Reactor Plant Safety Course FY2010
Winter Course

RPSC-Winter Course L-20

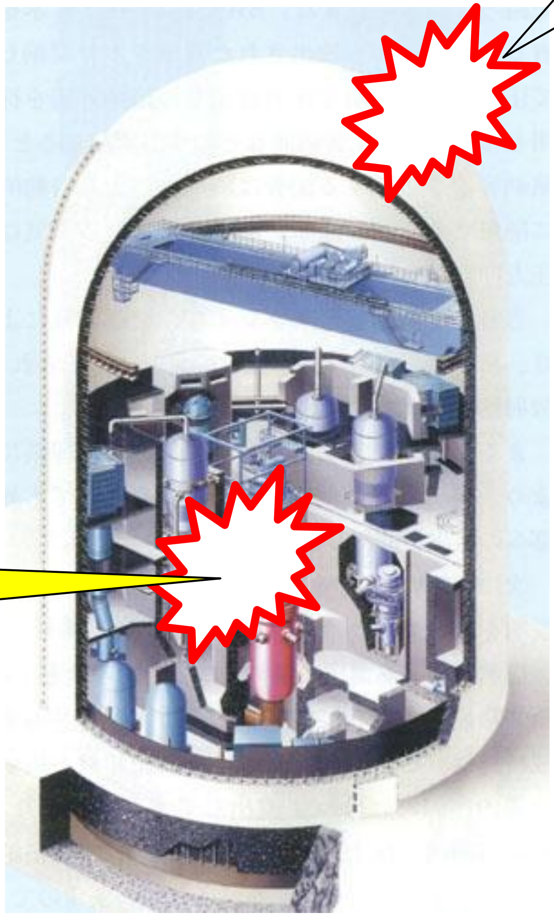
Safety Evaluation of Reactor Plant

February 2nd , 2011

Yoshiaki HIMENO
Technical Adviser, PESCO Co., Ltd.

Engineering Systems Need
Safety Design and Safety Assessment!

External Events



Natural Disaster:
Earth Quake
Typhoon
Tornado
Tsunami
Terrorism

by Site Selection
by Nuclear Plant Security

Internal Events

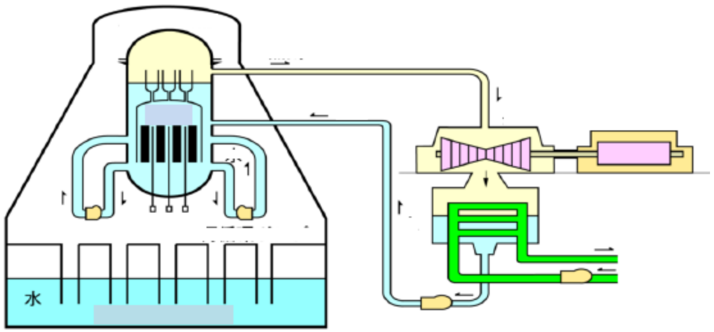
Transient
Accident

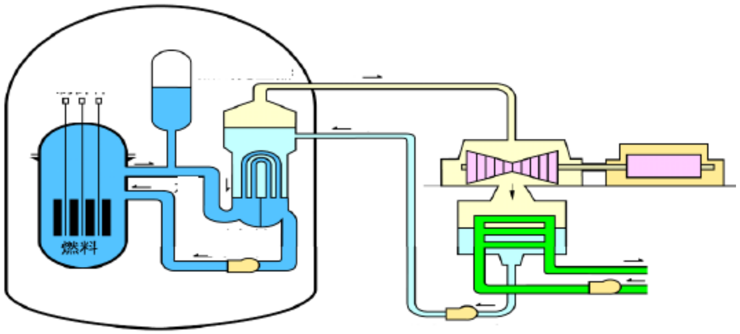
by Safety Design Evaluation

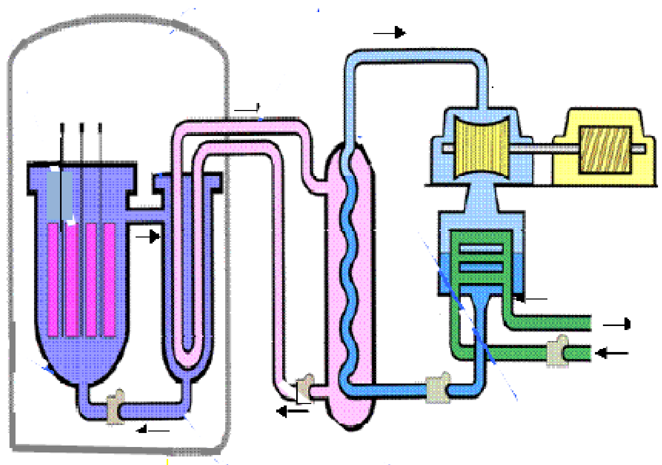
Contents

1. Safety Evaluation of LWR
1.1 Safety Design Evaluation of LWR
1.2 Site Evaluation of LWR

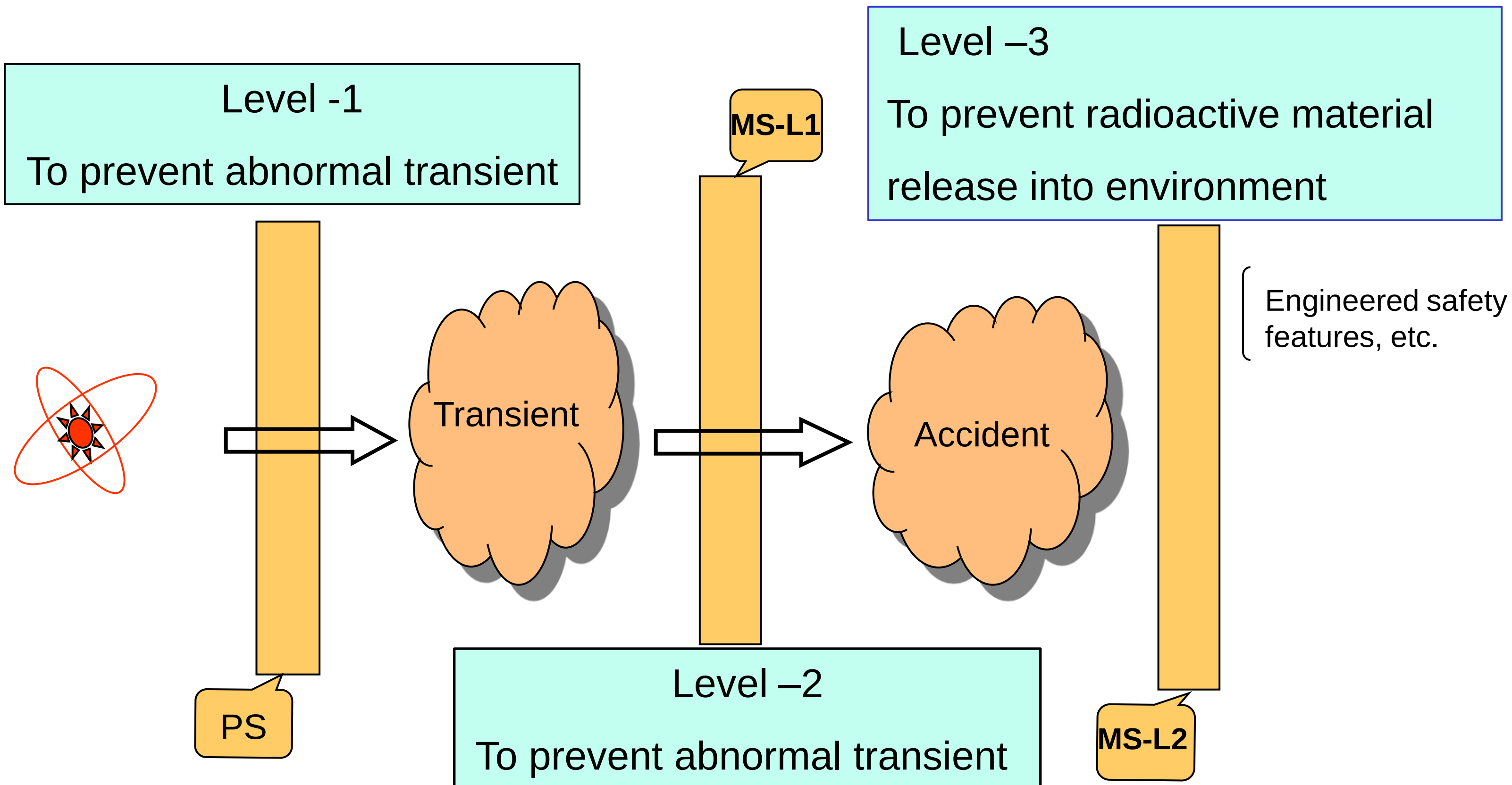
2. Safety Evaluation of FBR
2.1 Safety Design Evaluation
2.2 Site Evaluation







“Defense in Depth”

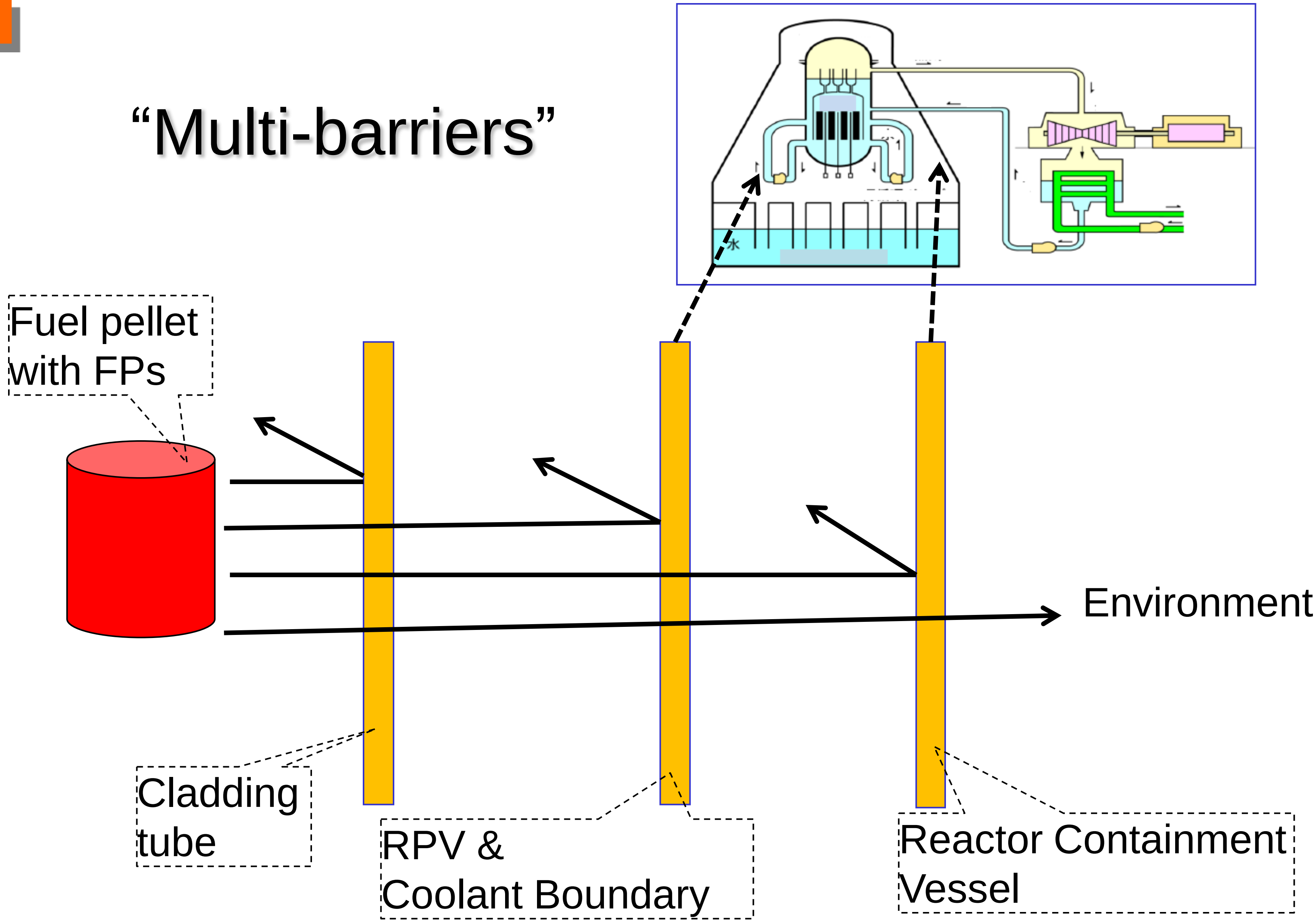


Reactor safety system, reactor shutdown system, etc.

PS, MS: defined in the “level of importance classification for safety function”

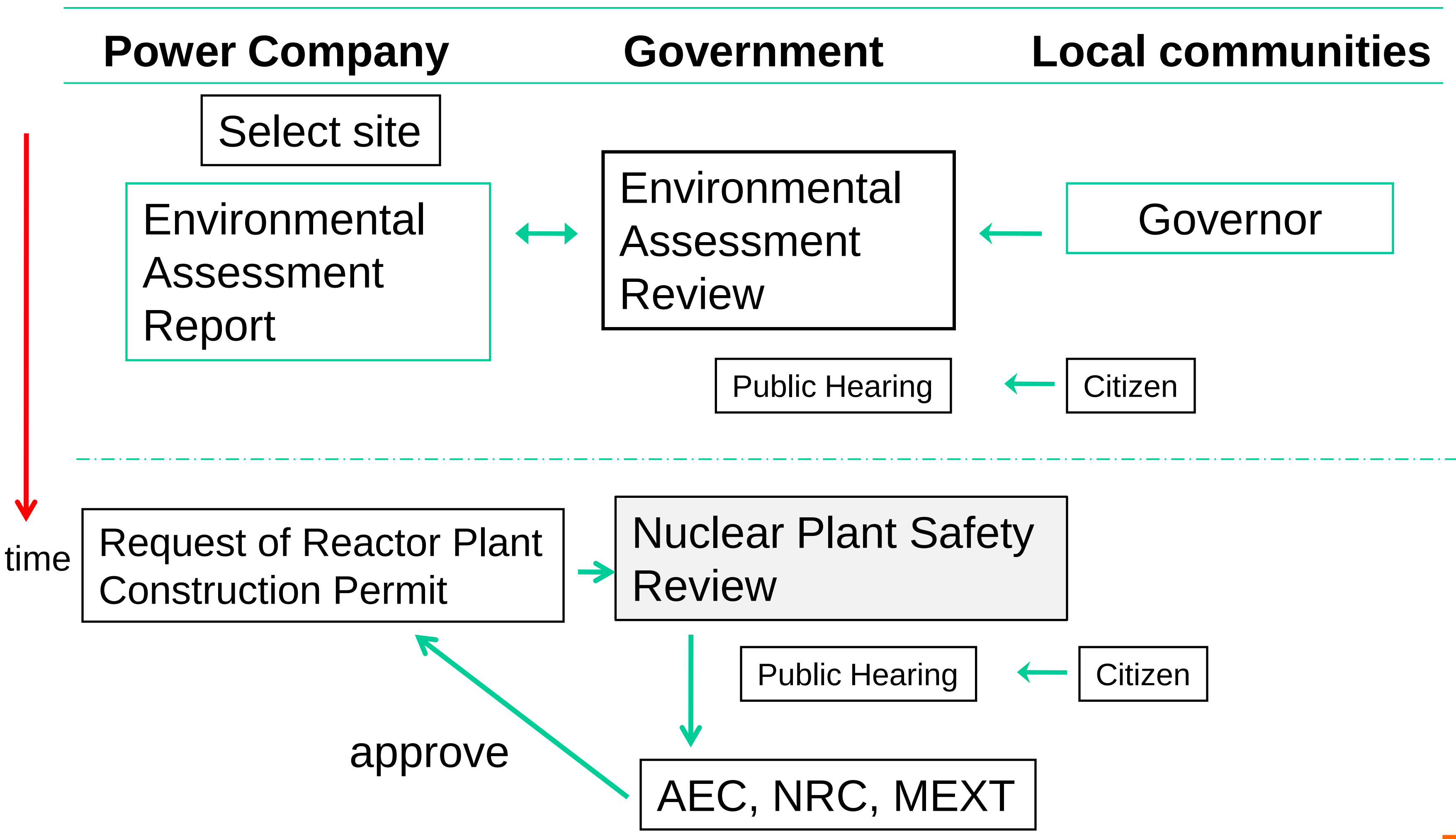
PS: protection system
MS: mitigation system

“Multi-barriers”



Shut-down DHR ECCS

Licensing Procedures for Power Reactor in Japan



Classification of Events for Safety Evaluation

Safety Design Evaluation	Abnormal Operational **Transient**	No possibility of radioactivity release	May occur
	Accident (**DBA**, **BDBA**)	Possibility of radioactivity release	In rare case, may occur
Site Evaluation	Maximum Credible Accident (**MCA**)	May occur in the worst case	Technically maximum possible radioactivity release
	Hypothetical Accident **(HA**)	Technically unlikely event	Larger radioactivity release than MCA
Prevention of Nuclear Disasters		Release larger than 10 times radioactivity from HA	For prevention measures within 8-10km from the site
Severe Accident		Serious impact	To strength safety and reliability of the plant

Deterministic approach, Probabilistic approach (PSA)

Typical Events Selected for Evaluation

Safety Design Evaluation	Abnormal Operational Transient	BWR/Withdrawal of CR PWR/Excess Feed-water to SG FBR/Loss of Off-site Power
	Accident (DBA, BDBA)	BWR/Main Steam Pipe Break PWR/Loss of Reactor Coolant FBR/1ry Coolant Leak FBR/1ry Pipe Break, ULOF (BDBA)
Site Evaluation	Maximum Credible Accident (MCA)	BWR/LOCA, Main Steam Pipe Break PWR/LOCA, SG –tube Break FBR/1ry Coolant Leak, 1ry Gas Leak
	Hypothetical Accident (HA)	BWR&PWR/Larger FP Release than MCA FBR/ do.
Prevention of Nuclear Disasters		Release of larger than 10 times FP than HA
Severe Accident		Hydrogen Generation, High Pressure Core Melt Ejection

Severity

Terminology

Site Evaluation

To investigate the site suitability of the plant.

