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Subject: Submittal of Technical Report "Phenomena Identification and Ranking Tables (PIRTs) for the 4S and Further Investigation Program - Loss of Offsite Power, Sodium Leakage from Intermediate Piping, and Failure of a Cavity Can Events - "

Enclosed is a copy of the non-proprietary "Phenomena Identification and Ranking Tables (PIRTs) for the 4S and Further Investigation Program - Loss of Offsite Power, Sodium Leakage from Intermediate Piping, and Failure of a Cavity Can Events - " for the 4S (Super-Safe, Small and Simple) reactor plant that is currently the subject of a pre-application review among NRC, Toshiba, and its 4S affiliates including Japan's Central Research Institute for Electric Power Industry (CRIEPI).

The pre-application review for the 4S reactor commenced in the fourth quarter of 2007. Pre-application review meetings were held among NRC, Toshiba and the 4S affiliates in October 2007, and February, May and August 2008.

Additional technical reports pertaining to the 4S design will be submitted as the pre-application review progresses. If you have any questions regarding this document, please contact Mr. Tony Grenci of Westinghouse at (623) 271-9992, or grencit@westinghouse.com.

Very truly yours,

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Enclosures: Technical Report "Phenomena Identification and Ranking Tables (PIRTs) for the 4S and Further Investigation Program - Loss of Offsite Power, Sodium Leakage from Intermediate Piping, and Failure of a Cavity Can Events - "

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Phenomena Identification and Ranking Tables (PIRTs) for the 4S and Further Investigation Program

Loss of Offsite Power, Sodium Leakage from Intermediate Piping, and Failure of a Cavity Can Events

May 2010

TOSHIBA CORPORATION

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LIST OF ACRONYMS AND ABBREVIATIONS

4S	Super-Safe, Small and Simple
AC	air cooler
ANL	Argonne National Laboratory
AOO	anticipated operational occurrence
ATWS	anticipated transients without scram
BDBA	beyond-design-basis accident
BOP	balance of plant
CDF	cumulative damage fraction
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CPT	component performance test
CRBR	Clinch River Breeder Reactor
CRIEPI	Central Research Institute of Electric Power Industry
CSAU	Code Scaling, Applicability and Uncertainty Evaluation Methodology
DA	Design Approval
DBA	design basis accident
DBE	design basis event
DRACS	direct reactor auxiliary cooling system
EBR	experimental breeder reactor
ECCS	emergency core cooling system
EMDAP	Evaluation Model Development and Assessment Process
EMF	electromagnetic flowmeter
EMP	electromagnetic pump
ESFAS	engineered safety features actuation system
FoM	figure of merit
FCC	failure of a cavity can
FFTF	Fast Flux Test Facility
FP	fission product
FR	fast reactor
GV	guard vessel
H, M, L	high, medium, low
I&C	Instrumentation and control
IC	inner core
IET	integral effects test
IHX	intermediate heat exchanger
IHTS	intermediate heat transport system
IRACS	intermediate reactor auxiliary cooling system
IRAP	independent review and advisory panel
ISI	in-service inspection
K, P, U	known, partially known, unknown
LMFR	liquid metal fast reactor
LMP	Larson-Miller parameter
LOCA	loss-of-coolant accident
LOF	loss of flow

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

LOSP	loss of offsite power
LWR	light water reactor
MC	middle core
MG	motor generator
MLD	master logic diagram
NRC	Nuclear Regulatory Commission
OC	outer core
PBC1	peak value at the 1st phase of base case
PBC2	peak value at the 2nd phase of base case
PCT	peak cladding temperature
PHTS	primary heat transport system
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
PRISM	Power Reactor Inherently Safe Module
PSAC1	peak value at the 1st phase of sensitivity analysis case
PSAC2	peak value at the 2nd phase of sensitivity analysis case
QS	quantitative standard
RHRS	residual heat removal system
RPS	reactor protection system
RV	reactor vessel
RVACS	reactor vessel auxiliary cooling system
S/A	subassembly
SBWR	simplified boiling water reactor
SET	separate effects test
SG	steam generator
SLIP	sodium leakage from intermediate piping
SoK	state of knowledge
SR	shutdown rod
SRP	Standard Review Plan
SSC	structures, systems, and component
SWR	steam-water reaction
ULOF	unprotected loss of flow
ULOHS	unprotected loss of heat sink
UTOP	unprotected transient of overpower

EXECUTIVE SUMMARY

Introduction

TOSHIBA Corporation is now planning to apply for Design Approval (DA) for the Super-Safe, Small and Simple (4S) reactor in the United States with the cooperation of the Central Research Institute of Electric Power Industry (CRIEPI), Westinghouse, and the Argonne National Laboratory (ANL). TOSHIBA and the aforementioned parties have held pre-application review meetings with the U.S. Nuclear Regulatory Commission (NRC) in preparation for the DA submittal.

In the DA submittal, TOSHIBA must demonstrate to the NRC that TOSHIBA has sufficient knowledge of phenomena related to safety, which appear in all the systems and components of the 4S, and that TOSHIBA can control those systems. Therefore, TOSHIBA developed the Phenomena Identification and Ranking Tables (PIRT) document that shows which phenomena are important for safety of the 4S.

This report summarizes the development process for the PIRT for the 4S, which was discussed in the 4th pre-application review meeting, and also outlines topics for further investigation that have been derived from the PIRT project.

The objectives of the 4S PIRT project are as follows:

- Classify the phenomena expected in the 4S by the level of importance and state of knowledge (SoK).
- For the categories above, set the priority for further investigation to be implemented to expand the SoK.
- Based on the priority determined above, clarify the content of the tests or analyses to be implemented in the near future.

This PIRT focuses on identifying the relative importance and associated SoK of phenomena related to the performance of safety-related structures, systems, and components (SSCs) in the 4S.

To benefit from significant prior PIRT development and applications, TOSHIBA and CRIEPI have consulted with an independent review and advisory panel (IRAP) with extensive knowledge and experience. The IRAP consisted of two Japanese professors and three U.S. experts whose roles in the PIRT effort were carefully defined at the initiation of the project. The Japanese IRAP members fully supported TOSHIBA and CRIEPI in all efforts such as definition of the 4S PIRT process and ranking of relative importance and state of knowledge. However, to ensure full independence of the U.S. experts, their role was limited to review of the TOSHIBA and CRIEPI results in the context of identifying any results that were ambiguous, unclear, or different from the experience of the U.S. members.

Procedure of PIRT

Figure ES-1 shows the process used for the 4S PIRT report. This process has evolved based on the research papers by Wilson [ES-1] and is composed of 11 steps. Listed below are the steps and the chapters of this report in which the steps are described.

Step 1: Define the issues to be addressed by the 4S PIRT (Chapter 1)

Step 2: Define objectives that are imbedded in the issue of Step 1 (Chapter 1)

Step 3: Define object event (Chapter 4)

Step 4: Partition selected events into time phases (Chapter 4)

Step 5: Partition plant system into multiple components (Chapter 5)

Step 6: Define Figures of Merit (Chapter 6)

Step 7: Identify plausible phenomena (Chapter 7)

Step 8: Rank relative importance for phenomena (Chapter 8)

Step 9: Rank SoK for the phenomena (Chapter 8)

Step 10: Perform sensitivity analysis (explained in Chapter 8)

Step 11: Identify the scope and priority for further analytical and experimental investigations that may be needed in the DA effort (Chapter 9)

This process allows for refining the results by returning to previous steps as needed in an iterative fashion.

To our knowledge, no previous PIRTs have been developed for liquid metal fast reactors (LMFRs) such as the 4S. Therefore, the 4S PIRT was initially established by reference to existing light water reactor (LWR) PIRTs such as [ES-2], [ES-3]. Subsequently, the 4S PIRT was then refined by adapting the appropriate process steps to adequately reflect the 4S design, specific features, and phenomena associated with the events of interest.

Description of Plant System

The 4S is a small LMFR that, in combination with power generation equipment, is designed for use as a power source in remote locations and is intended to operate for 30 years without refueling. A pool-type fast neutron reactor, the 4S, when coupled to power generation equipment, has a primary electrical output of 10 MWe (30 MWt).

Figure ES-2 is a schematic drawing of the overall 4S-based power generation facility, depicting its major components. The overall area covered by the below-grade and above-grade structures of the plant is approximately 50 meters long by 30 meters wide. The nuclear island is below grade (this includes the steam generator as well as the reactor vessel and all vital equipment). The balance of plant (BOP) includes facilities located within the security barrier (turbine building, gas turbine generator, switchgear, control area, and security facilities) and facilities located outside the security barrier, but within the owner-controlled area (administration facility, offices, and the like). The BOP facilities are designed to conventional industrial construction requirements.

Figure ES-3 is a schematic cross-sectional view of the 4S power generation facility, showing its main features. As can be seen in the figure, the 4S assembly is housed in a reactor building, which includes and supports the reactor vessel, guard vessel/top dome containment configuration, steam generator, and equipment cells. The reactor building itself is supported by seismic isolators, which provide horizontal seismic isolation. Isolation for vertical seismic shaking is not required because 4S has a very long (vertical) reactor vessel.

Event Definition

For the 4S PIRT, events that confirm the design validity of the reactor protection system (RPS) were selected from the anticipated operational occurrences (AOOs) and design basis accidents (DBAs).

As the event that confirm the design validity of the RPS, the loss of offsite power (LOSP) event was selected for shut down systems, the intermediate reactor auxiliary cooling system (IRACS) and the reactor vessel auxiliary cooling system (RVACS).

As the event that confirm the design validity of the RHRS, the sodium leakage from intermediate piping (SLIP) event was selected for the reactor vessel auxiliary cooling system (RVACS),.

The failure of a cavity can (FCC) event has the largest reactivity insertion rate and power level and was selected as the event that can confirm the validity of the shut down systems.

The events are briefly described below.

(1) LOSP

This event results from the loss of the power supply for the station, either from transmission system failure or offsite power electrical equipment failure during rated power operation. Due to the power loss, the primary pumps, intermediate pump, and feedwater pumps are tripped at the same time. Power for the primary and intermediate pump is switched to the coastdown power generated by a motor-generator (MG) set. The primary and intermediate coolant flow rates transit into the state of coastdown by decrease of pump head. The reactor is scrammed by the low voltage signal of the bus voltage (safety protection system signal). The reflector is lowered (withdrawn) and the shutdown rod is inserted into the core; reactor power then decreases rapidly.

Primary flow is gradually decreased by using the flow coastdown system after pump trip. Primary flow reaches 50 percent of rated flow at 30 seconds after pump trip. Natural circulation conditions are reached after 60 seconds.

The safety protection system signal is used for the RHRS startup signal. When the damper of the air cooler (AC) equipped in the intermediate system opens, residual heat starts to be removed by the AC. Also, the RVACS does not have an active component, and is always active regardless of operating condtition.

(2) SLIP

This event results from sodium that is leaked in the steam generator room and burns. This event assumes that the IRACS piping fails and intermediate sodium is leaked from the piping.

Residual heat removal using the AC in the intermediate system cannot be expected because the sodium is drained from the intermediate system after the event occurs. Therefore, only the RVACS is used for residual heat removal in this case.

Sodium leakage is detected by a leakage detection system, and the reactor is manually shut down by the operator as soon as possible. After the trip of the intermediate pump, intermediate sodium is drained into the dump tank. Residual heat in the core is removed by the RVACS.

(3) FCC

The reflector is divided into six segments circumferentially. Six cavity cans are connected in two rows of three high on each reflector segment. The cavity cans are made of ferritic stainless steel (modified 9Cr1Mo steel), and each can is filled with argon gas to equal the pressure outside the core during operation. This event results from failure of one of the 36 cavity cans, leakage of the argon gas in the cavity can into the reactor through the leakage point, and entrance of primary sodium into the cavity can to replace the argon. As a result, positive reactivity is inserted by the decrease of neutron leakage effect, and reactor power increases rapidly.

At the increase in power, a high power signal (RPS signal) from the power-range monitor is transmitted and the reactor is scrammed. The reflector and the shutdown rod are lowered and the reactor power decreases rapidly. Maximum reactor power is about 1.6P0 (1.6 times the rated reactor power).

Power for the primary and intermediate pump is switched to the power generated by the MG sets. The primary and intermediate coolant flow rates transit to coastdown by the gradual reduction of pump head. Since pump head will be lost within 60 seconds of the event initiation, the primary and intermediate coolant flow rates transit to the state of natural circulation.

The RPS signal is also used for the RHRS startup signal. When the AC damper in the intermediate system opens, residual heat starts to be removed by AC. Also, RVACS does not have an active component, and is already operating when the event starts.

The intermediate pump and blower will restart using backup power that initiates after the event occurs. As a result, the core is cooled to the state of cold standby because the intermediate coolant and AC air return to forced circulation.

Partitioning of Events into Time Phases

In the 4S PIRT report, phenomena seen in the events defined in the section above are identified and their relative importance is ranked. However, the importance of the phenomena may vary within the same event with its progress. For example, the LOSP event can be classified into two phases. The first is one in which the effect of forced circulation remains after the electromagnetic pump (EMP) shuts down, and the second is one in which the effect of natural circulation is dominant after that. Therefore, in this PIRT report, events are classified for each phase that have the same important phenomena to establish the relative importance ranking for those phenomena.

LOSP and SLIP: These are events having two time phases that can be separated into the time from when the effect diminishes after the EMP trips until natural circulation becomes dominant, and all the time that follows.

FCC: This is a one-phase event. The time phase for FCC PIRT establishment is not divided because this is a short-term event that can stabilize the transient quickly.

Partitioning of Plant System

In the PIRT process, to make the selection of plausible phenomena easier, the plant system is divided into subsystems or components. Here, "plant system" means the whole plant system, and "subsystem" means individual plant system, such as "Core and Fuel Assemblies" and "Reactor System."

The overall plant system is divided into five subsystems from the aspect of thermal-hydraulic behavior.

- Core and fuel assemblies
- Reactor system
- Primary heat transport system
- Intermediate heat transport system
- Residual heat removal system

Also, the subsystem of the RPS is included in the following partition.

- Instrumentation and control (I&C) system

These six subsystems are divided into components, and those components are further divided into subcomponents.

Figure of Merit

In a PIRT study, the degree of importance of a phenomenon is evaluated by its relative importance against a criterion called the Figure of Merit (FoM).

There is no precedent for the PIRT of an LMFR such as 4S and no authoritative reference material is available for the process of FoM selection. Experience in the process of FoM selection for an LWR is not sufficiently applicable to provide directly useful information for an LMFR.

When selecting the FoM for the 4S PIRT, the highest-level requirement to "protect public health and safety" was chosen, as described in 10 CFR 50 Appendix A and the Standard Review Plan (SRP), particularly as it relates to integrity of the fuel pins and integrity of the primary coolant boundary.

Based on these considerations, for the two conditions of fuel pin integrity and primary coolant boundary integrity, cladding temperature was selected as the FoM for all three events.

Phenomena Identification

In the PIRT process, plausible phenomena are all those that may have some influence on the FoM. The identification of plausible phenomena before ranking their relative importance is a primary means to help ensure that the full phenomena spectrum is identified.

In the 4S PIRT, plausible phenomena were selected for each subsystem/subcomponent. The number of phenomena for each system are follows:

- Core/fuel assemblies — 27 phenomena
- Reactor system — 23 phenomena
- Primary heat transport system — 13 phenomena
- Intermediate heat transport system — 8 phenomena
- RHRS — 15 phenomena
- I&C system — 5 phenomena

In addition, plausible phenomena are identified using the Phenomena Identification Tables for cases where the events differ for LOSP, FCC, and SLIP events. The plausible phenomena of other events are compared with the phenomena identified for the 4S LOSP, SLIP, and FCC events. It is confirmed that the identified phenomena in the three events are sufficient as the phenomena that indicate the validity for the design of the RPS. However, events in addition to the LOSP, SLIP, and FCC are evaluated in Appendix A and for anticipated transients without scram (ATWS) transients and beyond-design-basis accidents (BDBAs), which generally do not involve the RPS, and PIRTs will be developed in the future for those events.

PIRT Ranking Table

The process of establishing the ranking for relative importance of the phenomena seen in the subsystems and for their current SoK is the heart of the PIRT [ES-1].

The object of the ranking for PIRT is to clarify the order of priority and scope for theoretical evaluation and testing to be conducted using a combination of the relative importance and SoK rankings.

For further investigation to build the knowledge base for the 4S design, both testing and theoretical evaluation will be conducted. The theoretical evaluation will consist of theoretically based explanations for the phenomena and investigation using detailed analysis codes, such as computational fluid dynamics (CFD) codes.

In the 4S PIRT project, the ranking is established by using the following scale to classify the relative importance of phenomena into four levels. These classifications are based on how much each phenomenon affects the FoM.

High (H):	Phenomenon has large effect on FoM.
Medium (M):	Phenomenon has medium effect on FoM.
Low (L):	Phenomenon has a small effect on FoM.
Insignificant (N/A):	Phenomenon has little or no effect on FoM.

Also in this PIRT effort, sensitivity analysis was used as one of the methods of confirming the ranking results for relative importance of the phenomena. In the sensitivity analysis, the effect of the phenomena can be evaluated quantitatively by changing the parameter in the safety analysis code ARGO that is related to the selected phenomena. However, the final rankings of relative importance are determined by TOSHIBA, CRIEPI, and the Japanese members of the IRAP. The U.S. IRAP members reviewed the ranking and provided advice in the context of adequate technical justification for each rank.

In the 4S PIRT, the ranking is established by using the following scale to categorize the current SoK according to the following:

Known (K):	Phenomenon is well known. Model of the test data and analysis code contains little uncertainty.
Partially known (P):	Phenomenon is partially known. Model of the test data and analysis code contains moderate uncertainty.

Unknown (U): There is little knowledge regarding the phenomenon. Model of test data and analysis code contains large uncertainty.

Ranking for the SoK of the phenomena in the 4S PIRT is based on the use of high-reliability testing, surveys of industry papers involving analysis and theory, and the opinions of highly informed specialists. However, the final ranking for SoK is established by the specialists of TOSHIBA, CRIEPI, and the Japanese members of the IRAP in the same way as the ranking for relative importance. The U.S. IRAP members reviewed the ranking and provided advice in the context of adequate technical justification for each rank.

In the sensitivity analysis, it is important to determine the uncertainty range of the parameter used in the calculation. Based on review of the various PIRT references and engineering judgment, a standard deviation of 1σ is set as the criterion for uncertainty range of parameters in the sensitivity analysis. In addition, some calculations are performed for a larger deviation, such as 2~3σ, which may simply indicate that the sensitivity is very small. As a sample of sensitivity analysis, the result that indicates the large sensitivity in LOSP is shown in Figure ES-4.

Investigation methods used in the sensitivity analyses are discussed in the following paragraphs. Figure ES-5 outlines the methods. The meanings of the codes used in the figure are as follows.

QS: Quantitative standard
PBC1: Peak value at the 1st phase of base case
PBC2: Peak value at the 2nd phase of base case
PSAC1: Peak value at the 1st phase of sensitivity analysis case
PSAC2: Peak value at the 2nd phase of sensitivity analysis case

As described in Figure ES-5, sensitivity is evaluated for each time phase. As described in the following evaluation formula, sensitivity is defined by comparing the "difference between peak temperature of base case and quantitative standard value" and the "difference between peak temperature of sensitivity analysis case and quantitative standard value" in each phase.

$$\text{Sensitivity} = 1 - \frac{\text{FoM}_{QS} - \text{FoM}_{PSACn}}{\text{FoM}_{QS} - \text{FoM}_{PBCn}} \qquad \text{(ES-1)}$$

where the subscript n indicates the number of the time phase.

The quantitative standard described here is not equivalent to the acceptance criteria used in the safety analysis; it is only the guideline to investigate the sensitivity in the sensitivity analysis. It is set to 630°C for the FoM (cladding temperature), which indicates the highest temperature at which the cladding integrity can be maintained even if a long transient time duration is assumed.

As a sample of sensitivity analysis, the result that indicates the large sensitivity in LOSP is shown in Figure ES-4.

As a sample of the final PIRT result, part of the ranking results for the LOSP event is described in Table ES-1. This table contains not only the ranking results but also an explanation for the results with rationales. Here in Table ES-1, the right-hand column "Code" means a serial number assigned for each phenomenon included in each subsystem/component in the left-hand column.

Further Investigation

This section indicates how to select the phenomena that have high importance but are not sufficiently known based on the results of the PIRT, and how to expand the knowledge of these phenomena in the future.

Regarding further investigation for expanding knowledge of the phenomena, both testing and theoretical evaluation will be conducted. The theoretical evaluation will include detailed analysis using CFD codes.

In this report, the results of the PIRT are summarized, and the priorities for the further investigation for each phenomenon are established. These priorities are determined using a five-level scale by combining the relative importance of phenomena obtained from the result of the PIRT and the SoK, as shown in the matrix in Figure ES-6. These priorities are set based on the opinions of the Japanese experts of the IRAP of the 4S PIRT team. The U.S. IRAP members reviewed the priorities and provided advice in the context of adequate justification.

The numbers assigned in the nine cells in the matrix in Figure ES-6 indicate the order of priority. A smaller number indicates a higher priority.

The combinations of importance of the phenomena and the SoK for each priority are as follows.

Priority 1: Unknown (regardless of importance)
Priority 2: High importance/partially known
Priority 3: Medium importance/partially known
Priority 4: Low importance/partially known
Priority 5: Known (regardless of importance)

In the PIRT, the phenomena with priorities 1 through 3 are defined as the phenomena that require further investigation. Table ES-2 shows the details of the measures for further investigation for each priority. In Table ES-2, arrows in the table mean that test will be performed after consideration by theoretical evaluation and further investigation.

The final results for each phenomenon whose priority is from 1 through 3 are shown in Tables ES-3, ES-4, and ES-5. The number of phenomena selected for LOSP is 11 phenomena, for SLIP is 13 phenomena, and for FCC is 3 phenomena. Since there are some duplicated phenomena among the three events, the total number of phenomena is 16.

Table ES-1. Example of Final PIRT (Event: Loss of Offsite Power)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
	A	Core/Fuel Assemblies						
1	-	Pressure loss in core region	M	L	P	– Pressure loss in core region accounts for about 70% at normal operation and 90% in natural circulation of the whole primary system. Therefore, the coolant flow rate change corresponding to the pressure loss change has a relatively large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Data of flow rate regarding pressure loss coefficient of core region around rated operation were obtained by test [8-4 to 8-7]. In fact,data for the 4S design were also obtained [8-60], [8-61]. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a01
2	-	Pressure loss in reflector region	L	L	P	– Since coolant flow rate in the reflector region accounts for 2% of the whole primary system, the coolant flow rate change corresponding to the pressure loss change does not affect cladding temperature.	– Based on established knowledge [8-7, 8-8], form loss of inlet orifice part and friction loss can be evaluated – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a02
3	-	Natural convection	L	M	P	– In the 1st phase, since there is still an effect of forced circulation, pressure loss that affects cladding temperature is not generated by natural convection. – In the 2nd phase, since the core coolant flow rate is low, there is a potential to cause pressure loss that is likely to affect core flow rate by local natural convection.	– Flow behavior in and around the core at normal operation can be evaluated analytically. – In the natural circulation phase, it is difficult to estimate the flow velocity distribution and temperature distribution in and around the core. – Behavior of the local eddy generated in and around the core at the time of natural circulation and pressure loss attributed to the eddy can be estimated analytically. However, there are limited data to verify the results.	a03

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table ES-1. Example of Final PIRT (Event: Loss of Offsite Power) (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
4	-	Reactivity feedback	L	N/A	P	– In the LOSP event, subcritical reactivity can be maintained by the scram of the reactor. Therefore, reactivity feedback does not affect cladding temperature.	– As a result of criticality test, analysis code is sufficiently verified by the organized nuclear data. Therefore, reactivity feedback can be estimated. – There is insufficient knowledge of shape variation caused by temperature change at the transition. Therefore, evaluated value of reactivity feedback attributed to shape variation is uncertain.	a04
5	-	Gap conductance between fuel and cladding	L	L	K	– Compared to a reactor using oxide fuel, the gap between the cladding and fuel slug is filled with sodium having high thermal conductivity that corresponds to those of the fuel and cladding. Therefore, in the gap between cladding and fuel slug, gap conductance will not cause a large temperature difference that affects cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Since the uncertainty is small for thermal conductivity of fuel, sodium, and cladding (which affects the gap conductance evaluation), uncertainty of gap conductance is also small.	a05
6	-	Heat transfer between cladding and coolant	L	L	K	– Heat transfer coefficient (Nu number) is a function of Pe number (=Re·Pr) and effect of thermal conduction is dominant at Pe<100. Also, in the 4S, Pe number is about 100 even at normal operation. Since the heat transfer between cladding and coolant is controlled by thermal conductivity, which is a physical property not affected by coolant flow rate, sensitivity is small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Since the correlation equation of Nu number between cladding and coolant is investigated, there is sufficient knowledge for many fuel pins, such as those of the FFTF and CRBR [8-7] [8-9]. – For the 4S, a modified Lyon's correlation is used.	a06

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table ES-2. Procedure for Further Investigation

	Theoretical Evaluation	Test
Priority 1 (None currently identified)	Test planning ⇨	✓
Priority 2 and Priority 3	✓ ⇨	Depends on results of theoretical evaluation
Priority 4 and Priority 5	None	None

Table ES-3. Rearranged Final PIRT for LOSP

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature			Importance				
No.	Subsystem/Component		Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
-	Highly ranked phenomena with partially known SoK							
7	A	Core/Fuel Assemblies	Intra- and inter-assembly flow distribution	H	L	P	2	a07
24	A	Core/Fuel Assemblies	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	2	a24
-	Moderately ranked phenomena with partially known SoK							
1	A	Core/Fuel Assemblies	Pressure loss in core region	M	L	P	3	a01
3	A	Core/Fuel Assemblies	Natural convection	L	M	P	3	a03
8	A	Core/Fuel Assemblies	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	M	P	3	a08
23	A	Core/Fuel Assemblies	Inter-wrapper flow between wrapper tubes	L	M	P	3	a23
28	B	Reactor System	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	3	b02
30	B	Reactor System	Natural convection	L	M	P	3	b04
50	C	Primary Heat Transport System	Natural circulation	L	M	P	3	c01
64	D	Intermediate Heat Transport System	Natural circulation	L	M	P	3	d02
73	E	Residual Heat Removal System	Heat transfer between tube and air	L	M	P	3	e03

Table ES-4. Rearranged Final PIRT for SLIP

No.		Subsystem/Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Priority	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
-	Highly ranked phenomena with partially known SoK							
7	A	Core/Fuel Assemblies	Intra- and inter-assembly flow distribution	H	L	P	2	a07
24	A	Core/Fuel Assemblies	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	2	a24
-	Moderately ranked phenomena with partially known SoK							
1	A	Core/Fuel Assemblies	Pressure loss in core region	M	L	P	3	a01
3	A	Core/Fuel Assemblies	Natural convection	L	M	P	3	a03
8	A	Core/Fuel Assemblies	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	M	P	3	a08
23	A	Core/Fuel Assemblies	Inter-wrapper flow between wrapper tubes	L	M	P	3	a23
28	B	Reactor System	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	3	b02
30	B	Reactor System	Natural convection	L	M	P	3	b04
50	C	Primary Heat Transport System	Natural circulation	L	M	P	3	c01
67	E	Residual Heat Removal System	Thermal radiation between RV wall and GV wall	L	M	P	3	e11
68	E	Residual Heat Removal System	Thermal radiation between GV wall and heat collector wall	L	M	P	3	e12
69	E	Residual Heat Removal System	Thermal radiation between heat collector wall and concrete wall	L	M	P	3	e13
70	E	Residual Heat Removal System	Asymmetric airflow	L	M	P	3	e14

Table ES-5. Rearranged Final PIRT for FCC

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature						
No.	Subsystem/Component		Phenomenon	Importance	SoK	Priority	Code
-	Highly ranked phenomena with partially known SoK						
4	A	Core/Fuel Assemblies	Intra- and inter-assembly flow distribution	H	P	2	a07
-	Moderately ranked phenomena with partially known SoK						
5	A	Core/Fuel Assemblies	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	P	3	a08
20	A	Core/Fuel Assemblies	Reactivity insertion by cavity can failure	M	P	3	a27

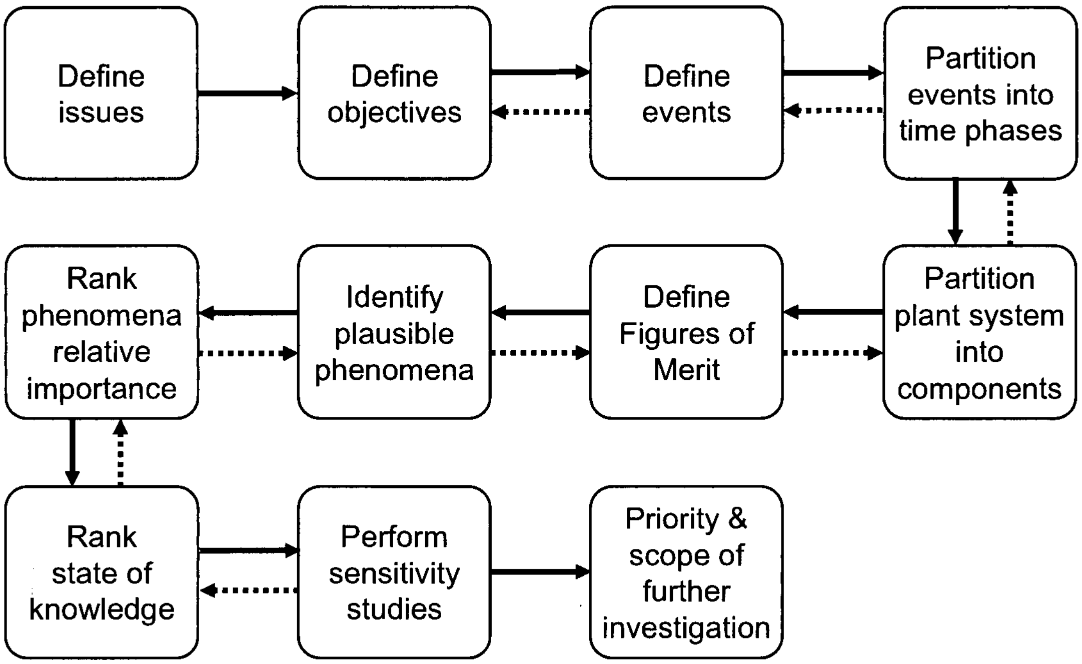


Figure ES-1. 11 Steps of the 4S PIRT Process

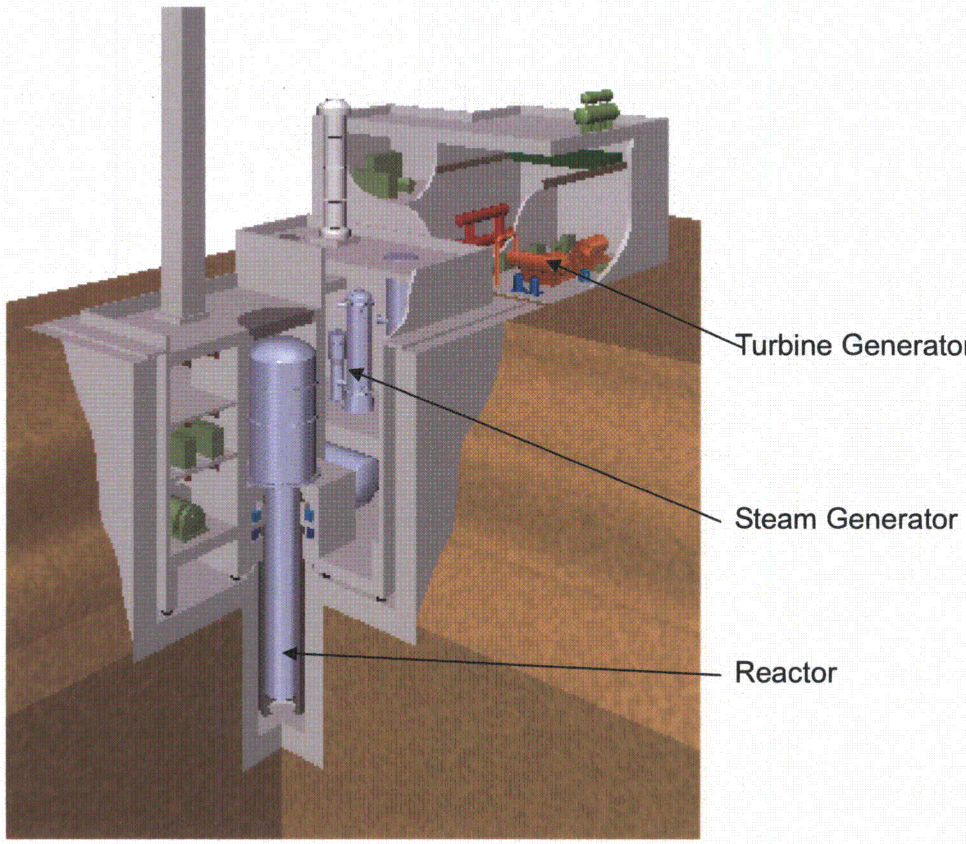


Figure ES-2. Schematic Drawing of 4S Power Generation Facility

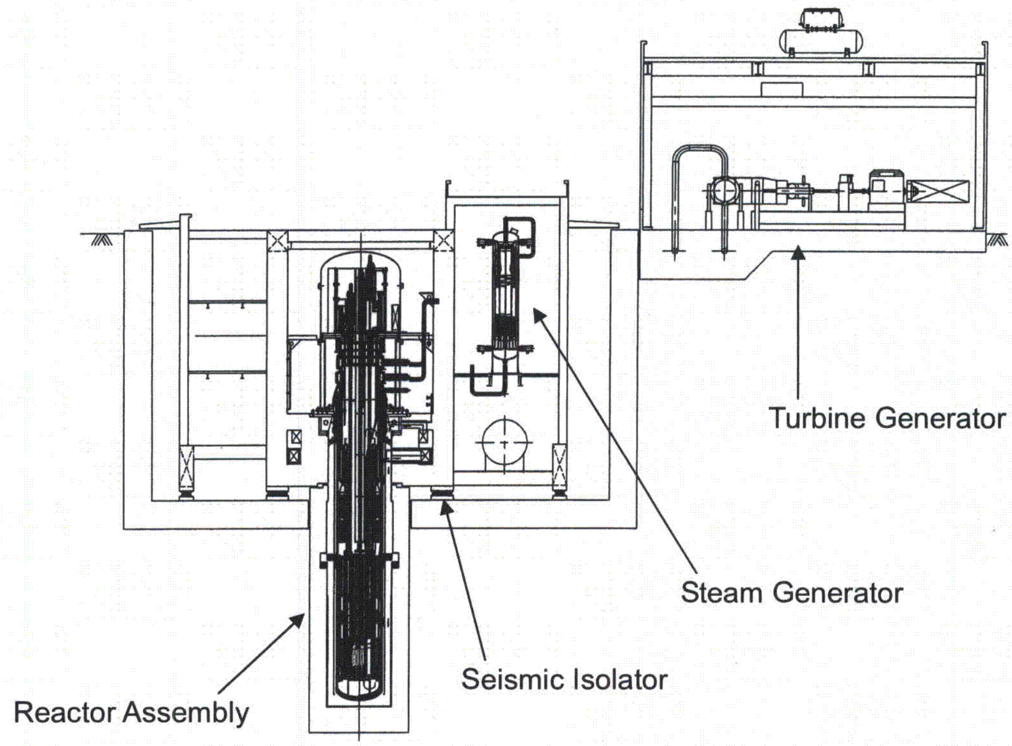


Figure ES-3. Section View of 4S Power Generation Facility

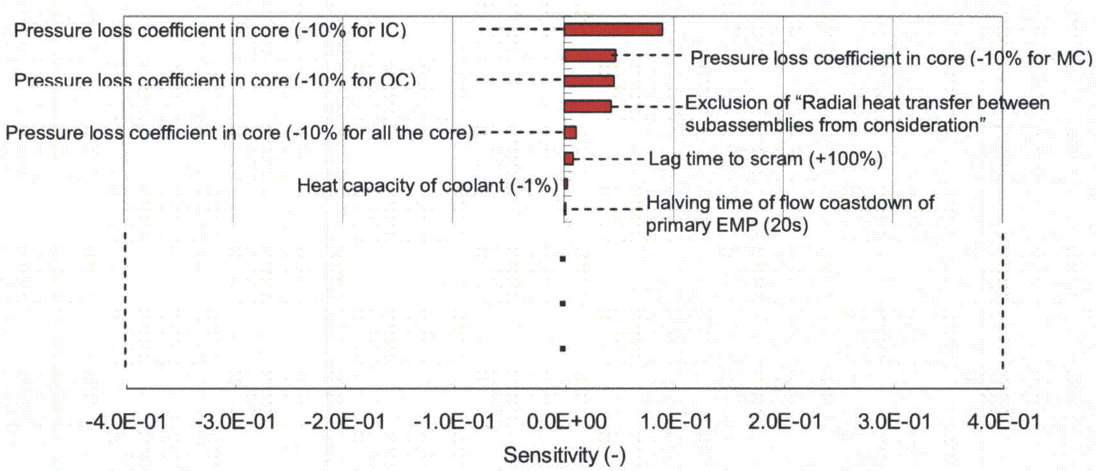


Figure ES-4. Example of Sensitivity Analysis Results (LOSP event, 1st phase)

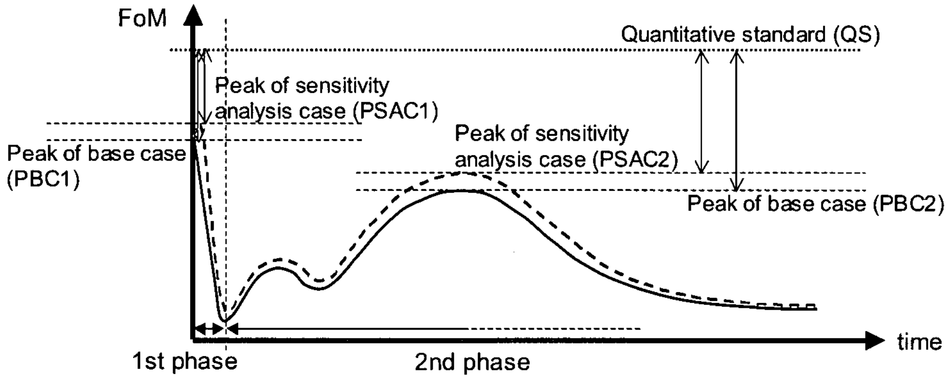


Figure ES-5. Evaluation Method to Determine Sensitivity of FoM to Phenomenon

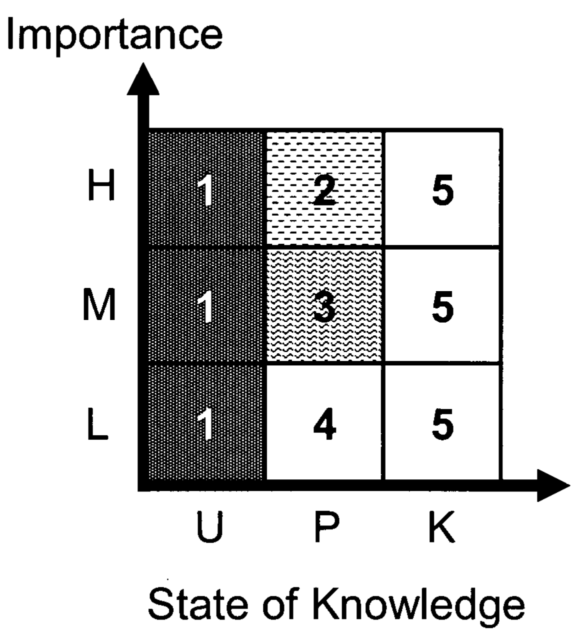


Figure ES-6. Priority for Further Investigation

References

[ES-1] G. E. Wilson and B. E. Boyack, "The Role of the PIRT Process in Experiments, Code Development and Code Applications Associated with Reactor Safety Analysis," *Nuclear Engineering and Design*, 186, pp. 23–37, 1998

[ES-2] G. E. Wilson, C. D. Fletcher, C. B. Davis, et al., "Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios," NUREG/CR-6541, U.S. NRC, Washington, DC, USA, June 1997.

[ES-3] Westinghouse Report WCAP-16318-NP, "IRIS Small Break LOCA Phenomena Identification and Ranking Table (PIRT)," 2004.

1 INTRODUCTION

TOSHIBA Corporation is planning to apply for Design Approval (DA) for the Super-Safe, Small and Simple (4S) reactor in the United States with the cooperation of the Central Research Institute of Electric Power Industry (CRIEPI), Westinghouse, and the Argonne National Laboratory (ANL). TOSHIBA and the aforementioned parties (hereafter the 4S design team) have held pre-application review meetings with the U.S. Nuclear Regulatory Commission (NRC) in preparation for the DA submittal [1-1], [1-2], [1-3], [1-4].

In the DA submittal, TOSHIBA must demonstrate to the NRC that TOSHIBA has sufficient knowledge of phenomena related to safety, which appear in all the systems and components of the 4S, and that TOSHIBA can control those systems. Therefore, TOSHIBA developed the Phenomena Identification and Ranking Tables (PIRT) to show which phenomena are important for safety of the 4S.

This report summarizes the outline of the development process for the 4S PIRT, which was discussed in the 4th pre-application review meeting, and also the outline of the further investigation that has been derived from the PIRT effort.

The objectives of the 4S PIRT project are as follows:

- Classify the phenomena expected in the 4S by the level of importance and state of knowledge (SoK).
- For the categories above, set the priority for further investigation to be implemented to expand the SoK.
- Based on the priority determined above, clarify the content of the test or analyses to be implemented in the near future.

This PIRT focuses on identifying the relative importance and associated SoK of phenomena related to the performance of safety-related structures, systems, and components (SSCs) in the 4S.

To benefit from significant of prior PIRT development and applications, TOSHIBA and CRIEPI have consulted with an independent review and advisory panel (IRAP) with extensive knowledge and experience. The IRAP consisted of two Japanese professors and three U.S. experts whose roles in the PIRT effort were carefully defined at the initiation of the project. The Japanese IRAP members fully supported TOSHIBA and CRIEPI in all efforts such as definition of the 4S PIRT process and ranking of relative importance and state of knowledge. However, to ensure full independence of the U.S. experts, their role was limited to review of the TOSHIBA and CRIEPI results in the context of identifying any results that were ambiguous, unclear, or different from the experience of the U.S. members.

U.S. members of IRAP:

Mario H. Fontana (University of Tennessee): Reactor safety, reactor design, and thermal-hydraulics

Frederick J. Moody (Consulting Engineer): Thermal-hydraulics and two-phase flow

Gary E. Wilson (KatJon Services Inc.): PIRT and reactor safety analysis and licensing. Also, thermal-hydraulic computer code development, assessment, and application with an emphasis on code scaling, applicability, and uncertainty evaluation (CSAU) methodology.

Japanese members of IRAP:

Hisashi Ninokata (Tokyo Institute of Technology): Thermal-hydraulic analysis of advanced nuclear reactors, fast reactor safety, probabilistic risk assessment (PRA), neutronics-thermal-hydraulics-coupled analysis of nuclear reactors

Akira Yamaguchi (Osaka University): Thermal-hydraulics, reactor safety, fast reactor safety, PRA, computational fluid dynamics (CFD), and nuclear engineering

Team members who were engineers from TOSHIBA and CRIEPI established the initial Preliminary PIRT in December 2007. These engineers are assigned to the design sections and research and development sections related to the 4S. Some members were engaged in the design, research, and development of the Joyo and Monju fast breeder reactors (FBRs) in Japan.

Furthermore, from December 2007 through October 2008, the IRAP reviewed and re-evaluated the PIRT development process and the ranking results. To implement this review and re-evaluation, the 4S design team held several meetings with the Japanese experts in Japan and four meetings with all members in the U.S.

This report describes establishment of the PIRT process, PIRT results, selection results of candidate phenomena that require further investigation, and the content of further investigation for the selected phenomena.

References

[1-1] ADAMS: ML072950025 "4S reactor First pre-application review meeting with NRC," October 2007.

[1-2] ADAMS: ML080510370 "4S reactor Second Meeting with NRC Pre-application review," February 2008.

[1-3] ADAMS: ML081400095 "4S reactor Third pre-application review meeting with NRC," May 2008.

[1-4] ADAMS: ML082190834 "4S reactor Fourth pre-application review meeting with NRC," August 2008.

2 PIRT PROCEDURE

PIRT is a method that was introduced as a process for CSAU [2-1]. Originally, PIRT was used as one of the processes in the evaluation for uncertainty of analysis codes to identify the important phenomena.

Application of PIRT techniques is not limited to clarifying phenomena that lack information or require highly detailed analysis for modeling uncertainty evaluation for analysis codes. PIRT is also available as an effective tool for establishment of test plans and for development of analysis code.

Figure 2-1 shows the process used for the 4S PIRT. This process has evolved based on the research papers by Wilson [2-2] and is composed of 11 steps. Listed below are the steps and the chapters of this report in which the steps are described.

Step 1: Define the issues (Chapter 1)

This step defines the issues to be addressed by the 4S PIRT.

Step 2: Define objectives (Chapter 1)

This step defines objectives that are imbedded in the issue of Step 1.

Step 3: Define event (Chapter 4)

This step defines the events addressed in the PIRT report. This report selects representative events from design basis accidents (DBAs) and anticipated operational occurrences (AOOs) that produce the greatest challenge to the safety systems based on preliminary safety analysis.

Step 4: Partition selected events into time phases (Chapter 4)

This step partitions selected events into time phases to facilitate understanding of how phenomena importance may change as each transient progresses, because relative importance of phenomena changes during event.

Step 5: Partition plant system into multiple components (Chapter 5)

This step partitions plant system into multiple subsystems/components to enhance plausible phenomena identification performed in step 7.

Step 6: Define Figures of Merit (Chapter 6)

This step defines Figures of Merit (FoM) that are the criteria with which the relative importance of each phenomenon is judged.

Step 7: Identify plausible phenomena (Chapter 7)

This step identifies plausible phenomena using all currently available information, including expert opinion. In the context of the PIRT process, plausible phenomena are those that may have some influence on the FoM.

Step 8: Rank relative importance for phenomena (Chapter 8)

This step ranks relative importance of plausible phenomena that impact FoM using all currently available information, including expert opinion.

Step 9: Rank SoK for the phenomena (Chapter 8)

This step ranks SoK for plausible phenomena that impact FoM using all currently available information, including expert opinion.

Step 10: Perform sensitivity analysis (Chapter 8)

Sensitivity studies are performed in this step to verify/refine preliminary ranking result of the importance of phenomena.

Step 11: Identify the scope and priority for further analytical and experimental investigations that may be needed in the DA effort (Chapter 9)

This step identifies the scope and priority for further analytical and experimental investigations that may be needed in the DA effort. Priority is determined by combination of "relative importance" and "SoK" of phenomena.

This process allows for refining the results by returning to previous steps as needed in an iterative fashion.

A number of PIRTs have been documented for light water reactors (LWRs) and gas-cooled reactors [2-3], [2-4], [2-5].

To our knowledge, no previous PIRTs have been developed for liquid-metal fast reactors (LMFRs) such as the 4S [2-2], [2-3]. Therefore, the 4S PIRT was initially established by reference to existing LWR PIRTs. Subsequently the 4S PIRT was then refined by adapting the appropriate process steps to adequately reflect the 4S design, specific features, and phenomena associated with the target events.

Also, this PIRT takes the following strategy to check the sufficiency of the selected phenomena. As stated in the step 3 above, this PIRT defines events from DBA and AOO. Then, in the step 7, plausible phenomena are identified in the defined events. These selected phenomena, however, must represent all the events occurring in the 4S. Hence, this PIRT checks if selected phenomena envelop important ones occurring in other events such as "ATWS" or "local fault." This is described in Appendix A.

