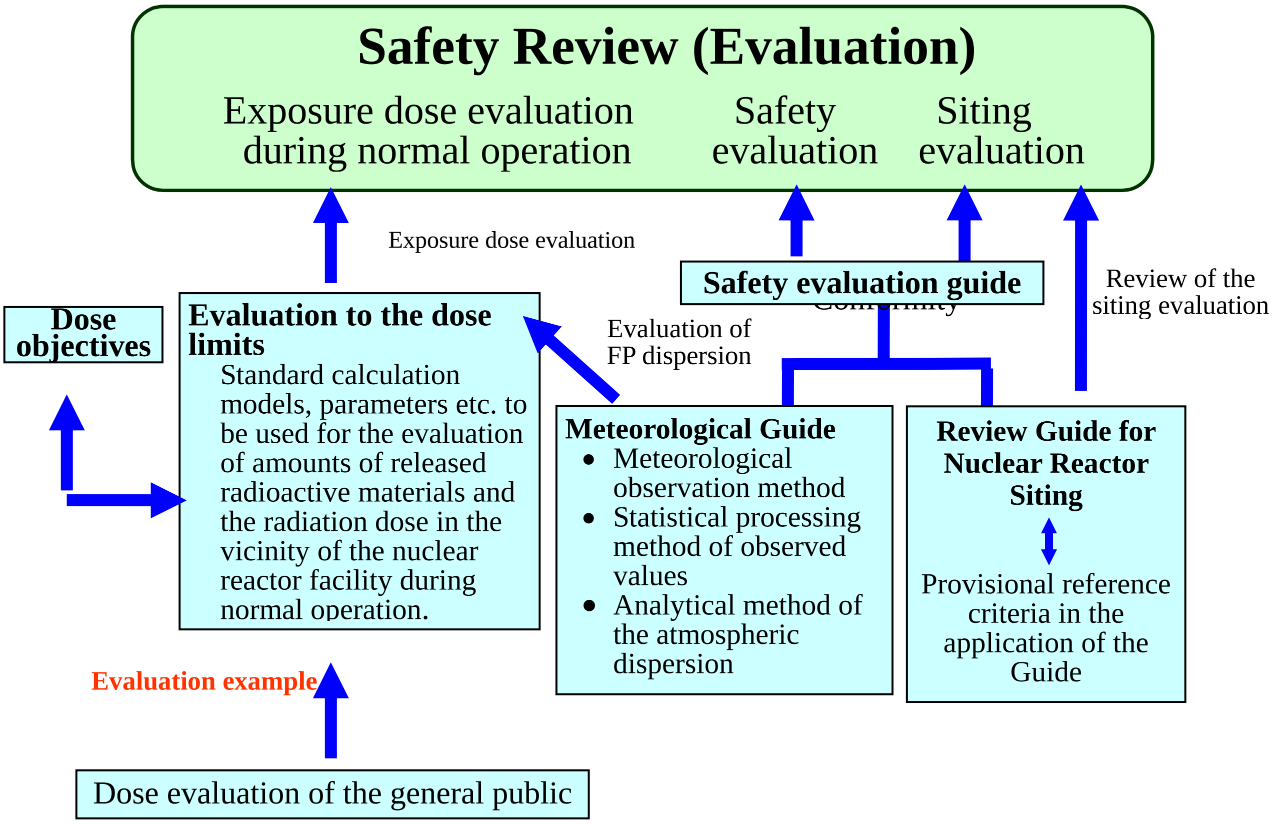
On the other hand, for evaluation of exposure dose during normal operation, the dose objectives to the general public in the vicinity of the facility are specified by the "**Guide for Dose**

Objectives Around Light Water Nuclear Power Reactor Facilities." And, the "**Evaluation Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities**" specifies the standard calculation models, parameters, etc. to be used for evaluations of the amount of released radioactive materials and the dose in order to evaluate the dose around nuclear reactor facilities during normal operation at the basic design stage of nuclear reactor facilities.

As for dose evaluation during accidents and normal operation, meteological observation method, statistical treatment of the observed data, and analysis methods on dispersion to estimate the dispersed conditions of radioacive materials in the air are stipulated in the "Meteorological Guide" (Reference sheet 1: Meteorological Guide for Safety Analysis of Nuclear Power Reactor Facilities)."

Relations among the review guides for safety evaluation etc. are shown in Figure 1-1.

The Safety Review is performed pursuant to the contents provided in these Review Guides.



Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities	Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities
Review Guide for Nuclear Reactor Siting	Review Guide for Nuclear Reactor Siting and Guideline of Judging its Application
Meteorological Guide	Meteorological Guide for Safety Analysis of Nuclear Power Reactor Facilities
Dose Objectives	Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities
Evaluation to dose objectives	Evaluation Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities
Dose evaluation for the general public	Evaluation of Dose Equivalents for the General Public at the Safety Review of Light Water Nuclear Power Reactor Facilities

Figure 1-1 Guides Related to the Safety Review (Evaluation)

2. Safety requirements

2.1 Requirements for siting conditions

The fundamental principles on siting conditions are provided in the "**Review Guide for Nuclear Reactor Siting**." This is used to judge the suitability of the siting conditions for possible accidents in the Safety Review performed prior to establishment permit of a reactor.

First of all, it is natural that a nuclear reactor shall be designed, constructed, operated, and maintained so as not to cause an accident wherever it is installed, but in order to ensure safety of the public, the following siting conditions are required in principle in preparation for possible accidents."

(1) There was no event which induced a large accident in the past and such an event is unlikely to happen even in the future. In addition events that escalate into a disaster are also rare.

(2) The reactor must be sufficiently away from the public in relation to the safety protective facility.

(3) The reactor site, including the surrounding area, should be such that appropriate measures can be taken for the public as necessary.

To put it shortly, a nuclear siting shall be at a location where a major accident would not occur due to an earthquake, wind, tsunami, landslide, etc. and a proper distance between the nuclear power plant and the public residential areas is secured.

Matters to be examined:

1. Site: The distance from the surrounding public is jueged based on the findings regarding location, site size and a site boundary of the nuclear power station.
2. Weather: When installing a nuclear power plant, weather conditions at the site and the vicnity are surveyed such as the lowest temperature, maximum instantaneous wind speed, snow accumulation. In the safety analysis, atmospheric dispersion is analyzed after surveing wind directions, wind speeds, etc. and statistically processing those data. The conformity to the Guide of the analytical methods is reviewed.
3. Ground: Test pitting to survey the ground characteristics of the site is performed to confirm the stability of the ground.
4. Hydrology: To confirm that a site would not be affected by tsunami or flood
5. Earthquake: To select the earthquakes to be taken into consideration for seismic design from the results of studies on the past earthquakes, active faults, etc. in the regions around a site
6. Social environment: To perform surveys on population distribution, industrial activities, traffic and transportation, etc., to determine that those would not affect the safety of a nuclear power plant

2.2 Requirements for evaluation of exposure dose during normal operation

In order to keep dose low, which is received by the public in the vicinity due to radioactive materials released to the environment during normal operation of light water nuclear power reactor facilities, dose objectives are specified and dose assessment is required. And, the evaluation guide for this purpose is established. (Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities)

(1) Dose objectives

As non-binding objectives to keep the dose low, which will be received by the public in the vicinity due to radioactive materials released to the environment during normal operation of light water nuclear power reactor facilities, **dose objectives** to the general public in the vicinity of the facilities shall be **50 micro-Sv per year** in effective dose.

However, at the dose evaluations, external exposure due to γ-rays from radioactive noble gas and internal exposure due to intake of radioactive iodine for radioactive materials in the gaseous waste and internal exposure due to intake of sea foods for radioactive materials in the liquid waste should be evaluated in terms of effective dose.

2.3 Requirements for safety evaluation

The "Safety Evaluation Guide" requires that "anticipated operational occurrences" and "accidents" be analyzed to confirm the validity of the safety design policy of the nuclear reactor facilities, and that the analytical results be evaluated in comparison with the criteria prescribed in the Guide, and they meet the criteria. Dose evaluations for accidents are required in the safety evaluation. The accident is an abnormal state that exceeds "anticipated operational occurrences." When an accident occurs, though the frequency of occurrence is rare, there is a possibility that radioactive materials are released from the nuclear reactor facility. Therefore the "accident" is an event that should be assumed from the viewpoint of evaluating the safety of nuclear reactor facilities.

The most severe events are selected as evaluation events from each of the groups of similar events that could cause a significant impact of radioactive materials released from the nuclear reactor facility on the surrounding area of the site. Analytical conditions, such as the plant state prior to the event occurrence, model and input data, which could give conservative results are used in the accident analysis.

The amount of fission products released during the accident and the effective doses are evaluated for the selected event. Validity of the safety design policy is examined by satisfying the criteria shown in the "Safety Evaluation Guide" for every accident assumed.

Refer to "Reference sheet 2" on dose evaluation to be performed in the Safety Evaluation.

(1) Criteria

According to the "Safety Evaluation Guide," the criteria for events with abnormal release of radioactive materials into the environment are as follows:

"There is no significant risk of radiation exposure to the surrounding public."

If the effective dose to the public living surrounding the site does not exceed 5 mSv per accident, the risk is judged small.

2.4 Requirements for siting evaluation

(1) Basic principle

(a) Siting conditions in principle

The following siting conditions are necessary in principle to ensure public safety in preparation for a possible accident:

- There was no event which induced a large accident in the past and such an event is unlikely to happen even in the future. In addition events that escalate into a disaster are also rare.
- The reactor must be sufficiently away from the public in relation to the safety protective facility.
- The reactor site, including the surrounding area, should be such that appropriate measures can be taken for the public as necessary.

(b) Basic objective

There is a policy to ensure public safety even in the event of an accident and to promote sound development of nuclear power and the following three basic objectives are intended to achieve:

(i) To not cause radiological hazards to the public in the surrounding area even assuming the occurrence of a critical accident (hereinafter referred to as a major accident) that is likely to occur in the worst case from a technical standpoint by considering events around the site, characteristics of the reactor, and safety protective facility.

(ii) To not cause a significant radiological disaster to the public in the surrounding area even if the occurrence of an accident is assumed that is unlikely to occur from a technical standpoint that exceeds a major accident (hereinafter referred to as a hypothetical accident). (For instance, it is assumed that some of the safety protective facilities are not operable, the effect of which was expected in the case of a major accident, and the release of comparable radioactive material is assumed.)

(iii) In the case of a hypothetical accident, the influence on the collective dose must be adequately small.

(2) Guide for siting review

It is necessary to examine the conformity to, at least, the following three conditions to achieve the above-mentioned basic objective upon judging the suitability of siting conditions.

(a) The area around the reactor of a certain distance from the reactor must be an exclusion area.

The "area of a certain distance" here is the area up to a distance that could cause radiological hazards to a person if a person stays inside that distance in the event of a major accident. The exclusion area is the area where the public do not reside as a rule.

(b) The zone outside of the exclusion area as well as of a certain distance from the reactor must be the low population zone (LPZ).

It is defined that, if some measures are not taken in the case of a hypothetical accident, the area that could give a significant radiological disaster to the public in the area is taken as the area of a certain distance. The low population zone (LPZ) is the zone in an environment in which appropriate measures can be taken so as not to give a significant radiological disaster.

(c) The reactor site must be a certain distance apart from the highly populated area.

"A certain distance" is defined as the distance to which the total integrated population dose to the whole body becomes small to an extent sufficiently acceptable from the viewpoint of the collective dose.

(3) The reference criteria

(a) Use the following dose as the reference to judge "the area of a certain distance" that is in (2)(a).

For the thyroid (*):   1.5Sv

*In the calculation, inhalation dose coefficient of one-year-old child are used.

For the whole body: 0.25Sv

(b) Consider the following dose as a rough reference to judge "the area of a certain distance" that is in (2)(b).

For the thyroid (adult):   3Sv

For the whole body: 0.25Sv

(c) Refer to examples of foreign countries (for instance, 20,000 person-Sv) as the reference to judge "a certain distance apart" that is in (2)(c).

3. Evaluations to be performed

Evaluations regarding exposure dose consisit of "dose evaluations during normal operating conditions," "dose evaluations for accidents in the safety evaluations," and "dose evaluations for major accidents and a hyopothetuical accident in the siting evaluation."

3.1 Evaluation of exposure dose during normal operation

The guide is provided o secure dose objectives which were set forth to keep public dose low associated with the release of radioactive material to the environment from light water nuclear power reactor facilities during normal operation (Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities).

(1) Guide for application of dose objectives

(a) In design of light water nuclear power reactor facilities, dose evaluations conducted taking into account the future formation of settlements in the vicinity of the facilities shall meet the dose objectives.

(b) In controlling a release of radioactive materials during normal operation of light water nuclear power reactor facilities, when the same method as that in (a) is used for dose evaluation, achievable dose objectives for an amount of annual release or an average release rate shall be defined, and efforts shall be made not to exceed these objectives.

When a release exceeds the controlling objectives, the following measures shall be taken;

(i) Dose evaluation shall be done to evaluate doses to persons with a standard dietary habit in the existing living areas in the vicinity of the facility using realistic calculation methods and parameters, taking into account actual situations such as the weather conditions during that period, living conditions of the public, and measured results of environmental monitoring samples.

(ii) When the evaluation results of (i) show that the calculated doses are likely to exceed the dose objectives even under standard weather conditions of a year, and also likely to repeatedly exceed the dose objectives after that, efforts shall be made to improve discharge method of radioactive materials and related facilities in order to meet the dose objectives.

The evaluation guide to evaluate these processes is set forth. (Evaluation Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities)

Relations between the Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities and the Evaluation Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities is given in Figure 3-1.

Guide for
Dose Objectives Around Light Water Nuclear Power Reactor Facilities

1. Dose Objectives

Light water nuclear power reactor facilities during normal operation

- The dose objectives which the general public in the vicinity will receive associated with a release of radioactive materials during normal operating conditions:
  50 micro-Sv / year (Effective dose)

2. Evaluation Items of Radiation Doses

Gaseous waste

- External exposure by the gamma ray from radioactive noble gas
- Internal exposure by intake of radioactive iodine into the living body

Liquid waste

- Internal exposure by intake of marine products containing radioactive materials

Evaluation Guide for
Dose Objectives Around Light Water Nuclear Power Reactor Facilities

<u>Exposure path</u>

- Calculation of the effective dose due to the external exposure by the gamma ray from the radioactive cloud moving and diffusing from the release source.
- Calculation of the effective internal exposure dose by inhalation and intakes of green-vegetables, milk etc.
- Calculation of the internal effective exposure dose by intake of marine products containing radioactive materials.

<u>Source of released radioactive material</u>

Calculate the quantity of radioactive materials released to the environment from ventilation stack and drain outlets.

<u>Radioactive material</u>

Radioactive materials with gaseous behavior
  Deal with radioactive noble gas and Iodine-131 and 133.
Radioactive materials with liquid behavior
  Deal with fission products and activated products.

Figure 3-1 Relation Between Dose Objectives and Evaluation Guide

(2) Source calculations of released radioactive materials

In the source calculations of released radioactive materials, the amount of radioactive materials released to the environment from the ventilation stack outlet and the drain outlet taking into account functions and capabilities of the radioactive material treatment system, etc. is evaluated.

The source calculations are performed, focusing on radioactive fission products that behave in gaseous state; noble gas and radioactive iodine (iodine-131, iodine-133) and on radioactive fission products and activated products that behave in liquid state.

- Radioactive materials in gaseous wastes

For PWR, specifying the following items as the calculation conditions, the amount of radioactive materials is calculated by calculation formulas.

<Items for the calculation>

- Load factor of a nuclear reactor facility: 80% per year
- Noble gas and iodine concentrations in the primary coolant
- Noble gas and iodine in the exhaust gas of off-gas attenuation system
- Noble gas and iodine released from ventilation systems
- Iodine-131 released during periodic inspection

The release pathways of noble gas and iodine in gaseous waste of PWR nuclear reactor facility buildings are as shown in Figure 3-2a and Figure 3-2b.

