Reactor Plant Safety Course FY2010
Winter Course

Application of Installation Permission of Nuclear Power Plant (NPP)

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6. ***Comparison with Standard Review Plan of Safety Analysis Report for NPP by U.S.NRC***

(Appendix)

Preliminary Safety Analysis Report (PSAR)

(Standard Review Plan for the Review of Safety Analysis Reports for NPP: LWR Edition)

1. Positioning of Application of Installation Permission in the Law

System of Law Concerning Safe Regulation of NPP

Atomic Fundamental Law

- **Objective**

  Energy resources are secured by progressing research and development and utilization of nuclear.

- **Basic Policy**

  Nuclear development and use is restricted to the purpose of peace and its results should be opened and should be contributed to international cooperation spontaneously.

Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors

Main Regulation Matters

- **Installation Permission of NPP**
- **Permission of Safety Operational Program**
- **Security Inspection**
- **Permission of Regulation regarding Physical Protection for Nuclear Materials**

Positioning of Application of Installation Permission in Construction Process of NPP (1/2)

Legal Procedure Flow	Non-legal Procedure Flow
Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors	**Selection of Candidate Site**
↓	↓
Regulations Concerning Installation and Operation of Reactors	Getting of Consent for Investigation from Local Residents
↓	
Various Environmental Surveys	
↓	
Environmental Examination and Permission by government	**Environmental Examination and Approval by Prefecture**
↓	
Application of Installation Permission of NPP to METI **Safety Assessment** by Screening Committee	Open Hearing by Participating of Ordinary Peoples including Habitants

Cont. to next page

Positioning of Application of Installation Permission in Construction Process of NPP (2/2)

Legal Procedure Flow	Non-legal Procedure Flow
Cont. from front page	*NISA: Nuclear and Industrial Safety Agency
Getting of Installation Permission of NPP from METI (NISA*)	Remark)
	Getting of Consent from concerned Local Governor and Mayer
Each Approval Application of Construction Method for the Concerned System to NISA	Remark) Although getting of consent from the local governor and Mayer is not necessary as legal procedure, it is essential actually. Additionally, a local agreement is concluded prior to start of operation.
Getting of Construction Permission from NISA via Examination	
Start of Construction	

Main Events until Starting Construction (for Monju)

1970	～	1976	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987
△/4 Candidate Site's Selection		△/7 Start of Environmental Surveys	△/8 Finish of Environmental Surveys	△/3 Start of Environmental Examination	△/2 Modification of Some Parts of Plan	△/9 Finish of Environmental Examination △/12 Application of Installation Permission		△/5 Cabinet Construction Consent △/7 Open Hearing	△/5 Getting of Installation permission	△/12 Approval Application of Construction Method	△/10 Start of Construction	△/2 Start of Basement Concrete △/7 Start of C/V Construction	△/9 Finish Approval Application of Construction Method

Environmental Surveys & Examination for at. 4 years

Safety Assessment for at. 3years

Construction

(for at. 6 years)

Source: JNC TN4440 2002-012「もんじゅ建設の歩みNo.1」ー用地選定から建設着工までー、p3、核燃料サイクル開発機構国際技術センター、2002年9月

2. Relation between Safety Assessment and Installation Permission of NPP

Installation Permission of NPP

Application

Installation of NPP

Permission

Safety Assessment

(Assessing whether application contents has satisfied the Safety and Permission Criterion)

Check Application Contends with **Permission Criterion**

Verdict

Consultation

◈ Atomic Committee

◈ Atomic Safety Committee

Siting・Design

1. Siting Condition

2. Safety Design for Basic Design

Assessment

3. Incident Evaluation

4. Accident Evaluation

5. Siting Evaluation (Safety Assessment)

- Major Accident
- Hypothetical Accident

Various Screening Committees

Screening Standard of Safety Assessment consisting of a total of 59 items

Main Flow until Getting Permission of Installation of NPP

Application of Installation Permission of NPP to METI (or MEXT)

↓

Implementation of Safety Assessment by Screening Committees organized by METI

- Screening Committee for Core Design
- Screening Committee for Structure Design
- Screening Committee for Seismic Design
- Screening Committee for Sodium Handling (FBR)

↓

Final Check by "Atomic Committee" and "Atomic Safety Committee"

Check Application Contents with Permission Criterion of Installation Permission of NPP (Non-Technical Matter)

↓

Pass the Final Check

↓

Giving Installation Permission of NPP from METI (or MEXT)

Permission Criterion for Installation of NPP

The concerned minister (METI or MEXT) has to <u>interview the opinion of both committees</u> before giving the permission to the plant owner.

Atomic Committee	Peaceful Use	There is no possibility that a reactor may be used besides the purpose of peace.
Atomic Committee	Planned Implementation	There is no possibility that giving permission never causes the trouble to the planned implementation of nuclear development and use.
Atomic Committee	Foundation	A plant owner has enough finance to be constructed a rector plant.
Atomic Safety Committee	Technical Capability	A plant owner has enough technical capability for construction and operation of a reactor plant.
Atomic Safety Committee	Prevention of Disaster	To prevent occurring disaster, a reactor plant must be taken suitable measures.

Screening Standard of Safety Assessment (59 Items)

System for Screening	Main Screening Contents	Number
1. Reactor Facility Whole	Conformity Standard, Natural Phenomenon, Fire, etc.	10 Items
2. Reactor Core and Reactor Shutdown System	Self Regulating Characteristics, Shutdown Capability, etc.	8 Items
3. Heat Removal System	Soundness of Pressure Boundary, ECCS, etc.	9 Items
4. Reactor Containment Vessel	Isolation Function, Ventilation Control, etc	6 Items
5. Reactor Protection System	Redundancy, Diversity, etc.	7 Items
6. Control Room and Emergency System	Emergency Reactor Shutdown Function from Outside of Control Room, etc.	6 Items
7. Instrumentation & Control System and Power System	Instrumentation & Control System and Power System	2 Items
8. Fuel Handling System	Fresh/Spent Fuel Handling, Critical Prevention, etc.	3 Items
9. Radioactive Waste Disposal System	Disposal of Gas, Liquid and Solid Radioactive Wastes	4 Items
10. Radiation Control	Radiation Protection, Monitoring, etc.	4 Items

Example of Safety Design Assessment Standard (Extract)

<Extract 1/2>

V) Reactor Core and Shutdown System

●Standard-11: Core Design

1. Core design should be a design which does not exceed the permissible design limit of fuel during normal and abnormal operations by functions of heat transfer system, reactor shutdown system, instrumentation control and safety protection system.
2. Core subassembly should be a design which can secure cooling during all operation modes including a accident.

●Standard-13: Core Characteristic

A core should be a design which has an inherent safety (negative feedback characteristic) and also can control easily even if power oscillation occurs.

<Extract 2/2>

VI) Reactor Heat Transfer System

●Standard-19: Soundness of a Reactor Coolant Pressure Boundary

1. A reactor coolant pressure boundary should be a design which can secure its soundness during normal and abnormal operation modes.
2. As a general rule, the piping system connected to reactor heat transfer system should have isolation valves.

●Standard-25: Emergency Core Cooling System (ECCS)

1. ECCS should be a design which can prevent a serious damage of fuel for the accident of loss of coolant and moreover can restrict a reaction between fuel cladding metal and a water to a small quantity sufficiently.
2. ECCS should have redundancy or diversity and independency in order to perform its safety function completely even if an external power cannot be used in addition to assumption of the single failure of an equipment.
3. ECCS should be a design which can conduct test and inspection periodically and can carry out test and inspection of each system independently for assuring its soundness and redundancy.

3. Configuration of Application Form for Installation Permission of NPP

Application Form

- **Main Text**
  - 1) Name and address of the applicant, Representative name
  - 2) Purpose of using
  - 3) Reactor type, thermal output and number of the reactors
  - 4) Name of the reactors and address of installation
  - 5) Location, structure and equipment of the reactor facilities
  - 6) Construction Plan
  - 7) Kind of Nuclear Fuel Materials
  - 8) Method for disposing of spent fuel
- **Attachment Materials***
  - Attachment-1 ~ Attachment-11

*To **explain each item in detail technically**, **eleven attachment materials appended** separately.

Compartment between Main Text and Attachment Materials

- Main text

Although the technical specifications related to a core, fuel, heat transfer system, fuel handling system, instrumentation and control system, power supply, etc. are mentioned in the term of 5 in attachment material, they are **only summaries**.

- Attachment Materials

The detail technical specifications are explained in the attachment material of 6, 8 and 10 as the following.

- Attachment-6: **Environment, Climate** and **Earthquake**
- Attachment-8: **Safety Design (Basic Design)**
- Attachment-10: **Safety Evaluation**

Positioning of Attachment-6,-8 & -10 in Safety Assessment

Safety Assessment
(Assessing whether application contents has satisfied the Safety and Permission Criterion)

Attachment-6

Attachment-8

Attachment-10

Siting・Design

1. Siting Condition

2. Safety Design for Basic Design

Assessment

3. Incident Evaluation

4. Accident Evaluation

5. Siting Evaluation (Safety Assessment)

- Major Accident
- Hypothetical Accident

Reactor Plant Design:
1) Satge-1: Concept Design
2) Stage-2: **Basic Design**
3) Stage-3: Detailed Design
4) Satge-4: Manufacturing Design

Configuration of Attachment Materials

Attachment Materials

1) Explanation regarding "Purpose of Using Reactor"

2) Explanation regarding "Reactor Thermal Output"

3) Explanation regarding "Construction Fee and Financing Plan"

4) Explanation regarding "Acquisition Plan of Nuclear Fuel Materials"

5) Explanation regarding "Technical Capability" for Installation and Operation "

6) Explanation regarding **Situations in Site such as Climate, Foundation, Hydrology, Earthquake, Social Environment**, etc.

7) Explanation regarding "Map of Site within 20km and 5km"

8) Explanation regarding **Safety Design** for **Basic Design**

9) Explanation regarding "Radiation Exposure Management and Disposing Method of Radioactive Waste"

10) Explanation regarding Evaluation **for Influence of Incident, Accident, Major and Hypothetical Accidents**

11) Explanation regarding "Financial Situation"

4. Contents of Attachment Materials

Attachment-6: Situations in Site such as Climate, Foundation, Hydrology, Earthquake, Social Environment (1/4)

1. Site

1.1 Conspectus of site

1.2 Overview of others

2. Climate

2.1 Climate of Tsuruga Area

2.1.1.Geographic features and climate

2.1.2. Climate of four seasons

2.2 Usual Climate based on the Materials of the Nearest Weather Office

2.2.1 Situation of the place of the weather office

2.2.2 The reason for having chosen the weather office in Fukui and Tsuruga

2.2.3 General climate conspectus in the nearest weather office

2.3 Climate Observation in the Site

2.3.1 Situation of observation points

2.3.2 Observation items

2.3.3 Qualification of climate observation equipments

2.4 Observation Results at the Site

2.5 Weather Conditions for applying to Safety Analysis

2.5.1 Study of representative climate data during climate observation period

Attachment-6: Situations in Site such as Climate, Foundation, Hydrology, Earthquake, Social Environment (2/4)

2.5 Weather Conditions for applying to Safety Analysis (Continuation)

2.5.2 Effective height of emission source applied to calculating atmospheric diffusion

2.5.3 Weather conditions applied to exposure evaluation in normal operation

2.5.4 weather conditions applied to exposure evaluation in assumption accident

3. Foundation

3.1 Process of Inspection

3.1.1 Survey of circumference of site area

3.1.2 Survey of neighboring of site area

3.1.3 Survey of reactor installation place

3.2 Geology/Geological Structure of Circumference of Site Area

3.2.1 Survey contents

3.2.2 Survey results

3.3 Geography/Geology/Geological Structure of Neighboring of Site Area

3.3.1 Survey contents

3.3.2 Survey result

3.4 Geology/Geological Structure and Foundation of Reactor Installation Place

3.4.1 Survey contents

3.4.2 Survey results

3.4.3 Evaluation of results

3.4.4 Verification regarding geology survey

Attachment-6: Situations in Site such as Climate, Foundation, Hydrology, Earthquake, Social Environment (3/4)

4. Hydrology

4.1 Inland Water

4.2 Oceanographic Phenomenon

4.2.1 Stream regime/flow speed

4.2.2 Tidal level

4.2.3 Ocean wave

4.2.4 Sea temperature

4.2.5 Drift sand

4.3 Water Utilization Plan

4.3.1 Necessary amount of water

4.3.2 Condenser cooling water and auxiliary cooling water

5. Earthquake

5.1 **Past Earthquakes**

5.1.1 Activity of earthquake

5.1.2 Earthquake damage history

5.1.3 Statistical expected value of Intensity of earthquake motion

5.2 **Active Fault**

5.2.1 Active faults to be considered as the strongest earthquake

5.2.2 Active faults to be considered as the limiting earthquake

Attachment-6: Situations in Site such as Climate, Foundation, Hydrology, Earthquake, Social Environment (4/4)

5. Earthquake (cont.)

5.3 **Regional Geological Structure**

5.4 **Vibration Characteristic of Foundation**

5.4.1 Earthquake observation

5.4.2 Regular microscopic observation

5.5 **Standard Earthquake Vibration**

5.5.1 Earthquakes which should be taken into consideration

5.5.2 Earthquake vibration characteristic

5.5.3 Standard earthquake vibration

6. Social Environment

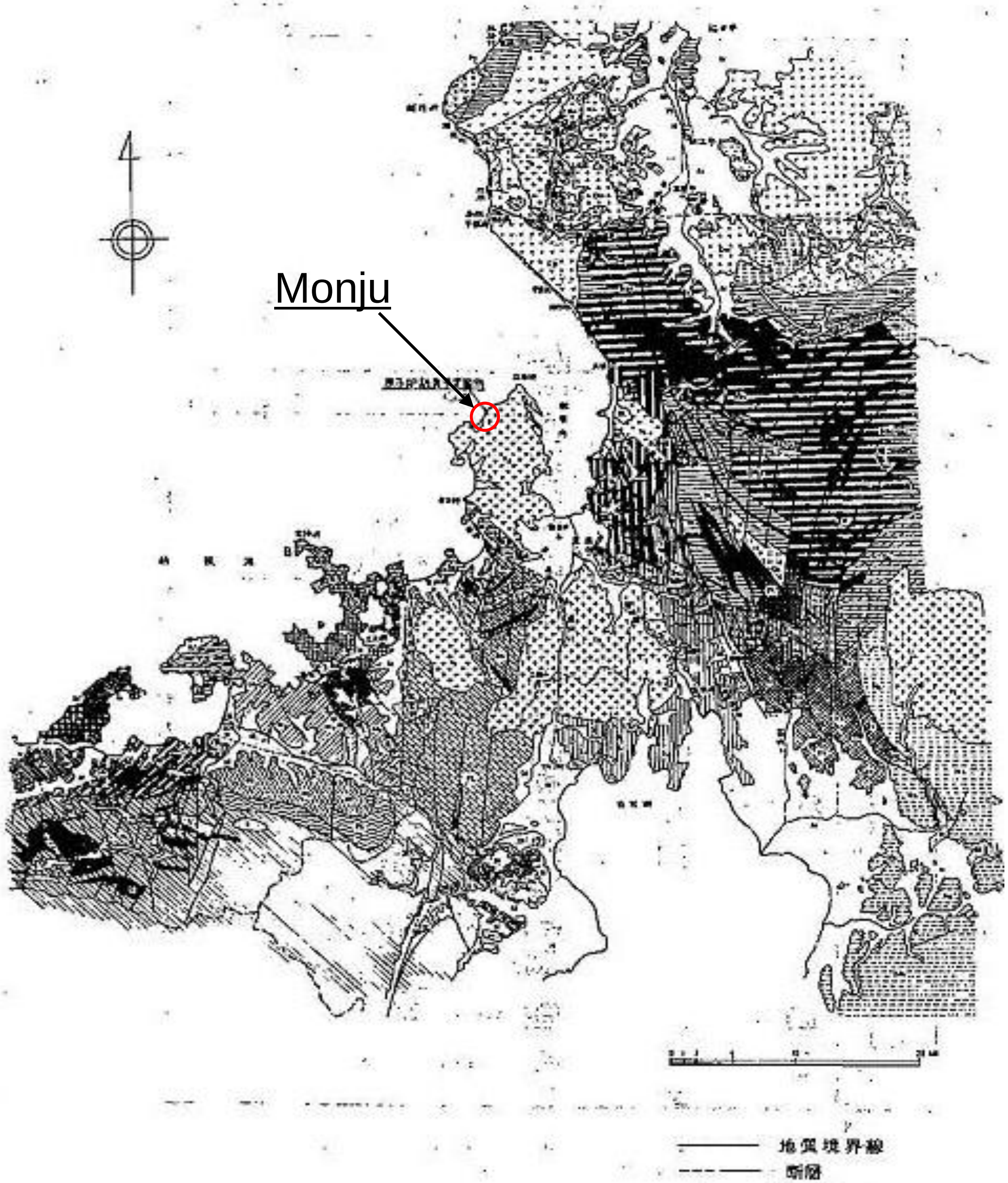
6.1 Population Distribution

6.2 Neighboring Settlement and Public Facilities

6.3 Industrial Activities

6.4 Transportation System

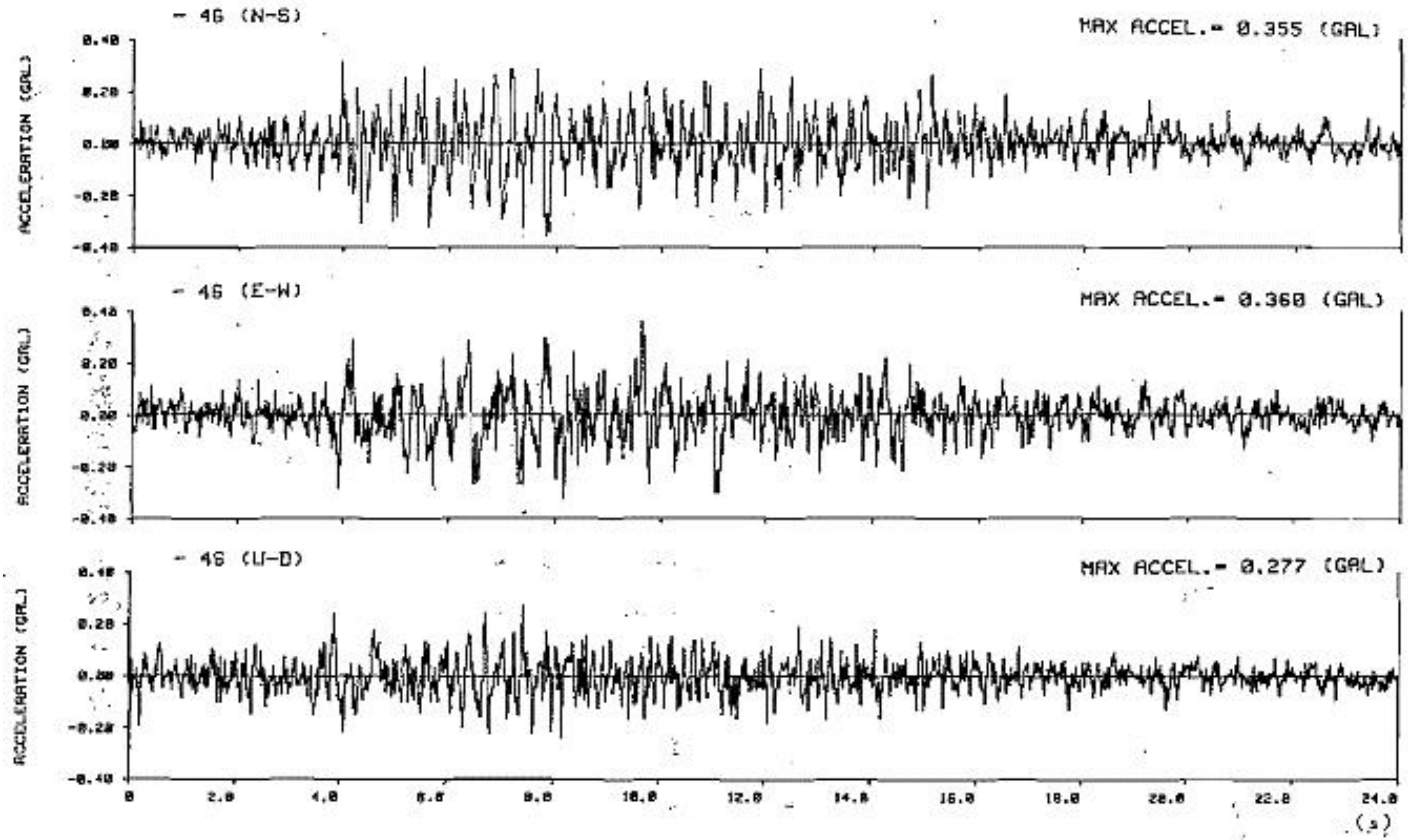
6.5 Development Plan



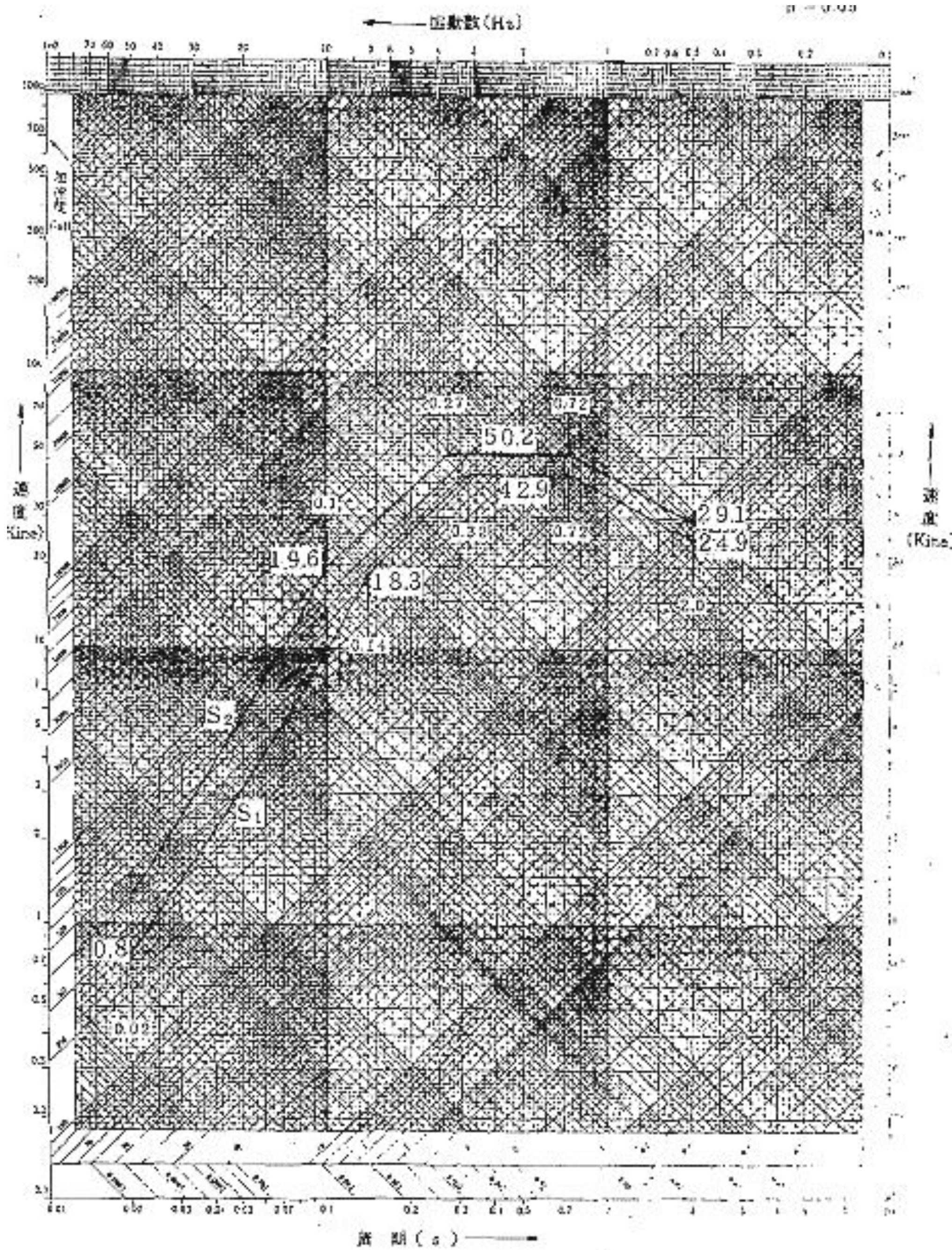
Sample attachment figure: Geologic map around a site (Monju)

垣見俊弘・他編．1978：「日本活断層図」地質調査所（部分拡大）

Sample attachment figure: Distribution map of active fault around a site (Monju)



Sample attachment figure: Seismic observation record at around a site (GL-46m)



Sample attachment figure: Response spectrum of standard earthquake motion (Monju)

Attachment-8: Explanation Regarding "Safety Design"

<Contents of Attachment-8> (1/25)

1. Safety Design

1.1 Safety Design Policy

1.1.1 Basic Policy of "Safety Design"

1.1.2 "Inherent Safety" of Reactor

1.1.3 Consideration on Safety in "Design and Manufacture of Reactor Plant Facilities"

1.1.4 Basic Policy of "Core Physics Design and Thermal Hydraulic Design"

(Sample Contents: Basic Policy of Core Physics Design)

- ✓A core is hexagon pillar form of which a diameter ratio is 0.52, and consists of a total of 198 fuel assemblies and 172 blanket fuel assemblies of hexagon form and so on.
- ✓A fuel region is divided into two parts of inner and outer sides, and the flat of power distribution is attained by rising the enrichment of an outer core.
- ✓Reactivity control is carried out by using control rods.
- ✓A reactor shutdown systems is composed of a main shutdown system which has both functions of reactivity control and emergency shutdown and a backup shutdown system for emergency shutdown.
- ✓Even if one control rod cannot insert, a reactor should design so that a reactor can be scrammed safely by remained another control rods and has enough negative reactivity for maintaining a sub-criticality situation of a reactor.
- ✓The maximum liner power density should be designed as less than 360w/cm so that fuel max. temperature never reach the melting point of a fuel pellet.
- ✓A reactor should design so that a power coefficient becomes minus due to an negative feed back characteristic.