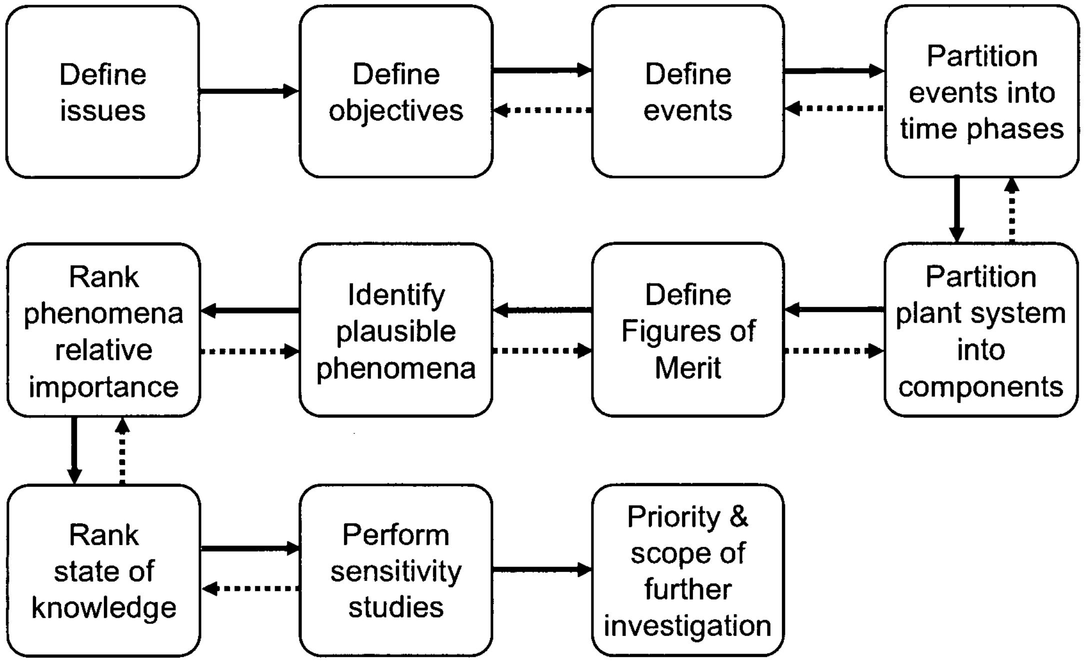


Figure 2-1. 11 Steps of the 4S PIRT Process

References

[2-1] NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of Coolant Accident," 1989.

[2-2] G. E. Wilson and B. E. Boyack, "The Role of the PIRT Process in Experiments, Code Development and Code Applications Associated with Reactor Safety Analysis," *Nuclear Engineering and Design*, 186, pp. 23–37, 1998.

[2-3] G. E. Wilson, et al., "Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generation Tube Rupture Scenarios," NUREG/CR-6541, 1997.

[2-4] Westinghouse Report WCAP-16318-NP, "IRIS Small Break LOCA Phenomena Identification and Ranking Table (PIRT)," 2004.

[2-5] S. J. Ball and S. E. Fisher, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)," NUREG/CR-6944, ORNL/TM-2007/147, 2008.

3 DESCRIPTION OF PLANT SYSTEM

This chapter briefly explains the plant system of the 4S in material excerpted from the "4S Design Description" [3-1].

The 4S is a small LMFR, which, in combination with power generation equipment, is designed for use as a power source in remote locations, and intended to operate for 30 years without refueling. A pool-type fast neutron reactor, the 4S, when coupled to power generation equipment, has an electrical output of 10 MWe (30 MWt).

Figure 3-1 is a schematic drawing of the overall 4S-based power generation facility, depicting its major components. The overall area covered by the below-grade and above-grade structures of the plant is approximately 50 meters long by 30 meters wide. The nuclear island is below grade (this includes the steam generator as well as the reactor vessel and all vital equipment). The balance of plant (BOP) includes facilities within the security barrier (turbine building, gas turbine generator, switchgear, control area, and security facilities) and facilities outside the security barrier, but within the owner-controlled area (administration facility, offices, and the like). The BOP facilities are designed to conventional industrial construction requirements.

Figure 3-2 is a schematic cross-sectional view of the 4S power generation facility, showing its main features. As can be seen in the figure, the 4S assembly is housed in a reactor building, which includes and supports the reactor vessel, the guard vessel/top dome containment configuration, the steam generator, and the equipment cells. The reactor building itself is supported by seismic isolators, which provide horizontal seismic isolation.

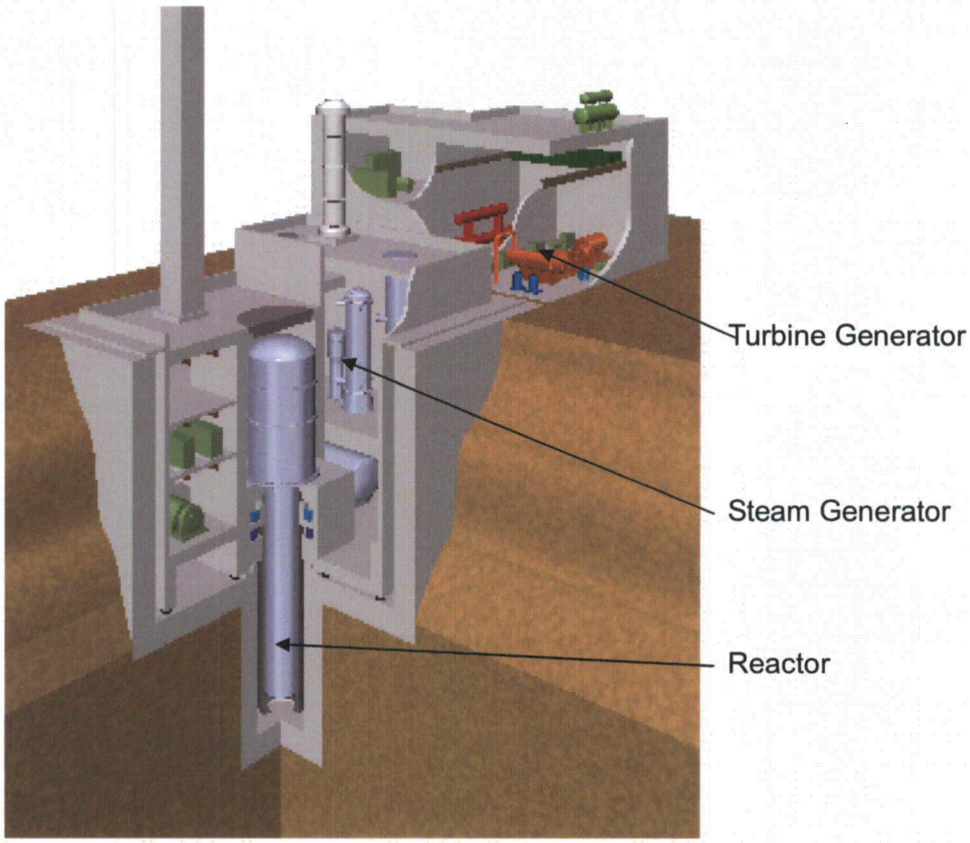


Figure 3-1. Schematic Drawing of 4S Power Generation Facility

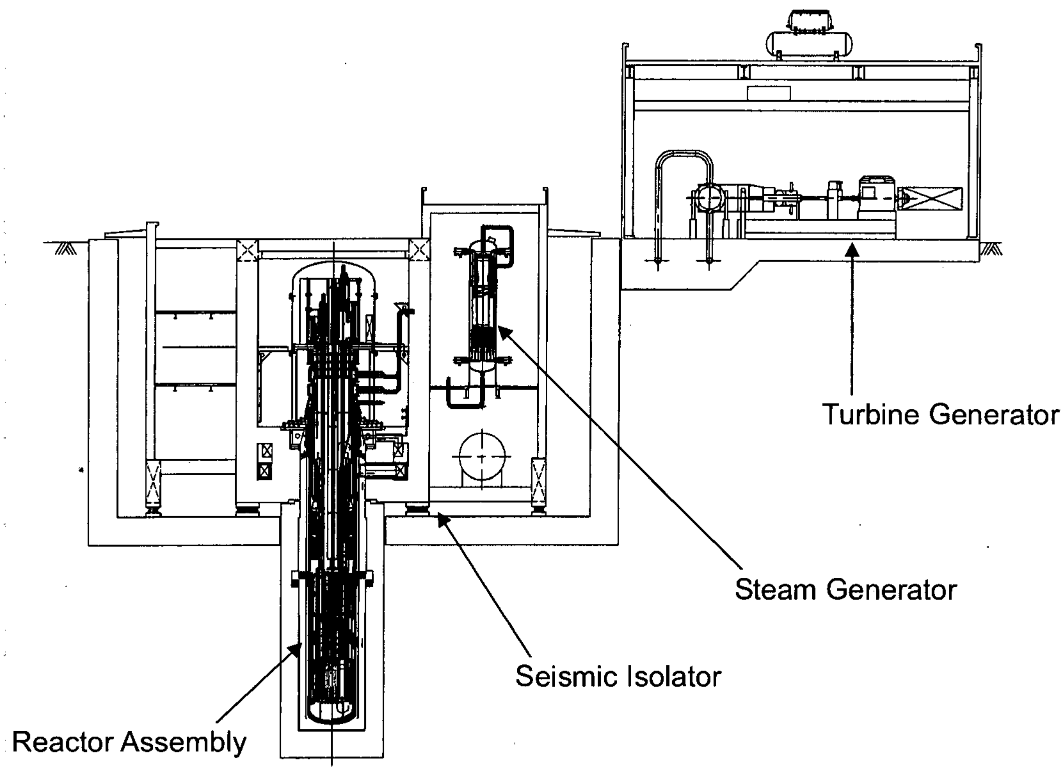


Figure 3-2. Section View of 4S Power Generation Facility

3.1 REACTOR AND CORE

The major components making up the reactor assembly are the reactor vessel, shielding plug, guard vessel (GV), and top dome. Structures internal to the reactor vessel include the core support structures, upper vertical baffle, EMPs, and intermediate heat exchanger (IHX).

Primary sodium is directed via the vessel outer plenum down through the IHX, exchanging heat with the intermediate loop. It then passes to the suction of the EMPs, where it is pumped downward to the lower head region and redirected up through the core. The sodium is heated in the core, exits the core, and is redirected downward into the IHX to continue the cycle.

The core height is 2500 mm. The high thermal conductivity of the metal fuel core allows it to be operated at relatively low temperatures. Inherent core reactivity feedback causes reactivity shutdown for beyond-design-basis accidents (BDBAs), such as an anticipated transient without scram (ATWS).

During normal operating conditions, reactor power is controlled by a movable reflector. The reflector drive consists of a combination of fine and fast adjustment mechanisms. To scram the reactor, the clutch of the fast adjustment mechanism is released and the reflector lowers (withdraws) via gravity, causing the reactor to shut down. The shutdown rod is also inserted for a scram at the core center position to increase neutron absorption. Figure 3.1-1 shows the core layout, reflector, and shutdown rod.

Burnup reactivity compensation and margins for uncertainties in the temperature effect, criticality, and fissile enrichment are considered in the reflector and fixed neutron absorber design. The radial reflectors are adjustable, and they are raised slowly and incrementally during the service life of the core via the fine adjustment mechanism to maintain neutron flux and power levels. In case of electrical power loss or failure of the reflector drive, the assembly lowers to the bottom of the reactor. Reflector lowering (withdrawal) reduces reactivity and stops the nuclear reaction.

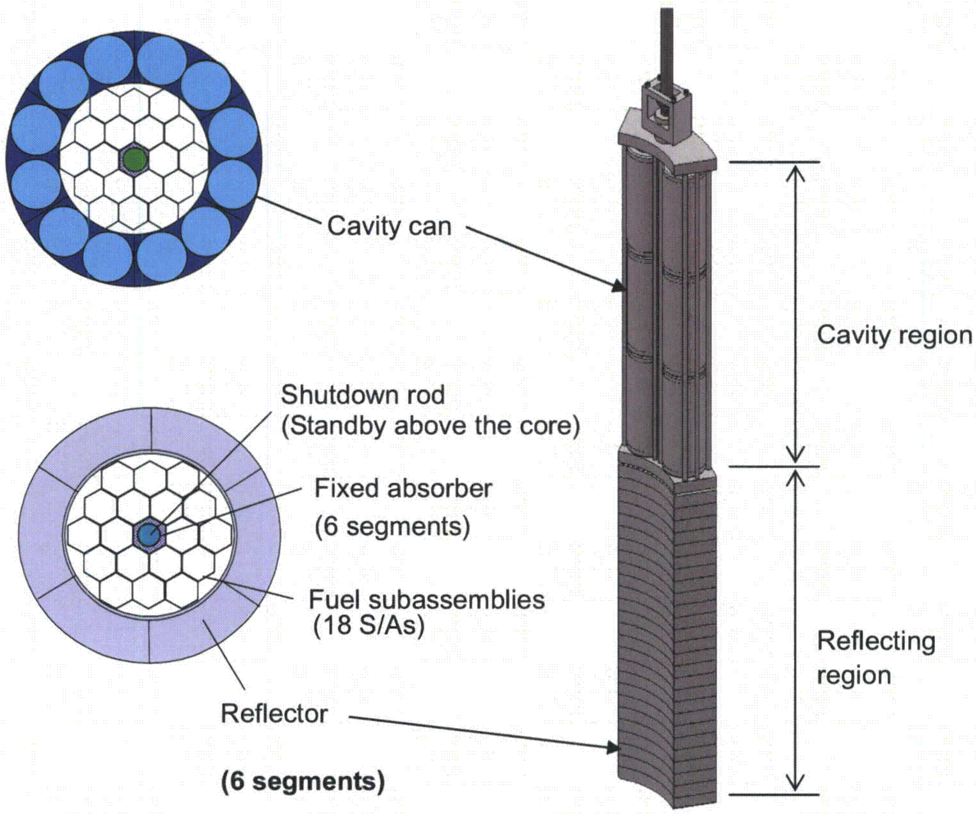


Figure 3.1-1. Core Layout and Reflector

3.2 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

The power generation facility for the reference design consists of the reactor, steam generator, and one turbine generator. Two sequentially oriented EMPs of 50 percent capacity each combine to circulate primary sodium within the reactor vessel. A single IHX transfers reactor thermal energy to the single intermediate heat transport system (IHTS) loop. Heat is transferred from the IHTS via the steam generator to a single steam turbine generator. The primary heat transport system (PHTS) is contained entirely within the reactor vessel. It consists of the hot pool, the tube side of the IHX, pumps, and pump plenum. As shown in Figure 3.2-1, sodium from the reactor enters and flows through the IHX where it is cooled as it heats the intermediate sodium. The primary sodium then exits the IHX and is drawn into the pump plenum. The primary EMPs discharge the sodium down into the bottom of the reactor. The sodium is then heated as it flows up through the core and back through the IHX. The IHX consists of upper and lower tube sheets separated by straight tubes. The cold leg intermediate sodium flows down through the IHX shell and, as it is being heated, flows up to exit the IHX through the intermediate outlet nozzle for use in the IHTS.

The IHTS transports heat from the primary system to the steam generator system. The IHTS consists of a piped loop thermally coupled to the primary system by the IHX in the reactor vessel and to the steam generator system in the steam generator compartment. The IHTS is a closed-loop system with an expansion space in the steam generator plenum using argon cover gas to accommodate thermally induced systemwide volume changes. An EMP, located separately from the steam generator, circulates intermediate sodium through the shell side of the IHX and steam generator. All materials for the IHTS piping and components are designed to minimize corrosion and erosion, and to ensure compatibility with the operating environment. The steam generator is a helical coil shell-and-tube heat exchanger with water/steam in the tube side and sodium in the shell side. The double-wall tubes, which include a wire mesh layer between the inner and outer tube, are adapted to the steam generator. The double-wall tube provides high reliability and significantly reduces the probability of sodium-water interaction.

The generator is powered via a steam turbine that operates at 3600 rpm at rated steam conditions. The generator exhausts to a single condenser. A feedwater pump system for the water/steam loop circulates water from the condenser back to the steam generator.

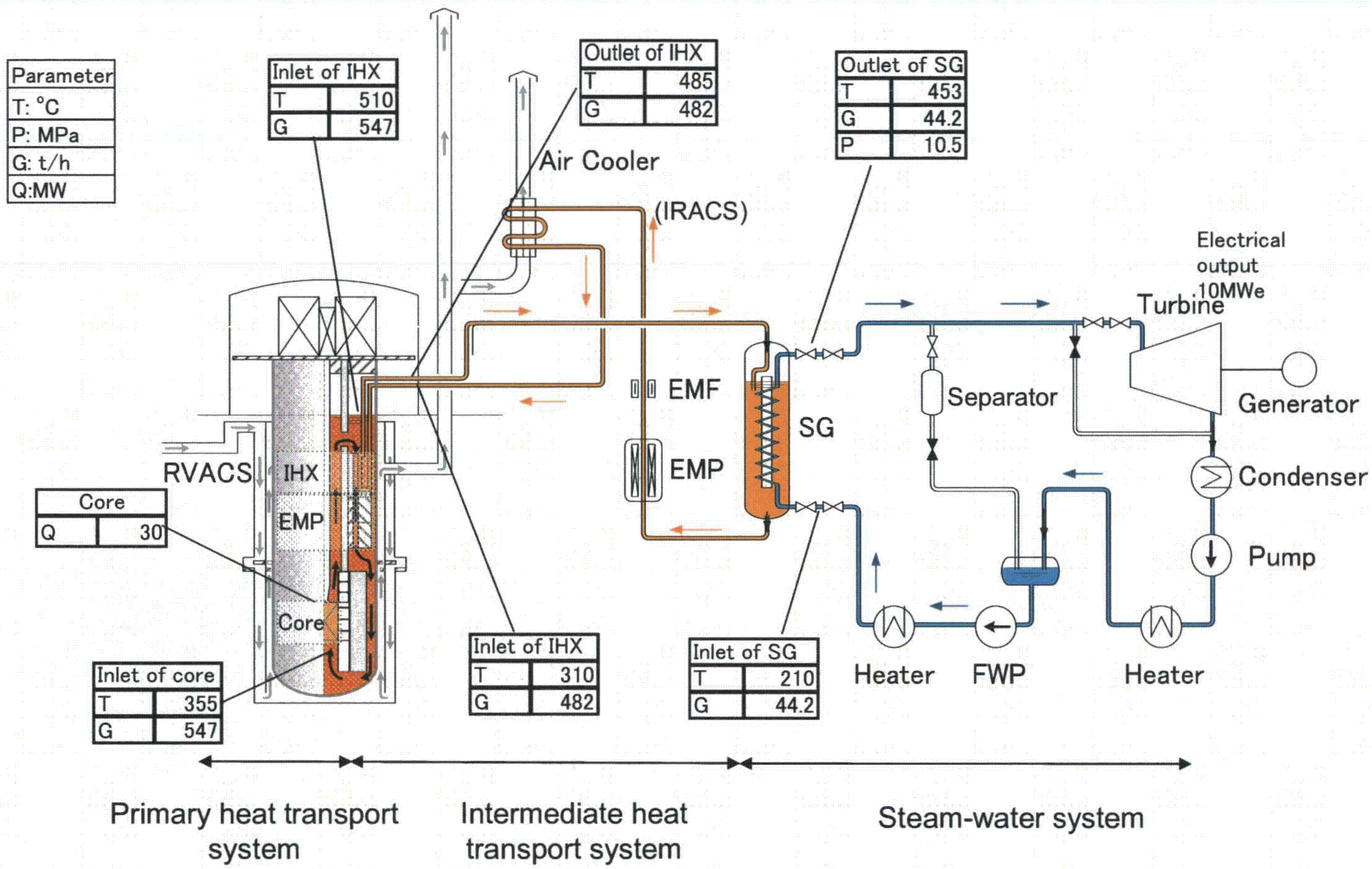


Figure 3.2-1. Heat Transport System Flow Diagram

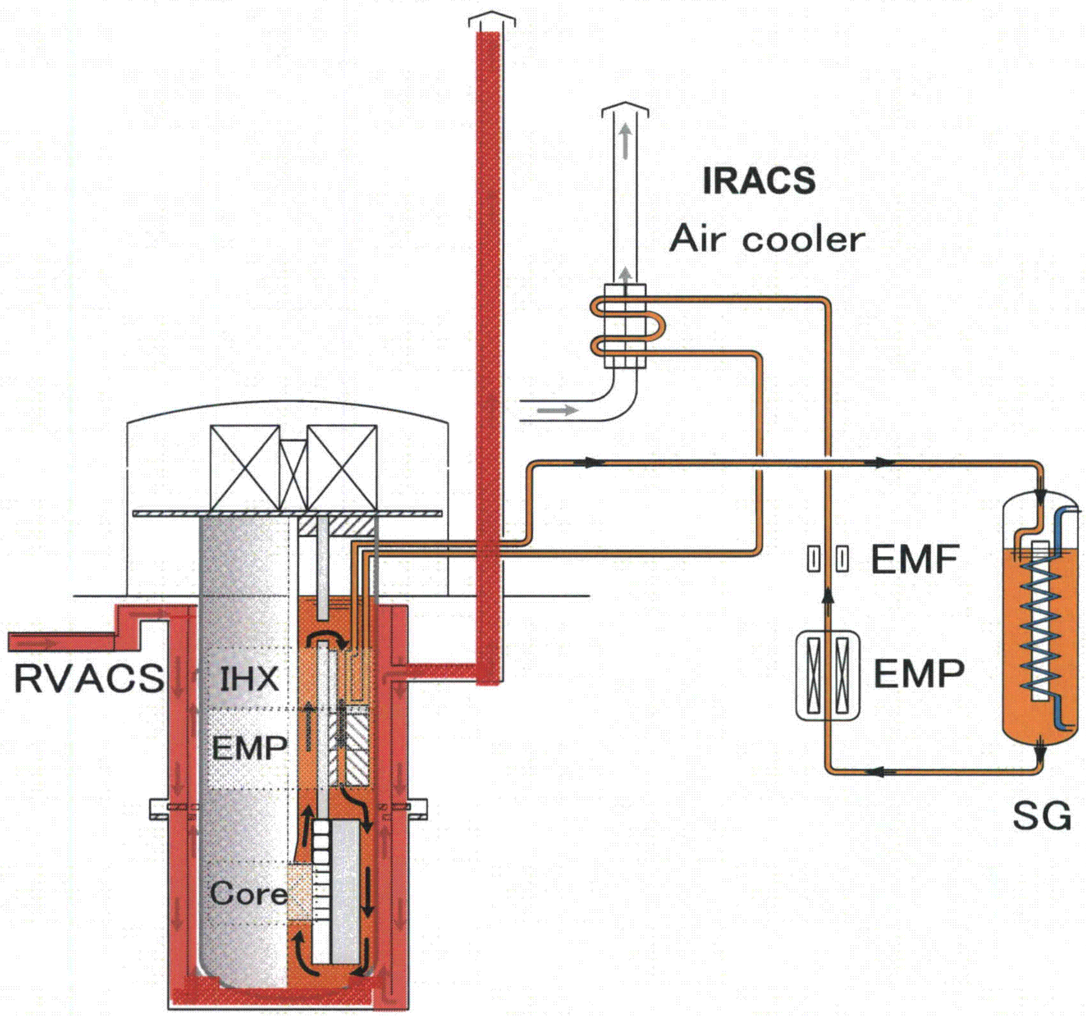
3.3 RESIDUAL HEAT REMOVAL SYSTEMS

The 4S shutdown heat removal systems consist of the main condenser cooling system; the IRACS, which uses an air cooler (AC) in the intermediate sodium loop; and the reactor vessel auxiliary cooling system (RVACS), which removes heat directly from the reactor vessel. Figure 3.3-1 shows the schematic diagram of residual heat removal systems.

When the reactor is brought from full power to cold shutdown conditions under normal operation, cooling is provided by routing steam through the steam generator bypass directly to the condenser. In the condenser, the steam condenses to water and, using the feedwater pumps, it is fed back to the steam generator. Feedwater flow to the steam generator is maintained by the local control system.

The EMP of the intermediate sodium loop maintains sodium flow within the IHTS with power supplied by an auxiliary generator. The IRACS can remove additional heat from the core via an air cooler, which is located in the piping between the IHX and the intermediate EMP.

In the event that normal condenser cooling is not available, as with a loss of power supply, decay heat can be removed by the RVACS. The RVACS is a passive system. The system transports heat to the atmosphere by natural circulation of air. It functions continuously with its heat transport rate governed by the reactor vessel temperature. The RVACS operates continuously at all operating conditions including normal power operation. The flow of sodium within the reactor is aided by natural convection caused by heating in the core and cooling along the reactor vessel wall caused by the RVACS. Airflow in the RVACS is maintained by natural draft in the main RVACS riser.



EMF: Electromagnetic flowmeter

SG: Steam generator

Figure 3.3-1. Schematic Diagram of Residual Heat Removal Systems

Reference

[3-1] ADAMS: ML081440765, "4S Design Description."

4 EVENT DEFINITION

For LWRs, performance of the emergency core cooling system (ECCS) during postulated loss-of-coolant accident (LOCA) events is required to be evaluated. This is because LOCA is the most severe design basis event regarding public exposure for the LWR. Therefore, a PIRT for an LWR mainly covers the LOCA event [4-1], [4-2]. On the other hand, for the sodium-cooled fast reactor, the event that should be treated in the PIRT has never been determined. Since a sodium-cooled reactor is a low-pressure system, a LOCA is not an applicable event for PIRT. Also, since a PIRT for a sodium-cooled reactor has never been performed, there is no reference that contains helpful information for event selection.

The 4S PIRT was implemented to establish the test program that is required to indicate the design viability of important systems.

The shut down system, residual heat removal system (RHRS), and containment system are the most important safety systems for the protection and mitigation functions in the 4S design.

Events to confirm the validity of the containment system are not selected in the 4S PIRT. Both the primary and intermediate systems have low pressure, and the containment system is configured to prevent excessive pressure and temperature by utilizing multiple boundaries and an inert nitrogen atmosphere to accommodate primary and intermediate sodium leakage in the containment system, and primary cover gas leakage.

Therefore, the containment system is designed with sufficient margin to secure the integrity and function for leakage-related DBAs. Also, these events will not pose a direct threat to fuel integrity if the shut down system and RHRS function as intended. Therefore, these containment system DBAs (accidents of primary coolant leakage from reactor vessel and primary cover gas boundary leakage) are not selected for this PIRT. The events treated in this PIRT are intended to confirm the validity of the shut down system and RHRS designs.

4.1 EVENT SELECTION

Events investigated in the safety analysis [4-1] were identified as anticipated operational occurrences (AOO), design basis accidents (DBA), or anticipated transient without scram (ATWS), based on the Standard Review Plan Section 15.0 [4-2].

In the 4S PIRT, events that confirm the design validity of the RPS are selected from the AOOs and DBAs.

As the event that confirm the design validity of the RPS, the loss of offsite power (LOSP) event was selected for shut down systems, the intermediate reactor auxiliary cooling system (IRACS) and the reactor vessel auxiliary cooling system (RVACS).

As the event that confirm the design validity of the RHRS, the sodium leakage from intermediate piping (SLIP) event was selected for the reactor vessel auxiliary cooling system (RVACS),.

The failure of a cavity can (FCC) event has the largest reactivity insertion rate and power level and was selected as the event that can confirm the validity of the shut down systems.

4.2 EVENT DESCRIPTION

4.2.1 Loss of Offsite Power (LOSP)

This event involves transmission system failure or offsite power electrical equipment failure during reactor rated operation, leading to loss of the power supply for the station. Due to the power loss, the primary pumps, intermediate pump, and feedwater pumps are tripped at the same time. Power for the primary and intermediate pump is switched to the power generated by an MG set. The primary and intermediate coolant flow rates transition into the state of coastdown by decrease of pump head. The reactor is scrammed by the low voltage signal, which is transmitted when bus voltage becomes low (safety protection system signal). The sectors of reflector are lowered (withdrawn), the shutdown rod is inserted into the core, and reactor power decreases rapidly.

Primary flow is gradually decreased by using the flow coastdown system after pump trip. Primary flow decreases to 50 percent of rated flow after 30 seconds from pump trip and reaches natural circulation flow after 60 seconds.

The RPS signal is used for the RHRS startup signal. When the air cooler (AC) damper in the intermediate system opens, the AC begins removing residual heat. Also, the RVACS does not have an active component, and is in the standby state during normal operation before the accident. Core residual heat will be removed by natural circulation of air in the AC and the RVACS.

The analysis results using the ARGO code are shown below to explain the transition.

The sequence, start time of each phenomenon, and system response are shown in Table 4.2.1-1.

The reactor power and the primary flow rate are shown in Figure 4.2.1-1, the fuel temperature of the highest-temperature pin and cladding temperature are shown in Figure 4.2.1-2, and the coolant average temperature at the core inlet and outlet are shown in Figure 4.2.1-3, respectively, from the analytical results using the ARGO code. The hottest pin is calculated under nominal condition at rated power.

When offsite power is lost, the primary pumps, intermediate pump, and feedwater pumps are tripped. The signal indicating the low bus voltage is transmitted, resulting in a scram signal in 1.0 seconds, and the reflector starts to lower after 1.5 seconds. The relative value of the reactor power exceeds the relative value of the primary coolant flow rate momentarily until the start of reflector descent; however, this imbalance is small and the fuel and cladding temperature are hardly changed due to the short time and the low power density of the core. The fuel peak temperature and the fuel cladding peak temperature of the maximum-temperature pin do not exceed 600°C and 571°C, respectively.

After the reflectors are lowered, the relative flow rate becomes larger than the relative power rate; hence, the core temperature decreases (see Figures 4.2.1-1 and 4.2.1-2). After that, the primary pump head decreases along with power generated by the MG set, and primary and intermediate flow change to natural circulation flow. Residual heat is removed by the RVACS and IRACS. In this analysis, emergency (backup) power is not expected to work. The RVACS and IRACS remove residual heat by natural circulation.

Table 4.2.1-1. Sequence of Events for LOSP

Time (sec)	Events
0.0	Loss of station power
0.0	Trip of the primary, intermediate, and feedwater pumps
0.0	Switch of status of the primary pumps from normal operation to flow coastdown
1.0	Transmittal of signals:
	"Low voltage" signal of bus voltage
	Scram initiation signal
	RHRS initiation signal
1.5	Start lowering of the reflector
2.0	Start opening of the AC damper
9.5	Finish lowering of the reflector
32	Finish opening of the AC damper
60	Primary and intermediate coolant flow changes to natural circulation, and residual heat is removed by RVACS and IRACS.

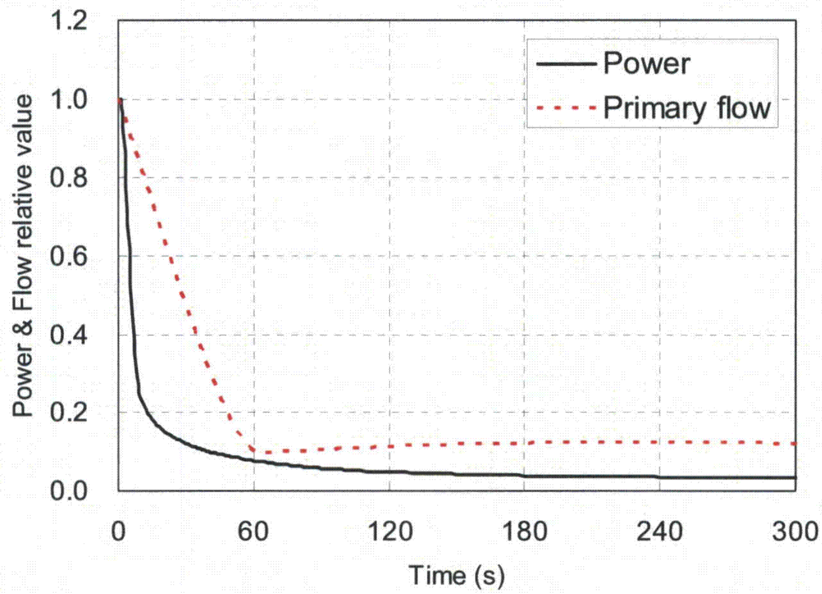


Figure 4.2.1-1. Power and Primary Flow (LOSP)

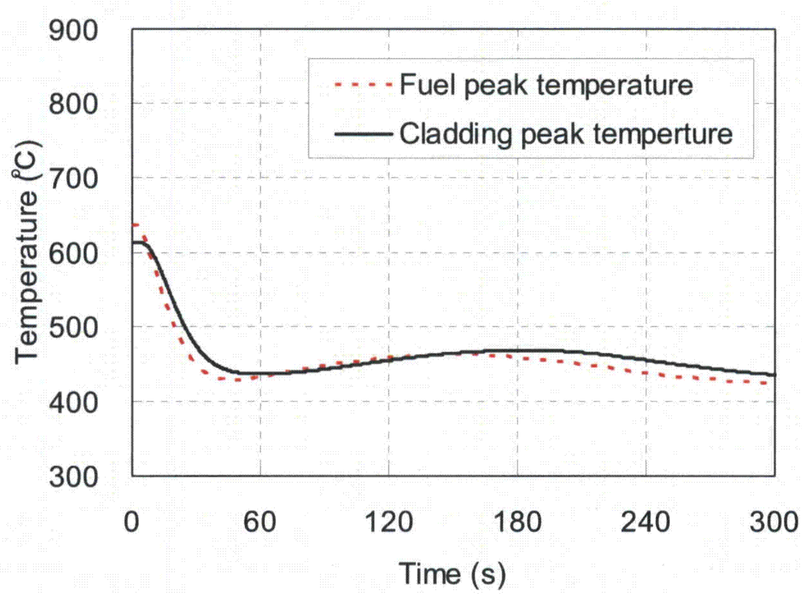


Figure 4.2.1-2. Fuel and Cladding Peak Temperatures (LOSP)

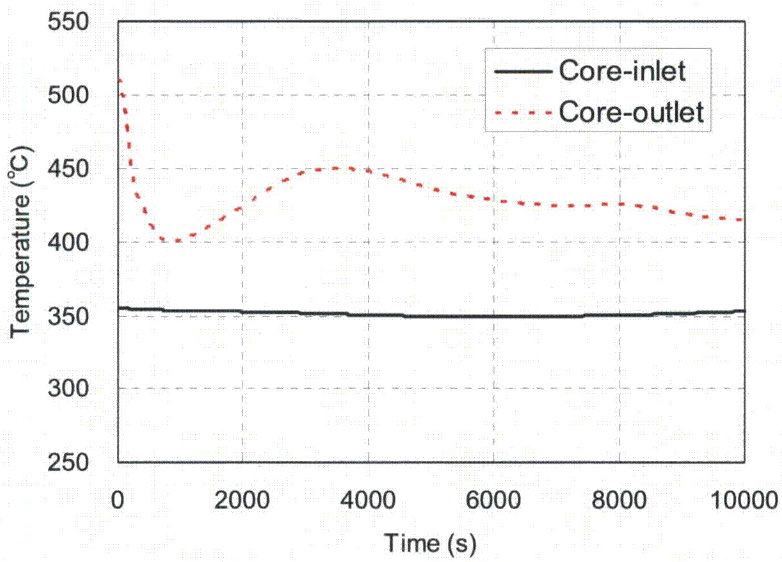


Figure 4.2.1-3. Core Inlet and Outlet Temperatures (LOSP)

4.2.2 Sodium Leakage from Intermediate Piping (SLIP)

The SLIP event results from the sodium leakage in the steam generator room. This event assumes that the intermediate system piping fails and intermediate sodium leaks from the piping.

Residual heat cannot be removed by the AC in the IRACS because the sodium is drained from the intermediate system. Therefore, only the RVACS is used for residual heat removal in this case.

Sodium leakage is detected by a leakage detection system, and the reactor is manually shut down by the operator as soon as possible. After the shutdown of the intermediate pump, intermediate sodium is drained into the dump tank. Residual heat in the core is removed by the RVACS, which is operating as designed. The analysis results using the ARGO code are shown below to explain the transition.

The sequence, start time of each phenomenon, and system response are shown in Table 4.2.2-1.

The reactor power and the primary flow rate are shown in Figure 4.2.2-1. The fuel and cladding temperature of the maximum-temperature pin are shown in Figure 4.2.2-2. The temperatures at the core outlet and inlet are shown in Figure 4.2.2-3.

After the event occurs, the primary pumps, intermediate pump, and feedwater pumps trip, and 0.5 seconds later, the reflector starts lowering. The initial power and flow behavior is somewhat similar to that of the loss of offsite power described in Section 4.2.1. The fuel peak temperature and cladding peak temperature of the maximum-temperature pin do not exceed 600°C and 570°C, respectively.

Regarding the long-term behavior, the flow of both the primary and intermediate coolant changes to natural circulation flow, and residual heat is removed by the RVACS (see Figure 4.2.2-3).

Table 4.2.2-1. Sequence of Events for SLIP

Time (sec)	Events
0.0	Sodium leakage from intermediate piping
0.0	Manual reactor trip
	Loss of heat removal from IHX (Immediate loss of heat removal from IRACS is conservatively assumed in the analysis considering draining of sodium from intermediate loop.)
0.5	Trip of the primary EMPs
	Start lowering of the reflector
8.5	Finish lowering of the reflector
60.5	The flow of the primary coolant changes to natural circulation flow, and residual heat is removed by RVACS.

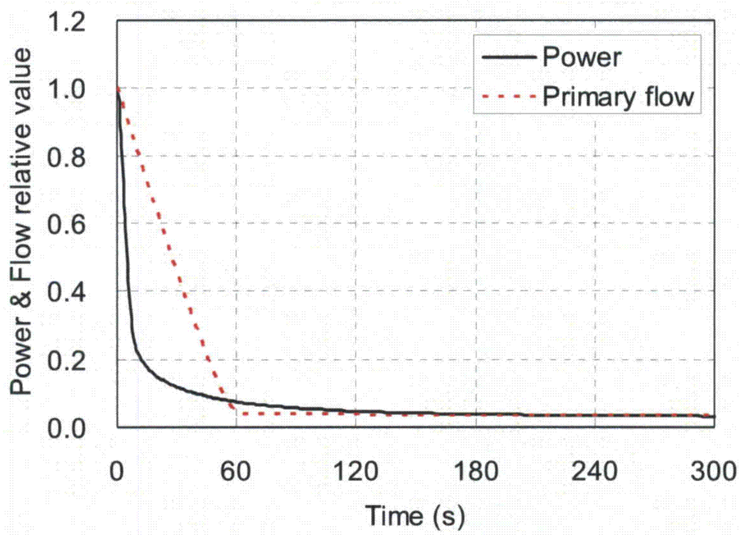


Figure 4.2.2-1. Power and Primary Flow (SLIP)

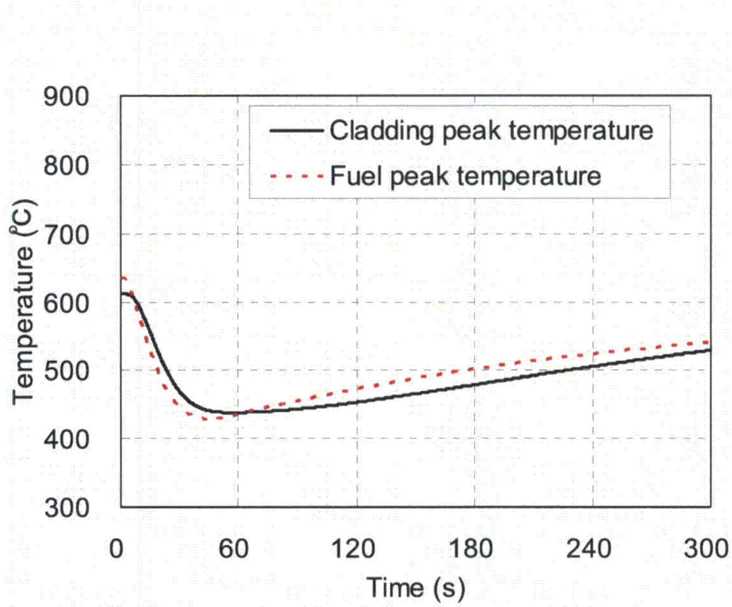


Figure 4.2.2-2. Fuel and Cladding Temperatures (SLIP)

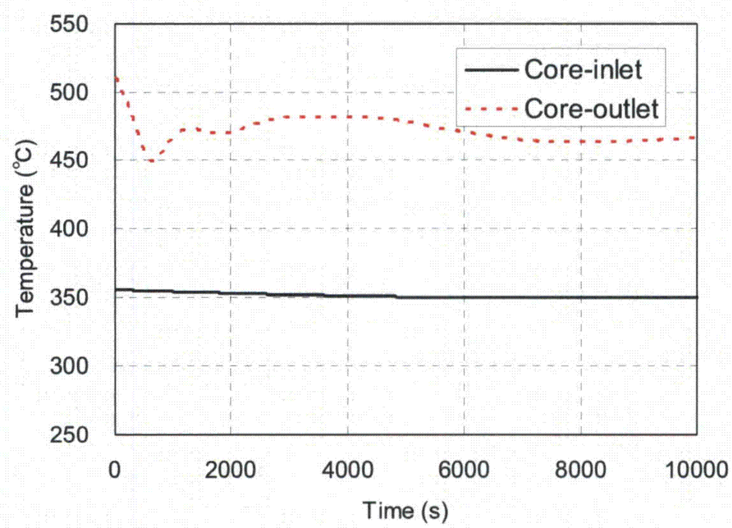


Figure 4.2.2-3. Core Inlet and Outlet Temperatures (SLIP)

4.2.3 Failure of a Cavity Can (FCC)

The reflector is divided into six segments circumferentially. Six cavity cans are arranged in two rows and three concentric layers on each reflector segment. The cavity cans are made of ferritic stainless steel (modified 9Cr1Mo steel), and each can is filled with argon gas to equal the pressure outside the core during operation. This event results from failure of one of the total of 36 cavity cans (failure of more than one cavity can is a BDBA), leakage of argon gas in the cavity can into the reactor through the leakage point, and entrance of primary sodium into the cavity can to replace the argon. As a result, positive reactivity is inserted by the decrease of neutron leakage effect, and reactor power increases rapidly.

At the increase of power, a signal indicating overpower (RPS signal) from the power-range monitor is transmitted and the reactor is scrammed. The reflector and the shutdown rod are lowered and the reactor power decreases rapidly. Maximum power of the reactor is about $1.6P_0$ (1.6 times the rated reactor power).

Power for the primary and intermediate pump is switched to the power generated by the MG sets. The primary and intermediate coolant flow rates transit to coastdown by the gradual reduction of pump head. Since pump head will be lost within 60 seconds of the event initiation, the primary and intermediate coolant flow rates transit to natural circulation.

The RPS signal is also used for the RHRS startup signal. When the AC damper in the intermediate system opens, residual heat starts to be removed by AC. Also, the RVACS does not have an active component, and is already operating when the event starts. Residual heat will be removed by natural circulation of air in the AC and the RVACS.

The intermediate pump will restart and blower will start using backup power that initiates after the event occurs. As a result, the core reaches the state of cold standby because the intermediate coolant and air in AC return to forced circulation.

The sequence, start time of each phenomenon, and system response are shown in Table 4.2.3-1.

The reactor power and the primary flow rate are shown in Figure 4.2.3-1. The fuel and cladding temperatures of the maximum-temperature pin are shown in Figure 4.2.3-2. The coolant temperatures at the core outlet and inlet are shown in Figure 4.2.3-3.

The signal indicating high neutron flux is transmitted in 0.6 second after the event occurrence, the primary and intermediate pumps trip after 1.6 seconds, and the reflector starts lowering 1.6 seconds later. Reactor power momentarily increases up to 1.6 times rated power, due to the reactivity insertion, until the reflectors start lowering. As a result, fuel and cladding temperatures increase. However, the increase of reactor power occurs for only a very short time and since the core power density is low, the rise in core temperature is small. The peak fuel temperature and the peak cladding temperature of the maximum-temperature pin do not exceed 621°C and 583°C, respectively.

The flow of primary and intermediate coolant changes to natural circulation flow, and residual heat is removed by the RVACS and IRACS. In this analysis, emergency (backup) power is not expected to work. RVACS and IRACS remove residual heat by natural circulation.

Table 4.2.3-1. Sequence of Events for FCC

Time (sec)	Events
0.0	Failure of one cavity can, insertion of positive reactivity
0.6	Transmittal of signals: Neutron flux level "High" signal Scram initiation signal RHRS initiation signal
1.6	Trip of the primary, intermediate, and feedwater pumps
1.6	Start lowering of the reflector
2.1	Start opening of the AC damper
9.6	Finish lowering of the reflector
32.1	Finish opening of the AC damper
61.6	The flow of primary and intermediate coolant changes to natural circulation flow, and residual heat is removed by the RVACS and IRACS.

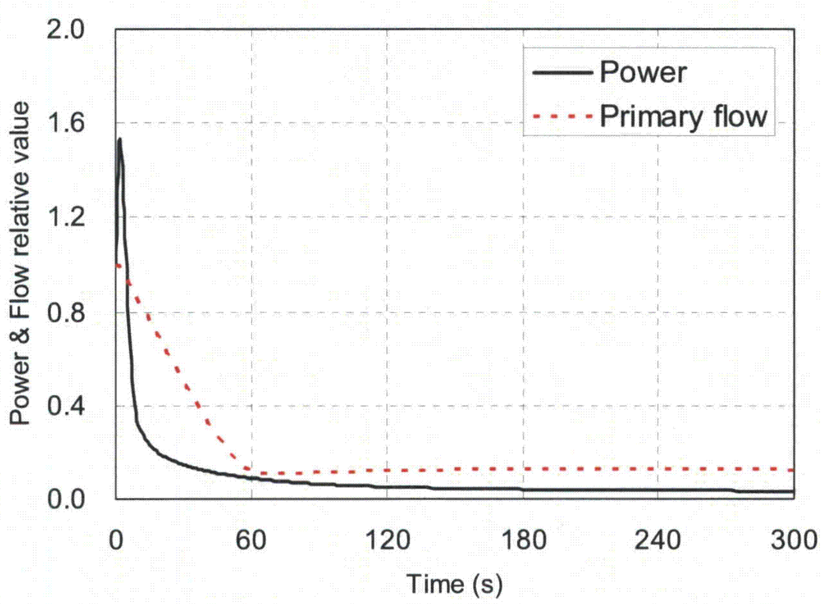


Figure 4.2.3-1. Power and Primary Flow (FCC)

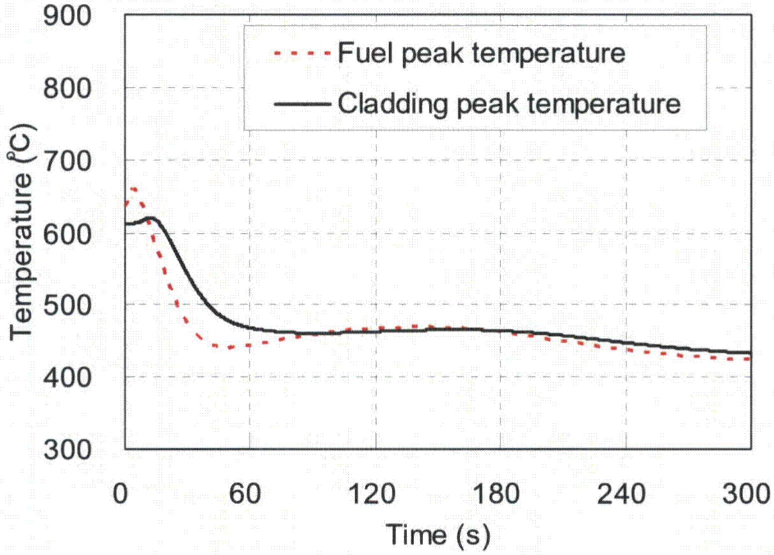


Figure 4.2.3-2. Fuel and Cladding Peak Temperatures (FCC)

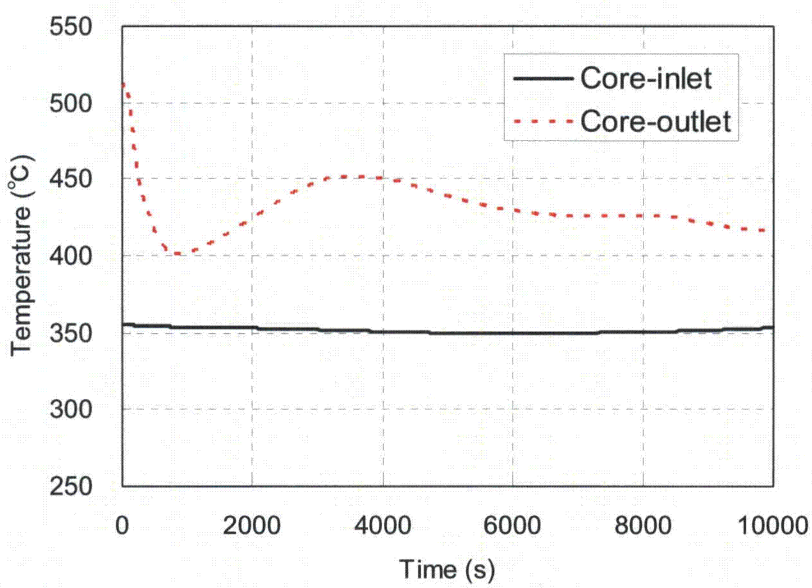


Figure 4.2.3-3. Core Inlet and Outlet Temperatures (FCC)

4.3 PARTITIONING OF EVENTS INTO TIME PHASES

As described below, in the 4S PIRT, phenomena seen in the events defined in Section 4.1 are identified and their ranking of relative importance is established. However, the importance of the phenomena may vary within the same event with its progress. For example, the LOSP event can be divided into two phases, which are the time from when the effect of forced flow diminishes after the EMP shuts down until natural circulation becomes dominant, and the time that follows. Therefore, in the 4S PIRT project, events are classified for each phase that contains the same important phenomena to establish the relative importance ranking for those phenomena. Specifically, the three events defined in the Section 4.1 were divided into the following time phases.