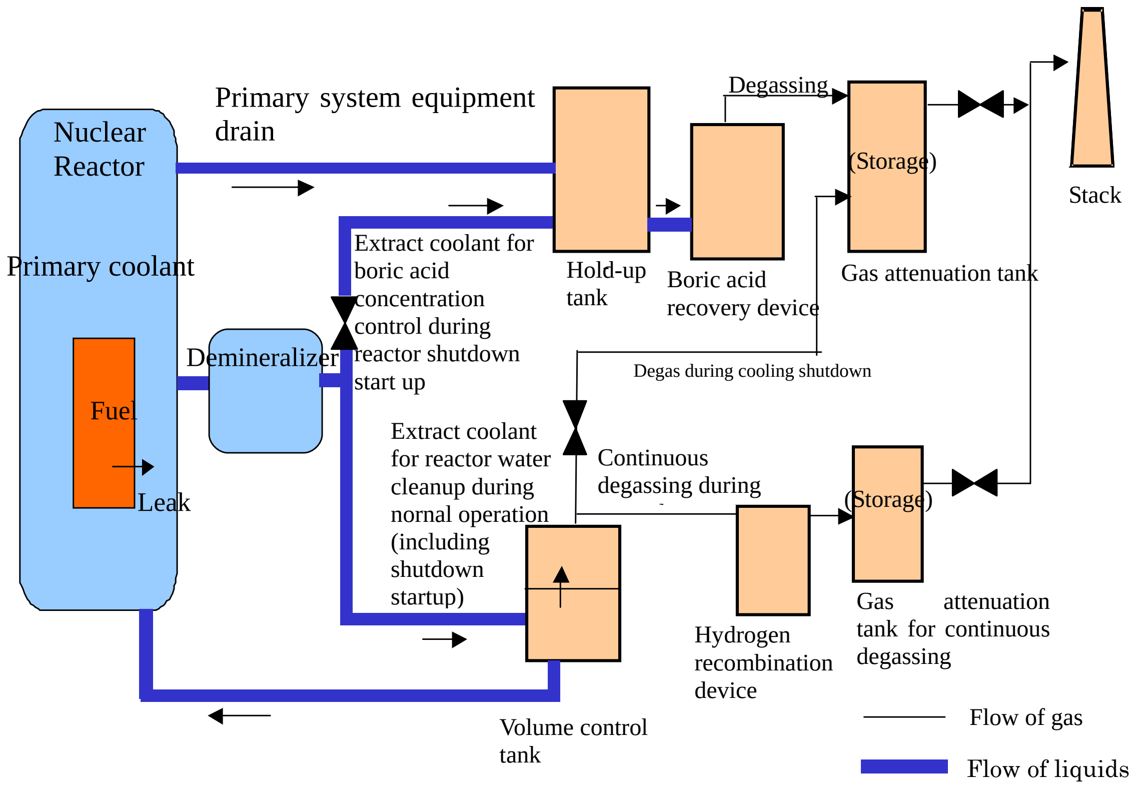


Figure 3-2a Release pathways of noble gas and iodine in gaseous waste of PWR nuclear reactor facilities (Part 1)

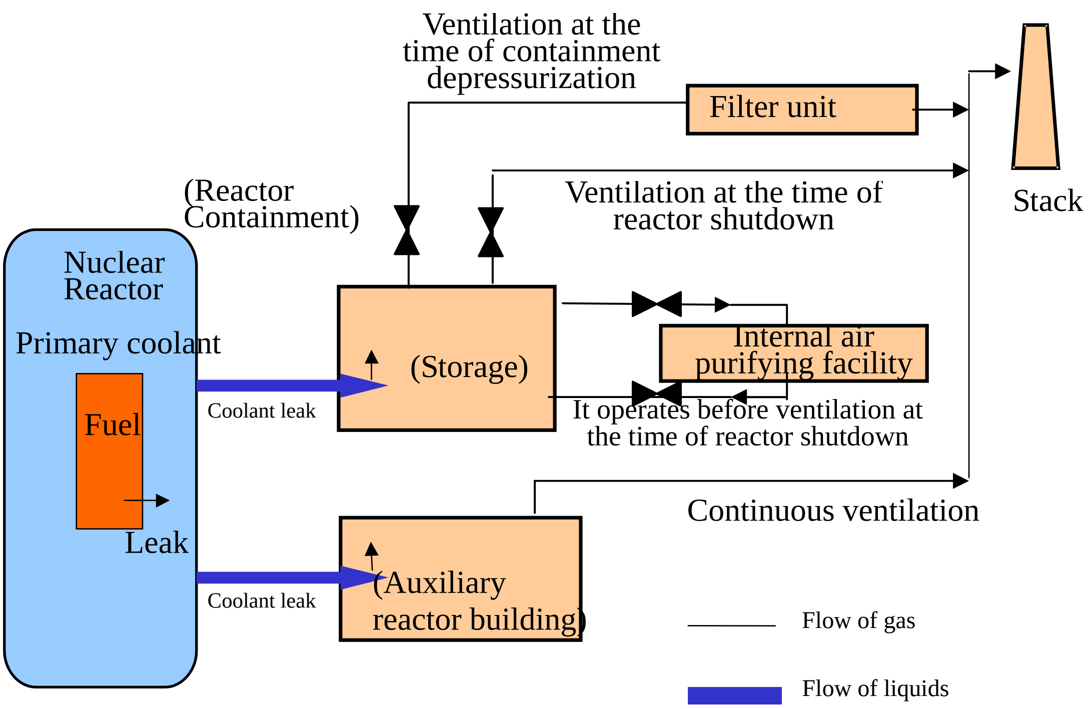


Figure 3-2b Release pathways of noble gas and iodine in gaseous waste of PWR nuclear reactor facilities (Part 2)

For BWR, specifying the following items as the calculation conditions, the amount of radioactive materials is calculated by calculation formulas.

<Items for the calculation>

- Load factor of a nuclear reactor facility: 80% per year
- Noble gas and iodine concentrations in the primary coolant
- Noble gas and iodine in main condenser ejector system exhaust gas
- Noble gas and iodine in turbine shaft seal steam system exhaust gas
- Noble gas and iodine in the exhaust gas of off-gas attenuation system
- Noble gas and iodine released from ventilation systems
- Iodine-131 released during periodic inspection

The release pathways of noble gas and iodine in gaseous waste of BWR nuclear reactor facility buildings are as shown in Figure 3-2c and Figure 3-2d.

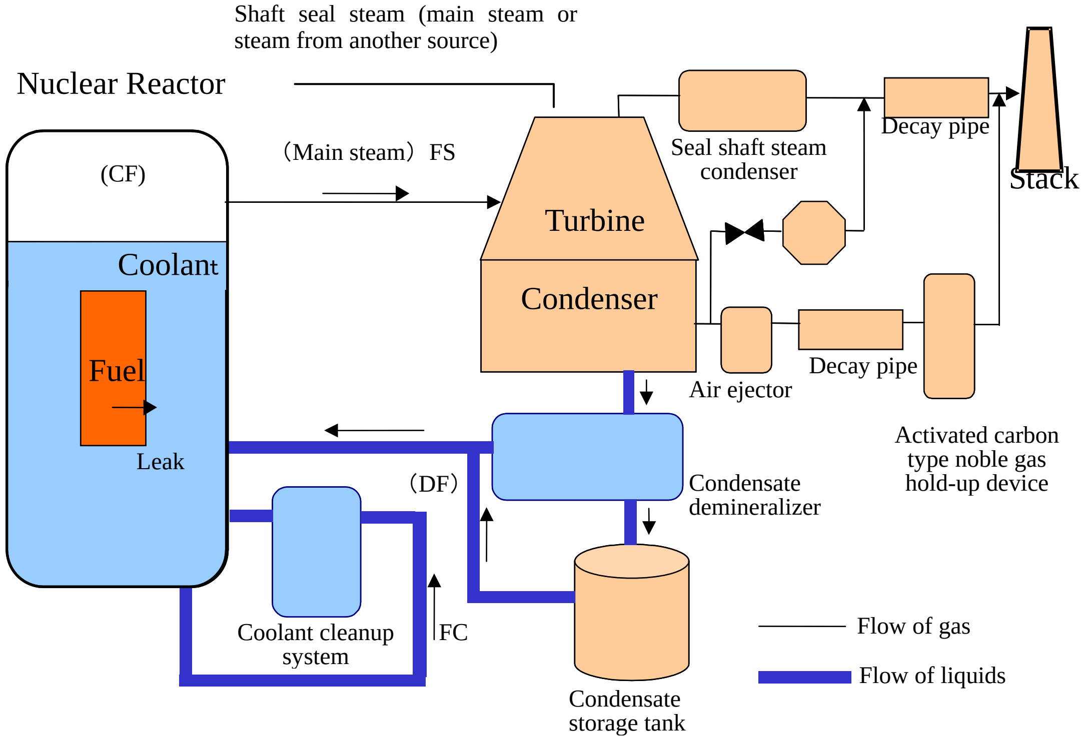


Figure 3-2c Release pathways of noble gas and iodine in gaseous waste of BWR nuclear reactor facilities (Part 1)

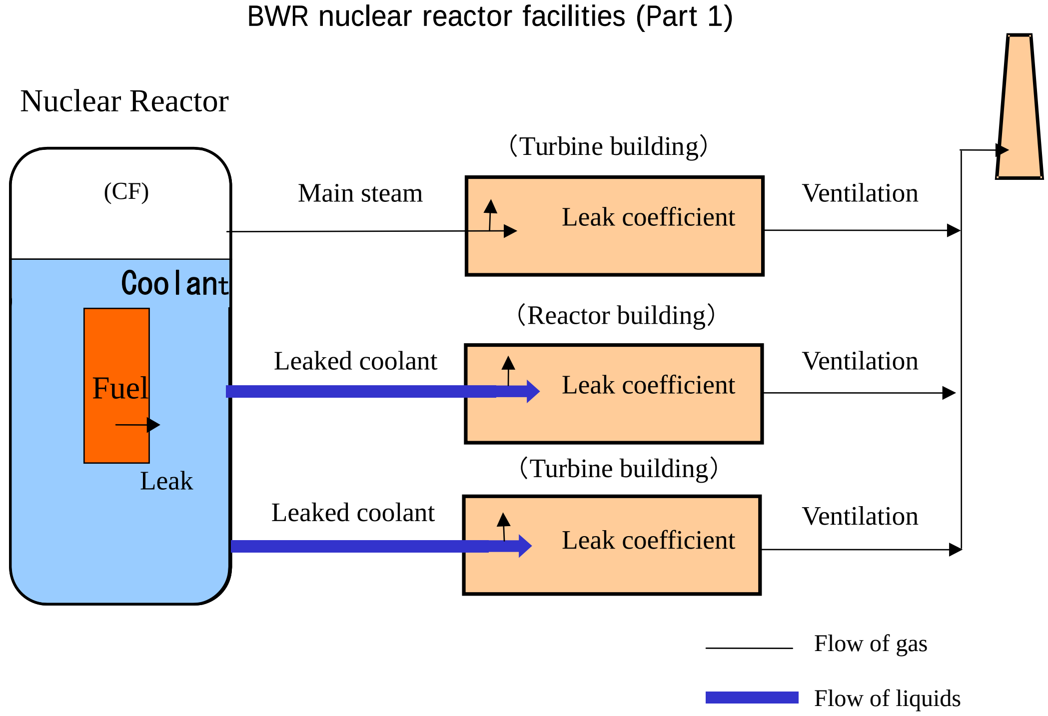


Figure 3-2d Release pathways of noble gas and iodine in gaseous waste of BWR nuclear reactor facilities (Part 2)

- Radioactive materials in liquid waste

  Specifying the following items as the calculation conditions, the amount of radioactive materials is calculated by calculation formulas.

  < Items for the calculation>

  - Amount of radioactive materials in liquid waste
  - Amount of radioactive materials released to the environment

(3) Exposure pathways and doses from the exposure involving intake of radioactive materials in the gaseous and liquid wastes

The main exposure pathways to the general public of radioactive materials in the gaseous waste are external exposure to radiations from a radioactive plume and radioactive materials deposited on the soil surface and internal exposure to radioactive materials taken into the body due to breadth and intakes of leafy vegetable and milk. Exposure pathways to the public opf radioactive materials in the liquid waste are external exposure to radiation emitted from radioactive materials that diffused in ocean and beach and internal exposure to radioactive materials taken in the body by intake of marine products.

In the case of the nuclear reactor facility, judging from the released quantity of radioactive nuclide, the most important exposure forms of the general public living around the facility are external exposure to gamma-ray from radioactive noble gas and internal exposure to radioactive materials taken by marine products and internal exposure by intake of radioactive iodine.

Therefore, the exposure pathways and doses that should be limited by the dose objectives are the following:

1) Calculate the effective dose due to the gamma-ray from the radioactive noble gas as the effective dose due to external exposure from gamma-ray from a radioactive cloud that diffuses and moves from the release source
2) Calculate the effective dose attributable to radioactive materials in the liquid waste as the effective dose due to internal exposure associated with the intake of marine products which include radioactive materials.
3) Calculate the effective dose attributable to the radioactive iodine contained in the gaseous waste as the effective dose due to internal exposure associated with inhalation and intakes of leafy vegetable and milk.

3.2 Siting evaluation

(1) Purpose of siting evaluation

The suitability of siting conditions of a nuclear reactor is reviewed in accordance with the "Review Guide for Nuclear Reactor Siting and Reference Criteria Concerning its Application." This guide provides that a certain distance shall be secured from a nuclear reactor so as not to cause radiological hazards to the general public in the vicinity and to limit the collective dose sufficiently small for ensuring public safety in case of a major accident or a hypothetical accident. The reference values are set forth in the guide. The above-mentioned guide requires that a certain distance to surrounding non-residential areas, low population area and densely populated area from a reactor facility is secured so that evaluated public doses are below the reference doses when "major accidents" and "hypothetical accident" are assumed. Therefore, it is required to perform an evaluation of "major accidents" and "hypothetical accidents" to judge the suitability of siting conditions of a nuclear reactor.

Events to be assumed, criteria, matters to be taken into consideration for the analysis, etc. in siting evaluation are provided in the following:

(2) Specific events for major accidents and hypothetical accidents

Specific events to be evaluated for major accidents and hypothetical accidents shall be as follows.

- Loss of reactor coolant (PWR, BWR)
- Steam generator tube break (PWR)
- Main steam line break (BWR)

(3) Specific conditions in evaluations of major accidents and hypothetical accidents, and methods to apply the criteria

(a) Loss of reactor coolant (PWR)

- For the major accident

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

(ii) 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.

(iii) The amount of fission products released 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.

(iv) 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.

(v) 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.

(vi) 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.

(vii) 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 in the "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.

(viii) 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.

(ix) 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%.

(x) 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%.

(xi) 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.

(xii) 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.

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

(xiv) 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 accidents excluding the following:

(iii) The amount of fission products released 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.

(ix) 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%.

(xi) 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.

(b) Loss of reactor coolant (BWR)

- For the major accidents

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

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

(iii) The amount of fission products released 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.

(iv) 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.

(v) 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.

(vi) 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.

(vii) 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 in the "Safety Design Evaluation."

(viii) 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.

(ix) 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%.

(x) 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.

(xi) 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.

(xii) 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.

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

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

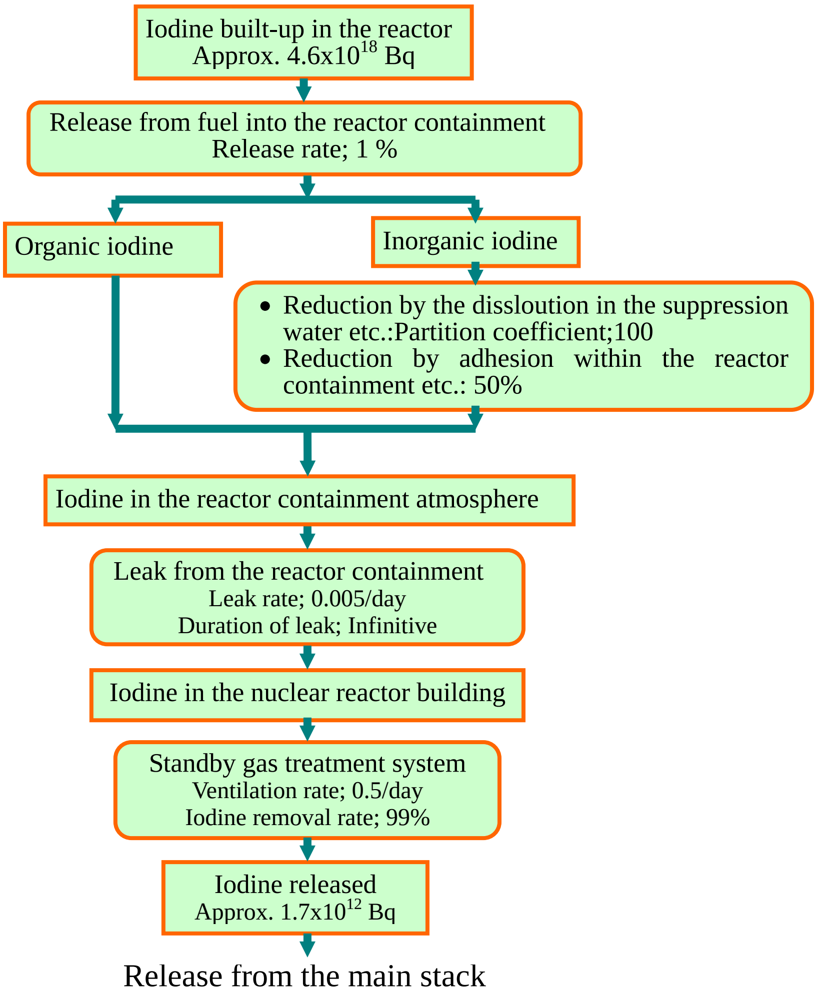


Figure 3-3 Example of iodine release process to the atmosphere during a loss of coolant accident (major accident) (amount equivalent to I-131)

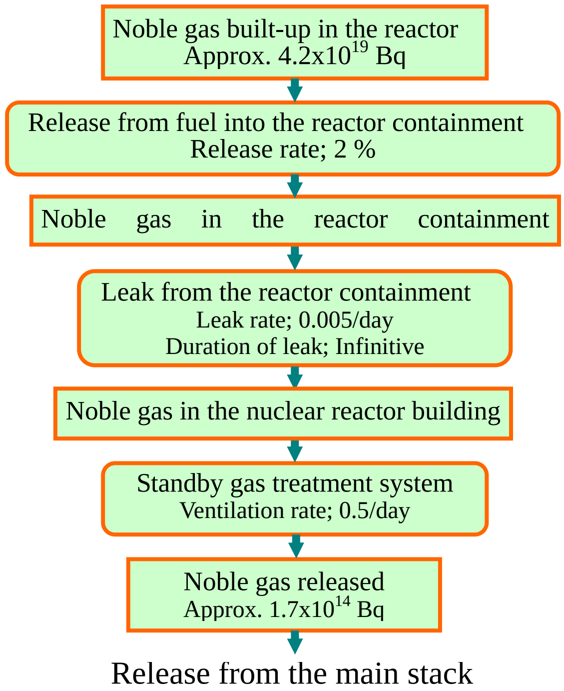


Figure 3-4 Example of noble gas release process to the atmosphere during a loss of coolant accident (major accident) (amount reduced to 0.5MeV Gamma-rays)

- For the hypothetical accident

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

(iii) The amount of fission products released 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.