<Contents of Attachment-8> (2/25)

1.1.5 Preventive Measures of "Radioactive Material's Diffusion"
1.1.6 Consideration on Design to "Sodium"
1.1.7 Basic Policy of "Instrumentation Control System's Design"
1.1.8 Basic Policy of "Engineered Safety System"
1.1.9 Consideration on Design to "Removal of Decay Heat"
1.1.10 Consideration on Design to "Fire"
1.1.11 Consideration on Design to "Loss of Power"
1.1.12 "Physical Separation"
1.1.13 Basic Policy of "Intensity Design"
1.1.14 Basic Policy of "Quality Assurance"

(Sample Contents: **Quality Assurance**)

- ✓ Organization, work assignment and responsibility are clarified, and quality assurance (QA) activity performs certainly.
- ✓ To assure that the QA activity of a plant maker is carried out correctly, their organization, capability, manual, etc. has to be checked beforehand, and the investigation by Monju should be done if necessary.
- ✓ Also, the external order articles by a plant maker is treated as the same treatment.
- ✓ Each stage on determination of specification, design, manufacture, installation, test and inspection, the statute, standard, requirement to apply should be approved after confirming by documentation check or on-the-spot inspection whether basic design is satisfied or not.
- ✓ About documentations, figures, specifications, record of QA, etc., treatment method and management manual are clearly and is kept surely.

<Contents of Attachment-8> (3/25)

1.2 Design for Fitness to Safety Design Policy

1.2.1 Definition of term
1.2.2 "Application Regulations and Standards"
1.2.3 Design for Conformity regarding "Reactor and Instrumentation Control System"
1.2.4 Design for Conformity regarding "Reactor Shutdown System, Reactivity Control System and Reactor Protection System"
1.2.5 Design for Conformity regarding "Reactor Coolant System"
1.2.6 Design for Conformity regarding "Containment Vessel System"
1.2.7 Design for Conformity regarding "Fuel Handling System and Waste Disposing System"
1.2.8 Design for Conformity regarding "Radiation Protection and Radiation Control Facility"

(Sample of design for fitness < Policy-50: Critical Prevention of Nuclear Fuel>)

- ✓ A storage rack for storing of spent fuels should taken enough interval of each assembly, and should design so that an effective multiplication factor (*keff*) is kept less than 0.95 even if fuels are received to the limit of capacity
- ✓ Also, a storage rack should be design with earthquake proof class of A in order to not damage and to not close fuel assemblies each other.
- ✓ A fresh fuel storage rack should fully taken interval of each assembly and should store a fresh fuel under the inert gas.
- ✓ A fresh fuel storage rack should be made arrangement which is not flooded, and should be designed so that a *keff* can be maintained under 0.95 even if filling with a water.
- ✓ Fuel handling system has to handle only one assembly at each time operation, and is made the design which prevents criticality.
- ✓ Furthermore, it is made the design so that a reactor can be maintained sufficiently sub-criticality when a control rod is withdrawn during fuel handling work.

<Contents of Attachment-8> (4/25)

<Contents of Attachment-8> (5/25)

3. Reactor and Core

3.1 Outline

3.2 Mechanical Design

3.2.1 Fuel

3.2.2 Structures in Reactor Vessel

3.2.3 Reactivity Control System

3.3 Core Physics Design

3.3.1 Outline

3.3.2 Design Policy

Core design should be designed so that a reactor satisfies the following conditions during the life time.

(1) Shutdown Margin r Reactivity

✓A reactor shutdown system should be a design which a reactor can be resulted in sub-criticality even if the control rod which has the maximum reactivity cannot be inserted due to its stick.

✓Therefore, a main shutdown system should be designed have a shutdown margin reactivity of 0.01⊿k/k or more even if one control rod which has the maximum reactivity cannot be inserted.

✓Moreover, even if it assumes that a main shutdown system doesn't work, a reactor should be maintained into sub-criticality under low temperature mode.

(2) Applying Ratio of Reactivity

✓ A maximum reactivity applying ratio should be made to less than 8x10-5 ⊿k/k in order to prevent occurring a reactor core damage due to the reactivity accident assumed.

<Contents of Attachment-8> (6/25)

3.3.2 Design Policy

(3) Excess Reactivity

✓An excess reactivity should be designed to compensate the following: reactivity change under from low temperature to the rated power operation mode; reactivity change by fuel burnup; operation margin.

✓This excess reactivity should be designed to control by reactor shutdown system including a reactor margin reactivity shown in the term of (1).

(4) Power Coefficient

✓A reactor should be designed to have a power coefficient which damps power fluctuation when a power level is changed.

✓Accordingly, the Doppler coefficient is negative as a reactor has negative feedback characteristic, and the power coefficient integrating, not only the Doppler also including the temperature coefficient of fuel, structure, coolant and core support structure, should be designed to become minus under the all operation modes.

(5) Power Distribution

✓The power distribution which exceeds a permissible design limit of fuel never happen under normal and abnormal operation conditions.

✓The power distribution is made flat in order to take out thermal power effectively under a normal operation mode.

✓Therefore, a reactor core is composed of two regions which is different plutonium enrichment, and the high plutonium enrichment fuels are loaded into outer core.

✓The maximum linear power density under the rated power should be designed less than 360W/cm.

<Contents of Attachment-8> (7/25)

3.3.3 Analysis Method

Nuclear calculation is performed based on a multi-group neutron diffusion theory and a transport theory.

(1) Group Constant

✓The group constant to be used is 26 group constant of ABBN type.

✓This group constant consists of an infinite dilution cross-section, and the table of the self shielding factor of composition dependence and temperature dependence.

✓An target energy range is from 10.5MeV to thermal energy range.

(2) Calculation of Power Distribution, Effective Multiplication Factor and Control Rod Worth

✓Basic calculation is carried out by a diffusion calculation code.

✓If needed, the nuclear characteristic amount is estimated by a transportation code and transportation theory compensation is performed.

✓Calculation of fuel composition is obtained in consideration in three dimensional effect by combining RZ system and three angle mesh system.

✓Power distribution, effective multiplication factor and control rod worth are calculated by RZ system and three angle mesh system

(3) Reactivity Coefficient

✓The reactivity coefficient is got by two-dimensional diffusion perturbative calculation.

(4) Verification by Critical Experiment Analysis

✓The group constant and design calculation method mentioned above are verified its reliability by critical experimental analysis such as FCA in Japan, ZPPR in USA.

<Contents of Attachment-8> (8/25)

3.3.4 Nuclear Design Value

- ✓ The design burnup of core fuel assembly is about 80,000MWD/t with average when taking out, while fuel pin maximum burnup is about 98,000MWD/t.
- ✓ The nuclear design values were calculated for all cores from an initial core to an equilibrium core.

3.3.5 Contents of Nuclear

3.3.5.1 Reactivity Control

- ➢ Reactivity control of a reactor is conducted by control rod.
- ➢ Control rod consists of regulation rods (main reactor shutdown system) and backup control rods (backup reactor shutdown system), moreover regulation rods are divided into fine control rods and coarse control rods.
- ➢ The usual start-up and shutdown operations are performed by regulation rods.
- ➢ Each reactor shutdown system has a function of emergency shutdown, respectively.
- ➢ Maximum excess reactivity of a reactor is about 0.057⊿k/k in the initial core and about 0.056⊿k/k in the equilibrium core.

(1) <u>Reactor Shutdown System</u>

- ✓ The reactor shutdown system has a function which can scram a reactor quickly and safely when a reactor is abnormal state.
- ✓ In case of necessary scram, a reactor should be able to be scramed by the remained system even if one of the systems doesn’t work.
- ✓ The reactor shutdown system has a reactor shutdown margin reactivity of 0.01 ⊿k/k or more at the lowest temperature on design (180^{o}C).

<Contents of Attachment-8> (9/25)

3.3.5.1 Reactivity Control

(2) Regulation Rod

✓ A regulation rod has a function which controls the required reactivity for operating.

✓ A regulation rod is composed of the fine control rods and the coarse control rods, and a reactivity applying curve of regulation rod is shown in figure.

1) Fine Control Rod

a). A fine control rod mainly controls the reactivity change causing by fuel temperature, structure and coolant temperature's change due to reactor power change.

b). A fine control rod has an enough control rod worth which can control a reactor without scram when a loading set value is changed with lamp change of ±5%/min, or step change of 10%.

c). The maximum reactivity applying ratio by withdrawing a regulation rod is decided with reactivity applying curve and max. driving speed. The max. driving speed is less than 0.3m/min, and the max. reactivity applying ratio is under 8x10-5 ⊿k/k.

d) Two fine control rod cannot be withdrawn simultaneously.

e) During the rated power operation, three fine control rods are adjusted so that they are in the fixed range in consideration of load following operation.

2) Coarse Control Rod

a). A coarse control rod mainly controls the reactivity change causing by fuel temperature, structure and coolant temperature's change due to startup operation from reactor stop state to about 30% power operation.

b). The max. driving speed of a coarse control rod is less than 0.12m/min, and the max. reactivity applying ratio is under 4x10-5 ⊿k/k.

c). Two fine control rod cannot be withdrawn simultaneously as well as fine control rod.

<Contents of Attachment-8> (10/25)

3.3.5.1 Reactivity Control

(3) Backup Control Rod

- ✓A backup control rod should be able to scram a reactor under all operation modes promptly when emergency scram is needed and can maintain sub-criticality state with sufficient predetermined reactor shutdown margin reactivity.
- ✓The backup control rods are always kept with the position of withdrawing completely from a core.

3.3.5.2 Required Reactivity and Functional Assignment of Control Rod

(1) Required Reactivity

A control rod should control reactivity is as follows: (Required reactivity)

1) Power Compensation

The reactivity change during from reactor shutdown state (180ºC) to the rated power state is compensated.

2) Burnup Compensation

The reactivity change accompanying fuel handling in each fuel cycle is compensated by a coarse control rod.

3) Error Cancellation of Reactor Reactivity

An error between the reactor reactivity and the reserve value of reactivity is canceled.

4) Operation Margin

A fine control rod has the necessary reactivity for operation and is inserted to the position where is assure the required differential reactivity worth for load following operation.

<Contents of Attachment-8> (11/25)

3.3.5.2 Required Reactivity and Functional Assignment of Control Rod

5) Reactivity Shutdown Margin

The reactor shutdown system has the predetermined reactivity shutdown margin respectively so that a reactor can be stopped during all operation modes and can be kept sufficient sub-criticality.

6) Functional Assignment of Control Rod

With respect to the reactivity which should be controlled by each control rod, the functional assignment is described in the table.

3.3.5.3 Reactivity Coefficient

- A reactivity coefficient is a coefficient showing reactivity change due to the change of fuel temperature, structural material temp. and coolant temp., etc. and is shown in the table.
- The Doppler coefficient is a ratio of reactivity change due to change of fuel effective temperature, and is negative in Monju which uses MOX fuel.
- The Doppler coefficient is dominant in the effect of reactivity change for the change of various reactor plant parameters.
- Therefore, a power coefficient integrated those is always maintained in the all operation modes and gives inherent safety to a reactor.
- Although a void factor in Monju due to boil of a coolant is positive, its value assumed in a design against boiling of one fuel assembly which has maximum reactivity, is smaller compared with the Doppler coefficient and the core support plate coefficient as shown in the table.

<Contents of Attachment-8> (12/25)

3.3.5.4 Power Distribution

In order to be flat power distribution of a reactor, the following items are taken into account.

1) A reactor core region is divided into two regions of an inner core and an outer core with the almost same volume.
2) Flat of power distribution is achieved by loading the fuel assemblies of the high plutonium enrichment into an outer core.
3) Fuel handling method adopts the distributed form.
4) The fine and coarse control rods should be operated so that it may not become unusual power distribution.
5) An output assignment ratio changes following fuel burnup, and an output assignment ration of blanket fuel region tends increasing due to accumulation of plutonium in there. An output assignment in each region is shown in the table.

3.3.5.5 Stability

- An output coefficient is negative during all operation modes because coefficients of Doppler and fuel temperature are bigger compared with other coefficients, and a reactor has self-regulating characteristics against power fluctuation.
- Also, neutron flux distribution in a core is not largely distorted locally when control rods are inserted, and power distribution is stable since spatial vibration of neutron flux distribution does not happen because fission products does not have a big absorption cross section within the neutron energy regions.

<System Specifications of Reactor Core>

原子炉熱出力		714 MW
1次冷却材流量		約 15.3×10^{6} kg／h
1次冷却材入口温度		約397℃
1次冷却材出口温度		約529℃
炉心燃料領域形状		
領　域　数		2
有　効　高　さ		約0.93 m
等　価　直　径		約1.8 m
軸方向ブランケット厚さ	上　部	約0.3 m
	下　部	約0.35 m
半径方向ブランケット等価厚さ		約0.3 m
初装荷燃料装荷量		
炉心燃料領域	プルトニウム及びウラン	約5.9 t
軸方向ブランケット	ウラン	約4.5 t
半径方向ブランケット	ウラン	約13 t
炉心燃料集合体数		
内　側　炉　心		108 体
外　側　炉　心		90 体
ブランケット燃料集合体数		172 体
制御棒集合体数		19 体
中性子源集合体数		2 体
中性子しゃへい体数		316 体
サーベイランス集合体数	中性子しゃへい体領域装荷	8 体
	炉内ラック装荷	最大8 体
炉心燃料平均取出し燃焼度		約80,000 MWD／T
増　殖　比		約1.2
炉心燃料領域組成比	燃　料	約33.5 vol％
	冷却材	約40.0 vol％
	構造材	約24.5 vol％
	空　隙	約2.0 vol％

（関連頁　8－3－1）

<Design Specifications of Fuel Assembly>

(1) 炉心燃料集合体	
(i) 燃　料	
炉心燃料材料	プルトニウム・ウラン混合酸化物
核分裂性プルトニウム富化度	
初装荷燃料	約15 wt％（内側炉心領域）
	約20 wt％（外側炉心領域）
取替え燃料	約16 wt％（内側炉心領域）
	約21 wt％（外側炉心領域）
ウラン235含有率	約0.3 wt％
ペレット密度	約85％理論密度
軸方向ブランケット燃料材料	二酸化ウラン
ウラン235含有率	約0.3 wt％
ペレット密度	約93％理論密度
ペレット直径	約5.4 mm
ペレット長さ	約 8 mm（炉心燃料）
	約10 mm（軸方向ブランケット燃料）
燃　焼　度	
初装荷炉心平均	約16,000 MWD／T
平衡炉心平均	約80,000 MWD／T
燃料集合体最高	約94,000 MWD／T
燃料要素最高	約98,000 MWD／T
ペレット最高	約130,000 MWD／T
ペレット最高温度	
定格出力時	約2,350℃
最大線出力密度時（過出力時）	約2,600℃
(ii) 被ふく管	
材　料	SUS316相当ステンレス鋼
外　径	約6.5 mm

<Core Design Specifications>

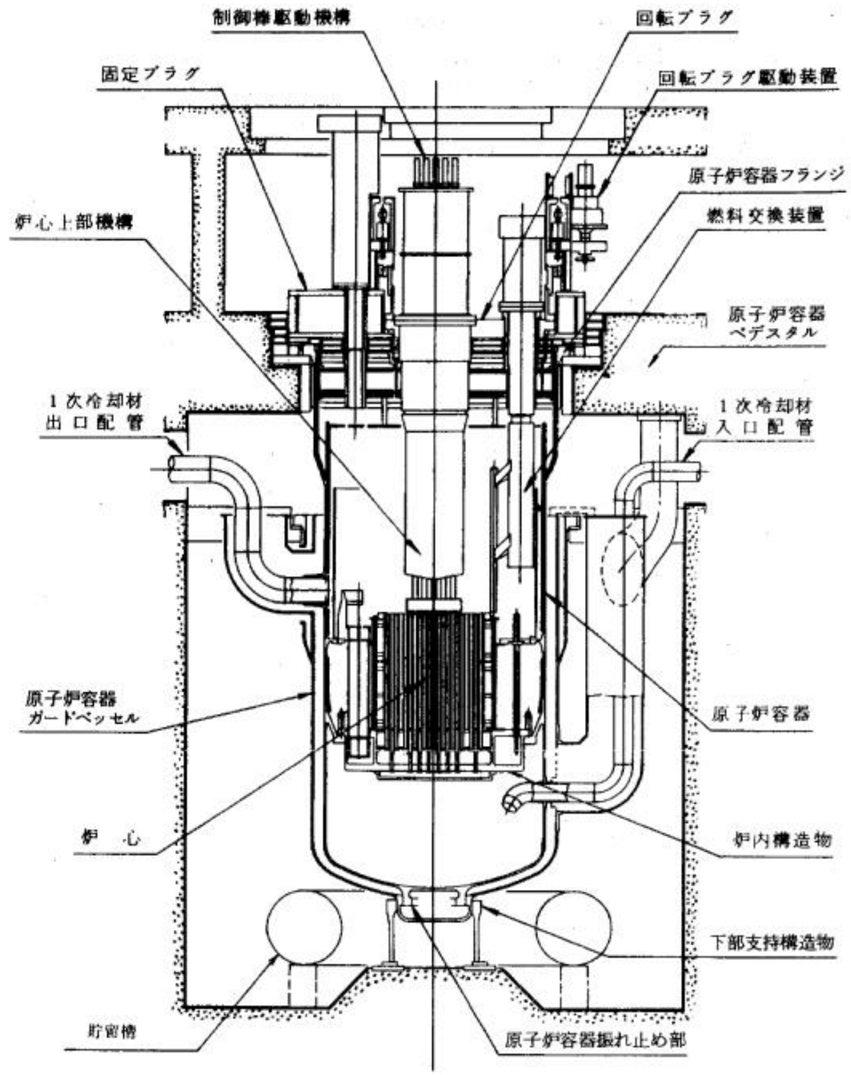
燃料交換法	
炉心燃料集合体	5バッチ分散方式
ブランケット燃料集合体	5バッチ分散方式
核分裂性プルトニウム富化度	
初装荷燃料（内側炉心，外側炉心）	約15wt％，約20wt％
取替え燃料（内側炉心，外側炉心）	約16wt％，約21wt％
増殖比	
初装荷炉心	約1.2
平衡炉心	約1.2
炉心燃料平均取出し燃焼度	
初装荷炉心	約16,000 MWD／T
平衡炉心	約80,000 MWD／T
線出力密度	
定格出力時炉心平均	約210W／cm
定格出力時炉心最高	約360W／cm
制御棒価値	初装荷炉心　平衡炉心
調整棒（最大価値調整棒1本未そう入時）	約0.07Δk／k　約0.07Δk／k
後備炉停止棒	約0.06Δk／k　約0.06Δk／k
反応度係数	
ドップラ係数	$-(5.7\sim7.6)\times10^{-3}\ T\frac{dk}{dT}$
燃料温度係数	$-(3.3\sim3.9)\times10^{-6}$ Δk／k／℃
構造材温度係数	$+(6.0\sim10)\times10^{-7}$ Δk／k／℃
冷却材温度係数	$+(1.0\sim14)\times10^{-7}$ Δk／k／℃
炉心支持板温度係数	$-(10\sim12)\times10^{-6}$ Δk／k／℃
出力係数	$-(9.4\sim11)\times10^{-6}$ Δk／k／MW
1炉心燃料集合体最大ボイド反応度	$+(1.1\sim1.5)\times10^{-4}$ Δk／k
即発中性子寿命	0.40〜0.45 μs
実効遅発中性子割合	0.0034〜0.0038

<Reactivity Balance>

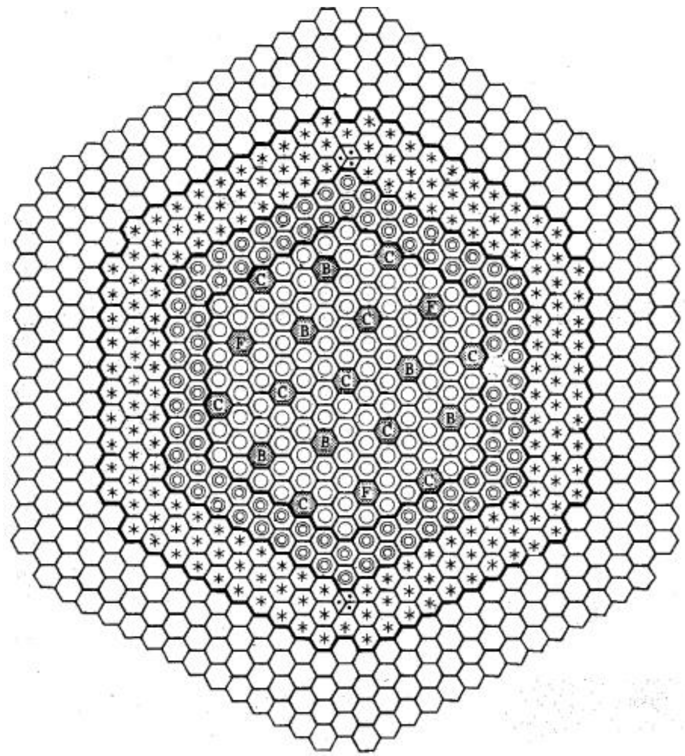
$\times10^{-2}$ Δk/k

炉心 / 原子炉停止系 / 制御棒 / 反応度バランス		初装荷炉心 主炉停止系 調整棒	初装荷炉心 後備炉停止系 後備炉停止棒	平衡炉心 主炉停止系 調整棒	平衡炉心 後備炉停止系 後備炉停止棒
所要反応度	出力補償	1.9	1.9	1.7	1.7
	燃焼補償	2.5	—	2.6	—
	運転余裕	0.3	—	0.3	—
	炉の反応度の誤差吸収	1.0	—	1.0	—
	所要反応度の合計	5.7	1.9	5.6	1.7
制御棒価値		7.1*	5.9	7.0*	5.8
余裕反応度		1.4	4.0	1.4	4.1