Safety Design Evaluation

To investigate whether the safety design of the plant meets the regulatory requirements.

Safety Evaluation

To investigate whether the request of reactor construction permit meets the regulatory requirements.

Guidelines from Regulatory Authorities

Review Guide for Safety Evaluation of Light Water Reactor Plant

Review Guide for Safety Design of LWR Plant

Review Guide for Nuclear Reactor Site Evaluation
and Application Criteria

1.Safety Evaluation of LWR
1.1 Safety Design Evaluation of LWR

1.1.1 Events selected for Evaluations

Abnormal Transient during operation (DBE)

- by a single failure, a single malfunction, or a single operational error.
- Has a possibility to damage integrities of fuel and coolant boundary.

Accident (DBE)

- Has a possibility to release fission products into the RCV.

Technically Unlikely Accident (Beyond DBA)

- Results are assumed to be significance.
- Has a possibility to release fission products into the environment.

1.1 Safety Design Evaluation of LWR

1.1.2 Judgment Criteria for Evaluation Results

Abnormal Transient during operation (DBE)

Should be:

no fuel failure, no radioactive material release.

plant should be ready to restart after the necessary restoration.

Criteria:

Minimum critical power ratio (MCPR) (for BWR) , or

minimum departure from nucleate boiling ratio (MDNBR) (for PWR)

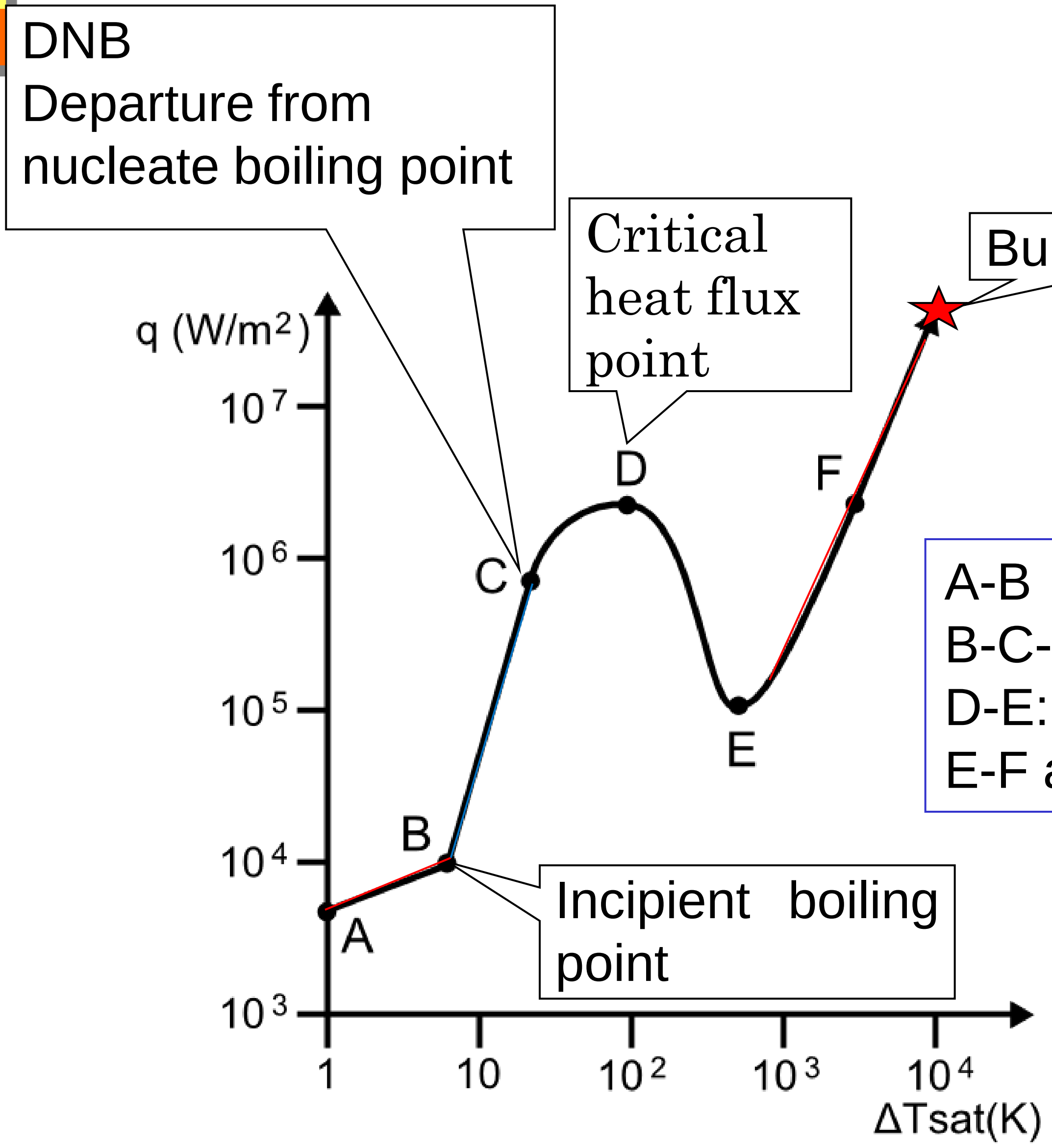
> the allowable limit(※1). See next page

Maximum fuel enthalpy (heat accumulated per unit weight of fuel)

< the allowable limit (40-110 depending fuel burn-up)

Pressure on the reactor coolant pressure boundary

< 1.1 times maximum operating pressure



A-B : Non-boiling
B-C-D: Nucleate boiling
D-E: Transient boiling
E-F and beyond: Film boiling

Boiling Curve (Nukiyama Curve)

1.1 Safety Design Evaluation of LWR

1.1.2 Judgment Criteria for Evaluation Results (cont'd)

Note ※1: MCRP (for BWR) and MDNBR (for PWR)

See boiling curve

MCRP: **Minimum Critical Power Ratio**

Critical Power (CP) :FA thermal power at onset of transition boiling.

Critical Power Ratio (CPR) = CP/ (FA thermal power).

MCPR: FA Thermal power which gives the Minimum CPR.

MDNBR : Minimum Departure from Nucleate boiling ratio

DNBR= (DNB heat flux)/(actual heat flux).

MDNBR: Minimum DNBR.

To avoid sudden clad temperature rise, transition boiling should be prevented.

Values recommended by Evaluation Guides are:

MCPR～1.06,and

MDNBR～1.17.

1.1 Safety Design Evaluation of LWR

1.1.2 Judgment Criteria for Evaluation Results (cont'd)

Accident (DBE)

Should be:

No core-melt, No marked core failure,

No secondary failure that may cause anticipated transient.

Keep integrities of multiple barrier against radioactivity release.

Criteria:

Core---- Max. clad temp. ＜ 1200 ℃

Fuel -----fuel enthalpy <the limit. (230 cal/gUO2）

Pressure on the pressure boundary < 1.2 times the design press.

Pressure on RCV < the maximum pressure.

No marked risk of radiation exposure to the public in the site periphery.

＜5 mSv

1.1 Safety Design Evaluation of LWR

1.1.2 Judgment Criteria for Evaluation Results (cont'd)

Evaluation Guide for ECCS Performance

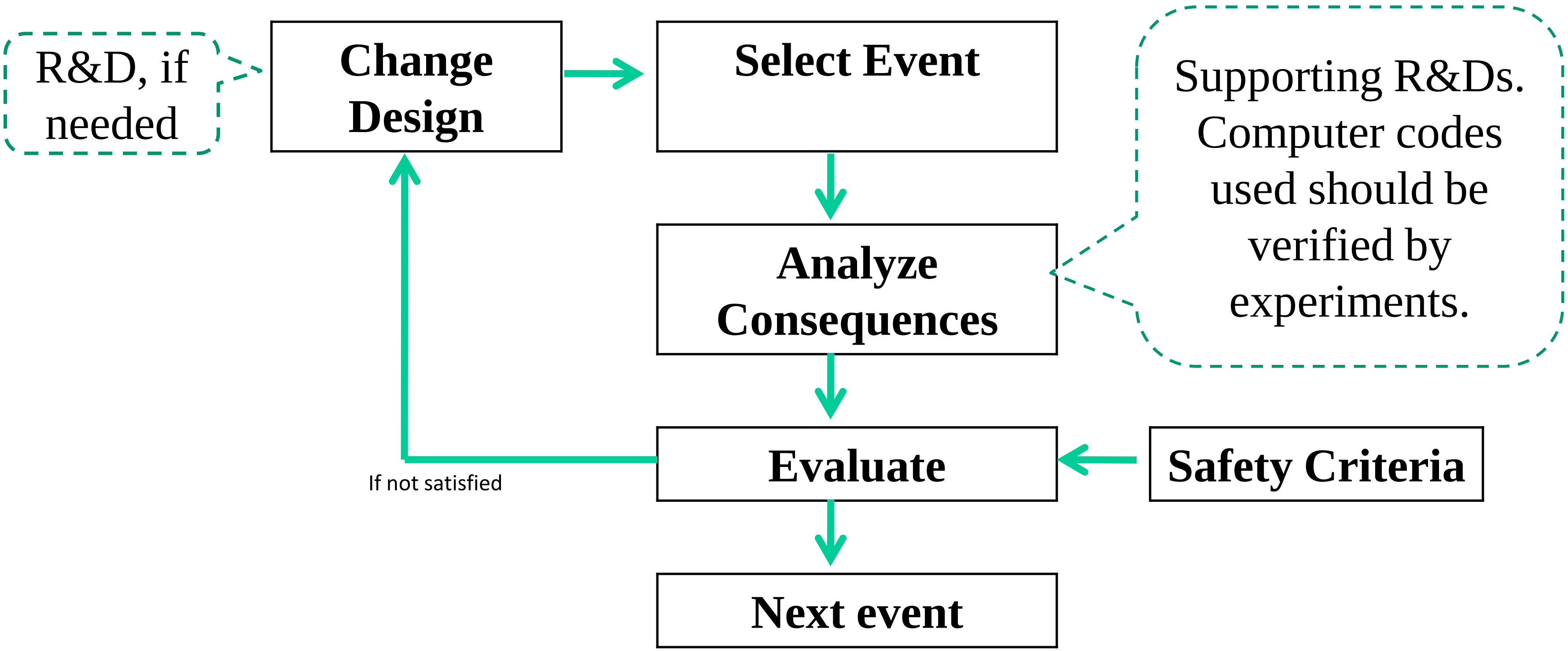
Judgment Criteria

- Maximum fuel cladding temperature < 1200 ℃.
- Stoichiometric oxidized cladding material < 15% of clad thickness.
- Hydrogen generated by water-Zr reaction in core:

  Should be small so that it dose not challenge to the RCV integrity.
- Decay heat removal capability should be maintained for a long period of time.

1.1 Safety Design Evaluation of LWR

1.1.3 Process of Evaluation



1.1 Safety Design Evaluation of LWR

1.1.4 Examples of Safety Design Evaluation of LWR

From next page, the following evaluations will be presented.

Anticipated operational transient

Trns-1: Abnormal withdrawal of control rod in BWR

Trns-2: Excess feed-water supply to SG of PWR

Accident

Acd-1: Main steam pipe break of BWR

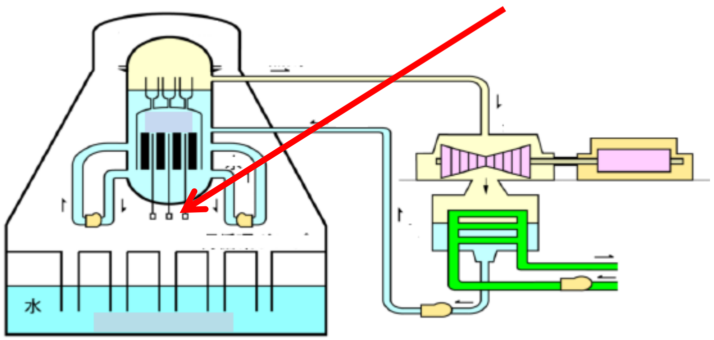
Acd-2: Loss of reactor coolant of PWR

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Trns-1 :Abnormal Withdrawal of Control Rod of BWR, (1/2)

Initiating Event:

During approach to criticality, an operator withdraws CR (control rod) continuously.



Protection Measures:

- Near criticality, operator proceeds with a help of a RWM ※.
- Reactor power excursion can be stopped by Doppler effect.
- Reactor scrams by "neutron flux- high" signal.

Note※: RWM: rod worth minimizer, i.e., CR operation monitoring equipment. When CR pattern goes out of allowable range, RWM prevents CR withdrawal/insertion.

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Trns-1 Abnormal Withdrawal of Control Rod of BWR, (2/2)

Results of analysis

- Neutron flux increases → reactor scram signal (at 9 s)

→ reactor automatic shuts down.

- fuel enthalpy increases, the value is smaller than the allowable limit.→ fuel integrity is secured.
- reactor pressure increases slightly. → coolant boundary is secured.

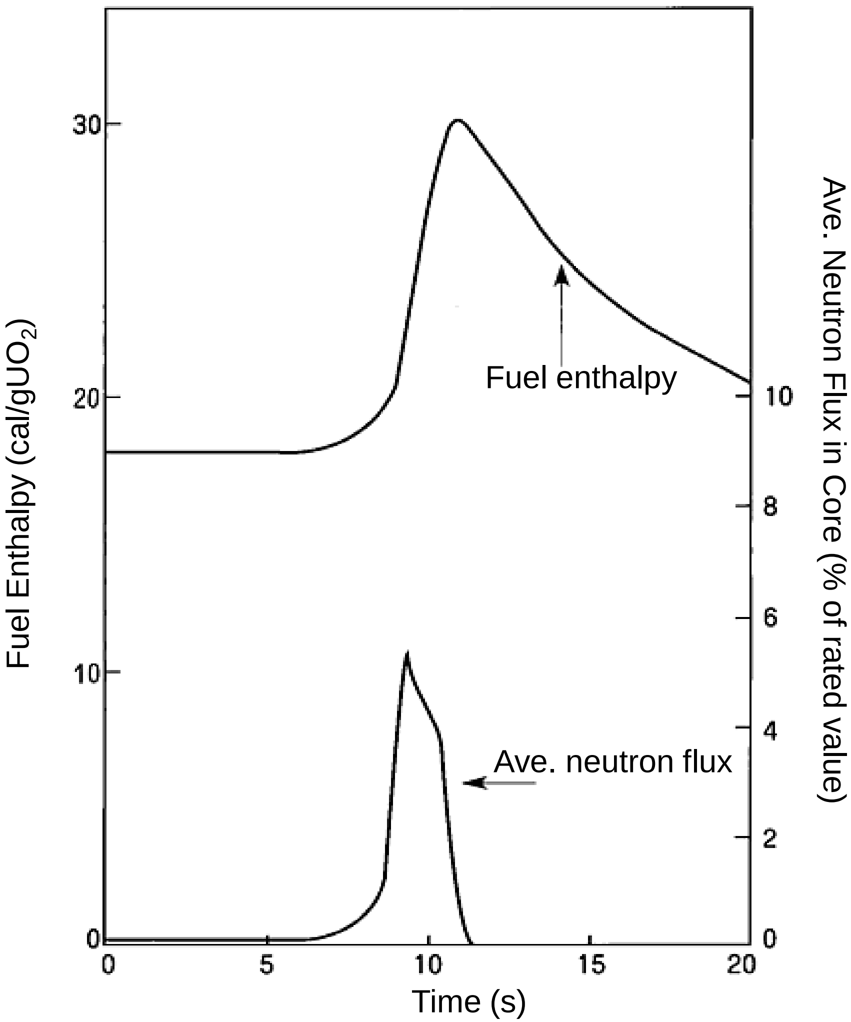


Fig. 1-1 Abnormal Transient due to Unexpected Withdrawal of Control Rod at Reactor Startup

〔出典〕 東京電力:柏崎刈羽原子力発電所原子炉設置変更許可申請書、平成4年10月

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Trns-2 Excess Feed Water Supply to SG of PWR, (1/3)

Initiating Event and Sequence:

At power operation, turbine bypass valve, main steam safety valve or main steam relief valve (MSRV) is fully opens by malfunction.