(1) Loss of Offsite Power (LOSP)

This event is divided into two time phases (Figure 4.3-1).
1st phase: Event initiation until natural circulation is established (0s ~ 100s)
2nd phase: Residual heat removal by IRACS and RVACS (100s ~)

(2) Sodium Leakage from Intermediate Piping (SLIP)

This event is divided into two time phases.
1st phase: Event initiation until natural circulation is established (0s ~ 100s)
2nd phase: Residual heat removal by RVACS (100s ~)

(3) Failure of a Cavity Can (FCC)

After the shutdown of the EMP and the transition to natural circulation, this event will develop in the same manner as the LOSP. Therefore, the only time phase that must be considered in this event is 0s ~ 100s, which is the time taken to establish natural circulation, and none of the phenomena following the 1st phase are considered.

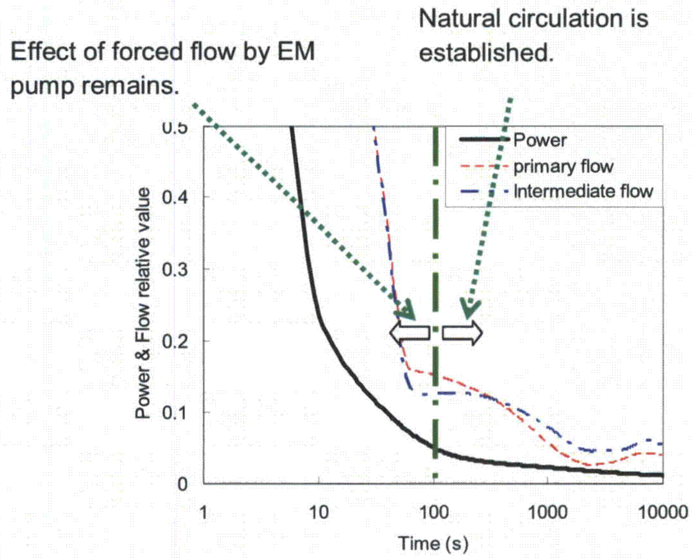


Figure 4.3-1. Partitioning of LOSP Event into Time Phases

References

[4-1] ADAMS: ML092170507, "4S Safety Analysis Technical Report," July 2009.

[4-2] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 15.0, Accident Analysis, U.S. NRC, March 2007.

5 PARTITIONING OF PLANT SYSTEM

In the PIRT process, to make the selection of plausible phenomena easier, the overall plant system is divided into subsystems or components. Here, "plant system" means the whole plant system, and "subsystem" means individual plant system, such as "Core and Fuel Assemblies," "Reactor System," and so on.

The overall plant system is divided into five subsystems from the aspect of thermal-hydraulic behavior.

- Core and fuel assemblies
- Reactor system
- Primary heat transport system (PHTS)
- Intermediate heat transport system (IHTS)
- Residual heat removal system (RHRS)

Also, the subsystem of the RPS is included in the following partition.

- Instrumentation and control (I&C) system

These six subsystems are divided into components and those components are further divided into subcomponents.

The results of this partitioning of the 4S plant system are shown in Table 5-1 and Figure 5-1; the six subsystems and components of the 4S plant system are described in detail in Table 5-2. The index (A, B, etc.) in Table 5-2 is the index of subsystem, component, or subcomponent in Table 5-1.

Figures are provided to help in understanding the description of subsystems and components.

- Figure 5-2 shows the core layout and the reflector.
- Figure 5-3 shows the structure of a fuel assembly.
- Figure 5-4 shows the cross-section of the reactor assembly.
- Figure 5-5 shows the reactor internals.
- Figure 5-6 shows the configuration of the primary EMP and the IHX.
- Figure 5-7 shows the flow coastdown system of the EMP.
- Figure 5-8 shows the SG configuration.
- Figure 5-9 shows the schematic diagram of the RHRS.
- Figure 5-10 shows the plant protection sensors.

Table 5-1. Subsystems, Components, and Subcomponents

Subsystem	Component	Subcomponent	Index		
Core/Fuel Assemblies	-	-	A	-	-
Reactor System	Reactor vessel	-	B	B1	-
	Reactor internal structures	General	-	B2	B20
		Reflector	-	-	B21
		Lower plenum	-	-	B22
		Upper plenum	-	-	B23
		Vertical shroud	-	-	B24
		Radial shield	-	-	B25
		Reactivity control drive mechanism	-	-	B26
Primary Heat Transport System	General	-	C	C0	-
	IHX	-	-	C1	-
	Primary EMP	-	-	C2	-
Intermediate Heat Transport System	General	-	D	D0	-
	Intermediate EMP	-	-	D1	-
	Steam generator system	-	-	D2	-
Residual Heat Removal System	Air cooler of IRACS	-	E	E1	-
	RVACS	-	-	E2	-
Instrumentation and Control System	I&C equipment	Plant protection sensors	F	F1	F11
		Others	-	-	F12

Table 5-2. Description of Subsystems and Components

Index	Subsystem/Component/ Subcomponent	Description
A	Core/fuel assemblies	The core is composed of two types of fuel assemblies with different enrichments of uranium (total 18 fuel assemblies). It is 2.5m high with a thermal power of 30 MWt and requires no refueling for 30 years. Reflectors that control neutron leakage are arranged circumferentially and control core power by moving the reflector up and down. An assembly that integrates the fixed absorber that compensates for burnup reactivity and the shutdown rod with shutdown function for backup protection system is placed in the center of the core. A five-layered neutron shield is installed on the outermost circumference of the core. Metallic fuel (alloy of uranium and zirconium) is used and is encased in the fuel cladding with sodium. One fuel assembly is composed of 169 fuel pins. The space between the fuel pins is maintained by grid spacers. Each fuel assembly obtains a predetermined flow rate using a flow orifice placed at the module of the core internal structure. A neutron shield function is incorporated at the top and bottom of the fuel assembly.
B	Reactor system	The reactor system is composed of reactor vessel, shield plug, reactor internal structures, shroud, upper and lower plenum, and primary heat transport system equipment. The core is supported by the core support structure and is placed in the axis center at the lower part of the reactor vessel. The core support structure have a protection function against mispositioning of the core assembly and a flow rate adjustment function, and is supported by the lower part of the reactor vessel. The backup core support structure is installed beneath the core support structure to prevent the unexpected falling of the core. The heat energy generated in the core transfers from the primary coolant through the IHX to the intermediate coolant. Reactivity of the core is controlled by the reflector placed on the outer circumference of the core, and the reflector control mechanisms are supported on the shield plug. Equipment in the reactor related to the reactor and core is designed considering nuclear radiation effects. The reactor vessel is a cylindrical vertical vessel with a lower-end half-ellipse-shaped plate, and a guard vessel surrounds the reactor vessel in case of unexpected leakage. The space between the reactor and guard vessels is filled with nitrogen gas.

Table 5-2. Description of Subsystems and Components (cont.)

Index	Subsystem/Component/ Subcomponent	Description
B1	Reactor vessel	The reactor vessel is a cylindrical vertical vessel with a lower-end half-ellipse-shaped plate, an upper flange support, and a core support flange at the lower part of the vessel. A shielding plug seals the upper part of the reactor vessel, and encloses the surface of the liquid sodium in an argon cover gas atmosphere. The inner diameter and height of the reactor vessel barrel are 3.5m and 24m, respectively.
B2	Reactor internal structures	The reactor internal structures include the core support structure, reflector, lower plenum, upper plenum, vertical shroud, and radial shield, and comprise the flow path for the primary coolant. The core support structure is composed of core support plate, upper core support plate, which support and fix components including core fuel assemblies, shutdown rod, and fixed absorbers.
B21	Reflector	The reflector controls reactivity by moving the six fan-shaped structures placed on the outer circumference of the core. The reflector is composed of a reflector region and a cavity region. The reflector is assembled from laminated plates of ferritic steel and is connected to the reflector drive unit on the shielding plug through the drive shaft. Each sector of the cavity region consists of six cavity cans for each of the six segments of the reflector. The cavity region is placed above the reflector region. The function of the cavity region is to enhance the neutron leakage to the surrounding sodium coolant. Each cavity can is cylindrical and is filled with argon gas.
B22	Lower plenum	The lower plenum is the region formed by the lower core support plate and lower end plate of the reactor vessel. It changes the flow direction of the primary coolant from the outlet of the radial shield to the core inlet, and also has the function of coolant mixing in the plenum.
B23	Upper plenum	The upper plenum is the region ranging from directly above the fuel assembly to the free surface level of sodium in the primary system. Sodium level varies due to the volume variation caused by the primary coolant temperature under conditions of normal operation, shutdown, and reactor scram. The sodium level under normal operation includes margin so that the liquid level exceeds the inlet of the IHX in case of unexpected leakage from the reactor vessel.

Table 5-2. Description of Subsystems and Components (cont.)

Index	Subsystem/Component/ Subcomponent	Description
B24	Vertical shroud	The upper vertical shroud is the structure that divides the upper plenum, IHX, and EMP, and suppresses the amount of heat transfer from the high-temperature (510°C) sodium that flows out of the core outlet to the low-temperature (355°C) sodium that flows out of the EMP. Heat-insulating material is contained inside the upper vertical shroud. Differences in thermal expansion caused by the difference in temperature between the inside and outside of the upper vertical shroud are accommodated by a bellows placed on the upper part of the upper vertical shroud.
B25	Radial shield	The radial shield is the structure having the functions to suppress the ductile reduction of the reactor vessel by neutron irradiation and to prevent radioactivation of the air outside the guard vessel. The radial shield is placed in the outermost circumference of the core. The shield is composed of an inner cylinder filled with B_4C powder and an outer cylinder to protect inner cylinder from thermal expansion and swelling of B_4C. The radial shield is composed of shield segments with different external diameters and consists of five layers.
B26	Reactivity control drive mechanism	The reactivity control system is composed of a reflector, reflector drive mechanism, shutdown rod, shutdown rod drive mechanism, and fixed absorbers. The cylindrical reflector has a six-segment structure that can be controlled individually and is located outside the core barrel. The reflector drive mechanism consists of three different drive systems for startup and shutdown by power cylinder, for power control, and for burnup swing compensation that consists of ball screw, motor, and reduction gear. When needed, the scram function releases the brake on the power cylinder for startup and shutdown, and the reflector descends by its own weight. At the center of the core, one shutdown rod and fixed absorbers divided into six fan-shaped pieces surrounding the shutdown rod are enclosed in a cylindrical protecting tube on the center of the hexangular wrapper tube. The shutdown rod is a cylindrical tube containing B_4C pellets. The drive system of the shutdown rod employs a motor-driven ball screw mechanism and is used to raise the shutdown rod before startup. At the time of a scram, the shutdown rod is detached from the drive system, and is inserted into the core by gravity. For a scheduled shutdown, the shutdown rod is inserted into the core by the shutdown rod drive system.

Table 5-2. Description of Subsystems and Components (cont.)

Index	Subsystem/Component/ Subcomponent	Description
B26 (cont.)		The fixed absorber is hafnium and sheathed in stainless steel. The fixed absorber is located in the core at the start of operation. After a predetermined operating period, the fixed absorber is manually withdrawn, and positive reactivity is added to compensate for fuel burnup. When moving the fixed absorber, the reactor is shut down once, so the fixed absorber is not moved during reactor operation.
C	Primary heat transport system (PHTS)	The PHTS cools the core and is composed of the IHX, primary EMPs, and shroud. Primary coolant that flows out of the core rises through the center of the reactor vessel, flows into the IHX, falls through the heat transfer tube, and reaches the suction port of the primary EMP. Coolant is then pumped down through the shield region by the driving force of the two EMPs in series, redirecting at the lower plenum, and returns to the core inlet.
C1	Intermediate heat exchanger (IHX)	The IHX is a vertical shell-and-tube, straight-tube, parallel counterflow type. Primary coolant flows from the upper side of the heat transfer tubes arranged in several concentric circles with different radii around the shell inside of the heat exchanger, falls through the tube, and flows out from the lower part of the heat transfer tube. The inlet and outlet flow paths of the intermediate coolant are formed by the structure of the double annular shell on the upper part of the heat exchanger. Intermediate coolant falls through the outside inlet annular flow path, is redirected at the lower tube plate, rises through the outside of the heat transfer tube, and flows out of the outlet annular flow path of the intermediate system. The IHX is supported by the upper flange and is combined with the primary EMPs as a single unit.

Table 5-2. Description of Subsystems and Components (cont.)

Index	Subsystem/Component/ Subcomponent	Description
C2	Primary EMP	To circulate the primary coolant, two primary EMPs are installed sequentially. The pumps are induction-type and the flow path has an annular structure. Power is supplied from the three-phase ac power supplies. Flow rate is controlled by fixing the ratio of voltage and frequency. Heat generation in the pumps is cooled by the primary coolant. The EMPs are positioned around the center of the reactor vessel axis direction and near the inner surface of the reactor vessel. Compared to a mechanical pump, an EMP has little fluid inertia. The pumps are equipped with a motor-generator set (MG set) with a flywheel as backup power to compensate for the lack of fluid inertia when the normal power supply is stopped. The MG set can supply sufficient power to maintain an adequate flow coastdown during a power loss. The EMP exciter sends an electric current to the rotor of the generator at the time of the flow coastdown and startup.
D	Intermediate heat transport system	The IHTS consists of one loop including a steam generator, air cooler, intermediate EMP, EMF, and piping. This system circulates the intermediate coolant (sodium) heated by the IHX of the primary heat transport system. The heat of the intermediate coolant is transferred to the water/steam in the steam generator. The air cooler is in the standby state during normal operation and removes decay heat during normal shutdown and transients.
D1	Intermediate EMP	To circulate the coolant, one EMP is provided on the intermediate loop cold leg. This pump is an induction type, and the flow path has an annular structure and is connected to intermediate system piping. Power is supplied from the three-phase ac power supplies. Flow rate is controlled by fixing the ratio of voltage and frequency. Heat generation in the EMP is cooled by the intermediate coolant. Compared to a mechanical pump, an EMP has little fluid inertia. The pumps are equipped with an MG set with a flywheel as backup power to compensate for the lack of fluid inertia when the normal power supply is stopped. The MG set can supply sufficient power to maintain an adequate flow coastdown during a power loss.

Table 5-2. Description of Subsystems and Components (cont.)

Index	Subsystem/Component/ Subcomponent	Description
D2	Steam generator (SG) system	The SG is a once-through shell-and-tube type equipped with double-wall heat transfer tubes of the helical-coil type. Heated intermediate sodium flows from the upper part of the SG through the distribution pipe into the upper part of the heat transfer tube bundle. When flowing down in the tube bundle, intermediate sodium exchanges heat with water/steam in the heat transfer tube, and flows out the outlet nozzle at the lower part of the SG. There are three points of the feedwater nozzle on the lower part of the SG, and water/steam flows from there into the heat transfer tube and is heated by sodium while rising through the helical-coiled heat transfer tube bundle, flowing out the three steam nozzles in the upper part of the SG. The heat transfer tubes have a double-wall tube structure with a wire mesh layer placed between the inner and outer tubes. The space between inner and outer tubes is filled with helium gas and is connected to a leak detection system.
E	Residual heat removal system	The RHRS is composed of the intermediate reactor auxiliary cooling system (IRACS) that removes decay heat using an air cooler in the intermediate heat transport system and the reactor vessel auxiliary cooling system (RVACS) that continuously removes heat by natural air draft from outside the guard vessel.
E1	Air cooler (AC) of IRACS	The AC is a fin-tube type with the inlet/outlet equipped with a damper and an air blower for forced-air cooling. It is placed in the cold leg piping area between the IHX and intermediate EMP, and has a stack 15m high. The heat exchange under forced-air circulation is 0.7 MWt.
E2	RVACS	The RVACS is composed of the guard vessel, cylindrical heat collector, cylindrical concrete structure surrounding the reactor vessel, air inflow and outflow paths, and an exhaust stack. The heat collector is placed outside the guard vessel, and cylindrical concrete wall is placed outside the heat collector. Cold ambient air flows from intake at the upper part of the reactor building, descends between the cylindrical concrete wall and heat collector, returns to the lower end of the heat collector, and rises through the space between the heat collector and guard vessel. High-temperature air is exhausted from the exhaust stack in the reactor building.

Table 5-2. Description of Subsystems and Components (cont.)

Index	Subsystem/Component/ Subcomponent	Description
E2 (cont.)		Heat transfers between the reactor vessel and guard vessel mainly by radiation. Heat transfers between the guard vessel and heat collector by both radiation and convection. The RVACS can remove heat using only passive means.
F	Instrumentation and control system	The I&C system is composed of reactor instrumentation and process instrumentation used to obtain information on the reactor, reactivity control system, reactor protection system for automatic reactor shutdown, engineered safety features actuation system (ESFAS) to activate systems preventing expansion of an accident, and control room.
F1	Instrumentation and control equipment	The reactor instrumentation includes neutron instrumentation, reactor vessel in-vessel instrumentation, failed fuel detector, and control element position indicators that are placed to collect information specifically about the reactor. Process instrumentation is used to conduct measurement for required process quantities in each auxiliary system including the primary system for proper and safe plant operation. Several signals are used for the reactor protection system, ESFAS, and reactor control system.
F11	Plant protection sensors	Instrumentation of the reactor protection system is composed of monitors for neutron flux, IHX primary outlet temperature, reactor vessel sodium liquid level, primary EMP voltage and current, seismic acceleration, and power line voltage. The reactor protection system accomplishes reactor shutdown automatically by initiating the reactor shutdown system to protect the core and reactor coolant boundary using the signals of these instruments.
F12	Other sensors	Other sensors include those for the ESFAS, control room, reactor control system and monitors such as for primary flow rate, IHX primary inlet temperature, intermediate flow rate, outlet sodium temperature of the air cooler, SG outlet temperature, reactor vessel cover gas pressure, and radiation levels.

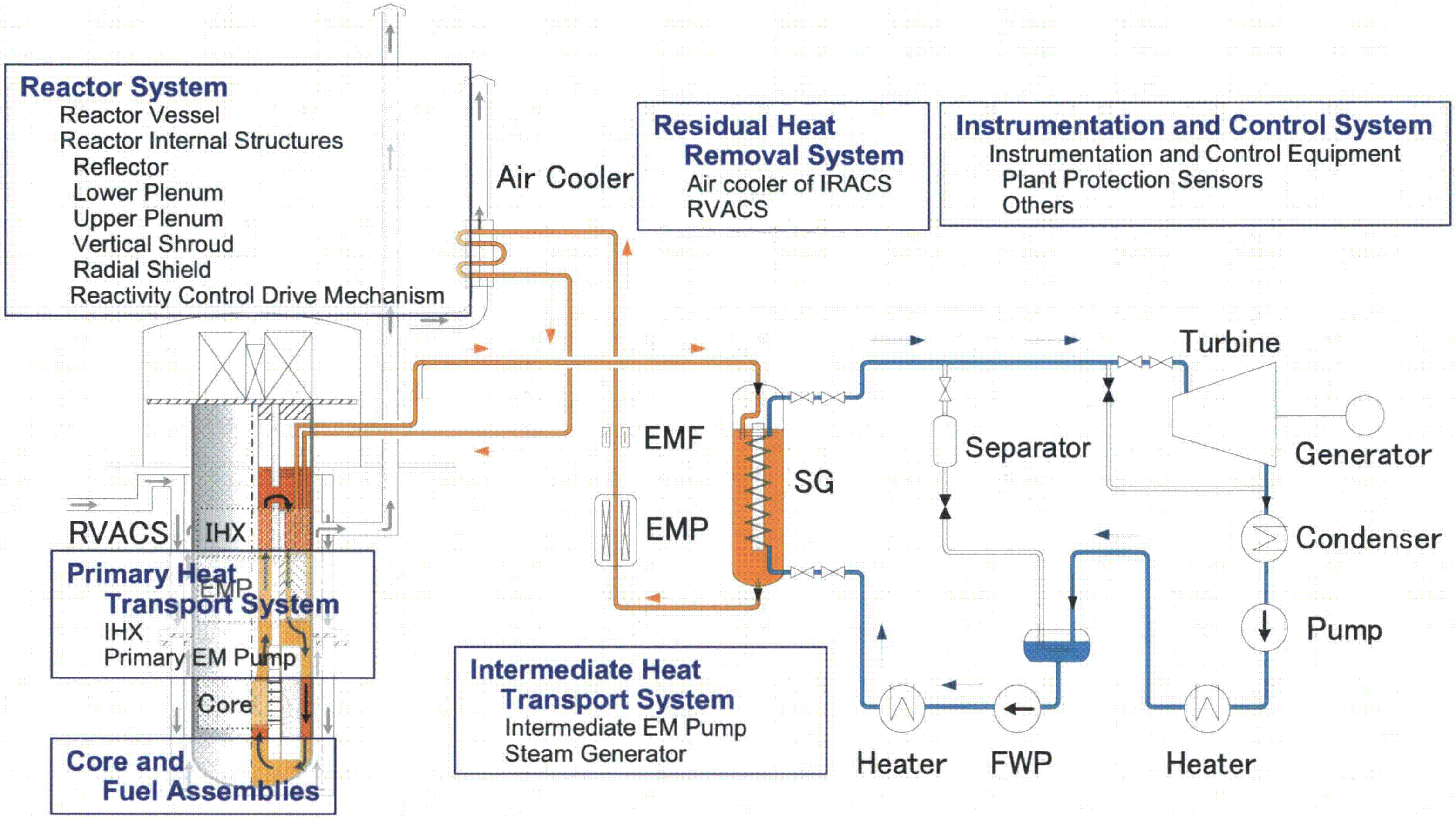


Figure 5-1. Subsystems and Components Considered in the 4S PIRT

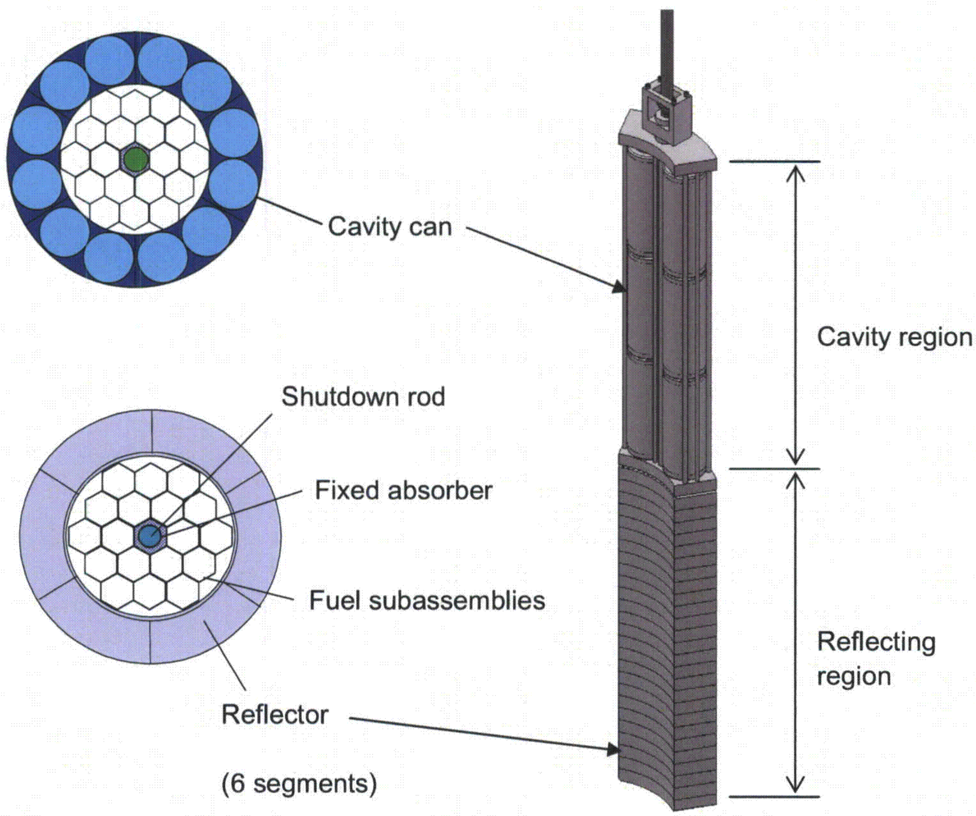


Figure 5-2. Core Layout and Reflector

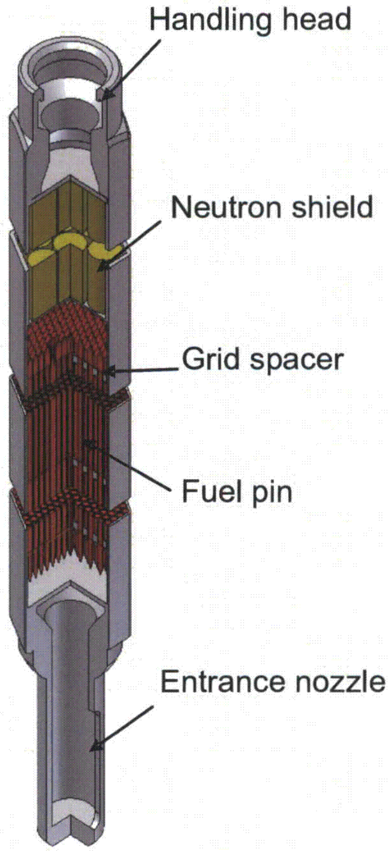


Figure 5-3. Structure of Fuel Assembly

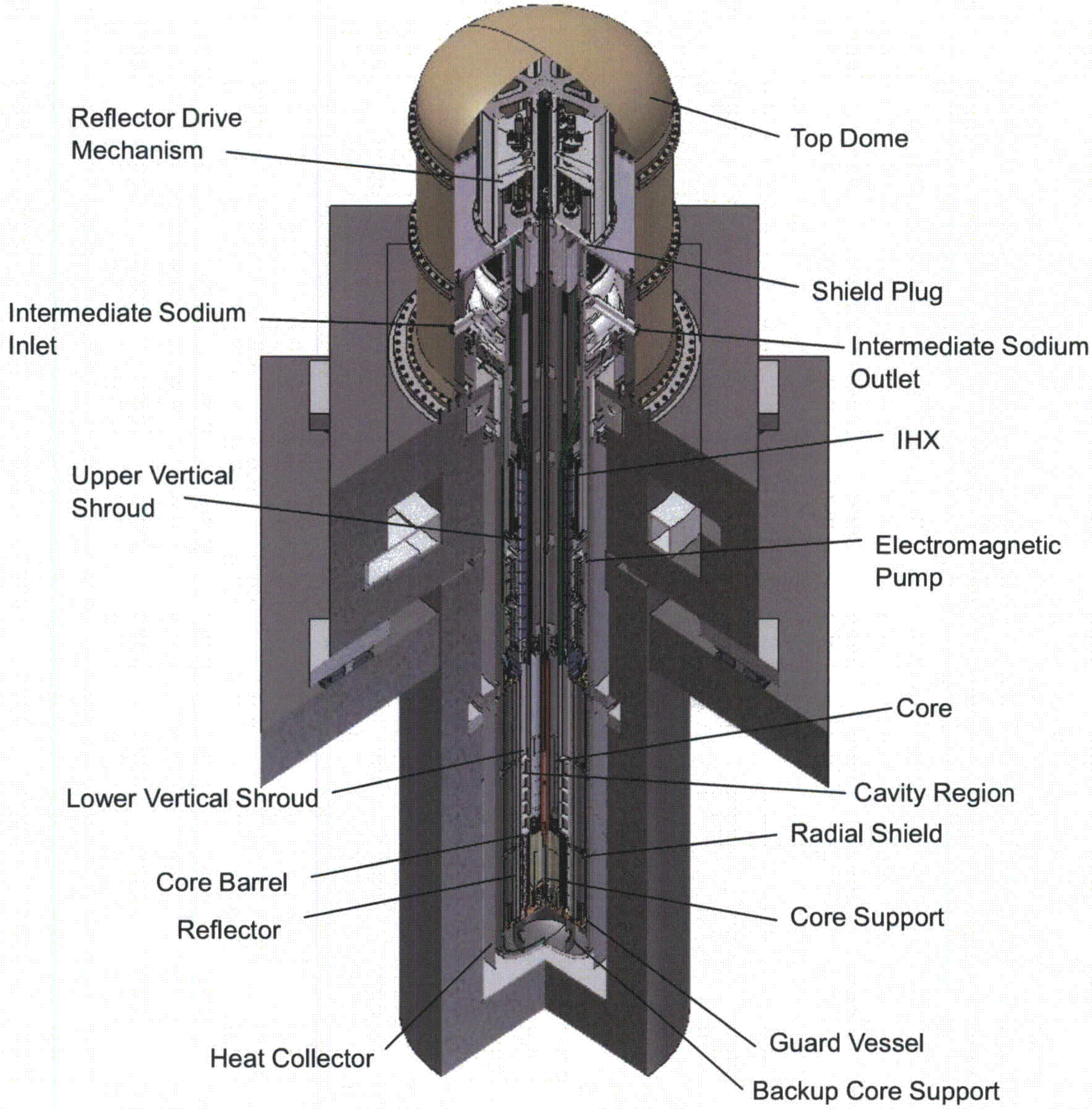


Figure 5-4. Cross-Section of Reactor Assembly

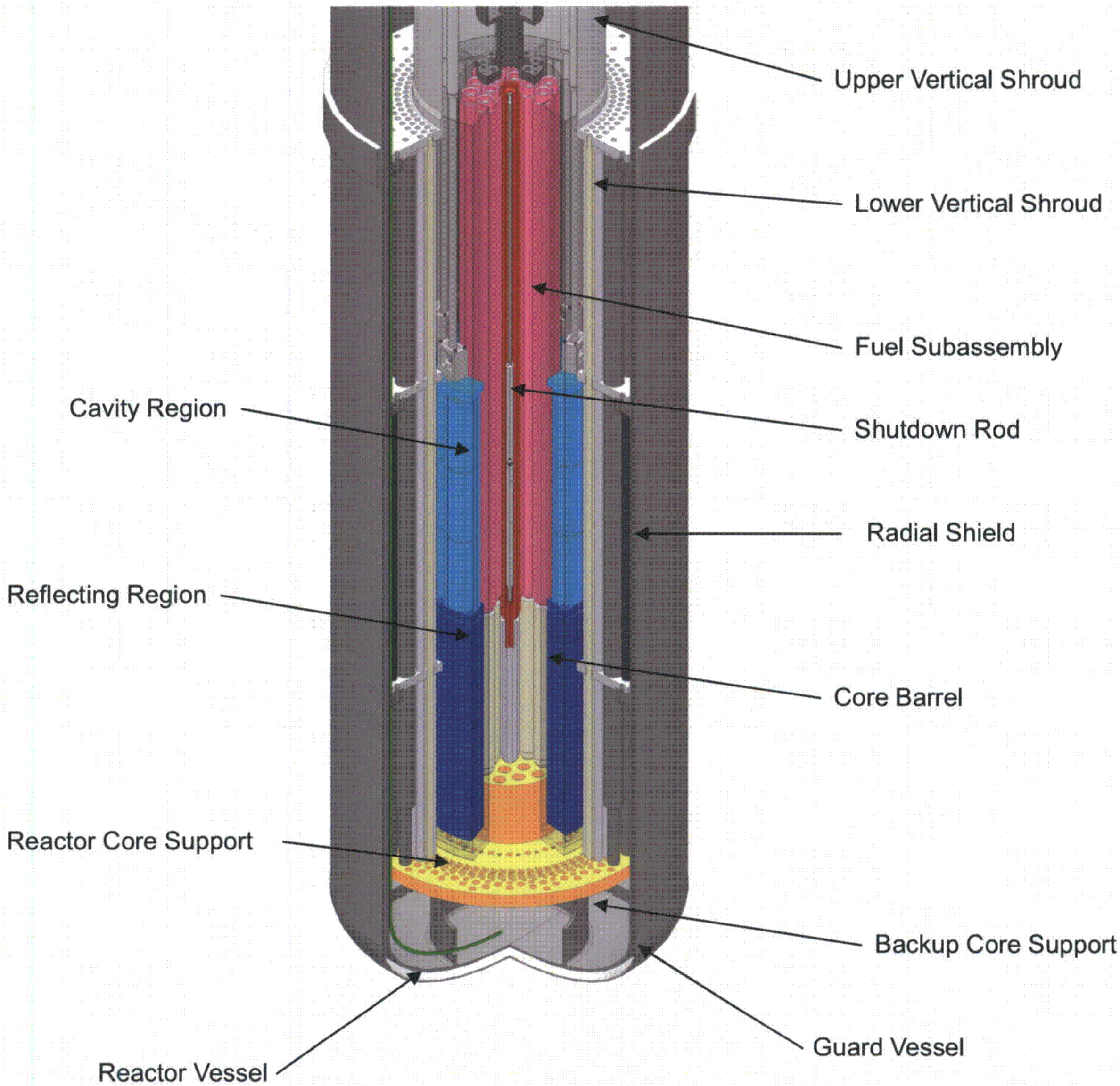


Figure 5-5. Reactor Internals

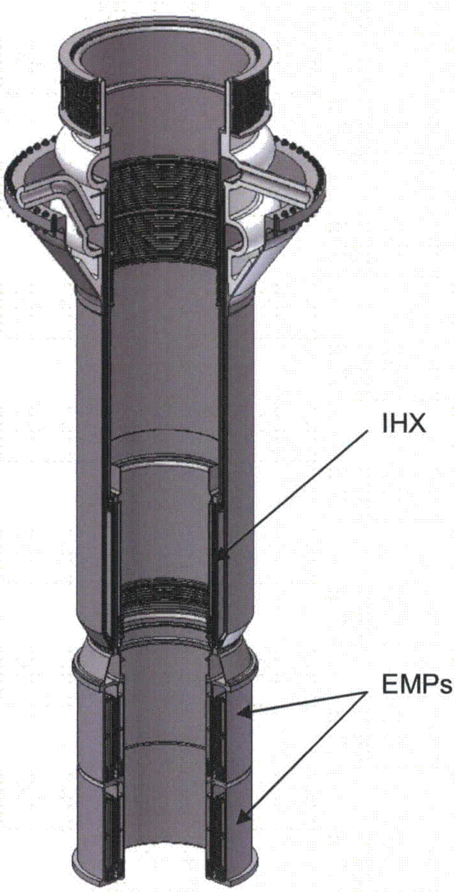


Figure 5-6. Configuration of Electromagnetic Pumps and Intermediate Heat Exchanger

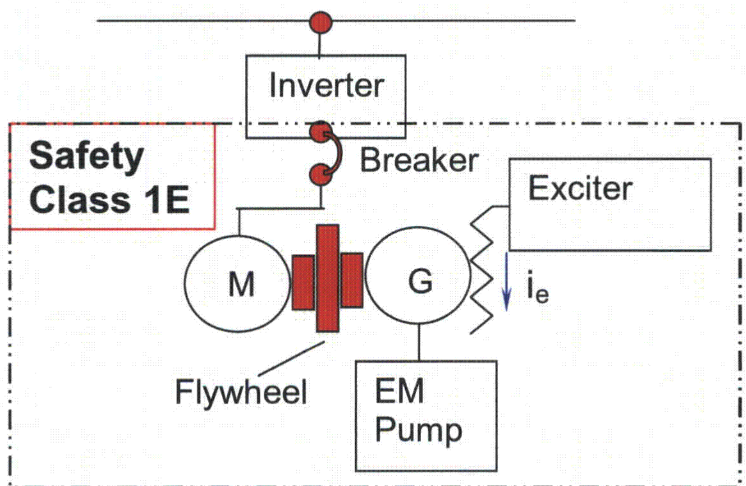


Figure 5-7. Flow Coastdown System of Electromagnetic Pump

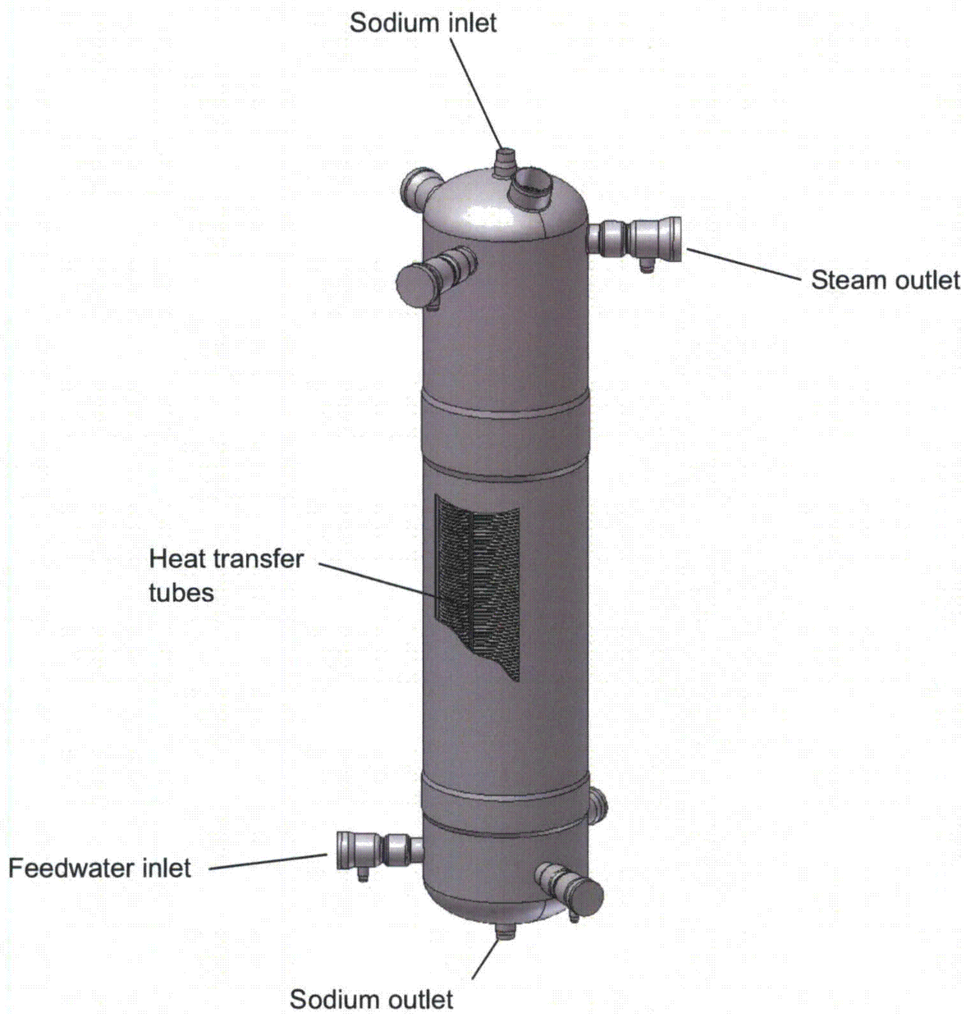


Figure 5-8. Steam Generator Configuration

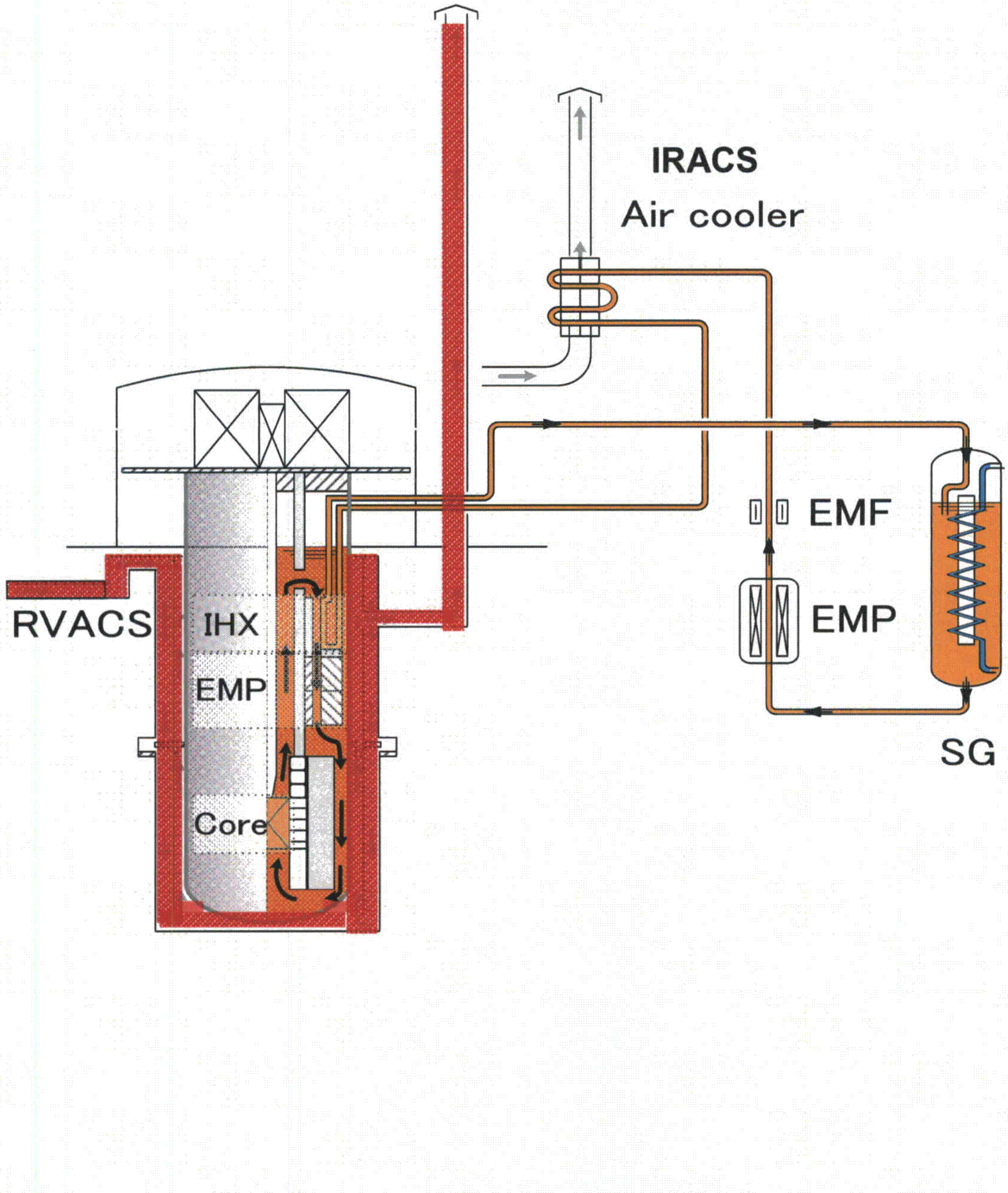


Figure 5-9. Schematic Diagram of Residual Heat Removal Systems

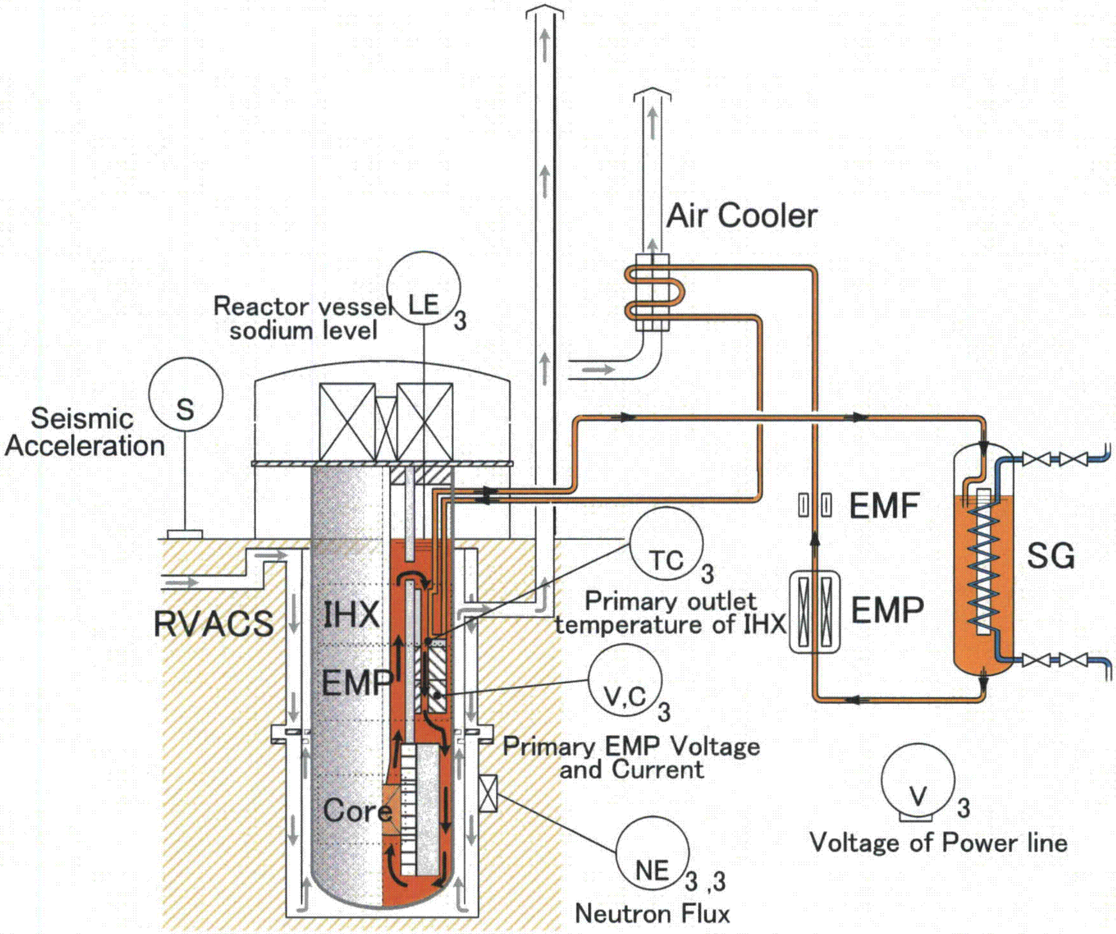


Figure 5-10. Plant Protection Sensors

6 FIGURE OF MERIT

In PIRT studies, the degree of importance of a phenomenon is evaluated by its relative importance against a criterion called the Figure of Merit (FoM).

The selection of the FoM is a very important step in the overall PIRT process.

In some past PIRTs, the acceptance criteria for the safety analysis were used as the FoM.

The acceptance criteria for the safety analysis mentioned here mean quantitative allowance criteria used to define an acceptable solution. For LWRs, NRC policies and the Code of Federal Regulations (CFR) such as 10 CFR 50 Appendix A [6-1] and Standard Review Plan (SRP) [6-2], provide these criteria. These regulatory safety requirements require that the reactor system be maintained in a safe condition during an accident, transient, and rated operation.

For example, 10 CFR 50.46 [6-3] provides the following five criteria for the LOCA event for an LWR.

(1) Peak cladding temperature (PCT): The evaluated fuel cladding temperature shall not exceed 2200°F (1221°C).

(2) Maximum cladding oxidation: The maximum oxide quantity of the cladding shall not exceed 17 percent of the thickness of the cladding before oxidization.

(3) Maximum hydrogen generation: The quantity of hydrogen generated by the reaction of cladding and water/steam, shall not exceed 1 percent of the quantity of the hydrogen that is generated when the whole cladding reacts.

(4) Coolable geometry: Change in shape of evaluated core shall not interrupt the cooling process for the fuel pin.

(5) Long-term cooling: During the period of time defined from the characteristics of long-life radioactive material, decay heat shall be removed in the specified time and the evaluation value of core temperature shall be maintained at an acceptable low value.

As shown in Table 6-1, the regulatory safety requirements above result from a hierarchy of requirements [6-4]. The most important requirement stipulated for the reactor system is "Protect public health and safety," 10 CFR 1.11 [6-5], which indicates this statement should be placed on the top level of the hierarchy. This is the essential part of safety for the reactor system, i.e., the primary regulatory issue, and the origin for the selection of the FoM. Also, the requirements of the lower hierarchy levels express the detailed content of upper levels, so the requirement items of each lower level must also encompass the items of higher levels. As an illustration, the purpose of "limit fission product release," which is the second requirement, is to "protect public health and safety," which is the top requirement. And the purpose of the "limit fuel failure," which is the third requirement, is to "limit fission product release," which is the second requirement. The five requirements above correspond to the fourth level for LOCA. So, to simplify and make the selection of FoM easier, a number of PIRTs select physical quantities that are directly related to the requirements above [6-4] as FoM.