* 最大反応度価値を持つ制御棒1本が，全引抜き位置のままそう入できないとした場合。

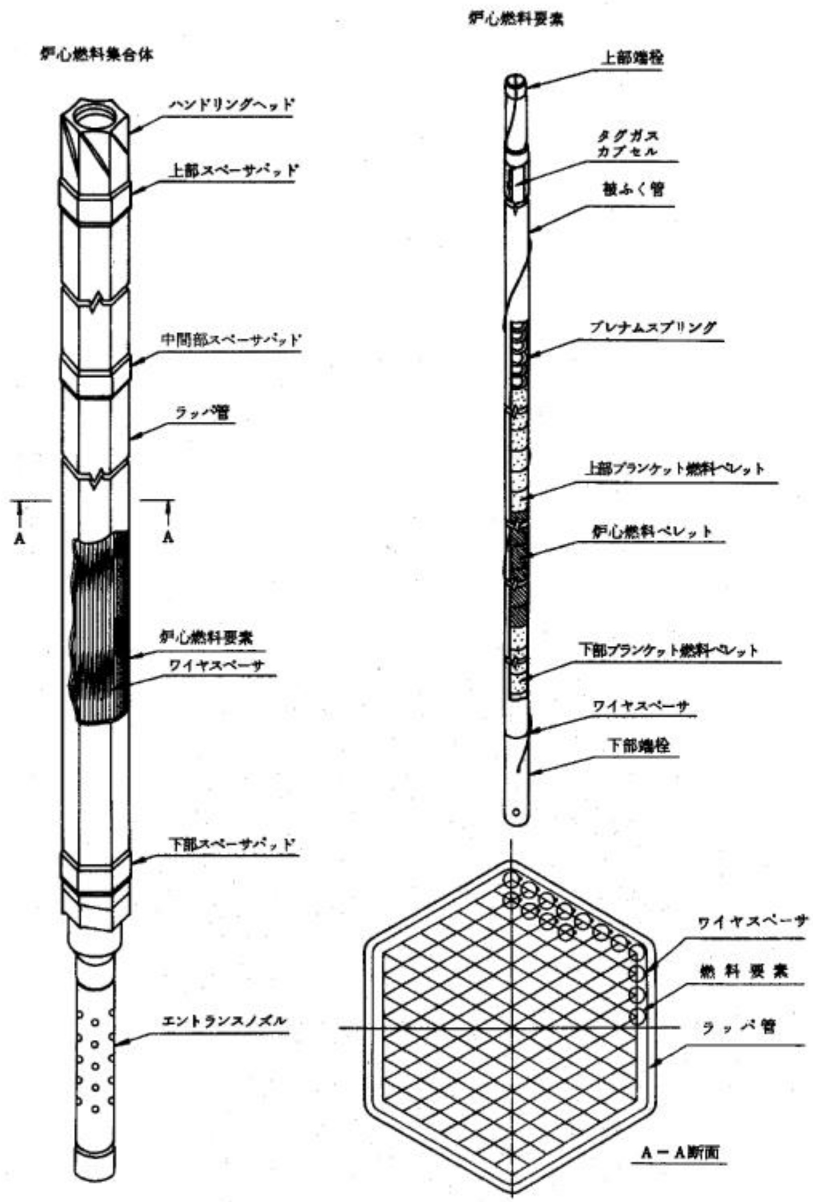


<Internal Structure of Reactor Core>

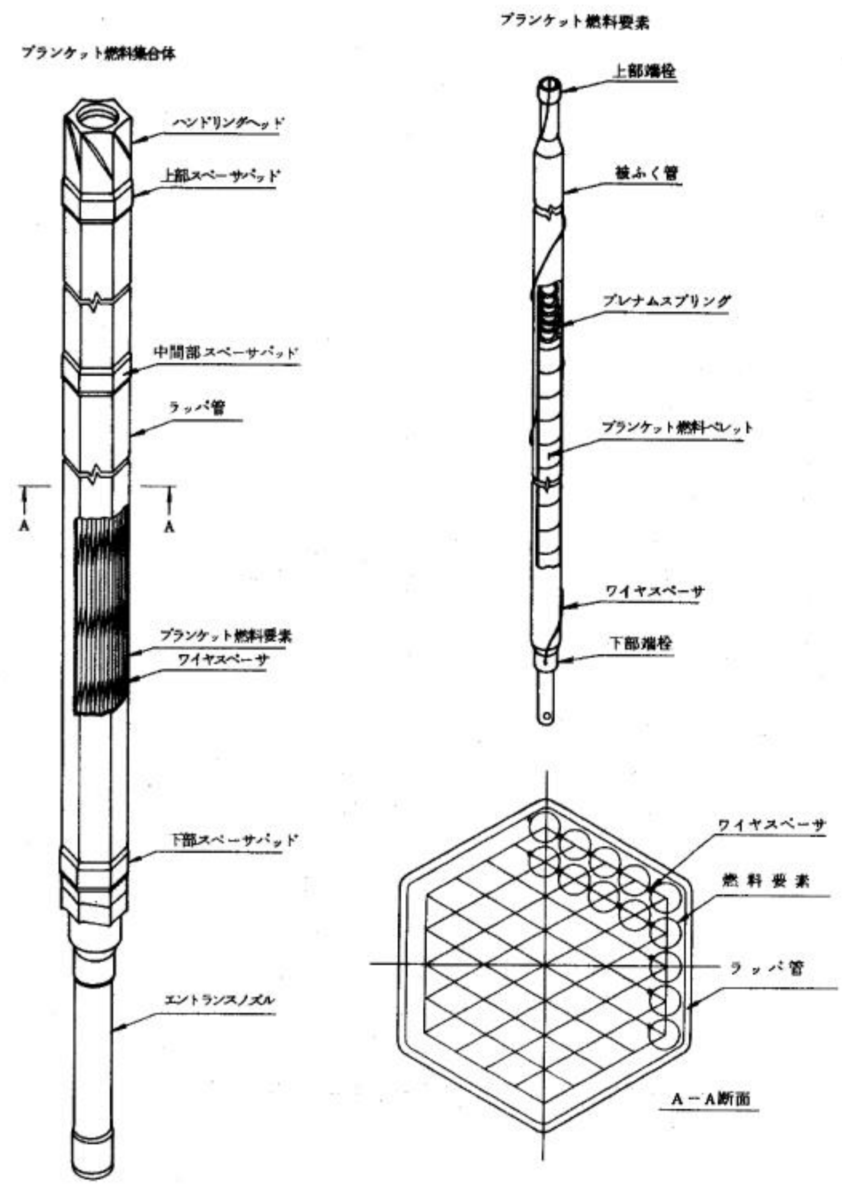


炉心構成要素		記号	数量
炉心燃料集合体	内側炉心		108
	外側炉心		90
ブランケット燃料集合体			172
制御棒集合体	微調整棒	F	3
	粗調整棒	C	10
	後備炉停止棒	B	6
中性子源集合体			2
中性子しゃへい体			316
サーベイランス集合体			8

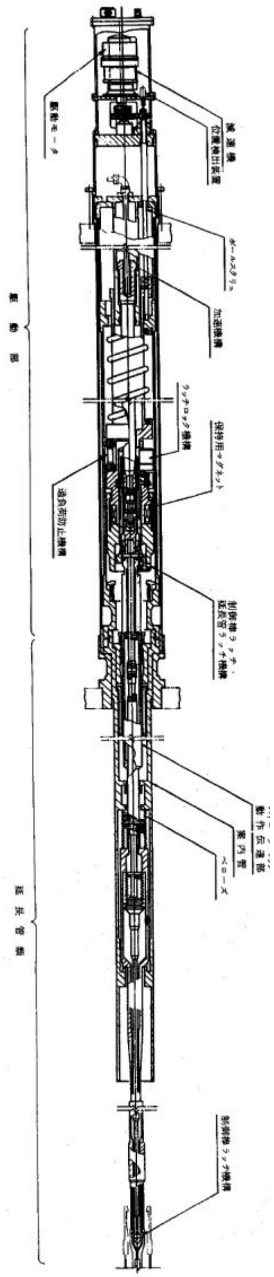
<Core Map of Initial Core>

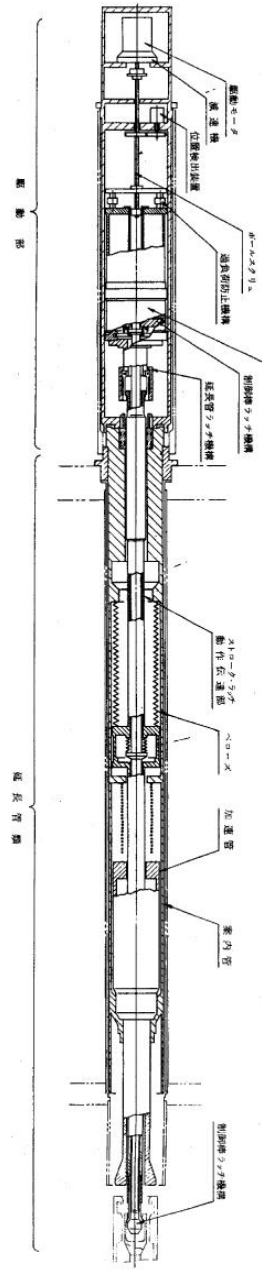


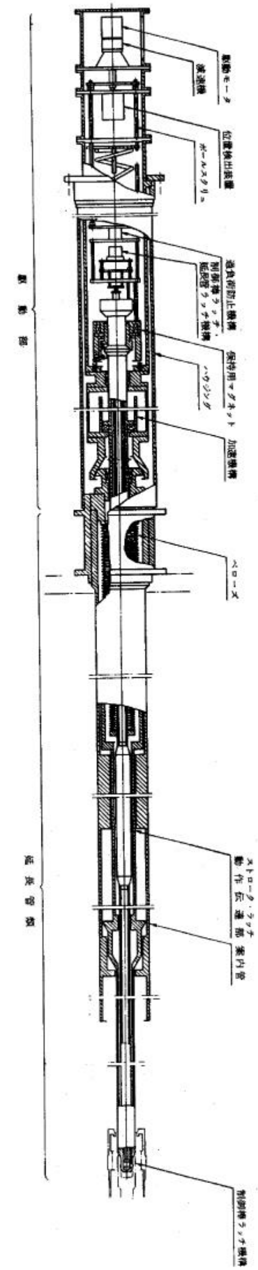
<Structure of Fuel Assembly >



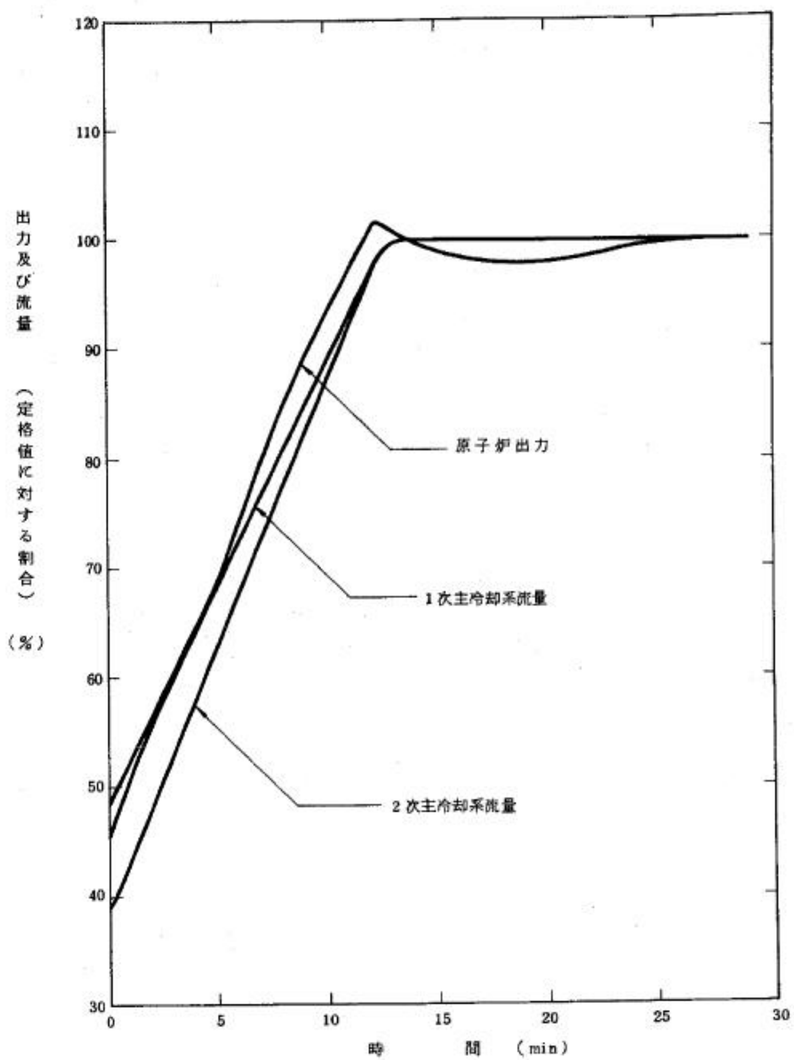
<Structure of Blanket Assembly >

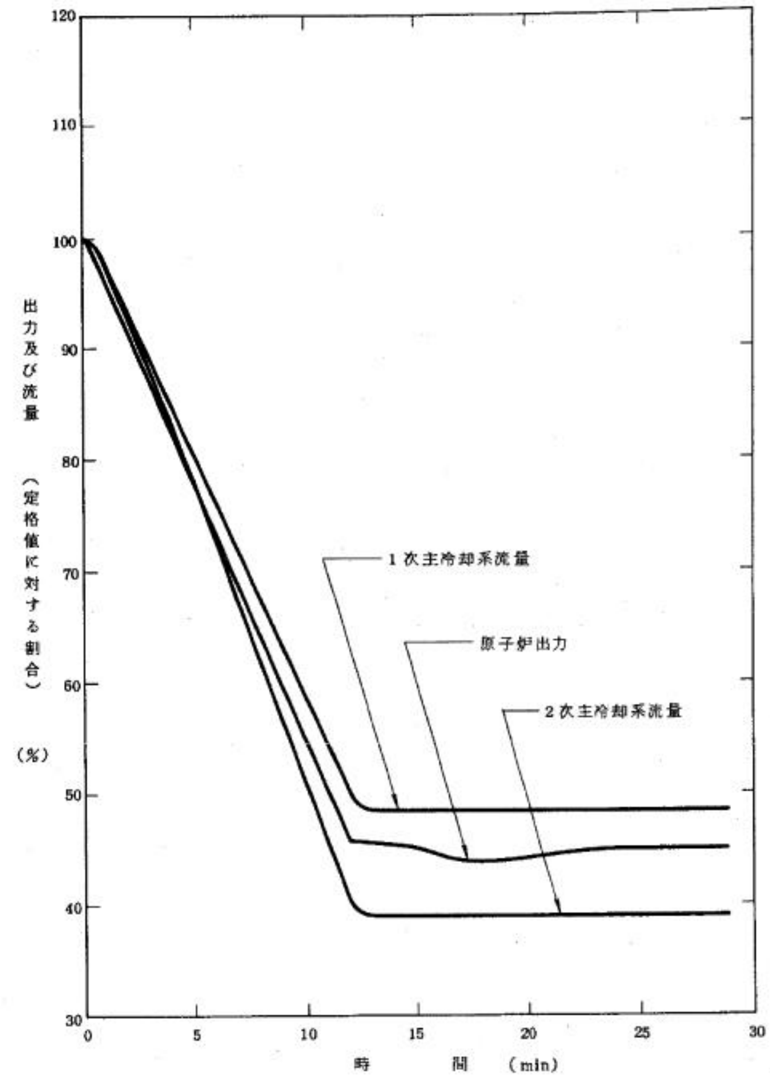






<Structure of Fine Control Rod > <Structure of Coarse Control Rod > <Structure of Backup Control Rod >





<Analysis of Power Up with 5%/min lump Ration>

<Analysis of Power Down with 5%/min lump Ration>

<Contents of Attachment-8> (13/25)

3.3.5.6 Fuel Handling and Change following Burnup

- Fuel handling method adopts five batch dispersion method, i.e., one fifth of each core (inner, outer and blanket) is exchanged with the interval of each half years.
- The duration of one fuel cycle is about 148 EFPD (effective full power day).
- In addition, fuel handling in the early stage of formal operation will be performed with four batch dispersion method with accumulating the irradiation results. That time's duration of one fuel cycle is about 123 EFPD.
- During the period mentioned above, change of power distribution and nuclear characteristics value are little.

3.4 Thermal Hydraulic Design

3.4.1 Outline

3.4.2 Design Policy

3.4.3 Analysis Method

3.4.4 Thermal Hydraulic Design Values

3.4.5 Contents of Thermal Hydraulic Design

3.4.5.1 Power distribution applied to thermal hydraulic design

3.4.5.2 Coolant flow distribution

3.4.5.3 Maximum temperatures of coolant and fuel cladding

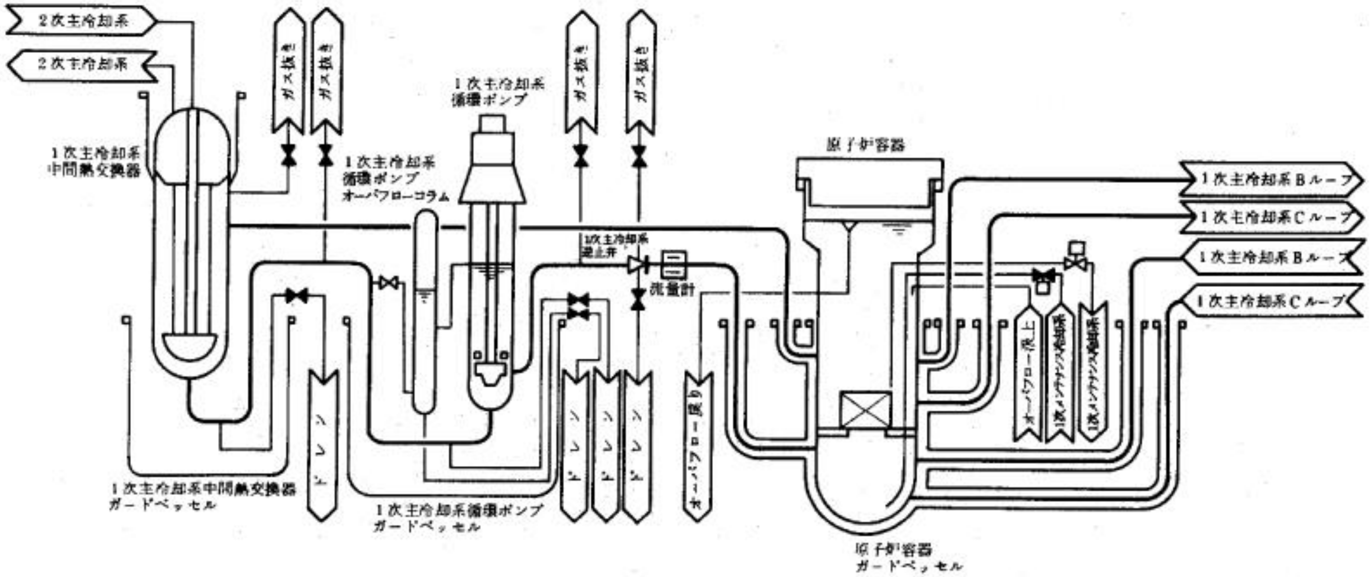
3.4.5.4 Fuel maximum temperature

<Contents of Attachment-8> (14/25)

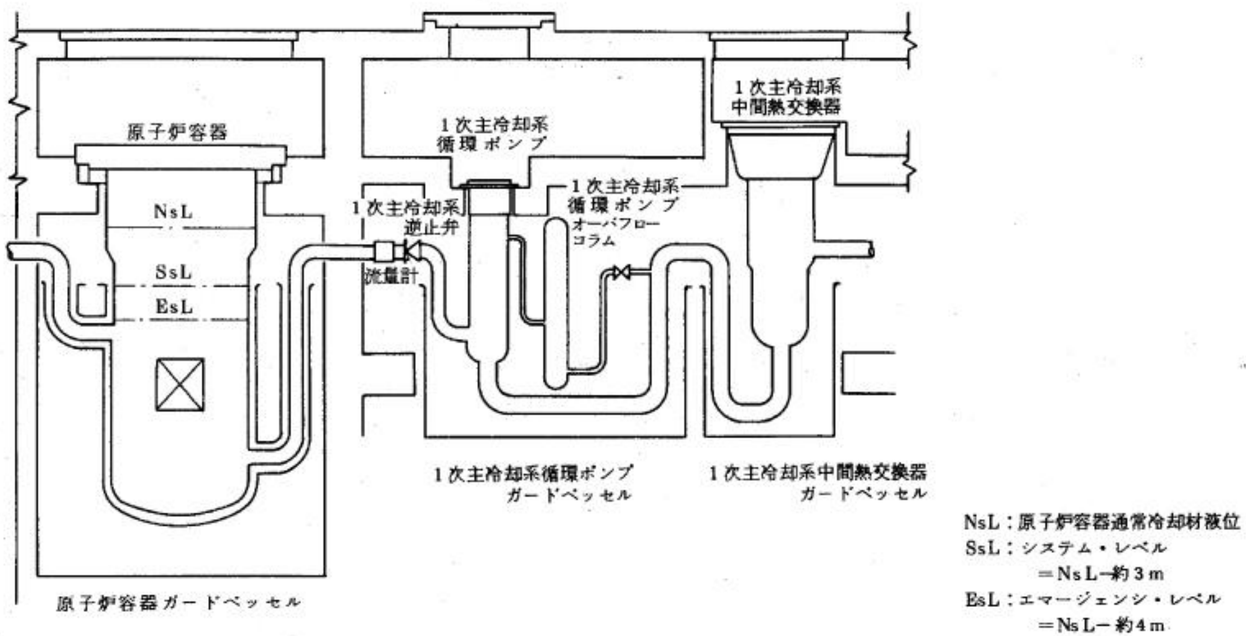
<Contents of Attachment-8> (15/25)

<Contents of Attachment-8> (16/25)

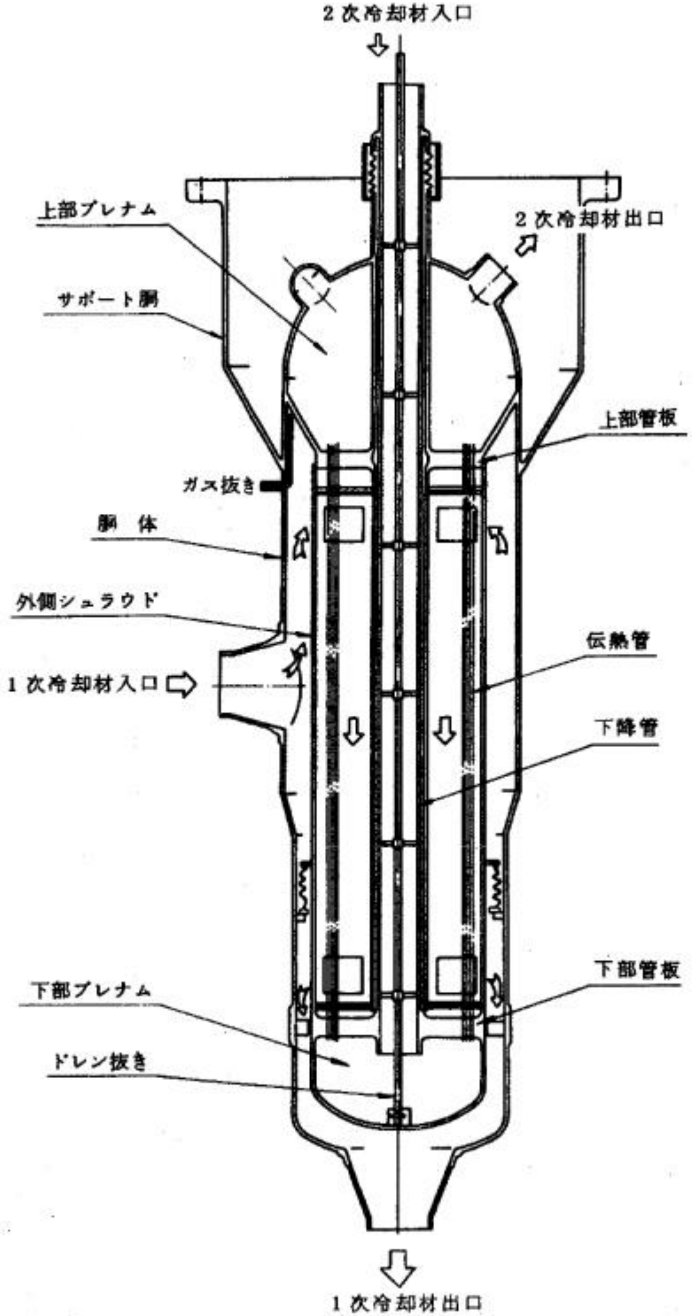
<Contents of Attachment-8> (17/25)