→ sudden increase in heat removal by SG→ reactor power increases

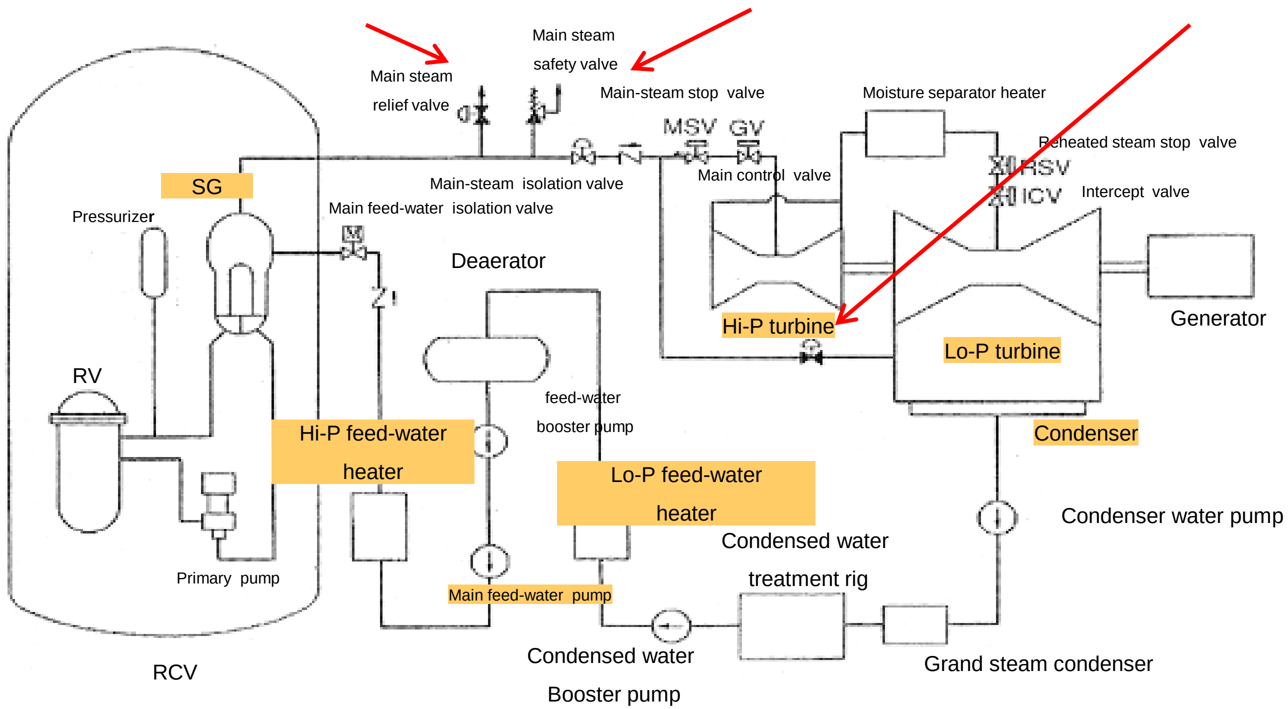


Fig. 1-2 Secondary cooling system (PWR)

[出展]（社）火力原子力発電技術協会：原子力発電所−全体計画と設備−（改訂版）、平成１４年６月、p20

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Trns-2 Excess Feed Water Supply to SG of PWR, (2/3)

Protection Measures:

Anomaly is detected by SG "water level" indicator and stream flow meter.

Reactor is shut-down by excess power/temperature difference.

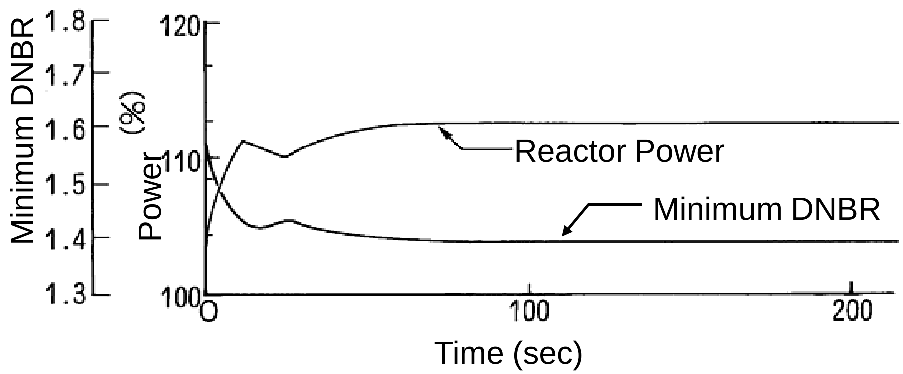
1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

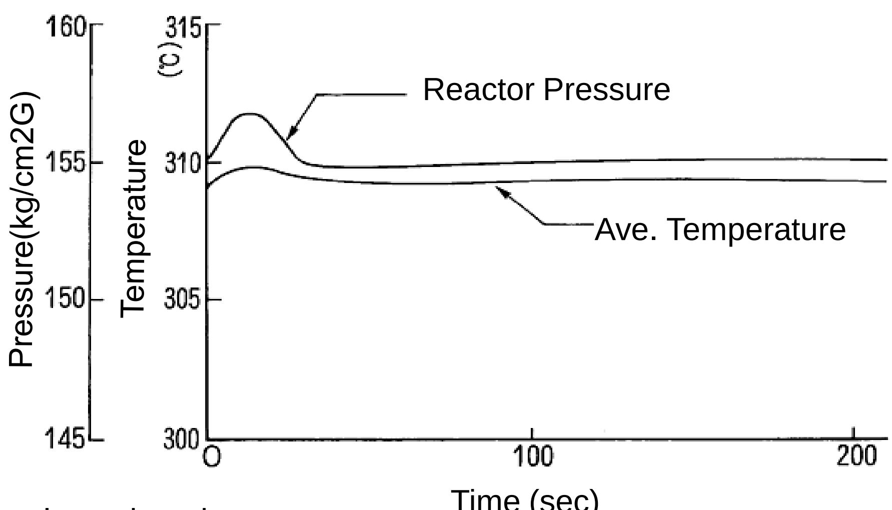
Trns-2 Excess Feed Water Supply to SG of PWR, (3/3)

Results of Analysis

- Excess SG's heat removal → lower RPV inlet coolant temp→ adding reactivity → power increasing.
- Reactor pressure increase is slight
- MDNBR (1.40) is safe → securing fuel & reactor coolant pressure boundary.
- Except initial transient, primary coolant temperature is kept constant.

Initial Condition : Reactor Power 102%
Coolant Density coefficient. : 0.51 (ΔK/K)/(g/cm^3)
Doppler Power coefficient : Negative value





〔出典〕 日本原子力発電:敦賀発電所
原子炉設置変更許可申請書、昭和55年8月

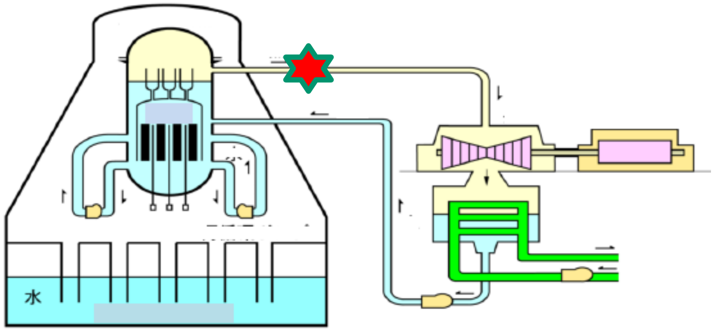
Fig. 1-3 Abnormal transient due to excess feed water supply to SG

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (1/7)

Initiating Event and Sequence:

At full power operation, 1 of 4 main steam pipes Guillotine breaks outside the RCV. Main steam flows out until closure of main steam isolation valve (MSIV).

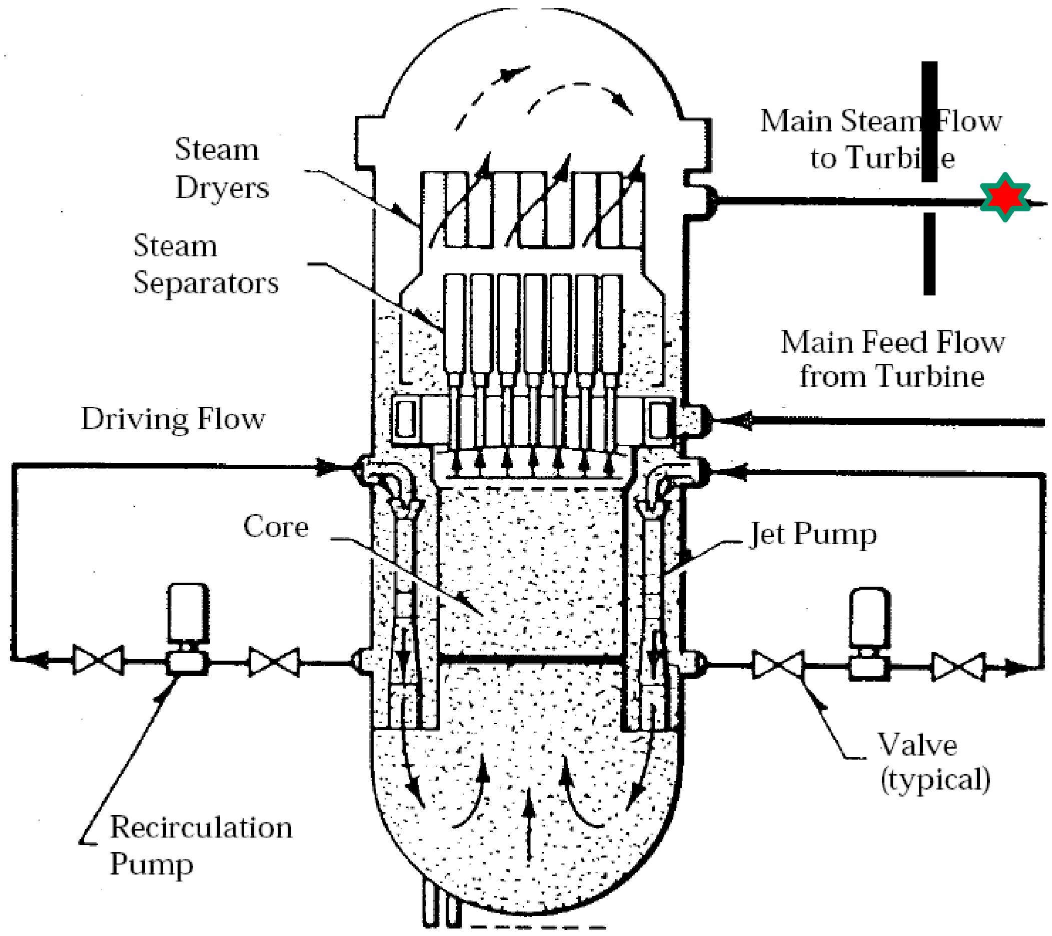


After the closure of MSIV, reactor pressure decreases to atmospheric pressure by ECCS.

In this event, loss of off-site power (causing the recirculation pump trip) and 1 out of 8 MSIV fails to close are assumed to happen.

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (2/7)



From NEA6846

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (3/7)

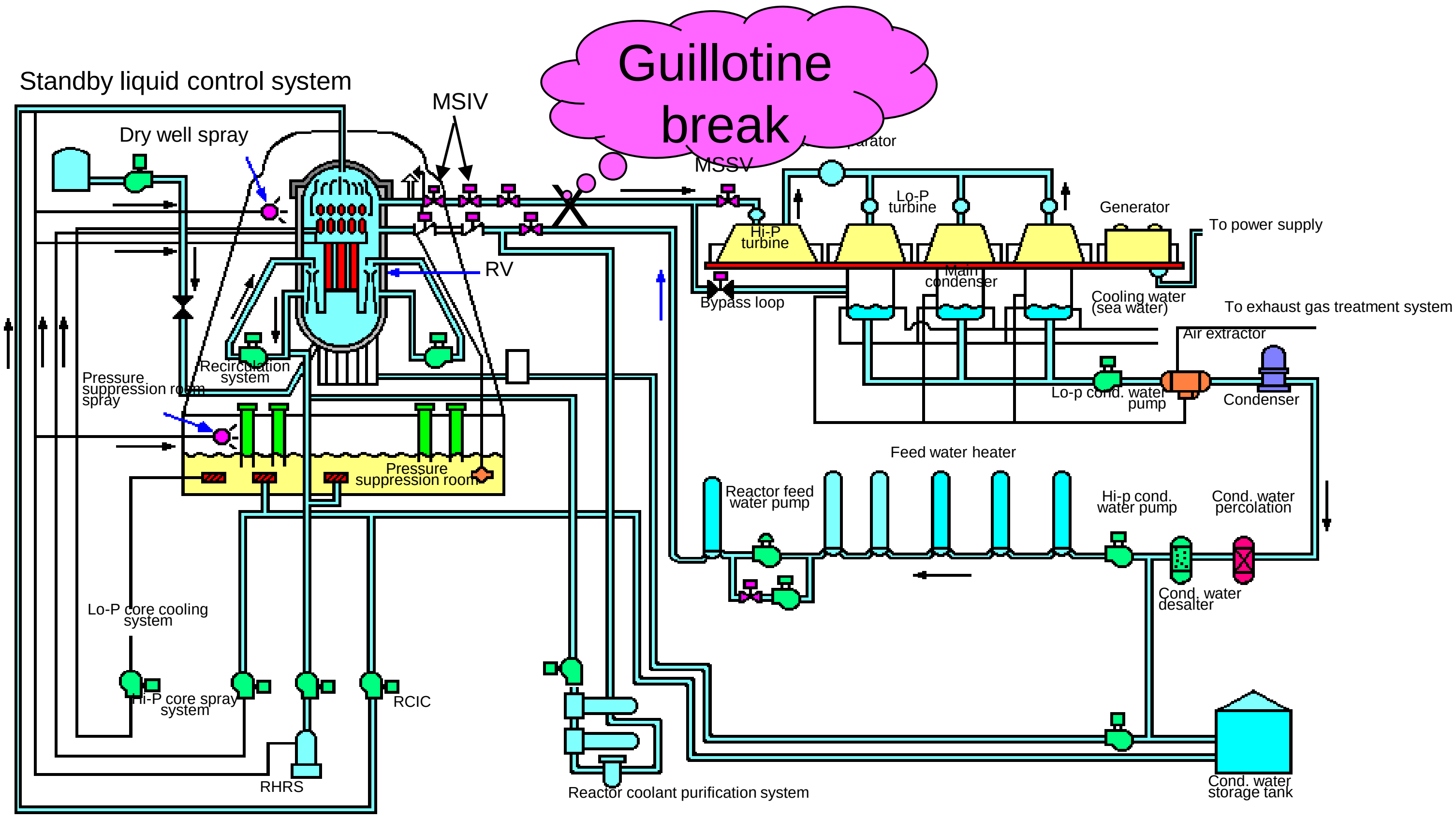


Fig. 1-4 Schematic diagram of BWR main system

【出典】資源エネルギー庁原子力発電課（編）：原子力発電便覧1997年版、電力新報社（1997）p.299

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (4/7)

Mitigation Measures

Coolant outflow control:

by outflow limiter (up to 200% of rated flow) located inside RCV of main pipe.

Automatic MSIVs close:

MSIVs at inside and outside of RCV automatically closed by signals of "main steam pipe flow-large", "main steam pipe tunnel temperature- high", "main steam pipe radiation level- high", "main steam pipe pressure-low", etc.

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (5/7)

Results of Analysis

- Steam in 3 intact pipes flows backward and flows out from the broken pipe.
- Steam outflow from the opening increases to about 3,500 kg/s which corresponds to the critical flow.
- Reactor pressure decreases, resulting in the rise of water level in the reactor due to the increase of the void.
- About 2 seconds later, the reactor water level reaches steam exit nozzle and steam outflow changes to two-phase flow thereafter.

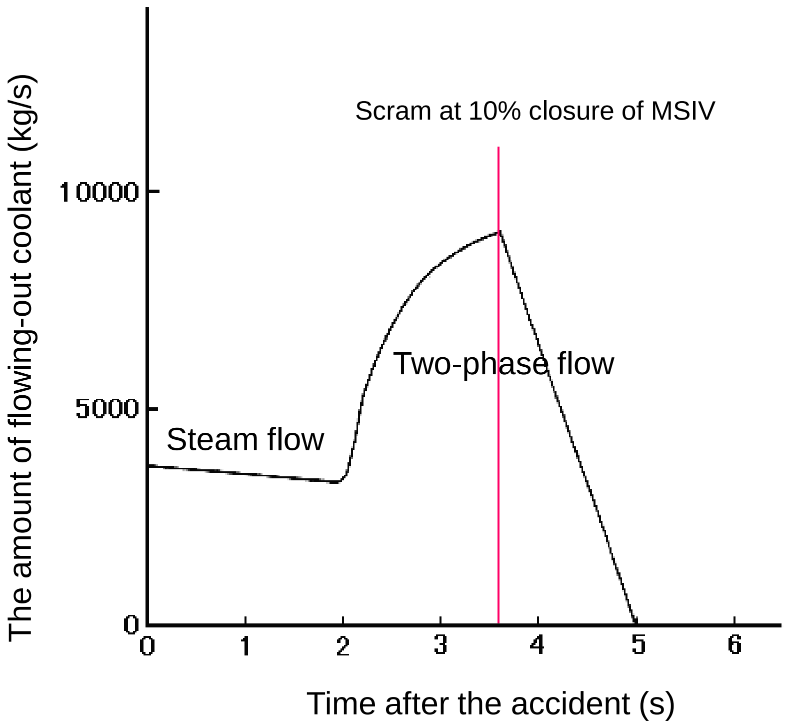


Fig.1-5 Change in the Amount of Flowing-out Coolant during the Guillotine break Accident of Main Steam Pipe

〔出典〕 東京電力:柏崎刈羽原子力発電所原子炉設置変更許可申請書、(平成4年10月)

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (6/7)

Results of Analysis (continued)

- The MSIV closes and reactor scrams in 5sec. Steam flowing out starts to decrease rapidly.
- Flow out steam/water until the MSIV closure is13ton steam and 22ton water.
- With this amount of coolant loss, core is not exposed, thus fuels dose not get thermal damage.
- After the accident, the reactor is cooled by the auto-depressurization system (ADS), the reactor core isolation cooling system (LPCS) and residual heat removal system (LPCI).