(ix) 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%.

(xi) 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.

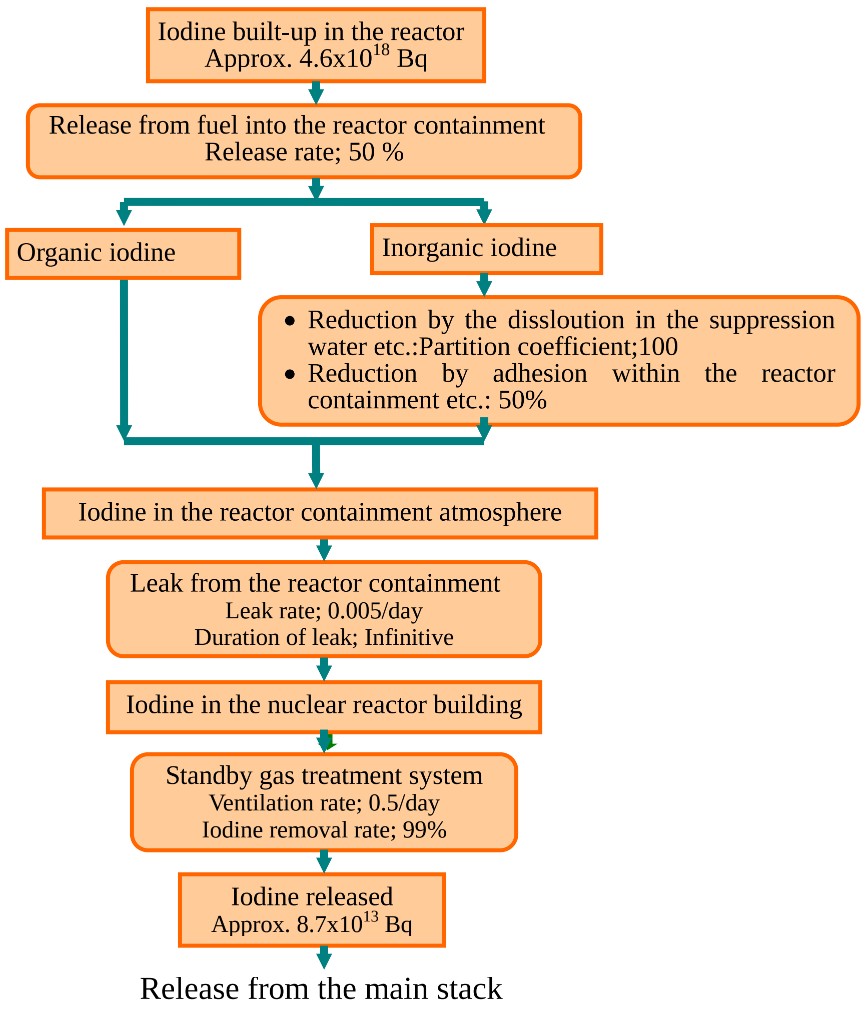


Figure 3-5 Example of iodine release process to the atmosphere during a loss of coolant accident (hypothetical accident) (amount equivalent to I-131)

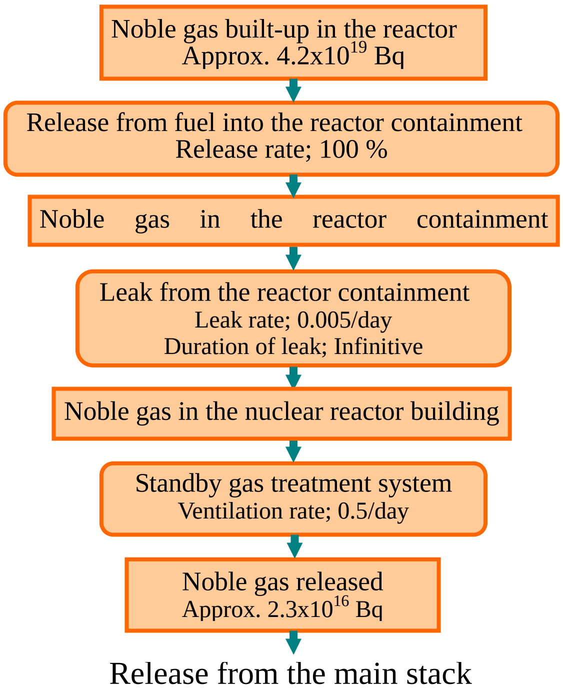


Figure 3-6 Example of noble gas release process to the atmosphere during a loss of coolant accident (hypothetical accident) (amount reduced to 0.5MeV Gamma -rays)

(c) Steam generator tube break (PWR)

- For the major accident

(i) 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.

(ii) It is assumed that the reactor has been operating at a power level with 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.

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

(iv) 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.

(v) 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.

(vi) 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.

(vii) 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.

(viii) 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.

(ix) 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.

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

(xi) 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:

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

(vii) 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.

(ix) 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.

(d) Main steam line break (BWR)

- For the major accident

(i) 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.

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

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

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

(v) 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.

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

(vii) 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.

(viii) 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.

(ix) 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.

(x) 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.

(xi) 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.

(xii) 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.

(xiii) 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.

(xiv) 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.

(xv) 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

(xvi) 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."

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

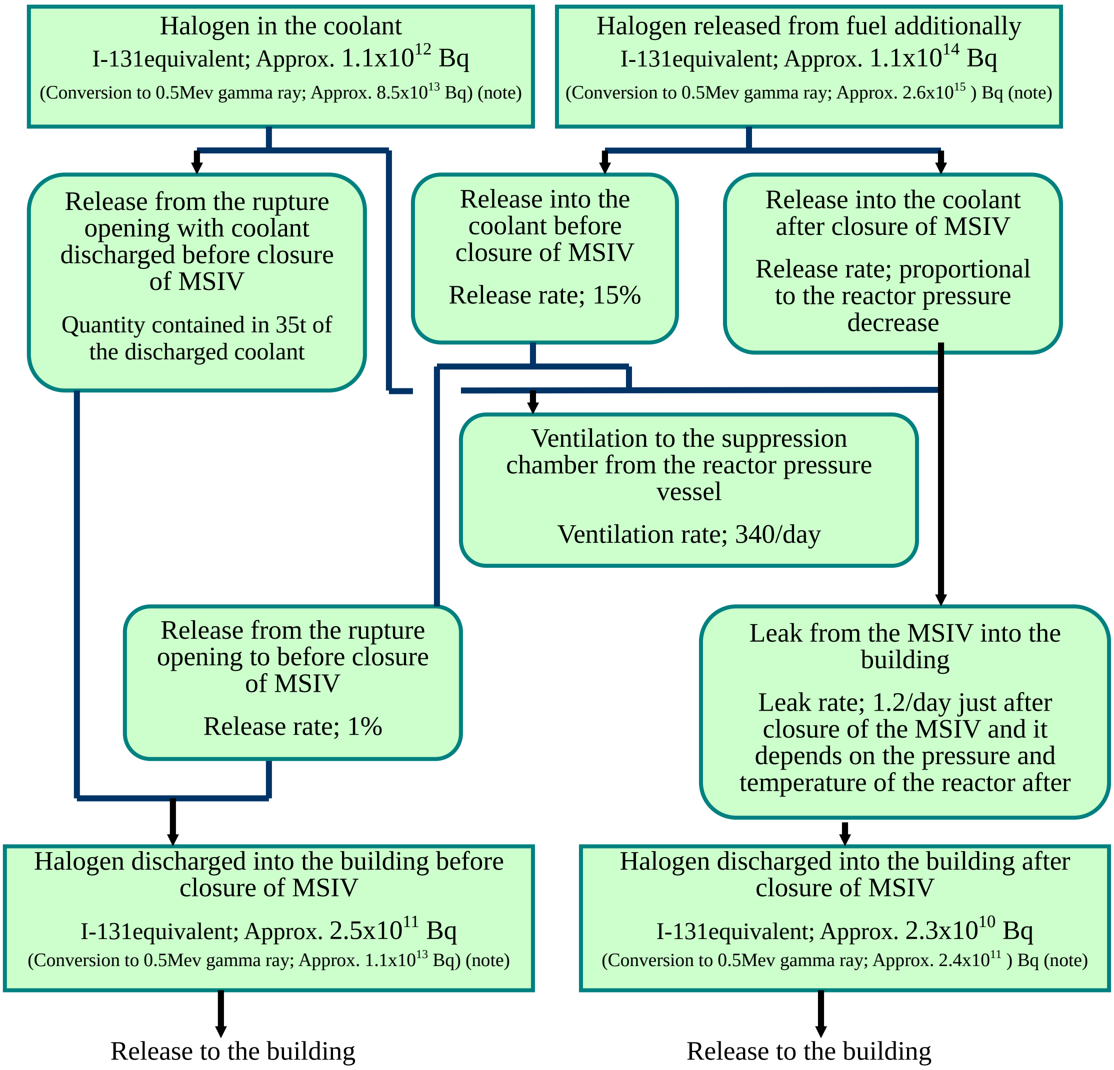
- For the hypothetical accident

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

(x) 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.

(xi) 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.

(xv) 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.The hypothetical accident shall be evaluated in the same manner as for the major accident except for the following items.



(Note) These numbers are to be used for the whole body dose evaluation together with those of noble gas.

Figure 3-7 Example of halogen release process to the atmosphere during a main steam line break accident (major accident)

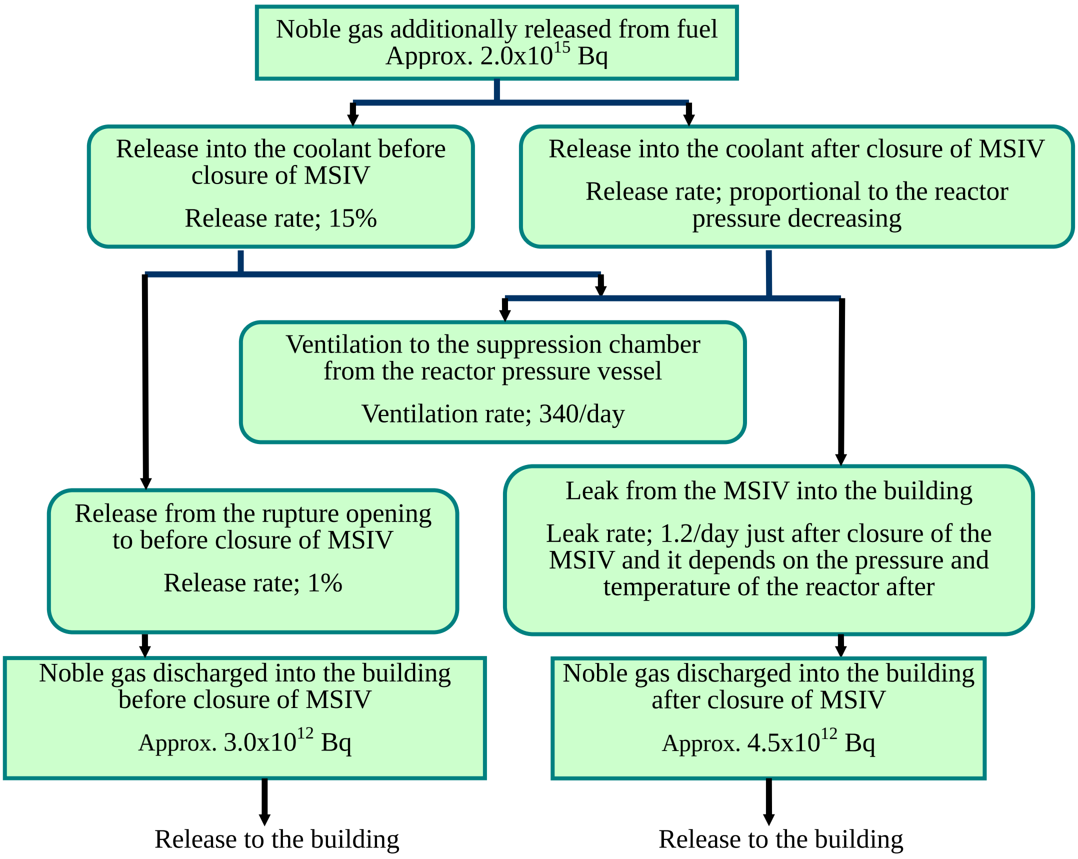
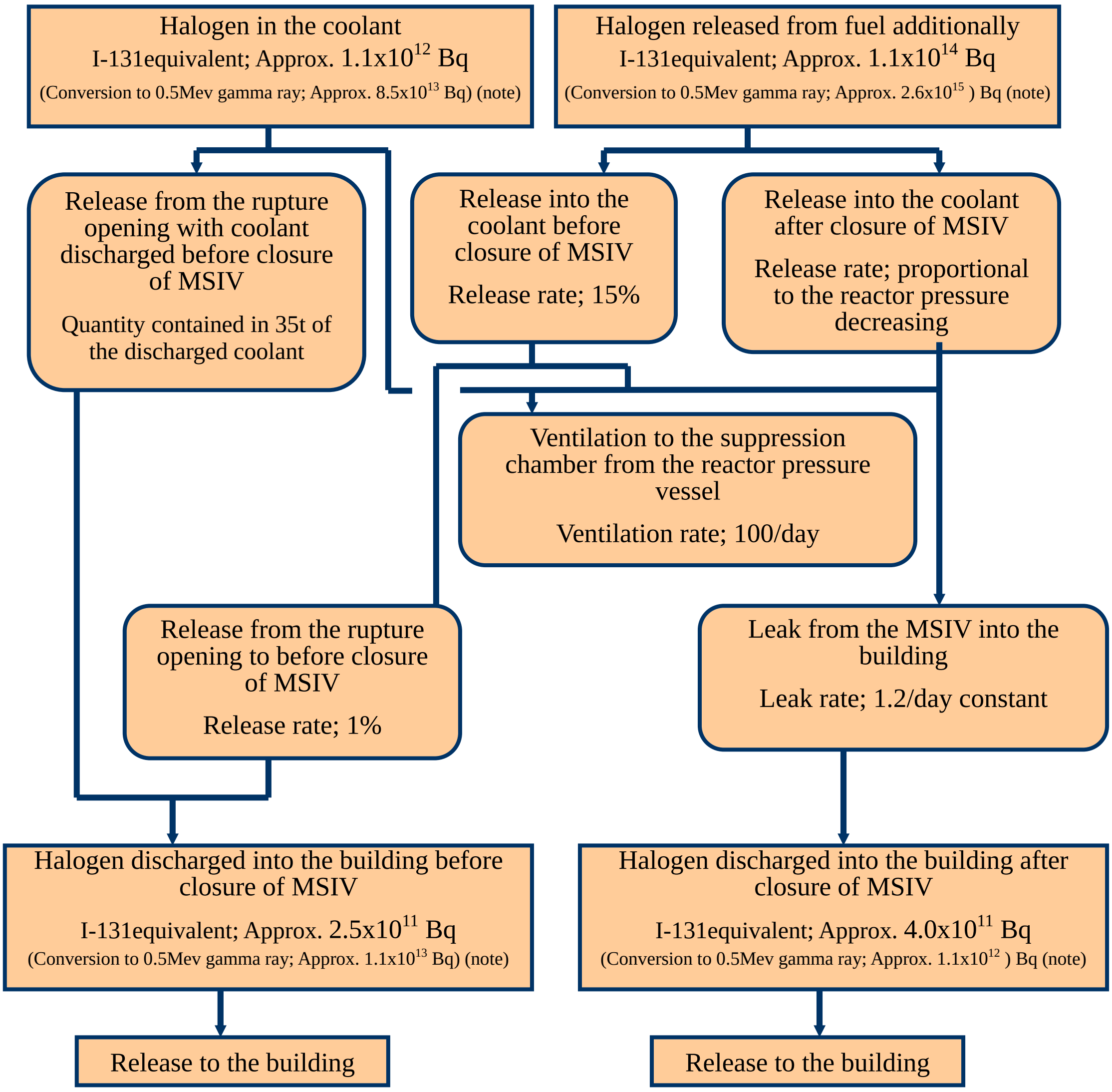


Figure 3-8 Example of noble gas release process to the atmosphere during a main steam line break accident (major accident) (amount reduced to 0.5MeV Gamma-rays)



(Note) These numbers are to be used for the whole body dose evaluation together with those of noble gas.