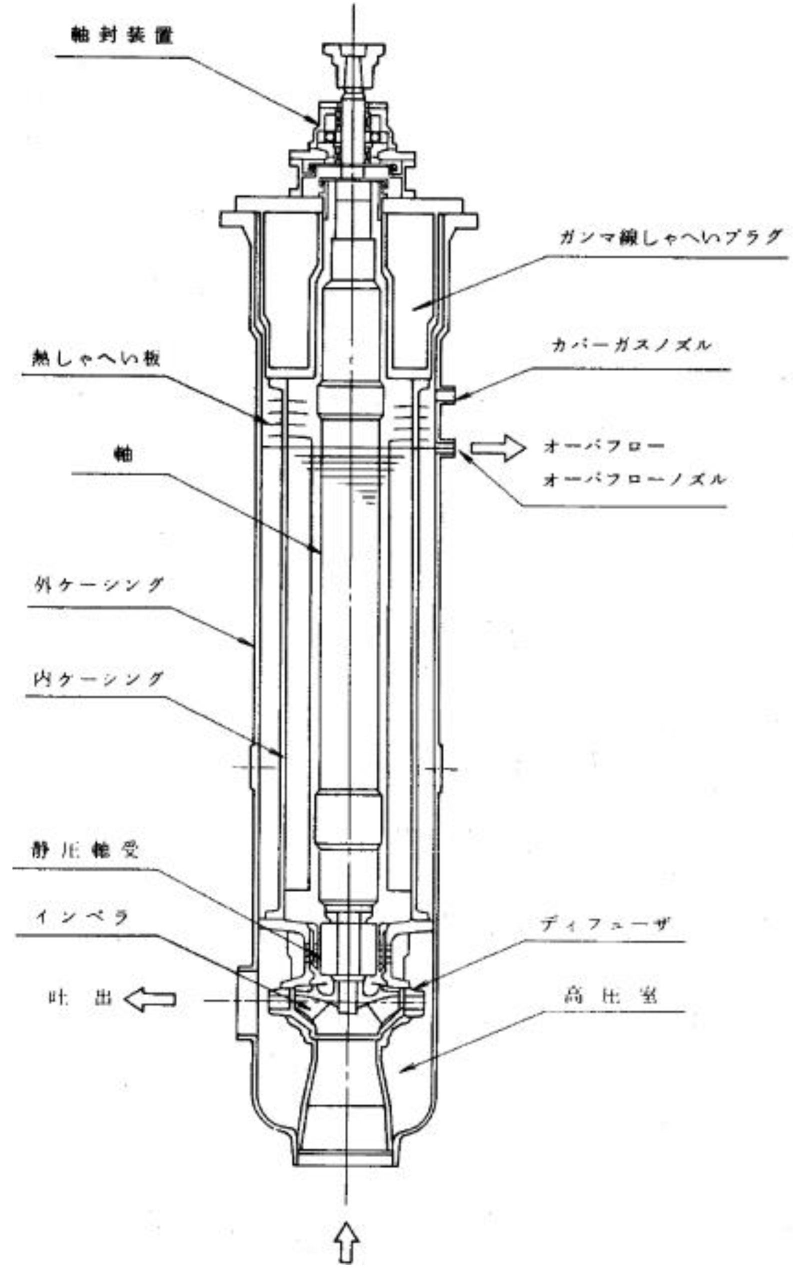
<Main Cooling System>



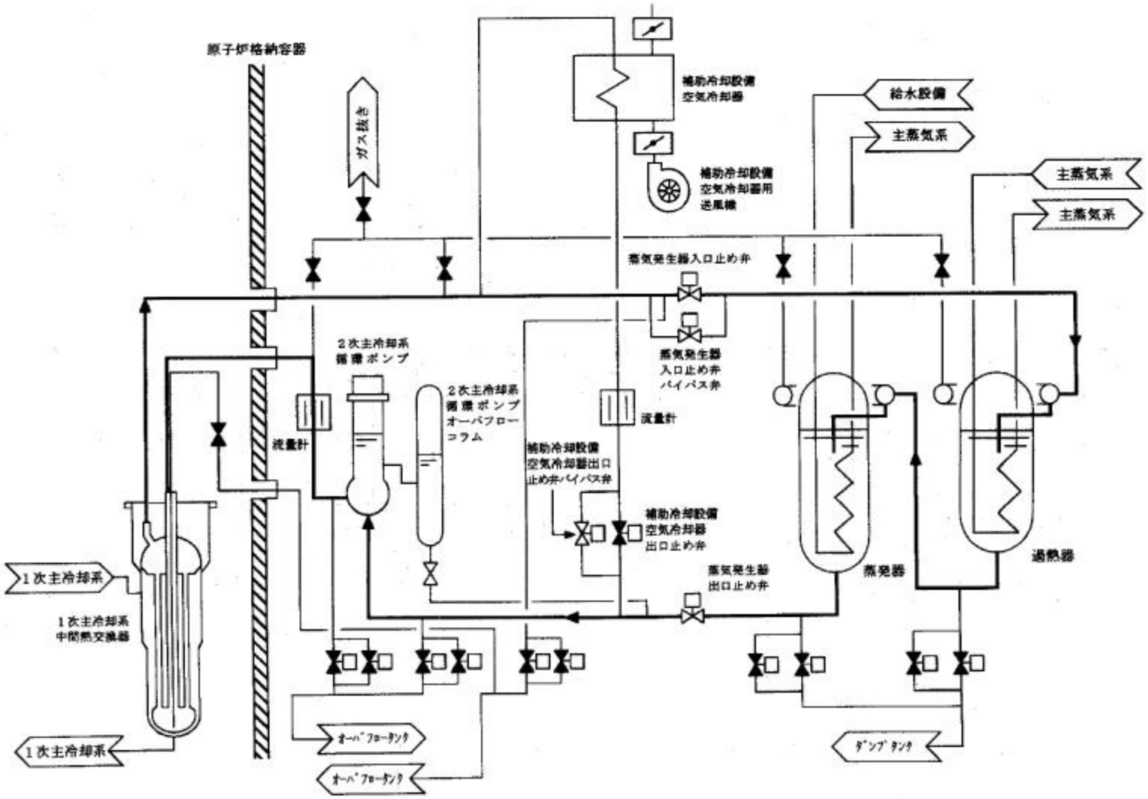
<Guard Vessel for Primary Main Cooling System>



<Structure of Primary Main IHX>

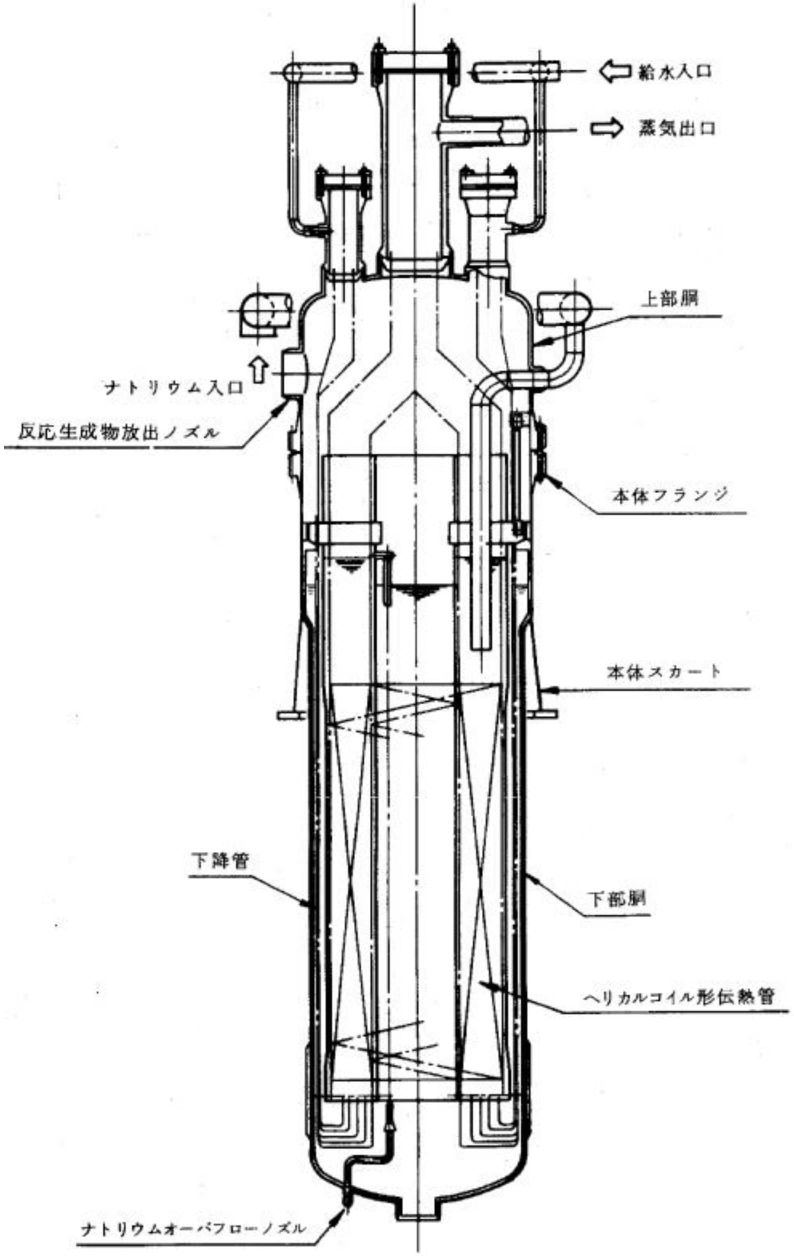


<Structure of Primary Main Pump>

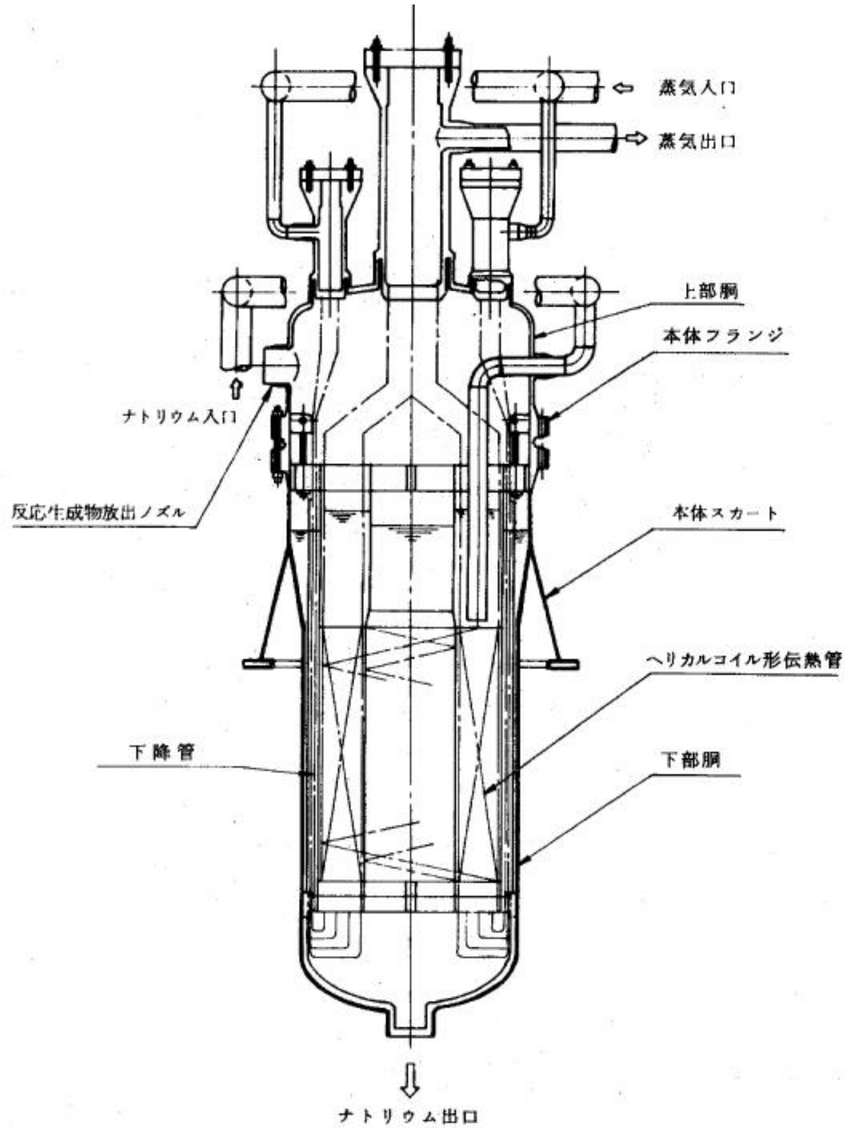


<Secondary Main Cooling System>

<Auxiliary Cooling System>



<Structure of Evaporator>

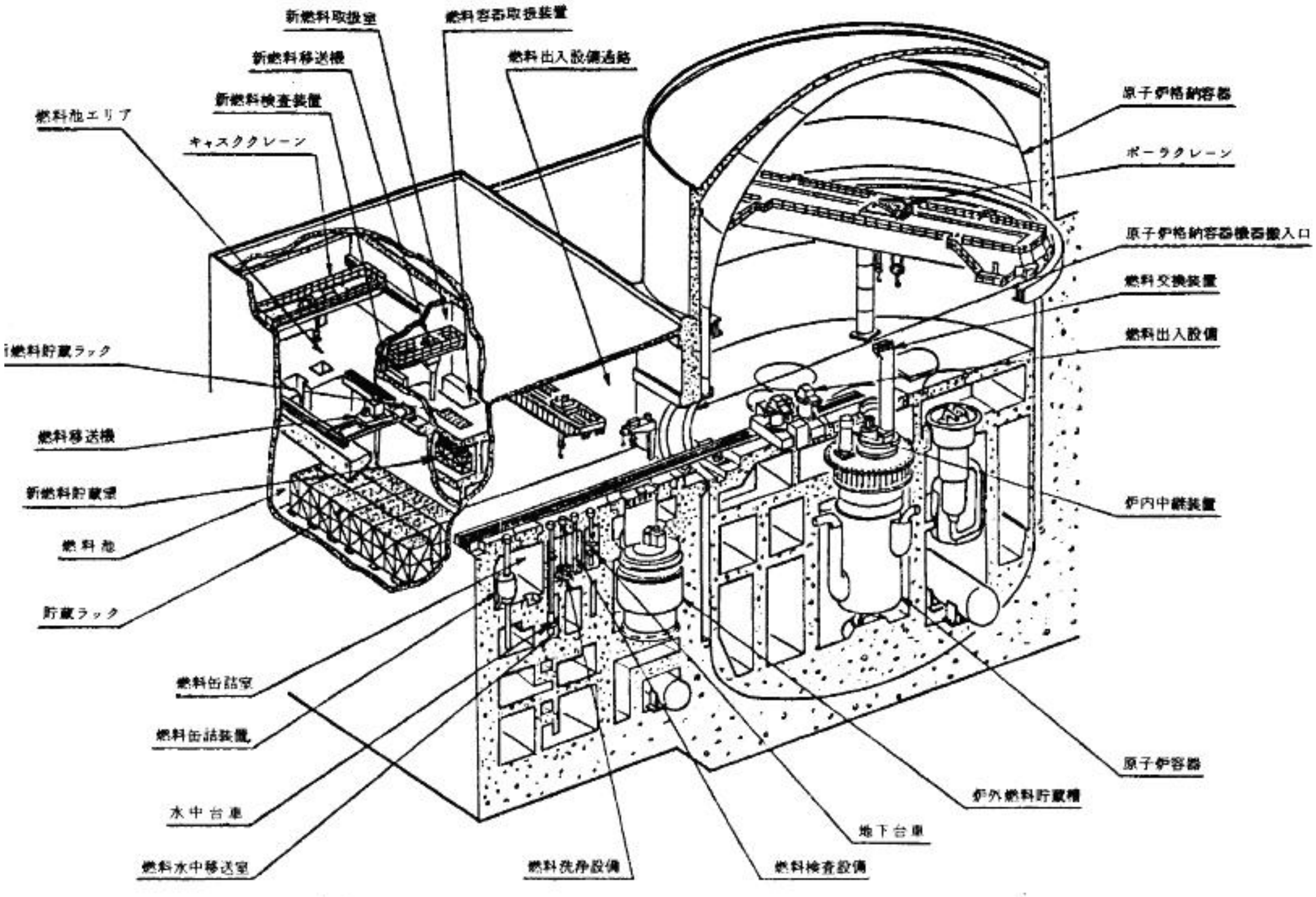


<Structure of Super-Heater>

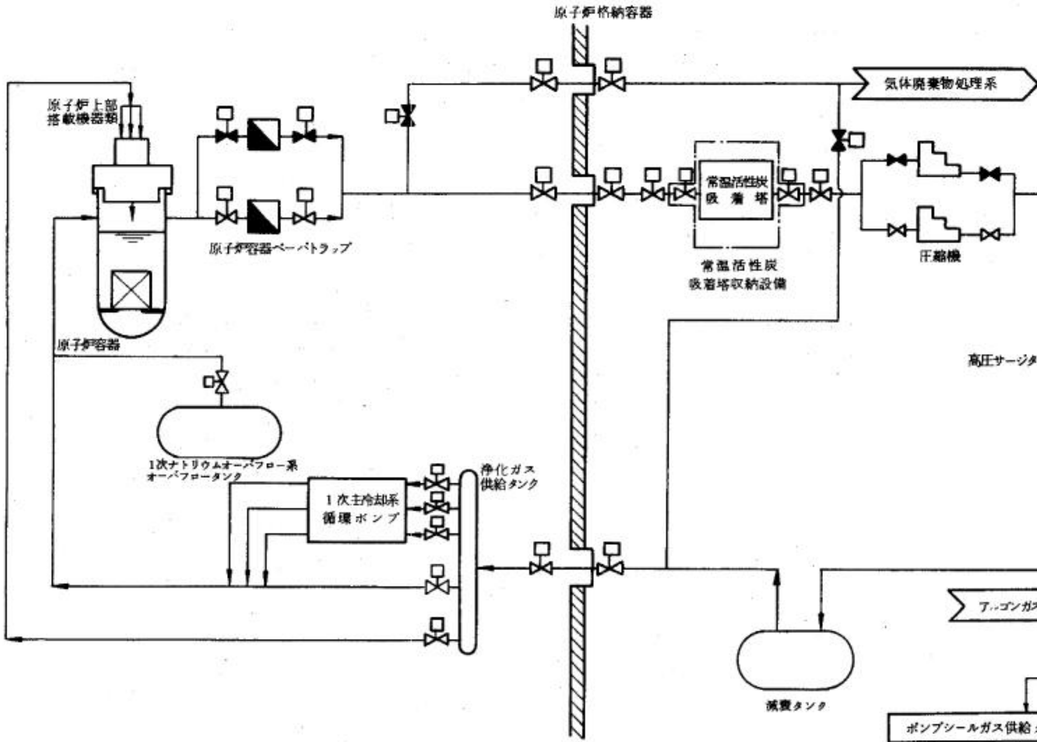
<Contents of Attachment-8> (18/25)

<Contents of Attachment-8> (19/25)

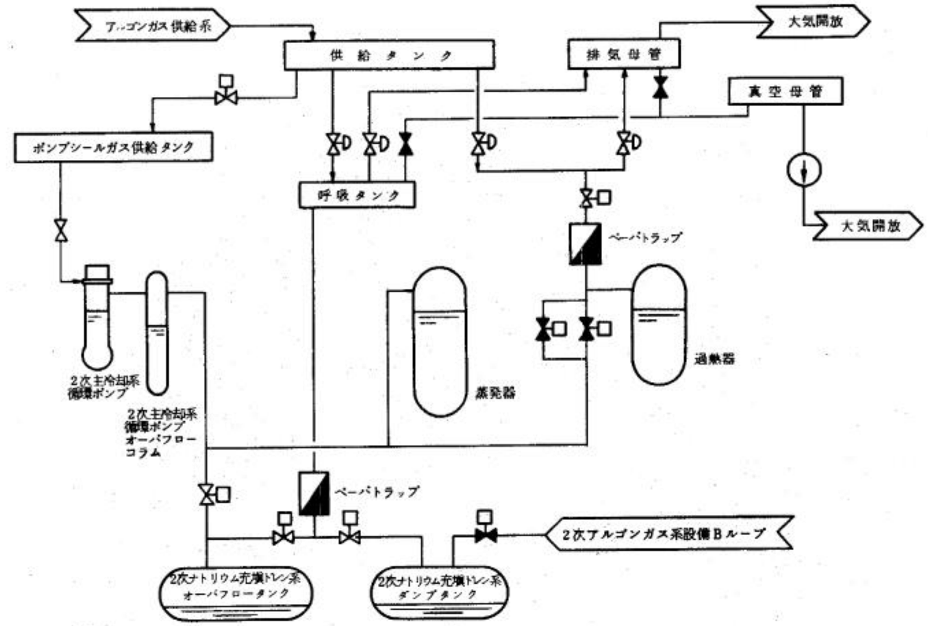
<Contents of Attachment-8> (20/25)