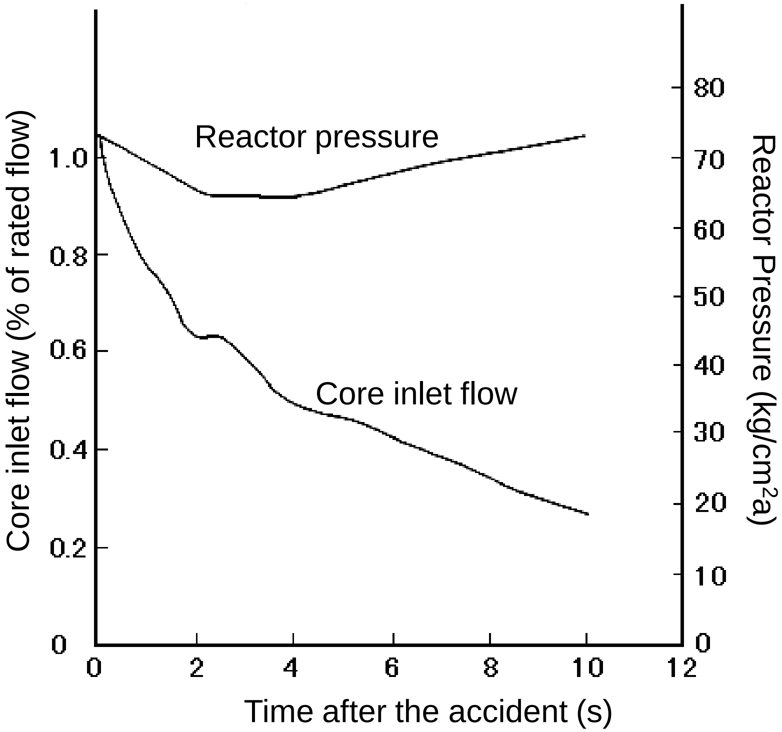


Fig.1-6 Change in the Core Flow and Reactor Pressure during the Guillotine Break Accident of Main Steam Pipe

〔出典〕 東京電力:柏崎刈羽原子力発電所原子炉設置変更許可申請書、(平成4年10月)

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-1: Main Steam Pipe Break of BWR, (7/7)

Radiological Consequences

- Before MSIV closure, radioactive materials and steam/water flow-out to the environment.

- After the MSIVs closure, coolant leaks from an isolation valve (1 out of 8 MSIVs failed to close by the single failure assumption) at a rate of 30%/day for 10hrs till reactor pressure reaches atmospheric pressure.

- Effective dose equivalent outside the border line is evaluated to be 0.02mSv.
- Thus the risk of radioactive exposure to the public in the vicinity of the plant is sufficiently low.

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (1/7)

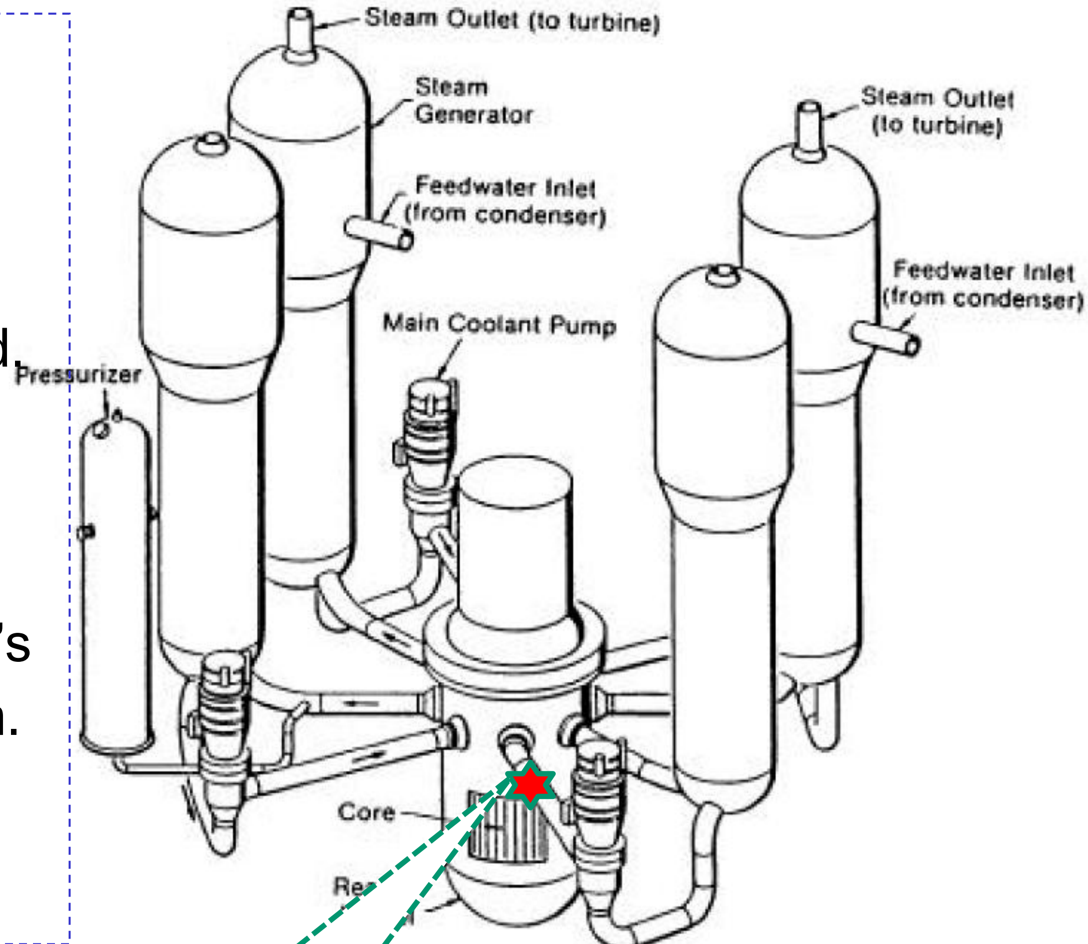
Initiating Event/ Sequence

At power operation, the 1ry pump outlet pipe **Double Ended Breaks**, thus, coolant leaks and RCV is pressurized.

Assume :

Loss of off-site power.

A single failure of 1 of ECCS's low pressure injection system.



Double ended guillotine

From NEA6846

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (2/7)

Mitigation Measures

1) ECCS (HPCI, AI, and LPCI) See also next page

Injects BAS (boric acid solution) into core to stop nuclear reaction, in the event of MSLB (Main Steam Line Break) /LOCA.

HPCI : At high reactor pressure, P_r , injects BAS, at first from refueling water tank, and in later, from recirculation sump.

AI : At P_r < 40atm, injects BAS in accumulator through 1ry piping system.

LPCI: At Pr <10atm, injects water, at first from refueling tank, and in later, from recirculation sump, into core through decay heat removal cooler.

HPCI: High Pressure Coolant Injection System
LPCI: Low pressure Coolant Injection System
AI : Accumulator Injection System. Accumulator is a big storage tanks connected to the reactor cooling system that have water pressurized with nitrogen

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (3/7)

Mitigation Measures (Continued)

2) Automatic reactor shutdown

"RCV pressure- high" signal or "Reactor pressure- low" signal actuates ECCS and shuts down reactor automatically.

3) RCV spray system

MSLB results in RCV's temp. and press. increases.Thus, RCV is cooled by spray system whose water is taken from refueling service water tank or RCV recirculation sump. Iodine is added to water.

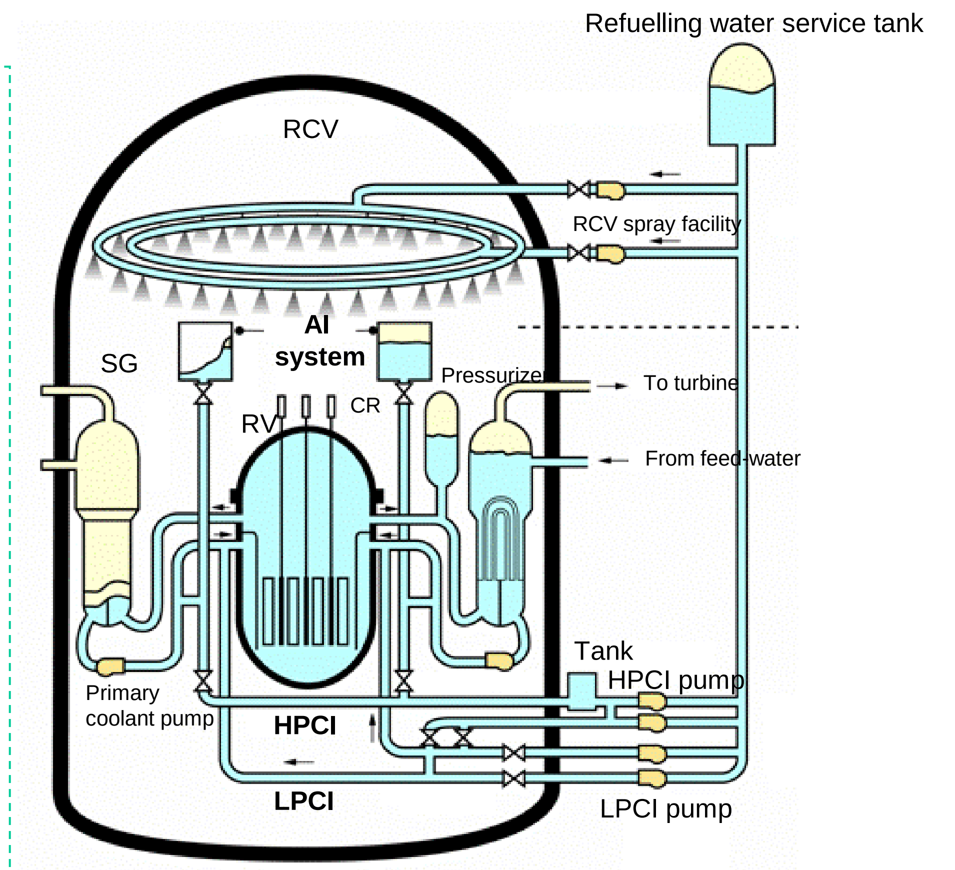


Fig. 1-7 Example of Engineered Safety Features (PWR)
- ECCS and RCV spray system -
[出展]資源エネルギー庁「原子力2004」より

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (4/7)

Results of Analysis

Blow-down

- Reactor pressure, P_r , decreases rapidly.
- After 26sec, P_r reaches RCV's pressure, thus, blow-down finishes.

Reactor scrum and ECCS startup:

- In 2sec, reactor shut down. But, ECCS start-up delays for 32s because of the assumption of the loss of off-site power.
- At 16sec, P_r decreases below 41atm, the holding press. of accumulator tank, thus, AI starts injecting boric acid solution automatically.

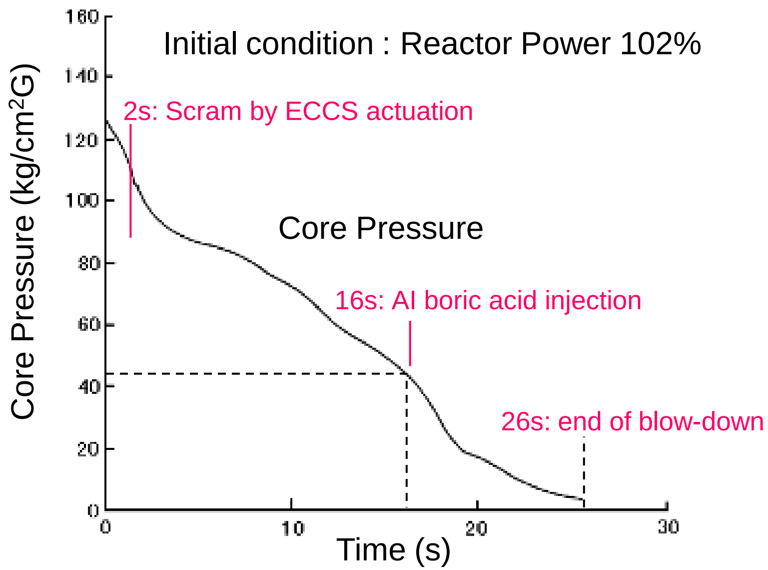


Fig. 1-8 Analytical Results of LOCA (1)

〔出典〕 日本原子力発電:敦賀発電所
原子炉設置変更許可申請書、昭和55年8月

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (5/7)

Results of Analysis (continued)

Re-flood :After 32sec, diesel generator voltage established, then, HPCI and LPCI start injecting water from around 34sec.

- After blow-down, water from accumulator starts to flood reactor lower plenum.
- At 37sec, core water level reaches bottom of core.
- Thus, core is cooled by steam generated in core and mixed flow with water droplet caught in the steam.

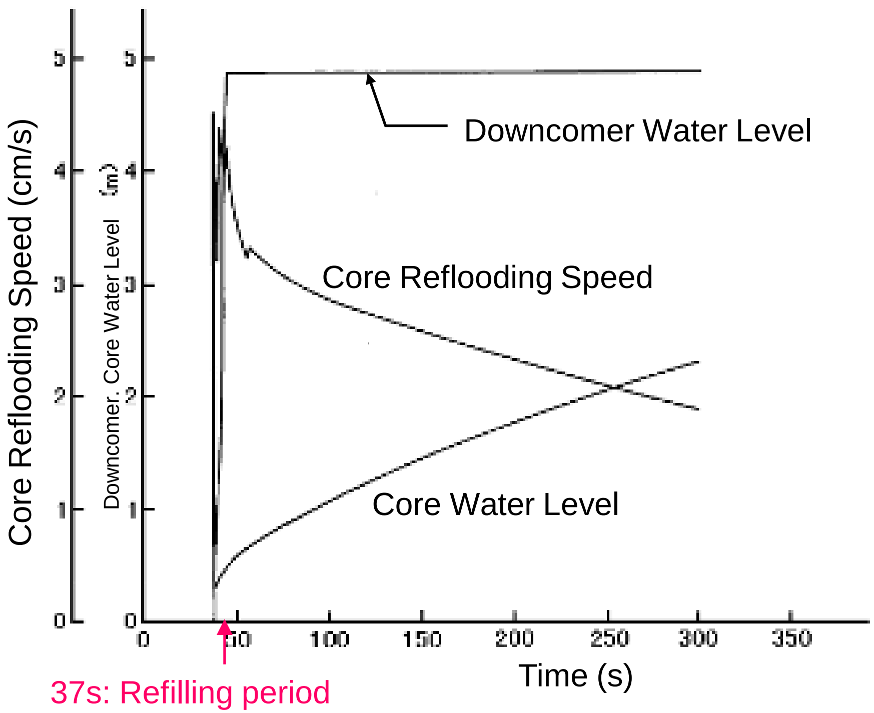


Fig. 1-9 Analytical Result of LOCA (2)

〔出典〕日本原子力発電:敦賀発電所原子炉設置変更許可申請書、昭和55年8月

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (6/7)

Results of Analysis (cont.)

Fuel Rod Temperature

・Cladding temperature, after reaching its maximum of 1086˚C at100sec, starts to decrease in accordance with core water level rising up, keeping its value <1200˚C.

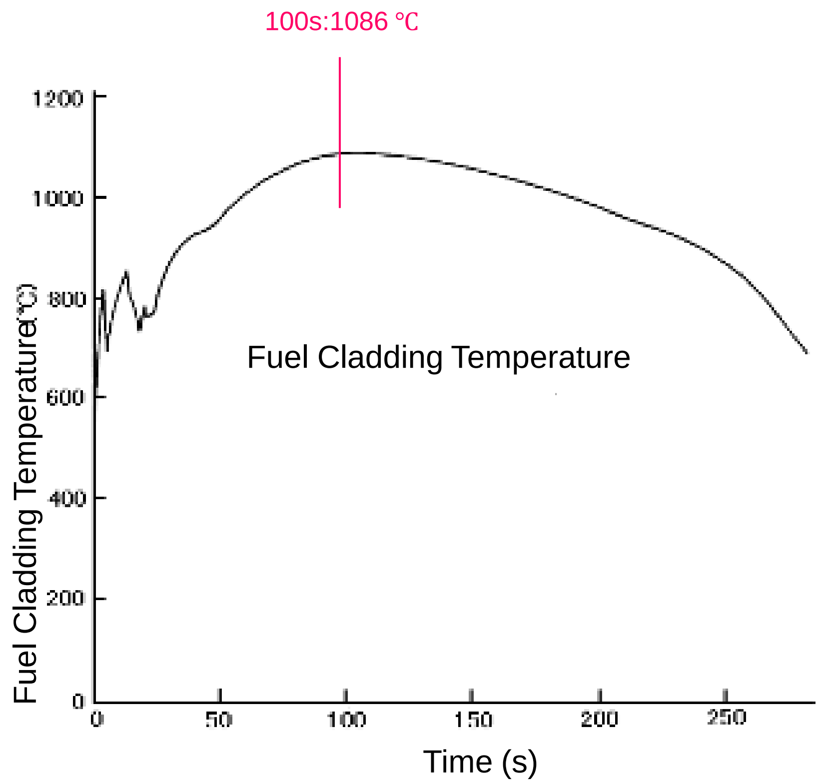


Fig. 1-10 Analytical Result of LOCA (3)

〔出典〕 日本原子力発電:敦賀発電所原子炉設置変更許可申請書、昭和55年8月

1.1.4 Examples of Safety Design Evaluation of LWR (cont'd)

Acd-2: Loss of Reactor Coolant of PWR, (7/7)

Results of Analysis (cont.)

RCV pressure: Reaches maximum of 3.4kg/cm^{2}G at 158sec, i.e., < maximum design pressure (4.0 kg/cm2G).

- Water injection continues.
- Water accumulated in RCV sump is re-circulated by the RHR pump, thus, core is cooled for long time.
- Start-up of annulus air cleanup equipment keeps radioactive material leaked to the environment to be so small that no marked risk to the public in the vicinity.