Figure 3-9 Example of halogen release process to the atmosphere during a main steam line break accident (hypothetical accident)

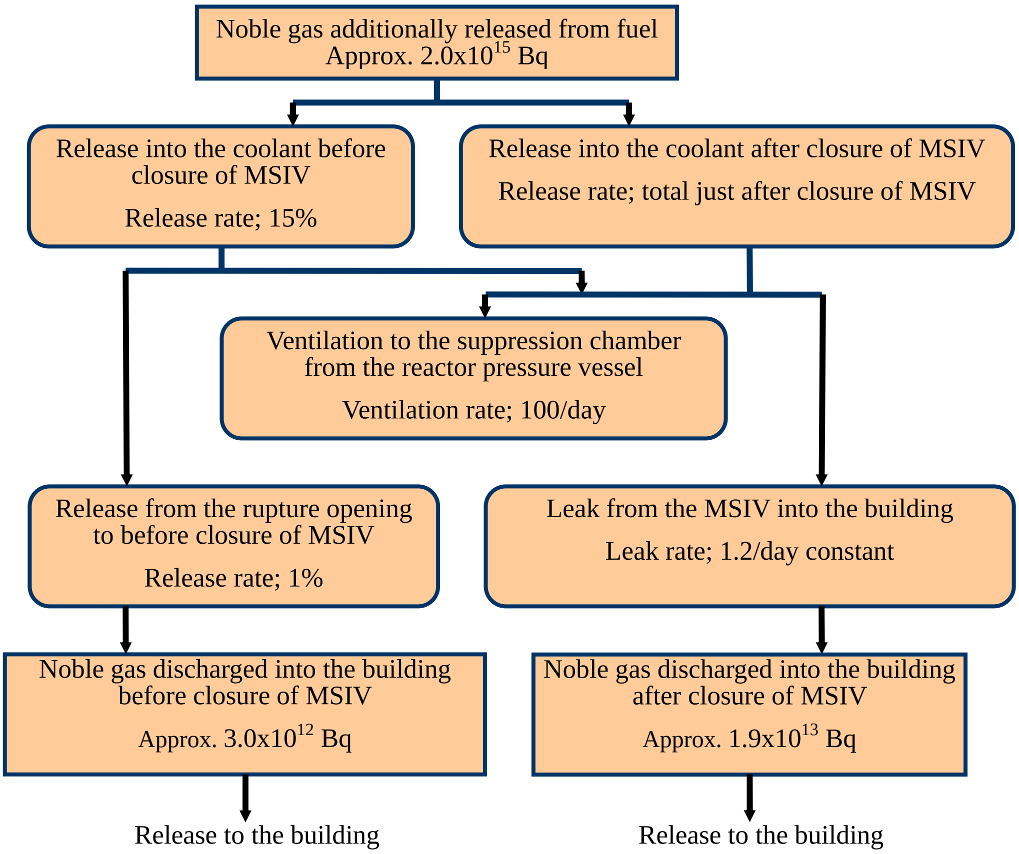


Figure 3-10 Example of noble gas release process to the atmosphere during a main steam line break accident (hypothetical accident) (amount reduced to 0.5MeV Gamma-rays)

(4) Evaluation of major accidents and hypothetical accidents

The matters to be noted on "major accidents" and the "hypothetical accidents" for the siting evaluations are provided below.

(a) Dose evaluations for "major accidents" and "hypothetical accidents"

(i) Dose evaluation to the whole body

- Dose due to radioactive material released to the atmosphere

The dose to the whole body from the γ-rays from radioactive clouds due to radioactive materials released to the atmosphere shall be evaluated based on the relative dose using the air kerma caused by radioactive materials. The conversion factor from the air kerma to the whole body shall be 1 Sv/Gy.

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

(ii) Dose due to radioactive materials in the nuclear reactor facility buildings

The dose to the whole body from the direct γ-rays and skyshine γ-rays due to radioactive materials released in the nuclear reactor facility buildings shall be evaluated appropriately taking into account locations of the facilities, shielding structures, geographical conditions, etc. The conversion factor from the air kerma to the whole body shall be 1 Sv/Gy. When it is apparent that the direct dose and skyshine dose do not make a significant contribution to the dose to the whole body in the case of the accident concerned, their evaluations may be omitted.

(b) Dose evaluation to the thyroid gland

The dose to the thyroid gland due to inhalation of iodine released to the atmosphere shall be evaluated in accordance with the "Meteorological Guide" using the following formula based on the relative concentration of iodine in the air of the ground surface and the amount of I-131 equivalent released. In addition, the parameters to be used for the calculation shall be the values for infants (1 year old) in the case of the "major accidents" and the values for adults in the case of the "hypothetical accidents" shown in Table-3.

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

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

$$Q_e = \sum_i (K_{Ti}/K_{Te}) \cdot Q_i$$

K_{Ti}: dose coefficient pertaining to the equivalent dose to the thyroid gland due to inhalation of nuclide i

Q_i: amount of nuclide i released

(c) Evaluation of cumulative dose to the whole body

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

Table-3 Parameters etc. to be used to evaluate the dose to the thyroid gland due to iodine

Parameter etc.	Symbol	Unit	Value	
Dose coefficient pertaining to the equivalent dose to the thyroid gland by inhalation of nuclide i	K_{Ti}	S_v/B_q	Infant	I-131: 3.2×10^{-6} I-132: 3.8×10^{-8} I-133: 8.0×10^{-7} I-134: 7.3×10^{-9} I-135: 1.6×10^{-7}
			Adult	I-131: 3.9×10^{-7} I-132: 3.6×10^{-9} I-133: 7.6×10^{-8} I-134: 7.0×10^{-10} I-135: 1.5×10^{-8}
Respiratory rate	M	m^3/h	0.31 (for infants when active) 1.2 (for adults when active	
		m^3/d	5.16 (day average for infants) 22.2 (day average for adults)	

(5) Study of the adaptability to the criteria

Performing dose evaluations of major accidents in relation to various kinds of engineered safety features, the doses outside of the site boundary are confirmed to be sufficiently below the doses (1.5Sv to the thyroid gland (infant) and 0.25Sv to the whole body).

Performing dose evaluations of hypothetical accidents in relation to various kinds of engineered safety features, the doses outside of the site boundary are confirmed to be sufficiently below the reference criteria as reference (3Sv to the thyroid gland (adult) and 0.25Sv to the whole body).

Furthermore, the cumulative dose to the whole-body is confirmed to be sufficiently below 20,000 man-Sv shown in the criteria.

Thereby, a nuclear facility to be installed in the site accommodates the siting conditions provided in the "Review Guide for Nuclear Reactor Siting."

References

1. “Safety Review of Nuclear power Plant” from the brochure of the Incorporated Agency, Nuclear Power Engineering Corporation

2. Following guides shown in the Review Guide for Safety Review etc. on the website of the Nuclear and Industrial Safety Agency

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

   Provisional Guideline of Application of Review Guide for Nuclear Reactor Siting

   Review Guide for Nuclear Reactor Siting and Reference Criteria Concernking its Application

   Evaluation of Dose Equivalents for the General Public at the Safety Review of Light Water Nuclear Power Reactor Facilities

   Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities

   Evaluation Guide for Dose Objectives Around Light Water Nuclear Power Reactor Facilities

   Meteorological Guide for Safety Analysis of Nuclear Power Reactor Facilities

3. Application for establishment permit of Kashiwazaki Kariwa Nuclear Power Station, Tokyo Electric Power Co., Inc.

Reference sheet 1: Summary of the Meteorological Guide for Safety Analysis of Nuclear Power Reactor Facilities

I. Purpose

This guide defines the meteorological observation methods, statistical processing methods of observed values, and the analytical methods of atmospheric diffusion that are required to estimate the diffusion condition of the radioactive materials in the atmosphere on an occasion of dose evaluation in cases of the normal operation and postulated accidents (major accidents and hypothetical accident) of nuclear power reactor facilities.

II. Meteorological observation method

Category

Routine observation: It is continuously performed from prior to establishment of a nuclear reactor facility to after its decommissioning to obtain the weather information directly related to dose evaluations.

Special observation: It is performed for a specific period of time in order to obtain the weather information on meteorological characteristics at a site and in its vicinity for safety analysis before establishment of a nuclear reactor facility.

Observation items

Observation items of routine observations: Wind direction, wind speed, amount of insolation, amount of radiation balance

Observation items of special observations: Wind direction, wind speed, winds at higher altitudes, temperature difference

Observation method

Ratio of missing measurements (including atmospheric stability) shall be, in principle, 10% or less for 12 continuous months.

Observation period

Routine observation: It shall start at least one year prior to application for approval of establishment permit of a nuclear reactor facility, and performed continuously until its decommissioning.

Special observation: Wind direction and wind speed are continuously observed for at least one year prior to application for establishment permit of a nuclear reactor facility, and winds at higher altitudes and temperature differences are observed at the appropriate time of this period.

III. Statistical processing method of observed values

(1) Hourly weather information

The hourly weather information specified below is used as the statistical base:

(a) Wind direction, wind speed, amount of insolation, and amount of radiation balance

For wind direction, wind speed, amount of insolation, and amount of radiation balance, the average value of each observed value for 10 minutes before each hour is treated as the observed value at the hour concerned.

(b) Atmospheric stability

Atmospheric stability is classified based on wind speed, amount of insolation and amount of radiation balance at the time concerned of "ground wind representing a site", and this result is used as the atmospheric stability at the time concerned.

(c) When any meteorological element of wind direction, wind speed, or atmospheric stability is missing, the weather information at the time concerned is treated as missing data.

The statistics obtained from the observed information excluding missing data shall be treated as representative data for that year.

(2) Statistical processing of weather information

(a) During normal operation

Hourly weather information is statistically processed on the following items:

(1) Total summation of inverse number of wind speed classified according to the wind direction and atmospheric stability

(2) Average for inverse number of wind speed classified according to wind direction and atmospheric stability

(3) Average for inverse number of wind speed classified according to wind direction

(4) Frequency of wind direction

(5) Frequency of wind direction with wind speed from 0.5 to 2.0m/s

In statistically processing, (1), (2) and (3) mentioned above, the observational data at windy conditions (wind speed 0.5 m/s or more) are used as they are, but at calm conditions (wind speed less than 0.5m/s), the observational data are distributed proportionally according to 0.5 m/s for wind speed and frequency of wind direction with wind speed from 0.5 to 2.0m/s for wind direction.

(b) For postulated accidents

Hourly weather information is arranged for each hour on wind direction, wind speed, and atmospheric stability.

IV. Basic diffusion equation

The concentrations of radioactive materials in air during normal operation and postulated accidents are calculated based on a diffusion equation which assumes that the special concentration distribution of radioactive materials is a normal distribution both in the horizontal direction and the vertical direction on the condition that all weather conditions such as wind direction and wind speed are uniformly stable, radioactive materials are regularly released from the release source, and the topography is flat. In this case, the coordinates of the diffusion equation are rectangular coordinates with the ground surface directly under the release source as the original point, leeward direction as x axis, the direction at right angles to x-axis as y-axis and the vertical direction as z-axis.

V. Analytical method of atmospheric diffusion during normal operation

(1) Concentrations on the ground surface to be used for dose calculations

Calculation of the concentrations in air on the ground surface to be used for the dose calculation during normal operation

However, when the concentrations in air on the ground surface need to be corrected due to the wind tunnel test results etc., appropriate corrections should be made.

(2) Calculation of yearly average concentrations

(a) In calculations of yearly average concentrations of radioactive materials, the contributions of the wind going to the direction including a location of interest (direction of interest) from a release point and the wind going to the adjacent directions should be added.

(b) In calculations of yearly average concentrations in the direction of interest, total summation of inverse numbers of wind speed classified according to the wind direction and atmospheric stability is to be used in the case of continuation release.

In the case of intermittent release, based on frequency of the wind direction to the direction of interest and the adjacent directions (sum of the frequencies in three directions) and the numbers of yearly release, the total number of the release going to the three directions is calculated using 67% as the reliability of the binomial probability distribution, and this result is distributed proportionally to the frequency of wind in three directions.

And, for wind speeds, the average of inverse numbers of wind speed classified according to the wind direction and atmospheric stability is to be used.

However, when the release frequency is large and the release duration is long, the number of releases to each direction should be proportional to the frequency of that wind direction.

(c) In calculations of yearly average concentrations in the direction of interest, the concentration should be averaged assuming that the wind direction varies uniformly within one direction.

VI. Analytical method of atmospheric diffusion during postulated accidents

The concentrations in air on the ground surface to be used for the dose calculation during postulated accidents are derived by multiplying the downwind concentrations per unit release rate (defined as relative concentration) by the release rate of radioactive materials during the accident period.

(1) Relative concentration to be used for dose calculation

(a) The relative concentration is calculated of a location of interest in each direction based on the hourly weather information and effective release duration (defined with account taken of temporal change in a release rate of radioactive materials, hereinafter referred to as effective release duration).

(b) The relative concentration at a location of interest should be the relative concentration with the accumulative frequency 97% when hourly relative concentrations are accumulated in order of smaller ones over one year.

(c) The relative concentration to be used for the dose calculation should be the maximum value among the relative concentrations obtained in the above (b).

(2) Calculation of relative concentration

(a) Short-time release

(b) Prolonged release

VII. Effective height of release point

The effective height of release point should be determined comprehensively taking into consideration the ground height of a stack, the spouting height of the stack, effects of buildings and topography, etc.

VIII. Wind tunnel test

When the topography of a site is complex, or effects of buildings etc. on the release point are expected to be significant, in order to study the adequacies of an effective height etc. of the release point, wind tunnel tests are to be conducted using models simulating each geometric condition.

Reference sheet 2: Summary of Dose Evaluation to be performed in the Safety Evaluation

(1) Dose evaluation to be performed in the safety evaluation

In order to judge whether the design of a nuclear power reactor is adequate or not from a viewpoint of ensuring safety, the safety evaluation is performed in accordance with the "Review Guide for Safety Evaluation (Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities)." In the safety evaluation, it is confirmed that the structures, components etc. of a nuclear reactor facility perform required functions not only under normal operation, but also under abnormal conditions to ensure safety. These abnormal conditions are analyzed and evaluated dividing them into anticipated operational occurrences and accidents.

In the course of these analysis and evaluation, the dose evaluation is performed to evaluate that "there is no significant risk of radiation exposure to the surrounding public." which is provided in the "Review Guide for Safety Evaluation" as one of criteria to be confirmed for accidents.

(a) Specific "accident" events for dose evaluation

In accordance with the "Review Guide for Safety Evaluation," events that could cause a significant impact on the vicinity of a nuclear reactor facility site due to radioactive materials released from the facility are classified into the "Loss of reactor coolant or considerable change in core cooling ", "abnormal reactivity insertion or a rapid change in reactor power", "abnormal release of radioactive materials to the environment", "abnormal change in pressure, atmosphere, etc. in the reactor containment" and other events necessary for evaluation depending or the design of the nuclear reactor facility.

From these events, the events that cause an "abnormal release of radioactive materials to the

Specific 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. Specific events which are performed in the accident analysis are as follows;

- Failure of a radioactive gaseous waste processing facility (PWR, BWR)
- Main steam line break (BWR)
- Steam generator tube break (PWR)
- Drop of a fuel assembly (PWR, BWR)
- Loss of the reactor coolant (PWR, BWR)
- Control rod ejection (PWR)

- Control rod drop (BWR)

(b) Evaluation

(i) Evaluation method

The effective dose is calculated using data of relative concentration obtained from the meteorological conditions around the nuclear power reactor facilities and relative dose, as well as a release of fission products. Moreover, for a loss of the reactor coolant and a control rod ejection for PWRs, the effective dose resulting from the direct rays and skyshine rays are also added together.

(ii) Evaluation conditions

For the evaluation, the initial operation conditions leading to the severest result in light of the criteria are selected.

(iii) Dose evaluation for "accidents"

a. Evaluation of effective dose due to external exposure

・ Effective dose due to radioactive materials released to the atmosphere

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

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

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

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

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

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

b. Evaluation of effective dose due to internal exposure

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

Effective dose = KHe x M x Qe - x (χ / Q)

KHe: effective dose coefficient of infants by intake of I-131 by inhalation
M: respiratory coefficient of infants
Qe: amount of iodine released (amount of I-131 equivalent)
(χ / Q) : relative concentration

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

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

$$Qe = \sum(K_{Hi}/K_{HE}) x Qi$$

KHi: effective dose coefficient of infants by intake of nuclide i by inhalation

Qi: amount of nuclide i released

(b) Criteria

According to the "Safety Evaluation Guide," the criteria for events with abnormal release of radioactive materials into the environment are as follows:

"There is no significant risk of radiation exposure to the surrounding public."

If the effective dose to the public living surrounding the site does not exceed 5 mSv per accident, the risk is judged small.

(2) Events to be evaluated

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

(i) Assumption

There are various kinds of radioactive materials in a nuclear reactor facility, but gaseous radioactive waste has a comparatively high possibility of release to the environment among them. Depending on damage etc. to a radioactive gaseous waste processing facility, the gaseous radioactive materials in storage or in process there could be released to the environment. Namely, 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.

(ii) Conditions

a 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) .

b 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 should be assumed. In addition, the damaged part is assumed so as to result in the most severe consequences in light of the criteria with account taken of the amount of the stored radioactive material and isolation time.