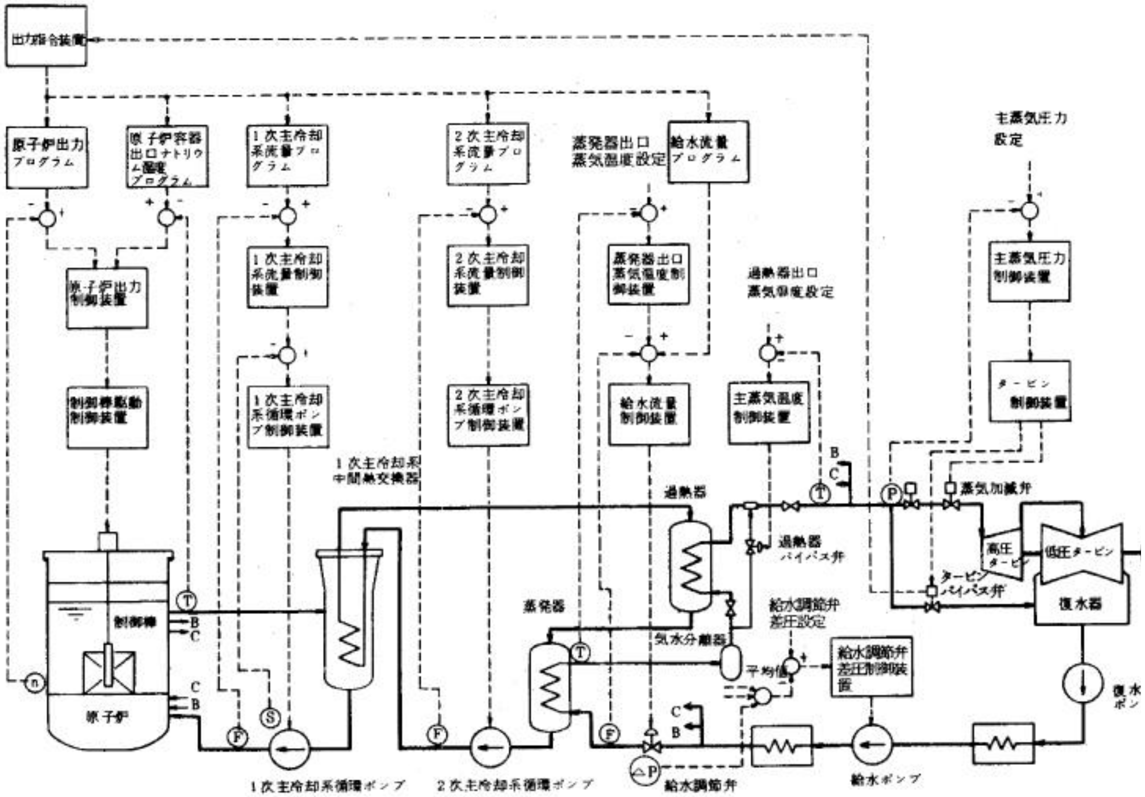
<Bird-View of Fuel Handling System>



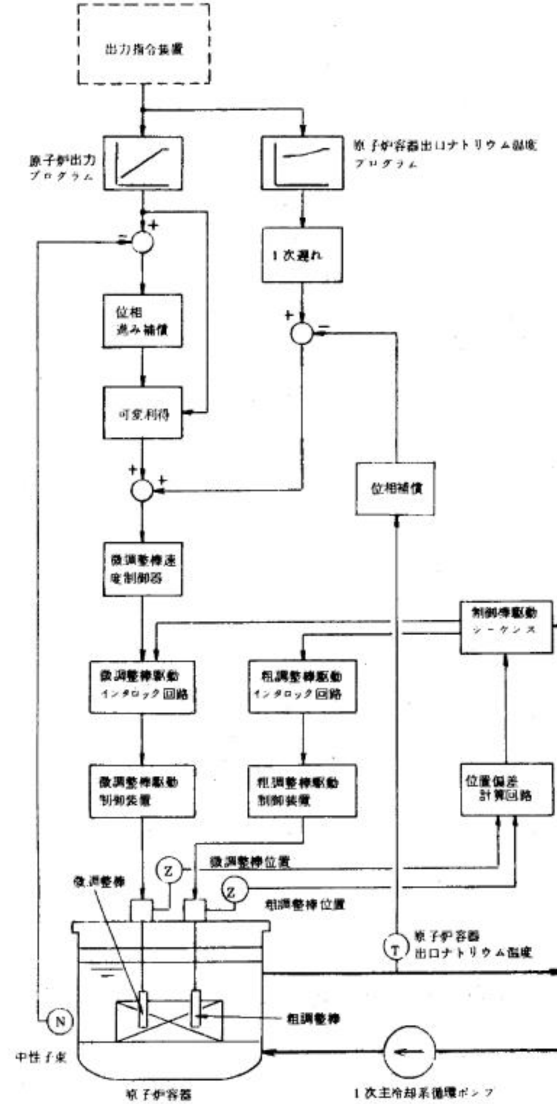
<Primary Argon Gas System>



<Secondary Argon Gas System>



<Reactor Control System>



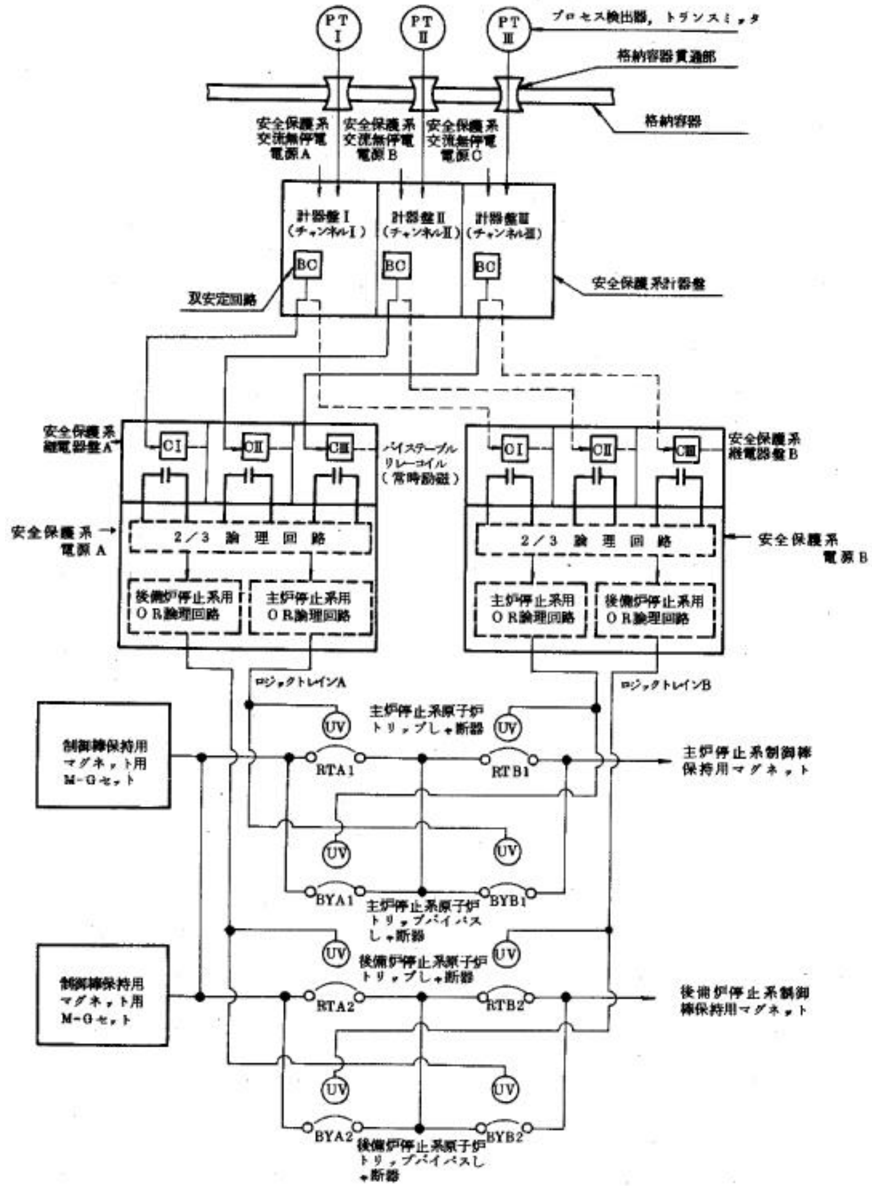
<Reactor Power Control System>

原子炉トリップ信号	検出器	作動ロジック	インタロック
線源領域中性子束高	線源領域中性子束検出器	1／2	（P－A）設定値以上で手動ブロック
広域中性子束高 a　低設定 b　高設定	 広域中性子束検出器 広域中性子束検出器	 2／3 2／3	（P－B）設定値以上で手動ブロック
出力領域中性子束高 a　低設定 b　高設定	 出力領域中性子束検出器 出力領域中性子束検出器	 2／3 2／3	（P－B）設定値以上で手動ブロック
出力領域中性子束変化率高	出力領域中性子束検出器	2／3	
原子炉容器ナトリウム液位低	原子炉容器ナトリウム液面計	2／3	（P－A）設定値以下で手動ブロック
原子炉容器出口ナトリウム温度高	原子炉容器出口ナトリウム温度検出器	各ループ2/3	
中間熱交換器1次側出口ナトリウム温度高	中間熱交換器1次側出口ナトリウム温度検出器	各ループ2/3	
1次主冷却系循環ポンプ回転数低	（1次主冷却系循環ポンプ回転数検出器 出力領域中性子束検出器）	各ループ2/3	（P－A）設定値以下で手動ブロック
1次主冷却系循環ポンプ回転数高	（1次主冷却系循環ポンプ回転数検出器 出力領域中性子束検出器）	各ループ2/3	
1次主冷却系流量低	（1次主冷却系流量検出器 広域中性子束検出器）	各ループ2/3	（P－A）設定値以下で手動ブロック
2次主冷却系循環ポンプ回転数低	（2次主冷却系循環ポンプ回転数検出器 出力領域中性子束検出器）	各ループ2/3	（P－A）設定値以下で手動ブロック
2次主冷却系流量低	（2次主冷却系流量検出器 広域中性子束検出器）	各ループ2/3	（P－A）設定値以下で手動ブロック
蒸発器出口ナトリウム温度高	蒸発器出口ナトリウム温度検出器	各ループ2/3	
タービントリップ	主蒸気止め弁位置検出器	2台閉	（P－C）設定値以下で自動ブロック
常用母線電圧低	常用母線電圧低電圧リレー	各母線 2/3	
燃料破損検出	遅発中性子束検出器	各ループ2/3	
原子炉格納容器隔離			第9.6－1表(1)
地震加速度大 a　水平方向加速度 b　垂直方向加速度	 水平方向加速度検出器 垂直方向加速度検出器	 2／3 2／3	
手　　動		1／2	

(注)　トリップ設定値は詳細設計で決定

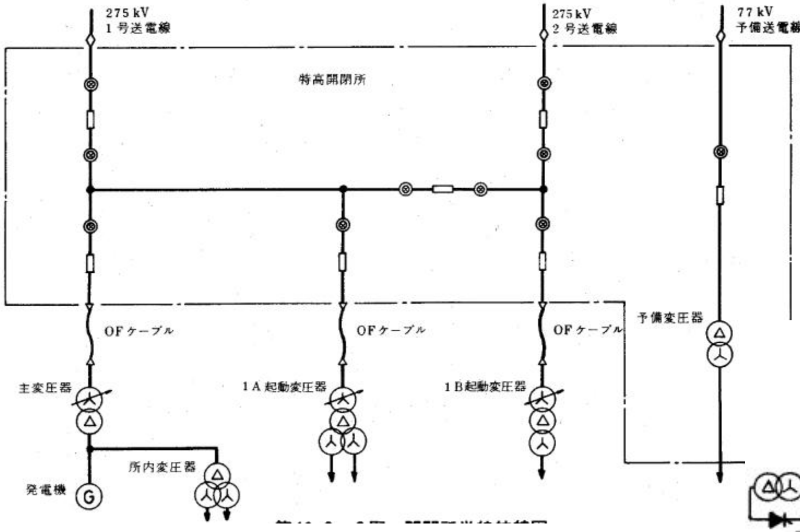
（関連頁8－9－24）

<List of Reactor Scram Items>

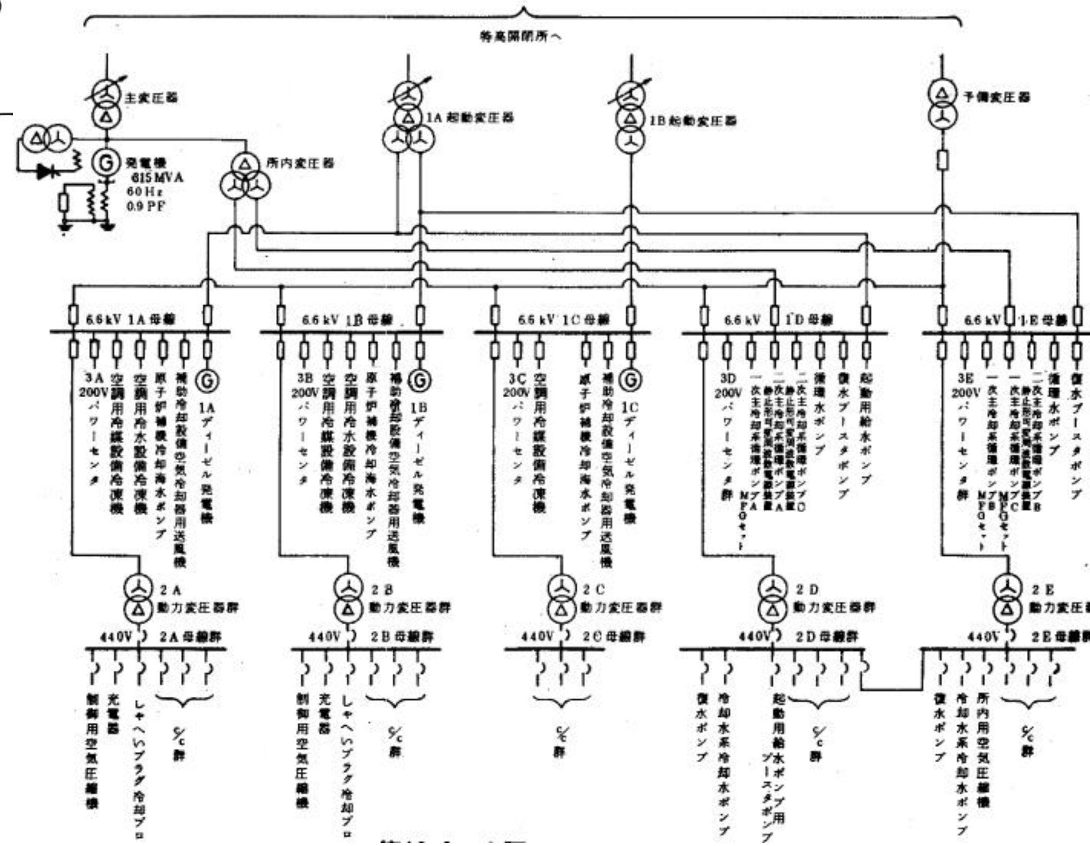


<Reactor Safety Protection System Flow>

<Contents of Attachment-8> (21/25)



<Power Supply Switching Line>



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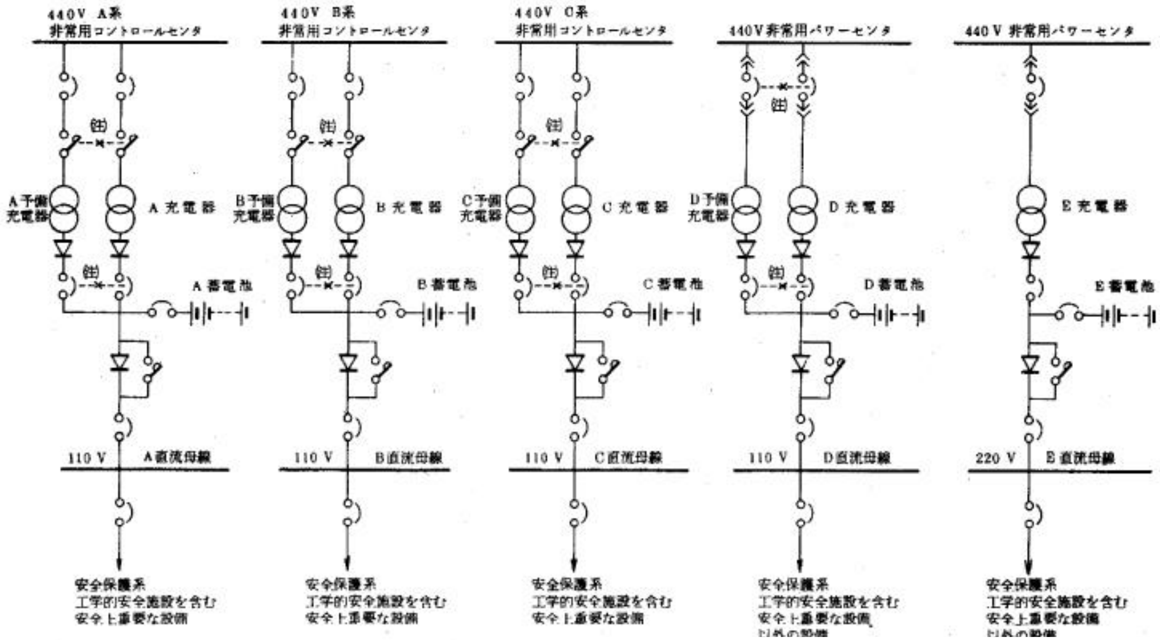

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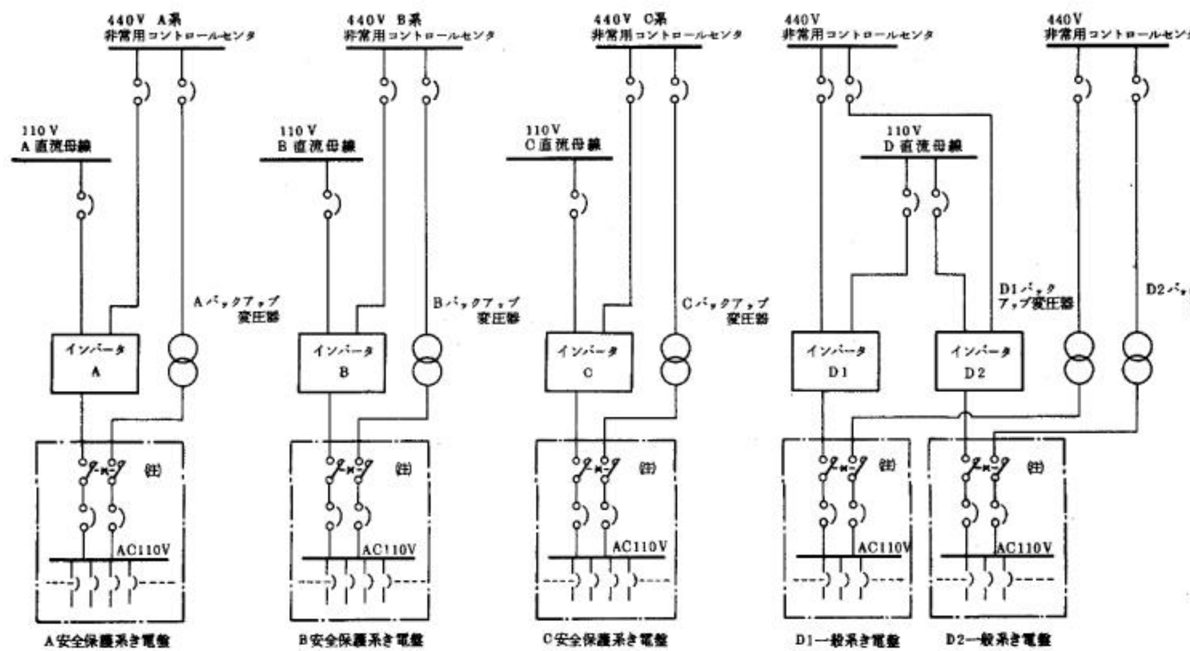

<AC Uninterruptible Power Line>

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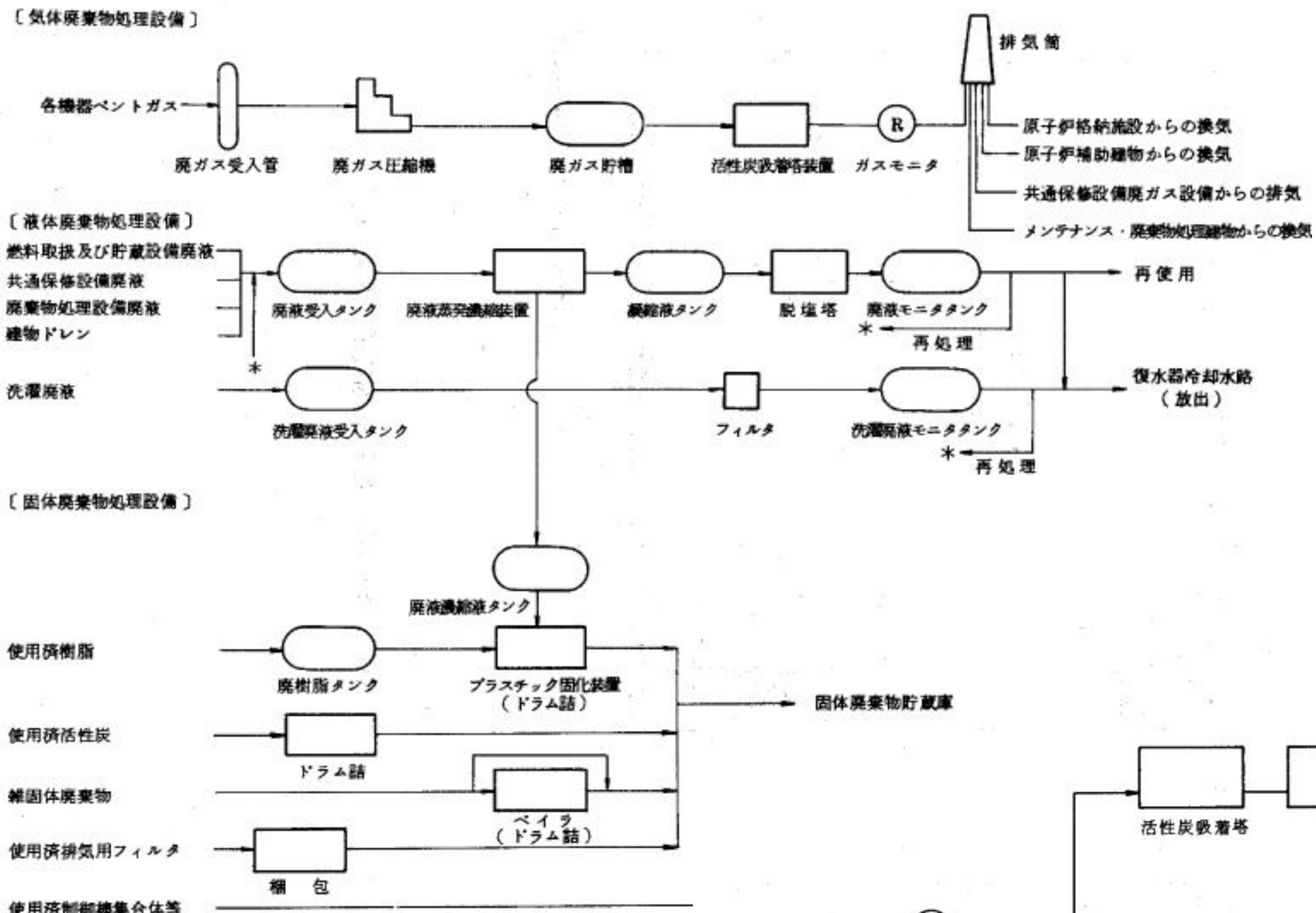
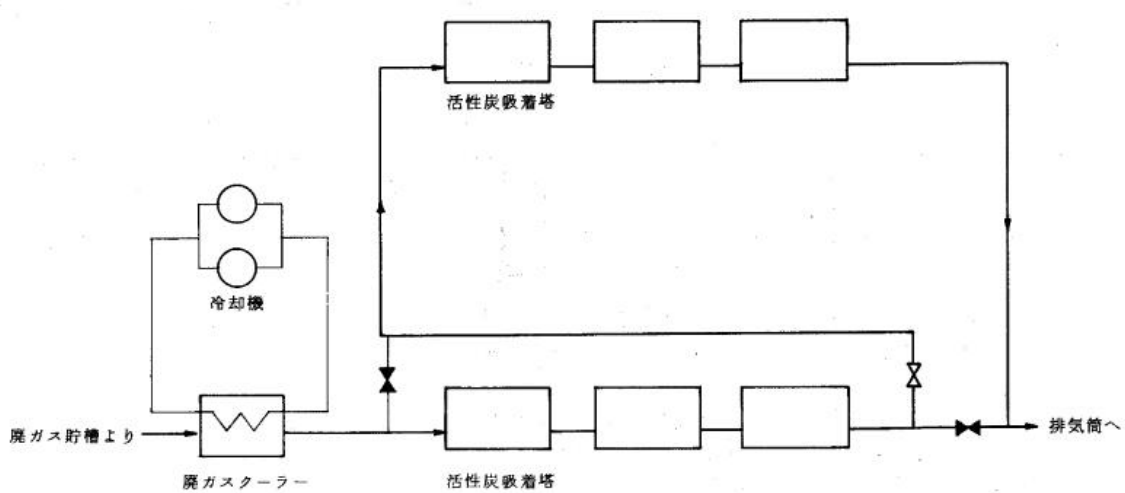


<Radioactive Waste Disposal System



<Charcoal Absorption System>

<Contents of Attachment-8> (24/25)

<Contents of Attachment-8> (25/25)

15. Operation and Maintenance

15.1 Basic Policy of Operation/Maintenance

15.2 Organization and Job

15.3 Operation Management

15.4 Fuel Management

15.5 Radioactive Waste Management

15.6 Radiation Control

15.7 Maintenance

15.8 Measures for Emergency

15.9 Education and Training

15.10 Health Management

15.11 Security Management for Staffs

15.12 Record and Report

Attachment-10: Explanation Regarding “??????”

Since Attachment-10 is mentioned in the “Safety Assessment of Reactor Plant” of Lecture-??” in detail, its contents is skipped in here.

5. Actual Example of Installation Permission of NPP

As an actual example, Change of Application of Installation Permission of Tomari and Onagawa NPP Accompanying introduction of Plutonium Fuel is mentioned in the following.

By introducing Plutonium Fuel, change the following items:

1) Fall of Melting Point and Thermal Conductivity of Fuel Pellet

2) Increase of Fission Product (FP) Gas Emission

3) Change of Power Distribution in Radial Direction

4) Fall of Rod Worth (Decrease of Boron Effect)

5) Change of Nuclear Characteristics (Increase of Fission Cross Section, Thermal Absorption Cross Section, Temperature Effect and Fall of Delayed Neutron Yield, etc.)

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Main Screening Contents of Safety Assessment

Safety Assessment

1. Safety Design of Reactor Plant
   - 1.1 Reactor Core Design

     (Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)
   - 1.2 Emergency Core Cooling System (ECCS)
   - 1.3 Fuel Storage Facility
2. Exposure Evaluation of General Public around NPP
3. Analysis of Abnormal Transient Change at Operation
4. Analysis of Accident
5. Analysis of Hypothetical Accident for Siting Evaluation
6. Technical Capability (Organization, Quality Assurance Activity for Design, Operation, etc.)