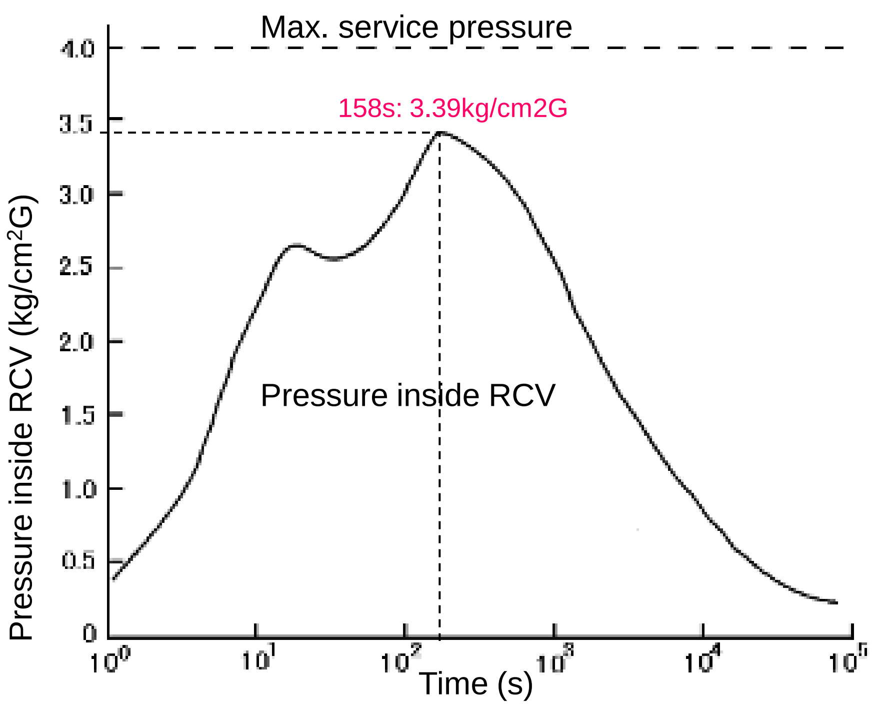


Fig. 1-11 Analytical Result of LOCA (4)

〔出典〕 日本原子力発電:敦賀発電所原子炉設置変更許可申請書、昭和55年8月

1.2 Site Evaluation of LWR

When site of a reactor is selected, a safety review is conducted prior reactor installation.

"Review Guide for Nuclear Reactor Site Evaluation and Application Criteria", or

"Review Guide for Site Evaluation",

is used in its safety review by the regulatory authorities to examine the adequacy of the site conditions.

1.2 Site Evaluation of LWR

1.2.1 Review Guide for Site Evaluation - Basic Concept (1) -

Site conditions in principle

1) **Natural disasters, which could trigger big reactor accidents, did not occur in the past and will not occur in the future.**

2) **Reactor site is sufficiently far from the public in the context of its safety protection facilities (Defense in Depth).**

3) **Reactor site including its periphery is located where appropriate measures can be undertaken for the public safety as necessary.**

1.2 Site Evaluation of LWR (cont'd)

1.2.1 Review Guide for Site Evaluation – Basic Concept (2) -

Fundamental requirements

1) Maximum credible accident:
Should not bring radiation impact to the public in the site periphery.

2) Hypothetical accident :
Should not bring marked radiation hazard to the public in the site periphery, and its impact on the collective dose should be small.

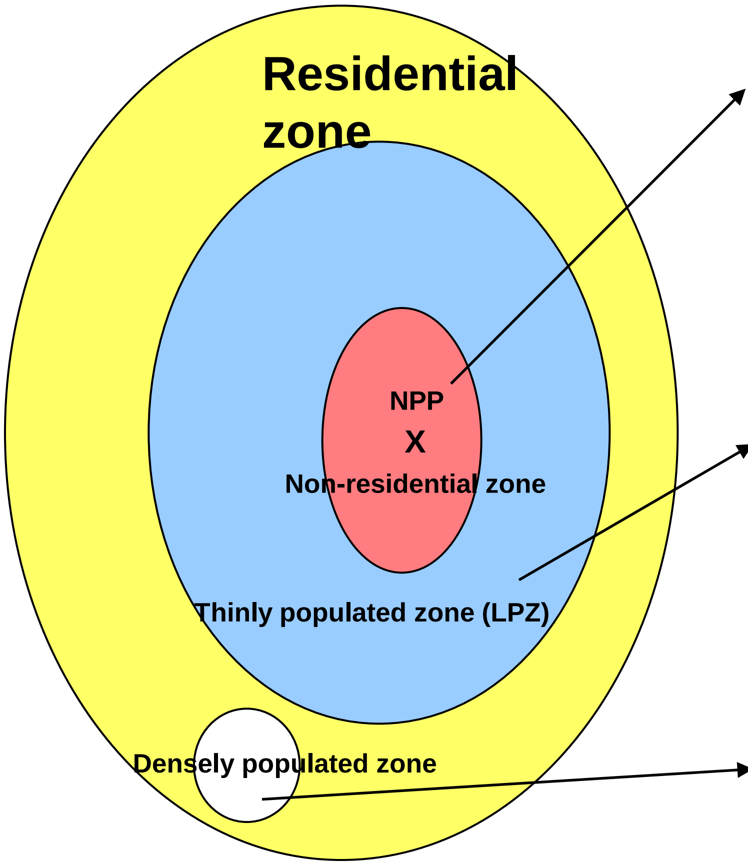
Where,

Maximum Credible Accident (MCA):
Unlikely accident in reality, but this is assume in order to evaluate site suitability under the conditions of large amount of radioactivity release from the reactor.

Hypothetical Accident (HA):
Beyond MCA, i.e. some of the safety protection systems do not work like they do under MCA; accordingly, more fission products are released.

1.2 Site Evaluation of LWR (cont'd)

1.2.1 Review Guide for Site Evaluation – Overview -



<u>Criteria for site evaluation</u>

Exposure dose calculated under MCAs should be lower than the value below:

- 1.5Sv for thyroid (infants)
- 0.25Sv for whole body

Exposure dose calculated under HAs should be lower than the value below:

- 3Sv for thyroid (adults)
- 0.25Sv for whole body

Accumulated exposure dose calculated under HAs should be lower than the value below:

- 20,000 man-Sv

1.2 Site Evaluation of LWR (cont'd)

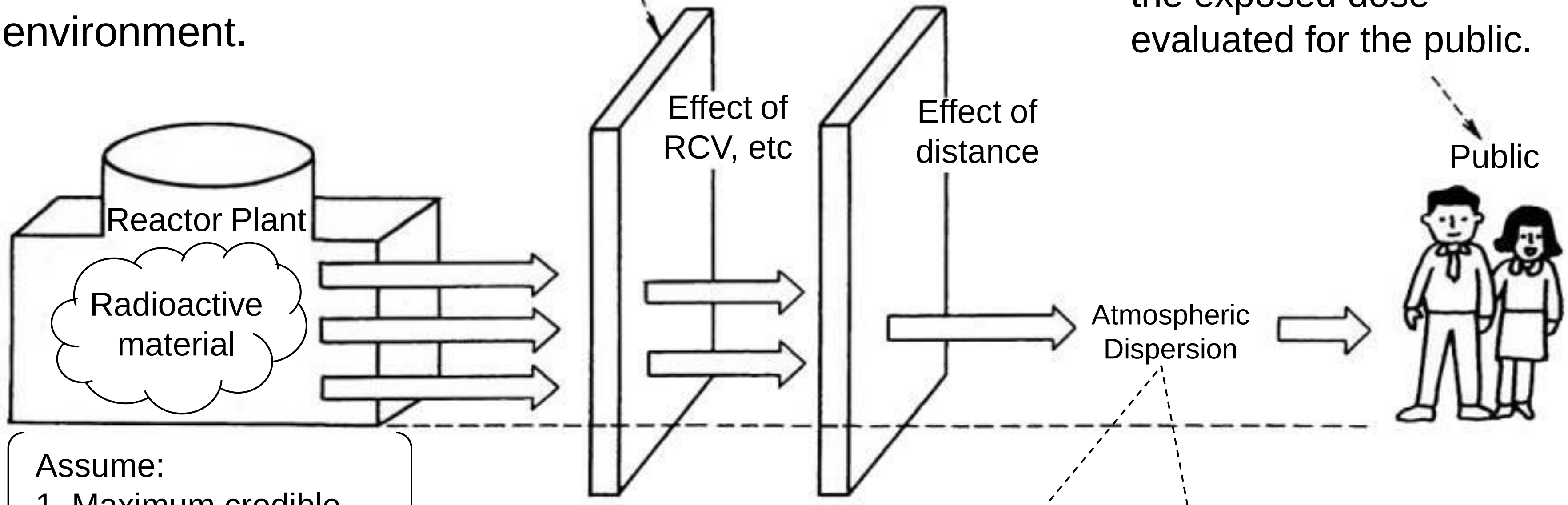
1.2.1 Review Guide for Site Evaluation – Overview

- Evaluation of Isolated Distance between Reactor and Public -

It is unlikely in reality, but assume a large radioactive material release to the environment.

Containment capacity of radioactive material has an effect equivalent to a long distance secured.

Isolated distance between the reactor and the public is judged from the exposed dose evaluated for the public.

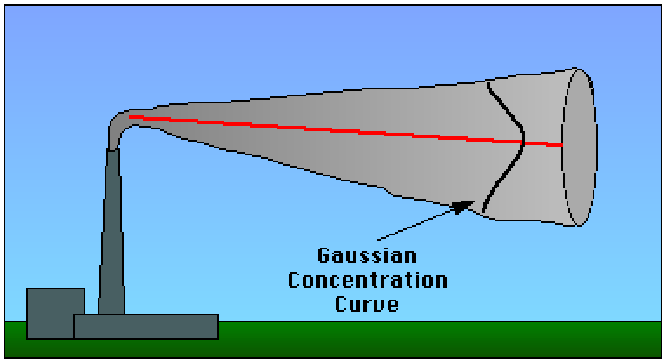


Assume:
1. Maximum credible accident (MCA)
2. Hypothetical accident (HA)

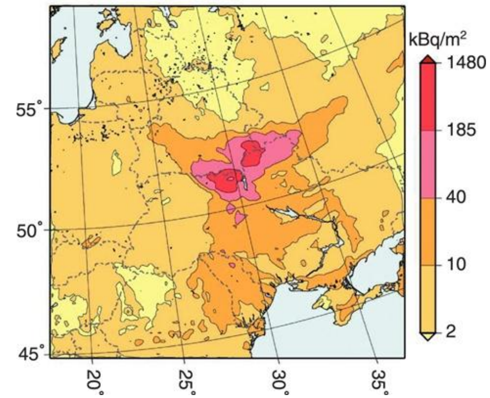
Diffusion Model or
3D Atmospheric Dispersion Model

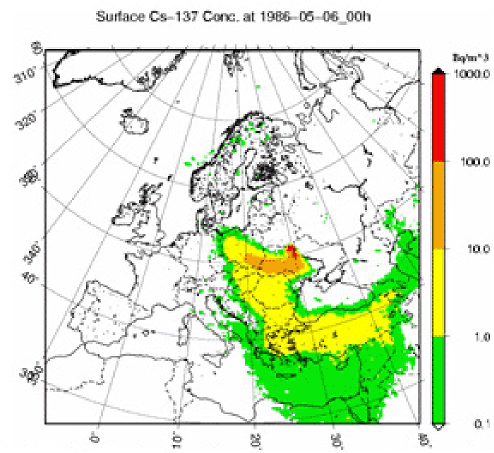
1.2 Site Evaluation of LWR (cont'd)

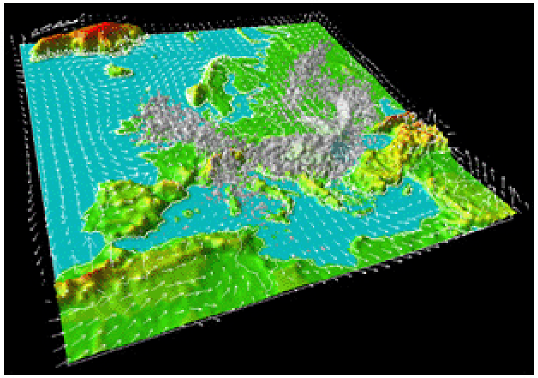
- Atmospheric Dispersion Models, Examples -



Gaussisn Model







Cs^{137} Precipitation and Radiation Cloud across Europe After Chernobyl Accident, by 3-D Numerical Analysis, JAEA

1.2 Site Evaluation of LWR (cont'd)

1.2.2 Accidents selected for Evaluation

Maximum Credible Accident, MCA

BWR: (1) LOCA due to the double ended guillotine break of main pipe

(2) Main steam pipe rupture outside the RCV

PWR: (1) LOCA due to the double ended guillotine break of primary pipe

(2) SG heat transfer tube rupture

EX.1: will be discussed later

Hypothetical Accident, HA

EX.2:will be discussed later

BWR/ PWR:

The amount of fission products released in the RCV is assumed to be much larger than the amount under MCA.

1.2.3 Examples of Site Evaluation of LWR

Ex.1 Maximum Credible Accident, PWR-LOCA, (1/4)

Assumed Accident Sequence and Way of Evaluation

- All fuel are damaged, and FPs in the fuels are released into RCV.

  FPs released in RCV are:

  - Iodine (I) : 1%
  - Rare gas : 2%

- They gradually leak out from RCV to the annulus with leak-rate adding a certain margin to its design value.

  - Leak-rate: 0.15%/d(0~1d),

    : 0.075%/d (1d~30d)

- Majority of FPs are removed by filter in the air cleanup system of the annulus.-Iodine removal efficiency of filter is 90%
- Diffusion of rest FPs released to the environment is evaluated by the "Meteorological Guideline on the safety analysis of power reactor facility."

1.2.3 Examples of Site Evaluation of LWR

Ex.1 Maximum Credible Accident, PWR-LOCA, (2/4)

Results of Analysis

- Iodine and rare gas leak in accordance with processes such as shown in Figs.1-12 .

▪Amount of iodine and rare gas leaked to the environment are:

  - Iodine : About 1.4E12Bq
  - Rare Gas : About 1.15E14Bq

Radiation dosage

- Maximum exposed dose on the outside of the site boundary :
  - About 0.12Sv for thyroid (of infant)
  - About 0.0002Sv for the whole body

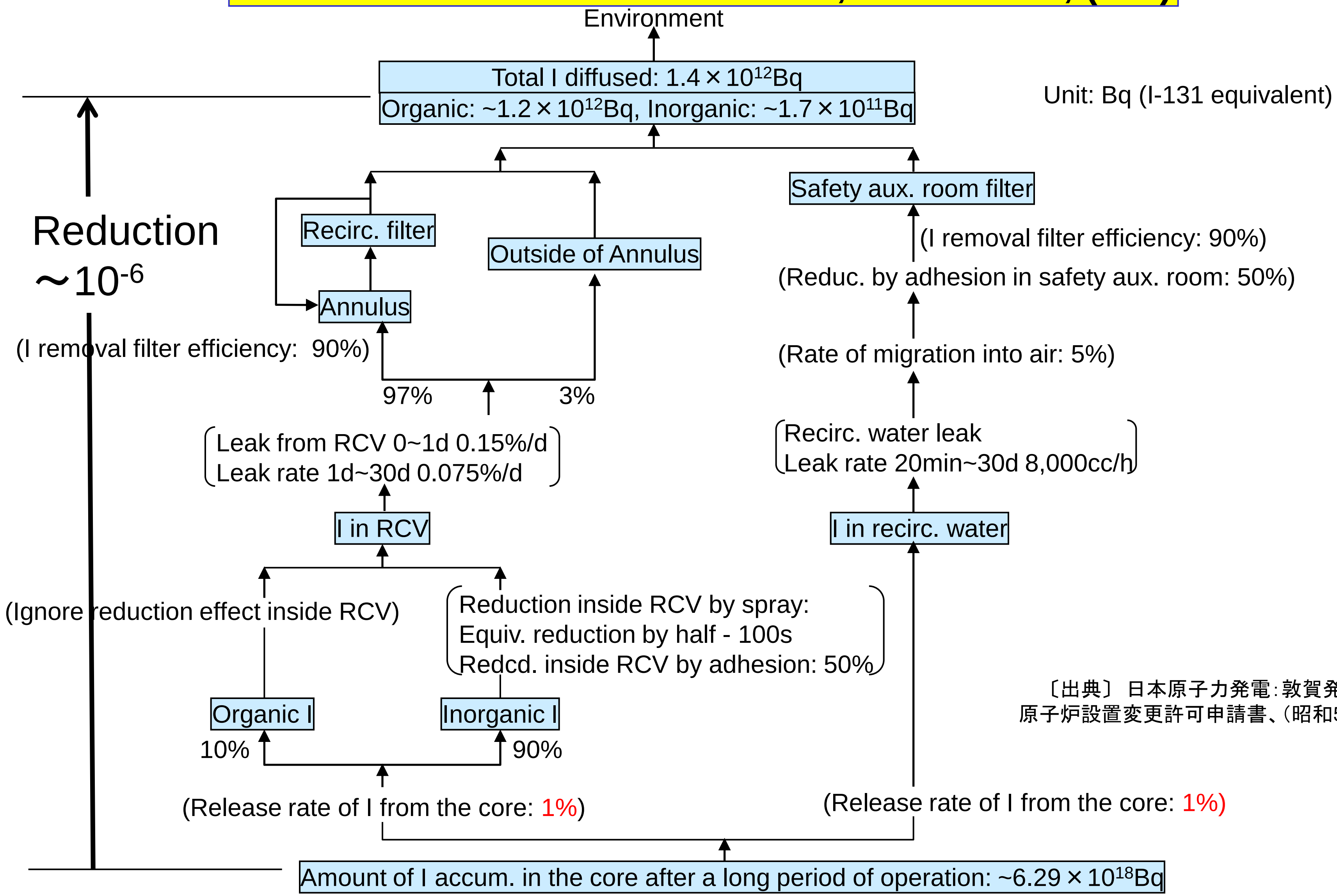
They are sufficiently low in compared to the allowable upper limit:

1.5Sv for thyroid (of infant) ,

0.25Sv for the whole body.