The following are the characteristics required for FoM [6-4]:

- Directly related to issue
- Directly related to phenomena
- Easily comprehended
- Explicit
- Measurable

The above-mentioned requirements on the top level, such as the regulatory safety requirement, are clearly related to the phenomenon directly, but cannot be easily comprehended or measured. On the other hand, PCT and similar factors, which are shown in level 4 in Table 6-1, can be selected as Figures of Merit, because these are measurable and explicit.

When evaluating the relative importance of a phenomenon, however, it is not necessary to select any level 4 parameter as an FoM. If there is a more specific physical quantity at a lower level that meets the requirements above, it is preferable to use it as surrogate FoM. Actually, some PIRTs select physical quantities from a lower level as FoM. For example, although the PIRT [6-6] for the Simplified Boiling Water Reactor (SBWR) plant selects "total reactor coolant system inventory" as one of the FoM, it is not described in SRP. In this case, "total reactor coolant system inventory" is a substitute for PCT.

In the selection of FoM for the 4S PIRT, the team began with "protect public health and safety," which is the highest level requirement in Table 6-1, and focused on "the integrity of the fuel pin" and "the integrity of the reactor vessel." This is because there is no precedent for the PIRT of an LMFR, including 4S, and no authoritative reference for FoM selection. Also, in the selection of FoM for an LMFR, it is considered that experience obtained in FoM selection for an LWR has insufficient value in providing useful information.

(1) Integrity of Fuel Pin Cladding

In the 4S PIRT, cladding temperature was selected as the criterion indicating soundness of the fuel pin cladding. In making this selection, the acceptance criteria for the safety analysis for reactors such as the Experimental Breeder Reactor EBR-II [6-7] and the Power Reactor Inherently Safe Module (PRISM) [6-8], which use metallic fuel and HT-9 for the fuel and the cladding, respectively, were considered. These reactors utilize four parameters for safety standards in the safety analysis: cladding temperature, cumulative damage fraction (CDF), a plastic hoop strain, and the liquid phase penetration value of cladding by eutectic formation. However, because the amount of plastic hoop strain, CDF, and the amount of liquid phase penetration of cladding by eutectic formation represent a summation of stresses throughout the plant lifetime, these are not suitable as FoM for the PIRT, which evaluates results of sensitivity analysis for only one event.

Therefore, only the cladding temperature is left as a candidate for FoM, and these three parameters can be removed from consideration as FoM in the 4S PIRT.

(2) Integrity of Primary Coolant Boundary

An FoM for "Integrity of fuel pin cladding" can be substituted for one for the primary coolant boundary because the requirements for "Integrity of fuel pin cladding" are more severe than those for the RV in the events treated in this report, as explained in the following.

From the transient change of cladding temperature for the three events described in Chapter 4, the maximum coolant temperature at the core inlet does not exceed the temperature at rated operation, or if it is exceeded, it is only by a few degrees. Also, generally, the maximum coolant temperature at the core inlet is equal to or higher than that of the "Primary coolant boundary." Therefore, it can be concluded that soundness of the "Primary coolant boundary" in a transient can be assured if the "Integrity of fuel pin cladding" is maintained.

Returning to clause (1), "Integrity of fuel pin cladding," cladding temperature defined in clause (1) either corresponds to or is higher than the temperature of "primary coolant boundary" at rated conditions.

As also discussed in Chapter 4, cladding temperature at transient does not remain higher than temperature at rated operation for a long time (i.e., 600 seconds).

Additionally, the cladding is continuously subjected to high inner pressure from the fission gas generated by fuel burnup, whereas the RV wall has been designed for low pressure. The cladding consists of HT-9, which has inferior creep strength compared to SUS304, which is the RV material.

Based on the reasons cited above, the requirement imposed for "fuel pin cladding" is more severe than that for the "primary coolant boundary."

Based on the preceding discussion, by using cladding temperature as the FoM for "Integrity of the primary coolant boundary," both the integrity of the fuel pin cladding and primary coolant boundary can be treated. Therefore, cladding temperature was selected as the FoM. This is common for all three events selected in Chapter 4.

Table 6-1. Hierarchy of Regulatory Safety Requirement and Example of Criteria

Level	Source	Criteria	Directly Related to Issue	Directly Related to Phenomena	Easily Comprehended	Explicit	Measurable
1	10 CFR 1.11 [6-5]	Protect public health and safety	Primary Regulatory Issue				
2	10 CFR 100.11 [6-9]	Limit fission product (FP) release	✓	✓	–	–	–
3	10 CFR 50 Appendix A [6-1]	Limit fuel failure and containment breach	✓	✓	–	–	–
4	SRP 6.2 Containments [6-2]	Limit containment pressure, temperature, etc.	✓	✓	✓	–	–
	SRP 15.1.4 to 15.6.1 Non-LOCA [6-2]	Fuel limits, energy deposition, fuel temperature, etc.	✓	✓	✓	✓	✓
	10 CFR 50.46 [6-3] and SRP 15.6.5 LCOA [6-2]	Peak cladding temperature, hydrogen generation, etc.	✓	✓	✓	✓	✓
5	AP600: NUREG/ CR-6541, INEL-94/0061 Rev. 2 [6-10]	Vessel inventory	✓	✓	✓	✓	✓
	SBWR: NUREG/ CR-6472, BNL-NUREG-52501 [6-6]	Vessel inventory	✓	✓	✓	✓	✓

References

[6-1] 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."

[6-2] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

[6-3] 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

[6-4] B. E. Boyack and G. E. Wilson, "Lessons Learned in Obtaining Efficient and Sufficient Applications of the PIRT Process," BE-2004-International Mtg. on updates in best estimate methods in nuclear installations safety analysis, 2004.

[6-5] 10 CFR 1.11, "The Commission."

[6-6] P. G. Kroeger, et al., "Preliminary Phenomena Identification and Ranking Tables for Simplified Boiling Water Reactor Loss-of-Coolant Accident Scenarios," NUREG/CR-6472, BNL-NUREG-52501, 1998.

[6-7] L. K. Chang, et al., "A Method for the Determination of Technical Specifications Limiting Temperature in EBR-II Operation," 4th International Topic Meeting on Nuclear Thermal Hydraulics, Operation and Safety, 1994.

[6-8] NUREG-1368, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," February 1994.

[6-9] 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

[6-10] G. E. Wilson. et al., "Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios," NUREG/CR-6541, INEL-94/0061, Rev. 2, 1997.

7 PHENOMENA IDENTIFICATION

7.1 PLAUSIBLE PHENOMENA

Plausible phenomena in the PIRT process are all those that may have some influence on the FoM. The identification of plausible phenomena before ranking their relative importance is a primary means to help ensure that the full phenomena spectrum is identified.

Table 7.1-1 describes the plausible phenomena that were identified in the 4S PIRT. In this table, "Code" in the rightmost column is the index provided for each phenomenon, which is also used in Chapter 8.

The numbers of phenomena for each system are follows.

- Core and fuel assemblies 27 phenomena
- Reactor system 23 phenomena
- Primary heat transport system 13 phenomena
- Intermediate heat transport system 8 phenomena
- Residual heat removal system 15 phenomena
- Instrumentation and control system 5 phenomena

Figures are provided to help in understanding the phenomena descriptions.

- Figure 7.1-1 shows coolant flow in the reactor vessel.
- Figure 7.1-2 shows coolant flow inside the upper part of the reactor.
- Figure 7.1-3 shows the internal structures of the upper part of the reactor and an enlarged view of the coolant flow around the IHX.
- Figure 7.1-4 shows the intermediate coolant flow in the IHX.

The subsystems and components in Table 7.1-1 are described in Chapter 5.

Table 7.1-1. Descriptions of Plausible Phenomena

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	Pressure loss in core region	– Pressure loss along the coolant flow path of the bundle in the core/fuel assemblies is caused by friction losses, losses from flow contraction and expansion depending on the inlet and outlet geometry, and losses from the grid spacer. – Under rated flow conditions, pressure loss through the core is about 4.1×10^4 Pa, which represents about 70% of the total pressure loss along the flow path of the primary system.	a01
	Pressure loss in reflector region	– Pressure loss of the coolant flow path in the reflector region consists primarily of friction losses and losses from flow contraction and expansion, depending on the inlet orifice geometry. – Pressure loss under rated flow conditions is about 4.1×10^4 Pa.	a02
	Natural convection	– Coolant natural convection in the core is driven by a buoyancy force as a result of the difference in fluid density at high- and low-temperature regions in the core. – Strength of natural convection in the core is dependent on the position of hot spots in the core and temperature variations.	a03
	Reactivity feedback	– Reactivity feedback in the core is caused by thermal expansion and contraction of the fuel, coolant, structure materials, and core support plate, which is caused by the decrease or increase of reactor temperature and by the Doppler effect. – Power changes when reactivity is inserted.	a04

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	Gap conductance between fuel and cladding	– Fuel pin utilizes a sodium bond fuel. The gap between the fuel slug and cladding is filled with liquid sodium. – Therefore, the gap conductance between the fuel slug and cladding is dependent on the thermal conductivity of sodium.	a05
	Heat transfer between cladding and coolant	– Heat transfer between the cladding and coolant is dependent on coolant flow velocity near the cladding surface, the spacer geometry, and P (fuel pin pitch)/D (fuel pin diameter).	a06
	Intra- and inter-assembly flow distribution	– Flow distribution in the core is controlled by flow orificing at the bottom of the assembly. – Flow distribution inside a fuel assembly is controlled by bundle geometry. In natural convection decay heat removal, it is also controlled by a temperature distribution inside the bundle. – It is noted that the temperature distribution inside the bundle is controlled by flow distribution inside the bundle. – Flow distribution inside the fuel assembly deviates from the initial prediction due to errors such as manufacturing errors for the fluid resistance in fuel assembly. – These variations result in cladding temperature fluctuation.	a07

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	Radial heat transfer between subassemblies (S/A <-->sodium <--> S/A)	– Interassembly heat transfer in the radial direction is caused by heat transfer from the wrapper tube of the fuel assembly to the coolant in the gap between wrapper tubes and heat transfer from coolant in the gap to the wrapper tubes of the neighboring fuel assemblies. This inter-subassembly heat transfer behavior is dependent on operating conditions. During rated power and flow conditions, heat transfer is determined by the temperature difference between two adjacent subassemblies. During low-flow natural circulation decay heat removal conditions, heat transfer, and thus temperature distribution and natural convection flow in each subassembly, are all coupled and interdependent.	a08
	Heat transfer between reflector and coolant	– There are coolant flow paths inside and outside the reflector. Heat transfer between the reflector and coolant is dependent on the coolant flow velocity in each flow path and the geometry and size of the coolant flow path.	a09
	Heat capacity of core assemblies	– Heat capacity of the core assemblies is determined by the specific heat and the mass of the structural material making up the core fuel assemblies. It influences the rate of core and fuel assembly temperature changes during transients and accidents.	a10

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	Coolant boiling	– Boiling temperature of sodium coolant is 881°C at 0.1 MPa and rises with cover gas and sodium hydraulic head pressure increases. – Coolant boiling under any power conditions is not likely to occur due to sufficient margins to the boiling temperature from the coolant normal as well as transient temperatures.	a11
	Core power	– There are two types of heating for core power, nuclear heating and gamma heating. The main heating for core power is nuclear heating. Rated core power is 30 MWt.	a12
	Decay heat	– Decay heat is heat released by the decay of radioactive material. Even if the reactor shuts down, the radioactive material contained in fission products will continue to decay and release heat. – The level of decay heat will be about 0.4 MWt after 1 hour, about 0.18 MWt after one day, and about 0.08 MWt after 10 days from reactor shutdown at the end of fuel lifetime.	a13
	Heat transfer between core support plate and sodium	– Heat transfer between the core support plate and coolant is caused by convection heat transfer of the coolant from the top and bottom surfaces of the core support plate.	a14
	Rate of scram reactivity insertion	– Insertion rate of scram reactivity is the reactivity per unit time when the reflector is lowered (withdrawn) at a scram.	a15
	Delay of scram reactivity insertion	– Delay of scram reactivity is the delay in delatch and time from when the signal is transmitted until the reflector or shutdown rod actuates.	a16

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	Eutectic reaction between fuel and cladding	– Eutectic reaction is the phenomena in which two solid phases melt simultaneously and form a liquid phase. – Eutectic reaction of uranium, plutonium, and FPs in the fuel alloy, and cladding materials such as iron can occur in metallic fuel. – Eutectic reaction occurs at high temperature (higher than 650°C) and becomes more severe as temperature rises.	a17
	Temperature dependence of physical properties of materials	– Physical properties such as specific heat, density, thermal conductivity, and creep characteristics of core components are temperature dependent.	a18
	FP release from fuel slug into gas plenum	– Pores are a result from the FP gas accumulating in the fuel slug, increasing its pore density. Holes that lead to the outside are formed when these pores combine and link. FP gas is released to the outside of the fuel slug through these holes and transferred to the gas plenum region. – Sodium enters the fuel slug through the hole. – Effective conductivity changes by a variation of the pore density and the penetration of sodium. – FP release rate from fuel slug is dependent on the fuel temperature.	a19
	FP transport from fuel to sodium bond, and sodium in primary system	– FP released from the fuel into the sodium bond leaks into and is transported in the primary system sodium in case of a fuel failure.	a20

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	FP transport from sodium in primary system to cover gas	– FPs are transported through the sodium in the primary system and to the cover gas in case of a fuel failure.	a21
	Flow-induced vibration in a subassembly	– Fuel pin bundle vibrates by fluid and structure interaction when coolant flow velocity around the fuel pin bundle in the fuel assembly is high.	a22
	Inter-wrapper flow between wrapper tubes	– Coolant flow at the gap between the fuel assemblies. This flow influences inter-subassembly heat transfer phenomena during natural circulation decay heat removal.	a23
	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	– Coolant flow rate in an assembly deviates from the predicted value due to manufacturing errors such as assembly orifice, fuel pin, grid spacer, and wrapper tube.	a24
	Radial power distribution	– The fixed absorber region and the center of the core where a shutdown rod is placed are not in the fuel region, where neutrons are not generated. Also, since neutrons leak outside the core, neutron flux decreases outside the core and near the fixed absorber region in the center of the core. These create radial changes in neutron flux distribution. – Radial power distribution is in proportion to the neutron flux.	a25

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component	Phenomenon	Description	Code
Core and Fuel Assemblies	Axial power distribution	– Due to the leakage of neutrons from the top and bottom of the axial reflector regions, neutron flux decreases at the top and bottom in the axial direction of the core. – In proportion to the neutron flux, the part of axial power distribution surrounded by reflector is high and the part at the top and bottom in the axial direction of the core is low. – Axial power distribution of the core changes during core life.	a26
	Reactivity insertion by cavity can failure	– When a cavity can fails, sodium leaks into the failed can, replacing the argon gas and reducing neutron leakage, and as a result, positive reactivity is inserted.	a27

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component			Phenomenon	Description	Code
Reactor System	Reactor vessel		Temperature fluctuation of reactor vessel by change of liquid level	– If the RV wall is in direct contact with sodium near the sodium and cover gas interface region, the RV wall would suffer temperature fluctuation due to sodium level changes. – The RV, however, is separated from the hot sodium pool by the annular IHX inserted. – Therefore, direct contact of the hot plenum sodium with the RV wall is avoided by design and the sodium level changes do not lead to temperature fluctuation on the RV wall.	b01
	Reactor internal structures	General	Coolant mixing effect in upper plenum including thermal stratification	– Since primary coolant from the core flows into the IHX after crossing the upper shroud, the mixing in the upper plenum is determined by the core outlet flow distribution. – Flow velocity of primary heat transport system at rated operation is about 10 cm/s on average.	b02
			Temperature dependence of physical properties of structural materials	– Physical properties such as specific heat, density, thermal conductivity, and creep characteristics of structure materials have temperature dependency.	b03
			Natural convection	– Natural convection in the lower plenum, upper plenum, reflector, vertical shroud, and radial shield are caused by buoyancy produced by nonuniform temperature distribution in the component. – A variation of natural convection in each component above is dependent on the position of each hot spot and the temperature variation.	b04

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component			Phenomenon	Description	Code
Reactor System	Reactor internal structures	General	Flow-induced vibration	– Flow-induced vibration is the phenomenon whereby a structure vibrates when coolant flow velocity around it is high.	b05
		Reflector	Deformation due to thermal effect and irradiation	– Thermal deformations by core temperature fluctuation and irradiation deformation by neutron flux.	b06
			Flow in reflector region	– Since heat is also generated in the reflector region, coolant flow is introduced into the reflector region to remove the heat.	b07
			Effect of generated heat by neutron capture and gamma rays	– Heat generation in the reflector region is caused by neutron capture and gamma rays.	b08
		Lower plenum	Pressure loss	– Pressure losses at the core inlet region including the backup support mechanism at the bottom part of the core support plate. – Pressure loss is mainly due to contraction and expansion of the flow.	b09
			Heat capacity	– Heat capacity of the coolant, backup support mechanism, and core support plate in the lower plenum. – Heat capacity can be calculated by the specific heat and mass of above-mentioned coolant and structure materials.	b10

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component			Phenomenon	Description	Code
Reactor System	Reactor internal structures	Lower plenum	Coolant mixing and thermal stratification	– Coolant mixing of momentum and energy due to turbulence and also by coolant flow distribution in the lower plenum. – It is dependent on the shape of the lower plenum. – Thermal stratification takes place when the mixing of cold and hot sodium is not sufficient and temperature gradient is governed by thermal diffusion. In the lower plenum, the phenomena are of no significance without differences in temperatures of the incoming sodium flows.	b11
			Heat release from half-ellipse-shaped plate at lower end of reactor vessel	– Heat transfer from the lower-end half-ellipse-shaped plate of reactor vessel to outside. – The heat transfers by radiation and convection.	b12
		Upper plenum	Pressure loss	– Pressure loss in the upper plenum (flow space of sodium from core outlet to coolant level) due to flow contraction, flow expansion, and wall friction.	b13
			Heat capacity	– Heat capacity of the coolant in the upper plenum and the upper core structure materials. – Heat capacity is determined by the specific heat and mass of the coolant and structure material.	b14
			Heat transfer between shielding plug and sodium	– Heat transfer between the shielding plug and sodium.	b15
			Coolant mixing, thermal stratification, and thermal striping	– Coolant temperature and flow velocity in the core outlet are distributed in a radial direction. The mixing effect of the coolant is caused by the distribution. – The mixing effect of high- and low-temperature coolant is dependent on the flow velocity and geometry of the upper plenum.	b16

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component			Phenomenon	Description	Code
Reactor System	Reactor internal structures	Vertical shroud	Radial heat transfer between inside and outside coolant through vertical shroud	– Heat transfer between the coolant inside the vertical shroud and the coolant outside. Heat transfers through the vertical shroud.	b17
		Radial shield	Flow in radial shield region	– Natural convection depends on the temperature gradient in the radial shield and the geometry of the radial shield and supporting structures.	b18
			Heat capacity	– Heat capacity of the coolant, structure materials, and shielding in the radial shield region. – Heat capacity is determined by the specific heat and mass of the coolant and structure materials.	b19
			Effect of generated heat by neutron capture and gamma rays	– Heat generation in structure materials due to neutron capture and gamma rays.	b20
			Radial heat transfer in radial shield region	– Radial heat transfer from the inner high-temperature region to the outer low-temperature region.	b21
	Reactivity control drive mechanism		Shutdown velocity of reflector	– When the reactor shuts down, the reflector drops by gravity by over a specified stroke (1m) within the specified time (8s), which are the design requirements.	b22
			Shutdown velocity of shutdown rod	– When the reactor shuts down, the shutdown rod drops by gravity over a specified stroke (2.5m) within the specified time (8s), which are the design requirements.	b23

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Primary Heat Transport System	General	Natural circulation	– Natural circulation is driven by the natural circulation head and the pressure losses along the natural circulation path. – The heat source for natural circulation head is mainly the core, and the heat sink is IHX and RVACS. The natural circulation driving head is evaluated by the average temperature difference between the heat source and heat sink. – Pressure losses of the natural circulation path are a sum of those by flow contraction, flow expansion, and wall friction of the path, all taking into account the local buoyancy-driven phenomena including the flow recirculation inside each subsystem or component.	c01
		Sodium inventory	– Total amount of sodium in the primary heat transport system.	c02
		Heat capacity of coolant	– Heat capacity of coolant in the primary heat transport system. – Heat capacity can be calculated from the specific heat and mass of coolant.	c03

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Primary Heat Transport System	IHX	Pressure loss	– Pressure loss along the flow path of IHX primary coolant. – Pressure loss of primary coolant of IHX is mainly due to the losses caused by the change in inlet and outlet geometry of the heat transfer tubes and the wall friction losses through the tube bundle. – Under the rated flow conditions, pressure loss through the IHX is about $1x10^3$ Pa, which represents about 2% of the total pressure loss along the flow path in the primary system.	c04
		Heat transfer from primary coolant to intermediate coolant	– Heat is transferred from primary coolant through IHX heat transfer tubes to intermediate coolant that flows in the opposite direction to the primary coolant flow.	c05
		Primary flow rate	– Flow rate of the primary coolant is about 152 kg/s at rated operation. – In natural circulation (at low flow rate), flow is governed by pressure losses along the circulation flow path and natural circulation driving head in the primary heat transport system.	c06
		Intermediate flow rate	– Flow rate of the intermediate coolant is about 134 kg/s at rated operation. – In natural circulation (at low flow rate), flow is governed by pressure losses along the circulation flow path and natural circulation driving head in the intermediate heat transport system.	c07

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Primary Heat Transport System	IHX	Heat capacity	– Heat capacity of structure materials such as the heat transfer tube, shielding, and annular flow path of the IHX. – Heat capacity can be calculated from the specific heat and mass of the structure materials.	c08
		Spatial distribution effect of intermediate flow path in IHX annulus shape	– Intermediate coolant flow path in the IHX has one inlet and one outlet, and its structure forms an annulus flow channel. – A baffle plate is placed to equalize the flow in the flow paths. The coolant flow distribution changes depending on its geometry and flow conditions.	c09
	Primary EMP	Flow coastdown performance	– After a primary EMP trip, the primary flow rate decreases following the flow coastdown design curve at a rate of the flow coastdown halving time of 30s.	c10
		Pressure loss	– Pressure loss of the coolant flow in the primary EMP is contributed by the losses from flow contraction and expansion, depending on the inlet and outlet geometry of the EMP and by the friction loss. – Pressure loss of the EMP at rated flow is about 0.008 MPa, which represents about 15% of the total losses in the primary heat transport system.	c11
		Pump head	– Rated pump head of the two EMPs arranged in series is designed at about 0.06 MPa.	c12
		Heat capacity and joule heat at flow coastdown	– Heat capacity for the iron core, coil, and structure materials of the EMP. Heat capacity can be calculated by the specific heat and mass of the structure materials. – EMP generates heat by the Joule effect during flow coastdown and releases it into the coolant.	c13

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Intermediate Heat Transport System	General	Pressure loss	– Pressure loss along the intermediate heat transport system is influenced by flow contraction or expansion dependent on flow path geometry, and also by friction loss. – Intermediate heat transport system consists of IHX (intermediate coolant side), AC heat transfer tubes, intermediate coolant flow path in SG, intermediate EMP, and piping. The total pressure loss at the rated load is about 2.7×10^5 Pa.	d01
		Natural circulation	– Natural circulation is driven by the natural circulation head and pressure losses along the natural circulation path. – Heat source and heat sink that govern natural circulation driving head are the IHX and SG, respectively. The AC works also as a heat sink during normal operation. Natural circulation driving head is due to the average temperature difference between the heat source and heat sink. – Pressure loss along the natural circulation path is influenced by flow contraction and expansion along the path, and by wall friction. – Flow rate of the intermediate heat transport system is governed by the natural circulation driving head and pressure loss.	d02
		Heat removal from SG	– Heat in the intermediate heat transport system transfers to the water/steam system through SG at the rated operation.	d03

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Intermediate Heat Transport System	General	Heat transfer between upper plenum and intermediate coolant external to IHX	– Heat from the primary coolant in the upper plenum transfers to the intermediate coolant of the IHX outlet annulus flow path though the wall of the annular flow path of the IHX.	d04
	Intermediate EMP	Flow coastdown performance	– After an intermediate EMP trip, the flow decreases following the flow coastdown design curve at a rate of the flow coastdown halving time of 30s.	d05
		Pressure loss	– Pressure loss of the coolant flow in the intermediate EMP is influenced by losses from flow contraction and expansion depending on the inlet and outlet geometry of the EMP, and by friction loss. – Pressure loss of the EMP at the rated flow is about $2x10^4$ Pa, which represents about 7% of the total losses in the intermediate heat transport system.	d06
		Pump head	– Rated pump head of the intermediate EMP is designed at 0.27 MPa.	d07
	Steam generator system	Heat capacity of structure, sodium, water, and steam	– Heat capacity for structure materials, intermediate sodium coolant, and water/steam in SG. – Heat capacity can be calculated from the specific heat and mass of the intermediate coolant, water/steam, and structure materials.	d08

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Residual Heat Removal System	Air cooler of IRACS	Pressure loss of sodium side	– Pressure loss along the sodium flow path in the AC is a sum of the losses from flow contraction and expansion depending on the geometry of collecting duct and U-tube type heat transfer tubes and friction losses.	e01
		Pressure loss of air side	– Pressure loss along the AC air flow path is a sum of the losses from flow contraction and expansion depending on the inlet and outlet geometry of the AC and losses through the tube bundle section.	e02
		Heat transfer between tube and air	– Heat transfers from the heat transfer tubes to the air outside the tube. – Heat transfer on the tube wall surface to the air is dependent on the air flow velocity and geometry of heat transfer tube array configuration. – About 0.7 MW of heat is removed in the AC.	e03
		Heat transfer between tube and sodium	– Heat transfers from the intermediate sodium coolant flowing inside the tube to the heat transfer tubes. – Heat transfer between intermediate sodium coolant and heat transfer tube is dependent on the sodium flow rate and geometry of heat transfer tube. – About 0.7 MW of heat is removed in the AC.	e04
		Inlet air temperature range	– Average of the highest temperatures ever recorded in all the states in the U.S. is 46.1°C [7-1] and average of the lowest is -40.0°C [7-2]. The air inlet temperature of the AC is designed as the temperature 50°C rounded up to the highest average.	e05

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Residual Heat Removal System	Air cooler of IRACS	Heat capacity	– Heat capacity of heat transfer tubes and structure materials of the AC. – Heat capacity can be calculated from the specific heat and mass of the structure materials.	e06
	RVACS	Pressure loss in air flow path	– Pressure loss along the RVACS air flow path is influenced by losses from flow contraction or expansion depending on the geometry of the air inlet, annular flow path folded part of the collector, the air outlet, and friction losses.	e07
		Heat transfer between GV wall and air	– Heat of the GV wall transfers to the air flowing in the RVACS air flow path mainly by convection heat transfer. – Heat transfer from GV wall to the air depends on air flow velocity and geometry of the RVACS air flow path.	e08
		Heat transfer between collector wall and air	– Heat of collector wall transfers to the air in the RVACS flow path mainly by convection heat transfer. – Heat transfer from heat collector wall to the air depends on airflow velocity and geometry of the RVACS airflow path.	e09
		Heat transfer between concrete wall and air	– Heat of concrete wall around the RV transfers to the air in the airflow path of RVACS mainly by convection heat transfer. – Heat transfer from concrete wall to the air depends on airflow velocity and geometry of the RVACS airflow path.	e10

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Residual Heat Removal System	RVACS	Thermal radiation between RV wall and GV wall	– Thermal radiation between the wall surfaces of the RV and GV is dependent on temperature, geometry, and emissivity of the wall surface of the RV and GV.	e11
		Thermal radiation between GV wall and heat collector wall	– Thermal radiation between the wall surfaces of the GV and heat collector is dependent on temperature, geometry, and emissivity of the wall surface of the GV and heat collector.	e12
		Thermal radiation between heat collector wall and concrete wall	– Thermal radiation of the heat collector wall and the concrete wall around the RV is dependent on the temperature, geometry, and emissivity of the wall surface of the heat collector wall and the concrete wall.	e13
		Asymmetric airflow	– The RVACS airflow path assumes an annular form surrounded by the GV and concrete structural objects and has the potential to generate an asymmetric flow around, and hence an asymmetric temperature distribution in, the reactor.	e14
		Inlet air temperature range	– Air temperature conditions on the standard site are -40°C~+46.1°C; air inlet temperature of RVACS is designed as 50°C. – Average of the highest temperatures ever recorded in all the states in the U.S. is 46.1°C and average of the lowest is -40.0°C. The air inlet temperature of the RVACS is designed as the temperature 50°C rounded up to the highest average.	e15

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component			Phenomenon	Description	Code
Instrumentation and Control System	I&C equipment	Plant protection sensors	Delay of scram signal of primary EMP voltage and current	– The voltage/current signal of the primary EMP is a scram signal. The delay is the response time from when the process signal (voltage/current value) exceeds the setpoint until the trip breaker opens. – This delay is designed to be 1s or less.	f01
			Delay of scram signal of power line voltage	– The low-voltage signal of the power line is a scram signal. The delay is the response time from when the process signal (bus voltage) exceeds the setpoint until the trip breaker opens. – This delay is designed to be 1s or less.	f02
			Delay of scram signal of neutron flux	– The neutron flux signal is a scram signal. The delay is the response time from when the process signal (neutron flux) exceeds the setpoint until the trip breaker opens. – This delay is designed to be 0.5s or less.	f03
			Delay of scram signal of IHX primary outlet temperature	– The temperature signal of the IHX primary sodium outlet is a scram signal. The delay is the response time from when the process signal (sodium temperature) exceeds the setpoint until the trip breaker opens and is dead time, such as for relay operation delay, and response delay of the thermometer well. – The dead time such as relay operation delay is designed to be 1s or less. Primary delay time of the response delay for the thermometer well is designed to be 30s or less.	f04

Table 7.1-1. Description for Plausible Phenomena (cont.)

Subsystem/Component		Phenomenon	Description	Code
Instrumentation and Control System	Others	Delay of interlock signal of SG outlet temperature	– The interlock signal is transmitted when intermediate coolant temperature at SG outlet is lower than the specified set value. The feedwater pumps and intermediate pump are tripped by this signal. The delay of this signal is the response time from when the process signal (sodium temperature) exceeds the setpoint until the trip breaker opens and is dead time, such as relay operation delay, and the response delay of the thermometer well. – The dead time such as relay operation delay is designed to be 1s or less. Primary delay time of the response delay for the thermometer well is designed to be 30s or less.	f05

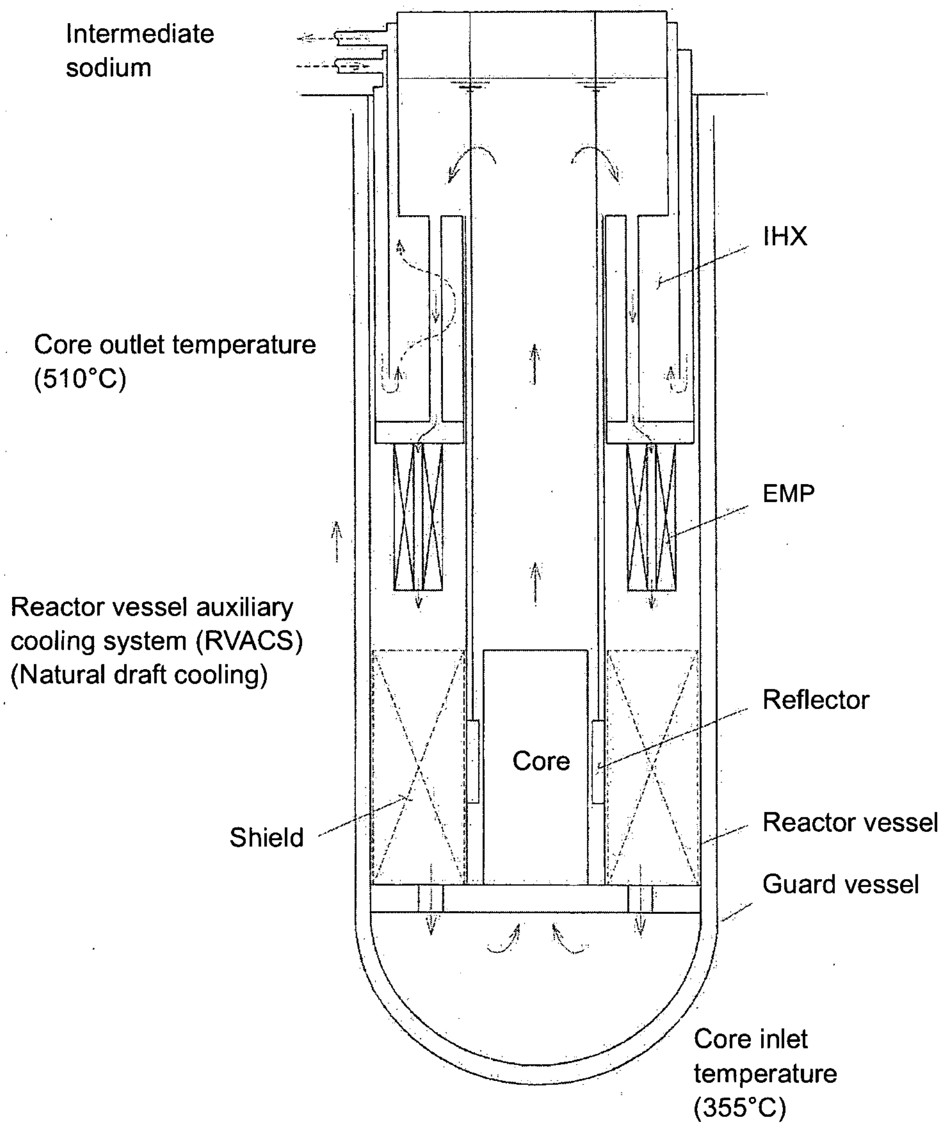


Figure 7.1-1. Coolant Flow in the Reactor

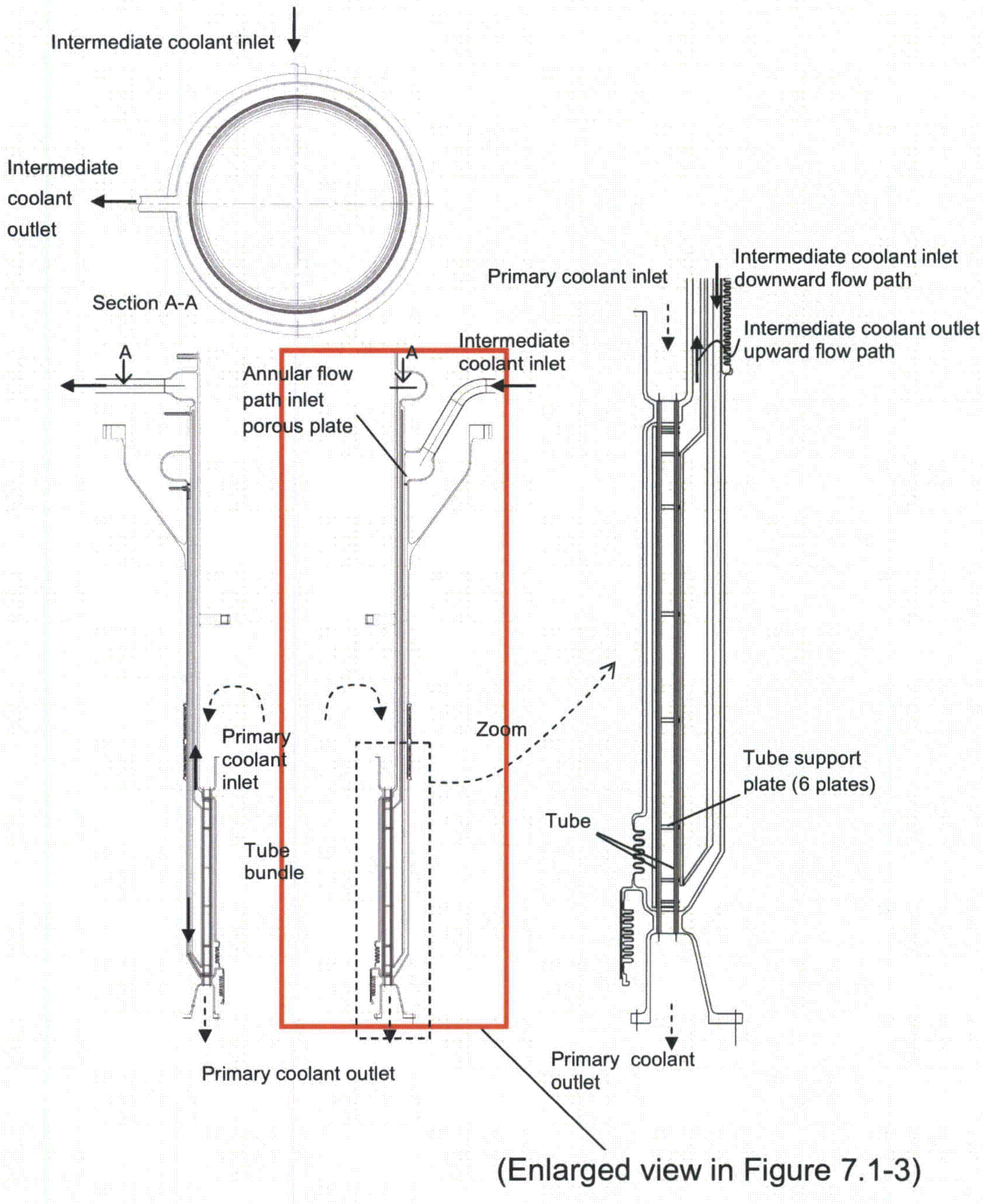


Figure 7.1-2. Coolant Flow in the Internal Structures of Upper Part of Reactor

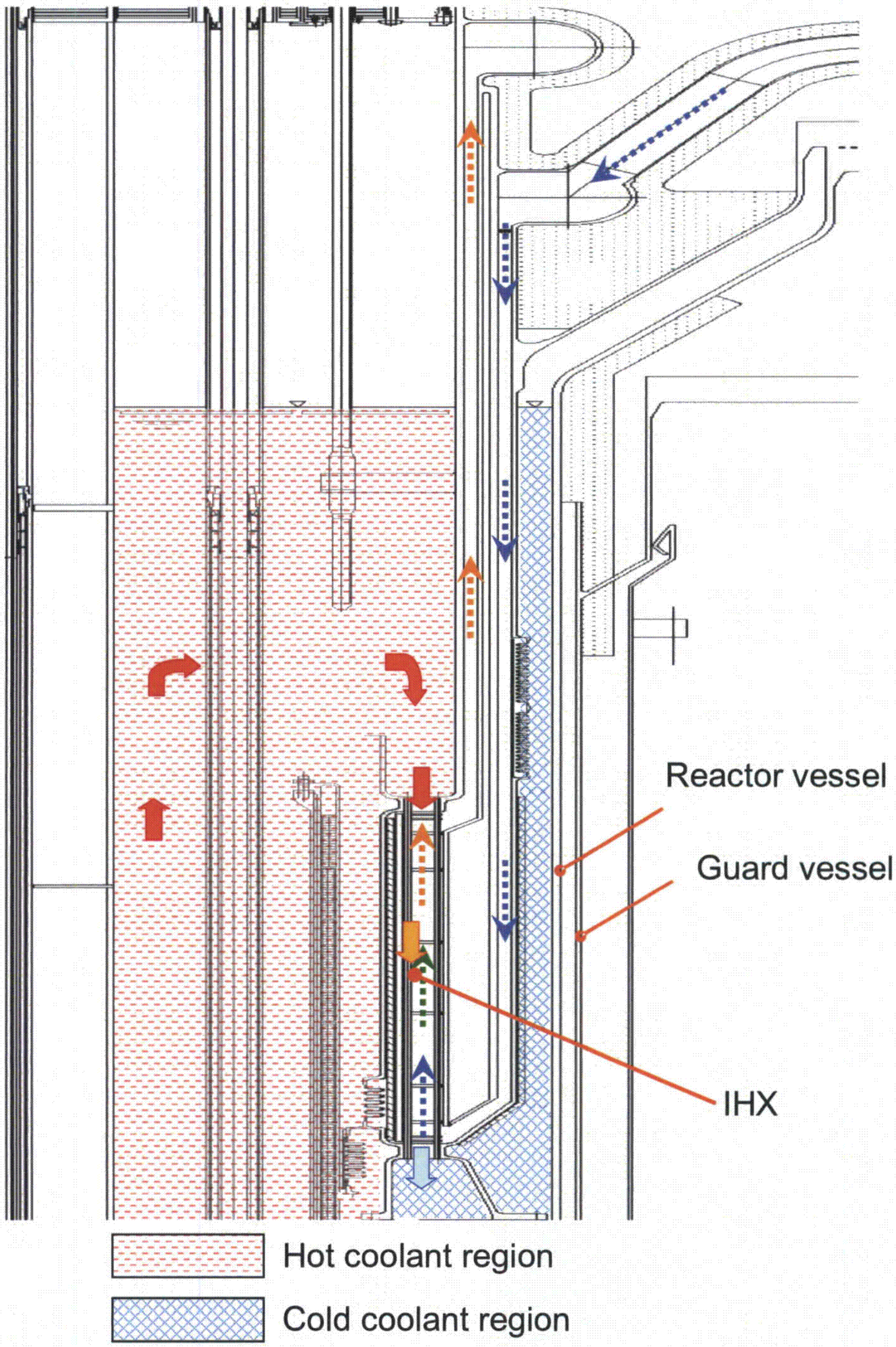


Figure 7.1-3. Enlarged View of Coolant Flow Around the IHX and Internal Structures of Upper Reactor

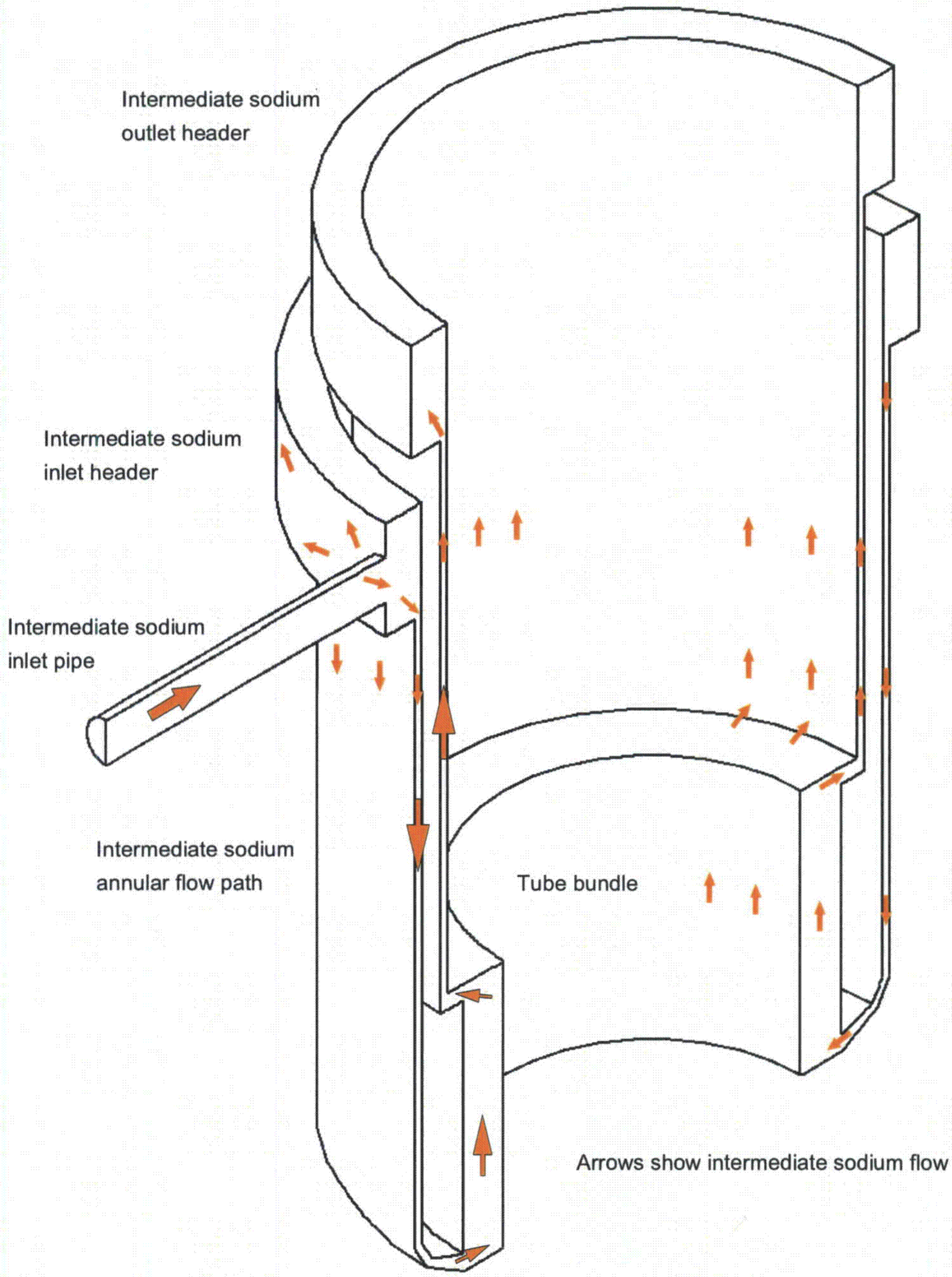


Figure 7.1-4. Intermediate Coolant Flow in IHX

7.2 SUFFICIENCY OF SELECTED PHENOMENA: EVENT AND PHENOMENA MATRIX

The objectives of the 4S PIRT project are as follows:

- Classify the phenomena expected in the 4S by the level of importance and SoK.
- For the categories above, set the priority for further investigation to be implemented to expand the SoK.
- Based on the priority determined above, clarify the content of the test or analyses to be implemented in the near future.

This PIRT focuses on identifying the relative importance and associated SoK of phenomena related to the performance of safety-related structures, systems, and components (SSCs) in the 4S. In this 4S PIRT, the important safety systems are the RPS. To investigate the sufficiency of the selected phenomena, plausible phenomena are identified using the Phenomena Identification Tables for the events whose sequences are different from the LOSP, SLIP, and FCC events. The phenomena are compared with the phenomena identified for these three events for 4S. It is confirmed that the identified phenomena in the three events are sufficient as the phenomena that indicate the validity for the design of the RPS and RHRS. The detailed investigation for additional events arising from ATWS and BDBA is described in Appendix A. For these events, additional PIRT studies are now being developed.

References

[7-1] U.S. National Climatic Data Center, Record Highest Temperatures by State: http://www.ncdc.noaa.gov/oa/pub/data/special/maxtemps.pdf

[7-2] U.S. National Climatic Data Center, Record Lowest Temperatures by State: http://www.ncdc.noaa.gov/oa/pub/data/special/mintemps.pdf

8 PIRT RANKING RESULTS

The process of establishing the ranking for relative importance of the phenomena seen in subsystems, and for their current SoK, is the primary activity of the PIRT process [2-2].

Initially, for the performance of the 4S PIRT, rankings for relative importance of phenomena and current SoK were established by the PIRT team, consisting of engineers from TOSHIBA and CRIEPI. These engineers are assigned to the design section and the research and development section related to the 4S; some of them had previously been engaged in design and R&D for other fast breeder reactors (FBRs) in Japan, namely Joyo and Monju.

Subsequently, industry specialists making up the IRAP met with the Toshiba and CRIEPI engineers to revise the contents of the PIRT rankings, and reached consensus for the final ranking results for each phenomenon. Here, the role of the Japanese and the U.S. members of the IRAP is as described in Chapter 1.

8.1 RANKING SCALE

It is important to identify the criteria for the importance and SoK rankings to enhance the objectivity of the PIRT ranking. This section describes the criteria used in establishing the ranking for both the importance and the current SoK of the evaluated phenomena.

8.1.1 Ranking Scale of Relative Importance of Phenomena

In this PIRT, ranking is established by using the following scale to classify the relative importance of phenomena into four levels. These classifications are based on how much each phenomenon affects the FoM.

High (H):	Phenomenon has a large effect on FoM.
Medium (M):	Phenomenon has a medium effect on FoM.
Low (L):	Phenomenon has a small effect on FoM.
Not applicable (N/A):	Phenomenon has little or nothing to do with FoM.

Many of the phenomena selected in Chapter 7 are related to the physical model of the safety analysis code. Therefore, relative importance of those phenomena can be investigated quantitatively by sensitivity analysis using the parameter in the analysis code that is made to correspond with the physical model. Therefore, in this PIRT, sensitivity analysis is used as one of the methods of confirming the ranking results for relative importance of the phenomena.

If the safety analysis code does not model a particular phenomenon, the phenomenon will not be subject to sensitivity analysis. In such cases, final judgments for the ranking of the phenomena were made by TOSHIBA, CRIEPI, and the Japanese members of the IRAP.