→ **Confirmation of Safety of NPP**

Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant

1.1 Reactor Core

(Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)

1.2 Emergency Core Cooling System (ECCS)

1.3 Fuel Storage Facility

2. exposure Evaluation of General Public around NPP
3. Analysis of Abnormal Transient Change (Incident) during Normal Operation
4. Analysis of Accident
5. Analysis of Hypothetical Accident for Siting Evaluation
6. Technical Capability

1.1 (1) Reactor Core (Mechanical Design)

Screening Points for Soundness Assessment of Fuel Assembly

1) Central Max. Temp. of Fuel Pellet

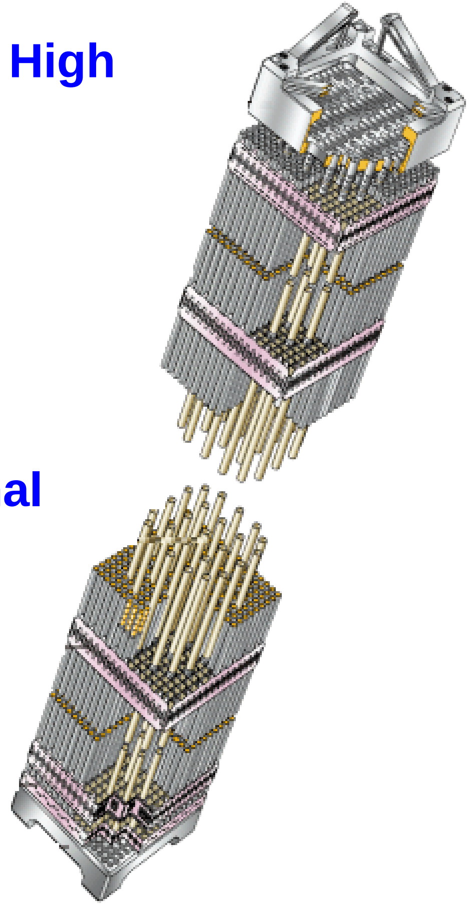
☞ The possibility of **Melt of Fuel Pellet** **due to Abnormal High Temperature** (Comparison with Criterion)

2) Inner Pressure of Fuel Pin

☞ The possibility of **Rapture of Fuel Pin** **due to FP Gas Expansion** (Comparison with Criterion)

3) Thermal Stress of Fuel Cladding Tube

☞ **Thermal Stress of Fuel Cladding Tube** **due to Thermal Loading (During in Normal operation, Incident and Accident)** (Comparison with Criterion)



4) Tensile Strain of Fuel Cladding Tube

☞ **Axial Thermal Stress of Fuel Cladding Tube** **due to Thermal Loading** (Comparison with Criterion)

5) Cumulative Fatigue of Fuel Cladding Tube

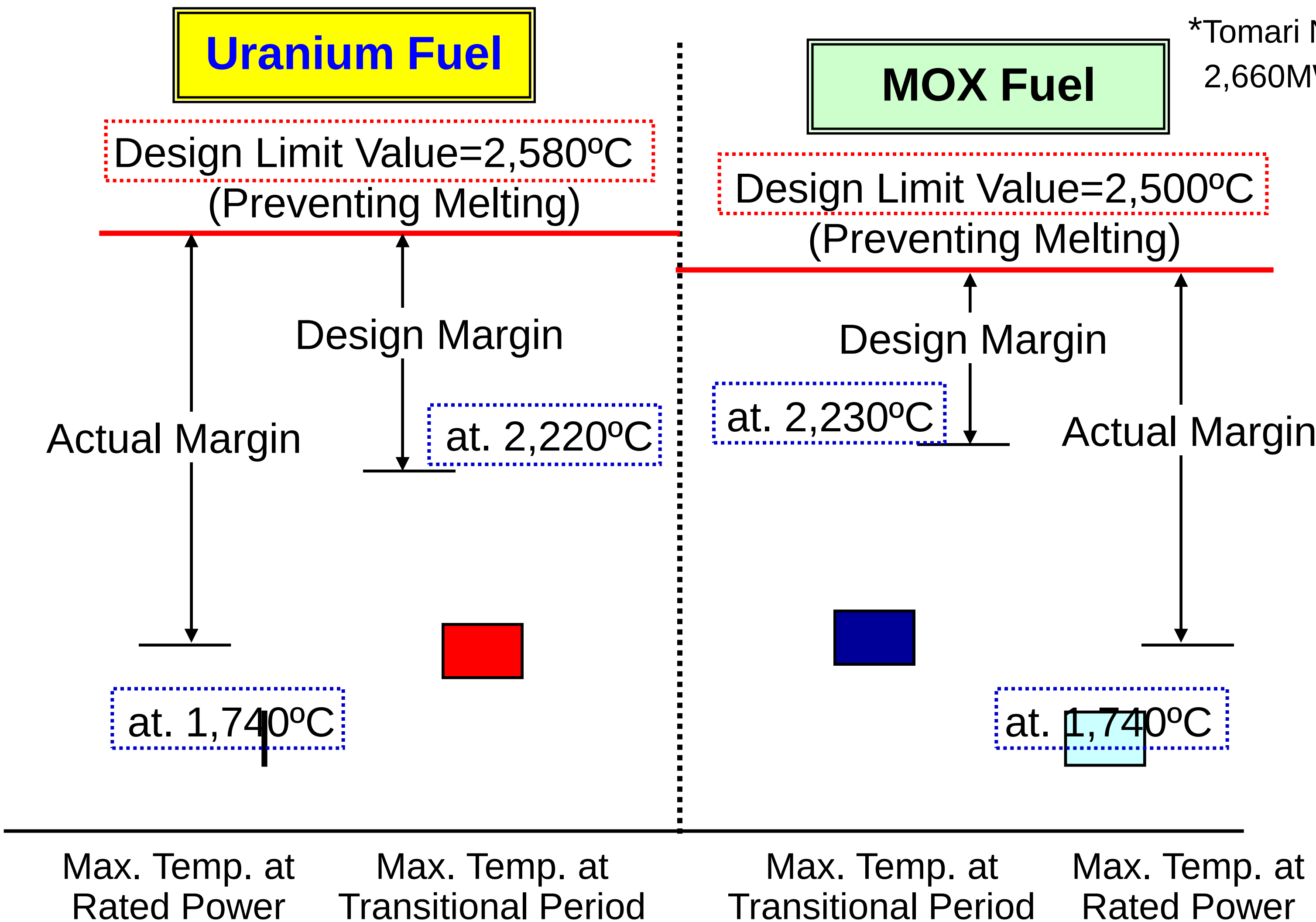
☞ **Cumulative Fatigue Damage Coefficient of Fuel Cladding Tube** **due to Repeat Thermal Loading** (Comparison with Criterion)

Example of Analysis Results (1)*: Central Max. Temp. of Fuel Pellet (1/2)

<Screening Point>

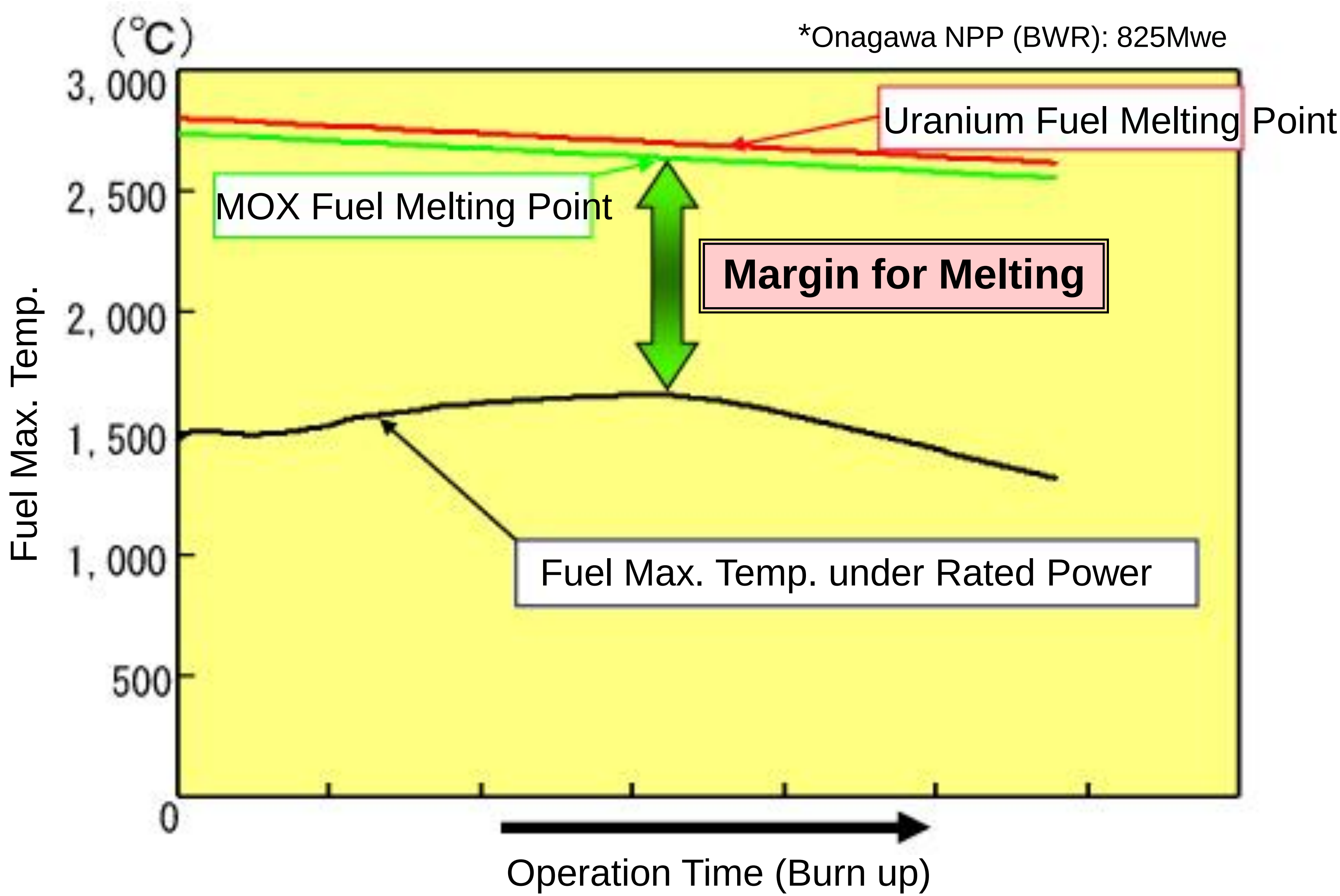
Isn't a fuel melted when reactor's temperature is risen to abnormal high temp.?

*Tomari NPP (PWR): 2,660MWt (912Mwe)



Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の 一次審査結果、P16、平成22年5月、原子力安全・保安院

Example of Analysis Results (1)*: Central Max. Temp. of Fuel Pellet (2/2)



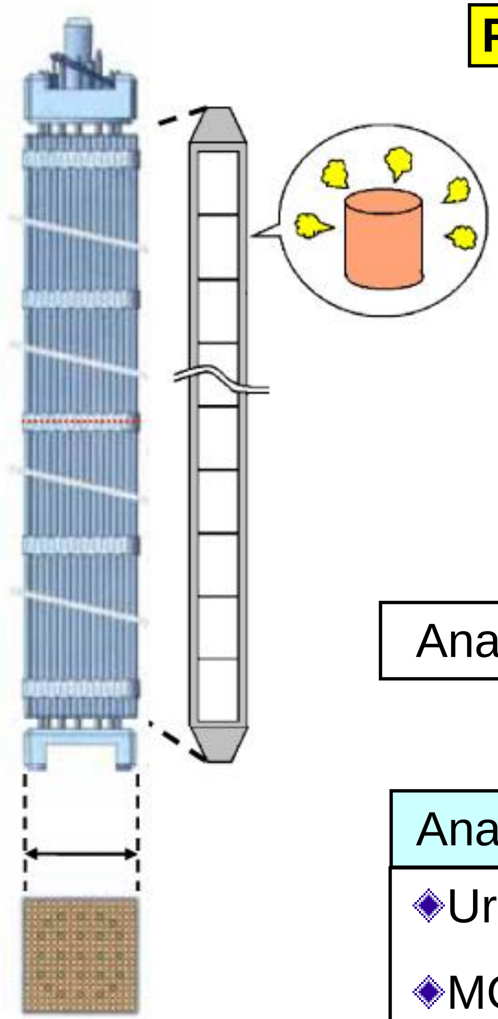
Source:http://www.nisa.meti.go.jp/genshiryoku/doukou/files/onagawa=女川原子力発電所3号機のプルサーマルについて、p20、原子力安全・保安院、平成21年9月

Example of Analysis Results (2)*: Inner Pressure of Fuel Pin

*Tomari NPP (PWR): 912Mwe

<Screening Point>

Isn't a fuel pin damaged by unusual expansion due to the filling fission products (FP) gas within a pin?



Pellet Mechanical Characteristic

Inner Pressure Increasing by Expanding FP Gas

*1:Since a Fuel pin receives high pressure from outside, helium gas is filled within a pin in order to prevent being crushed.

Counter Measure

Adjusting the gas quantity enclosing within a pin (helium gas)*1 adequately

Analysis of Inner Pressure within a Pin

Analysis Results

*2:Criterion of Judgment ≦1.0

- ◈Uranium Fuel: 0.73 (Contrast with Design)*2
- ◈MOX Fuel: 0.80 (Contrast with Design)*2

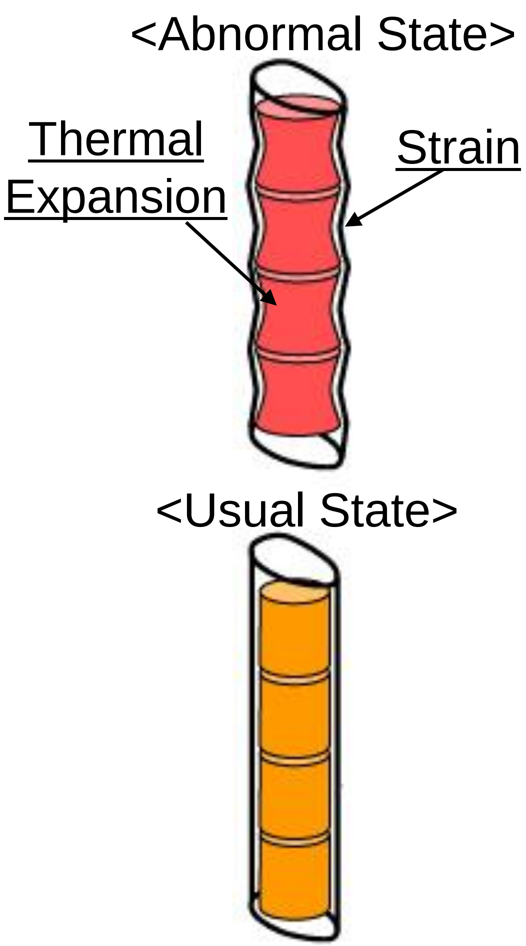
Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の 一次審査結果、P17、平成22年5月、原子力安全・保安院

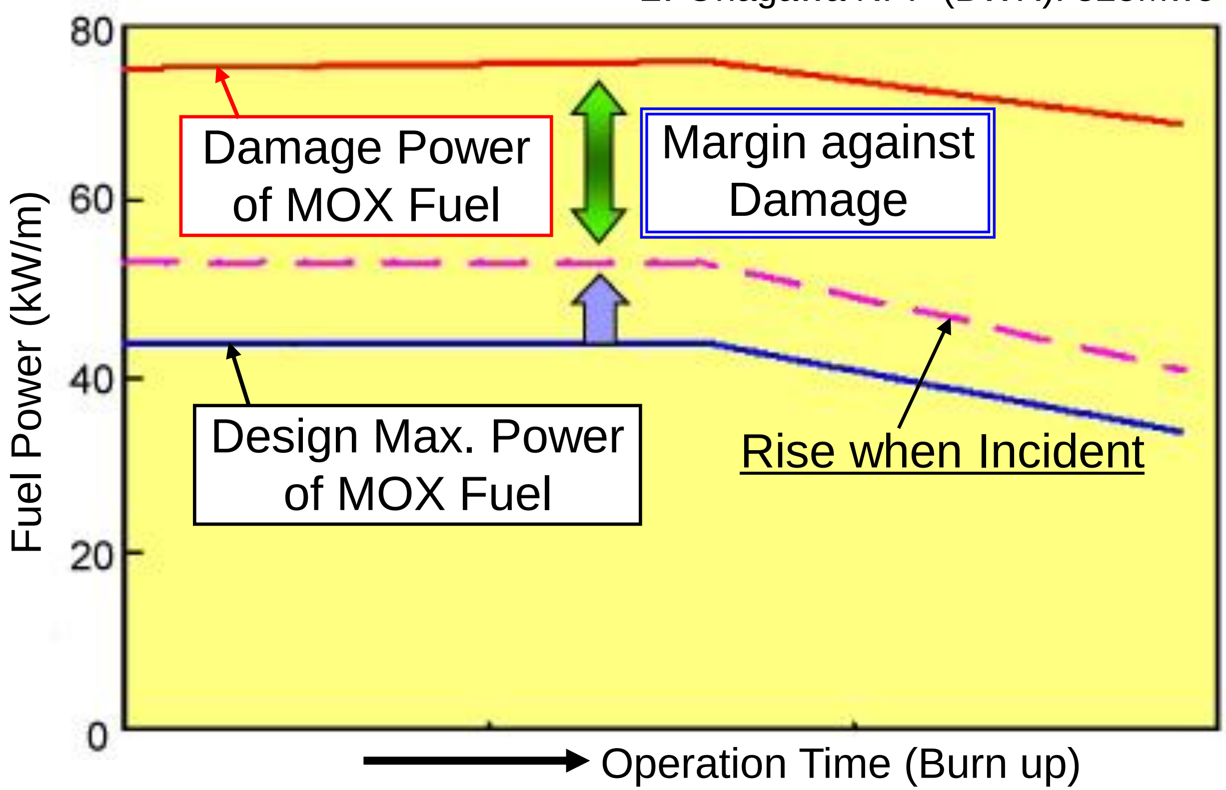
Example of Analysis Results (3): Thermal Stresses of Fuel Cladding Tube

<Screening Point>

Isn't a fuel cladding tube damaged by thermal stress due to the unusual rising up of reactor power?

*1: Onagawa NPP (BWR): 825Mwe





Analysis Results
◈Uranium Fuel: 0.75 (Contrast with Design)*2
◈MOX Fuel: 0.84 (Contrast with Design)*2

*2:Criterion of Judgment ≦1.0

Source:http://www.nisa.meti.go.jp/genshiryoku/doukou/files/onagawa=女川原子力発電所3号機のプルサーマルについて、p21、原子力安全・保安院、平成21年9月

Example of Analysis Results (4)*: Summarization of Mechanical Design of Fuel Assembly

(Representative Example)

Assessment Items		Uranium Fuel		MOX Fuel	
		Calculated Value	Criterion of Judgment	Calculated Value	Criterion of Judgment
Central Max. Temp. of Fuel Pellet	Rated Power	at.1,740ºC	<2,580ºC	at.1,740ºC	<2,500ºC
	Abnormal Operation	at. 2,220ºC		at. 2,230ºC	
Inner Pressure of Fuel Pin		0.73	≦1.0	0.80	≦1.0
Thermal Stress of Fuel Cladding Tube		0.75	≦1.0	0.84	≦1.0
Tensile Strain of Fuel Cladding Tube		0.36%	≦1.0%	0.42%	≦1.0%
Cumulative Fatigue of Fuel Cladding Tube		0.18	≦1.0	0.13	≦1.0

*Tomari NPP (PWR): 2,660MWt (912Mwe)

Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の 一次審査結果、P18、平成22年5月、原子力安全･保安院

1.1 (2) Reactor Core (Nuclear Design)

Main Important Design Points on Core Design

- Even if a control rod which has **Maximum Rod Worth (reactivity) cannot insert** from the whole withdrawing position, a reactor can be maintained in the state of **Sub-Criticality***1 under high temperature condition by remained control rods.

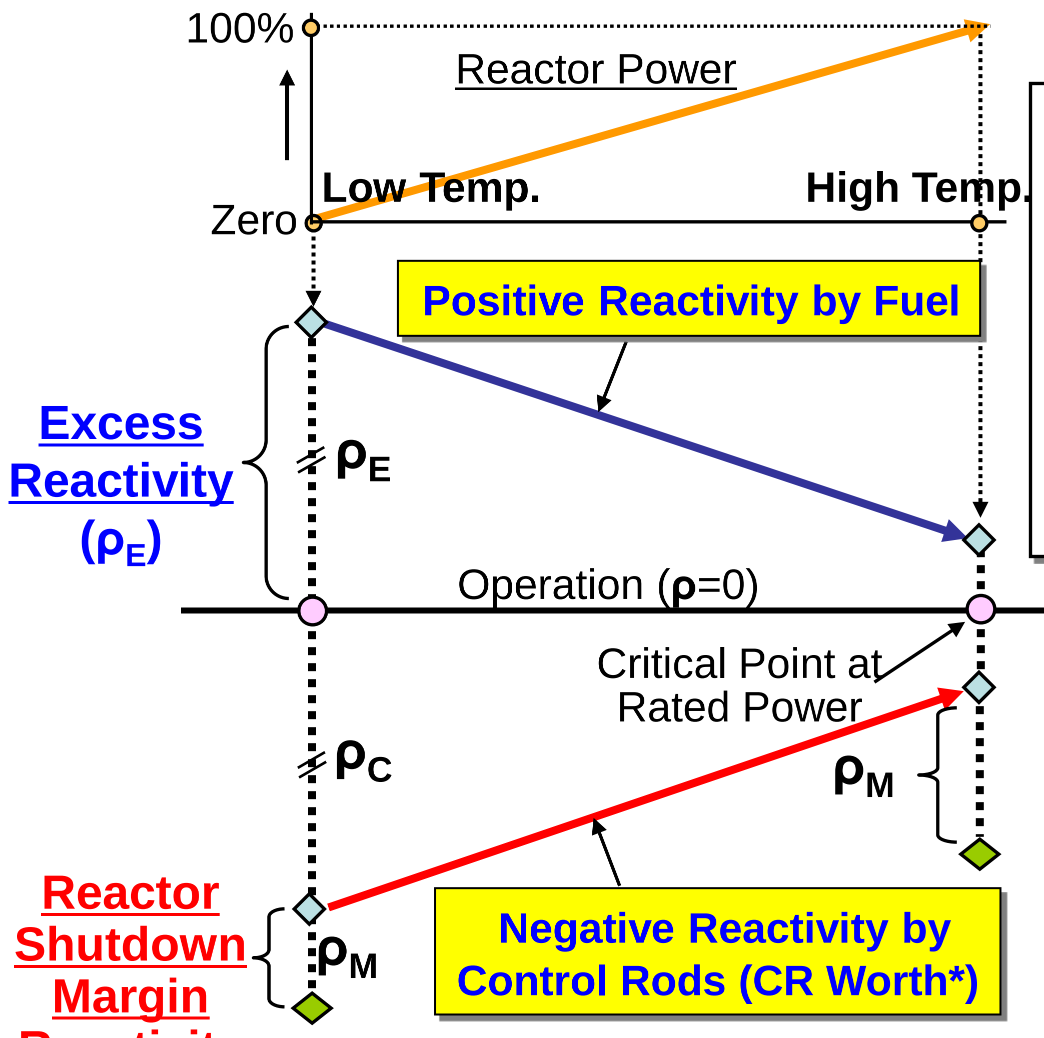
  (*1: **Reactor Shutdown Margin Reactivity >1.8%⊿k/k**)

- Moreover, a core can be kept in the state of sub-criticality under low temperature condition by **Chemical Control System** (Boric Acid).

  (*2: Reactor Shutdown Margin Reactivity >1.0%⊿k/k)

- A core should be designed so that **Doppler Effect** and **Moderator Temperature Effect become negative** and has the inherent rapid power control effect.

Reference: Reactor Shutdown Margin



*CR Worth=ρ_C (ρ_E) +ρ_M

<Excess Reactivity>

The Excess Reactivity decreases accompanying with operation due to the following reasons:

1) Core Conversion Ratio is small, especially in LWR. (at 0.6)

2) Poison FP (^{135}Xe, ^{149}Sm) increases accompanying with operation

<Reactor Shutdown Margin at Rated Power>

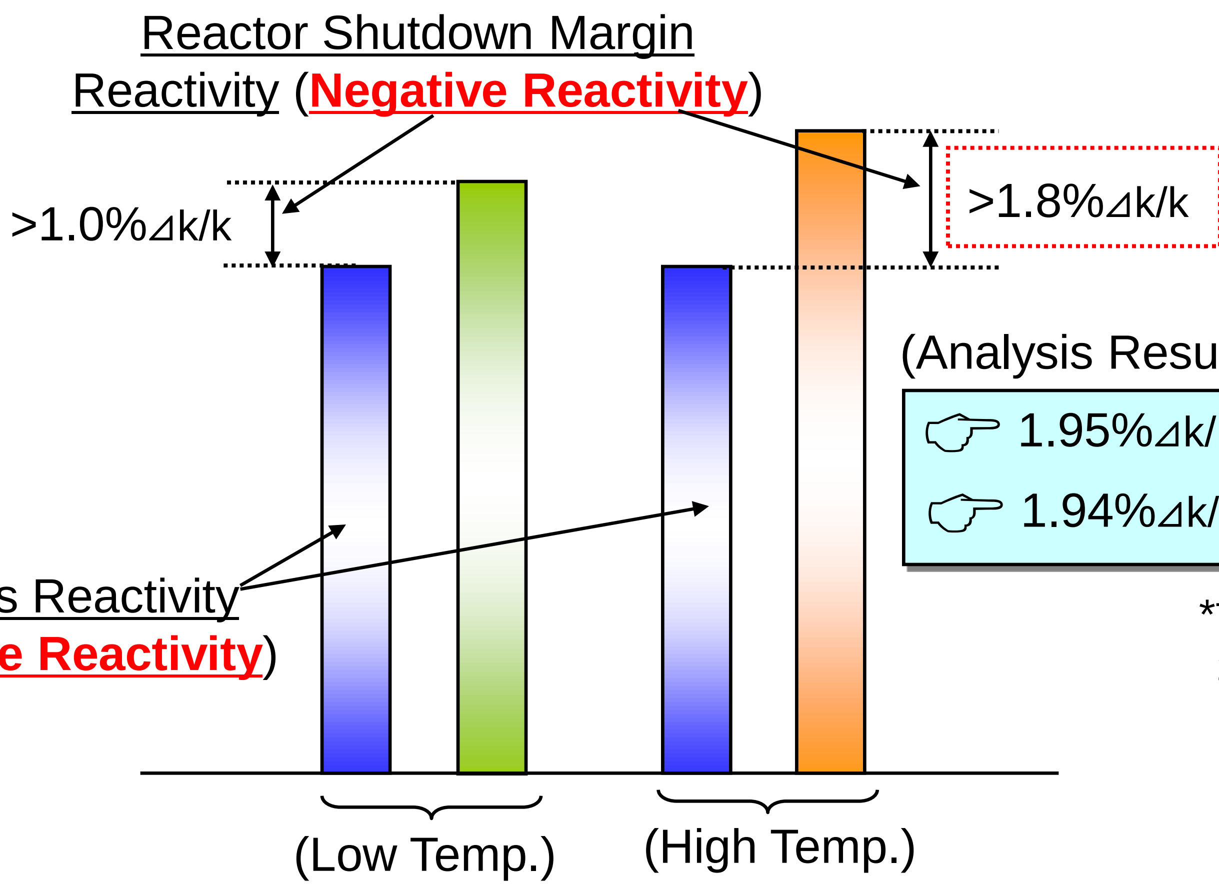
(%⊿k/k)

	CR Worth	ρ_E	ρ_M
PWR	at 31%	29%	1.8%
BWR	at 27%	25%	at 2.0%
FBR	7%	5.6%	1.4%

Example of Analysis Results (1): Reactor Shutdown Margin Reactivity

<Screening Point>

Does a core have enough reactor shutdown capability (Reactor Shutdown Margin Reactivity)?