1.2.3 Examples of Site Evaluation of LWR

Ex.1 Maximum Credible Accident, PWR-LOCA, (3/4)



〔出典〕 日本原子力発電：敦賀発電所
原子炉設置変更許可申請書、(昭和55年8月)

Fig. 1-12 Process of **Iodine** Leak to Environment during LOCA (MCA)

1.2.3 Examples of Site Evaluation of LWR

Ex.2 Hypothetical Accident, PWR-LOCA, (1/4)

Assumed Conditions

- All the conditions are same as those of the MCA, except the initial condition, i.e., amount of FPs leak into RCV is assumed to be:
  - Iodine (I): 50%
  - Rare gas: 100%

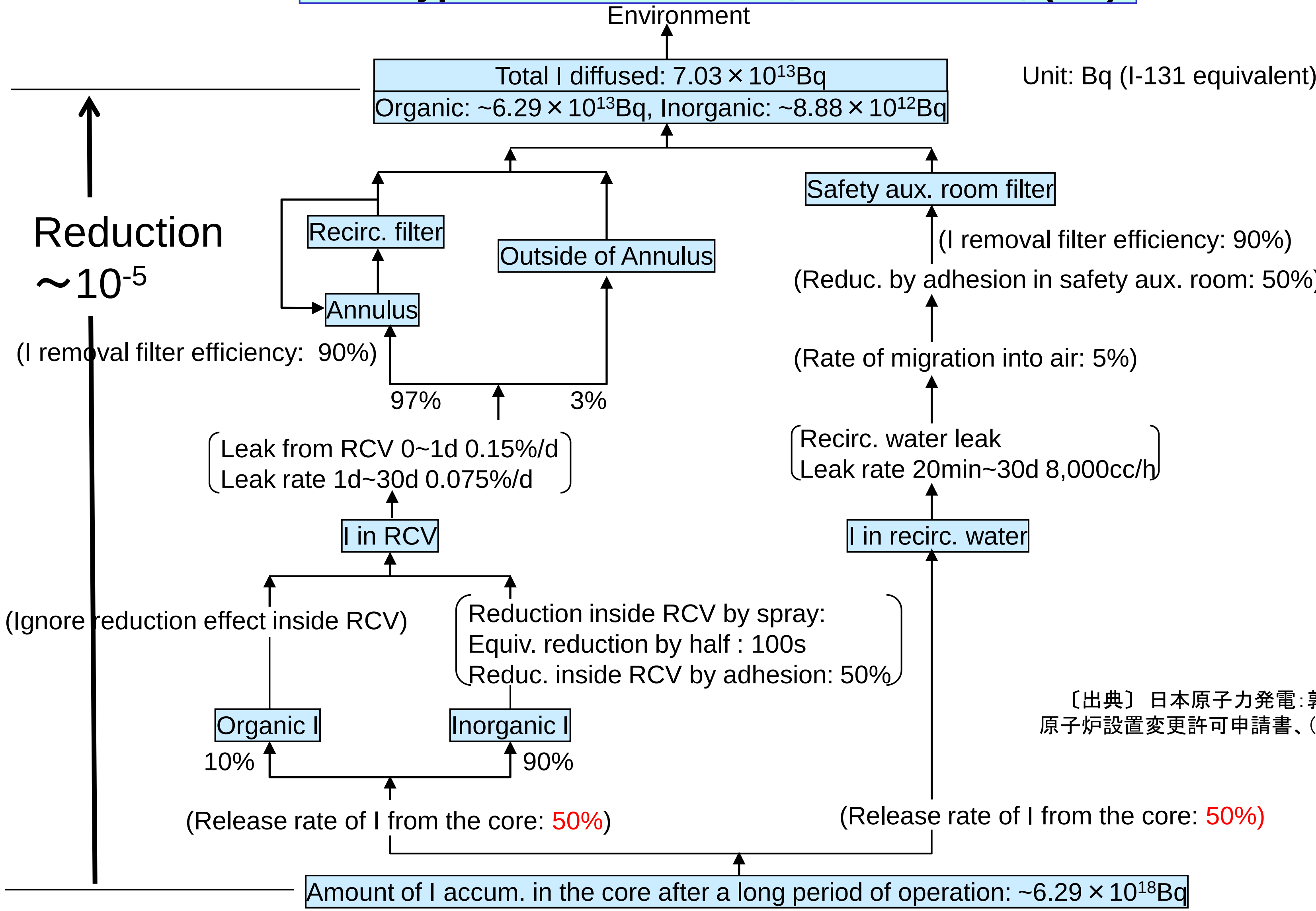
Results of Analysis/Released FPs

- Processes of leak to the environment are such as shown in Fig.1-13.

  Calculated amount of iodine and rare gas leaked to the environment is:
  - Iodine : About 7.03E13Bq
  - Rare Gas : About 5.92E15Bq

1.2.3 Examples of Site Evaluation of LWR

Ex.2 Hypothetical Accident, PWR-LOCA, (3/4)



〔出典〕 日本原子力発電:敦賀発電所
原子炉設置変更許可申請書、(昭和55年8月)

Fig. 1-13 Process of Iodine Leak to Environment during LOCA (Hypothetical Accident)

1.2.3 Examples of Site Evaluation of LWR

Ex.2 Hypothetical Accident, PWR-LOCA, (2/4)

Results of Analysis (Continued)/Radiological Consequences

Maximum exposed dose on the outside of the reactor site boundary is

- About 0.15Sv for thyroid (of adult)
- About 0.01Sv for whole body due to external γ-ray
- About 690men-Sv for the accumulated exposed dose of the whole body (for the estimated future population)

They are sufficiently low in compared to the allowable upper limit:

3Sv for thyroid (of adult),

0.25Sv for the whole body,

20,000men-Sv for the accumulated exposed dose for the whole body.

2. Safety Evaluation of FBR

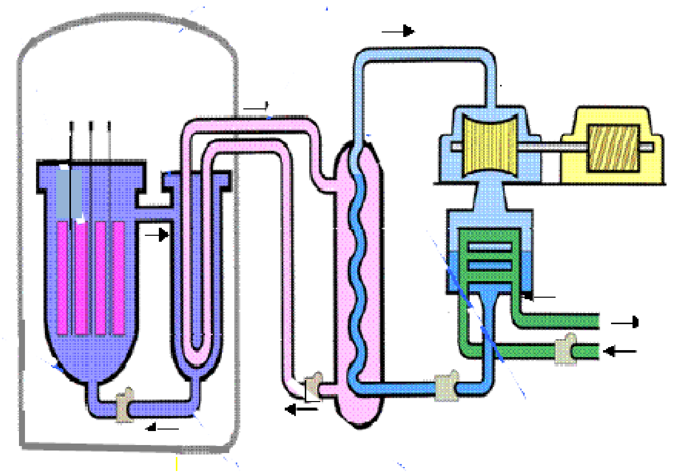
Guideline: "Philosophy in Safety Evaluation of Fast Breeder Reactors"
(prepared by NSC in Nov. 1980, latest amendment in Mar. 2001)

To be refered : "Review Guide for Safety Evaluation of LWR facilities"

2.1 Safety Design Evaluation of FBR

2.1.1 Scope of Evaluation

(1) **Design basis events (DBE):**
- abnormal transient during opera
- accident



(2) **Beyond DBE (BDBE):**
- technically unlikely accident
  To confirm the release of
  large amount radioactive mater
  is within the regulatory requirements.

2.1 Safety Design Evaluation of FBR

2.1.2 Events selected for Evaluation (1/2)

Abnormal Transient During Operation (DBE)

1) Abnormal change in reactivity or power distribution
- Abnormal CR withdrawal from a sub-critical condition
- Abnormal CR withdrawal during a power operation
- etc.

2) Abnormal change in heat generation or heat removal of the core
- Partial loss of primary flow
- Increase of primary flow
- Loss of off-site power (LOSP)
- Partial loss of secondary flow
- Increase of secondary flow
- Loss of feed water flow
- Increase of feed water flow
- etc.

LOSP: will be discussed later

3) Chemical reaction of sodium
- Small leak in a SG heat transfer tube

2.1 Safety Design Evaluation of FBR

2.1.2 Events selected for Evaluation (2/2)

Accident (DBE), its Examples:

1) Increase in core reactivity

Rapid CR withdrawal, Fuel slumping, Passage of bubble

2) Loss of core coolant flow

Coolant flow blockage, PHTS or SHTS pump stick,

Main Feed-water pump stick, Primary coolant leak (PCL),

PCL: will be discussed later

3) Chemical reaction of sodium

primary sodium leak, sodium-water reaction in SG

Technically Unlikely Events (Beyond DBE), their Examples

LPB: will be discussed later

1) Local Fuel Faults

Local overheat, channel blockage

ULOF:will be discussed later

2) Large PHTS Pipe Break (LPB)

3) Anticipated Transient Without Scram (ATWS)

- Unprotected loss of PHTS flow (ULOF)

2.1 Safety Design Evaluation of FBR

2.1.3 Requirements for Evaluation (1/2)

Abnormal transient during operation

Should be:

no fuel failure,

no radioactive materials release to the environment ,

plant should be ready for restart after the necessary restoration

Criteria

1) Cladding mid-wall temp. < 830˚C (to avoid pin rapture)

2) Sodium temp. in core < Boiling point

3) Fuel temp. < Melting point

4) <u>Reactor coolant boundary</u> < Either value below:

(a) 600˚C

(b) 1.4 times maximum design temp.

2.1 Safety Design Evaluation of FBR

2.1.3 Requirements for Evaluation (2/2)

Accidents

Should be:

no core-melt, no marked core failure, no following abnormal conditions, controlled radioactivity release to environment.

Criteria:

・Reactor coolant boundary < Either value bellow:

650˚C & 1.6 times maximum design press.

・Temp. & Press. of the RCV boundary

< 150˚C (max. design temp.), and

< 0.5kg/cm²G (max. design press.)

・Radiation exposure to the public in site periphery < 5mSv

Technically unlikely accident

Radiological consequences should be below the allowable limits

given by the guide.

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Transient, Loss of off-site power (LOSP) , (1/4)

Initiating Event and Sequence:

At full reactor operation,

due to partial or total off-site power loss,

PHTS and SHTS pumps loss power,

thus coolant flows decrease,

resulting in <u>difficulty of core cooling</u>.

<u>reactor is shut down automatically</u> by the safety protection system,

emergency diesel generators start up,

ACS starts up to remove <u>decay heat</u>

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Transient, Loss of off-site power (LOSP) , (2/4)

Preventive measures

- reactor is automatically shut down by "voltage of station-service bus- low" signal.
- Reactor is also automatically shut down by
  "primary pump speed- low",
  "primary coolant flow- low",
  "secondary pump speed- low", or
  "secondary coolant flow- low" signals.
- Each of 3 ACSs has capacity to remove decay
- Pump pony motor operation (**Decay Heat Removal mode operation**)
  in PHTS and SHTS secure 4% of rated core flow even with a single loop.

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Transient, Loss of off-site power (LOSP) , (3/4)

Results of Analysis

- After 0.9sec, reactor is automatically shut down.
- Coolant flows of PHTS and SHTS coast down, and 1ry and 2dry pumps shift to a low speed pony motor operation to ensure 7% of rated flow.
- Sodium temperature at RV exit momentarily increased to 540 ℃.
- Sodium temperature at RV inlet: up to 430 ℃.

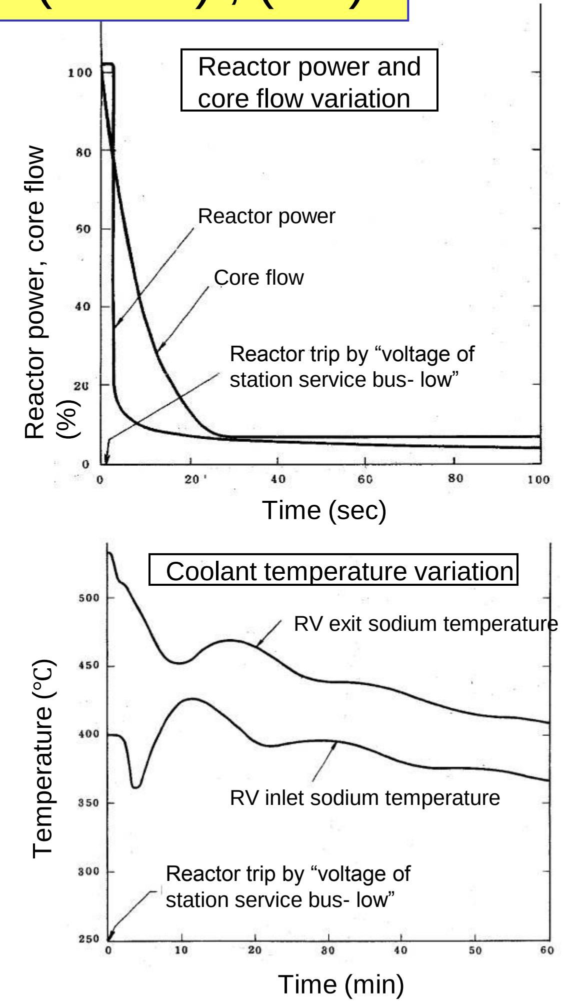


Fig.2-1 Loss of off-site power

〔出典〕 高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Transient, Loss of off-site power (LOSP) , (4/4)

Results of Analysis (cont'd)

- Max. cladding mid-wall temp.:
  730℃ < burst
- Maxi. sodium temp.:
  720℃ < boiling point
- Max. fuel temp.:
  < melting point

Above results indicate:

fuel is secured,

reactor coolant boundary is secured.

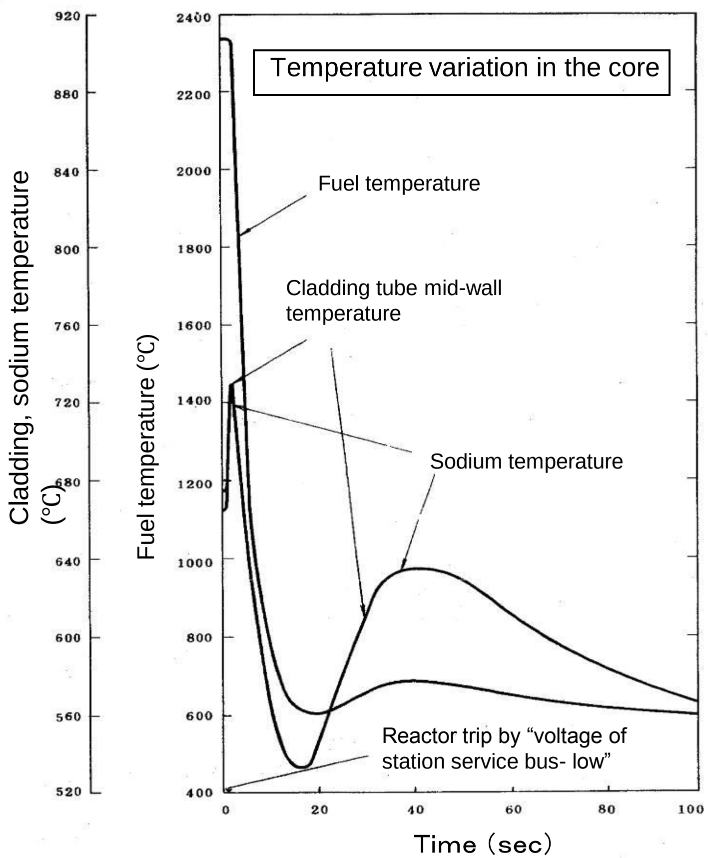


Fig.2-2 Loss of off-site power

〔出典〕 高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Accident: Primary coolant leak (PCL), (1/5)

Initiating Event and Sequence:

- During power operation, sodium leaks from PHTS piping crack whose size is equal to a **Double Ended Guillotine Break** of branch piping.
- Location of leak is RV inlet that give the maximum leak rate.
- Loss of off-site power occurs simultaneously.
- Reactor automatically shut-down by “RV sodium level– low” signal.
- Increases in cell and liner temperatures in PHTS by leaked sodium.
- Cell pressure and temperature are released to RCV together with sodium aerosol.
- RCV pressure and temperature increase which leads to sodium aerosol leak into the environment.

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Accident: Primary coolant leak (PCL), (2/5)

Mitigation measures

・Guard Vessel, GV, is provided to ensure sodium level in RV well above core and above RV exit nozzle, thus core can be cooled by the primary coolant circulation.

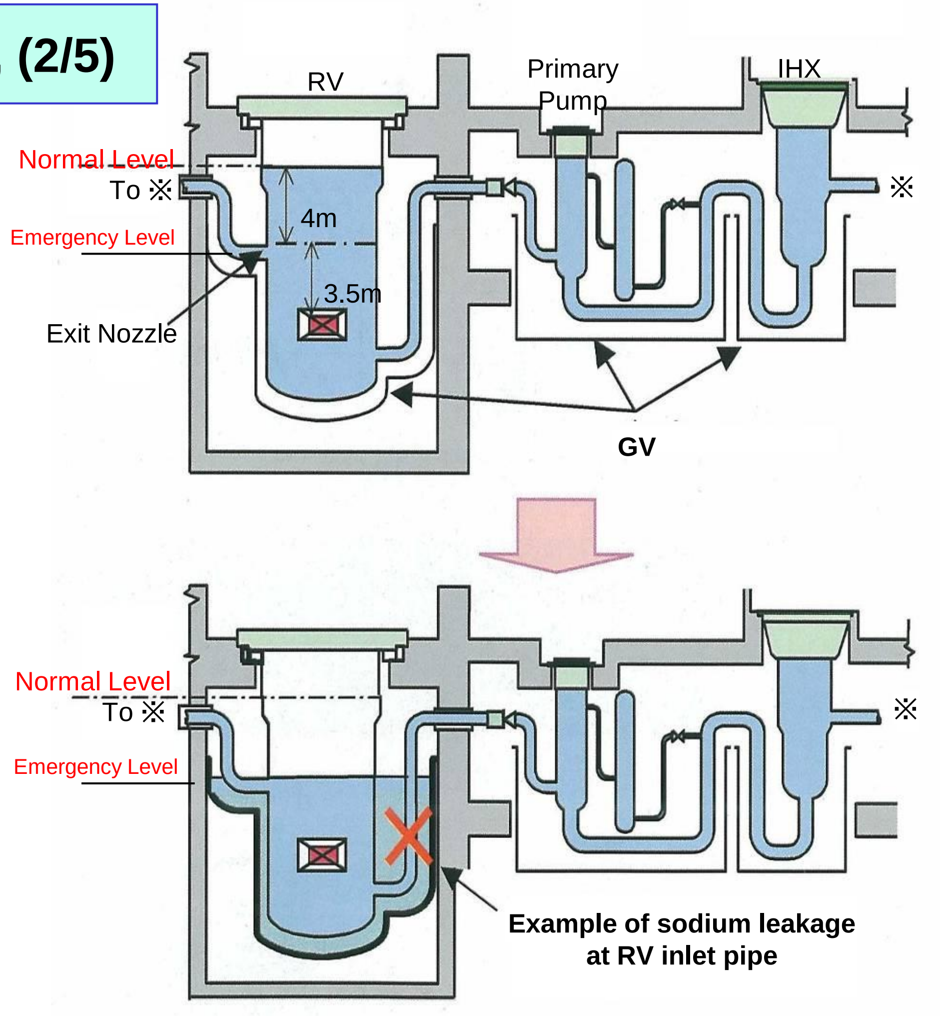


Fig.2-3 Preservation of core cool-ability by the GV

〔出典〕 高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Accident: Primary coolant leak (PCL), (3/5)

Results of Analysis (cont.)