8.1.2 Ranking Scale of State of Knowledge of Phenomena

In this PIRT, the ranking is established by using the following scale to classify the current SoK according to three levels.

Known (K):	Phenomenon is well-known. Model of the test data and analysis code contains little uncertainty.
Partially known (P):	Phenomenon is partially known. Model of the test data and analysis code contains moderate uncertainty.
Unknown (U):	There is little knowledge regarding the phenomenon. Model of test data and analysis code contains large uncertainty.

Ranking for the SoK of the phenomenon in the 4S PIRT is based on high-reliability testing, surveys of papers involving analysis and theory, and the expert knowledge of specialists. However, the final ranking for SoK is established by TOSHIBA, CRIEPI, and the Japanese members of the IRAP in the same way as the ranking for relative importance. In other words, the SoK of the phenomena for which the specialists have insufficient knowledge is judged as "Unknown" and conversely, the SoK of the phenomena with few elements of uncertainty that specialists have sufficient knowledge is judged as "Known."

8.2 INITIAL RANKING RESULTS

As described at the beginning of this chapter, the initial ranking of the PIRT phenomena was established by the TOSHIBA and CRIEPI engineers based on their available knowledge and experience.

Tables 8.2-1, 8.2-2, and 8.2-3 show the initial ranking results. The tables show the results for LOSP, SLIP, and FCC, respectively. However, phenomena regarding IHTS in SLIP and RHRS in FCC are not dealt with, because those subsystems have nothing to do with each target event. In the following section, these results are revised by the Toshiba and CRIEPI staff based on discussions with the IRAP. Final results are shown in Section 8.4.

Table 8.2-1. Initial PIRT Ranking Results (LOSP)

No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Code
	Event: Loss of Offsite Power (LOSP)					
	Figures of Merit (FoM): Cladding Temperature		Importance			
	A	Core and Fuel Assemblies				
1	-	Pressure loss in core region	L	L	P	a01
2	-	Pressure loss in reflector region	L	L	P	a02
3	-	Natural convection	L	M	P	a03
4	-	Reactivity feedback	L	N/A	P	a04
5	-	Gap conductance between fuel and cladding	L	L	K	a05
6	-	Heat transfer between cladding and coolant	L	L	K	a06
7	-	Intra- and inter-assembly flow distribution	H	M	P	a07
8	-	Radial heat transfer between subassemblies (S/A<-->sodium<-->S/A)	M	M	P	a08
9	-	Heat transfer between reflector and coolant	L	L	P	a09
10	-	Heat capacity of core assemblies	M	M	K	a10
11	-	Coolant boiling	N/A	N/A	K	a11
12	-	Core power	M	M	K	a12
13	-	Decay heat	M	M	K	a13
14	-	Heat transfer between core support plate and sodium	L	L	P	a14
15	-	Rate of scram reactivity insertion	H	L	K	a15
16	-	Delay of scram reactivity insertion	H	L	K	a16
17	-	Eutectic reaction between fuel and cladding	L	L	P	a17
18	-	Temperature dependence of physical properties of materials	M	M	K	a18
19	-	FP release from fuel slug into gas plenum	N/A	N/A	K	a19
20	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	P	a20
21	-	FP transport from sodium in primary system to cover gas	N/A	N/A	P	a21
22	-	Flow-induced vibration in a subassembly	L	L	P	a22
23	-	Inter-wrapper flow between wrapper tubes	L	L	P	a23
24	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	a24
25	-	Radial power distribution	L	L	K	a25
26	-	Axial power distribution	L	L	K	a26
	B	Reactor System				
	B1	Reactor Vessel				
27	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	K	b01
	B2	Reactor Internal Structures				
	B20	General				
28	-	Coolant mixing effect in upper plenum including thermal stratification	M	M	P	b02
29	-	Temperature dependence of physical properties of structural materials	L	L	K	b03
30	-	Natural convection	L	M	P	b04
31	-	Flow-induced vibration	L	L	P	b05

Notes:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-1. Initial PIRT Ranking Results (LOSP) (cont.)

	Event: Loss of Offsite Power (LOSP)						
	Figures of Merit (FoM): Cladding Temperature			Importance			
No.	Subsystem/ Component		Phenomenon	1st Phase	2nd Phase	SoK	Code
	B21	Reflector					
32	-		Deformation due to thermal effect and irradiation	L	L	P	b06
33	-		Flow in reflector region	L	L	P	b07
34	-		Effect of generated heat by neutron capture and gamma rays	L	L	P	b08
	B22	Lower Plenum					
35	-		Pressure loss	L	L	P	b09
36	-		Heat capacity	L	L	K	b10
37	-		Coolant mixing and thermal stratification	L	L	P	b11
38	-		Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	L	P	b12
	B23	Upper Plenum					
39	-		Pressure loss	L	L	P	b13
40	-		Heat capacity	L	M	K	b14
41	-		Heat transfer between shielding plug and sodium	L	L	K	b15
42	-		Coolant mixing, thermal stratification, and thermal striping	L	L	P	b16
	B24	Vertical Shroud					
43	-		Radial heat transfer between inside and outside coolant through vertical shroud	L	L	P	b17
	B25	Radial shield					
44	-		Flow in radial shield region	L	L	P	b18
45	-		Heat capacity	L	L	K	b19
46	-		Effect of generated heat by neutron capture and gamma rays	L	L	P	b20
47	-		Radial heat transfer in radial shield region	L	L	P	b21
	B26	Reactivity Control Drive Mechanism					
48	-		Shutdown velocity of reflector	M	L	K	b22
49	-		Shutdown velocity of shutdown rod	M	L	K	b23
	C	Primary Heat Transport System					
	C0	General					
50	-		Natural circulation	L	M	P	c01
51	-		Sodium inventory	L	M	K	c02
52	-		Heat capacity of coolant	L	L	K	c03
	C1	IHX					
53	-		Pressure loss	L	L	P	c04
54	-		Heat transfer from primary coolant to intermediate coolant	L	L	P	c05
55	-		Primary flow rate	L	L	P	c06
56	-		Intermediate flow rate	L	L	P	c07
57	-		Heat capacity	L	L	K	c08
58	-		Spatial distribution effect of intermediate flow path in IHX annulus shape	L	L	P	c09

Notes:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-1. Initial PIRT Ranking Results (LOSP) (cont.)

No.	Subsystem/ Component		Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Code
	Event: Loss of Offsite Power (LOSP)						
	Figures of Merit (FoM): Cladding Temperature						
	C2	Primary EMP					
59	-		Flow coastdown performance	M	L	P	c10
60	-		Pressure loss	L	L	P	c11
61	-		Pump head	L	L	K	c12
62	-		Heat capacity and joule heat at flow coastdown	L	L	P	c13
	D	Intermediate Heat Transport System					
	D0	General					
63	-		Pressure loss	L	L	P	d01
64	-		Natural circulation	L	L	P	d02
65	-		Heat removal from SG	L	N/A	K	d03
66	-		Heat transfer between upper plenum and intermediate coolant external to IHX	L	L	P	d04
	D1	Intermediate EMP					
67	-		Flow coastdown performance	M	L	K	d05
68	-		Pressure loss	L	L	P	d06
69	-		Pump head	L	L	K	d07
	D2	Steam Generator System					
70	-		Heat capacity of structure, sodium, water, and steam	L	L	K	d08
	E	Residual Heat Removal System					
	E1	Air cooler of IRACS					
71	-		Pressure loss of sodium side	L	L	P	e01
72	-		Pressure loss of air side	L	M	P	e02
73	-		Heat transfer between tube and air	L	M	P	e03
74	-		Heat transfer between tube and sodium	L	L	P	e04
75	-		Inlet air temperature range	L	M	K	e05
76	-		Heat capacity	L	L	K	e06
	E2	RVACS					
77	-		Pressure loss in airflow path	L	L	K	e07
78	-		Heat transfer between GV wall and air	L	L	K	e08
79	-		Heat transfer between collector wall and air	L	L	K	e09
80	-		Heat transfer between concrete wall and air	L	L	K	e10
81	-		Thermal radiation between RV wall and GV wall	L	L	P	e11
82	-		Thermal radiation between GV wall and heat collector wall	L	L	P	e12
83	-		Thermal radiation between heat collector wall and concrete wall	L	L	P	e13
84	-		Asymmetric airflow	L	L	P	e14
85	-		Inlet air temperature range	L	L	K	e15

Notes:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-1. Initial PIRT Ranking Results (LOSP) (cont.)

	Event: Loss of Offsite Power (LOSP)						
	Figures of Merit (FoM): Cladding Temperature			Importance			
No.	Subsystem/ Component		Phenomenon	1st Phase	2nd Phase	SoK	Code
	F	Instrumentation and Control System					
	F1	Instrumentation and Control Equipment					
	F11	Plant Protection Sensors					
86		-	Delay of scram signal of primary EMP voltage and current	M	L	K	f01
87		-	Delay of scram signal of power line voltage	M	L	K	f02
88		-	Delay of scram signal of neutron flux	L	L	K	f03
89		-	Delay of scram signal of IHX primary outlet temperature	L	L	K	f04
	F12	Others					
90		-	Delay of interlock signal of SG outlet temperature	L	L	K	f05

Notes:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-2. Initial PIRT Ranking Results (SLIP)

	Event: Sodium Leakage from Intermediate Piping (SLIP)					
	Figures of Merit (FoM): Cladding Temperature		Importance			
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Code
	A	Core and Fuel Assemblies				
1	-	Pressure loss in core region	L	L	P	a01
2	-	Pressure loss in reflector region	L	L	P	a02
3	-	Natural convection	L	M	P	a03
4	-	Reactivity feedback	L	N/A	P	a04
5	-	Gap conductance between fuel and cladding	L	L	K	a05
6	-	Heat transfer between cladding and coolant	L	L	K	a06
7	-	Intra- and inter-assembly flow distribution	H	M	P	a07
8	-	Radial heat transfer between subassemblies (S/A<-->sodium<-->S/A)	M	M	P	a08
9	-	Heat transfer between reflector and coolant	L	L	P	a09
10	-	Heat capacity of core assemblies	M	M	K	a10
11	-	Coolant boiling	N/A	N/A	K	a11
12	-	Core power	M	M	K	a12
13	-	Decay heat	M	M	K	a13
14	-	Heat transfer between core support plate and sodium	L	L	P	a14
15	-	Rate of scram reactivity insertion	H	L	K	a15
16	-	Delay of scram reactivity insertion	H	L	K	a16
17	-	Eutectic reaction between fuel and cladding	L	L	P	a17
18	-	Temperature dependence of physical properties of materials	M	M	K	a18
19	-	FP release from fuel slug into gas plenum	N/A	N/A	K	a19
20	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	P	a20
21	-	FP transport from sodium in primary system to cover gas	N/A	N/A	P	a21
22	-	Flow-induced vibration in a subassembly	L	L	P	a22
23	-	Inter-wrapper flow between wrapper tubes	L	L	P	a23
24	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	a24
25	-	Radial power distribution	L	L	K	a25
26	-	Axial power distribution	L	L	K	a26
	B	Reactor System				
	B1	Reactor Vessel				
27	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	K	b01
	B2	Reactor Internal Structures				
	B20	General				
28	-	Coolant mixing effect in upper plenum including thermal stratification	M	M	P	b02
29	-	Temperature dependence of physical properties of structural materials	L	L	K	b03
30	-	Natural convection	L	M	P	b04
31	-	Flow-induced vibration	L	L	P	b05

Notes:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-2. Initial PIRT Ranking Results (SLIP) (cont.)

No.	Subsystem/ Component		Phenomenon	Importance 1st phase	Importance 2nd phase	SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)						
	Figures of Merit (FoM): Cladding Temperature						
	B21	Reflector					
32		-	Deformation due to thermal effect and irradiation	L	L	P	b06
33		-	Flow in reflector region	L	L	P	b07
34		-	Effect of generated heat by neutron capture and gamma rays	L	L	P	b08
	B22	Lower Plenum					
35		-	Pressure loss	L	L	P	b09
36		-	Heat capacity	L	L	K	b10
37		-	Coolant mixing and thermal stratification	L	L	P	b11
38		-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	L	P	b12
	B23	Upper Plenum					
39		-	Pressure loss	L	L	P	b13
40		-	Heat capacity	L	M	K	b14
41		-	Heat transfer between shielding plug and sodium	L	L	K	b15
42		-	Coolant mixing, thermal stratification, and thermal striping	L	L	P	b16
	B24	Vertical Shroud					
43		-	Radial heat transfer between inside and outside coolant through vertical shroud	L	L	P	b17
	B25	Radial shield					
44		-	Flow in radial shield region	L	L	P	b18
45		-	Heat capacity	L	L	K	b19
46		-	Effect of generated heat by neutron capture and gamma rays	L	L	P	b20
47		-	Radial heat transfer in radial shield region	L	L	P	b21
	B26	Reactivity Control Drive Mechanism					
48		-	Shutdown velocity of reflector	M	L	K	b22
49		-	Shutdown velocity of shutdown rod	M	L	K	b23
	C	Primary Heat Transport System					
	C0	General					
50		-	Natural circulation	L	M	P	c01
51		-	Sodium inventory	L	M	K	c02
52		-	Heat capacity of coolant	L	L	K	c03
	C1	IHX					
53		-	Pressure loss	L	L	P	c04
54		-	Heat transfer from primary coolant to intermediate coolant	L	L	P	c05
55		-	Primary flow rate	L	L	P	c06
56		-	Intermediate flow rate	L	L	P	c07
57		-	Heat capacity	L	L	K	c08
58		-	Spatial distribution effect of intermediate flow path in IHX annulus shape	L	L	P	c09

Notes:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-2. Initial PIRT Ranking Results (SLIP) (cont.)

No.	Subsystem/ Component		Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)						
	Figures of Merit (FoM): Cladding Temperature						
	C2	Primary EMP					
59	-		Flow coastdown performance	M	L	P	c10
60	-		Pressure loss	L	L	P	c11
61	-		Pump head	L	L	K	c12
62	-		Heat capacity and joule heat at flow coastdown	L	L	P	c13
	E	Residual Heat Removal System					
	E2	RVACS					
63	-		Pressure loss in airflow path	L	M	K	e07
64	-		Heat transfer between GV wall and air	L	L	K	e08
65	-		Heat transfer between collector wall and air	L	L	K	e09
66	-		Heat transfer between concrete wall and air	L	L	K	e10
67	-		Thermal radiation between RV wall and GV wall	L	M	P	e11
68	-		Thermal radiation between GV wall and heat collector wall	L	M	P	e12
69	-		Thermal radiation between heat collector wall and concrete wall	L	M	P	e13
70	-		Asymmetric airflow	L	M	P	e14
71	-		Inlet air temperature range	L	M	K	e15
	F	Instrumentation and Control System					
	F1	Instrumentation and Control Equipment					
	F11	Plant Protection Sensors					
72	-		Delay of scram signal of primary EMP voltage and current	M	L	K	f01
73	-		Delay of scram signal of power line voltage	M	L	K	f02
74	-		Delay of scram signal of neutron flux	L	L	K	f03
75	-		Delay of scram signal of IHX primary outlet temperature	L	L	K	f04
	F12	Others					
76	-		Delay of interlock signal of SG outlet temperature	L	L	K	f05

Notes:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-3. Initial PIRT Ranking Results (FCC)

	Event: Failure of a Cavity Can (FCC)					
	Figures of Merit (FoM): Cladding Temperature					
No.	Subsystem/ Component		Phenomenon	Importance	SoK	Code
	A	Core and Fuel Assemblies				
1	-		Reactivity feedback	L	P	a04
2	-		Gap conductance between fuel and cladding	L	P	a05
3	-		Heat transfer between cladding and coolant	L	K	a06
4	-		Intra- and inter-assembly flow distribution	H	P	a07
5	-		Radial heat transfer between subassemblies (S/A<-->sodium<-->S/A)	L	P	a08
6	-		Heat capacity of core assemblies	M	K	a10
7	-		Coolant boiling	N/A	K	a11
8	-		Core power	M	K	a12
9	-		Decay heat	M	K	a13
10	-		Heat transfer between core support plate and sodium	L	P	a14
11	-		Rate of scram reactivity insertion	H	P	a15
12	-		Delay of scram reactivity insertion	H	P	a16
13	-		Eutectic reaction between fuel and cladding	L	P	a17
14	-		Temperature dependence of physical properties of materials	L	K	a18
15	-		Flow-induced vibration in a subassembly	L	P	a22
16	-		Inter-wrapper flow between wrapper tubes	L	P	a23
17	-		Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	P	a24
18	-		Radial power distribution	L	K	a25
19	-		Axial power distribution	L	K	a26
20	-		Reactivity insertion by cavity can failure	M	P	a27
	B	Reactor System				
	B1	Reactor Vessel				
	B2	Reactor Internal Structures				
	B20	General				
21	-		Coolant mixing effect in upper plenum including thermal stratification	M	P	b02
	B21	Reflector				
22	-		Flow in reflector region	L	P	b07
23	-		Effect of generated heat by neutron capture and gamma rays	L	P	b08
	B22	Lower Plenum				
24	-		Heat capacity	L	K	b10
25	-		Coolant mixing and thermal stratification	L	P	b11
26	-		Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	P	b12
	B23	Upper Plenum				
27	-		Heat capacity	L	K	b14
28	-		Heat transfer between shielding plug and sodium	L	K	b15
29	-		Coolant mixing, thermal stratification, and thermal striping	L	P	b16

Notes:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.2-3. Initial PIRT Ranking Results (FCC) (cont.)

No.	Subsystem/ Component		Phenomenon	Importance	SoK	Code
	Event: Failure of a Cavity Can (FCC)					
	Figures of Merit (FoM): Cladding Temperature					
	B24	Vertical Shroud				
30		-	Radial heat transfer between inside and outside coolant through vertical shroud	L	K	b17
	B25	Radial shield				
31		-	Flow in radial shield region	L	P	b18
32		-	Heat capacity	L	K	b19
33		-	Effect of generated heat by neutron capture and gamma rays	L	P	b20
34		-	Radial heat transfer in radial shield region	L	P	b21
	B26	Reactivity Control Drive Mechanism				
35		-	Shutdown velocity of reflector	H	K	b22
36		-	Shutdown velocity of shutdown rod	H	K	b23
	C	Primary Heat Transport System				
	C0	General				
37		-	Sodium inventory	L	K	c02
38		-	Heat capacity of coolant	M	K	c03
	C1	IHX				
39		-	Heat transfer from primary coolant to intermediate coolant	L	P	c05
40		-	Primary flow rate	L	P	c06
41		-	Intermediate flow rate	L	P	c07
42		-	Heat capacity	L	K	c08
43		-	Spatial distribution effect of intermediate flow path in IHX annulus shape	L	P	c09
	D	Intermediate Heat Transport System				
	D0	General				
44		-	Heat removal from SG	L	P	d03
45		-	Heat transfer between upper plenum and intermediate coolant external to IHX	L	P	d04
	D1	Intermediate EMP				
	E	Residual Heat Removal System				
	F	Instrumentation and Control System				
	F1	Instrumentation and Control Equipment				
	F11	Plant Protection Sensors				
46		-	Delay of scram signal of neutron flux	H	K	f03
47		-	Delay of scram signal of IHX primary outlet temperature	L	K	f04
	F12	Others				
48		-	Delay of interlock signal of SG outlet temperature	L	K	f05

Notes:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

8.3 SENSITIVITY ANALYSIS

As described previously, in the 4S PIRT, sensitivity analysis is used as one method of confirming the ranking results for the relative importance of phenomena. Sensitivity analysis in this report is implemented for every event, even for the same phenomena, because the sensitivity of phenomena may differ for each event.

8.3.1 Procedure for Sensitivity Analysis

The analysis code used in the sensitivity analysis is a one-point approximation reactor kinetics flow/network code, ARGO, which was developed by TOSHIBA, and is also used in the safety analysis [8-1]. Appendix B describes the ARGO code and provides some items needed for validation.

The calculations are conducted as follows.

- Base case

  (B-1) Calculate the transient.

  (B-2) Evaluate the temperature variation of the hottest fuel pin under nominal calculation after the transient calculation.

- Sensitivity analysis

  (S-1) Set value of parameter in the calculation to target value.

  (S-2) Recalculate steady state for the value above.

  (S-3) Reset coolant temperatures at the inlet and the outlet of core by adjusting the core coolant flow rate so that they satisfy design values, which are 355°C at the inlet and 510°C at the outlet, respectively.

  (S-4) Calculate the transient.

  (S-5) Evaluate the temperature variation of the hottest fuel pin under nominal calculation after the transient calculation.

Note that, in case the parameter is related only to the phenomenon that occurs after the initiating event, e.g., the halving time of the flow coastdown system of EMP, the steps (S-2) and (S-3) are not required because they do not affect the calculation result at rated conditions.

8.3.2 Specification of Uncertainty Width of Parameter for Sensitivity Analysis

Sensitivity analysis is conducted by changing the parameter related to the objective phenomena from the value in the base case. At this point, it is important to determine the uncertainty width of the parameter for sensitivity analysis. Based on engineering judgment in the experiment and values defined in the design, a standard deviation of 1σ is set as the criterion for the uncertainty width of the parameters. In the sensitivity analysis, this 1σ, which is the variation of the parameter, is used as the criterion value. In addition, some calculations are conducted with larger variation such as 2~3σ. This may simply indicate that the sensitivity is very small.

In addition, for a parameter in the calculation for which 1σ is difficult to define, such as volume of RV upper plenum described later, that value is changed into an extremely large value.

Scales of standard deviation (σ) for the main parameters are set as follows and are based on design accuracy or engineering judgment translated from previous LMFR design experience.

Pressure loss: 10%
Coefficient of heat transfer: 20%
Core power: 2%
Decay heat: 5%
Flow halving time for EMP coastdown system: 33%
Scram insertion velocity: 10%
Delay time at the scram start: 10%
Heat transfer area: 10%
Reactivity insertion amount caused by cavity can failure: 10%
Reactivity insertion rate caused by cavity can failure: 10%

The following physical values are set for each material:

Fuel density: <5%
Fuel specific heat: <5%
Sodium density: <1%
Sodium specific heat: <1%
Air density: <1%
Air specific heat: <1%
B_4C density: ~10%
B_4C specific heat: ~10%
HT-9 density: <3%
HT-9 specific heat: <10%
SUS304 density: <1%
SUS304 specific heat: 2~3%

In addition, parameters for which it is difficult to define 1σ due to the nature of the parameter are shown below:

- Gap conductance between fuel and cladding

The base case used the conductance value of liquid sodium. In the sensitivity analysis, the worst case assumes that a large amount of FP gas released from the fuel slug accumulates inside the fuel pin, and a representative value of oxide fuel in the fuel pin is set as the worst case.

- Pressure loss in the core under the condition with 20 percent Reynolds number (Re) of rated flow

  In the sensitivity analysis, the worst case assumes the amount of pressure loss as twice that of the base case so that excessive conservatism is applied because knowledge of pressure loss characteristics in the low Re number regimes is insufficient.

- Radial heat transfer between and within subassemblies (intra- and inter-subassembly)

  This effect consists of two terms, "thermal conduction" and "forced convection." In sensitivity analysis, the case without both terms and the case without forced convection term are assumed.

- Volume in upper plenum

  In the base case, volume is set smaller than the design volume. This is based on the knowledge that coolant in the upper part of the upper plenum makes little contribution to circulation in the RV and that coolant flowing out of the core rises to the upper plenum and falls though the vertical shroud. In actual sensitivity analysis, the upper plenum in a calculation system consists of eight units that are equal in volume.

- Elevation of AC

  The position of the AC as a heat sink is one factor that determines the natural circulation force of the intermediate system. To investigate the effect on natural circulation, as the worst case, the value is changed from the position of the base case by ±1m.

- Air temperature at IRACS inlet/air temperature at RVACS inlet

  In the base case, a conservative value is used. In the sensitivity analysis, a realistic value is set. The temperature of the base case (50°C) is a value obtained by rounding up a value (46.1°C) that is the average of the highest temperatures ever recorded in all the states in the U.S. [8-3].

- Emissivity of radiation heat transfer between the RV and GV, and GV and heat collector

  There is little knowledge of the change in emissivity caused by age deterioration, and the uncertainty is large. Therefore, in the sensitivity analysis, the emissivity of the base case is changed in the range between -90 and -10 percent.

- Maximum insertion reactivity/reactivity insertion rate (cavity can failure)

The reactivity insertion is caused by the gas in the cavity can being displaced by the surrounding coolant. The uncertainty is small because the reactivity insertion amount assumes failure of one cavity can. However, because the failure itself assumes several patterns depending on the manufacturing defect, the reactivity insertion rate is significantly affected by the pattern.

Also, sensitivity analysis is not applied to the following phenomena, because ARGO does not have a corresponding model for them. As stated in Section 8.1.1, final judgments for the ranking of such phenomena are made by TOSHIBA, CRIEPI, and the Japanese members of the IRAP.

- Coolant boiling
- Eutectic reaction
- FP release from fuel slug into gas plenum
- FP transport from fuel to sodium bond, and sodium in primary system
- FP transport from sodium in primary system to cover gas
- Temperature fluctuation of reactor vessel by change of liquid level
- Flow-induced vibration
- Deformation due to thermal effect and radiation
- Effect of generated heat by neutron capture and gamma rays
- Spatial distribution effect of intermediate flow path in IHX annulus shape
- Asymmetric air flow

8.3.3 Results of Sensitivity Analysis

This section describes the results of the sensitivity analysis.

As described above, the results of the sensitivity analysis are evaluated by TOSHIBA, CRIEPI, and the Japanese members of the IRAP to refine the relative phenomena importance rankings, but they are not used as an absolute metric.

The investigation methods for sensitivity in the sensitivity analysis are as follows. Figure 8.3.3-1 shows the outline. The meanings of the codes used in the figure are as follows:

QS:	Quantitative Standard
PBC1:	Peak value at the 1st phase of base case
PBC2:	Peak value at the 2nd phase of base case
PSAC1:	Peak value at the 1st phase of sensitivity analysis case
PSAC2:	Peak value at the 2nd phase of sensitivity analysis case

As shown in Figure 8.3.3-1, sensitivity is evaluated for each time phase defined in Chapter 4. As described in the following evaluation formula, for the 4S analyses, sensitivity is defined by the comparison between the "difference between peak temperature of base case and quantitative standard value" and the "difference between peak temperature of sensitivity analysis case and quantitative standard value" in each phase.

$$\text{Sensitivity} = 1 - \frac{\text{FoM}_{\text{QS}} - \text{FoM}_{\text{PSACn}}}{\text{FoM}_{\text{QS}} - \text{FoM}_{\text{PBCn}}} \tag{8.3.3-1}$$

The subscript *n* indicates the number of the time phase.

The quantitative standard described here is not equivalent to the acceptance criterion used in the safety analysis; it is only the guideline to investigate the sensitivity in the sensitivity analysis. It is set to 630°C for the FoM, i.e., cladding temperature, which indicates the highest limit that can ensure the integrity of the cladding if only the temperature below 630°C is kept.

This is based on the study results [1-2] [8-2] by ANL that demonstrate that the integrity of the cladding is expected to be maintained at 630°C.

Subsections 8.3.3.1 to 8.3.3.3 show the main sensitivity analysis results and the summary for each one of the three events selected in Chapter 4.

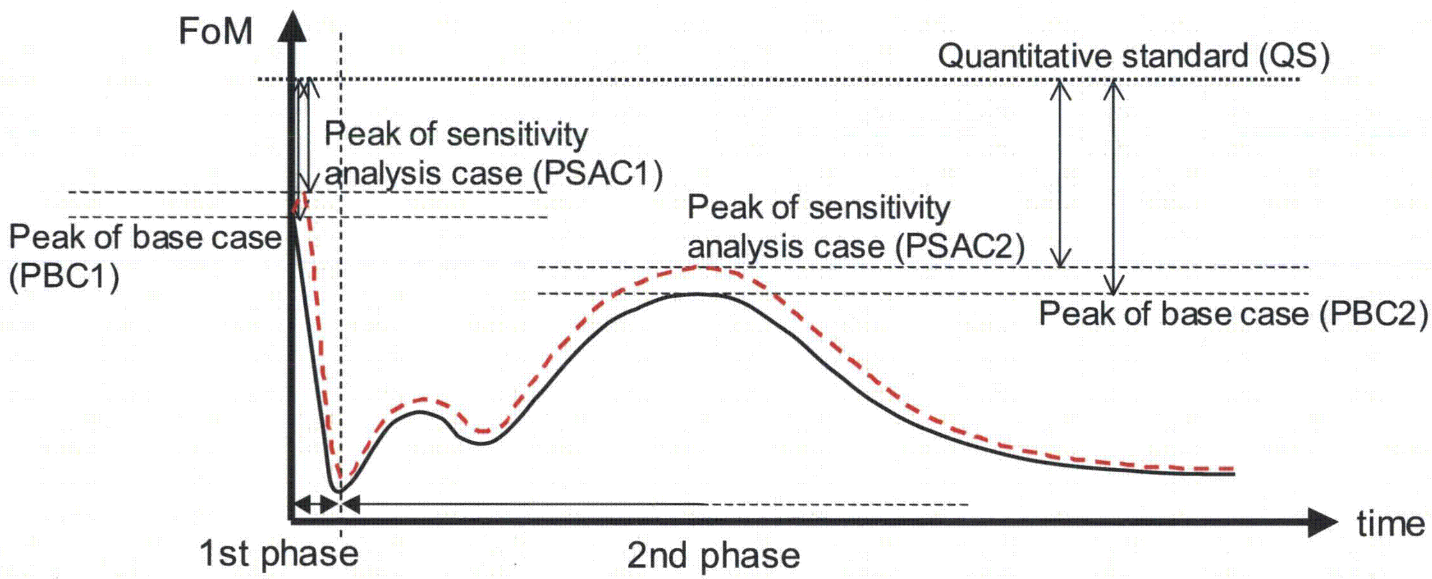


Figure 8.3.3-1. Evaluation Method to Determine Sensitivity of FoM

8.3.3.1 Loss of Offsite Power (LOSP)

Table 8.3.3.1-1 summarizes all the case of the sensitivity analysis conducted for the LOSP. The columns starting from leftmost column are as follows.

Case No.: Case number

Subsystem/Component: Subsystem and component that are related to the parameter in the calculation. These were explained in Chapter 5.

Explanation of Parameter: Explanation of meaning for parameter in the calculation

Set Value (range of variation from base case): Set value of parameter in the calculation

Code: Symbol to identify each phenomenon and the same as code defined in Table 7.1-1

Sensitivity (1st Phase): Sensitivity expressed by equation (8.3.3-1) and phenomena importance ranking defined by the sensitivity

Sensitivity (2nd Phase): Sensitivity expressed by equation (8.3.3-1) and phenomena importance ranking defined by the sensitivity

The results of the base case are as described in Section 4.2.

Figures 8.3.3.1-1 and 8.3.3.1-2 show a summary of the sensitivity analysis 1st phase and 2nd phase, respectively, arranged in order of the sensitivity scale obtained in the sensitivity analysis. A dashed line in these figures indicates the value of 0.04 dividing sensitivities M and L as shown just below. Figure 8.3.3.1-3 shows a concrete example for the time variation of cladding temperature in this event. This is the result of Case nos. 2 and 3 in Table 8.3.3.1-1.

In the 4S PIRT, the ranking for the results of the sensitivity analysis is defined, corresponding to the sensitivity value in the case that amplitude of the parameter is equivalent to 1σ. The sensitivity is evaluated by the amount of the change from the base case, regardless of the sign of the value. However, as stated in the Section 8.3.2, for some parameters such as emissivity of radiation in the RVACS, the value of 1σ is difficult to define. For those parameters, more cases than general parameters are set and the sensitivities are investigated. As stated later in this section, 8.3.3.2 and 8.3.3.3, just 90 percent of all the results of sensitivity analysis show sensitivity less than 0.04 that is calculated from in eq. (8.3.3-1). The, sensitivity 0.04 is set as a boundary between rank "M" and "L" as follows. In addition to that, the value of 0.08, which is twice as large as 0.04, is set as a boudanry between rank "M" and "H".

$0 < \|\text{Sensitivity}\| < 0.04$:	L
$0.04 < \|\text{Sensitivity}\| < 0.08$:	M
$0.08 < \|\text{Sensitivity}\|$:	H

In fact, the sensitivity of 0.04 in the LOSP event corresponds to about 2.4°C of fuel cladding temperature (FoM) in the 1st phase and about 6.1°C in the 2nd phase.

The cases with significant sensitivity, i.e., sensitivity of M or more shown in the parameter amplitude of 1σ, are described below.

(a) Parameter: Pressure loss coefficient in core (Case nos. 1 to 7)

The coefficient of pressure loss is used as a parameter and affects the coolant flow rate of the core. Sensitivity for the case in which the overall pressure loss coefficient in the core is changed by 1σ (Case no.1) corresponds to L and is small. However, the 1st phase of the cases where the pressure loss coefficient is locally changed by 1σ (Case nos. 2 through 7) shows a sensitivity of M or more. Specifically, sensitivity in the case where only the pressure loss coefficient of the inner core (IC) is changed is about 0.1 and corresponds to H. Setting of this latter parameter relates to flow distribution in the core and shows that the uncertainty of flow rate distribution of intra- and inter-assembly coolant has a strong effect on cladding temperature. The core is divided into six junctions in this ARGO calculation (j=1: IC, j=2: MC, j=3: OC, j=4: shutdown rod (SR), j=5: barrel, j=6: reflector) as shown in Figures 8.3.3.1-4 and 8.3.3.1-5. Here, j, shown in Case nos. 1 through 7 in Table 8.3.3.1-1, indicates the core junction number of the calculation.

(b) Parameter: Radial heat transfer between subassemblies (Case nos. 18 and 19)

The radial heat transfer of the core is used as a parameter. Sensitivity in the case, without considering the whole of the radial heat transfer of the core while it is assumed in the base case, is 0.043 and corresponds to M. This result shows that the effect of radial transfer of the heat generated in the core has a relatively large effect on cladding temperature.

(c) Parameter: Heat transfer between sodium and air in IRACS (Case nos. 68 to 73)

Heat transfer between the heat transfer tube wall and air, and between the tube wall and sodium, is used as a parameter. This parameter specifies the amount of heat transfer from the sodium side of IRACS to the air side, which is the heat removal capacity of the air cooler (AC). Sensitivity in the 2nd phase of the case, where the coefficient of heat transfer on the air side is changed by 1σ (Case nos. 68 and 69), corresponds to M. On the other hand, the coefficient of heat transfer sensitivity on the sodium side is almost 0 (Case nos. 70 and 71). This result shows that uncertainty relates to the heat transfer on the air side as to the effect on cladding temperature and there is little uncertainty related to the heat transfer on the sodium side.

(d) Parameter: Air temperature at IRACS inlet (Case nos. 74 to 77)

The air temperature at the IRACS inlet is used as a parameter. This parameter is the environmental factor outside the plant and affects the heat removal capacity of IRACS. However, the exterior environment of the plant is dependent on the actual setting of the site. In the cases that have possible temperatures set for sensitivity analysis, some cases indicate sensitivity equivalent to M. This result shows that the air temperature at the IRACS inlet affects cladding temperature.

Table 8.3.3.1-1. Setup Value of Each Parameter in Sensitivity Analysis (LOSP)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	-	**Base case: ±0%**	-	-	-
-	A: Core/Fuel Assemblies	-	-	-	-
-	-	Pressure loss coefficient in core Note: ARGO uses the analysis system for core part that is divided into 6 junctions. (See Figs. 8.3.3.1-4 and. 8.3.3.1-5) (j: the number of junction) j=1: IC, 2: MC, 3: OC, 4: shutdown rod, 5: barrel, and 6: reflector	-	-	-
1	-	+10% (for all the junctions 1 to 6)	a01	1.1E-02: L	1.6E-03: L
2	-	-10% (for only IC (j=1))	a01,a07	9.0E-02: H	4.3E-04: L
3	-	+10% (for only IC (j=1))	a01,a07	-9.7E-02: H	-1.4E-03: L
4	-	-10% (for only MC (j=2))	a01,a07	4.7E-02: M	-1.0E-03: L
5	-	+10% (for only MC (j=2))	a01,a07	-4.0E-02: M	2.5E-04: L
6	-	-10% (for only OC (j=3))	a01,a07	4.6E-02: M	-8.6E-04: L
7	-	+10% (for only OC (j=3))	a01,a07	-4.0E-02: M	1.2E-04: L
-	-	Area between coolant and radial shield Note: This parameter is related to natural circulation head. Four upper units of radial shield that consists of eight units are used as parameters in the calculation. (See Fig. 8.3.3.1-5)	-	-	-
8	-	-50% (4/8 times)	a03	< 1.0E-05: L	2.0E-02: L
9	-	-10%	a03	< 1.0E-05: L	1.1E-02: L
10	-	+10%	a03	< 1.0E-05: L	-6.8E-03: L
-	-	Gap conductance between fuel and cladding	-	-	-
11	-	5000 (assuming fuel pin of oxide fuel as the worst case)	a05	-2.0E-03: L	-1.5E-03: L
-	-	Heat transfer between structure and coolant in primary system Note: Heat transfer coefficient used in ARGO between "coolant and duct wall" and between "coolant and cladding" consists of two terms, that is, "thermal conduction term" and "forced convection term" as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
12	-	-20% (for both terms: C1, C2: -20%)	a06,c01	-1.2E-03: L	3.7E-04: L
13	-	+20% (for both terms: C1, C2: +20%)	a06,c01	6.7E-04: L	-1.8E-04: L
14	-	Not consider forced convection term (C2: 1.0e-10)	a06,c01	1.7E-04: L	< 1.0E-05: L
-	-	Pressure loss in the core in the case under 20% Re number of rated power Note: This parameter is related to flow distribution of the intra-assemblies.	-	-	-
15	-	2.0 times (IC)	a07	< 1.0E-05: L	3.4E-03: L
16	-	2.0 times (MC)	a07	< 1.0E-05: L	3.3E-03: L
17	-	2.0 times (OC)	a07	< 1.0E-05: L	2.3E-03: L
-	-	Radial heat transfer between subassemblies Note: Radial heat transfer coefficient used in ARGO between subassemblies consists of two terms, that is, "thermal conduction term" and "forced convection term."	-	-	-
18	-	Exclusion of all the terms from consideration	a08	4.3E-02: M	3.2E-02: L
19	-	Exclusion of forced convection term in this model from consideration	a08	< 1.0E-05: L	< 1.0E-05: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

Table 8.3.3.1-1. Setup Value of Each Parameter in Sensitivity Analysis (LOSP) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	A: Core/Fuel Assemblies	Heat transfer between reflector and coolant Note: Heat transfer coefficient used in ARGO between "reflector and coolant" consists of two terms, that is, "thermal conduction term" and "forced convection term" as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
20	-	-20% (for both terms: C1, C2: -20%)	a09	< 1.0E-05: L	3.7E-04: L
21	-	+20% (for both terms: C1, C2: +20%)	a09	< 1.0E-05: L	-1.8E-04: L
-	-	Heat capacity of core structure Note: Density ρ and specific heat Cp of core structure.	-	-	-
22	-	ρ: -3%, Cp: -10%	a10	3.3E-04: L	-5.2E-03: L
23	-	ρ: +3%, Cp: +10%	a10	-1.7E-04: L	5.7E-03: L
-	-	Heat capacity of core structure (HT-9) Note: Density ρ and specific heat Cp of core structure (IC, MC, OC).	-	-	-
24	-	ρ: +1%, Cp: +1%	a10	< 1.0E-05: L	1.2E-03: L
-	-	Density ρ and specific heat Cp of fuel	-	-	-
25	-	ρ: -5%, Cp: -5%	a10	< 1.0E-05: L	< 1.0E-05: L
26	-	ρ: +5%, Cp: +5%	a10	< 1.0E-05: L	< 1.0E-05: L
-	-	Power and amount of heat removal by SG	-	-	-
27	-	Power: -2%	a12	1.7E-03: L	-2.3E-03: L
28	-	Power: +2%	a12	-5.0E-04: L	2.0E-03: L
-	-	Decay heat	-	-	-
29	-	-5%	a13	< 1.0E-05: L	-9.1E-03: L
30	-	+5%	a13	< 1.0E-05: L	7.9E-03: L
-	-	Scram insertion rate	-	-	-
31	-	1/2 times	a15, a16, b22, b23	1.5E-03: L	4.7E-03: L
32	-	2 times	a15, a16, b22, b23	-8.4E-04: L	-2.6E-03: L
-	-	Lag time to scram	-	-	-
33	-	-50% (0.75 s)	a15, a16	-1.8E-03: L	-1.4E-03: L
34	-	+100% (3.0 s)	a15, a16	8.9E-03: L	2.8E-03: L
-	B: Reactor System	-	-	-	-
-	-	Volume in upper plenum Note: Parameter is volume of the highest unit in upper plenum, which consists of eight units (see Fig. 8.3.3.1-5). From the point of view that coolant in upper position in the upper plenum does not contribute to natural circulation, the unit volume of the highest unit is set to the smaller value than actual volume.	-	-	-
35	-	3/2 times (this means that the highest unit has five times larger volume than base case volume) (See Fig. 8.3.3.1-5)	b02	< 1.0E-05: L	2.4E-02: L
36	-	17/8 times (this means that the highest unit has 10 times larger volume than base case volume) (See Fig. 8.3.3.1-5)	b02	< 1.0E-05: L	2.9E-02: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter

Table 8.3.3.1-1. Setup Value of Each Parameter in Sensitivity Analysis (LOSP) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	B: Reactor System	Pressure loss coefficient in lower plenum	-	-	-
37	-	-10%	b09	< 1.0E-05: L	< 1.0E-05: L
38	-	+10%	b09	< 1.0E-05: L	< 1.0E-05: L
-	-	Heat capacity of radial shield structure Note: Density ρ and specific heat Cp of radial shield structure.	-	-	-
39	-	ρ: -10%, Cp: -10%	b19	< 1.0E-05: L	-6.2E-05: L
40	-	ρ: +10%, Cp: +10%	b19	< 1.0E-05: L	1.2E-04: L
-	C: PHTS	-	-	-	-
-	-	Pressure loss coefficient in primary system	-	-	-
41	-	+10%	c01,c04 c06	-1.7E-04: L	2.2E-03: L
-	-	Heat transfer area of IHX	-	-	-
42	-	-20%	c01	< 1.0E-05: L	-5.4E-03: L
-	-	Entire mass of sodium	-		-
43	-	-1%	c02	< 1.0E-05: L	-3.0E-03: L
-	-	Density ρ of sodium	-	-	-
44	-	ρ: -1%	c02,c03	1.0E-03: L	-6.2E-04: L
45	-	ρ: +1%	c02,c03	-1.0E-03: L	6.8E-04: L
-	-	Heat capacity of coolant (Specific heat Cp of coolant)	-	-	
46	-	Cp: +1%	c03	-2.0E-03: L	4.3E-04: L
-	-	Heat capacity of coolant Note: Density ρ and specific heat Cp of sodium.	-	-	-
47	-	ρ: +1%, Cp: +1%	c03	-2.0E-03: L	1.0E-03: L
48	-	ρ: -1%, Cp: -1%	c03	3.3E-03: L	-9.9E-04: L
-	-	Heat transfer between heat transfer tube and coolant in IHX Note: Heat transfer coefficient used in ARGO between "heat transfer tube and coolant" consists of two terms, that is, "thermal conduction term" and "forced convection term" for both primary and intermediate sides as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
49	-	-20% (for both terms in both sides: C1, C2: -20%)	c05	< 1.0E-05: L	-3.9E-03: L
50	-	+20% (for both terms in both sides: C1, C2: +20%)	c05	< 1.0E-05: L	2.9E-03: L
51	-	-20% (for both terms in primary side: C1, C2: -20%)	c05	< 1.0E-05: L	-1.5E-03: L
52	-	+20% (for both terms in primary side: C1, C2: +20%)	c05	< 1.0E-05: L	1.1E-03: L
53	-	-20% (for both terms in intermediate side: C1, C2: -20%)	c05	< 1.0E-05: L	-2.5E-03: L
54	-	+20% (for both terms in intermediate side: C1, C2: +20%)	c05	< 1.0E-05: L	1.8E-03: L
-	-	Heat capacity of structure in IHX Note: Density ρ and specific heat Cp of structure in IHX.	-	-	-
55	-	ρ: -5%, Cp: -5%	c08	< 1.0E-05: L	1.8E-04: L
56	-	ρ: +5%, Cp: +5%	c08	< 1.0E-05: L	-1.8E-04: L
	-	Halving time of flow coastdown of primary EMP	-	-	-
57	-	-33%	c10	2.0E-03: L	1.6E-03: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

Table 8.3.3.1-1. Setup Value of Each Parameter in Sensitivity Analysis (LOSP) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	D: IHTS	-	-	-	-
-	-	Pressure loss coefficient in intermediate system	-	-	-
58	-	-10%	d01,d02	< 1.0E-05: L	7.7E-03: L
59	-	+10%	d01,d02	< 1.0E-05: L	-8.7E-03: L
-	-	Elevation of AC	-	-	-
60	-	-1m	d02	-8.4E-04: L	7.7E-03: L
61	-	+1m	d02	-3.3E-04: L	-7.9E-03: L
-	-	Halving time of flow coastdown of intermediate EMP	-	-	-
62	-	-33%	d05	< 1.0E-05: L	3.3E-03: L
63	-	+33%	d05	< 1.0E-05: L	-6.9E-03: L
-	E: RHRS	-	-	-	-
-	-	Pressure loss coefficient in sodium side of IRACS	-	-	-
64	-	-10%	e01	< 1.0E-05: L	9.2E-04: L
65	-	+10%	e01	< 1.0E-05: L	-8.6E-04: L
-	-	Pressure loss coefficient in air side of IRACS	-	-	-
66	-	-10%	e01,e02	< 1.0E-05: L	-1.3E-02: L
67	-	+10%	e01,e02	< 1.0E-05: L	1.1E-02: L
-	-	Heat transfer between sodium and air in IRACS Note: Heat transfer coefficient used in ARGO between "sodium and air" consists of two terms, that is, "thermal conduction term" and "forced convection term" for both primary and intermediate sides as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
68	-	-20% (for both terms in air side: C1, C2: -20%)	e03	< 1.0E-05: L	4.2E-02: M
69	-	+20% (for both terms in air side: C1, C2: +20%)	e03	< 1.0E-05: L	-3.4E-02: L
70	-	-20% (for both terms in sodium side: C1, C2: -20%)	e04	< 1.0E-05: L	6.2E-05: L
71	-	+20% (for both terms in sodium side: C1, C2: +20%)	e04	< 1.0E-05: L	< 1.0E-05: L
72	-	-20% (for both terms in both sides: C1, C2: -20%)	e04	< 1.0E-05: L	4.2E-02: M
73	-	+20% (for both terms in both sides: C1, C2: +20%)	e04	< 1.0E-05: L	-3.4E-02: L
-	-	Air temperature at IRACS inlet (Reference temperature (Tref) = 50°C)	-	-	-
74	-	Tref + 10°C (60°C)	e05	< 1.0E-05: L	1.7E-02: L
75	-	Tref + 20°C (70°C)	e05	< 1.0E-05: L	3.3E-02: L
76	-	Tref - 60°C (-10°C)	e05	< 1.0E-05: L	-1.1E-01: H
77		Tref - 30°C (20°C)	e05	< 1.0E-05: L	-5.3E-02: M
-	-	Heat capacity of structure (sus304) in IRACS Note: Density ρ and specific heat Cp of structure (sus304) in IRACS.	-	-	-
78	-	ρ: -5%, Cp: -5%	e06	< 1.0E-05: L	1.2E-04: L
79	-	ρ: +5%, Cp: +5%	e06	< 1.0E-05: L	-6.2E-05: L
-	-	Pressure loss coefficient of airflow in RVACS	-	-	-
80	-	-10%	e07	< 1.0E-05: L	-6.2E-05: L
81	-	+10%	e07	< 1.0E-05: L	1.2E-04: L
-	-	Heat transfer area between GV and air in RVACS	-	-	-
82	-	-10%	e08	< 1.0E-05: L	1.8E-04: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