(Analysis Results)

- 1.95%⊿k/k **(Uranium Fuel)***
- 1.94%⊿k/k **(MOX Fuel)***

*Tomari NPP (PWR): 2,660MWt (912Mwe)

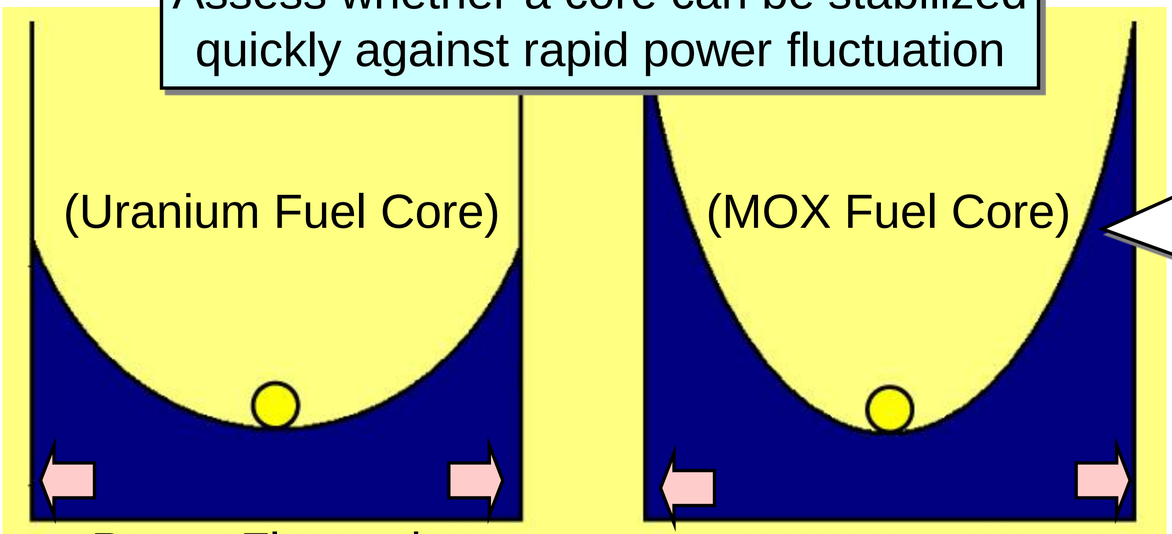
Assess whether a core can be shutdown with having margin safely

Example of Analysis Results (2): **Self Regulating Characteristics (1/2)**

<Screening Point>

When a reactor power is fluctuated rapidly, can a core be returned to a stable state by self regulating characteristics of Doppler Effect and Moderator Temperature Effect?

Assess whether a core can be stabilized quickly against rapid power fluctuation



Power Fluctuation

Power Fluctuation

The Self Regulating Characteristics of MOX fuel core is stronger than uranium fuel core one.

X10^{-5}(⊿k/k/°C)

Item	Uranium Fuel Core	MOX Fuel Core
Doppler Coe.	−3.5～−2.4	−3.6～−2.6
Moderator Temp. Coe.	−66～−7.9	−71～−11.5

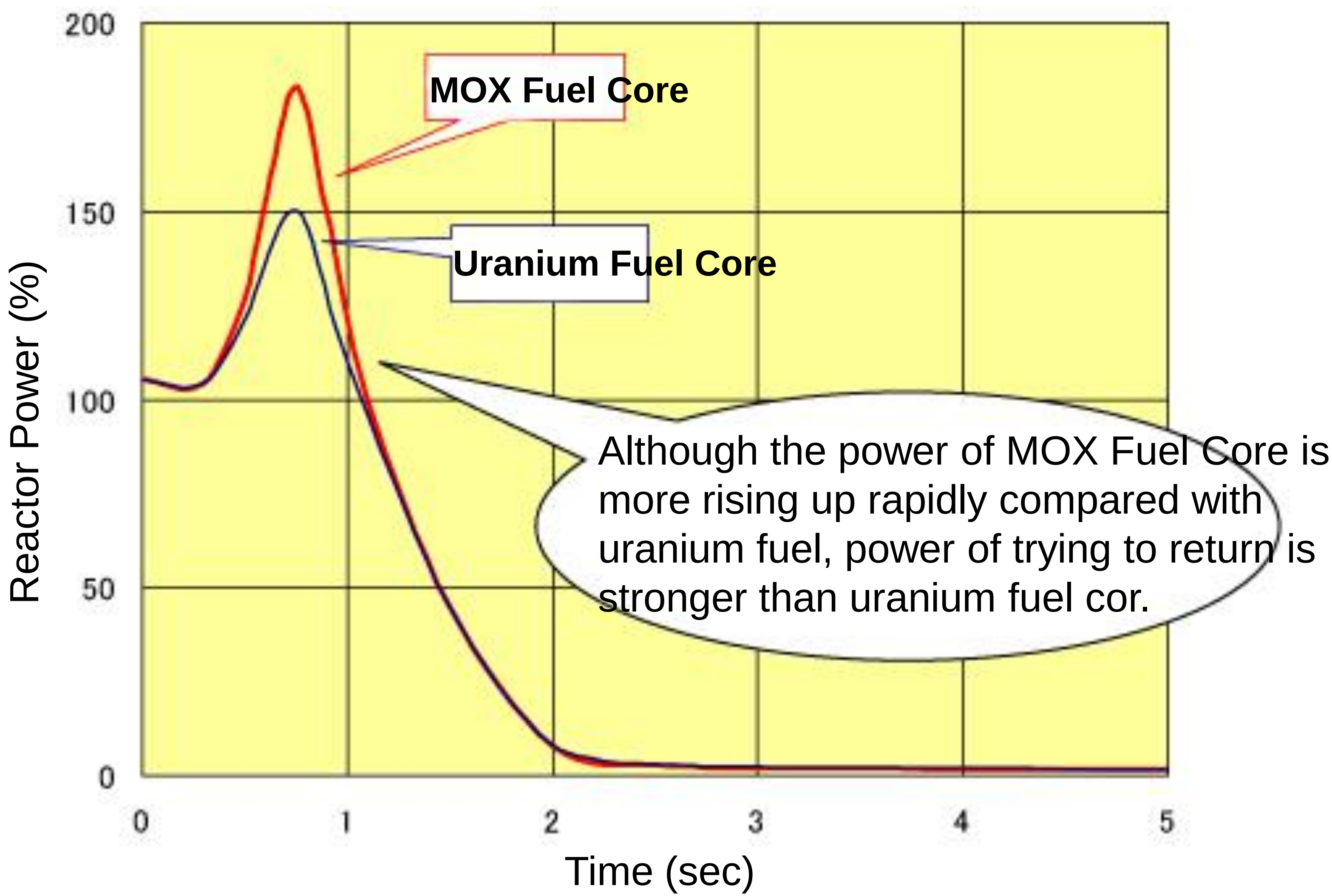
*Tomari NPP (PWR)

Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の 一次審査結果、P23、平成22年5月、原子力安全・保安院

Example of Analysis Results (2)*: **Self Regulating Characteristics (2/2)**

*Onagawa NPP (BWR): 825Mwe

(Analysis Result of Reactor Power Change When Cutting Off of Generator Loading)



Source:http://www.nisa.meti.go.jp/genshiryoku/doukou/files/onagawa=女川原子力発電所3号機のプルサーマルについて、p17、原子力安全・保安院、平成21年9月

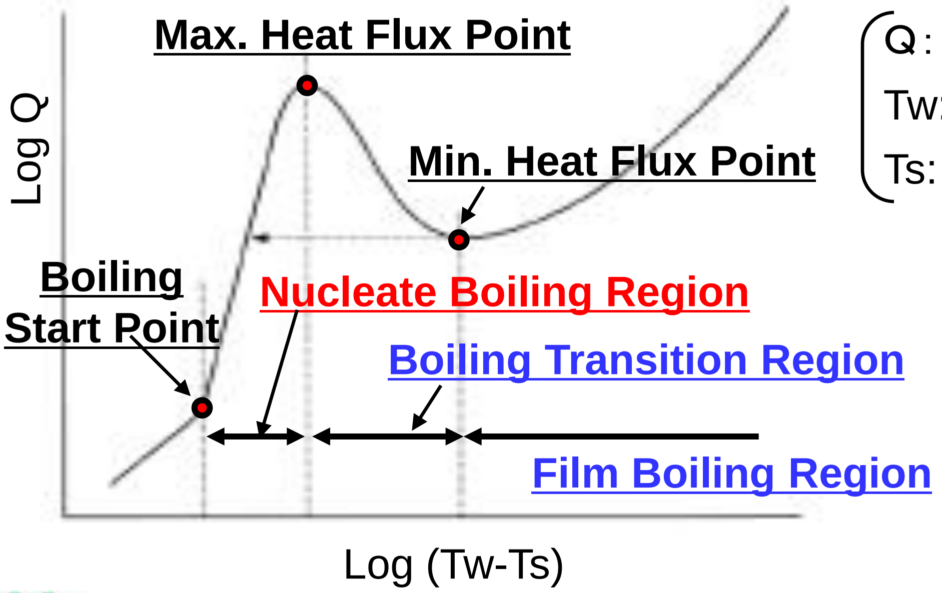
1.1 (3) Reactor Core (Thermal-Hydraulic Design)

Most Important Design Point on Thermal-Hydraulic Design

- The most important point on thermal-hydraulic design is to prevent occurring of "**Boiling Transition**".
- Boiling Transition is the phenomenon moving from **Nucleate Boiling** with good heat transfer to bad **Film Boiling**.
- Whether Boiling Transition occurs or not is judged by the value of "**Departure from Nucleate Boiling Ratio**" (**DNBR**).
- Thermal-hydraulic design of a core should be designed so that Minimum DNBR at rated power becomes bigger than Allowable Limit DNBR.
- DNBR is given as he following:

$$\text{DNBR} = \frac{\left(\begin{array}{l}\text{Heat Flux when fuel cladding tube's temperature begins rising up} \\ \text{rapidly causing by heat transfer between cladding and coolant falls}\end{array}\right)}{\left(\text{Actual Heat Flux}\right)}$$

Boiling Curve and Analysis Result of DNBR

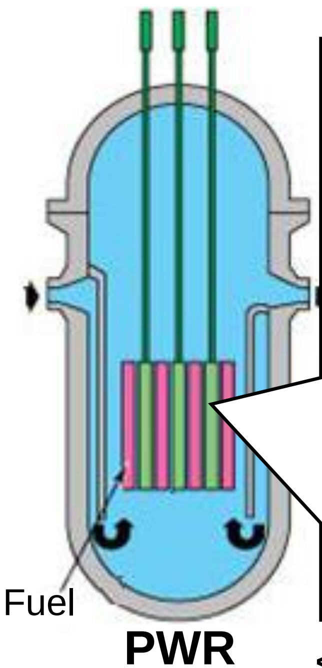


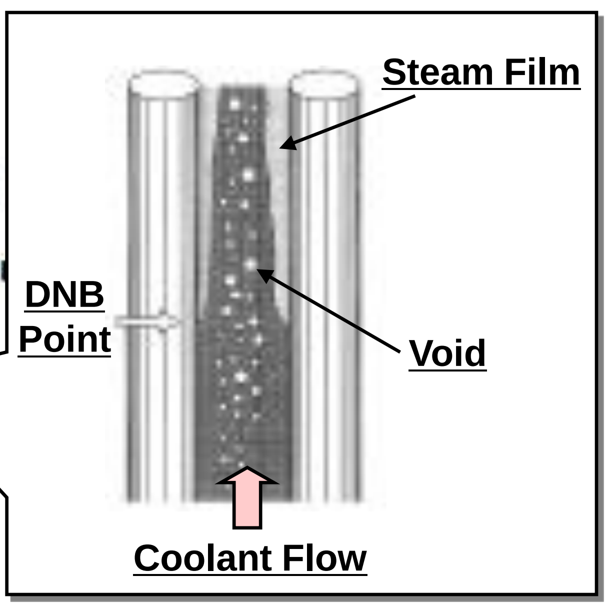
Q : Heat Flux at Heating Plane
Tw:Temp. at Heating Plane
Ts: Saturation Temp.

(Analysis Results)

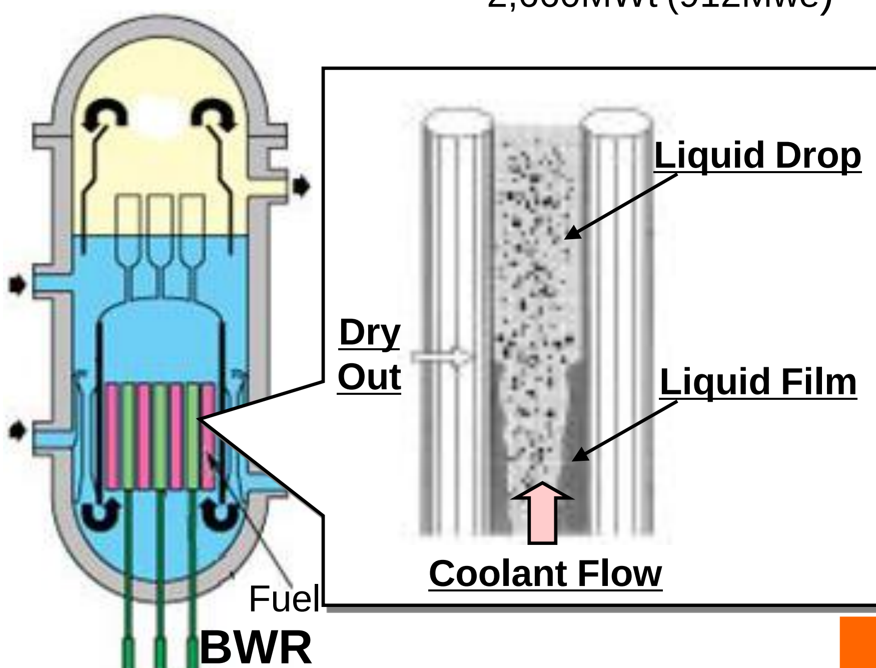
☞ Min. DNBR=2.15
>Allowable Limit DNBR=1.42

*Tomari NPP (PWR):
2,660MWt (912Mwe)





<Image of Boiling Transition>



Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant

1.1 Reactor Core

(Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)

1.2 Emergency Core Cooling System (ECCS)

1.3 Fuel Storage Facility

2. Exposure Evaluation of General Public around NPP

3. Analysis of Abnormal Transient Change (Incident) during Normal Operation

4. Analysis of Accident

5. Analysis of Hypothetical Accident for Siting Evaluation

6. Technical Capability

1.2 Emergency Core Cooling System (ECCS)

<Screening Point>

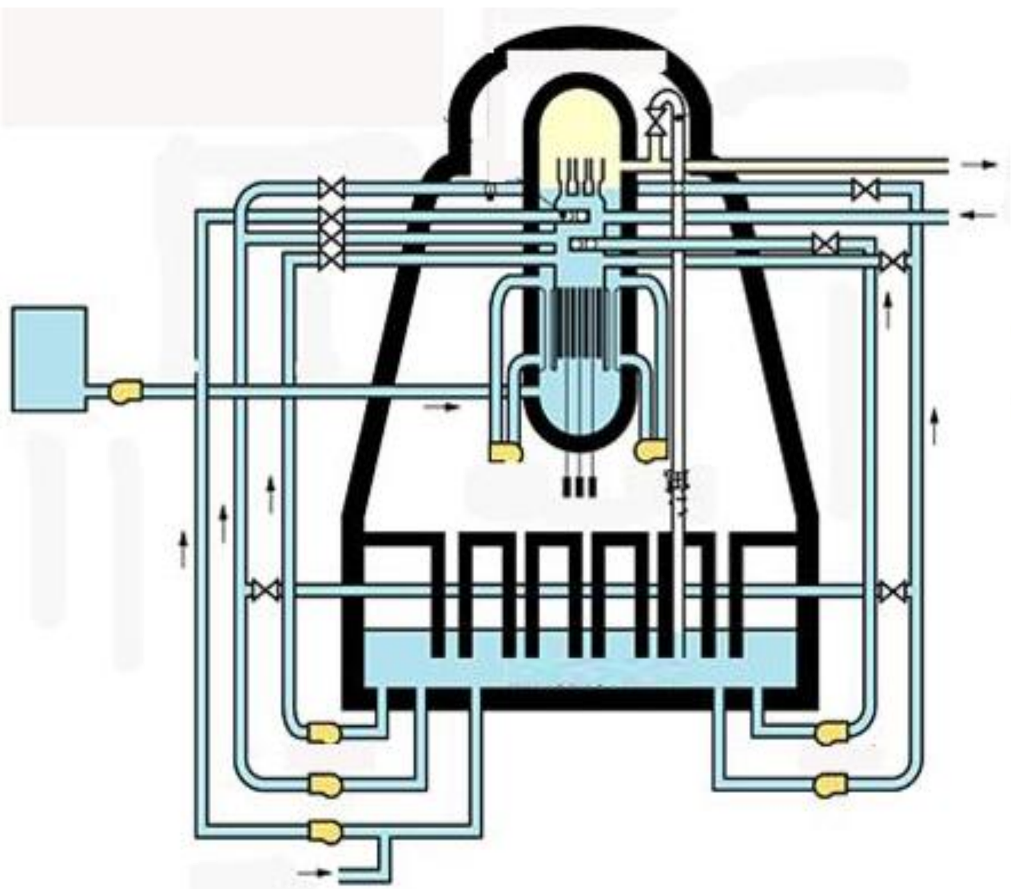
- ECCS should be a design which can prevent a serious damage of fuel for the accident of loss of coolant and moreover can restrict a reaction between fuel cladding metal and a water to a small quantity sufficiently
- ECCS should have redundancy or diversity and independency in order to perform its safety function completely even if an external power cannot be used in addition to assumption of the single failure of an equipment.
- ECCS should be a design which can conduct test and inspection periodically and can carry out test and inspection of each system independently for assuring its soundness and redundancy.

(Reference: For MOX Fuel Core)

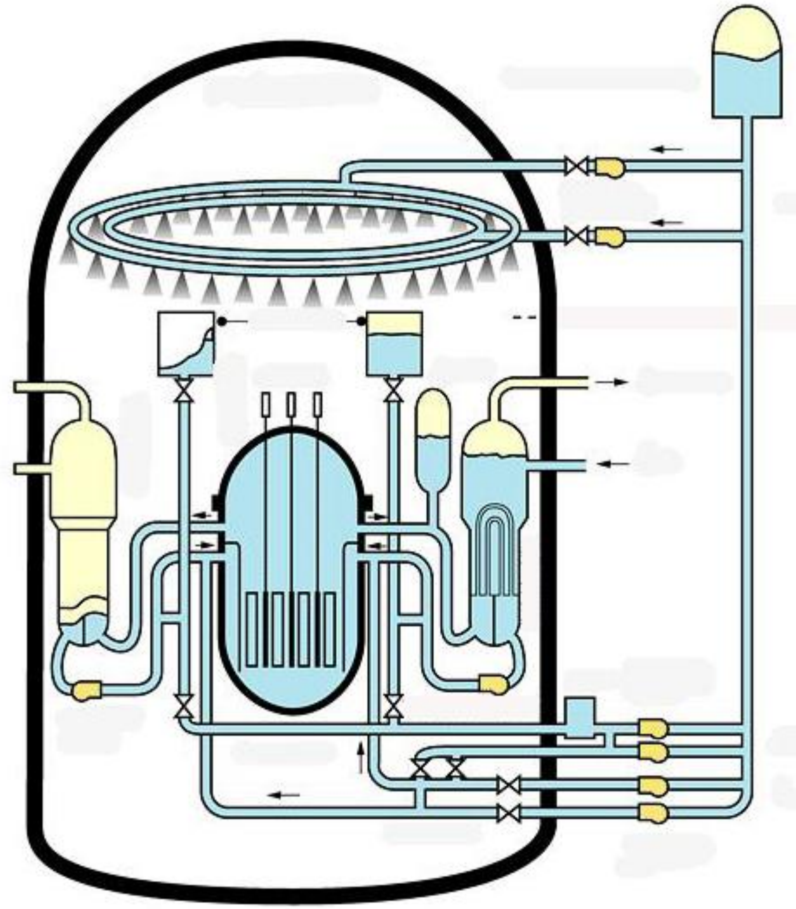
- Since **Reactivity Worth of Boric Acid Solution** is fallen by using plutonium, its concentration has to be increased.

  ☞ Example: From **3,000ppm** to ≧**3,200ppm** at Tomari NPP

Emergency Core Cooling System (ECCS) in BWR and PWR



<ECCS in BWR>



<ECCS in PWR>

Source: "Genshiroku 2007", 5shou Genshiryoku shisetsu no anzen kakuho, `62, 34th version, June 2007

Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant

1.1 Reactor Core

(Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)

1.2 Emergency Core Cooling System (ECCS)

1.3 Fuel Storage Facility

2. Exposure Evaluation of General Public around NPP

3. Analysis of Abnormal Transient Change (Incident) during Normal Operation

4. Analysis of Accident

5. Analysis of Hypothetical Accident for Siting Evaluation

6. Technical Capability

Example of Analysis Results*: Fuel Storage of MOX Fuel

<Screening Point>

Can Decay Heat of MOX Fuel stored in spent fuel storage pool be removed safely?

*Onagawa NPP (BWR): 825Mwe

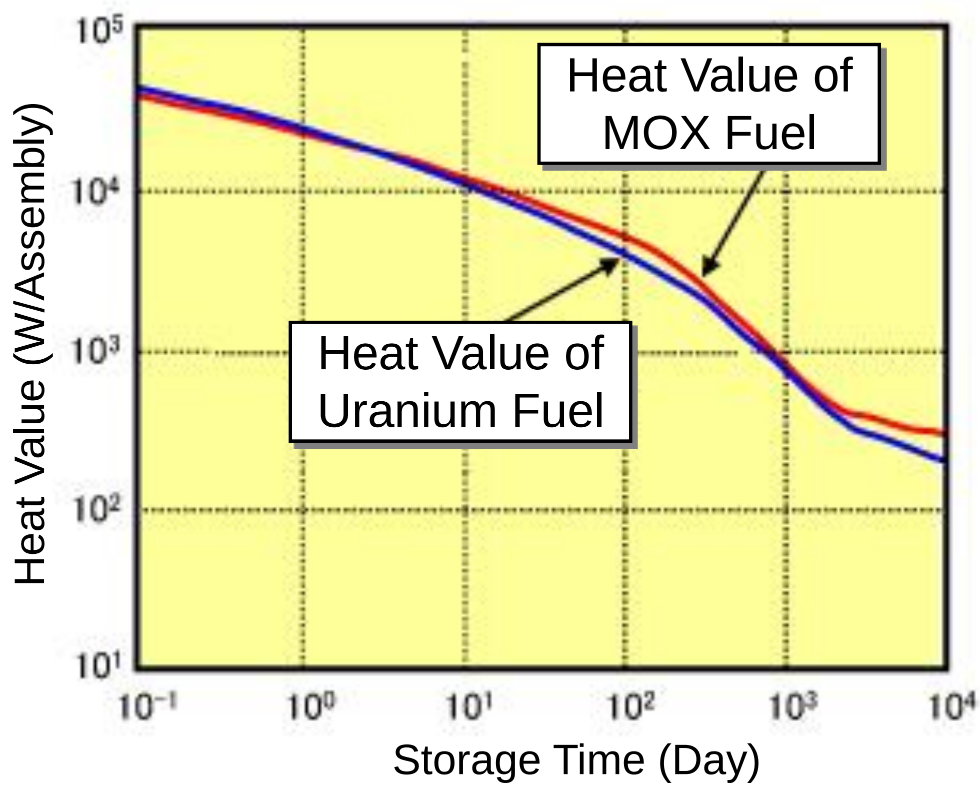
Decay Heat of MOX Fuel is larger than Uranium Fuel

↓

Analysis of Pool Temp. in the Most Severe Condition (A Pool is altogether occupied with MOX Fuels.)