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

d When the damaged part is located indoors, the air ventilation system etc. of the auxiliary-building or turbine building is assumed to be in such an operating condition so as to result in the most severe consequences.

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

(iii) Release path

Examples of fission products release paths of PWR and BWR are shown in Figures 1 and 3, respectively. The damaged parts are assumed to be a gas surge-tank outlet and downstream of the isolation valve for PWR and BWR, respectively.

(iv) Release process

Examples of noble gas release processes to the atmosphere of PWR and BWR are shown in Figures 2 and 4, respectively.

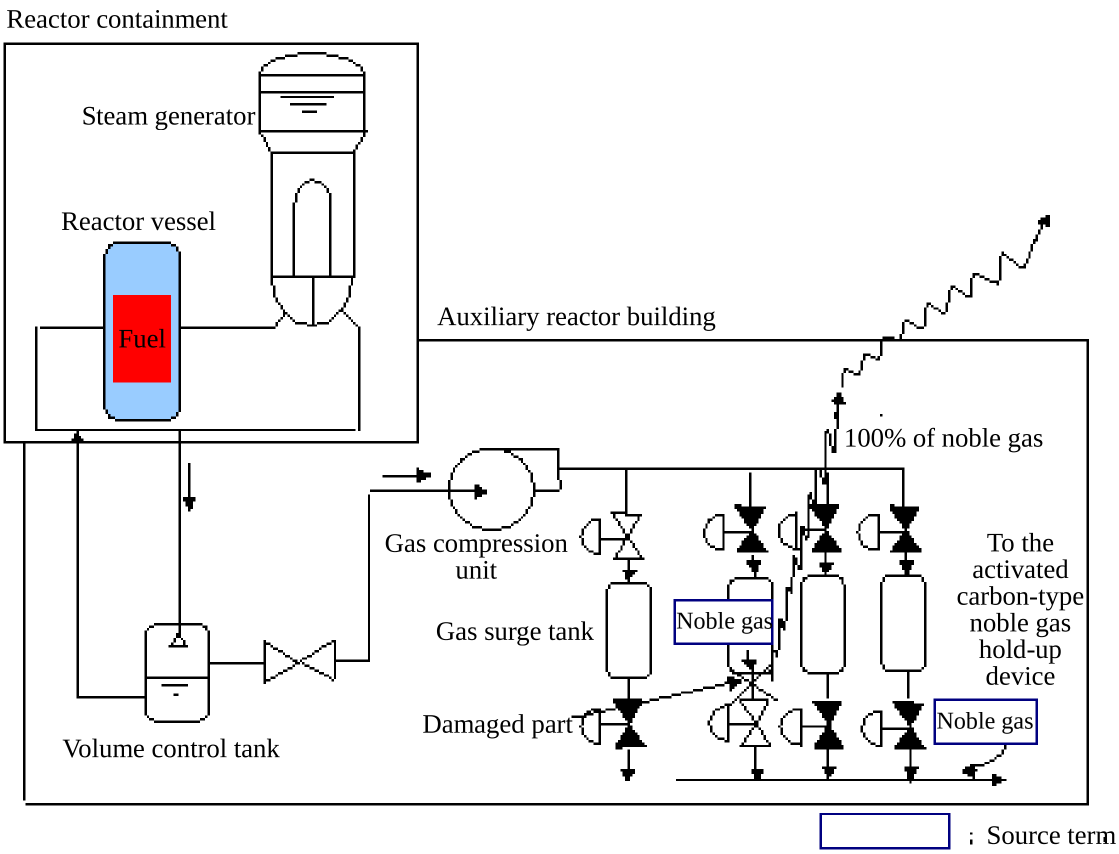


Figure 1 Example of a release path of fission products during a failure of radioactive gaseous waste processing device (PWR)

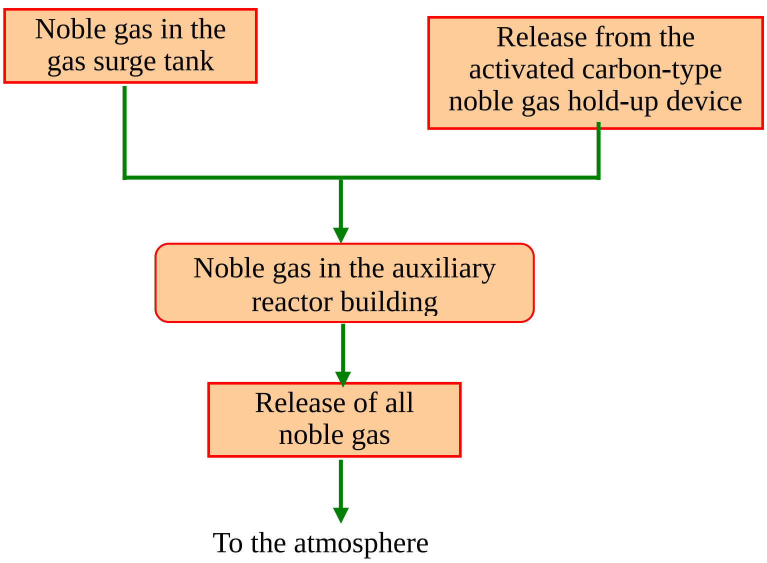


Figure 2 Example of a release process of noble gas to the atmosphere during a failure of radioactive gaseous waste processing device (PWR)

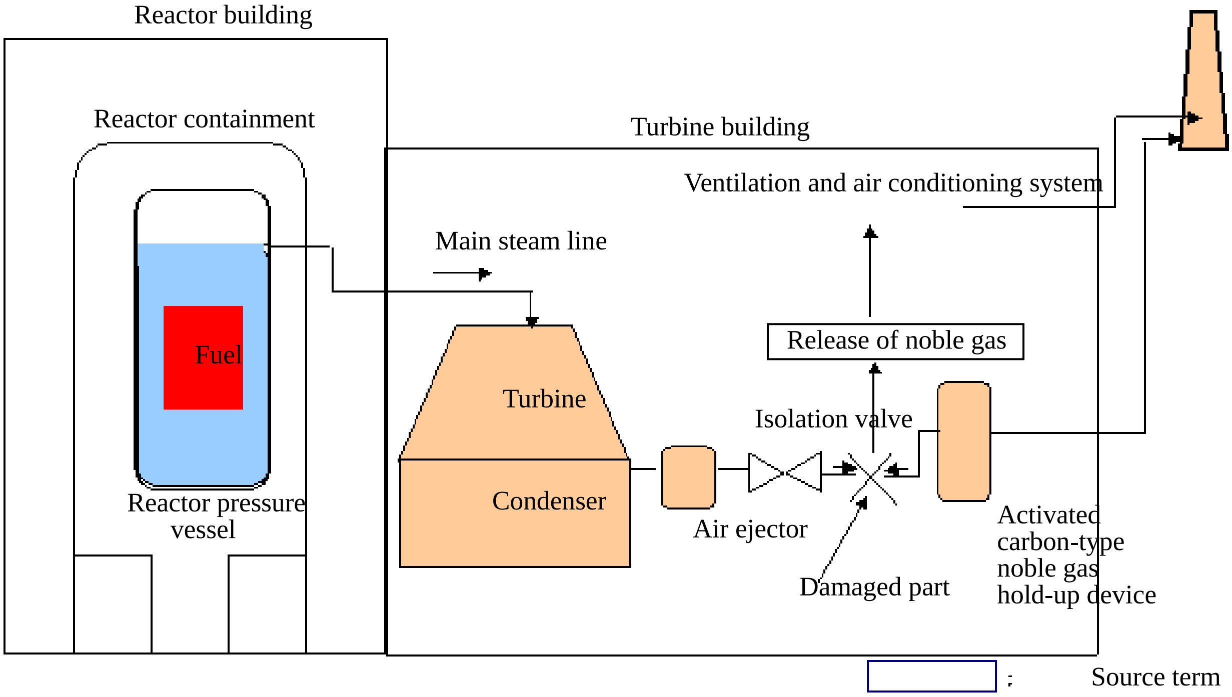


Figure 3 Example of a release path of fission products during the failure of radioactive gaseous waste processing device (BWR)

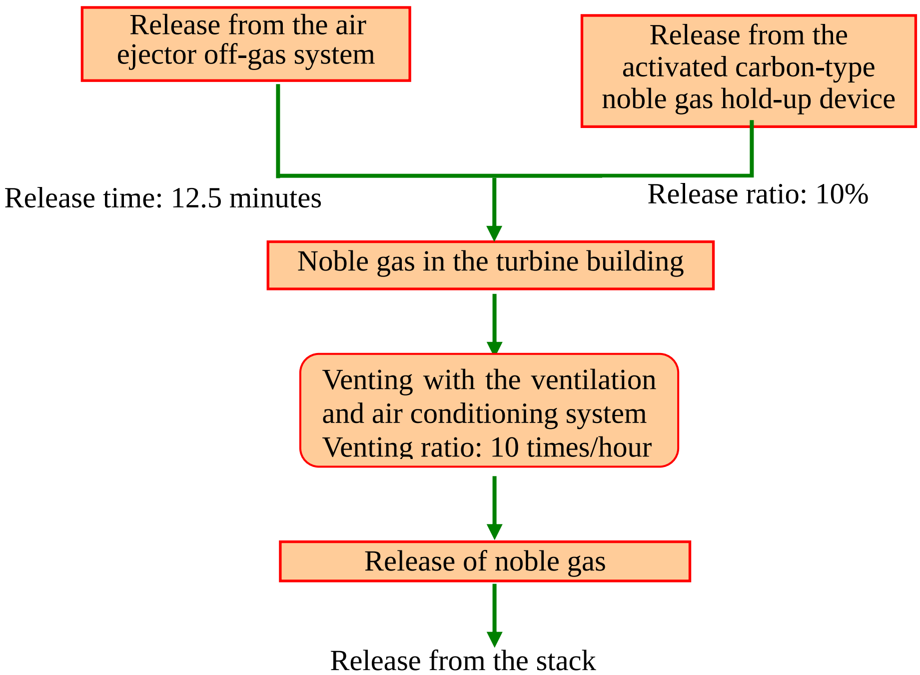


Figure 4 Example of release processes of noble gas to the atmosphere during the failure of radioactive gaseous waste processing device (BWR)

(b) Main steam line break (BWR)

(i) Assumption

In BWR, part of the reactor coolant changes to steam, and then the steam flows to the turbine outside the reactor containment through main steam lines. When the steam is released due to a break etc. of the main steam line outside the reactor containment, the radioactive materials in the steam will be directly released with the steam to the outside the reactor containment. Then, an event of the release of radioactive material to the environment due to a main steam pipe break outside the reactor containment vessel and discharge of the reactor coolant from the break opening shall be assumed during reactor power operation.

(ii) Conditions

a The reactor is assumed to be in operation at a power level with account taken of a margin for safety analysis to the rated power for a sufficiently long period of time.

b An instantaneous double-ended break of one main steam line outside of the reactor containment is assumed.

c The main steam isolation valves are assumed to close fully with the longest design operation delay time and closing time.

d 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 valves, the flow limiting effect of the valves shall not be taken into consideration until the critical flow is generated at the valves.

e A loss of offsite power supply is assumed simultaneously with the occurrence of the event.

f 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.

g The amount of additional release from the fuel rod following a decrease in the reactor pressure 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 a decreasing rate of the reactor pressure. The time required for the fission products released from the fuel rods to reach the main steam isolation valves before their closure during the course of the event may be taken into consideration for the evaluation.

h The organic iodine is assumed to be 4% of the released iodine from the fuel rod during the course of the event, and the remaining 96% is assumed to be inorganic iodine. It is assumed that 10% of the organic iodine transfers instantaneously to the gas phase and the rest of it decomposes. 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.

i 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.

j One of the main steam isolation valves shall be assumed not to close. In addition, it shall be assumed that the closed isolation valves have the leakage determined by the design leak rate, temperature and pressure.

k 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.

l After closure of the main steam isolation valves, the reactor pressure is assumed to decrease to the atmospheric pressure linearly in either time for the reactor pressure to decrease to the atmospheric pressure by the reactor core isolation cooling system etc. or 24 hours, whichever is longer.

m 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."

n It is confirmed as a criterion that any additional fuel rod failure would not occur,.

(iii) Release path

An example of a release path of fission products is shown in Figure 5.

An example of a release process of iodine and noble gas to the atmosphere is shown in Figure 6.

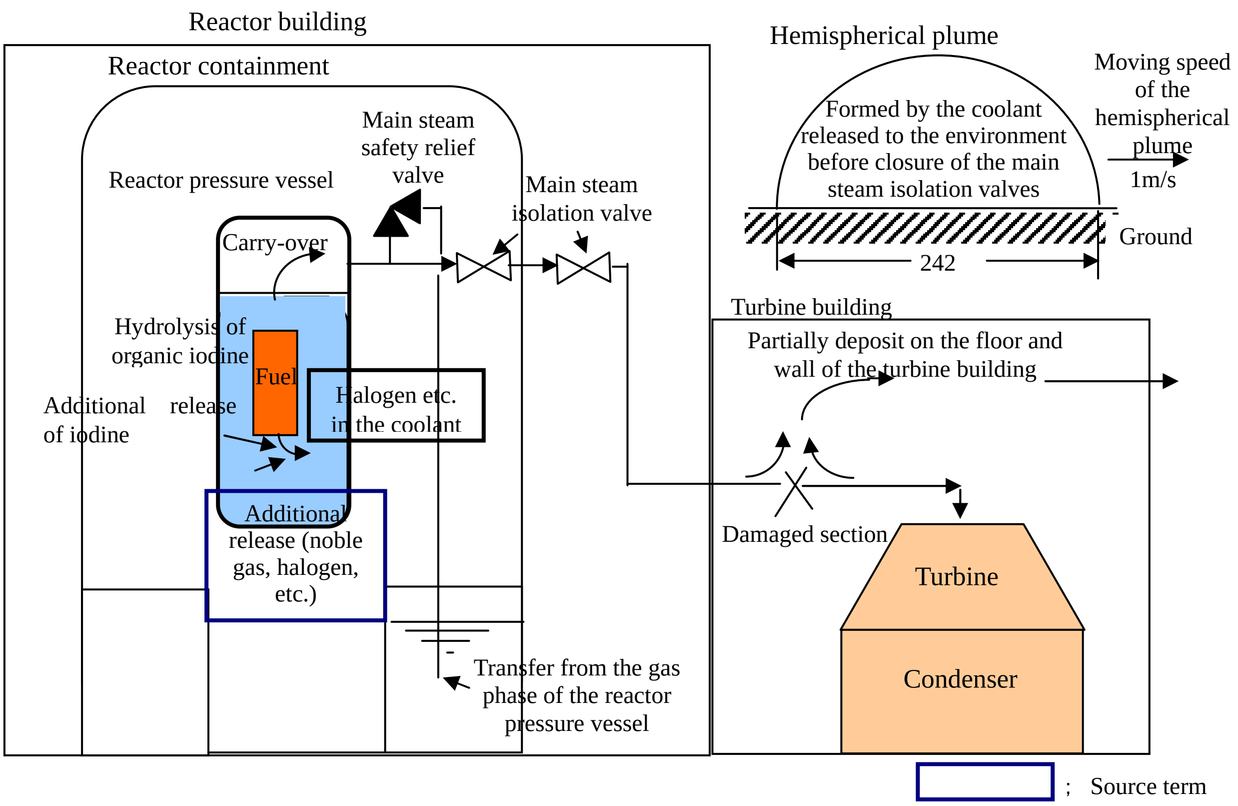


Figure 5 Example of release paths of fission products during a main steam line break (accident) (BWR)