Table 8.3.3.1-1. Setup Value of Each Parameter in Sensitivity Analysis (LOSP) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	E: RHRS	Emissivity of radiation heat transfer between RV and GV (e1) and emissivity of radiation heat transfer between GV and heat collector (e2)	-	-	-
83	-	e1: -10%, e2: -10%	e08,e09, e10	< 1.0E-05: L	1.4E-03: L
84	-	e1: -25%, e2: -25%	e08,e09, e10	< 1.0E-05: L	3.7E-03: L
85	-	e1: -50%, e2: -50%	e08,e09, e10	< 1.0E-05: L	7.8E-03: L
86	-	e1: -75%, e2: -75%	e08,e09, e10	< 1.0E-05: L	1.2E-02: L
	-	Air temperature at RVACS inlet	-	-	-
87	-	60°C	e15	< 1.0E-05: L	1.8E-04: L
88	-	70°C	e15	< 1.0E-05: L	3.7E-04: L
89	-	-10°C	e15	< 1.0E-05: L	1.6E-03: L
90	-	20°C	e15	< 1.0E-05: L	-5.5E-04: L
-	-	Heat transfer between air and GV in RVACS Note: Heat transfer coefficient used in ARGO between "air and GV" consists of two terms, that is, "thermal conduction term" and "forced convection term" for both of primary and intermediate sides as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
91	-	-20% (for forced convection term: C2: -20%)	e09,e10	< 1.0E-05: L	3.7E-04: L
92	-	+20% (for forced convection term: C2: +20%)	e09,e10	< 1.0E-05: L	-3.1E-04: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

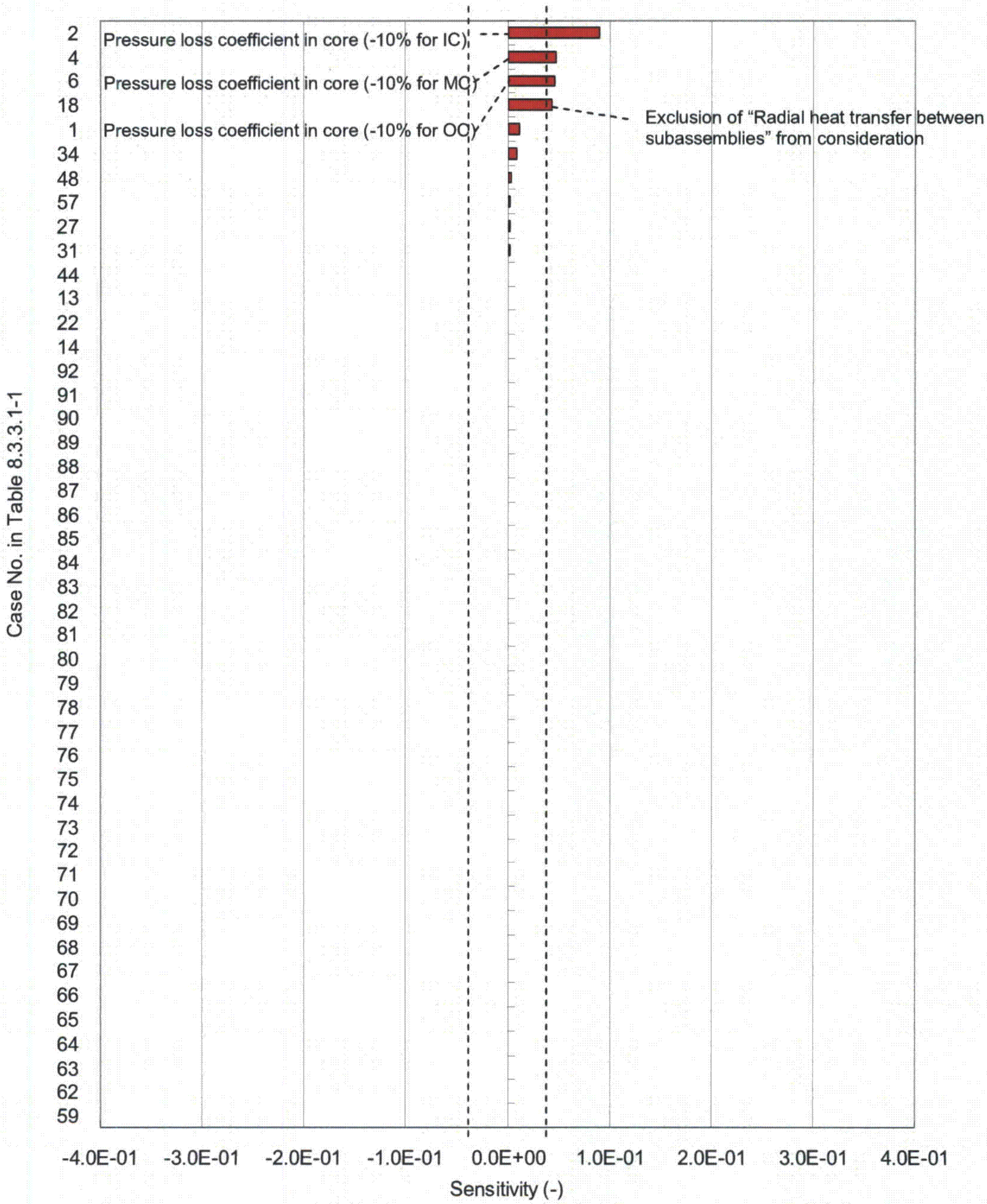


Figure 8.3.3.1-1. Summary of Sensitivity Analysis in LOSP (1st Phase: in order of sensitivity)

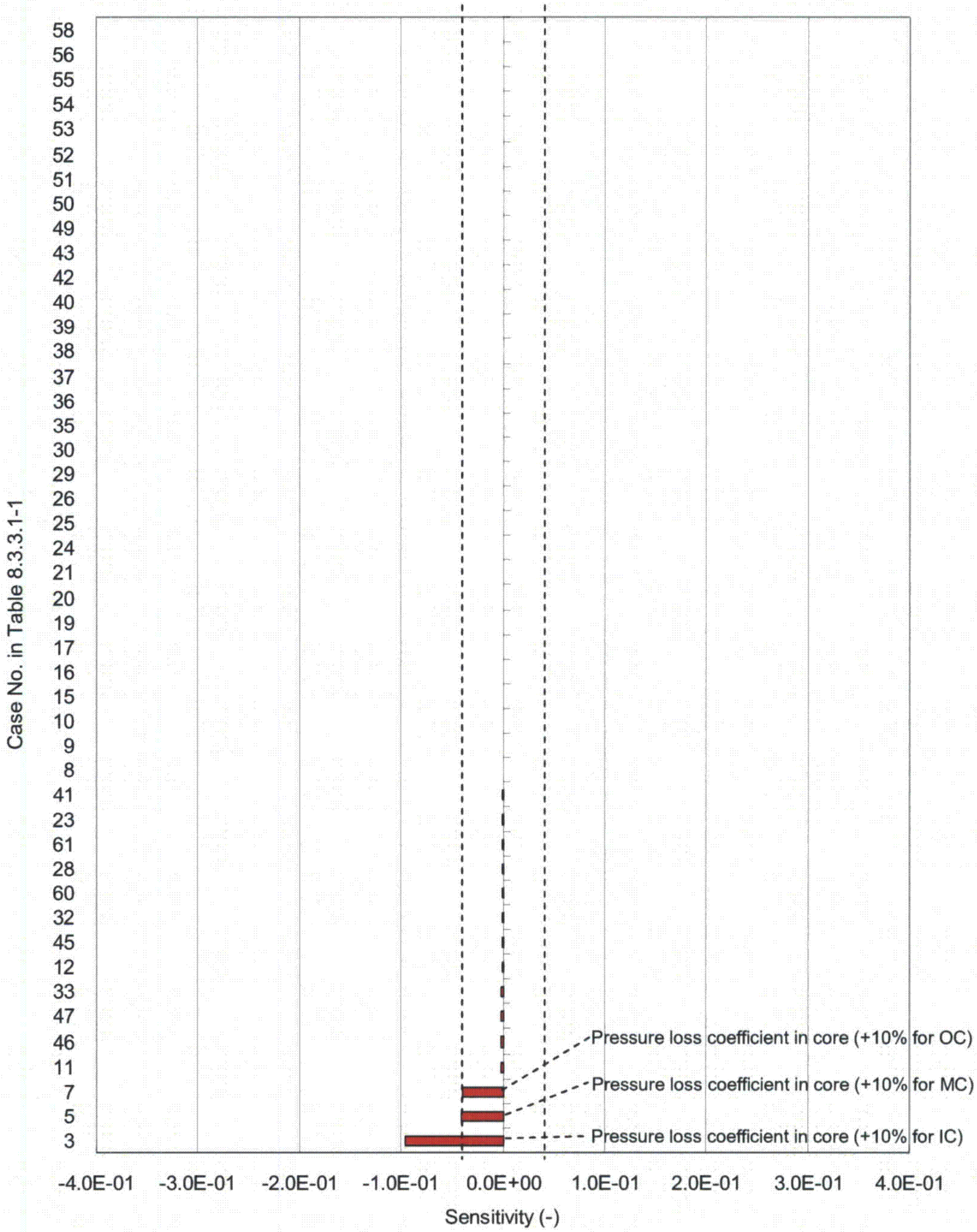


Figure 8.3.3.1-1. Summary of Sensitivity Analysis in LOSP (1st Phase: in order of sensitivity) (cont.)

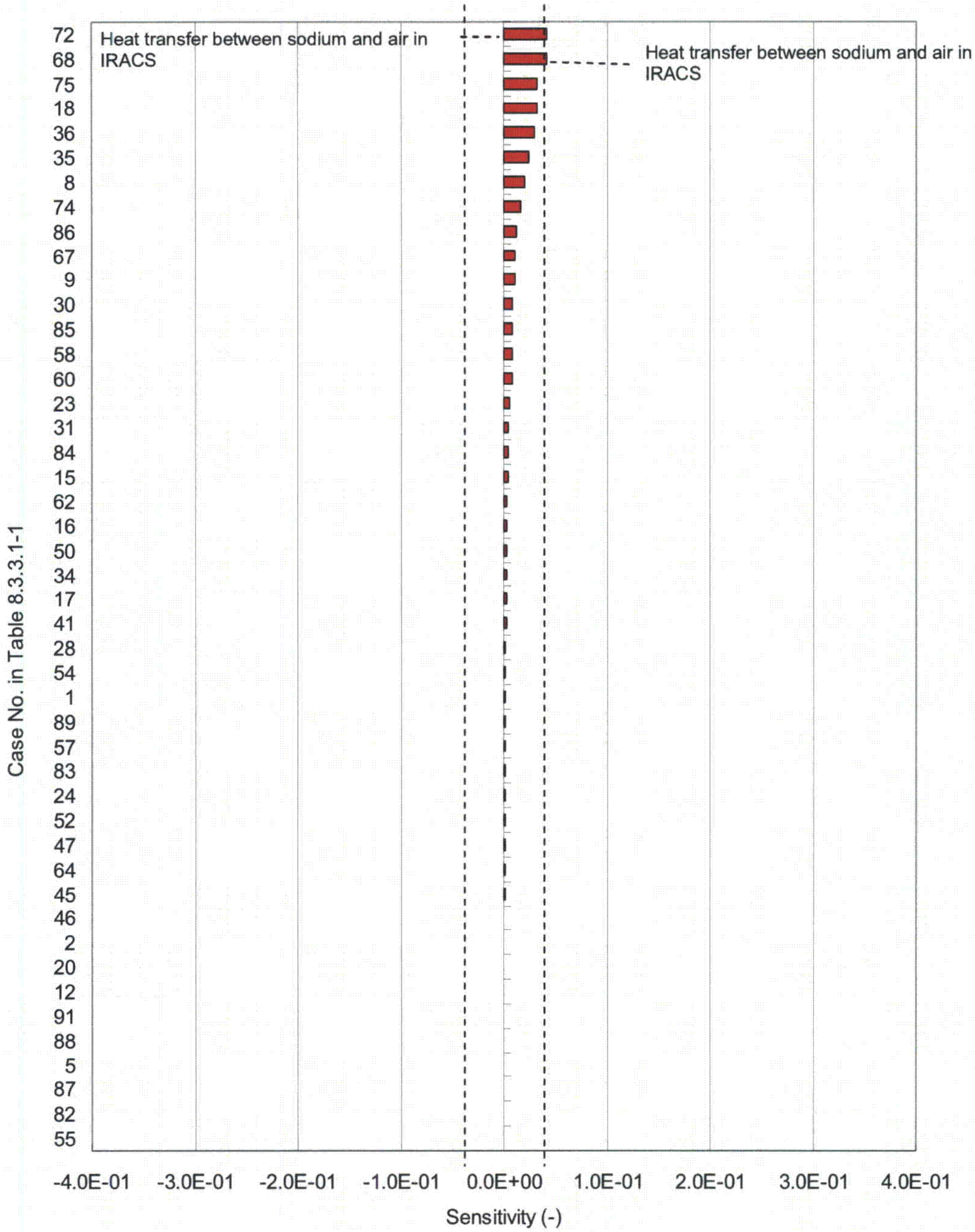


Figure 8.3.3.1-2. Summary of Sensitivity Analysis in LOSP (2nd Phase: in order of sensitivity)

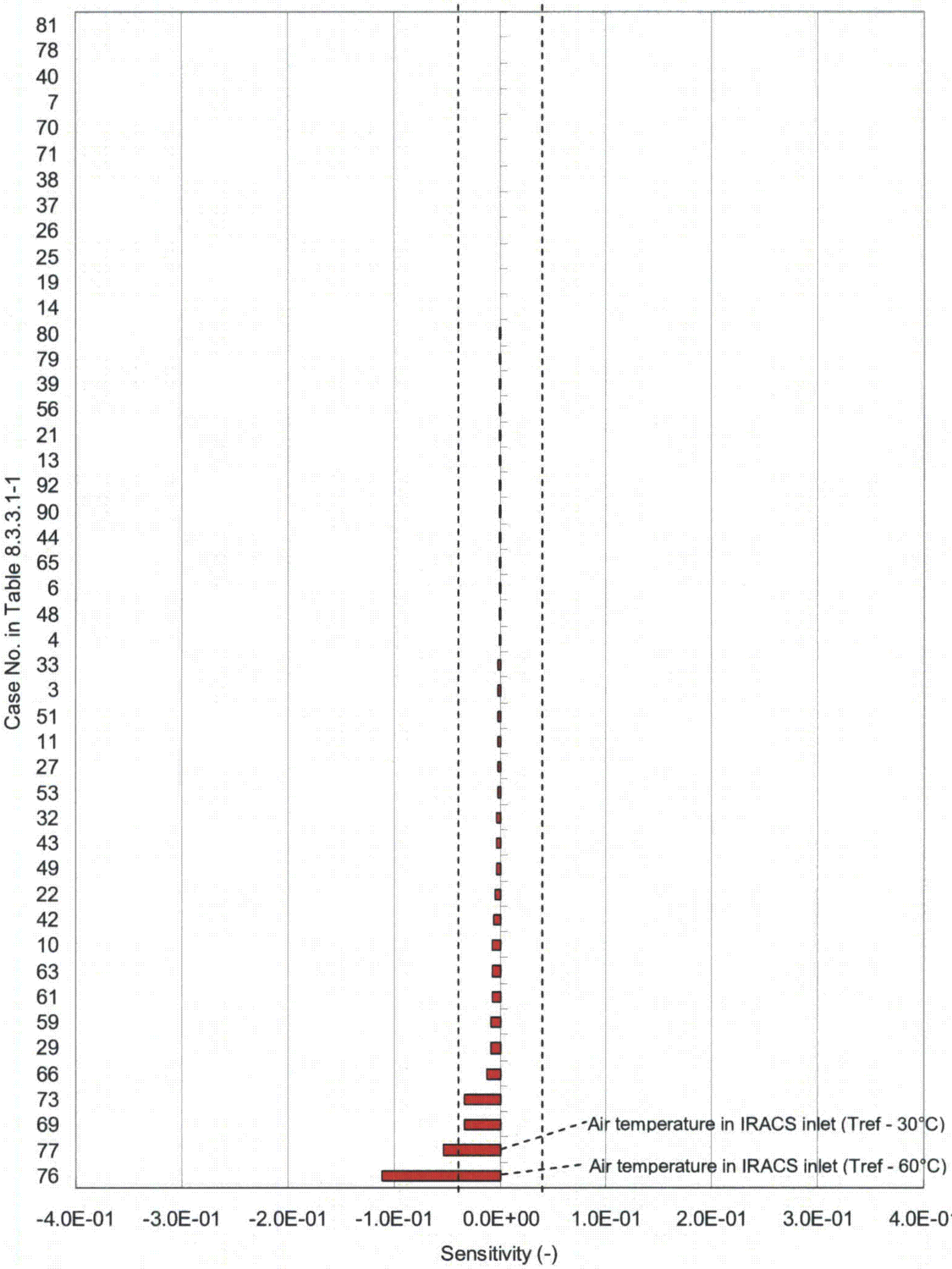


Figure 8.3.3.1-2. Summary of Sensitivity Analysis in LOSP (2nd Phase: in order of sensitivity) (cont.)

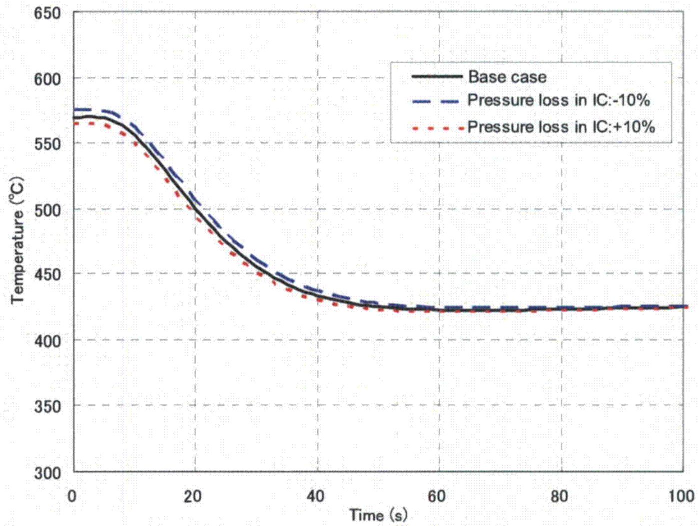


Figure 8.3.3.1-3. Example Result of Sensitivity Analysis in LOSP

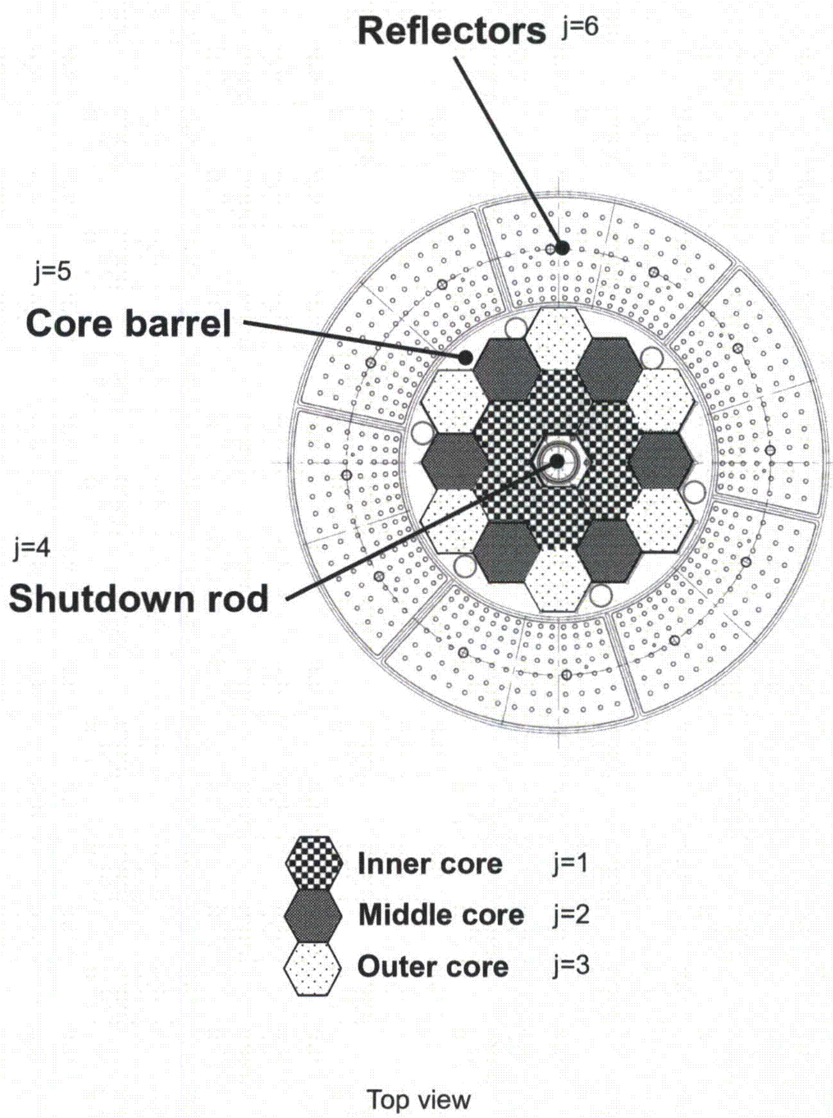
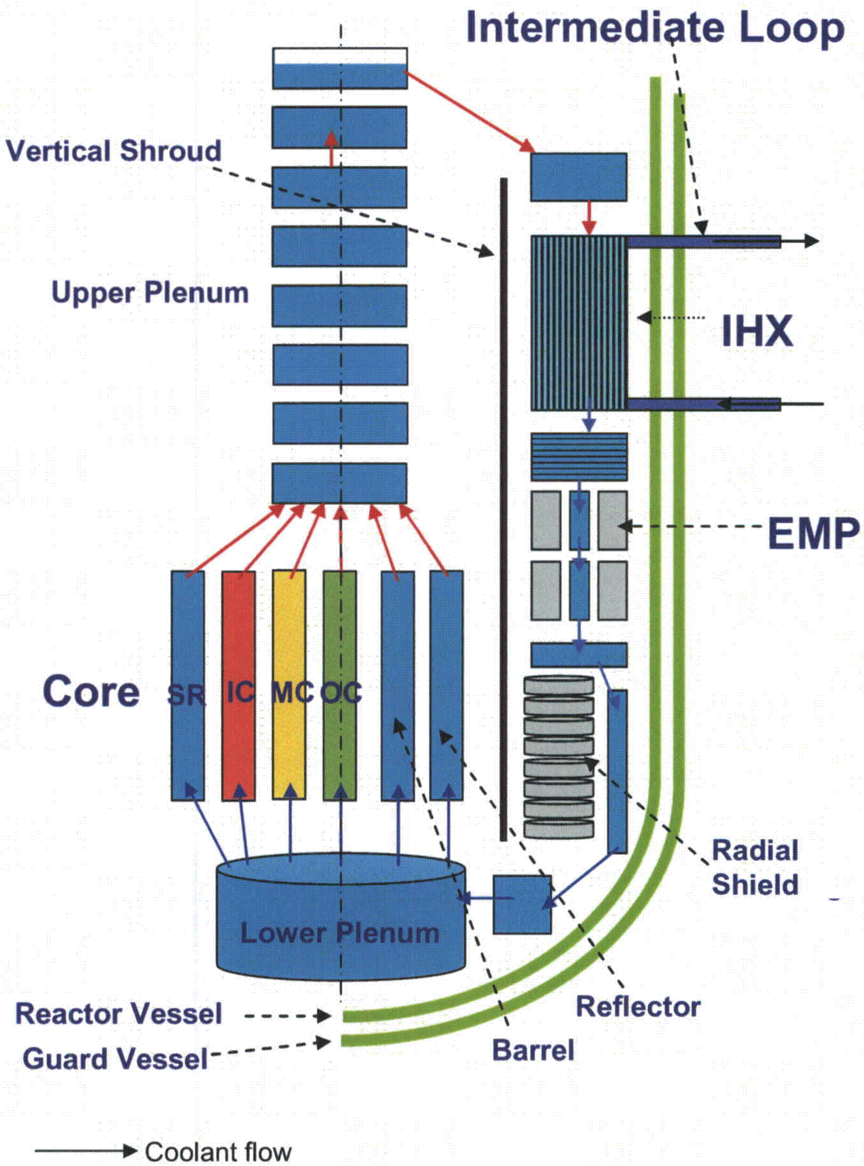


Figure 8.3.3.1-4. Configuration of Core



Configuration of Junctions and Units

Figure 8.3.3.1-5. Calculation Model of ARGO in Primary System

8.3.3.2 Sodium Leakage from Intermediate Piping (SLIP)

Table 8.3.3.2-1 summarizes all the case of the sensitivity analysis conducted for the SLIP. The columns are the same as in Table 8.3.3.1-1. Also, the results of the base case are as described in Section 4.2.

Figures 8.3.3.2-1 and 8.3.3.2-2 show a summary of the sensitivity analysis. The dotted line in these figures indicates the value of 0.04 separating sensitivities M from L. These are summaries of the 1st phase and 2nd phase, respectively, and are arranged in order of the scale of sensitivity. Figure 8.3.3.2-3 shows a concrete example for time variation of cladding temperature in this event, which is the result of Case no. 31 in Table 8.3.3.2-1.

A sensitivity of 0.04 for the SLIP event corresponds to about 2.4°C of fuel cladding temperature (FoM) in the 1st phase and about 4.4°C in the 2nd phase.

The cases that resulted in significant sensitivity (sensitivity of M or more) are described below.

(a) Parameter: Pressure loss coefficient in core (Case nos. 1 to 7)

The coefficient of pressure loss is used as a parameter and affects the coolant flow rate of the core. Sensitivity for the case in which the overall core pressure loss coefficient is changed by 1σ (Case no. 1) corresponds to L and is small. However, the 1st phase of the cases where the pressure loss coefficient is locally changed by 1σ (Case nos. 2 through 7) shows a sensitivity of M or more. Specifically, sensitivity in the case where only the pressure loss coefficient of the IC is changed is about 0.1 and corresponds to H. Setting of this latter parameter relates to flow distribution in the core and shows that the uncertainty of the flow rate distribution of intra- and inter-assembly coolant has a strong effect on cladding temperature.

(b) Area between coolant and radial shield (Case nos. 8 to 10)

Heat transfer area between the radial shield and coolant is used as a parameter. This parameter affects the amount of heat transfer between the radial shield and coolant, and natural circulation in the reactor vessel. The case where the area of heat transfer is reduced by 10 percent shows a sensitivity corresponding to M. This result shows that uncertainty in the amount of heat transfer between the radial shield and coolant has a relatively large effect on cladding temperature.

(c) Radial heat transfer between subassemblies (Case nos. 17 and 18)

Radial heat transfer in the core is used as a parameter. Sensitivity in the case, without considering the whole of the radial heat transfer, is 0.1 or more and shows a sensitivity corresponding to H. This result shows that radial heat transfer in the core has a large effect on cladding temperature.

(d) Volume in the upper plenum (Case nos. 31 and 32)

The coolant volume of the upper plenum is used as a parameter (See Figure 8.3.3.2-4). For this parameter, the set value itself is conservative as described in Section 8.3.2. Specifically, it uses about 60 percent of the designed volume. The volume of the top unit in the upper plenum is divided into eight equal units. The sensitivity is M, even in Case no. 31, which has five times the volume of the top unit as the base case, equal to 150 percent of the overall upper plenum volume. This result shows that the coolant on top of the upper plenum, while considered to make little contribution to natural circulation, has a relatively large effect on cladding temperature.

(e) Emissivity of radiation heat transfer between RV and GV, and GV and heat collector (Case nos. 49 to 55)

The shape coefficient of radiation in the RVACS is used as a parameter. This parameter specifies the heat removal capacity by RVACS radiation. Case no. 55 is the case wherein the value 0.5 of emissivity used in design, ε, is reduced by 60 percent (ε=0.2) and shows the sensitivity corresponding to M. There is little information regarding the emissivity changes caused by age deterioration, and its uncertainty is large. Therefore, this level of emissivity change should be considered within the range of possibility. This result shows that the uncertainty of emissivity has a relatively large effect on cladding temperature.

Table 8.3.3.2-1. Setup Value of Each Parameter in Sensitivity Analysis (SLIP)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	-	Base Case: ±0%	-	-	-
-	A: Core/Fuel Assemblies	-	-	-	-
-	-	Pressure loss coefficient in core Note: ARGO uses the analysis system for core part that is divided into 6 junctions. (See Figs. 8.3.3.1-4 and 8.3.3.1-5) (j: the number of junction) j=1: IC, 2: MC, 3: OC, 4: shutdown rod, 5: barrel, and 6: reflector	-	-	-
1	-	+10% (for all the junctions 1 to 6)	a01	-1.7E-04: L	5.6E-03: L
2	-	+10% (for only IC (j=1))	a01,a07	9.0E-02: H	2.2E-03: L
3	-	-10% (for only IC (j=1))	a01,a07	-9.7E-02: H	-4.1E-03: L
4	-	+10% (for only MC (j=2))	a01,a07	-4.1E-02: M	1.4E-03: L
5	-	-10% (for only MC (j=2))	a01,a07	4.7E-02: M	-3.3E-03: L
6		+10% (for only OC (j=3))	a01,a07	-4.0E-02: M	4.5E-04: L
7		-10% (for only OC (j=3))	a01,a07	4.6E-02: M	-2.2E-03: L
-	-	Area between coolant and radial shield Note: This parameter is related to natural circulation head. Four upper units of radial shield, which consists of eight units, are used as a parameter in the calculation. (See Fig. 8.3.3.1-5)	-	-	-
8	-	-50% (4/8 times)	a03	< 1.0E-05: L	2.4E-01: H
9	-	-10%	a03	< 1.0E-05: L	5.7E-02: M
10	-	+10%	a03	< 1.0E-05: L	-1.7E-02: L
	-	Gap conductance between fuel and cladding	-	-	-
11	-	5000 (assume oxide fuel)	a05	< 1.0E-05: L	-4.1E-03: L
-	-	Heat transfer between structure and coolant in primary system Note: Heat transfer coefficient used in ARGO between "coolant and duct wall" and between "coolant and cladding" consists of two terms, that is, "thermal conduction term" and "forced convection term" as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
12	-	-20% (for both terms: C1, C2: -20%)	a06,c01	< 1.0E-05: L	-4.5E-04: L
13	-	+20% (for both terms: C1, C2: +20%)	a06,c01	3.3E-04: L	3.6E-04: L
-	-	Pressure loss in the core in the case under 20% Re number of rated power Note: This parameter is related to flow distribution of the intra-assemblies.	-	-	-
14	-	1.5 times (IC)	a07	< 1.0E-05: L	8.1E-03: L
15	-	1.5 times (MC)	a07	< 1.0E-05: L	7.5E-03: L
16	-	1.5 times (OC)	a07	< 1.0E-05: L	6.1E-03: L
-	--	Radial heat transfer between subassemblies Note: Radial heat transfer coefficient used in ARGO between subassemblies consists of two terms, that is, "thermal conduction term" and "forced convection term"	-	-	-
17	-	Exclusion of all the terms from consideration	a08	1.7E-01: H	1.3E-01: H
18	-	Exclusion of forced convection term in this model from consideration	a08	< 1.0E-05: L	< 1.0E-05: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

Table 8.3.3.2-1. Setup Value of Each Parameter in Sensitivity Analysis (SLIP) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	A: Core/Fuel Assemblies	Heat transfer between reflector and coolant Note: Heat transfer coefficient used in ARGO between "reflector and coolant" consists of two terms, that is, "thermal conduction term" and "forced convection term" as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
19	-	-20% (for both terms: C1, C2: -20%)	a09	< 1.0E-05: L	-2.7E-04: L
20	-	+20% (for both terms: C1, C2: +20%)	a09	< 1.0E-05: L	2.7E-04: L
-	-	Heat capacity of core structure. Note: Density ρ and specific heat Cp of core structure.	-	-	-
21	-	ρ: -3%, Cp: -10%	a10	< 1.0E-05: L	2.4E-03: L
22	-	ρ: +3%, Cp: +10%	a10	< 1.0E-05: L	-2.2E-03: L
-	-	Density ρ and specific heat Cp of fuel	-	-	-
23	-	ρ: -5%, Cp: -5%	a10	< 1.0E-05: L	< 1.0E-05: L
24	-	ρ: +5%, Cp: +5%	a10	< 1.0E-05: L	< 1.0E-05: L
-	-	Power and amount of heat removal by SG	-	-	-
25	-	Power: -2%	a12	< 1.0E-05: L	-5.0E-03: L
26	-	Power: +2%	a12	3.3E-04: L	5.1E-03: L
-	-	Decay heat	-	-	-
27	-	-5%	a13	< 1.0E-05: L	-1.3E-02: L
28	-	+5%	a13	< 1.0E-05: L	1.3E-02: L
-	-	Scram insertion rate	-	-	-
29	-	1/2 times	a15, a16, b22, b23	< 1.0E-05: L	-1.6E-03: L
30	-	2 times	a15, a16, b22, b23	< 1.0E-05: L	9.9E-04: L
-	B: Reactor System	-	-	-	-
-	-	Volume in upper plenum Note: Parameter is volume of the highest unit in upper plenum, which consists of eight units (see Fig. 8.3.3.1-5). From the point of view that coolant in upper position in the upper plenum does not contribute to natural circulation, the unit volume of the highest unit is set to the smaller value than actual volume.	-	-	-
31	-	3/2 times (this means that the highest unit has five times larger volume than base volume) (See Fig. 8.3.3.1-5)	b02	< 1.0E-05: L	5.0E-02: M
32	-	17/8 times (this means that the highest unit has 10 times larger volume than base volume) (See Fig. 8.3.3.1-5)	b02	< 1.0E-05: L	7.1E-02: M
-	-	Pressure loss coefficient in lower plenum	-	-	-
33	-	-10%	b09	< 1.0E-05: L	< 1.0E-05: L
34	-	+10%	b09	< 1.0E-05: L	< 1.0E-05: L
-	-	Heat capacity of radial shield structure Note: Density ρ and specific heat Cp of radial shield structure.	-	-	-
35	-	ρ: -10%, Cp: -10%	b19	< 1.0E-05: L	1.2E-02: L
36	-	ρ: +10%, Cp: +10%	b19	< 1.0E-05: L	-9.5E-03: L
-	C: PHTS	-	-	-	-
-	-	Pressure loss coefficient in primary system	-	-	-
37	-	+10%	c01,c04, c06	-3.3E-04: L	5.9E-03: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

Table 8.3.3.2-1. Setup Value of Each Parameter in Sensitivity Analysis (SLIP) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)	Sensitivity (2nd Phase)
-	C: PHTS	Density ρ of sodium	-	-	-
38	-	ρ: -1%	c03	1.7E-04: L	8.1E-04: L
39	-	ρ: +1%	c03	< 1.0E-05: L	-7.2E-04: L
-	-	Heat capacity of coolant Note: Density ρ and specific heat Cp of sodium.	-	-	-
40	-	ρ: +1%, Cp: +1%	c03	< 1.0E-05: L	-1.3E-03: L
41	-	ρ: -1%, Cp: -1%	c03	5.0E-04: L	1.4E-03: L
-	-	Halving time of flow coastdown of primary EMP			
42	-	-33%	c10	< 1.0E-05: L	-7.3E-03: L
-	E: RHRS	-	-	-	-
-	-	Pressure loss coefficient of airflow in RVACS	-	-	-
43	-	-10%	e07	< 1.0E-05: L	-2.7E-04: L
44	-	+10%	e07	< 1.0E-05: L	3.6E-04: L
45	-	+100%	e07	< 1.0E-05: L	1.89E-03: L
46	-	+400%	e07	< 1.0E-05: L	3.87E-03: L
47	-	+900%	e07	< 1.0E-05: L	4.95E-03: L
-	-	Heat transfer area between GV and air in RVACS	-	-	-
48	-	-10%	e08	< 1.0E-05: L	7.2E-04: L
-	-	Emissivity of radiation heat transfer between RV and GV (e1) and emissivity of radiation heat transfer between GV and heat collector (e2)	-	-	-
49	-	e1: -10%, e2: -10%	e08,e09, e10	< 1.0E-05: L	5.0E-03: L
50	-	e1: -25%, e2: -25%	e08,e09, e10	< 1.0E-05: L	1.3E-02: L
51	-	e1: -50%, e2: -50%	e08,e09, e10	< 1.0E-05: L	2.8E-02: L
52	-	e1: -75%, e2: -75%	e08,e09, e10	< 1.0E-05: L	Note 1
53	-	e1: 0.053, e2: 0.053	e08,e09, e10	< 1.0E-05: L	Note 1
54	-	e1: 0.11, e2: 0.11	e08,e09, e10	< 1.0E-05: L	4.1E-02: M
55	-	e1: 0.18, e2: 0.18	e08,e09, e10	< 1.0E-05: L	2.6E-02: L
-	-	Heat transfer between air and GV in RVACS Note: Heat transfer coefficient used in ARGO between "air and GV" consists of two terms, that is, "thermal conduction term" and "forced convection term" for both of primary and intermediate sides as follows. Nu = C1 + C2 • Re^C3 • Pr^C4 eq. (8.3.3.1-1)	-	-	-
56	-	-20% (for both terms: C2: -20%)	e09	< 1.0E-05: L	1.8E-04: L
57	-	+20% (for both terms: C2: +20%)	e09	< 1.0E-05: L	-9.0E-05: L
-	-	Air temperature at RVACS inlet	-	-	-
58	-	60°C	e15	< 1.0E-05: L	7.2E-04: L
59	-	70°C	e15	< 1.0E-05: L	1.4E-03: L
60	-	-10°C	e15	< 1.0E-05: L	-3.9E-03: L
61	-	20°C	e15	< 1.0E-05: L	-2.3E-03: L

Note:
1. The 2nd peak does not appear during the transient calculation for 400,000 seconds. Hence, the sensitivity of this phase cannot be evaluated.

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

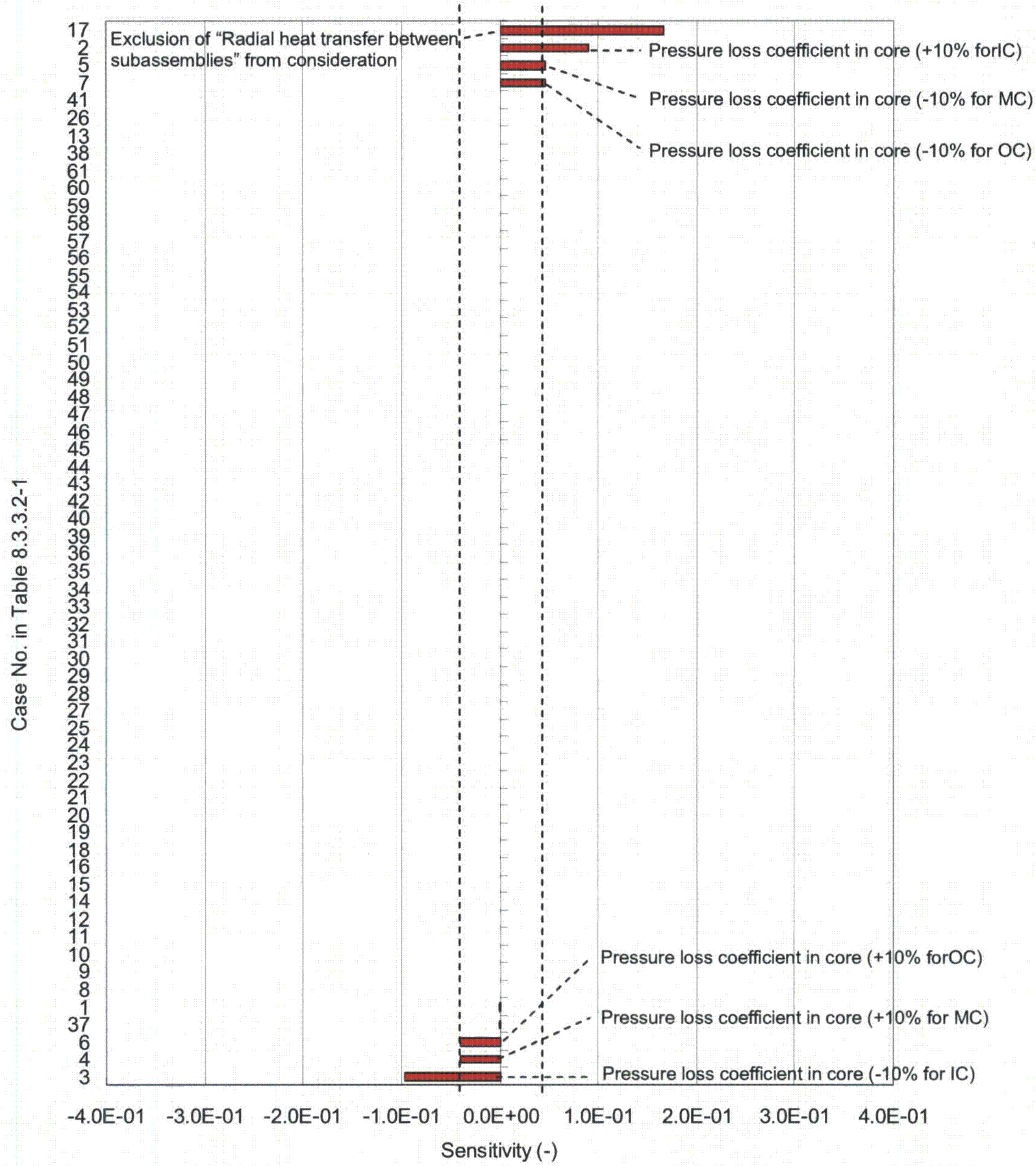


Figure 8.3.3.2-1. Summary of Sensitivity Analysis in SLIP (1st Phase: in order of sensitivity)

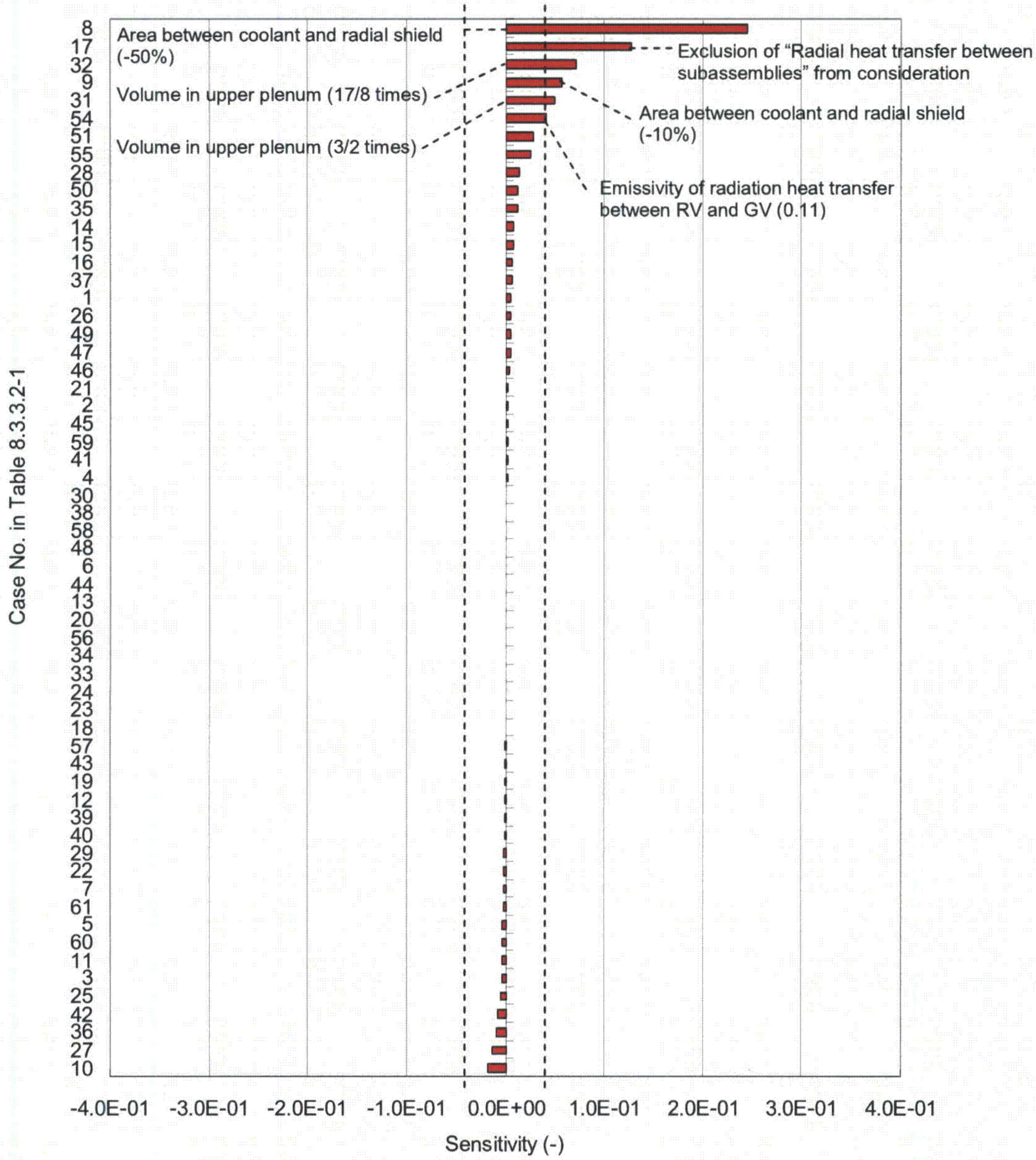


Figure 8.3.3.2-2. Summary of Sensitivity Analysis in SLIP
(2nd Phase: in order of sensitivity)

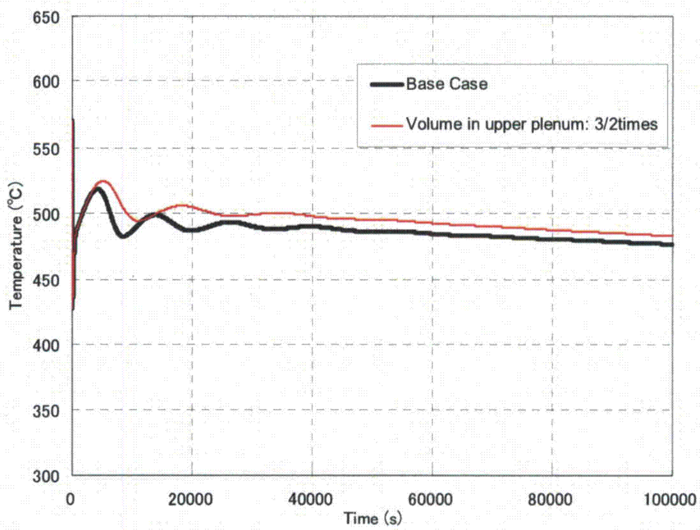


Figure 8.3.3.2-3. Example Result of Sensitivity Analysis in SLIP

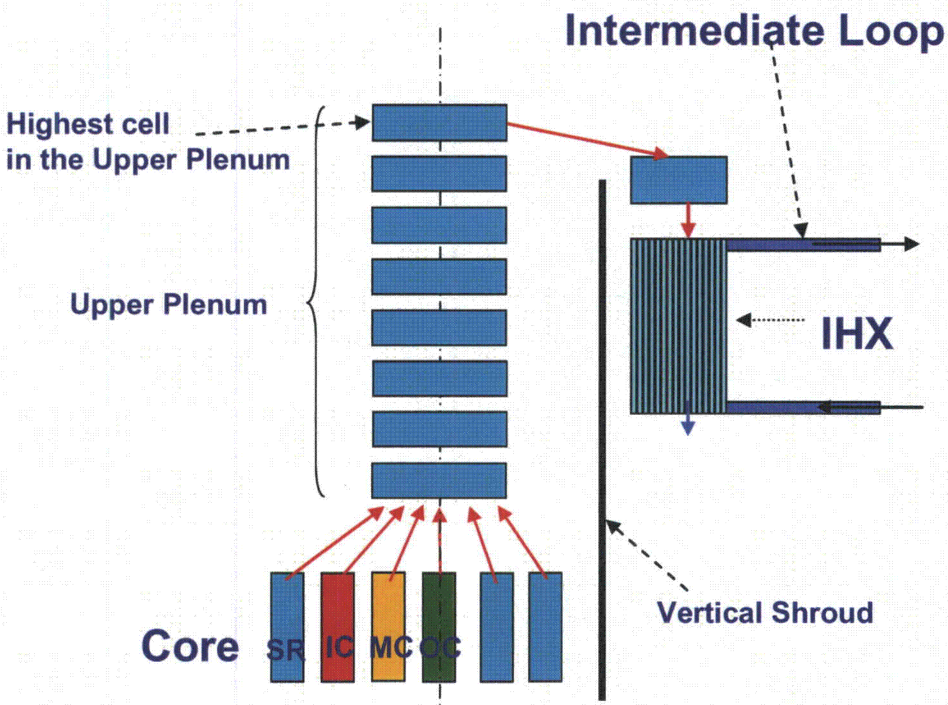
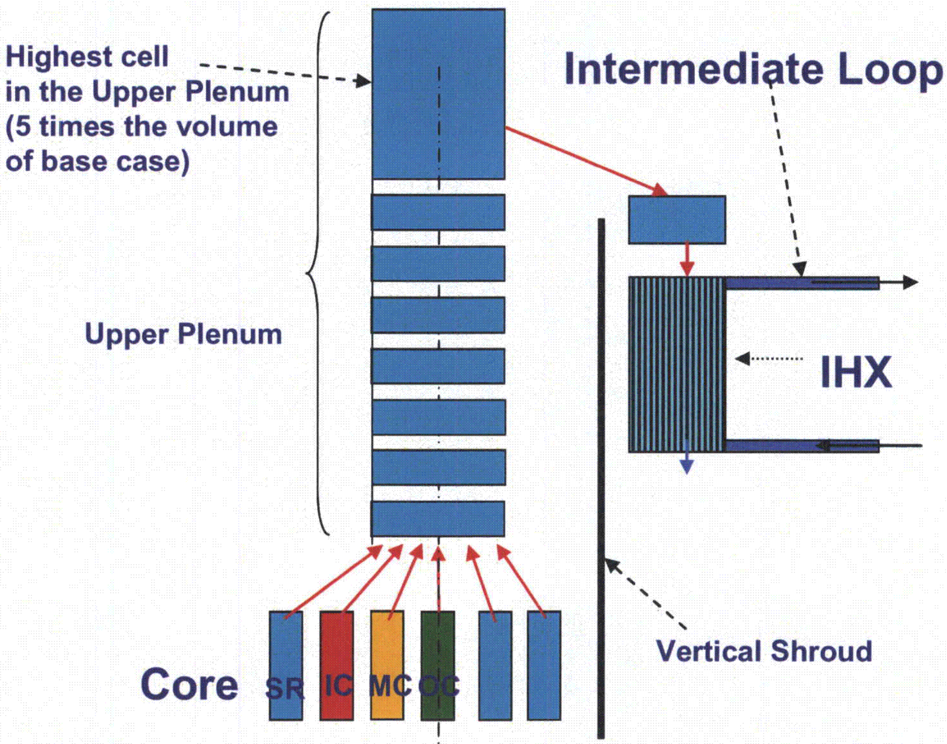
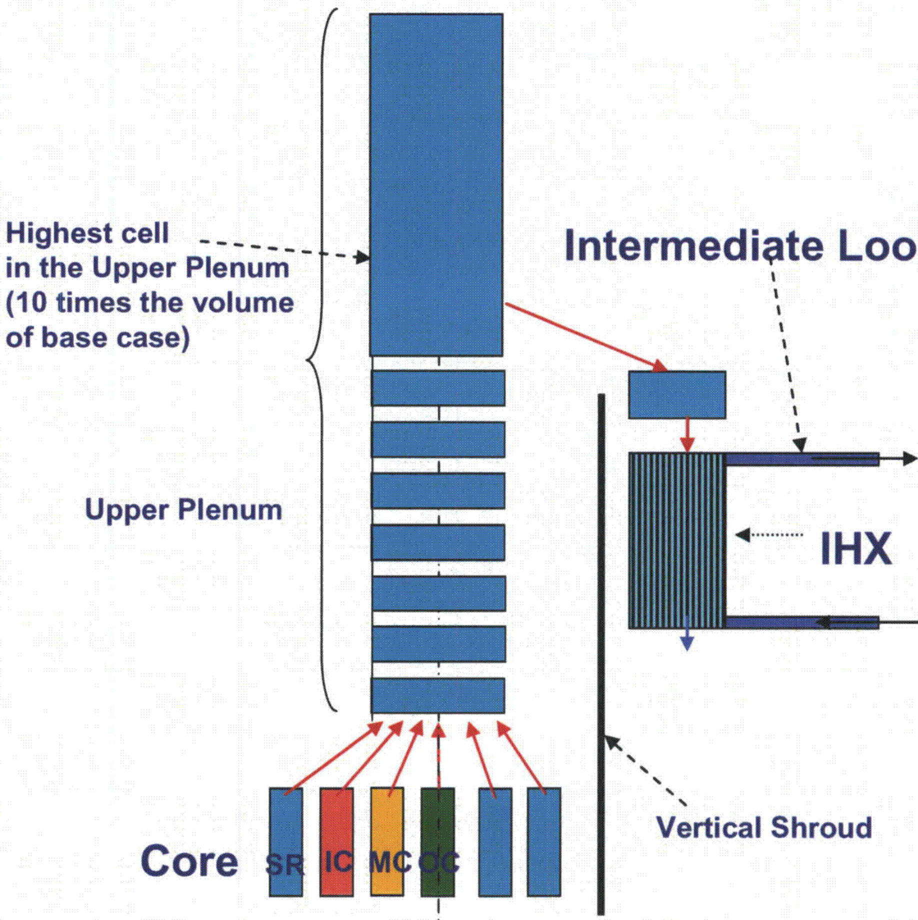


Figure 8.3.3.2-4 (1/3). Coolant Volume in the Highest Unit of Upper Plenum



(b) 5 times volume of base case

Figure 8.3.3.2-4 (2/3). Coolant Volume in the Highest Unit of Upper Plenum



(c) 10 times volume of base case

Figure 8.3.3.2-4 (3/3). Coolant Volume in the Highest Unit of Upper Plenum

8.3.3.3 Failure of a Cavity Can

Table 8.3.3.3-1 summarizes the sensitivity analysis cases conducted for the FCC. The columns are the same as in Table 8.3.3.1-1. Also, the results of the base case are as described in Section 4.2.