↓

It was confirmed that MOX Fuel can be stored safely.



	Water Temp. of Pool	Standard Value
Storage of MOX Fuel	at 54ºC	65ºC

Source:http://www.nisa.meti.go.jp/genshiryoku/doukou/files/onagawa=女川原子力発電所3号機のプルサーマルについて、p24、原子力安全・保安院、平成21年9月

Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant
   - 1.1 Reactor Core
     (Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)
   - 1.2 Emergency Core Cooling System (ECCS)
   - 1.3 Fuel Storage Facility
2. **Exposure Evaluation of General Public around NPP**
3. Analysis of Abnormal Transient Change (Incident) during Normal Operation
4. Analysis of Accident
5. Analysis of Hypothetical Accident for Siting Evaluation
6. Technical Capability

Dose Evaluation of General Public around NPP under Normal Operation

Screening Item

1) Confirmation that the Dose which the General Public receives under normal operation can fully be less than the **Allowable Dose defined by low**. (**≦1mSv per year**)

2) Confirmation that the Dose Evaluation Value can be less than the **Dose Target Value***. (≦50μSv per year)

*Dose Target Value is defined in order to reduce rationally the dose general public receiving as much as possible.

<Sample Evaluation Result>

It was confirmed that Tomari NPP is designed that the following both values can be satisfied:

◈Allowable Dose at Out of Circumference Observing Sector defined by Law is sufficiently less than 1mSv/year.

Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant
   1.1 Reactor Core
   (Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)
   1.2 Emergency Core Cooling System (ECCS)
   1.3 Fuel Storage Facility
2. Exposure Evaluation of General Public around NPP
3. **Analysis of Abnormal Transient Change (Incident) during Normal Operation**
4. Analysis of Accident
5. Analysis of Hypothetical Accident for Siting Evaluation
6. Technical Capability

Analysis Events for Incident Evaluation

(1) Abnormal Change of Reactivity or Power Distribution in a Core

- Unusual Withdrawing of a Control Rod during Reactor Startup Operation
- Unusual Withdrawing of a Control Rod during Rated Power Operation
- Unusual Dilution of a Boric Acid Solution in a Coolant

(2) Abnormal Change of Heat Generation or Heat Removal in a Core

- Loss of Commercial Power Line (External Power)
- Loss of Feed Flow
- Unusual Increasing of Steam Load
- Unusual Pressure Drop of Secondary Cooling System
- Excess Feed into Steam Generator

(3) Abnormal Change of Coolant Pressure or Coolant Possession Amount

- Miss Startup of ECCS during Rated Operation

Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant
   1.1 Reactor Core
   (Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)
   1.2 Emergency Core Cooling System (ECCS)
   1.3 Fuel Storage Facility
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3. Analysis of Abnormal Transient Change (Incident) during Normal Operation
4. **Analysis of Accident**
5. Analysis of Hypothetical Accident for Siting Evaluation
6. Technical Capability

Analysis Events for Accident Evaluation

(1) Loss of Coolant or Remarkable Change of Cooling State

- Loss of a Reactor Coolant
- Axial Sticking of a Primary Cooling Circulation Pump
- Rupture of Main Feed Piping
- Rupture of Main Steam Piping

(2) Abnormal Input of Reactivity or Rapid Change of Reactor Power

- Rapid Withdrawing of a Control Rod

(3) Unusual Releasing of Radioactive Materials into Environment

- Damage of Radioactive Waste Disposal Facility
- Rapture of Heat Transfer Tube of Steam Generator
- Loss of a Reactor Coolant

(4) Abnormal Change of Pressure or Environment of Containment Vessel

- Loss of a Reactor Coolant
- Generating of Combustible Gas

Example of Analysis Results (1): Loss of Coolant in BWR (Influence to Circumference)

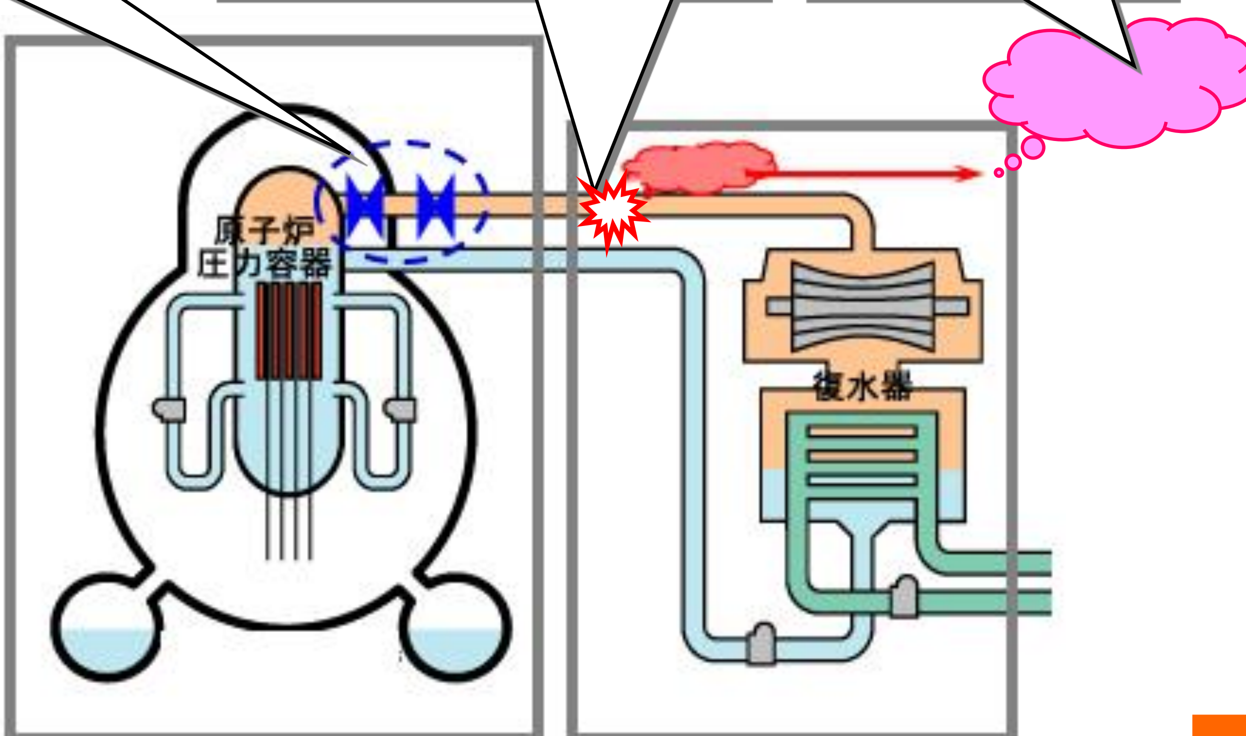
<Screening Point>

Doesn't an accident affect to environment even if radioactive materials are released into environment by rupturing of main steam piping?

(3) Blowout of steam is stopped by closing of isolation valves.

(1) Contaminated steam are flows out into facility by rupturing of main steam piping.

(2) Radioactive materials are released into environment.



Analysis Results

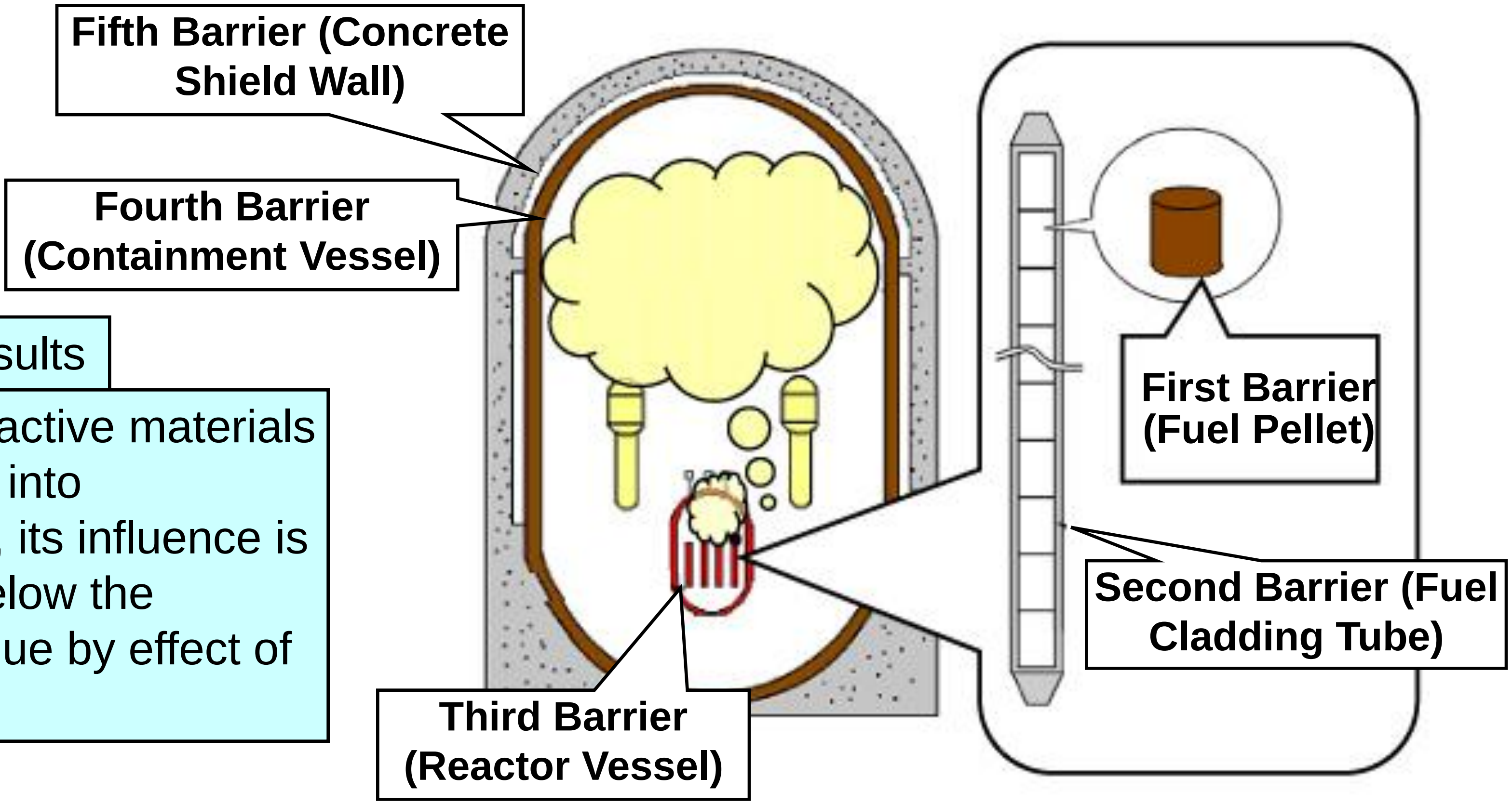
Even if radioactive materials are released into environment by rupturing of main steam piping, dose rate will be controlled to slight 0.09mSv.

Source:http://www.nisa.meti.go.jp/genshiryoku/doukou/files/onagawa=女川原子力発電所3号機のプルサーマルについて、p22、原子力安全・保安院、平成21年9月

Example of Analysis Results (1): Loss of Coolant in PWR (Influence to Circumference)

<Screening Point>

Doesn't an accident affect to environment even if radioactive materials are released into environment by leaking of primary coolant into out of the system?



Analysis Results

Even if radioactive materials are released into environment, its influence is restrained below the allowable value by effect of five barriers.

- ◆Effective Dose Rate: at 0.23mSv≦5mSv (for Rupture of Primary Piping)
- ◆Effective Dose Rate: at 0.29mSv≦5mSv (for Rupture of SG Heat Transfer Tube)

Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の一次審査結果、P40、平成22年5月、原子力安全・保安院

Example Concrete Screening Items in Safety Assessment

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   1.1 Reactor Core
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   1.2 Emergency Core Cooling System (ECCS)
   1.3 Fuel Storage Facility
2. Exposure Evaluation of General Public around NPP
3. Analysis of Abnormal Transient Change (Incident) during Normal Operation
4. Analysis of Accident
5. **Analysis of Hypothetical Accident for Siting Evaluation**
6. Technical Capability

Analysis of Hypothetical Accident for Siting Evaluation

<Analysis Events>

1) Rupture of SG Heat Transfer Tube (As for Major Accident)
2) Loss of Coolant (AS for Hypothetical Accident)

<Guide for Judgment>

Major Accident

- For Thyroid Gland (Child): 1.5Sv
- For Whole Body: 0.25Sv

Hypothetical Accident

- For Thyroid Gland (Adult): 3.0Sv
- For Whole Body: 0.25Sv

<Analysis Results>

Analysis Events	Parts	Analysis Results	Standard
Major Accident	Thyroid Gland (Child)	at. $1.3\text{x}10^{-2}$Sv	1.5Sv
	Whole Body	at. $3.5\text{x}10^{-4}$Sv	0.25Sv
Hypothetical Accident	Thyroid Gland (Adult)	at. $1.3\text{x}10^{-1}$Sv	3.0Sv
	Whole Body	at. $1.1\text{x}10^{-2}$Sv	0.25Sv

Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の 一次審査結果、P42、平成22年5月、原子力安全･保安院

Example Concrete Screening Items in Safety Assessment

1. Safety Design of Reactor Plant
   1.1 Reactor Core
   (Mechanical Design, Nuclear Design, Thermal-Hydraulic Design)
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   1.3 Fuel Storage Facility
2. Exposure Evaluation of General Public around NPP
3. Analysis of Abnormal Transient Change (Incident) during Normal Operation
4. Analysis of Accident
5. Analysis of Hypothetical Accident for Siting Evaluation
6. **Technical Capability**

Analysis of Hypothetical Accident for Siting Evaluation

<Analysis Items>

1) Organization for Design and Construction

2) Provision of Engineers for Design and Construction

3) Experience of Design and Construction

4) Quality Assurance Activity related to Design and Construction

5) Organization for Operation and Maintenance

6) Provision of Engineers for Operation and Maintenance

7) Experience for Operation and Maintenance

8) Quality Assurance Activity related to Operation and Maintenance

9) Education and Training for Engineer

10) Appointment and Arrangement of Qualified Persons

Source: http://www.pref.hokkaido.lg.jp/泊発電所3号機プルサーマル計画に係る原子炉設置変更許可変更申請の 一次審査結果、P44、平成22年5月、原子力安全・保安院

6. Comparison with Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

The contents of attachment material of application for installation permission of NPP mentioned in the next page is mostly in accordance with "**Standard Review Plan for the Review of Safety Analysis Reports for NPP: LWR Edition**" by U.S.NRC (USA Nuclear Regulatory Committee).
http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800

Contents of Safety Analysis Reports (1/2)

- ✓Captrer-1: Introduction and Interfaces → Attachment-1-5 & 7
- ✓Cahpter-2: Sites Characteristics and Site Parameters → **Attachment-6 (Climate, Foundation, Earthquake…)**
- ✓Chapter-3: Design of Structure, Components, Equipments and Systems (→See sample in p20)
- ✓Chapter-4: Reactor (→See sample in p22)
- ✓Chapter-5: Reactor Coolant System, and Connected System
- ✓Chapter-6: Engineered Safety Features
- ✓Chapter-7: Instrumentation and Controls
- ✓Chapter-8: Electric Power
- ✓Chapter-9: Auxiliary Systems
- ✓Chapter-10: Steam and Power Conversion System

Attachment-8 (Safety Design)

Contents of Safety Analysis Reports (2/2)

- ✓Chapter-11: Radioactive Waste Management
- ✓Chapter-12: Radiation Protection

→ **Attachment-9 (Radiation Management & Disposal of Radioactive Waste)**

- ✓Chapter-13:Conduct of Operators → **Operational Safety Program**
- ✓Chapter-14: Initial Test Program and ITACC-Design Certification
- ✓Chapter-15:Transient and Accident Analysis → **Attachment-10 (Safety Assessment)**
- ✓Chapter-16: Technical Specifications
- ✓Chapter-17: Quality Assurance → **Operational Safety Program**
- ✓Chapter-18: Human Factors Engineering → **Attachment-8 (Safety Design)**
- ✓Chapter-19: Severe Accidents → **Attachment-10 (Safety Assessment)**

(Remark)
Please **refer to appendix-1** for the **sample of safety analysis** report based on the standard review plan for the review of safety analysis reports for NPP: LWR edition (U.S.NRC)

Sample: Chapter-3 Design of Structure, Components, Equipments & Systems

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition — Design of Structures, Components, Equipment, and Systems (NUREG-0800, Chapter 3)

The following links on this page are to documents in our Agencywide Documents Access and Management System (ADAMS). ADAMS documents are provided in either Adobe Portable Document Format (PDF) or Tagged Image File Format (TIFF). To obtain free viewers for displaying these formats, see our Plugins, Viewers, and Other Tools. If you have problems with viewing or printing documents from ADAMS, please contact the Public Document Room staff.

Section	Title	Rev.	Date Updated
3.2.1	Seismic Classification	Rev. 2	03/2007
		Draft Rev. 2	06/1996
	see next page	Rev. 1	07/1981
		Rev. 0	11/1975
3.2.2	System Quality Group Classification	Rev. 2	03/2007
		Draft Rev. 2	06/1996
		Rev. 1	07/1981
		Rev. 0	11/1975
3.3.1	Wind Loadings	Rev. 3	03/2007
		Draft Rev. 3	06/1996
		Rev. 2	07/1981
		Rev. 1	08/1978
		Rev. 0	11/1975
3.3.2	Tornado Loadings	Rev. 3	03/2007
		Draft Rev. 3	06/1996
		Rev. 2	07/1981
		Rev. 1	08/1978

source:http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/

Sample: Chapter-3 Design of Structure, Components, Equipments & Systems

NUREG-0800

UNITED STATES NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

3.2.1 SEISMIC CLASSIFICATION

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - Organizations responsible for the review of component performance and testing.

I. AREAS OF REVIEW

General Design Criterion (GDC) 2 of 10 CFR Part 50, Appendix A, in part, requires that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The earthquake against which these plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which SSCs important to safety are designed to remain functional. Appendix S also requires consideration of surface deformation. Those plant features that are designed to remain functional if an SSE occurs are designated Seismic Category I in Regulatory Guide (RG) 1.29.

source:http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/

Sample: Chapter-4 Reactor

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition — Reactor (NUREG-0800, Chapter 4)

The following links on this page are to documents in our Agencywide Documents Access and Management System (ADAMS). ADAMS documents are provided in either Adobe Portable Document Format (PDF) or Tagged Image File Format (TIFF). To obtain free viewers for displaying these formats, see our Plugins, Viewers, and Other Tools. If you have problems with viewing or printing documents from ADAMS, please contact the Public Document Room staff.

Section	Title	Rev.	Date Updated
4.2	Fuel System Design	Rev. 3	03/2007
		Draft Rev. 3	06/1996
		Rev. 2	07/1981
		Rev. 1	09/1978
		Rev. 0	11/1975
4.3	Nuclear Design	Rev. 3	03/2007
		Draft Rev. 3	06/1996
		Rev. 2	07/1981
		Rev. 1	04/1978
		Rev. 0	11/1975
4.4	Thermal and Hydraulic Design	Rev. 2	03/2007
		Draft Rev. 2	06/1996
		Rev. 1	07/1981
		Rev. 0	11/1975
4.5.1	Control Rod Drive Structural Materials	Rev. 3	03/2007
		Draft Rev. 3	06/1996
		Rev. 2	07/1981
		Rev. 1	01/1978

see next page

source:http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/

Sample: Chapter-4 Reactor

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - The organization responsible for the review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

The organization responsible for the review of transient and accident analyses evaluates the thermal, mechanical, and materials design of the fuel system. The fuel system consists of arrays (assemblies or bundles) of fuel rods, including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. This section discusses the reactivity control elements of the control rods that extend from the coupling interface of the control rod drive mechanism into the core.

The fuel system safety review provides assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the

Appendix

Preliminary Safety Analysis Report (PSAR)

(Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition)

- Country: Taiwan
- Power Plant: Longmen N.P.P (ABWR) No.1 & No.2
- Output Power: 2,700 MWe (1,350MWe/unit)
- Start of Commercial Operation: December 2011 (Unit-No.1)

  December 2012 (Unit-No.2)

Implementation of Preliminary Safety Analysis based on LWR Edition (NUREG-0800, Formerly issued as NUREG-75/087)

http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (1/10)

Safety Analysis Reports

Nuclear power plant operators, i.e. licensees, are expected to maintain updated safety analysis reports on how the plant is designed, operated, and maintained. Initially, a Preliminary Safety Analysis Report (PSAR) must be filed with the regulatory agency. The agency reviews the information, asks questions, and may specify changes in the plant design or additional requirements. After the plant receives its operating license, the report is often referred to as the Final (or Updated) Safety Analysis Report.

Updated copies of these reports are maintained in the local public document center, usually a public library near the plant. In the United States, the safety analyses reports are reviewed using NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, as a guide. NUREG-0800 identifies the requirements that the NRC expects the power plant operator to comply with.

No Nukes Asia Forum Japan has graciously provided the Lungmen Power Plant Units 1 and 2 Preliminary Safety Analysis Report on their website. The Table of Contents links below lead to the various chapters of the report that are available. This report is typical of the information that a utility is expected to provide to the regulatory agency. In this specific case, the Taiwan nuclear laws require compliance with the same US Nuclear Regulatory Commission standards as US plants must meet.

The table of contents below provides additional information unavailable on the linked site. Total volume of the PSAR is about 87 MB and includes hundreds of drawings and tables. Practically all the documents are PDF files that can be read using Adobe Acrobat. In a few cases, the documents are DOC files that can be read using Microsoft Word. The DOC type documents are identified in the table of contents (e.g. Chapter 11).

Complete PSAR Table of Contents and Acronyms

Preliminary Safety Analysis Report (PSAR)

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (2/10)

1.0 Introduction and General Description of Plant

2.0 Site Characteristics

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (3/10)

3.0 Design of Structures, Components, Equipment and Systems

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (4/10)

4.0 Reactor

5.0 Reactor Coolant System and Connected Systems

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (5/10)

6 Engineered Safety Features

7.0 Instrumentation and Control Systems

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (6/10)

8.0 Electric Power

9.0 Auxiliary Systems

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (7/10)

10.0 Steam and Power Conversion System

11.0 Radioactive Waste Management

12.0 Radiation Protection

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (8/10)

13.0 Conduct of Operations

14.0 Initial Test Program

15.0 Accident and Analysis

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (9/10)

16.0 Preliminary Technical Specifications (PSAR)

Volume 2 Technical Specification Bases

17 Quality Assurance

Attachments 17A, 17B, 17C
17A TPC Nuclear Projects Quality Assurance Program
17B Quality Assurance Program for Lungmen Nuclear Power Project (GE)
17C Quality Assurance Program for Lungmen Nuclear Power Project (S&W)

18.0 Human Factors Engineering

Chapter 18
18.0 Human Factors Engineering
18.1 Introduction
18.2 Design Goals and Design Bases
18.3 Planning, Development, and Design
18.4 Control Room Standard Design Features
18.5 Remote Shutdown System
18.6 Systems Integration
18.7 Human-System Interface (HSI) Design Implementation

source:http://www.nucleartourist.com/psar/index.html

Safety Analysis Reports (10/10)

19.0 Severe Accident Analysis

Appendices

source:http://www.nucleartourist.com/psar/index.html