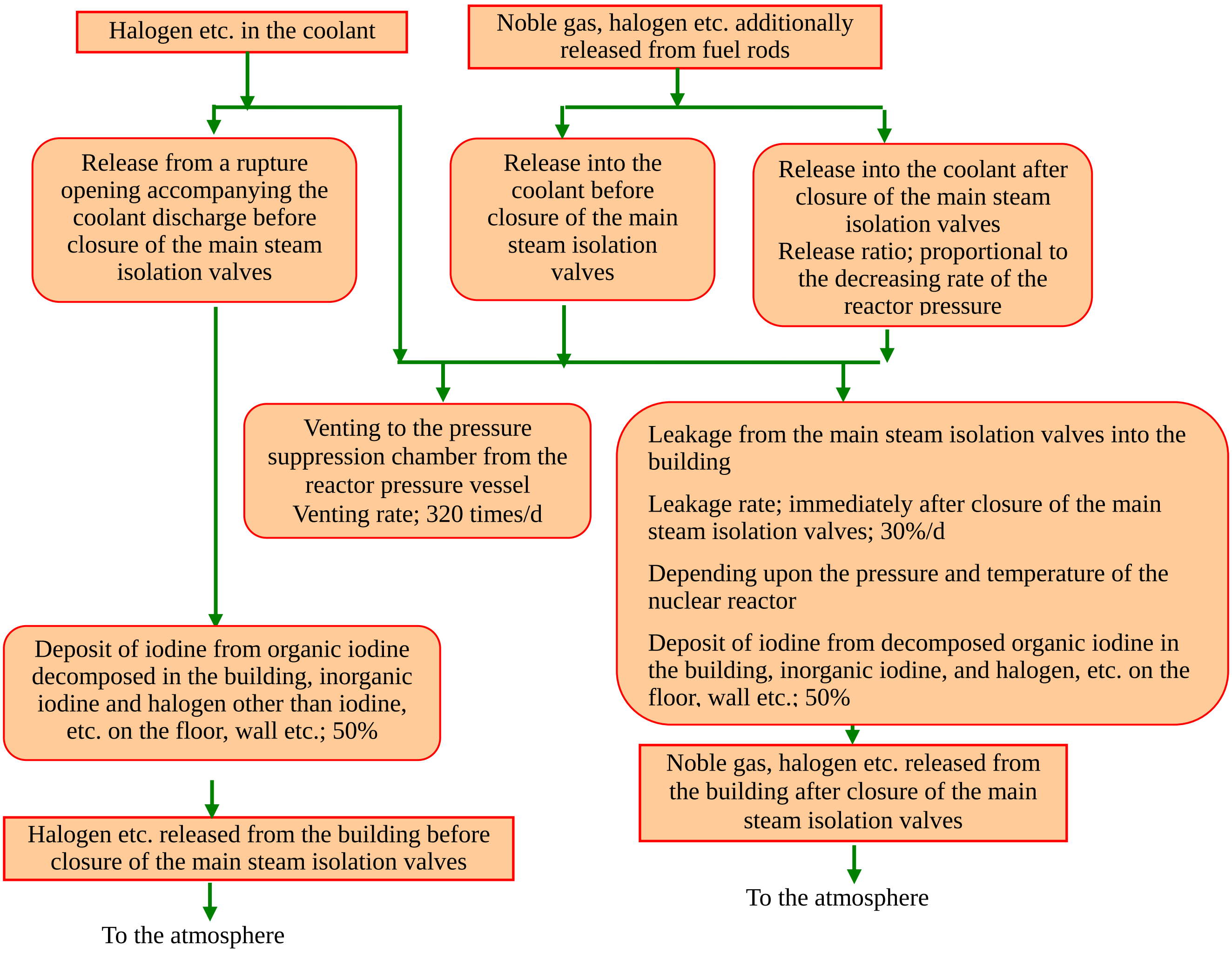


Figure 6 Example of release processes of noble gas and halogen to the atmosphere during a main steam line break (accident) (BWR)

(c) Steam generator tube break (PWR)

(i) Assumption

When a heat transfer tube of a steam generator fails, the primary coolant flows into the secondary cooling system and is released to the outside of the reactor containment through the main steam relief valves of the secondary cooling system etc., and the radioactive materials in the primary coolant could be released due to this event to the environment. Then, an event is assumed that the primary coolant is released outside the reactor containment through the secondary cooling system due to damage to a heat transfer tube of the steam generator during reactor power operation.

(ii) Conditions

a It is assumed that the reactor is in operation at a power level with account taken of a margin for safety analysis to the rated power for a sufficiently long period of time and the reactor pressure is the highest one during normal operation.

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

c Cases with and without offsite power available are to be considered. Moreover, when the ECCS starts automatically, its operation shall be assumed to result in the larger discharge rate of the primary coolant.

d 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.

e 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.

f 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.

g When isolation of the damaged steam generator requires operator actions, 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 in either time for the reactor pressure to decrease to the atmospheric pressure by the operable cooling system or 24 hours, whichever is longer, and the isolated valves have the leakage determined by the design leakage rate, temperature and pressure.

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

i It shall be confirmed as a criterion that an additional fuel rod failure would not occur.

(iii) Release path

An example of release paths of fission products is shown in Figure 7.

(iv) Release process

An example of release process of iodine and noble gas to the atmosphere is shown in Figure 8.

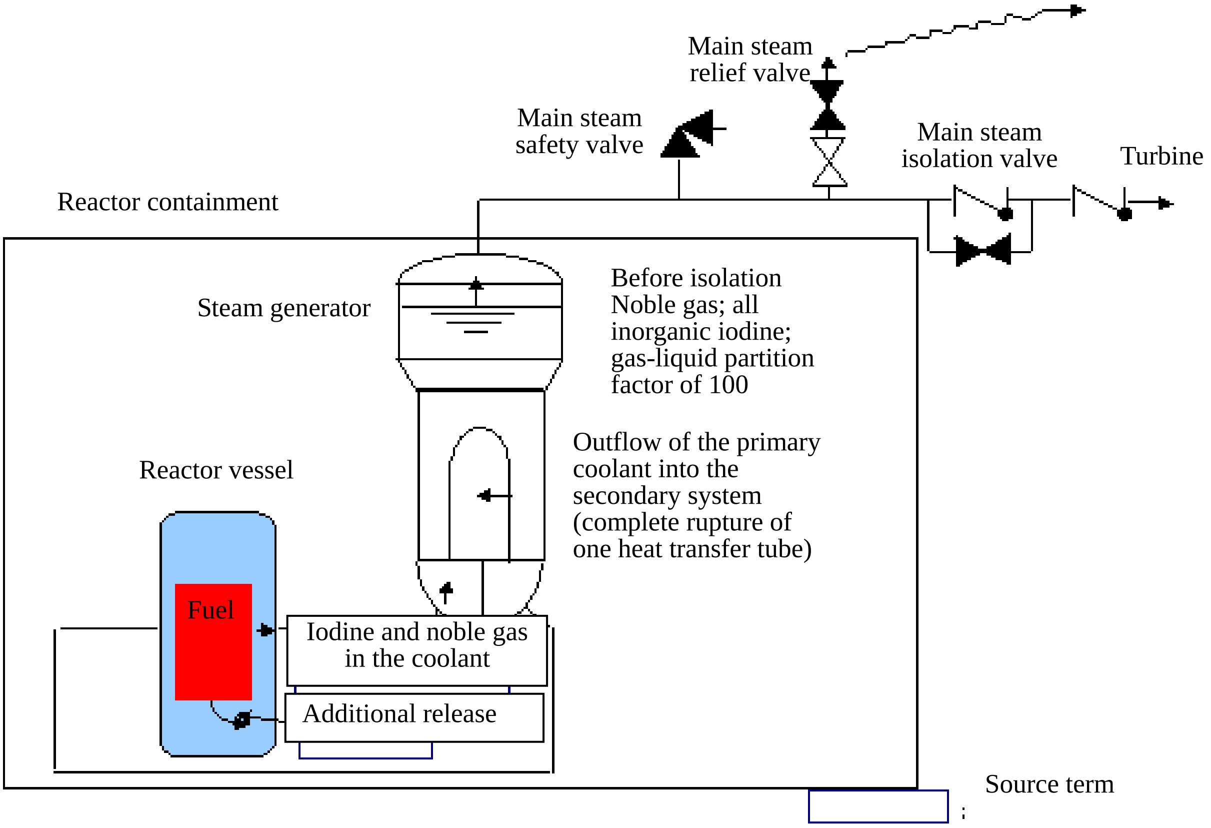


Figure 7 Example of release paths of fission products during a steam generator tube break (accident) (PWR)

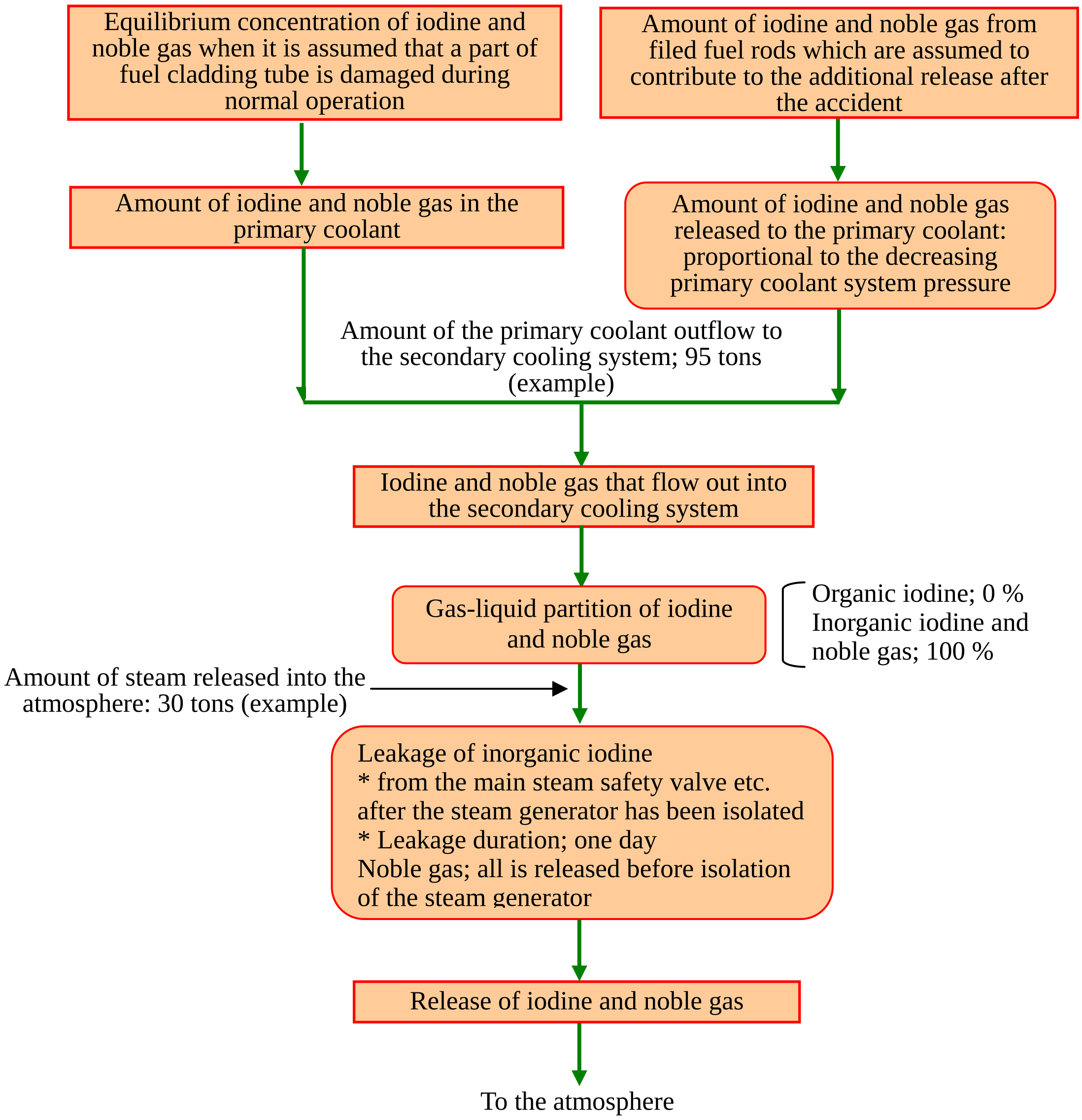


Figure 8 Example of release processes of iodine and noble gas to the atmosphere during a steam generator tube break (accident) (PWR)

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

(i) Assumption

One of the locations other than a reactor core in a nuclear reactor facility where a not-negligible amount of radioactive materials exists is a spent fuel handling facility. An event is assumed that a fuel assembly drops for some reason and fails during reactor refueling, resulting in a release of radioactive material to the environment.

(ii) Conditions

a For PWR, it is assumed that one fuel assembly being handled within the spent fuel pit drops from the highest position during the handling.

b For BWR, it is assumed that one fuel assembly being handled above the reactor core drops from the highest position during the handling into the core.

c 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 account taken of a margin for safety analysis to the rated power level 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.

d The number of failed fuel rods due to the drop shall be the maximum number of the fuel rods in that assembly unless there is any experimental basis.

e 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.

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

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

(iii) Release path

An example of a release path of fission products is shown in Figure 9 and Figure 11 for PWR and BWR, respectively.

(iv) Release process

An example of release processes of iodine and noble gas to the atmosphere is shown in Figure 3-10 and Figure 3-12 for PWR and BWR, respectively.

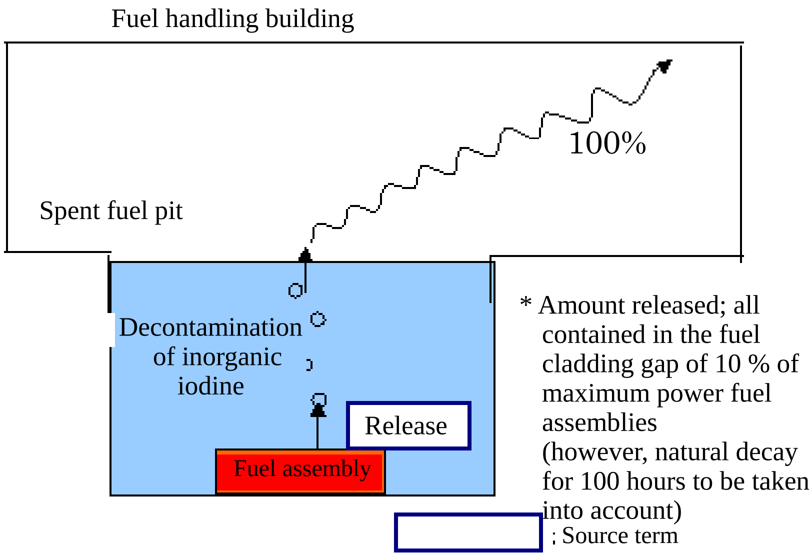


Figure 9 Example of release paths during a drop of fuel assembly (PWR)

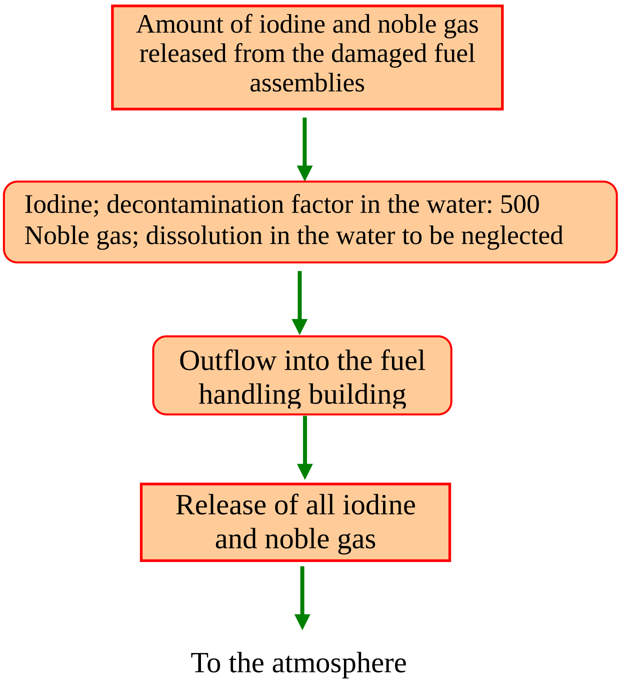


Figure 10 Example of release processes of iodine and noble gas during a drop of fuel assembly (PWR)

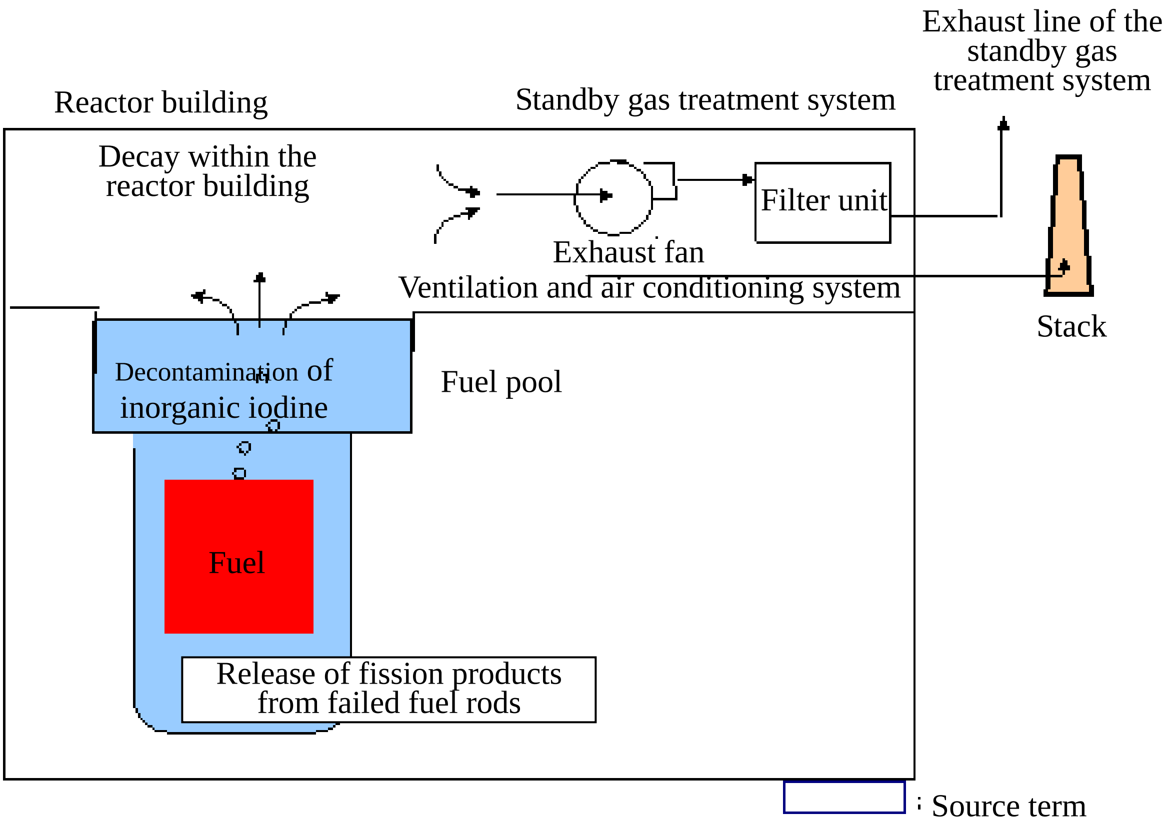


Figure 11 Example of release paths during a drop of fuel assembly (BWR)