Figure 8.3.3.3-1 shows a summary of the sensitivity analysis, arranged in order of scale of sensitivity obtained from the analysis. The dotted line in this figure indicates the value of 0.04, separating sensitivities M from L. Figure 8.3.3.3-2 shows a concrete example for the time variation of cladding temperature in this event, which is the result of Case nos. 24 and 25 in Table 8.3.3.3-1.

A sensitivity value of 0.04 for the FCC event corresponds to about 1.9°C of fuel cladding temperature (FoM).

The cases resulted in significant sensitivity (sensitivity of M or more) are described below.

(a) Parameter: Pressure loss coefficient in core (Case nos. 1 to 6)

The coefficient of pressure loss is used as a parameter and affects the coolant flow rate of the core. Sensitivity in the cases where the pressure loss coefficient is locally changed by 1σ is M or more. Specifically, sensitivity in the case in which the pressure loss coefficient is changed only in the IC is about 0.1, and it corresponds to H. Setting of this parameter relates to the flow distribution of the core and shows that the uncertainty in distribution of the intra- and inter-assembly coolant flow rate has a large effect on cladding temperature.

(b) Radial heat transfer between subassemblies (Case nos. 9 and 10)

The occurrence of radial heat transfer in the core is used as a parameter. Sensitivity in the case, without considering the radial heat transfer of the core while it is considered in base case, is 0.23, which corresponds to H. This result shows that radial transfer of heat generated in the core has a large effect on cladding temperature.

(c) Maximum insertion reactivity (Case nos. 17 and 18)

The amount of the insertion reactivity is used as a parameter. This parameter relates to the amount of the insertion reactivity caused by the cavity can failure. The case where the amount of the insertion reactivity is changed by 1σ (Case nos. 17 and 18) shows a sensitivity corresponding to M. On the other hand, the cases where the insertion rate is used as parameter (Case nos. 19 to 23) do not indicate sensitivity of M or more except in extreme cases. This result shows that uncertainty of the reactivity insertion caused by cavity can failure has a relatively large effect on cladding temperature.

(d) Scram insertion rate (Case nos. 24 to 27)

The scram insertion velocity is used as a parameter. This parameter defines the time from the initiation to the completion of the reactor shutdown. Sensitivity in the case where scram insertion velocity changes by 1σ is not M or more. However, compared with the results of the LOSP and SLIP events, it is a large value that is close to the criterion value for M. Also, in the case where the parameter value is changed by an extremely large amount, a sensitivity corresponding to H is shown. This result shows that the uncertainty in scram insertion velocity has a relatively large effect on cladding temperature.

(e) Lag time to scram (Case nos. 28 to 31)

The delay time until the scram starts is used as a parameter. Like (d), in the case where scram insertion velocity is changed by 1σ, this parameter does not show a sensitivity of M or more. However, compared with the results of the LOSP and SLIP events, it shows a large value that is close to the criterion value for M. Also, the result of the case in which the parameter value is changed by an extremely large amount shows the same trend as (d). This result shows that the uncertainty in delay time until the scram starts has a relatively large effect on cladding temperature.

Table 8.3.3.3-1. Setup Value of Each Parameter in Sensitivity Analysis (FCC)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)
-	-	Base Case: ±0%	-	-
-	A: Core/Fuel Assemblies	-	-	-
-	-	Pressure loss coefficient in core Note: ARGO uses the analysis system for core part that is divided into 6 junctions. (See Figs. 8.3.3.1-4 and 8.3.3.1-5) (j: the number of junction) j=1: IC, 2: MC, 3: OC, 4: shutdown rod, 5: barrel, and 6: reflector	-	-
1	-	-10% (for only IC (j=1))	a07	-1.2E-01: H
2	-	+10% (for only IC (j=1))	a07	1.1E-01: H
3	-	-10% (for only MC (j=2))	a07	5.6E-02: M
4	-	+10% (for only MC (j=2))	a07	-4.9E-02: M
5	-	-10% (for only OC (j=3))	a07	5.6E-02: M
6	-	+10% (for only OC (j=3))	a07	-4.8E-02: M
-	-	Doppler reactivity coefficient Note: wdopl(l,k) l=1,2, k=1,mchnl (k: the number of channel, 1: IC, 2: MC, 3: OC)	-	-
7	-	wdopl: -10% (l=1,2, k=1,3)	a04	1.3E-03: L
8	-	wdopl: +10% (l=1,2, k=1,3)	a04	-1.3E-03: L
-	-	Radial heat transfer between subassemblies Note: Radial heat transfer coefficient used in ARGO between subassemblies consists of two terms, that is, "thermal conduction term" and "forced convection term."	-	-
9	-	Exclusion of all the terms from consideration	a08	2.3E-01: H
10	-	Exclusion of forced convection term in this model from consideration	a08	< 1.0E-05: L
-	-	Heat capacity of core structure Note: Density ρ and specific heat Cp of core structure.	-	-
11	-	ρ: -3%, Cp: -10%	a10	2.4E-02: L
12	-	ρ: +3%, Cp: +10%	a10	-2.3E-02: L
-	-	Core power	-	-
13	-	Power: -2%	a12	-5.7E-03: L
14	-	Power:+2%	a12	6.8E-03: L
-	-	Decay heat	-	-
15	-	-20%	a13	2.8E-03: L
16	-	+20%	a13	-1.9E-03: L
-	-	Maximum insertion reactivity	-	-
17	-	-10%	a26	-5.2E-02: M
18	-	+10%	a26	5.8E-02: M
-	-	Reactivity insertion rate	-	-
19	-	-10%	a26	< 1.0E-05: L
20	-	+10%	a26	1.7E-03: L
21	-	+100%	a26	3.2E-03: L
22	-	+200%	a26	2.8E-03: L
23	-	-50%	a26	-1.3E-02: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

Table 8.3.3.3-1. Setup Value of Each Parameter in Sensitivity Analysis (FCC) (cont.)

Case No.	Subsystem/ Component	Explanation of Parameter Set Value (Range of Variation from Base Case)	Code*	Sensitivity (1st Phase)
-	A: Core/Fuel Assemblies	Scram insertion rate	-	-
24	-	Velocity: 2 times	a15, a16, b22, b23	-1.5E-01: H
25	-	Velocity: 1/2 times	a15, a16, b22, b23	3.7E-01: H
26	-	time: -10%	a15, a16, b22, b23	-3.2E-02: L
27	-	time: +10%	a15, a16, b22, b23	3.4E-02: L
-	-	Lag time to scram	-	-
28	-	-50% (0.5 s)	a15, a16	-9.0E-02: M
29	-	+100% (2.0 s)	a15, a16	2.1E-01: H
30	-	-10% (0.9 s)	a15, a16	-2.0E-02: L
31	-	+10% (1.1s)	a15, a16	2.0E-02: L
-	B: Reactor System	-	-	-
-	-	Volume in upper plenum Note: Parameter is volume of the highest unit in upper plenum (see Fig. 8.3.3.1-5). From the point of view that coolant in upper position in the upper plenum does not contribute to natural circulation, the unit volume of the highest unit is set to the smaller value than actual volume.	-	-
32	-	3/2 times (this means that the highest unit has five times larger volume than original volume) (See Fig. 8.3.3.1-5)	b02	< 1.0E-05: L
33	-	17/8 times (this means that the highest unit has 10 times larger volume than original volume) (See Fig. 8.3.3.1-5)	b02	< 1.0E-05: L
-	-	Heat capacity of radial shield structure Note: Density ρ and specific heat Cp of radial shield structure.	-	-
34	-	ρ: -10%, Cp: -10%	b19	< 1.0E-05: L
35	-	ρ: +10%, Cp: -10%	b19	< 1.0E-05: L
-	C: PHTS	-	-	-
-	-	Heat capacity of coolant Note: Density ρ and specific heat Cp of coolant.	-	-
36	-	ρ: -1%, Cp: -1%	c03	1.5E-03: L
37	-	ρ: +1%, Cp: +1%	c03	6.4E-04: L
-	-	Heat capacity of structure of IHX Note: Density ρ and specific heat Cp of structure in IHX.	-	-
38	-	ρ: -5%, Cp: -5%	c08	< 1.0E-05: L
39	-	ρ: +5%, Cp: +5%	c08	< 1.0E-05: L

* Code in Table 7.1-1 to identify each phenomenon. Here, code indicates phenomena related to each parameter.

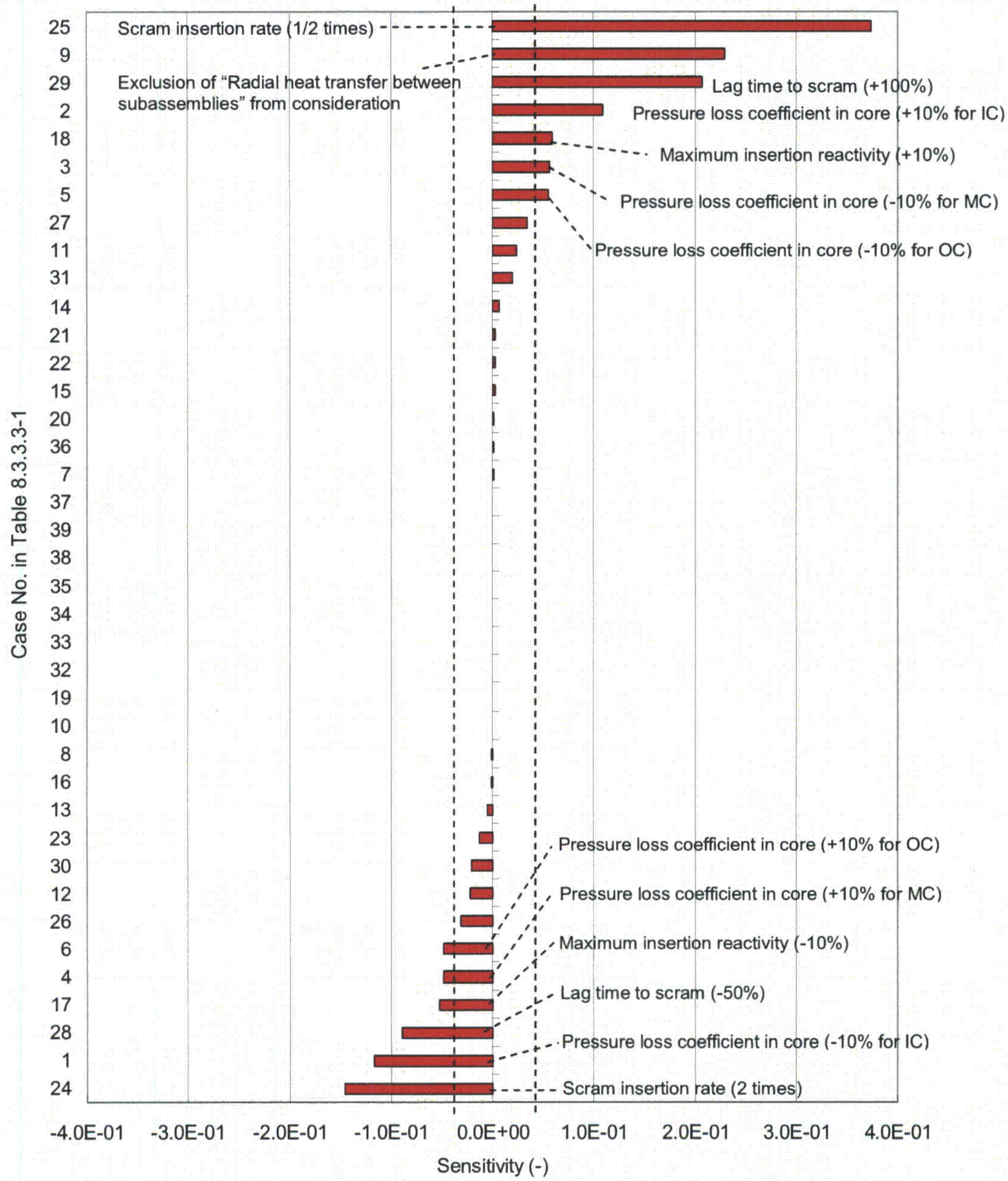


Figure 8.3.3.3-1. Summary of Sensitivity Analysis in FCC (in order of sensitivity)

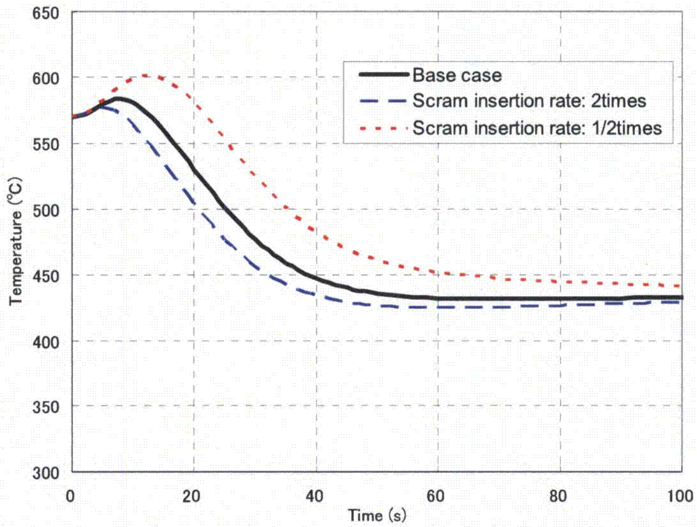


Figure 8.3.3.3-2. Example Result of Sensitivity Analysis in FCC

8.4 FINAL RANKING RESULTS

In this section, the importance rankings of the phenomena are refined, considering the sensitivity analysis results.

As described previously, the results of the sensitivity analysis are not the only element that determines the importance ranking of phenomena. The results of the sensitivity analysis are one reference used by TOSHIBA, CRIEPI, and the Japanese members of the IRAP to investigate the relative importance of phenomena.

8.4.1 Loss of Offsite Power (LOSP)

Table 8.4.1-1 shows the ranking results for the LOSP event. This table contains not only the ranking results but also an explanation of the results with rationales.

Importance rankings of 22 phenomena were changed from the initial ranking results as shown in Section 8.2. Among these 22 phenomena, three importance rankings increased and 19 decreased.

As a result of sensitivity analysis and expert opinion, Table 8.4.1-2 shows the phenomena with revised rankings.

Table 8.4.1-1. Final PIRT Results for LOSP

	Event: Loss of Offsite Power (LOSP)								
	Figures of Merit (FoM): Cladding Temperature		Importance						
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	A	Core and Fuel Assemblies						
1	-	Pressure loss in core region	M	L	P	– Pressure loss in core region accounts for about 70% at normal operation and 90% in natural circulation of the whole primary system. Therefore, the coolant flow rate change corresponding to the pressure loss change has a relatively large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Data of flow rate regarding pressure loss coefficient of core region at around rated operation were obtained by test [8-4] [8-5] [8-6] [8-7]. In fact, data for 4S design are also obtained [8-60] [8-61]. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a01
2	-	Pressure loss in reflector region	L	L	P	– Since coolant flow rate in the reflector region accounts for only 2% of the whole primary system, the coolant flow rate change corresponding to the pressure loss change does not affect cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss of inlet orifice part and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a02
3	-	Natural convection	L	M	P	– In the 1st phase, since there is still an effect of forced circulation, pressure loss that affects cladding temperature is not generated by natural convection. – In the 2nd phase, since core coolant flow rate is low, there is a potential to cause pressure loss that is likely to affect core flow rate by local natural convection.	– Flow behavior in and around the core at normal operation can be evaluated analytically. – In the natural circulation phase, it is difficult to estimate the flow velocity distribution and temperature distribution in and around the core. – Behavior of the local eddy generated in and around the core at the time of natural circulation and pressure loss attributed to the eddy can be estimated analytically. However, there are limited data to verify the results.	a03

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
4	-	Reactivity feedback	L	N/A	P	– In the LOSP event, subcritical reactivity can be maintained by reactor scram. Therefore, reactivity feedback does not affect cladding temperature.	– As a result of criticality test, analysis code is sufficiently verified by the organized nuclear data. Therefore, reactivity feedback can be estimated. – There is insufficient knowledge of shape variation caused by temperature change at the transition. Therefore, evaluated value of reactivity feedback attributed to shape variation is uncertain.	a04
5	-	Gap conductance between fuel and cladding	L	L	K	– Compared to a reactor using oxide fuel, the gap between the cladding and fuel slug is filled with sodium having high thermal conductivity that corresponds to those of the fuel and cladding. Therefore, in the gap between cladding and fuel slug, gap conductance will not cause a large difference in temperature that affects cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Since the uncertainty of thermal conductivity for fuel, sodium, and cladding, which affects the evaluation of gap conductance, is small, uncertainty of gap conductance is also small.	a05
6	-	Heat transfer between cladding and coolant	L	L	K	– Heat transfer coefficient (Nu number) is a function of Pe number (=Re·Pr), and effect of thermal conduction is dominant at Pe<100. Also, in the 4S, Pe number is about 100 even at normal operation. Since the heat transfer between cladding and coolant is controlled by thermal conductivity, which is a physical property not affected by coolant flow rate, sensitivity is small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Since the correlation equation of Nu number between cladding and coolant is investigated, there is sufficient knowledge for many fuel pins, such as those of the FFTF and CRBR [8-7] [8-9]. – For the 4S, a modified Lyon's correlation is used.	a06

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
7	-	Intra- and inter-assembly flow distribution	H	L	P	– If the flow distribution among the assemblies deviates from the design specification, the coolant flow rate in each assembly will change. This has a direct effect on the cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Flow distribution among assemblies at normal operation can be evaluated by pressure loss coefficient of each assembly [8-7] [8-8]. – Data of the pressure loss coefficient of core region can be obtained by test. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a07
8	-	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	M	P	– By the effect of radial heat transfer, heat transfers from the high-temperature core region to low-temperature peripheral part. – As stated in the a06, in the heat transfer between core structural material and coolant, thermal conduction is dominant. Also, since liquid metal sodium is used as coolant, it has a great effect on the cladding temperature. Therefore, the effect on cladding temperature by the presence of radial heat transfer is not small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There are sufficient data for physical properties of fuel assembly material and sodium [8-7] [8-8] [8-9] [8-10]. – However, since this phenomenon at natural circulation relates to phenomena such as "natural convection" in a03, it has an uncertainty.	a08
9	-	Heat transfer between reflector and coolant	L	L	P	– Since flow rate near the reflector is very low, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a09

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
10	-	Heat capacity of core assemblies	L	L	K	– Since evaluation error of specific heat and density is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	a10
11	-	Coolant boiling	N/A	N/A	K	– Peak value of cladding temperature in the LOSP event (less than 650°C) is lower than boiling point (1,000°C ~ 1,100°C). Therefore, coolant boiling has no effect on cladding temperature.	– Cover gas pressure and hydraulic head pressure from coolant surface in the reactor vessel to the core can be estimated by design. – There are sufficient data of the vapor pressure curve for liquid metal sodium coolant [8-15] [8-16] [8-17] [8-18].	a11
12	-	Core power	L	L	K	– In the LOSP event, power falls by scram right after the event starts. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Since nuclear data are well-organized, power can be calculated [8-65] [8-66].	a12
13	-	Decay heat	L	L	K	– 1st phase: Contribution to power by decay heat is small. Therefore, it has a small effect on cladding temperature. – 2nd phase: Compared to the amount of heat generation by decay heat, the coolant and structural material have larger heat capacity and remove more heat. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Nuclear data are well-organized [8-67]. – Since decay data of FPs are also well-organized, decay heat can be calculated.	a13

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
14	-	Heat transfer between core support plate and sodium	L	L	P	– Heat transfer coefficient between core support plate and coolant is large and temperature difference between core support plate and coolant is small. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – In low Re number phases such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	a14
15	-	Rate of scram reactivity insertion	L	L	K	– Since overall power is small, change in power caused by the reactivity change is also small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Reactivity distribution can be estimated by the three-dimensional calculation with the verified calculation code [8-68].	a15
16	-	Delay of scram reactivity insertion	L	L	K	– The description for a15 can be applicable for the change in power by the delay in reactivity insertion. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Same as a15.	a16
17	-	Eutectic reaction between fuel and cladding	L	L	P	– In this transition event, peak fuel peak temperature and initial fuel temperature are almost same and the power density is small. Therefore, the progress of the liquefied phase formation can be neglected and it has a small effect on cladding temperature.	– There are data from existing reactors regarding temperature at the start of eutectic reaction and growth rate of eutectic reaction [1-2] [8-2] [8-19]. – Data have been obtained by the heating test using a fuel pin after irradiation. – Burnup composition of fuel affects start temperature of eutectic reaction and growth rate of eutectic reaction, but there are limited data for the eutectic reaction start temperature.	a17

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
18	-	Temperature dependence of physical properties of materials	L	L	K	– Since the error of the physical property related to the core region is small, it has a small effect on cladding temperature.	– There are sufficient data for physical properties of component materials in the core [8-20] [8-21] [8-22] [8-23] [8-24].	a18
19	-	FP release from fuel slug into gas plenum	N/A	N/A	K	– Behavior of FP gas in the fuel pin does not affect cladding temperature.	– There are sufficient data for gas evolution rate in the fuel pin [1-2] [8-7].	a19
20	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	P	– Behavior of FP gas in the fuel pin does not affect cladding temperature.	– There is sufficient knowledge of nuclides resolved in sodium [1-3] [8-26] [8-27] [8-28] [8-29]. – Depending on the nuclide, the uncertainty of solubility data is large.	a20
21	-	FP transport from sodium in primary system to cover gas	N/A	N/A	P	– In this event, since FP gas does not transport to cover gas, this event does not affect cladding temperature.	– There is sufficient knowledge of nuclides resolved in sodium [1-3] [8-26] [8-27] [8-28] [8-29]. – Depending on the nuclide, the uncertainty of solubility data is large.	a21
22	-	Flow-induced vibration in a subassembly	L	L	P	– The vibration of structural material attributed to flow-induced vibration has no effect on cladding temperature.	– The evaluation of the flow-induced vibration has been generalized by existing test data using water as a fluid. – However, there are limited test data using sodium [8-62] [8-63]. Since data such as the dimensions and supporting structure of the 4S differ from those of existing sodium reactors, it is difficult to apply the test data of sodium to the 4S.	a22

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
23	-	Inter-wrapper flow between wrapper tubes	L	M	P	– Since the flow rate among assemblies is small, inter-wrapper flow has a small effect on the heat transfer between wrapper tube wall and coolant. Therefore, at normal operation, inter-wrapper flow has little effect on cladding temperature. – In the natural circulation phase, if pressure loss as stated in a03 is generated, it may affect cladding temperature. – The effect becomes pronounced when using direct reactor auxiliary cooling system (DRACS).	– In the natural circulation phase, there are test data of sodium flow behavior in gaps between fuel assemblies and their flow resistance to the part [8-64]. – However, the data are highly relative to test scale and geometry, and there is insufficient knowledge regarding scaling.	a23
24	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	– Flow distribution change of coolant has a direct effect on cladding temperature.	– Knowledge of natural circulation behavior contains uncertainty [8-6] [8-7] [8-8] [8-11] [8-12] [8-13] [8-17] [8-18] [8-30] [8-31] [8-32] [8-33] [8-34] [8-35] [8-36] [8-37] [8-38] [8-39] [8-40] [8-41] [8-42] [8-43] [8-44] [8-45] [8-56] [8-47]. With very low Re numbers, there is little knowledge for flow distribution of heat removal. – It is difficult to quantitatively estimate the pressure loss and heat transfer coefficient in the natural circulation phase.	a24
25	-	Radial power distribution	L	L	K	– Radial power distribution affects cladding temperature. Time for transitional power distribution change is short, so transitional effect is small.	– Much power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.	a25

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
26	-	Axial power distribution	L	L	K	– Axial power distribution affects cladding temperature. Time for transitional power distribution change is short and transitional effect is small. – Accident analyses were performed with some cases of axial power distribution, considering the condition of burn-up core (i.e. beginning of life and end of life). As a result, the influence of the axial power distribution change due to the difference in the burn-up core was small against FOM.	– Much power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.	a26
	B	Reactor System						
	B1	Reactor Vessel						
27	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	K	– Wall surface temperature of the vessel varies with the fluctuation of coolant level in the reactor vessel, but coolant temperature change in the upper plenum varies with the change of vessel wall surface temperature and can be neglected.	– Fluctuation of coolant level in the reactor vessel and axial temperature distribution of reactor vessel wall caused by that fluctuation can be evaluated analytically.	b01

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
	B2	Reactor Internal Structures						
	B20	General						
28	-	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	– In the 2nd phase, mixing effect of coolant for upper plenum depends on how much the coolant of upper side in the upper plenum contributes to the circulation of coolant in the primary system. As a result, it has a large effect on cladding temperature. – Since there is still an effect of forced circulation in the 1st phase, the mixing effect of coolant for upper plenum is strong. – For the 2nd phase, since Re number of the coolant in the upper plenum is about 10,000, the mixing effect is smaller than in 1st phase. Also, since low-temperature coolant flows from the core region into the upper plenum where the high-temperature coolant remains, there is a possibility of causing thermal stratification, which may affect natural circulation in the reactor vessel. – Therefore, mixing effect of coolant has a large effect on cladding temperature. See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There are sufficient data for flow tests using water, but it is difficult to apply these results to a system using sodium. – For low Re number phases, such as natural circulation, it is difficult to measure the mixing effect, and data contain uncertainty. – There is insufficient knowledge of thermal stratification in the natural circulation phase [8-48] [8-58] [8-59].	b02
29	-	Temperature dependence of physical properties of structural materials	L	L	K	– Errors in physical properties related to the core are small. Therefore, they have a small effect on cladding temperature.	– There are sufficient data for physical properties of the reactor internal structures.	b03

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
30	-	Natural convection	L	M	P	– Since there is still forced circulation in the 1st phase, natural convection does not affect cladding temperature. – In the 2nd phase, since coolant flow rate around the reactor internal structure is low, there is a possibility of causing pressure loss that affects the flow rate of primary coolant by local natural convection generated around the reactor internal structure. Therefore, this affects cladding temperature.	– Flow behavior around the reactor internal structure at normal operation can be evaluated analytically. – In the natural circulation phase, it is difficult to analytically estimate the flow velocity distribution and temperature distribution around the reactor internal structure. – Behavior of the local eddies generated around the reactor internal structure in natural circulation, and pressure loss attributed to them, can be evaluated analytically. However, there are limited data to verify the results, and uncertainty is large.	b04
31	-	Flow-induced vibration	N/A	N/A	P	– Flow-induced vibration does not affect cladding temperature.	– It can be estimated analytically. – This phenomenon shows different behavior (or result) depending on geometry of facility. However, since there are no test data established for the same geometry, there is insufficient knowledge.	b05

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	B21 Reflector							
32	-	Deformation due to thermal effect and irradiation	L	L	P	– The coolant flow rate in the reflector region accounts for only 2% of that of the whole primary system. – Therefore, even if deformation of the reflector or cavity is generated, it has little effect on cladding temperature.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Since analysis result of flow behavior at the transition has not been verified yet, it is difficult to estimate the temperature distribution. – There are limited data for the change in strength structural material affected by irradiation.	b06
33	-	Flow in reflector region	L	L	P	– The coolant flow rate in the reflector region accounts for only 2% of that of the whole primary system. Therefore, flow rate change in this area has little effect on cladding temperature.	– For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss and data contain uncertainty.	b07
34	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	– Compared to decay heat or rated power, heat generation by neutron capture and gamma rays has a small effect on cladding temperature and can be neglected.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Analysis result of flow behavior at the transition has not been verified yet. Hence, it is difficult to analytically estimate the temperature distribution at transient.	b08
	B22 Lower Plenum							
35	-	Pressure loss	L	L	P	– Pressure loss in the lower plenum region accounts for only 10% of that of the whole primary system. Therefore, the coolant flow rate change corresponding to the pressure loss change has little effect on cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	b09

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
36	-	Heat capacity	L	L	K	– Evaluation error of specific heat and density is small, and heat capacity of lower plenum accounts for only certain percentage of heat capacity of the whole primary system. Therefore, it has a small effect on cladding temperature.	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b10
37	-	Coolant mixing and thermal stratification	L	L	P	– Heat capacity of the coolant of lower plenum accounts for only certain percentage of heat capacity of the whole primary system. – Therefore, mixing effect of coolant of lower plenum has a small effect on cladding temperature.	– Although there are sufficient flow test data using water, it is difficult to scale the result of a water test to a sodium system. – In low Re number phases such as natural circulation, it is difficult to measure the mixing effect and data contain uncertainty. – There is insufficient knowledge of thermal stratification during natural circulation phase [8-58] [8-59].	b11
38	-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	L	P	– Heat transfer from lower head of reactor vessel to the outside is mainly by radiation, but the heat transfer area accounts for only a certain percentage of the RV wall. Therefore, it has a small effect on cladding temperature.	– There are limited data related to the change of radiation factor by age deterioration. – There is little knowledge of convective heat transfer for spheres.	b12

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
	B23 Upper Plenum							
39	-	Pressure loss	L	L	P	– Compared to the pressure loss of the overall primary system, the pressure loss of most of upper plenum region is 0. – Therefore, the coolant flow rate change corresponding to the pressure loss change has a small effect on cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	b13
40	-	Heat capacity	L	L	K	– Evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b14
41	-	Heat transfer between shielding plug and sodium	L	L	K	– In the heat transfer between shielding plug and coolant, radiation is dominant. From the aspect of the heat transfer area, the amount of heat transfer is small compared to that from RV wall surface to outside. Therefore, it has a small effect on cladding temperature.	– Amount of heat transfer by radiation governing heat transfer between shielding plug and coolant level can be estimated analytically [8-9] [8-10].	b15
42	-	Coolant mixing, thermal stratification, and thermal striping	L	L	P	– Since the length of upper plenum is long, radial distribution of coolant temperature becomes uniform regardless of the degree of mixing effect in the outlet of the core region. Therefore, it has a small effect on cladding temperature.	– There are limited experimental data for thermal striping [8-57]. – Uncertainty of analytical prediction method is large, since it strongly depends on geometry.	b16

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
	B24	Vertical Shroud						
43	-	Radial heat transfer between inside and outside coolant through vertical shroud	L	L	P	– Since vertical shroud is equipped with heat insulation material, heat transfer from the inside to the outside can mostly be neglected. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	b17
	B25	Radial Shield						
44	-	Flow in radial shield region	L	L	P	– Pressure loss in radial shield region accounts for only 8% of that of the whole primary system. – Therefore, the coolant flow rate change corresponding to the pressure loss change has little effect on cladding temperature.	– For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss, and data contain uncertainty.	b18
45	-	Heat capacity	L	L	K	– Ratio of the heat capacity of the radial shield to total heat capacity of primary system is large. However, evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b19
46	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	– Compared to decay heat or rated power, heat generation by neutron capture and gamma rays has a small effect on cladding temperature and can be neglected.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Analysis result of flow behavior at the transition has not been verified yet. Therefore, it is difficult to analytically estimate the temperature distribution.	b20

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
47	-	Radial heat transfer in radial shield region	L	L	P	– Since the heat capacity is large, radial shield has the role of a heat sink. – In LOSP event, the ratio of the heat removal by radial shield to that by the whole system is small since heat is removed by IRACS functions. – Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	b21
	B26	Reactivity Control Drive Mechanism						
48	-	Shutdown velocity of reflector	L	L	K	– Since shutdown velocity of the reflector is specifically designed and the error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Design reflector descent time is 8s.	b22
49	-	Shutdown velocity of shutdown rod	L	L	K	– Since shutdown velocity of the shutdown rod is specifically designed, and the error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Design shutdown rod descent time is 8s.	b23

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
	C	Primary Heat Transport System						
	C0	General						
50	-	Natural circulation	L	M	P	– In the 1st phase, since there is still an effect of forced circulation by flow coastdown system, natural circulation has a small effect on cladding temperature. – In the 2nd phase, natural circulation flow rate in the primary system is the dominant factor that determines amount of heat removal, and hence effect on cladding temperature. – Also, the amount of natural circulation flow rate is greatly affected by the geometric shape of flow path and pressure loss. – As mentioned in a03 and b04, it has a large effect on cladding temperature at core region, because there is still uncertainty in the pressure loss.	– Natural circulation flow rate can be evaluated analytically. – It is difficult to evaluate the pressure loss attributed to local eddies around the core and each reactor internal structure in natural circulation phase. – Since natural circulation testing has been conducted many times in the past [8-6] [8-7] [8-8] [8-11] [8-12] [8-13] [8-17] [8-18] [8-30] [8-31] [8-32] [8-33] [8-34] [8-35] [8-36] [8-37] [8-38] [8-39] [8-40] [8-41] [8-42] [8-43] [8-44] [8-45] [8-46] [8-47], there is sufficient knowledge, but natural circulation has strong dependence on geometric shape, so application of measured data for 4S contains uncertainty.	c01
51	-	Sodium inventory	L	L	K	– Since an amount of sodium is specifically designed and its error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– An amount of sodium can be evaluated by design requirement [8-9] [8-15] [8-16].	c02
52	-	Heat capacity of coolant	L	L	K	– Error of the specific heat and density evaluation for coolant is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Amount of sodium can be evaluated by design requirement. – There are sufficient data for specific heat [8-9] [8-15] [8-16].	c03

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss Of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	C1	IHX						
53	-	Pressure loss	L	L	P	– Pressure loss in IHX region accounts for only a certain percentage of all the intermediate system. Therefore, the coolant flow rate change corresponding to the pressure loss change has a small effect on cladding temperature.	– Based on the established knowledge [8-7] [8-9], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	c04
54	-	Heat transfer from primary coolant to intermediate coolant	L	L	P	– It takes about 400s for primary system coolant to go around the flow path at normal operation. Therefore, a change in heat transfer at the IHX in the 1st phase does not affect cladding temperature. – Flow rate in the 2nd phase is small. Since its Pe (=Re·Pr) number is 10 or less and the effect by thermal conductivity is dominant in heat transfer, the heat transfer from primary system coolant to intermediate system coolant through heat transfer tube has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	c05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss Of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
55	-	Primary flow rate	L	L	P	– Since primary flow rate at normal operation is specifically designed and its error is small, it has a small effect on cladding temperature in the 1st phase. – Primary flow rate in the natural circulation phase is determined by the natural circulation head and pressure loss in flow path. – As mentioned in c05, thermal conduction is dominant in the heat transfer, which has an effect on cladding temperature. Therefore, primary flow rate in the IHX has a small effect on cladding temperature.	– There is knowledge to be able to set coolant flow rate in the primary system at normal operation in the value based on design requirement. – It is difficult to measure the primary coolant flow rate in natural circulation, and the data contain uncertainty.	c06
56	-	Intermediate flow rate	L	L	P	– Same as c06. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is knowledge to be able to set coolant flow rate in the Intermediate system at normal operation in the value based on design requirement. – It is difficult to measure the flow rate of intermediate coolant in the natural circulation phase.	c07
57	-	Heat capacity	L	L	K	– Since evaluation error of specific heat and density is low, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	c08
58	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	L	L	P	– As state in c04, since pressure loss of IHX has little effect, even if the local pressure loss caused by natural convection is generated in the annulus part on the intermediate system side at the low flow rate, the effect is small.	– Coolant in the annulus part at normal operation shows spatially uniform flow. – In natural circulation, since uncertainty for pressure loss of baffle plate placed in annulus part is large, it is difficult to evaluate the flow behavior.	c09

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	C2	Primary EMP						
59	-	Flow coastdown performance	L	L	P	– The halving time for flow coastdown of EMP is specifically designed and its error is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There are test data for EMPs but no test data for EMPs of similar size as those in the primary system [8-49].	c10
60	-	Pressure loss	L	L	P	– Pressure loss of EMP accounts for only 10% of that of the whole primary system. Therefore, the coolant flow rate change corresponding to the pressure loss change has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Based on the established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	c11
61	-	Pump head	L	L	K	– 1st phase: The halving time for EMP flow coastdown is specifically designed and its error is small. Therefore, it has a small effect on cladding temperature. – 2nd phase: It has a small effect on cladding temperature since the pump is not running.	– There is knowledge to be able to set pump head at normal operation in the value based on design requirement.	c12
62	-	Heat capacity and joule heat at flow coastdown	L	N/A	P	– Evaluation error of specific heat and density is small. – Also, amount of heat generation by joule heat is less than that of decay heat. – Therefore, it has a small effect on cladding temperature.	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat data. – Since there is uncertainty in the flow coastdown performance as mentioned in c10, there is also uncertainty in joule heat evaluation.	c13

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)								
	Figures of Merit (FoM): Cladding Temperature		Importance						
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	D	Intermediate Heat Transport System						
	D0	General						
63	-	Pressure loss	L	L	P	– 1st phase: As stated in c05, flow rate change caused by pressure loss has a small effect on cladding temperature. – 2nd phase: As stated in c05, thermal conduction is dominant in the heat transfer from primary system coolant to intermediate system through heat transfer tube. – Therefore, flow rate change caused by pressure loss change in intermediate system has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	d01

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP) — Figures of Merit (FoM): Cladding Temperature							
64	-	Natural circulation	L	M	P	– Since there is still an effect of forced circulation in the 1st phase, natural circulation has a small effect on cladding temperature. – In the 2nd phase, flow rate of natural circulation in the intermediate system is the dominant factor that indirectly determines cladding temperature. – Flow rate of natural circulation in the intermediate system is defined by the natural circulation head and pressure loss of flow path, which are defined by the performance of the IHX (heat source) and AC (heat sink). – As stated in d01, pressure loss does not affect cladding temperature. However, as stated in heat transfer for IRACS (see e03), in the LOSP event, performance of heat removal by air cooler has some effect on cladding temperature. Therefore, the effect on cladding temperature by natural circulation is not small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Natural circulation flow rate can be evaluated analytically. – It is difficult to measure the pressure loss attributed to the local eddies generated in and around each structure in the intermediate system during natural circulation. – Natural circulation testing has been conducted many times [8-6] [8-7] [8-8] [8-11] [8-12] [8-13] [8-17] [8-18] [8-30] [8-31] [8-32] [8-33] [8-34] [8-35] [8-36] [8-37] [8-38] [8-39] [8-40] [8-41] [8-42] [8-43] [8-44] [8-45] [8-46] [8-47], so there is sufficient knowledge. Since natural circulation has strong dependence on geometric shape, however, application of measured data to 4S contains uncertainty.	d02
65	-	Heat removal from SG	L	N/A	K	– In the event of LOSP, water-steam system does not remove heat. Therefore, it has no effect on cladding temperature.	– Amount of heat removal by SG can be evaluated by design requirement.	d03

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
66	-	Heat transfer between upper plenum and intermediate coolant external to IHX	L	L	P	– Since the amount of coolant contained in this part is small, it has a small effect on cladding temperature.	– There is sufficient knowledge of about heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity and data contain uncertainty.	d04
	D1	Intermediate EMP						
67	-	Flow coastdown performance	L	L	K	– Same as c10. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There are test data for EMPs of the same size as those in the 4S intermediate system [1-2] [8-49].	d05
68	-	Pressure loss	L	L	P	– Same as d01.	– Based on the established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	d06
69	-	Pump head	L	L	K	– Same as c12. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is knowledge to be able to set pump head at normal operation in the value based on design requirement.	d07
	D2	Steam Generator System						
70	-	Heat capacity of structure, sodium, water, and steam	L	L	K	– In the water-steam system, since water is drained immediately after shutdown, heat capacity of water and steam has a small effect on cladding temperature.	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	d08

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	E	Residual Heat Removal System						
	E1	Air cooler of IRACS						
71	-	Pressure loss of sodium side	L	L	P	– 1st phase: Same as the following e02. – 2nd phase: In the 2nd phase, the amount of heat removal by the AC is dependent on thermal conductivity of air, which is smaller than sodium. As mentioned in e02, since the effect of pressure loss of the air side is also small, the effect by pressure loss of the flow pass on the sodium side is small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	e01

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
72	-	Pressure loss of air side	L	L	P	– It takes 280s for intermediate system coolant to travel the flow path. In the short time of the 1st phase, the effect of heat removal by the AC does not reach the IHX that exchanges heat with the primary system. Therefore, it has only a small effect on cladding temperature. – 2nd phase: As mentioned in e01, the amount of heat removal by the AC depends on thermal conductivity of air. As a result of the sensitivity analysis, cladding temperature shows a small change (by 1.6°C) for uncertainty (10%) of pressure loss assumed in the 4S PIRT. – Therefore, pressure loss on the air side has only a small effect on cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – Pressure loss of AC may depend on the status (open/closed) of the damper and clogging of filter placed on the air inlet, but there is insufficient knowledge.	e02
73	-	Heat transfer between tube and air	L	M	P	– Same as e02. – 2nd phase: As mentioned in e01, amount of AC heat removal depends on its performance on the air side. – As a result of the sensitivity analysis, cladding temperature shows the change by 6–7°C, which cannot be neglected for uncertainty (20%) of heat transfer assumed in the 4S PIRT. Therefore, the effect of the pressure loss in the air side on cladding temperature is not small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is sufficient knowledge of the heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – As stated in e02, since there is insufficient knowledge of the status (open/closed) of damper, the uncertainty of the estimated flow velocity is not small.	e03

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
74	-	Heat transfer between tube and sodium	L	L	P	– Same as e01. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is sufficient knowledge of the heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – In the low Re number region such as natural circulation, it is difficult to measure the coolant flow velocity, and the data contain uncertainty.	e04
75	-	Inlet air temperature range	L	M	K	– Temperature difference is one of the factors that has a large effect on the amount of heat removal by the air side of AC. Therefore, the effect on cladding temperature is not small. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– The highest and lowest temperatures of all the states in U.S. are shown [8-3].	e05
76	-	Heat capacity	L	L	K	– Evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	e06
	E2 RVACS							
77	-	Pressure loss in air flow path	L	L	K	– Most of heat removal in the LOSP is achieved by IRACS. (Ratio of heat removal amount for IRACS and RVACS is about 2:1.) Therefore, the heat removal behavior in RVACS has little effect on cladding temperature.	– Based on established knowledge, form loss and friction loss can be evaluated [8-8].	e07

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
78	-	Heat transfer between GV wall and air	L	L	K	– In LOSP, the effect of heat removal in RVACS is small, because IRACS is active. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-10] [8-50] [8-51] [8-52] [8-53] [8-54] [8-55].	e08
79	-	Heat transfer between collector wall and air	L	L	K	– Same as e08.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-10] [8-50] [8-51] [8-52] [8-53] [8-54] [8-55].	e09
80	-	Heat transfer between concrete wall and air	L	L	K	– Same as e08.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-10] [8-50] [8-51] [8-52] [8-53] [8-54] [8-55].	e10
81	-	Thermal radiation between RV wall and GV wall	L	L	P	– Same as e08.	– Amount of heat transfer by radiation can be evaluated analytically. – There are sufficient data for radiation factor [8-10] [8-55] [8-56]. – There are limited data related to the change of radiation factor by age deterioration.	e11
82	-	Thermal radiation between GV wall and heat collector wall	L	L	P	– Same as e08.	– Amount of heat transfer by radiation can be evaluated analytically. – There are limited data related to the change of radiation factor by age deterioration.	e12
83	-	Thermal radiation between heat collector wall and concrete wall	L	L	P	– Same as e08.	– Amount of heat transfer by radiation can be evaluated analytically. – There are limited data related to the change of radiation factor by age deterioration.	e13

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-1. Final PIRT Results for LOSP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature							
84	-	Asymmetric airflow	L	L	P	– Same as e07.	– In the natural circulation phase, nonuniform flow distribution in the IHX annulus piping region has a potential to show asymmetric behavior. – There is a possibility that such asymmetric flow would produce asymmetric diversity to air current in the RVACS, but there is insufficient knowledge and data contain uncertainty.	e14
85	-	Inlet air temperature range	L	L	K	– Same as e07.	– The highest and lowest temperatures of all the states in U.S. are shown [8-3].	e15
	F	Instrumentation and Control System						
	F1	Instrumentation and Control Equipment						
	F11	Plant Protection Sensors						
86	-	Delay of scram signal of primary EMP voltage and current	L	L	K	– Scram signal of primary EMP voltage and current is specifically designed, and its error is small. Therefore, it has a small effect on cladding temperature.	– Response time is specifically set according to the design requirement.	f01
87	-	Delay of scram signal of power line voltage	L	L	K	– Scram signal of low power line voltage is specifically designed, and its error is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.1-1).	– Same as f01.	f02
88	-	Delay of scram signal of neutron flux	L	L	K	– Same as f02.	– Same as f01.	f03
89	-	Delay of scram signal of IHX primary outlet temperature	L	L	K	– Same as f02.	– Same as f01.	f04
	F12	Others						
90	-	Delay of interlock signal of SG outlet temperature	L	L	K	– Same as f02.	– Same as f01.	f05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-2. Phenomena with Difference Between Initial and Final Ranking Results for LOSP

	Event: Loss of Offsite Power (LOSP)			Change from Initial Ranking	
	Figures of Merit (FoM): Cladding Temperature			Importance	
No.	Subsystem/Component		Phenomenon	1st Phase	2nd Phase
-	A	Core and Fuel Assemblies			
1	-		Pressure loss in core region	L → M	L
7	-		Intra- and inter-assembly flow distribution	H	M → L
10	-		Heat capacity of core assemblies	M→L	M → L
12	-		Core power	M → L	M → L
13	-		Decay heat	M → L	M → L
15	-		Rate of scram reactivity insertion	H → L	L
16	-		Delay of scram reactivity insertion	H → L	L
18	-		Temperature dependence of physical properties of materials	M → L	M → L
23	-		Inter-wrapper flow between wrapper tubes	L	L → M
-	B	Reactor System			
-	B2	Reactor Internal Structures			
-	B20	General			
28	-		Coolant mixing effect in upper plenum including thermal stratification	M → L	M
31	-		Flow-induced vibration	L → N/A	L → N/A
-	B23	Upper Plenum			
40	-		Heat capacity	L	M → L
-	B26	Reactivity Control Drive Mechanism			
48	-		Shutdown velocity of reflector	M → L	L
49	-		Shutdown velocity of shutdown rod	M → L	L
-	C	Primary Heat Transport System			
-	C0	General			
51	-		Sodium inventory	L	M → L
-	C2	Primary EMP			
59	-		Flow coastdown performance	M → L	L
62	-		Heat capacity and joule heat at flow coastdown	L	L → N/A

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.1-2. Phenomena with Difference between Initial and Final Ranking Results for LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)			Change from Initial Ranking	
	Figures of Merit (FoM): Cladding Temperature			Importance	
No.	Subsystem/Component		Phenomenon	1st Phase	2nd Phase
-	D	Intermediate Heat Transport System			
-	D0	General			
64	-		Natural circulation	L	L → M
-	D1	Intermediate EMP			
67	-		Flow coastdown performance	M → L	L
-	E	Residual Heat Removal System			
-	E1	Air cooler of IRACS			
72	-		Pressure loss of air side	L	M → L
-	F	Instrumentation and Control System			
-	F1	Instrumentation and Control Equipment			
-	F11	Plant Protection Sensors			
86	-		Delay of scram signal of primary EMP voltage and current	M → L	L
87	-		Delay of scram signal of power line voltage	M → L	L

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

8.4.2 Sodium Leakage from Intermediate Piping (SLIP)

Table 8.4.2-1 shows the ranking results for the SLIP event. This table contains not only the ranking results but also explanations for the results, with rationales.