- Reactor shutdown at 190sec.
- Sodium leak-rate: 80kg/s initially (less 2% of PHTS rated flow), and 34kg/s after DHR operation.
- core flow-rate by DHR operation: settles at 6% of rated flow-rate at 230sec after the accident.
- Decay heat of core is 4% of rated reactor power.

→ <u>Core cooling is secured.</u>

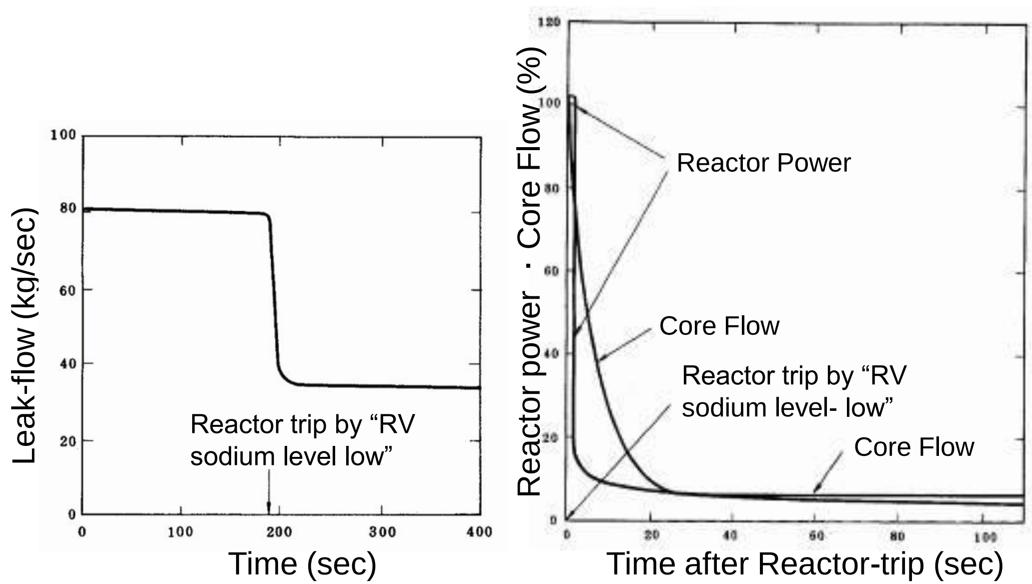


Fig.2-4 Primary coolant leak accident core cool-ability

〔出典〕高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Accident: Primary coolant leak (PCL), (4/5)

Results of Analysis (cont.)

- Max. sodium temperature at reactor exit: 540˚C (inlet:440˚C)
- Cladding temp.: 740˚C < burst temp. (830˚C)
- Max. sodium temp. in core: 730˚C < boiling point
- Max. fuel temp.: almost same as its initial temperature
- Sodium level in RV:secured above the emergency level by GV.

→ <u>Core cooling capacity is adequate.</u>

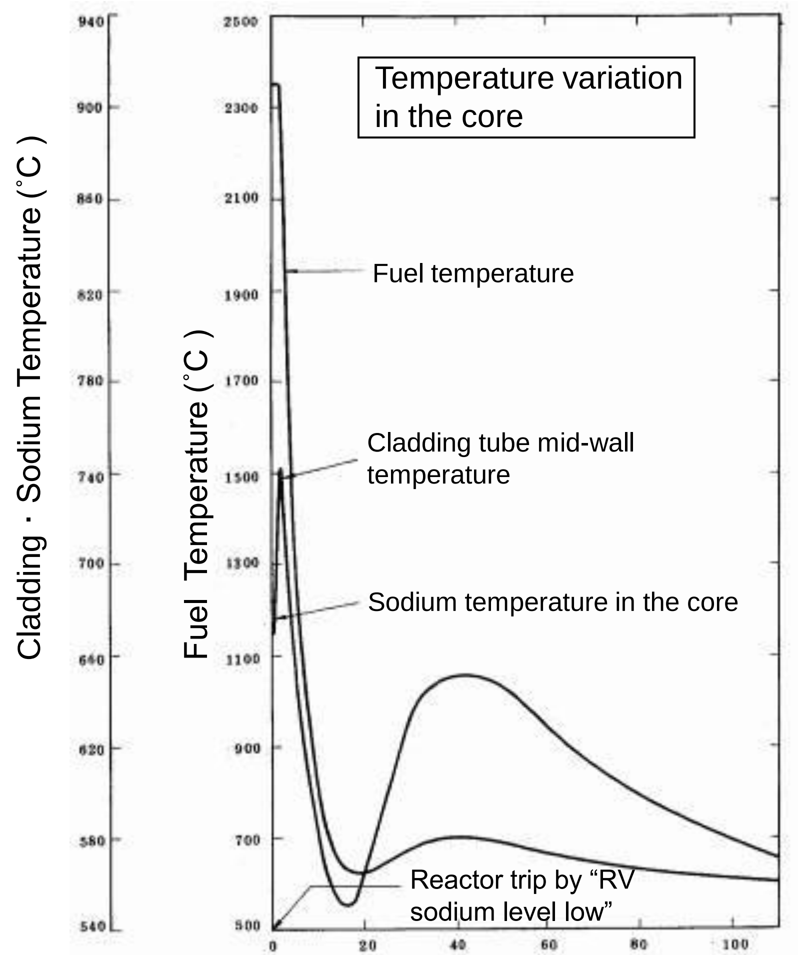


Fig.2-5 Primary coolant leak accident – core cool-ability

〔出典〕 高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Accident: Primary coolant leak (PCL), (5/5)

Results of Analysis (cont.)/Radiological Consequences:

- FPs released into the environment:
  - Iodine : 6.3×10^{8} Bq
  - Rare gas : 3.3×10^{12} Bq
- Maximum exposed dose on the outside of the reactor site boundary:
  - Exposed dose of infant thyroid : 0.0013 mSv < 1.5 Sv ※
  - Exposed dose of the whole body by **γ**- ray: 0.026 mSv < 0.25 Sv ※

(Note) Results are same for both hot-leg and cold-leg pipe break

※ allowable values given by “Guidelines for reactor site criteria”.

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

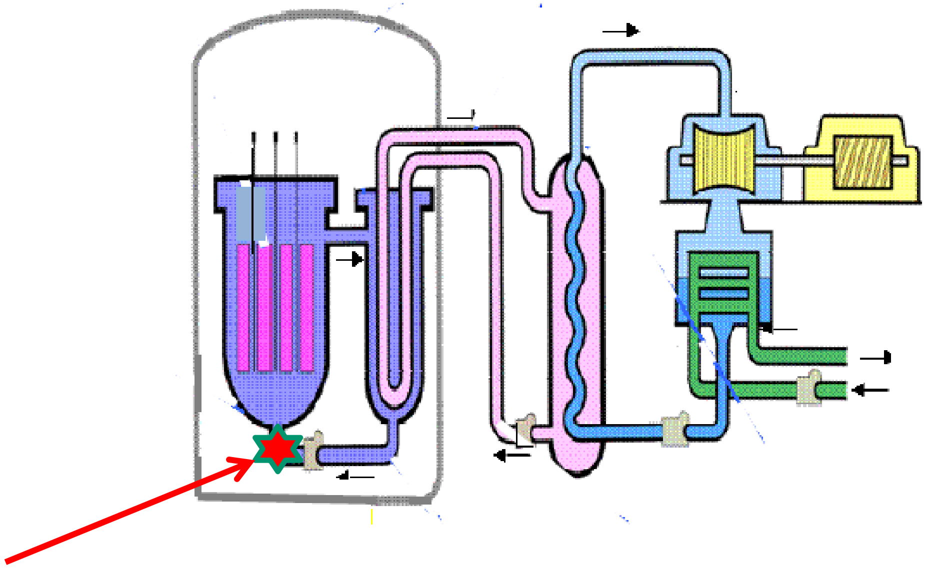
Beyond DBE, Large PHTS Pipe Break (LPB), (1/4)

Initiating Event and Sequence:

During a reactor full power operation,

a Guillotine pipe break occurs near the RV inlet,

thus a large amount of primary sodium leakage occurs.



2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Beyond DBE, Large PHTS Pipe Break (LPB), (2/4)

Results of Analysis

Assumption: Pipe break at RV inlet

Core cooling capacity

- Core flow: flow coasts down to 5% .

  At 70sec, pump switching to DHR operation, the core flow settles at 8%.
- RV exit/inlet temp.: increase up to 570˚C/ 450˚C, respectively.
- Core sodium max. temp.: 990˚C (Sodium boiling for short time)
- Cladding mid-wall temp.: 990˚C< melting point
- Fuel max. temp.: 2,390˚C< 2,650˚C

→ damaged fuel: ~3% of all core fuels

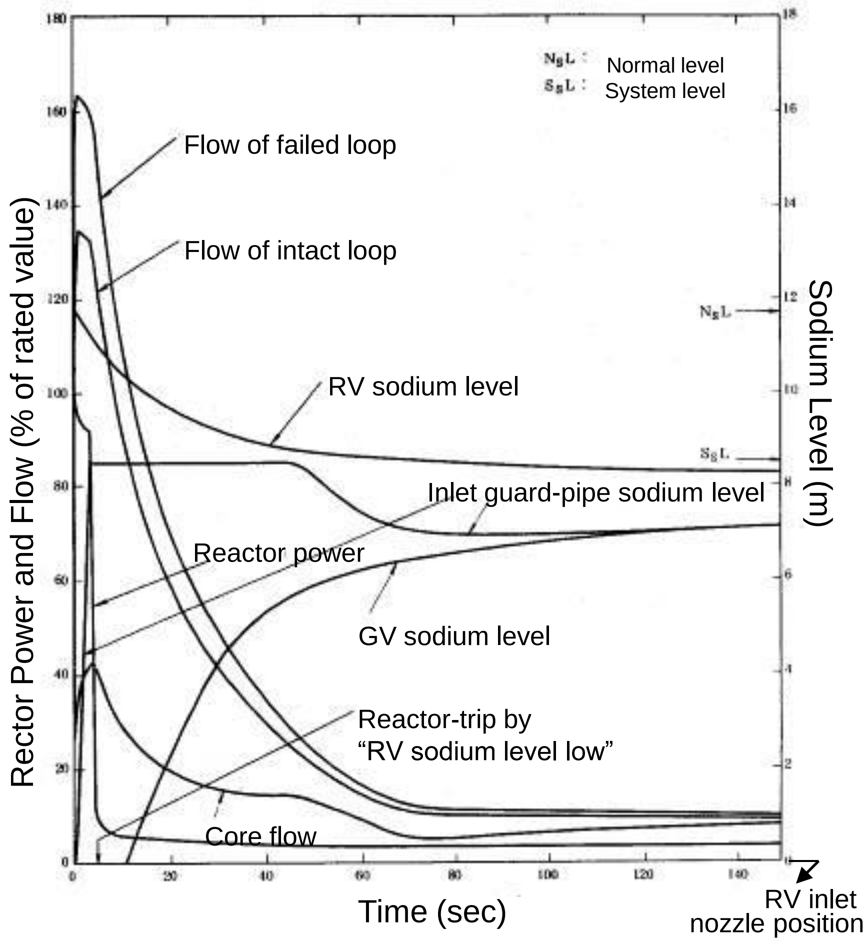


Fig. 2-6 Large Pipe Break in PHTS (Beyond DBE)

〔出典〕高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Beyond DBE, Large PHTS Pipe Break (LPB), (3/4)

Results of Analysis (cont.)

Thermal effect of leaked sodium

- Max. temp. of PHTS cell liner: 510˚C
  < 530˚C (design temp.)
- Inner pressure of RCV: up to 0.02kg/cm^2
  <0.5kg/cm^2 design press.
- Ambient temp. in RCV: increases only slightly

→ RCV integrity is secured

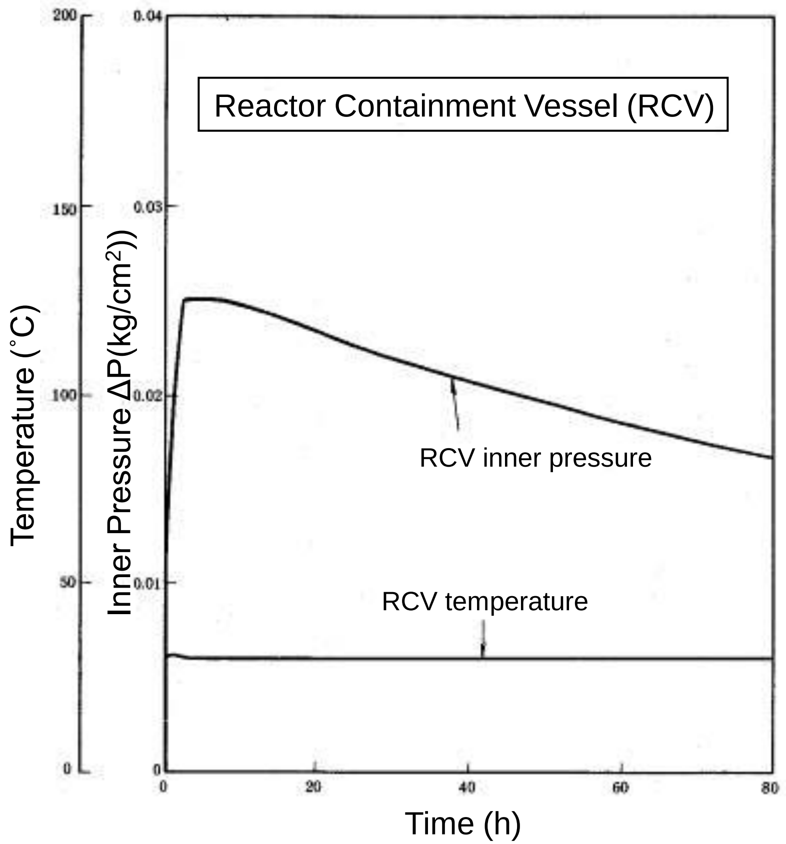


Fig. 2-7 Large Pipe Break in PHTS (Beyond DBE)

〔出典〕高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Beyond DBE, Large PHTS Pipe Break (LPB), (4/4)

Results of Analysis (cont.)

Radiological Consequences

Assumptions for evaluation

- FPs released in the PHTS cell:

  Rare gas ･･･ 10% of that in all fuel pellet-clad gap

  Iodine ･･････10% of that in all fuel pellet-clad gap

Results:

Exposed dose from FPs released to the environment is too small to be concerned.

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Beyond DBE, Unprotected Loss of Flow (ULOF), (1/3)

Initiating Event and Sequence:

- At power operation, Loss of Off-site Power occurs, thus, PHTS and SHTS pumps trip simultaneously. →Loss of Flow
- CR-control is also lost. → Reactor trip failure
- Sodium boiling, cladding melting, fuel relocation and slumping are followed by prompt critical.
- Core expands by the critical energy and thus becomes sub-critical.
- RV inflates by prompt critical energy but not breaches.
- No breach of PHTS components nor piping.
- Flow path is secured in PHTS for natural circulation decay heat removal, and also heat removal capacity by SHTS and auxiliary cooling system are secured.

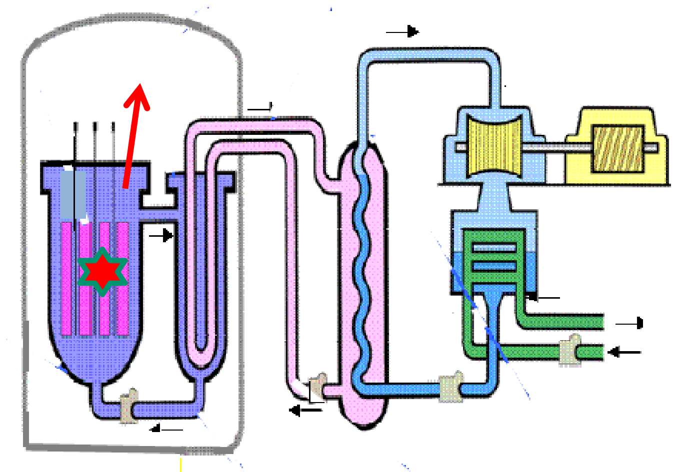
2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Beyond DBE, Unprotected Loss of Flow (ULOF), (2/3)

Results of Analysis

- As a result of sodium injection in RCV, temp./press. of RCV atmosphere initially increase to 140˚C/0.33kg/cm^{2}G that are both below design values, thereafter they start to decrease.
- Therefore, RCV integrity of is secured and release of radioactive material is suppressed.