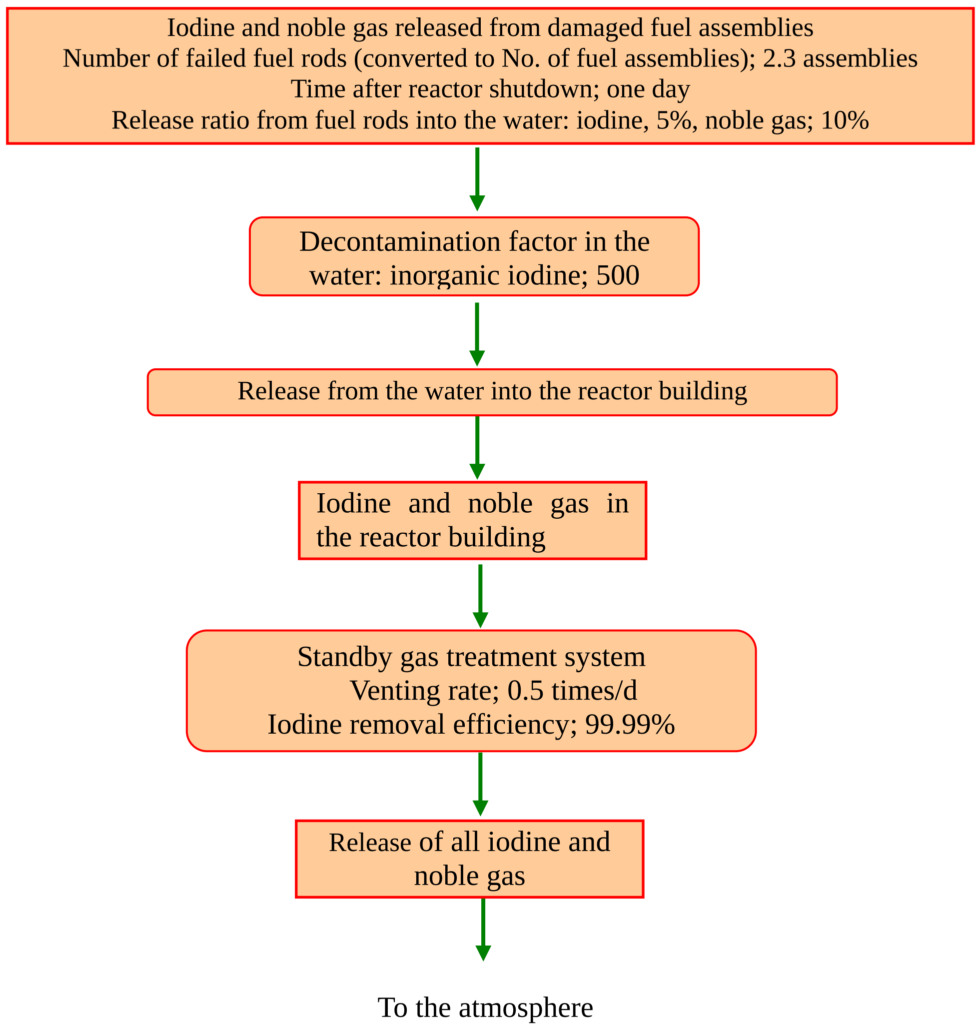


Fig. 12 Example of release processes of iodine and noble gas during a drop of fuel assembly (BWR)

(e) Loss of reactor coolant (PWR, BWR)

(i) Assumption

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

(ii) Conditions

a The reactor is assumed to be in operation at a power level with account taken of a margin for safety analysis to the rated power for a sufficiently long period of time.

b The concentration of the fission products in the reactor coolant before the occurrence of the

c 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 similarly evaluated as that of a steam generator tube rupture or main steam line break.

d 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.

e 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 a loss of the reactor coolant accident. 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.

f 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.

g 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 b. and c. 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%.

h The direct dose rate and skyshine 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.

i 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.

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

(iii) Release path

An example of release paths of fission products is shown in Figure 13 and Figure 15 for PWR and BWR, respectively.

(iv) Release process

An example of release processes of iodine and noble gas to the atmosphere is shown in Figure 14 and Figure 16 for PWR and BWR, respectively.

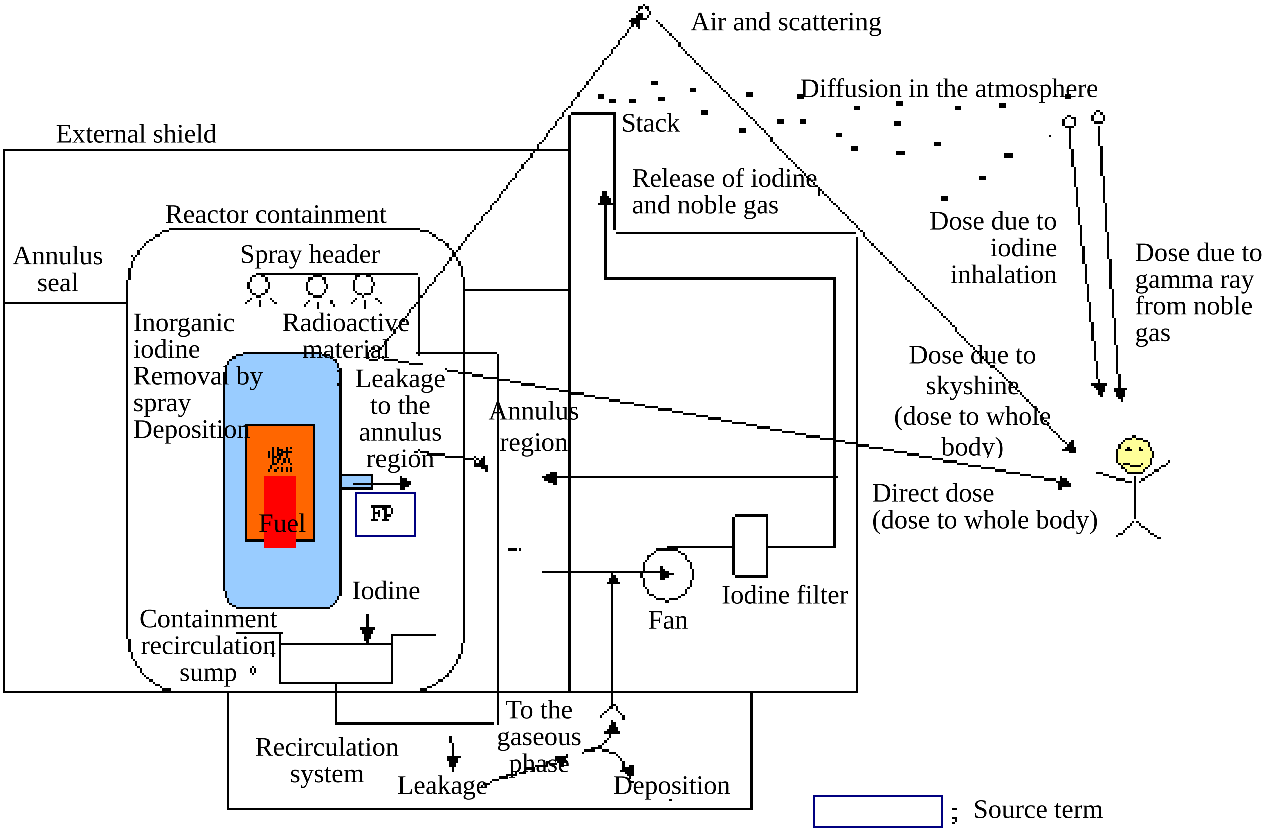


Figure 13 Example of release paths during a loss of reactor coolant (accident) (PWR)

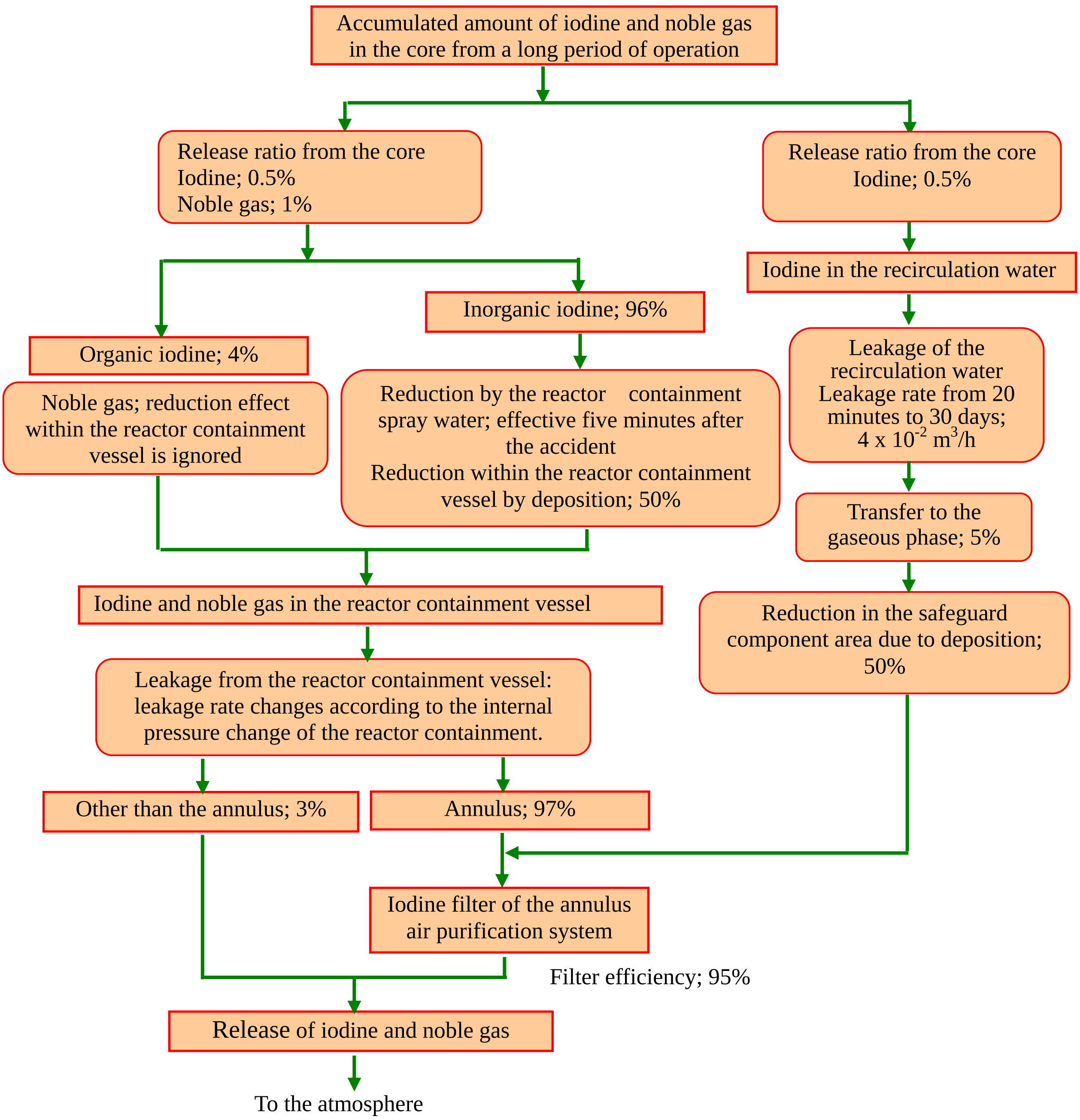


Figure 14 Example of release processes during a loss of reactor coolant (accident) (PWR)

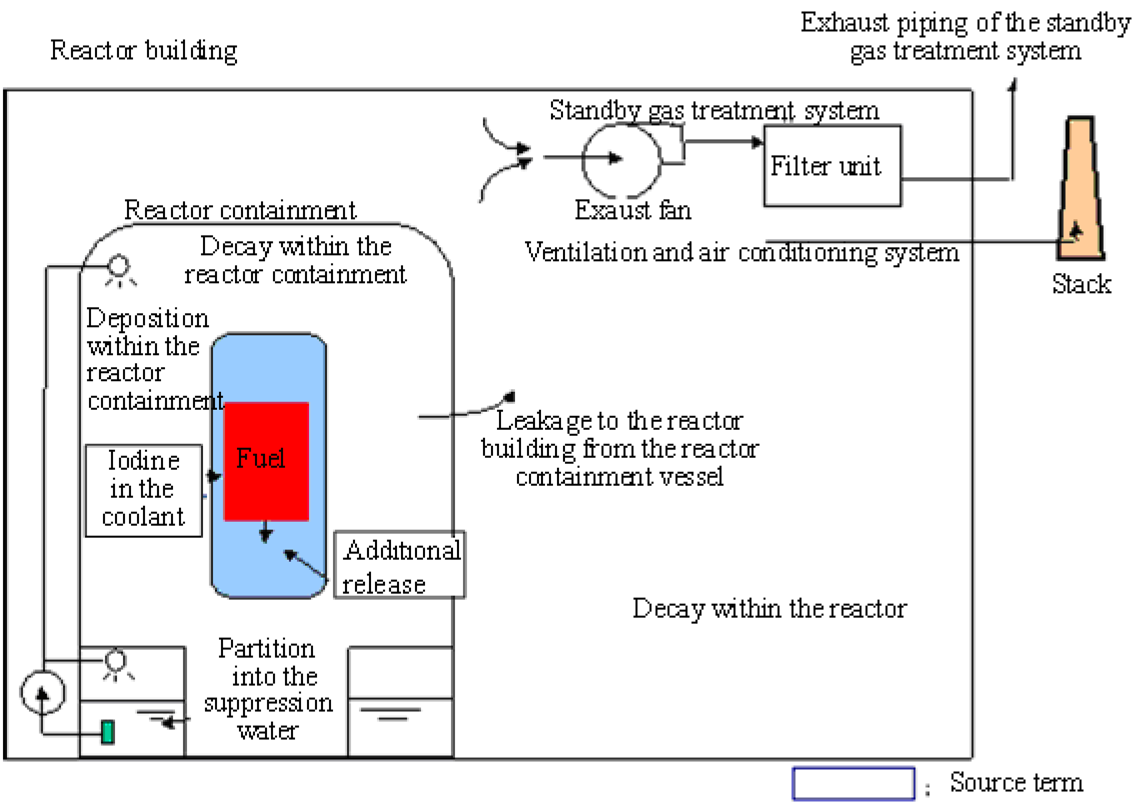


Figure 15 Example of release paths during a loss of the reactor coolant (accident) (BWR)

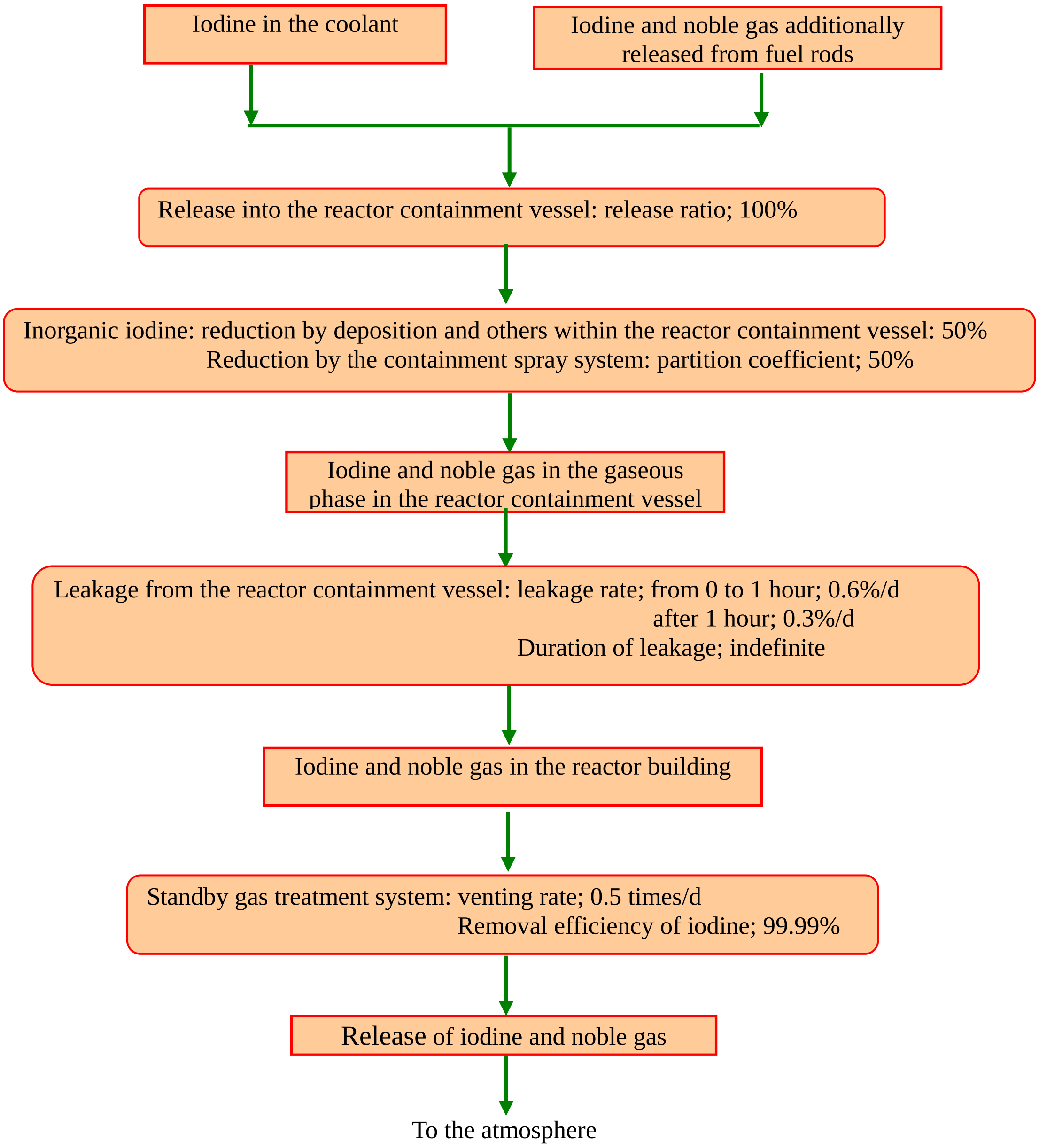


Fig. 16 Example of release processes during a loss of the reactor coolant (accident) (BWR)

(f) Control rod ejection (PWR)

(i) Assumption

An event of radioactive material release to the environment when one control rod cluster ejects out of a core due to a failure of control rod drive system, etc, is assumed, when a nuclear reactor is in a state of criticality or near to a criticality.

(ii) Conditions

a The reactor is assumed to be in operation at a power level with account taken of a margin for safety analysis to the rated power level for a sufficiently long period of time.

b Evaluation of the effective dose accompanying the outflow of the primary coolant is performed according to the case of a loss of the reactor coolant.

(iii) Release path

An example of a release path of fission products is shown in Figure 17.

(iv) Release process

An example of release processes of iodine and noble gas to the atmosphere is shown in Figure 18.

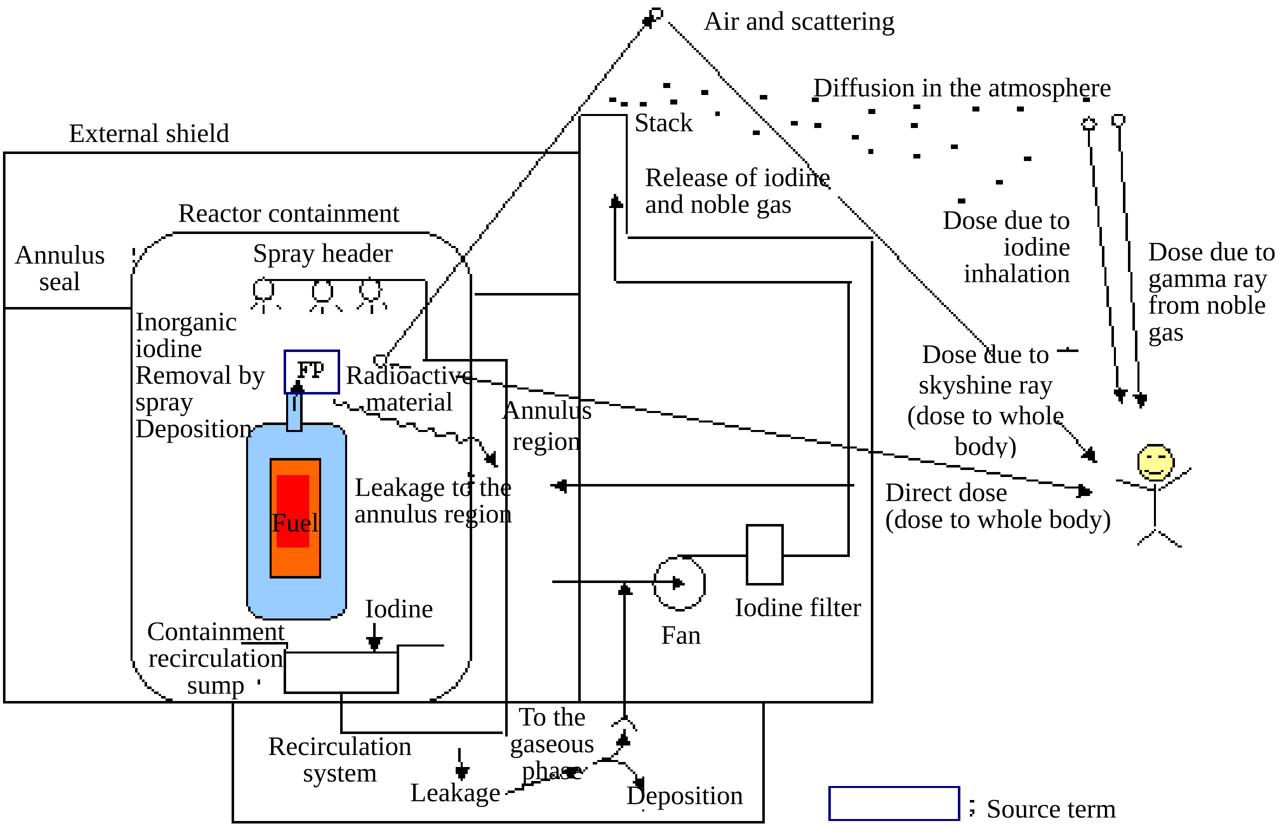


Figure 17 Example of release paths during control rod ejection (PWR)

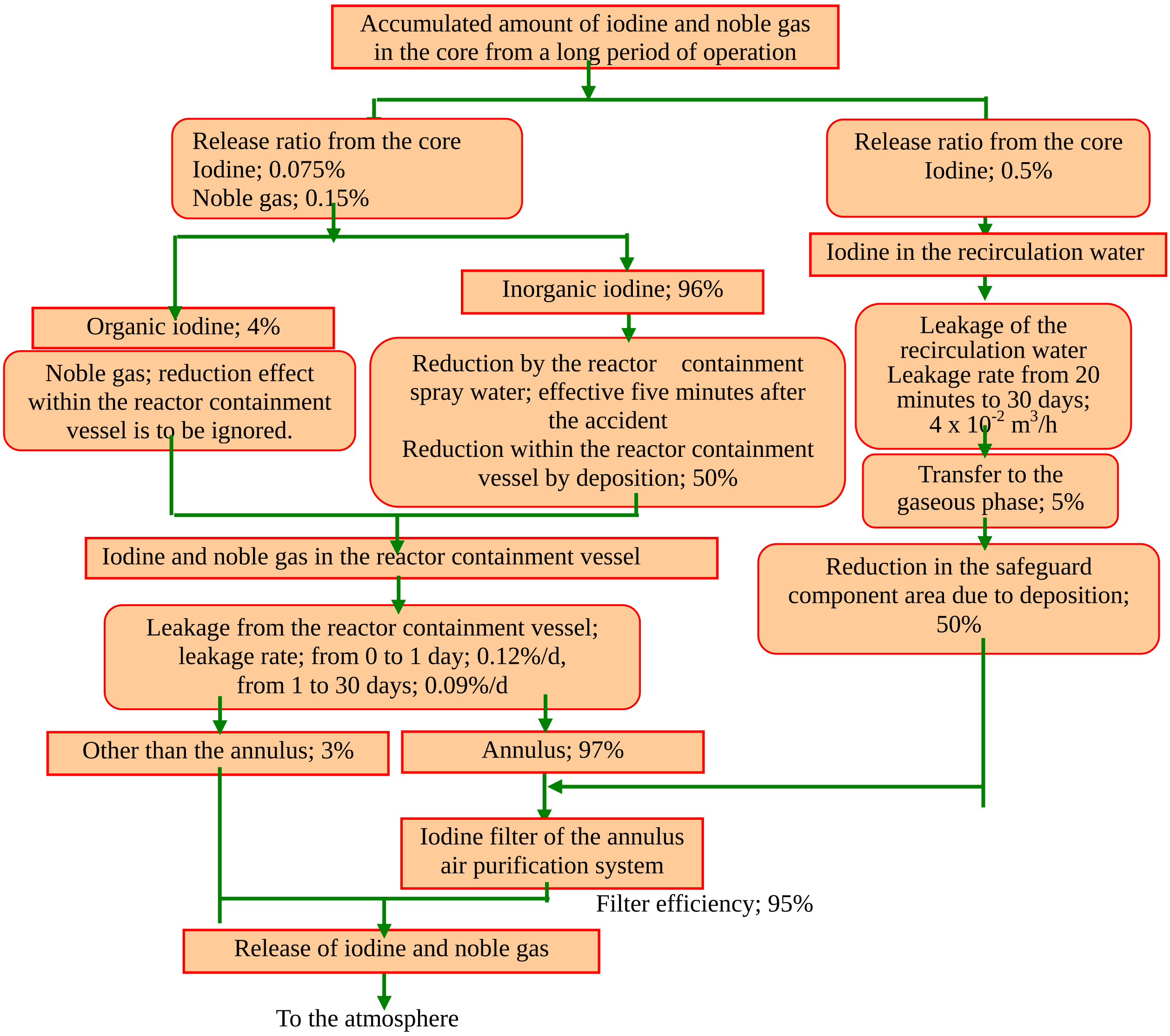


Figure 18 Example of release processes during control rod ejection (PWR)

(g) Control rod drop (BWR)

(i) Assumption

An event is assumed that rapid reactivity insertion and power-distribution change due to a fall from a core of a disjoined control rod from a control-rod driving shaft occurs while the reactor is critical or near criticality, resulting in a fuel rod failure and radioactive material release to the environment.

(ii) Conditions

a When an event occurs during high temperature standby or during partial power operation, the reactor is assumed to be in operation at a power level with account taken of a margin for safety analysis to the rated power for a sufficiently long 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 level with account taken of a margin for safety analysis to the rated power for a sufficiently long period of time up until 24 hours before the occurrence of the event.

b Fission products are 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 carryover ratio of iodine to the gas phase from the decomposed organic iodine and the inorganic iodine shall be 2%. The noble gas is assumed to transfer instantaneously to the gas phase.

c 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.

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

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

(iii) Release path

An example of release paths of fission products is shown in Figure 19.

(iv) Release process

An example of release processes of iodine and noble gas to the atmosphere is shown in Figure 20.

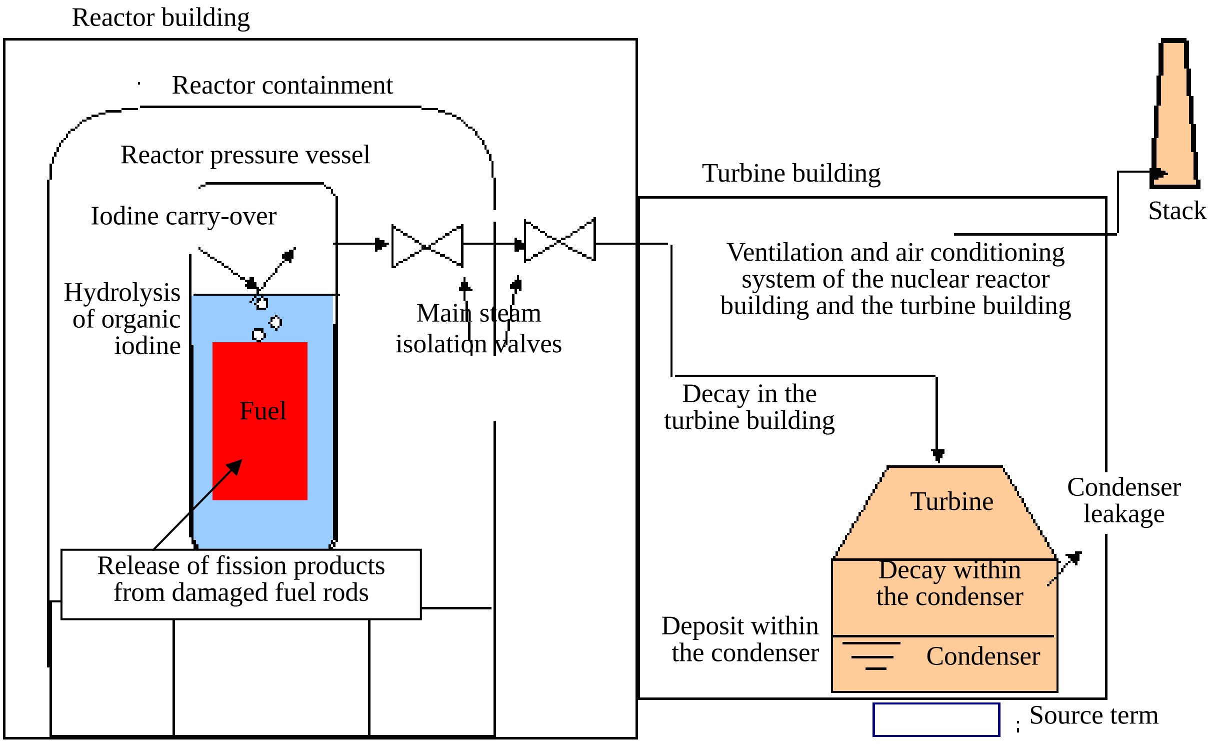


Figure 19 Example of release paths of fission products during control rod drop (BWR)

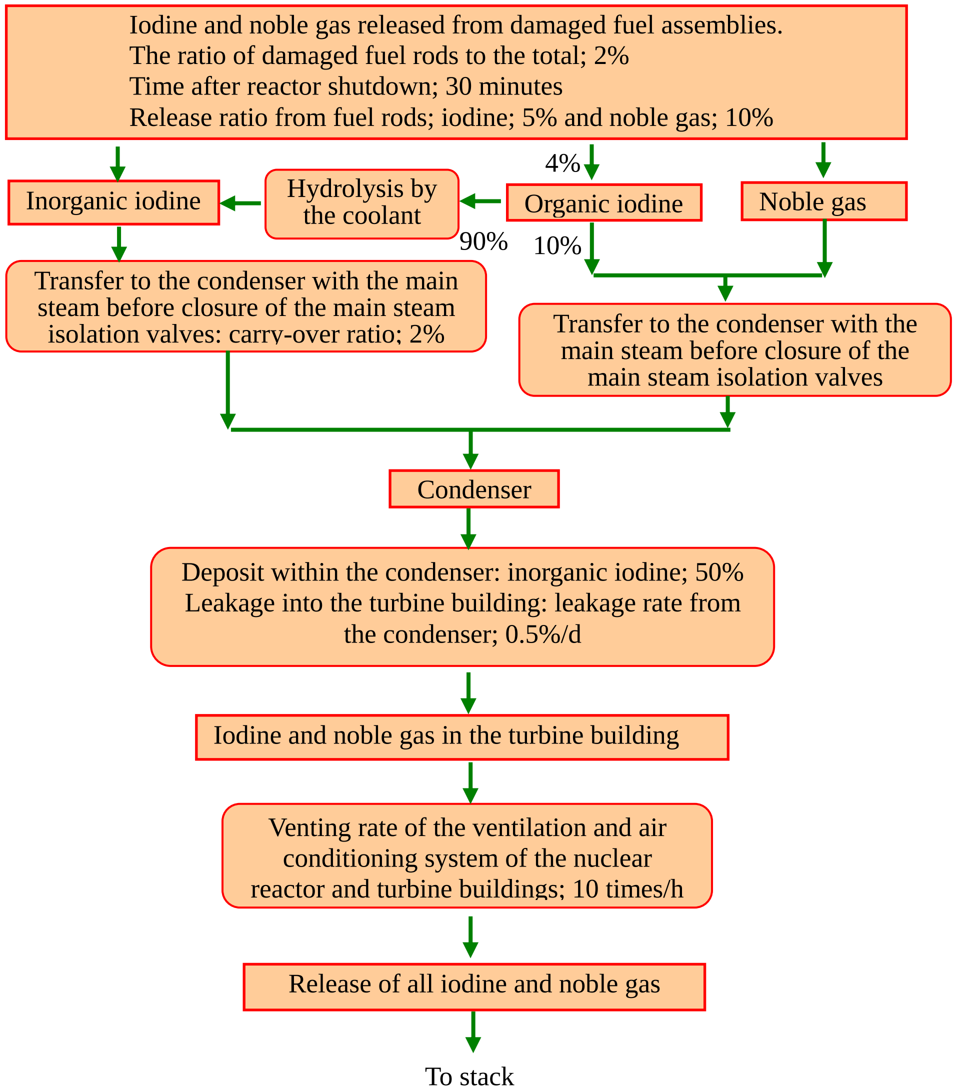


Figure 20 Example of release processes of iodine and noble gas during control rod drop (BWR)