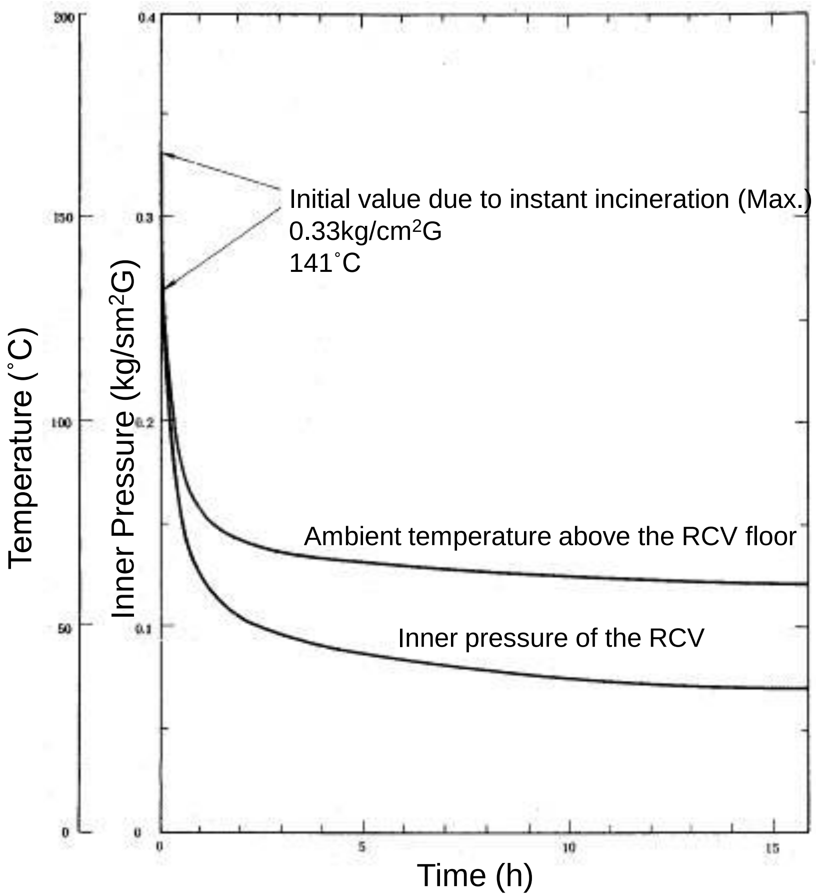


Fig. 2-8 Unprotected Loss of Flow (Beyond DBE)

〔出典〕高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.1 Safety Design Evaluation of FBR

2.1.4 Examples of Safety Design Evaluation of FBR

Beyond DBE, Unprotected Loss of Flow (ULOF), (3/3)

Results of Analysis (cont.)/Radiological Consequences

- Assumed FPs released in RCV (in % of that present in the reactor)

  Rare gas ･･･ 1%, Iodine ･･･ 1%, Pu ･･･ 0.1%

- A 95% of Iodine is released in the form of aerosol, thus, they plated out in RCV reducing their radiation.

- Max. exposure dose outside the site boundary

  1mSv for infant thyroid

  2.7mSv for adult thyroid

  0.69mSv for whole body by γ-ray

  0.00014Gy (lung), 0.00071Gy (bone surface), 0.00015Gy(liver) by Pu

2.2 Site Evaluation of FBR

Guideline:

"Review Guide for Nuclear Reactor Site Evaluation and Application Criteria".

Scope of Evaluation:

-Maximum Credible Accident (MCA)

- FPs in core that are assumed to release in RCV:
  10% of rare gas, 1% of iodine.

- Hypothetical Accident (HA)

- The same assessment as MCA is made for HAs, but with the assumption of a larger amount of FPs in core to release in RCV:
  100% of rare gas, 10% of iodine, 1% of Pu .

2.2.1 Example of Site Evaluation

Maximum Credible Accident, (1/2)

Assumed Conditions:

- A 90% of Iodine released is in the form of aerosol.
- Leak-rate from RCV to environment atmosphere: 1%/day
- Iodine removal efficiency of annulus filter: 95%
- The “Accident” continues for 30days

Results of Analysis

▪ FPs released into the environment :

Iodine: 240Ci, Rare gas: 47,000Ci

▪ Maximum exposed dose on the outside of the reactor site boundary:

1.8rem (<150rem) for of child thyroid:

0.15rem (<25rem) for whole body by γ-ray:

2.2.1 Example of Site Evaluation

Maximum Credible Accident, (2/2)

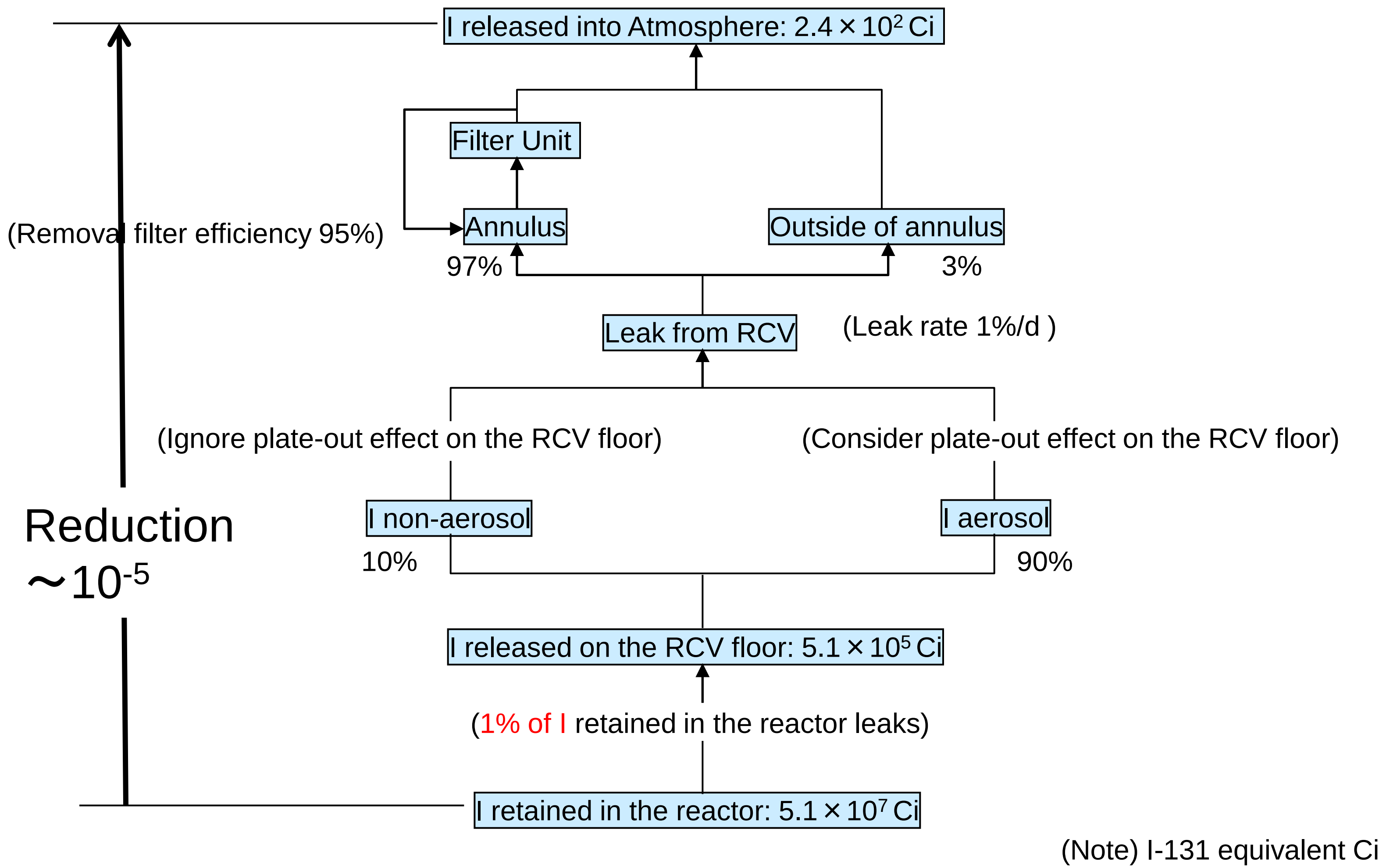


Fig. 2-9 Process of I Released to Atmosphere during Primary Coolant Leak Accident (MCA)

〔出典〕 高速増殖炉もんじゅ発電所原子炉設置許可申請書

2.2.2 Example of Site Evaluation

Hypothetical Accident, (1/2)

Assumed Conditions:

- Pu is in the form of aerosol and its filter removal efficiency is 95%.

Results of Analysis

- FPs released to the environment :

  Iodine: 2,300Ci, Rare gas: 470,000Ci, Pu: 51Ci
- Maximum exposed dose on the outside of the reactor site boundary:

  4.5rem (<300rem)* for thyroid, 1.4rem (<25rem)* for whole body

  Exposed dose by Pu:

  less than 0.9rad for lung, bone surface, and liver (<12rad) *
- Accumulated exposed dose for the whole body:

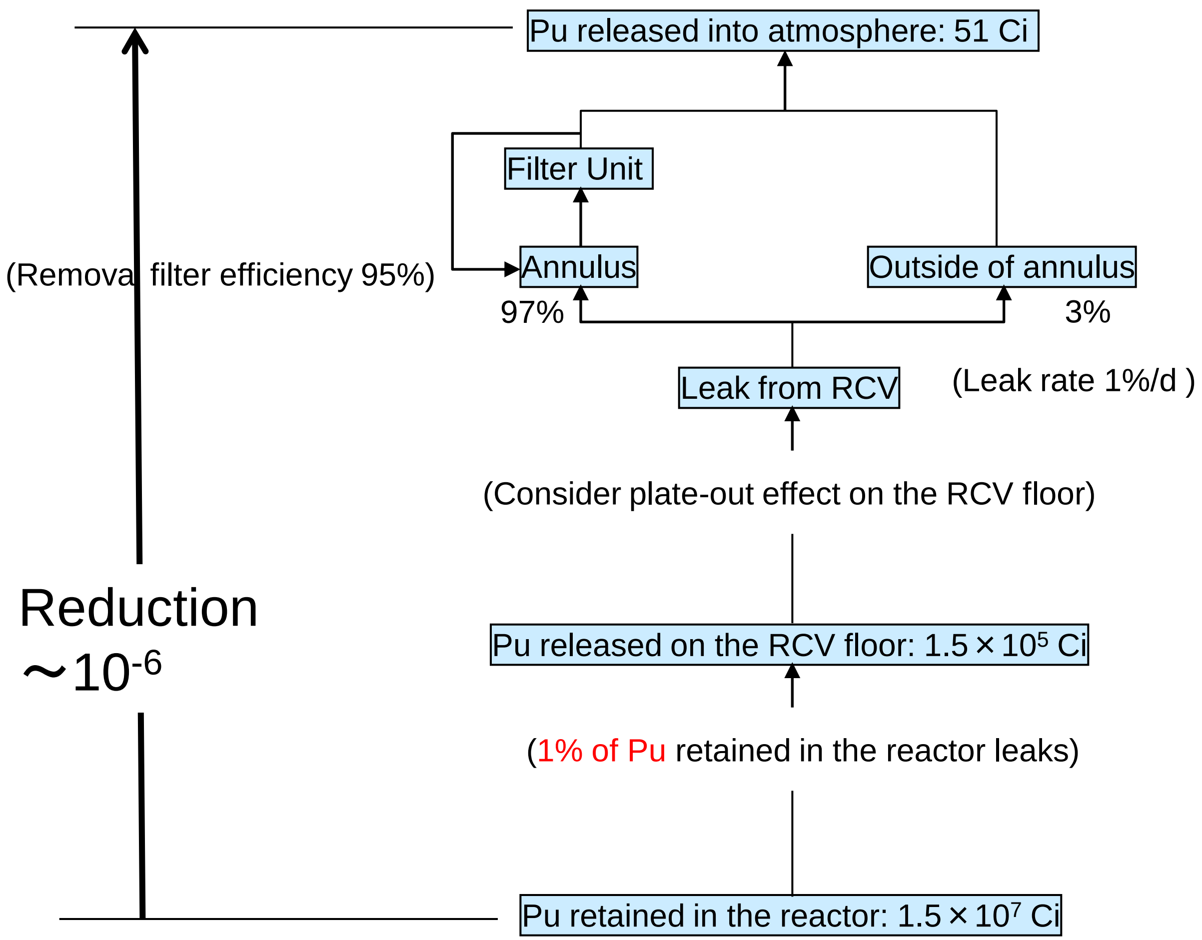
  130,000man.rem (<200,000man.rem)*,

.

Note: * is allowable values given by Guidelines.

2.2.2 Example of Site Evaluation

Hypothetical Accident, (2/2)



(Note) in terms of γ-ray 0.5MeV

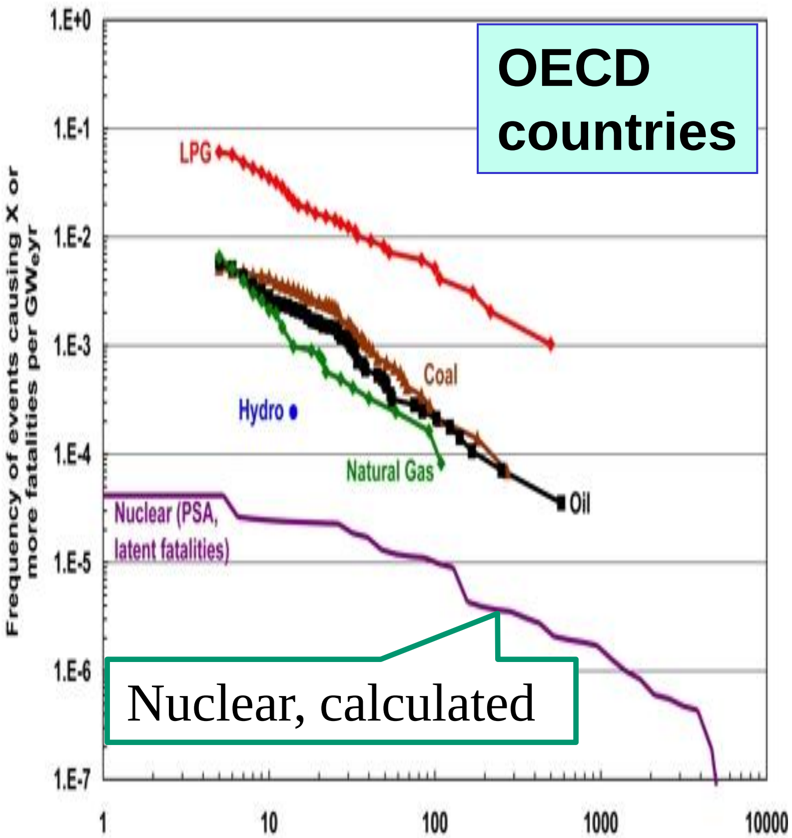
Fig. 2-10 Process of Pu Released to the Environment during Hypothetical Accident

〔出典〕 高速増殖炉もんじゅ発電所原子炉設置許可申請書

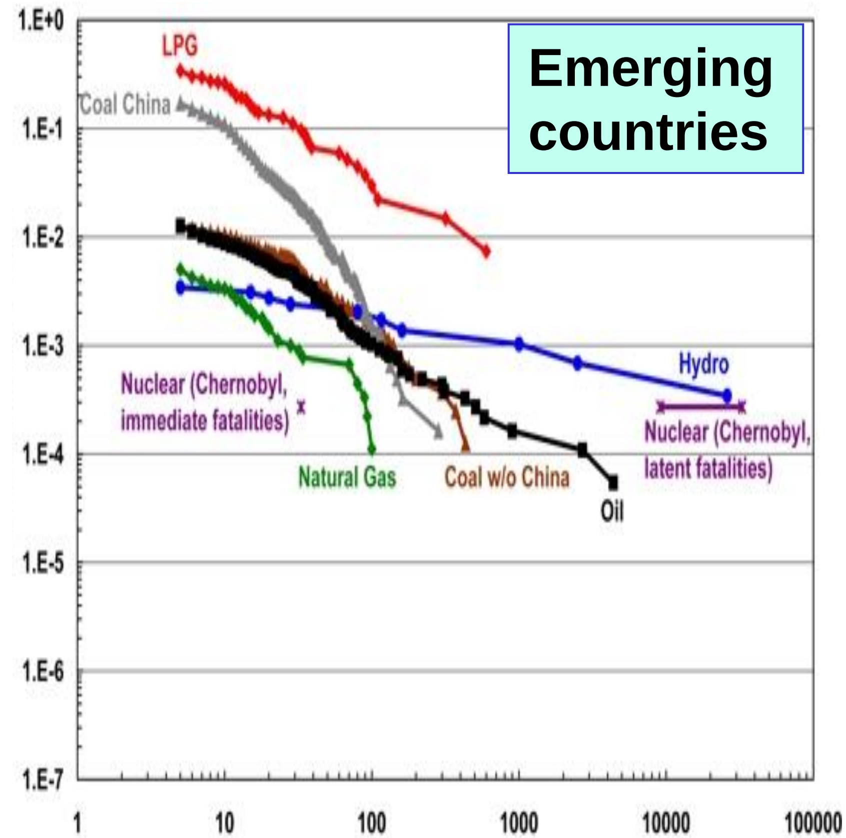
Risk of Nuclear Energy

- far lower than Risk of LPG, Coal, Oil, Hydro, and Natural Gas -

No public fatality (only fatalities of plant personnel)	Lower Risk even Chernobyl is included



Fatalities, X



Fatalities, X

OECD/NEA6862, 2010 by Swiss Paul Scherrer Institute

Between 1969～2000. Higher risk with LPG, coal and oil.