There were 24 phenomena that changed their importance rankings from the initial ranking results in Section 8.2. Two phenomena increased its ranking, while 22 phenomena decreased.

Table 8.4.2-2 shows the phenomena with revised rankings.

Table 8.4.2-1. Final PIRT Results for SLIP

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	A Core and Fuel Assemblies							
1	-	Pressure loss in core region	M	L	P	– Pressure loss in core region accounts for about 70% at normal operation and 90% in natural circulation of the whole primary system. Therefore, coolant flow rate change corresponding to the pressure loss change has a relatively large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Data of flow rate regarding pressure loss coefficient of core region at around rated operation were obtained by test [8-4] [8-5] [8-6] [8-7]. In fact, data for 4S design are also obtained [8-60] [8-61]. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a01
2	-	Pressure loss in reflector region	L	L	P	– Since coolant flow rate in the reflector region accounts for only 2% of the whole primary system, the coolant flow rate change corresponding to the pressure loss change does not affect cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss of inlet orifice part and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	a02

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
3	-	Natural convection	L	M	P	– In the 1st phase, since there is still an effect of forced circulation, pressure loss that affects cladding temperature by natural convection is not generated. – In the 2nd phase, since the core coolant flow rate is low, there is a possibility of pressure loss, which is likely to affect the core flow rate by local natural convection.	– Flow behavior in and around the core at normal operation can be evaluated analytically. – In the natural circulation phase, it is difficult to estimate the flow velocity distribution and temperature distribution in and around the core. – Behavior of the local eddy generated in and around the core at natural circulation and pressure loss attributed to the eddy can be estimated analytically. However, there are limited data to verify the results.	a03
4	-	Reactivity feedback	L	N/A	P	– In the SLIP event, subcritical reactivity can be maintained by reactor scram. Therefore, reactivity feedback does not affect cladding temperature.	– As a result of criticality test, analysis code is sufficiently verified by the organized nuclear data. Therefore, reactivity feedback can be estimated. – There is insufficient knowledge of shape variation caused by temperature change at the transition. Therefore, evaluated value of reactivity feedback attributed to shape variation is uncertain.	a04

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
5	-	Gap conductance between fuel and cladding	L	L	K	– Compared to a reactor using oxide fuel, the gap between the cladding and fuel slug is filled with sodium having high thermal conductivity corresponding to those of the fuel and cladding. Therefore, in the gap between cladding and fuel slug, gap conductance will not cause a large difference in temperature that affects cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Since the uncertainty of thermal conductivity for fuel, sodium, and cladding, which affects the gap conductance evaluation, is small, uncertainty of gap conductance is also small.	a05
6	-	Heat transfer between cladding and coolant	L	L	K	– Heat transfer coefficient (Nu number) is the function of Pe number (=Re·Pr) and effect of the thermal conduction is dominant at Pe<100. Also, in the 4S, Pe number is about 100 even at normal operation. Since heat transfer between cladding and coolant is controlled by thermal conductivity, which is a physical property not affected by coolant flow rate, the sensitivity is small. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Since the correlation equation of Nu number between cladding and coolant is investigated, there is sufficient knowledge for many fuel pins, such as those of the FFTF and CRBR [8-7] [8-9]. – For the 4S, a modified Lyon's correlation is used.	a06

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
7	-	Intra- and inter-assembly flow distribution	H	L	P	– If the flow distribution among the assemblies deviates from the design specification, coolant flow rate in each assembly will change. This has a direct effect on the fuel pin temperature, and so doest the cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Flow distribution among assemblies at normal operation among the assemblies can be evaluated by pressure loss coefficient of each assembly [8-7] [8-8]. – Data of the pressure loss coefficient of core region can be obtained by test. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	a07
8	-	Radial heat transfer between subassemblies (S/A <-> sodium <-> S/A)	M	M	P	– By the effect of radial heat transfer, heat transfers from the high-temperature core region to low-temperature peripheral part. – As stated in the a06, in the heat transfer between core structural material and coolant, thermal conduction is dominant. Also, since liquid metal sodium is the coolant material, its high thermal conductivity has a great effect on cladding temperature. Therefore, the effect on cladding temperature by radial heat transfer is not small. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– There are sufficient data for physical properties of fuel assembly material and sodium [8-7] [8-8] [8-9] [8-10]. – However, since this phenomenon of natural circulation relates to phenomena such as "natural convection" in a03, it has an uncertainty.	a08

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
9	-	Heat transfer between reflector and coolant	L	L	P	– Since the flow rate near the reflector is low, there is little effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	a09
10	-	Heat capacity of core assemblies	L	L	K	– Since evaluation error of specific heat and density is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	a10
11	-	Coolant boiling	N/A	N/A	K	– Peak value of cladding temperature in the SLIP event (less than 650°C) is lower than boiling point (1,000°C ~ 1,100°C). Therefore, coolant boiling has no effect on cladding temperature.	– Cover gas pressure and hydraulic head pressure from coolant surface in the reactor vessel to the core can be estimated by design requirement. – There are sufficient data of the vapor pressure curve for liquid metal sodium coolant [8-15] [8-16] [8-17] [8-18].	a11
12	-	Core power	L	L	K	– In the SLIP event, power falls off by scram right after the event starts. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Since nuclear data are well-organized, power can be estimated by calculation [8-65] [8-66].	a12

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
13	-	Decay heat	L	L	K	– 1st phase: Contribution to power by decay heat is small. Therefore, it has a small effect on cladding temperature. – 2nd phase: Compared to the amount of heat generation by decay heat, value of system heat removal, coolant, and structural material have larger heat capacity. Therefore, decay heat has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Nuclear data are well-organized [8-67]. – Since decay data of FPs are also well-organized, decay heat can be estimated by calculation.	a13
14	-	Heat transfer between core support plate and sodium	L	L	P	– Heat transfer coefficient between core support plate and coolant is large and temperature difference between core support plate and coolant is small. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – In low Re number phases such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	a14
15	-	Rate of scram reactivity insertion	L	L	K	– Since the power itself is small, change in power caused by the reactivity change is also small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Reactivity distribution can be estimated by the three-dimensional calculation with the verified calculation code [8-68].	a15
16	-	Delay of scram reactivity insertion	L	L	K	– The description for a15 can be applicable for the change in power by the delay in reactivity insertion. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Same as a15.	a16

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	**Subsystem/ Component**	**Phenomenon**	**1st Phase**	**2nd Phase**	**SoK**	**Ranking Rationale for Importance**	**Ranking Rationale for SoK**	**Code**
17	-	Eutectic reaction between fuel and cladding	L	L	P	– In this transition event, peak fuel temperature and initial fuel temperature are almost the same and power density is small. Therefore, since the progress of the liquefied phase formation can be neglected, it has a small effect on cladding temperature.	– There are data from existing reactors regarding temperature at the start of eutectic reaction and growth rate of eutectic reaction [1-2] [8-2] [8-19]. – Data have been obtained by the heating test using a fuel pin after irradiation. – Burnup composition of fuel affects start temperature of eutectic reaction and growth rate of eutectic reaction, but there are limited data of the eutectic reaction start temperature.	a17
18	-	Temperature dependence of physical properties of materials	L	L	K	– Since the error in physical properties related to the core region is small, it has a small effect on cladding temperature.	– There are sufficient data for physical properties of component materials in the core [8-20] [8-21] [8-22] [8-23] [8-24].	a18
19	-	FP release from fuel slug into gas plenum	N/A	N/A	K	– Behavior of FP gas in the fuel pin does not affect cladding temperature.	– There are sufficient data for gas evolution rate in the fuel pin [1-2] [8-7].	a19
20	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	P	– Behavior of FP gas in the fuel pin does not affect cladding temperature.	– There is sufficient knowledge of nuclides resolved in sodium [1-3] [8-26] [8-27] [8-28] [8-29]. – Depending on the nuclide, the uncertainty of solubility data is large.	a20

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
21	-	FP transport from sodium in primary system to cover gas	N/A	N/A	P	– In this event, since transport of FP gas to cover gas is not made, this event does not affect cladding temperature.	– There is sufficient knowledge of nuclides resolved in sodium [1-3] [8-26] [8-27] [8-28] [8-29]. – Depending on the nuclide, the uncertainty of solubility data is large.	a21
22	-	Flow-induced vibration in a subassembly	L	L	P	– The vibration of structural material attributed to the flow-induced vibration has no effect on cladding temperature.	– The evaluation of the flow-induced vibration has been generalized by existing test data using water as a fluid. – However, there are limited test data using sodium [8-62] [8-63]. Since data such as the dimensions and supporting structure of the 4S differ from those of existing sodium reactors, it is difficult to apply the test data of sodium to the 4S.	a22
23	-	Inter-wrapper flow between wrapper tubes	L	M	P	– Since the flow rate among the assemblies is small, inter-wrapper flow has a small effect on the heat transfer between wrapper tube wall and coolant. Therefore, at normal operation, inter-wrapper flow has little effect on cladding temperature. – In the natural circulation phase, if pressure loss as stated in a03 is generated, it may affect cladding temperature. – The effect becomes pronounced when using DRACS.	– In the natural circulation phase, there are test data of sodium flow behavior in gaps between fuel assemblies and their flow resistance [8-64]. – However, the data are highly relative to test scale and geometry, and there is insufficient knowledge regarding scaling.	a23

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
24	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	– Flow distribution change of coolant has direct effect on cladding temperature.	– Knowledge of natural circulation behavior [8-6] [8-7] [8-8] [8-11] [8-12] [8-13] [8-17] [8-18] [8-30] [8-31] [8-32] [8-33] [8-34] [8-35] [8-36] [8-37] [8-38] [8-39] [8-40] [8-41] [8-42] [8-43] [8-44] [8-45] [8-56] [8-47] contains uncertainty. With very low Re numbers, there is little knowledge for flow distribution of heat removal. – It is difficult to quantitatively estimate the pressure loss and heat transfer coefficient in the natural circulation phase.	a24
25	-	Radial power distribution	L	L	K	– Flow distribution error of coolant has a direct effect on cladding temperature.	– Much power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.	a25
26	-	Axial power distribution	L	L	K	– Axial power distribution affects cladding temperature. Time for transitional power distribution change is short and transitional effect is small. – Accident analyses were performed with some cases of axial power distribution, considering the condition of burn-up core (i.e. beginning of life and end of life). As a result, the influence of the axial power distribution change due to the difference in the burn-up core was small against FOM.	– Much power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.	a26

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)								
	Figures of Merit (FoM): Cladding Temperature		Importance						
No.	**Subsystem/ Component**	**Phenomenon**	**1st Phase**	**2nd Phase**	**SoK**	**Ranking Rationale for Importance**	**Ranking Rationale for SoK**	**Code**
	B	Reactor System						
	B1	Reactor Vessel						
27	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	K	– Axial power distribution affects cladding temperature to a great extent than changes in liquid level. Time for transitional power distribution change is short and transitional effect is small.	– Fluctuation of coolant level in the reactor vessel and axial temperature distribution of reactor vessel wall caused by that fluctuation can be evaluated analytically.	b01

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
	B2	Reactor Internal Structures						
	B20	General						
28	-	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	– Mixing effect of coolant in upper plenum depends on how much the coolant of upper part in the upper plenum contributes to the circulation of coolant in the primary system. As a result, it has a large effect on cladding temperature. – Since there is still an effect of forced circulation in the 1st phase, the mixing effect of coolant in the upper plenum is strong. – For the 2nd phase, since Re number of upper plenum coolant is about 10,000, the mixing effect is smaller than in the 1st phase. Also, in SLIP, coolant temperature difference between the hottest point and the coldest point in RV is less than LOSP. Moreover, since low-temperature coolant flows from the core region into the upper plenum where high-temperature coolant remains, thermal stratification may occur and affect the natural circulation in the reactor vessel. – Therefore, the mixing effect of coolant has a relatively large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– There are sufficient data for flow tests using water, but it is difficult to apply these results to a system using sodium. – For low Re number phases, such as natural circulation, it is difficult to measure the mixing effect, and data contain uncertainty. – There is insufficient knowledge of thermal stratification in the natural circulation phase [8-48] [8-58] [8-59].	b02
29	-	Temperature dependence of physical properties of structural materials	L	L	K	– Since the error of physical properties of the core region is small, it has a small effect on cladding temperature.	– There are sufficient data for physical properties of the reactor internal structures.	b03

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
30	-	Natural convection	L	M	P	– Natural convection itself does not affect the cladding temperature, because there is still the effect of forced circulation in the 1st phase. – In the 2nd phase, since coolant flow rate around the reactor internal structure is low, it is possible to cause pressure loss that affects the flow rate of primary coolant by local natural convection generated around the reactor internal structure. Therefore, this affects cladding temperature.	– Flow behavior around the reactor internal structure at normal operation can be evaluated analytically. – In the natural circulation phase, it is difficult to analytically estimate the flow velocity distribution and temperature distribution around the reactor internal structure. – Behavior of the local eddies generated around the reactor internal structure in natural circulation, and pressure loss attributed to them, can be evaluated analytically. However, there are limited data to verify the results, and uncertainty is large.	b04
31	-	Flow-induced vibration	N/A	N/A	P	– Flow-induced vibration does not affect cladding temperature.	– It can be estimated analytically. – This phenomenon shows different behavior (or result) depending on geometry of facility. However, since there are no test data established for the same geometry, there is insufficient knowledge.	b05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
	B21 Reflector							
32	-	Deformation due to thermal effect and irradiation	L	L	P	– The coolant flow rate in the reflector region accounts for only 2% of that of the whole primary system. Therefore, even if deformation of the reflector or cavity occurs, it has little effect on cladding temperature.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Since analysis result of flow behavior at the transition has not been verified yet, it is difficult to estimate the temperature distribution. – There are limited data for the change in strength of structural material affected by irradiation.	b06
33	-	Flow in reflector region	L	L	P	– The coolant flow rate of reflector region accounts for only 2% of the whole primary system. Therefore, flow rate change in this area has little effect on cladding temperature.	– For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss and data contain uncertainty.	b07
34	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	– Compared to decay heat or rated power, heat generation by neutron capture and gamma rays has a small effect on cladding temperature and can be neglected.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Analysis result of flow behavior at the transition has not been verified yet. Hence, it is difficult to analytically estimate the temperature distribution at transient.	b08
	B22 Lower Plenum							
35	-	Pressure loss	L	L	P	– Pressure loss in the lower plenum region accounts for only 10% of that of the whole primary system. Therefore, coolant flow rate change corresponding to the pressure loss change has little effect on cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	b09

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
36	-	Heat capacity	L	L	K	– Evaluation error of specific heat and density is small, and heat capacity of lower plenum accounts for a small percentage of heat capacity of the whole primary system. Therefore, it has a small effect on cladding temperature.	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b10
37	-	Coolant mixing and thermal stratification	L	L	P	– Heat capacity of the coolant of lower plenum accounts for a certain percentage of heat capacity of the whole primary system. Therefore, mixing effect of coolant of lower plenum has a small effect on cladding temperature.	– Although there are sufficient flow test data using water, it is difficult to scale the result of a water test to a sodium system. – In low Re number phases such as natural circulation, it is difficult to measure the mixing effect and data contain uncertainty. – There is insufficient knowledge of thermal stratification during natural circulation phase [8-58] [8-59].	b11
38	-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel.	L	L	P	– Heat transfer from lower head of reactor vessel to the outside is mainly by radiation but the heat transfer area accounts for a small percentage of the RV wall. Therefore, it has a small effect on cladding temperature.	– There are limited data related to the change of radiation factor by age deterioration. – There is little knowledge of convective heat transfer for spheres.	b12

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
	B23 Upper Plenum							
39	-	Pressure loss	L	L	P	– Compared to the pressure loss of overall primary system, most of the pressure loss of upper plenum region is 0. Therefore, the coolant flow rate change corresponding to the pressure loss change has a small effect on cladding temperature.	– Based on established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	b13
40	-	Heat capacity	L	L	K	– Evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b14
41	-	Heat transfer between shielding plug and sodium	L	L	K	– In the heat transfer between shielding plug and coolant, radiation is dominant. From the aspect of heat transfer area, the amount of heat transfer is small compared to that from RV wall surface to outside. Therefore, it has a small effect on cladding temperature.	– Amount of heat transfer by radiation governing heat transfer between shielding plug and coolant level can be estimated analytically [8-9] [8-10].	b15
42	-	Coolant mixing, thermal stratification, and thermal striping	L	L	P	– Since the upper plenum is long, radial distribution of coolant temperature becomes uniform regardless of the degree of mixing effect in the core outlet region. Therefore, it has a small effect on cladding temperature.	– There are limited experimental data for thermal striping [8-57]. – Uncertainty of analytical prediction method is large, since it strongly depends on geometry.	b16

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)								
	Figures of Merit (FoM): Cladding Temperature			Importance					
No.	Subsystem/ Component		Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	B24	Vertical Shroud							
43	-		Radial heat transfer between inside and outside coolant through vertical shroud	L	L	P	– Since vertical shroud is equipped with heat insulation material, heat transfer from the inside to the outside can mostly be neglected. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	b17
	B25	Radial Shield							
44	-		Flow in radial shield region	L	L	P	– Pressure loss in radial shield region accounts for only 8% of that of the whole primary system. Therefore, coolant flow rate change corresponding to the pressure loss change has little effect on cladding temperature.	– For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss, and data contain uncertainty.	b18
45	-		Heat capacity	L	L	K	– Ratio of the heat capacity of the radial shield to total heat capacity of primary system is large. However, evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b19

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
46	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	– Compared to decay heat or rated power, heat generation by neutron capture and gamma rays has a small effect on cladding temperature and can be neglected.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Analysis result of flow behavior at the transition has not been verified yet. Therefore, it is difficult to analytically estimate the temperature distribution.	b20
47	-	Radial heat transfer in radial shield region	L	L	P	– Since the heat capacity is large, radial shield has the role of a heat sink – Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	b21
	B26	Reactivity Control Drive Mechanism						
48	-	Shutdown velocity of reflector	L	L	K	– Since shutdown velocity of reflector is specifically designed and the error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Design reflector descent time is 8s.	b22
49	-	Shutdown velocity of shutdown rod	L	L	K	– Since shutdown velocity of the shutdown rod is specifically designed and the error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Design shutdown rod descent time is 8s.	b23

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
	C	Primary Heat Transport System						
	C0	General						
50	-	Natural circulation	L	M	P	– In the 1st phase, since there is still an effect of forced circulation by flow coastdown system, natural circulation has a small effect on cladding temperature. – In the 2nd phase, natural circulation flow rate in the primary system is the dominant factor that determines amount of heat removal, and hence impact on cladding temperature. Also, the natural circulation flow rate is greatly affected by the geometric shape of flow path and pressure loss. – As mentioned in a03 and b04, uncertainty in the pressure loss at core region may have a large effect on cladding temperature.	– Natural circulation flow rate can be evaluated analytically. – It is difficult to evaluate the pressure loss attributed to local eddies around the core and each reactor internal structure in natural circulation phase. – Since natural circulation testing has been conducted many times in the past [8-6] [8-7] [8-8] [8-11] [8-12] [8-13] [8-17] [8-18] [8-30] [8-31] [8-32] [8-33] [8-34] [8-35] [8-36] [8-37] [8-38] [8-39] [8-40] [8-41] [8-42] [8-43] [8-44] [8-45] [8-46] [8-47], there is sufficient knowledge, but natural circulation has strong dependence on geometric shape, so application of measured data for 4S contains uncertainty.	c01
51	-	Sodium inventory	L	L	K	– Since fill ration of sodium is specifically designed and its error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– An amount of sodium can be evaluated by design requirement [8-9] [8-15] [8-16].	c02

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
52	-	Heat capacity of coolant	L	L	K	– Errors in coolant specific heat and density are small. Therefore, heat capacity has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Amount of sodium can be evaluated by design requirement. – There are sufficient data for specific heat [8-9] [8-15] [8-16].	c03
	C1	IHX						
53	-	Pressure loss	L	L	P	– Pressure loss in IHX region accounts for only a certain percentage of the intermediate system. Therefore, coolant flow rate change corresponding to the pressure loss change has a small effect on cladding temperature.	– Based on the established knowledge [8-7] [8-9], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	c04
54	-	Heat transfer from primary coolant to intermediate coolant	N/A	N/A	P	– In the SLIP event, since sodium of intermediate system is leaking, there is no heat transfer from primary system to intermediate system. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	c05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
55	-	Primary flow rate	L	L	P	– Since primary flow rate at normal operation is specifically designed and its error is small, it has a small effect on cladding temperature in the 1st phase. – Primary flow rate in the natural circulation phase is determined by natural circulation head and pressure loss in the flow path. As mentioned in c05, thermal conduction is dominant in the heat transfer, which affects cladding temperature. Hence, primary flow rate in IHX has a small effect on FoM.	– There is sufficient knowledge to set primary system coolant flow rate in normal operation at the design requirement value. – It is difficult to measure the primary coolant flow rate in natural circulation, and the data contain uncertainty.	c06
56	-	Intermediate flow rate	N/A	N/A	P	– Same as c05.	– There is sufficient knowledge to set intermediate system coolant flow rate in normal operation at the design requirement value. – It is difficult to measure the flow rate of intermediate coolant in the natural circulation phase.	c07
57	-	Heat capacity	L	L	K	– Since evaluation error of specific heat and density is low, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	c08
58	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	N/A	N/A	P	– Same as c05.	– Coolant in the annulus part at normal operation shows spatially uniform flow. – In natural circulation, since uncertainty for pressure loss of baffle plate placed in annulus part is large, it is difficult to evaluate the flow behavior.	c09

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)								
	Figures of Merit (FoM): Cladding Temperature		Importance						
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	C2	Primary EMP						
59	-	Flow coastdown performance	L	L	P	– The halving time for flow coastdown of EMP is specifically designed and its error is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– There are test data for EMPs but no test data for EMPs of similar size as those in the primary system [8-49].	c10
60	-	Pressure loss	L	L	P	– Pressure loss of EMP accounts for only 10% of that of the whole primary system. Therefore, coolant flow rate change corresponding to the pressure loss change has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Based on the established knowledge [8-7] [8-8], form loss and friction loss can be evaluated. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient) and data contain uncertainty.	c11
61	-	Pump head	L	L	K	– 1st phase: The halving time for EMP flow coastdown is specifically designed and its error is small. Therefore, it has a small effect on cladding temperature. – 2nd phase: It has a small effect on cladding temperature since the pump is not running.	– It can be evaluated by design requirement.	c12
62	-	Heat capacity and joule heat at flow coastdown	L	N/A	P	– Evaluation error of specific heat and density is small. – Also, amount of heat generation by joule heat is lower than that of the decay heat. – Therefore, it has a small effect on cladding temperature.	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat. – Since there is uncertainty in the flow coastdown performance as mentioned in c10, there is also uncertainty in joule heat evaluation.	c13

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

No.	Subsystem/ Component	Phenomenon	Importance 1st Phase	Importance 2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature							
	E	Residual Heat Removal System						
	E2	RVACS						
63	-	Pressure loss in air flow path	L	L	K	– In the SLIP event, since IRACS does not function, decay heat removal is carried out by RVACS. – Heat removal by the RVACS is carried out by heat transfer by air and radiation. Heat transfer by air is carried out by natural convection, and compared to radiation, its rate is small. Therefore, the amount of heat transferred to air, which is affected by pressure loss in the airflow path that affects air velocity, does not have significant effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	Based on established knowledge, form loss and friction loss can be evaluated [8-8].	e07
64	-	Heat transfer between GV wall and air	L	L	K	– As the result of sensitivity analysis, thermal radiation is dominant in heat removal in RVACS. Hence, heat transfer by air does not have significant effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-10] [8-50] [8-51] [8-52] [8-53] [8-54] [8-55].	e08
65	-	Heat transfer between collector wall and air	L	L	K	– Same as e08. – See the result of the sensitivity analysis (Fig. 8.3.2.2-1).	– Same as e08.	e09
66	-	Heat transfer between concrete wall and air	L	L	K	– Same as e08. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Same as e08.	e10

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
67	-	Thermal radiation between RV wall and GV wall	L	M	P	– As stated in e07, radiation is dominant in heat removal in RVACS. Therefore, radiation has a comparatively large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Amount of heat transfer by radiation can be evaluated analytically. – There are sufficient data for radiation factor [8-10] [8-55] [8-56]. – There are limited data related to the change of radiation factor by age deterioration.	e11
68	-	Thermal radiation between GV wall and heat collector wall	L	M	P	– Same as e11. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Amount of heat transfer by radiation can be evaluated analytically. – There are limited data related to the change of radiation factor by age deterioration.	e12
69	-	Thermal radiation between heat collector wall and concrete wall	L	M	P	– Same as e11. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Amount of heat transfer by radiation can be evaluated analytically. – There are limited data related to the change of radiation factor by age deterioration.	e13
70	-	Asymmetric airflow	L	M	P	– Same as e11. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– In the natural circulation phase, nonuniform flow distribution in the IHX annulus piping region has a potential to show asymmetric behavior. – There is a possibility that such asymmetric flow would produce asymmetric differences in air current in the RVACS, but there is insufficient knowledge and data contain uncertainty.	e14

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-1. Final PIRT Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
71	-	Inlet air temperature range	L	L	K	– This is one of the factors that has a large effect on the amount of heat transfer to the RVACS air (temperature difference), but heat transfer to the air is not dominant for RVACS heat removal. Therefore, it does not have a significant effect on cladding temperature.	– The highest and lowest temperatures of all the states in U.S. are shown [8-3].	e15
	F	Instrumentation and Control System						
	F1	Instrumentation and Control Equipment						
	F11	Plant Protection Sensors						
72	-	Delay of scram signal of primary EMP voltage and current	L	L	K	– Scram signal of primary EMP voltage and current is specifically designed and its error is small. Therefore, it has a small effect on cladding temperature.	– Response time is specifically set according to the design requirement.	f01
73	-	Delay of scram signal of power line voltage	L	L	K	– Scram signal of low power line voltage is specifically designed and its error is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.2-1).	– Same as f01.	f02
74	-	Delay of scram signal of neutron flux	L	L	K	– Same as f02.	– Same as f01.	f03
75	-	Delay of scram signal of IHX primary outlet temperature	L	L	K	– Same as f02.	– Same as f01.	f04
	F12	Others						
76	-	Delay of interlock signal of SG outlet temperature	L	L	K	– Same as f02.	– Same as f01.	f05

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.2-2. Phenomena with Difference Between Initial and Final Ranking Results for SLIP

	Event: Sodium Leakage from Intermediate Piping (SLIP)			Change from Initial Ranking	
	Figures of Merit (FoM): Cladding Temperature			Importance	
No.	Subsystem/Component		Phenomenon	1st Phase	2nd Phase
-	A	Core and Fuel Assemblies			
1	-		Pressure loss in core region	L → M	L
7	-		Intra- and inter-assembly flow distribution	H	M → L
10	-		Heat capacity of core assemblies	M → L	M → L
12	-		Core power	M → L	M → L
13	-		Decay heat	M → L	M → L
15	-		Rate of scram reactivity insertion	H → L	L
16	-		Delay of scram reactivity insertion	H → L	L
18	-		Temperature dependence of physical properties of materials	M → L	M → L
23	-		Inter-wrapper flow between wrapper tubes	L	L → M
-	B	Reactor System			
-	B2	Reactor Internal Structures			
-	B20	General			
28	-		Coolant mixing effect in upper plenum including thermal stratification	M → L	M
31	-		Flow-induced vibration	L → N/A	L → N/A
-	B23	Upper Plenum			
40	-		Heat capacity	L	M → L
-	B26	Reactivity Control Drive Mechanism			
48	-		Shutdown velocity of reflector	M → L	L
49	-		Shutdown velocity of shutdown rod	M → L	L
-	C	Primary Heat Transport System			
-	C0	General			
51	-		Sodium inventory	L	M → L
-	C1	IHX			
54	-		Heat transfer from primary coolant to intermediate coolant	L → N/A	L → N/A
56	-		Intermediate flow rate	L → N/A	L → N/A
58	-		Spatial distribution effect of intermediate flow path in IHX annulus shape	L → N/A	L → N/A

Table 8.4.2-2. Phenomena with Difference Between Initial and Final Ranking Results for SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)			Change from Initial Ranking	
	Figures of Merit (FoM): Cladding Temperature			Importance	
No.	**Subsystem/Component**		**Phenomenon**	**1st Phase**	**2nd Phase**
-	C2	Primary EMP			
59	-		Flow coastdown performance	M → L	L
62	-		Heat capacity and joule heat at flow coastdown	L	L →N/A
-	E	Decay Heat Removal System			
-	E2	RVACS			
63	-		Pressure loss in airflow path	L	M → L
71	-		Inlet air temperature range	L	M → L
-	F	Instrumentation and Control System			
-	F1	Instrumentation and Control Equipment			
-	F11	Plant Protection Sensors			
72	-		Delay of scram signal of primary EMP voltage and current	M → L	L
73	-		Delay of scram signal of power line voltage	M → L	L

8.4.3 Failure of a Cavity Can (FCC)

Table 8.4.3-1 shows the ranking results for the FCC event. This table contains not only the ranking results but also explanations for the ranking results, with rationales.

There were 11 phenomena that changed their importance rankings from the initial ranking results shown in Section 8.2. One phenomenon increased its ranking, while 10 phenomena decreased.

Table 8.4.3-2 shows the phenomena with changed rankings.

Table 8.4.3-1. Final PIRT Results for FCC

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	A	Core and Fuel Assemblies					
1	-	Reactivity feedback	L	P	– In the FCC event, subcritical reactivity can be maintained by reactor scram. Therefore, reactivity feedback does not affect cladding temperature.	– As a result of criticality test, analysis code is sufficiently verified by the organized nuclear data. Therefore, reactivity feedback can be estimated. – There is insufficient knowledge of shape variation caused by temperature change at the transition. Therefore, evaluated value of reactivity feedback attributed to shape variation is uncertain.	a04
2	-	Gap conductance between fuel and cladding	L	K	– Unlike a reactor using oxide fuel, the gap between the cladding and fuel slug in 4S is filled with sodium having high thermal conductivity corresponding to those of the fuel and cladding. Therefore, in the gap between cladding and fuel slug, gap conductance will not cause a large temperature difference that affects cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Since the uncertainty of thermal conductivity for fuel, sodium, and cladding, which is used to estimate the gap conductance, is small, uncertainty of gap conductance is also small.	a05

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

No.	Subsystem/ Component	Phenomenon	Importance (No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature						
3	-	Heat transfer between cladding and coolant	L	K	– Heat transfer coefficient (Nu number) is the function of Pe number (=Re·Pr) and contribution by thermal conduction is dominant at Pe<100. Also, in the 4S, Pe number is about 100 even at normal operation. Therefore, since the heat transfer between cladding and coolant is governed by thermal conductivity, which is a physical property not affected by coolant flow rate, sensitivity is small. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Since the correlation equation of Nu number in the space between cladding and coolant is investigated, there is sufficient knowledge for many fuel pins, such as those of the FFTF and CRBR [8-7] [8-9]. – For the 4S, a modified Lyon's correlation is used.	a06
4	-	Intra- and inter-assembly flow distribution	H	P	– If flow rate distribution among the assemblies deviates from the design specification, coolant flow rate flowing in each assembly will change. This has direct effect on the fuel pin temperature, so does the cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Flow distribution among assemblies at normal operation can be evaluated by pressure loss coefficient of each assembly [8-7] [8-8]. – Data of the pressure loss coefficient of core region can be obtained by test. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), and data contain uncertainty.	a07

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

No.	Subsystem/ Component	Phenomenon	Importance (No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature						
5	-	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	P	– By the effect of radial heat transfer, heat transfers from the high-temperature core region to low-temperature peripheral part. – As stated in the a06, thermal conductivity is dominant in heat transfer from structural material to coolant. Also, since liquid metal sodium is used as coolant, it has a great effect on cladding temperature. Especially since reactivity is inserted in the FCC event, core temperature increases and heat is more likely to transfer to the outside. Therefore, the effect on cladding temperature by radial heat transfer is not small. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– There are sufficient data for physical properties of fuel assembly material and sodium [8-7] [8-8] [8-9] [8-10]. – However, since this phenomenon at natural circulation relates to phenomena such as "Natural convection" in a03, it has an uncertainty.	a08
6	-	Heat capacity of core assemblies	L	K	– Since evaluation error of specific heat and density is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	a10
7	-	Coolant boiling	N/A	K	– Peak value of cladding temperature in the FCC event (less than 650°C) is lower than boiling point (1,000°C ~ 1,100°C). Therefore, coolant boiling has no effect on cladding temperature.	– Cover gas pressure and hydraulic head from coolant surface in the reactor vessel to the core can be estimated by design requirement. – There are sufficient data of the vapor pressure curve for liquid metal sodium [8-15] [8-16] [8-17] [8-18].	a11

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
8	-	Core power	L	K	– In the FCC event, power falls by scram right after the event starts. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Since nuclear data are well-organized, power can be calculated [8-65] [8-66].	a12
9	-	Decay heat	L	K	– Contribution to power by decay heat is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Nuclear data are well-organized [8-67]. – Since decay data of FPs are also well organized, decay heat can be calculated.	a13
10	-	Heat transfer between core support plate and sodium	L	P	– Heat transfer coefficient between core support plate and coolant is large, so the temperature difference between core support plate and coolant is small. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – In low Re number phases such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	a14
11	-	Rate of scram reactivity insertion	M	K	– FCC is the event during which reactivity is applied. Therefore, this phenomenon relating to the insertion of scram that controls reactivity has a major effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Reactivity distribution can be estimated by the three-dimensional calculation with the verified calculation code [8-68].	a15
12	-	Delay of scram reactivity insertion	M	K	– The description for a15 can be applicable for the change in power by the delay in reactivity insertion. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Same as a15.	a16

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

No.	Subsystem/ Component	Phenomenon	Importance (No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature						
13	-	Eutectic reaction between fuel and cladding	L	P	– In this transition event, peak fuel temperature and initial fuel temperature are nearly equivalent, and power density is small. Therefore, since the progress of liquid phase formation can be neglected, it has a small effect on cladding temperature.	– There are data from existing reactors regarding temperature at the start of eutectic reaction and growth rate of eutectic reaction [1-2] [8-2] [8-19]. – Data have been obtained by the heating test using an irradiated fuel pin. – Burnup composition of fuel affects start temperature of eutectic reaction and growth rate of eutectic reaction, but there are limited data of the eutectic reaction start temperature.	a17
14	-	Temperature dependence of physical properties of materials	L	K	– Since physical properties of the core region are well known, it has a small effect on cladding temperature.	– There are sufficient data for physical properties of component materials in the core [8-20] [8-21] [8-22] [8-23] [8-24].	a18
15	-	Flow-induced vibration in a subassembly	L	P	– The vibration of structural material attributed to flow-induced vibration has no effect on cladding temperature.	– The evaluation of the flow-induced vibration has been generalized by existing test data using water as a fluid. – However, there are limited test data using sodium [8-62] [8-63]. Since data such as the dimensions and supporting structure of the 4S differ from those of existing sodium reactors, it is difficult to apply the test data of sodium to the 4S.	a22

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
16	-	Inter-wrapper flow between wrapper tubes	L	P	– Since the flow rate among the assemblies is small, inter-wrapper flow has a small effect on the heat transfer between wrapper tube wall and coolant. Therefore, at normal operation, inter-wrapper flow has little effect on cladding temperature. – The effect becomes pronounced when using direct reactor auxiliary cooling system (DRACS).	– In the natural circulation phase, there are test data of sodium flow behavior in gaps among fuel assemblies and their flow resistance to the part [8-64]. – However, the data are highly relative to test scale and geometry, and there is insufficient knowledge regarding scaling.	a23
17	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	P	– Flow distribution change of coolant has direct effect on cladding temperature.	– Knowledge of natural circulation behavior [8-6] [8-7] [8-8] [8-11] [8-12] [8-13] [8-17] [8-18] [8-30] [8-31] [8-32] [8-33] [8-34] [8-35] [8-36] [8-37] [8-38] [8-39] [8-40] [8-41] [8-42] [8-43] [8-44] [8-45] [8-56] [8-47] contains uncertainty. – It is difficult to quantitatively estimate the pressure loss and heat transfer coefficient in the natural circulation phase.	a24
18	-	Radial power distribution	L	K	– Radial power distribution affects cladding temperature. Time for transitional power distribution change is short, so transitional effect is small.	– Much power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.	a25

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)							
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
19	-	Axial power distribution	L	K	– Axial power distribution affects cladding temperature. Time for transitional power distribution change is short and transitional effect is small. – Accident analyses were performed with some cases of axial power distribution, considering the condition of burn-up core (i.e. beginning of life and end of life). As a result, the influence of the axial power distribution change due to the difference in the burn-up core was small against FOM.	– Much power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.	a26
20	-	Reactivity insertion by cavity can failure	M	P	– This phenomenon relating to the reactivity insertion has a large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– The amount of reactivity insertion is limited by the volume of one cavity can because this phenomenon assumes the failure of one out of 36 cavity cans. – However, uncertainty of reactivity insertion rate is large because of the range of factors that could cause failure, such as manufacturing error.	a27

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	B	Reactor System					
	B1	Reactor Vessel					
	B2	Reactor Internal Structures					
	B20	General					
21	-	Coolant mixing effect in upper plenum including thermal stratification	L	P	– This phenomenon deals only with the situation before natural circulation phase. Hence, this phenomenon does not influence cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– There are sufficient data for flow tests using water, but it is difficult to apply these results to a system using sodium. – For low Re number phases, such as natural circulation, it is difficult to measure the mixing effect, and data contain uncertainty. – There is insufficient knowledge of thermal stratification in the natural circulation phase [8-48] [8-58] [8-59].	b02
	B21	Reflector					
22	-	Flow in reflector region	L	P	– Coolant flow rate of reflector region accounts for only 2% of that of the whole primary system. Therefore, flow rate change in this area has little effect on cladding temperature.	– For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss, and data contain uncertainty.	b07
23	-	Effect of generated heat by neutron capture and gamma rays	L	P	– Compared to decay heat or rated power, heat generation by neutron capture and gamma rays has a small effect on cladding temperature and can be neglected.	– Neutron flux distribution and temperature distribution can be estimated analytically. – Analysis result of flow behavior at the transition has not been verified yet. Therefore, it is difficult to analytically estimate the temperature distribution at transient.	b08

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	B22	Lower Plenum					
24	-	Heat capacity	L	K	– Evaluation error of specific heat and density is small, and heat capacity of lower plenum accounts for a small percentage of that of the whole primary system. Therefore, it has a small effect on cladding temperature.	– Materials used and weights of component structures can be evaluated by design data. – Some data for specific heat exists.	b10
25	-	Coolant mixing and thermal stratification	L	P	– Heat capacity of the coolant at lower plenum accounts for a small percentage of heat capacity of the whole primary system. – Therefore, mixing effect of coolant in lower plenum has a small effect on cladding temperature.	– Although there are sufficient flow test data using water, it is difficult to scale the result of a water test to a sodium system. – In low Re number phases such as natural circulation, it is difficult to measure the mixing effect, and data contain uncertainty. – There is insufficient knowledge of thermal stratification during natural circulation phase [8-58] [8-59].	b11
26	-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel.	L	P	– Heat transfer from lower head of reactor vessel to the outside is governed mainly by radiation, but the heat transfer area accounts for a small percentage of that of the RV wall. Therefore, it has a small effect on cladding temperature.	– There are limited data related to the change of radiation factor by age deterioration. – There is little knowledge of convective heat transfer for spheres.	b12
	B23	Upper Plenum					
27	-	Heat capacity	L	K	– Evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b14

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

No.	Subsystem/ Component	Phenomenon	Importance (No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature						
28	-	Heat transfer between shielding plug and sodium	L	K	– In the heat transfer between shielding plug and coolant, radiation is dominant. From the aspect of heat transfer area, the amount of heat transfer is small compared to that from RV wall surface to outside. Therefore, it has a small effect on cladding temperature.	– Amount of heat transfer by radiation governing the heat transfer between shielding plug and coolant surface can be estimated analytically [8-9] [8-10].	b15
29	-	Coolant mixing, thermal stratification, and thermal striping	L	P	– Since the upper plenum is long, radial distribution of coolant temperature becomes uniform regardless of the degree of mixing in core outlet region. – Therefore, it has a small effect on cladding temperature.	– There are limited experimental data for thermal striping [8-57]. – Uncertainty of analytical prediction method is large, since it strongly depends on geometry.	b16
	B24	Vertical Shroud					
30	-	Radial heat transfer between inside and outside coolant through vertical shroud	L	P	– Since vertical shroud is equipped with heat insulation material, heat transfer from the inside to the outside can mostly be neglected. Therefore, it has a small effect on cladding temperature.	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	b17
	B25	Radial Shield					
31	-	Flow in radial shield region	L	P	– Pressure loss in radial shield region accounts for only 8% of that of the whole primary system. Therefore, coolant flow rate change corresponding to the pressure loss change has little effect on cladding temperature.	– For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss, and data contain uncertainty.	b18

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
32	-	Heat capacity	L	K	– Ratio of radial shield heat capacity to total heat capacity of primary system is large. However, evaluation error of specific heat and density is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	b19
33	-	Effect of generated heat by neutron capture and gamma rays	L	P	– Compared to decay heat or rated power, heat generation by neutron capture and gamma rays has a small effect on cladding temperature and can be neglected.	– Neutron flux distribution and temperature distribution at normal operation can be estimated analytically. – Analysis result of flow behavior at the transition has not been verified yet. Hence, it is difficult to analytically estimate the temperature distribution at transient.	b20
34	-	Radial heat transfer in radial shield region	L	P	– Since the heat capacity is large, radial shield has the role of a heat sink. – In FCC event, ratio of heat removal by the radial shield to that of the entire primary system is small due to heat removal by IRACS. – Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity and data contain uncertainty.	b21

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	B26	Reactivity Control Drive Mechanism					
35	-	Shutdown velocity of reflector	M	K	– As stated in a15, FCC is the event during which reactivity is applied. This phenomenon relating to scram insertion by the reflector that controls reactivity has a large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Design reflector descent time is 8s. – This phenomenon is caused by gravitational force. Hence, there are sufficient data.	b22
36	-	Shutdown velocity of shutdown rod	M	K	– As stated in a15, since FCC is the event during which reactivity is applied, this event relating to scram insertion by the shutdown rod that controls reactivity has a large effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Design shutdown rod descent time is 8s.. – This phenomenon is caused by gravitational force. Hence, there are sufficient data.	b23
	C	Primary Heat Transport System					
	C0	General					
37	-	Sodium inventory	L	K	– Since filling volume of sodium is specifically designed and its error is small, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– An amount of sodium can be evaluated by design requirement [8-9] [8-15] [8-16].	c02
38	-	Heat capacity of coolant	L	K	– Error in specific heat and density evaluation for coolant is small. Therefore, it has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Amount of sodium can be evaluated by design requirement. – There are sufficient data for specific heat [8-9] [8-15] [8-16].	c03

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

No.	Subsystem/ Component	Phenomenon	Importance (No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature						
	C1 IHX						
39	-	Heat transfer from primary coolant to intermediate coolant	L	P	– It takes 400s for primary system coolant to travel the flow path at normal operation. Therefore, the change in heat transfer of the IHX in the 1st phase does not affect cladding temperature. – Flow rate in the 2nd phase is small. Since its Pe (=Re·Pr) number is 10 or less, the effect by thermal conductivity is dominant in heat transfer. Hence, heat transfer from primary system to intermediate system through heat transfer tube has a small effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– There is sufficient knowledge of heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	c05
40	-	Primary flow rate	L	P	– Since primary flow rate at normal operation is specifically designed and its error is small, it has a small effect on cladding temperature in the 1st phase. – Primary flow rate in the natural circulation phase is determined by the natural circulation head and pressure loss in flow path. As mentioned in c05, thermal conduction is dominant in the heat transfer. Therefore, primary flow rate in IHX has a small effect on cladding temperature.	– There is knowledge to be able to set coolant flow rate in the primary system at normal operation in the value based on design requirement. – It is difficult to measure the primary coolant flow rate in natural circulation, and the data contain uncertainty.	c06

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
41	-	Intermediate flow rate	L	P	– Same as c06. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– There is knowledge to be able to set coolant flow rate in the intermediate system at normal operation in the value based on design requirement. – It is difficult to measure the flow rate of intermediate coolant in the natural circulation phase.	c07
42	-	Heat capacity	L	K	– Since evaluation error of specific heat and density is small, it has a little effect on cladding temperature. – See the result of the sensitivity analysis (Fig. 8.3.3.3-1).	– Materials used and weights of component structures can be evaluated by design data. – There are sufficient data for specific heat.	c08
43	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	L	P	– As stated in c04 of LOSP (Table 8.4.1-1), since pressure loss in IHX has little effect, the effect on cladding temperature is small even if the local pressure loss caused by the natural convection is generated at the annulus part in the intermediate system at the low flow rate.	– Coolant in the annulus part at normal operation shows spatially uniform flow. – At natural circulation, since uncertainty of pressure loss at baffle plate placed in annulus part is large, it is difficult to evaluate the flow behavior.	c09
	D	Intermediate Heat Transport System					
	D0	General					
44	-	Heat removal from SG	L	K	– In the event of FCC, water-steam system does not remove heat. Therefore, it has no effect on cladding temperature.	– Amount of heat removal by SG can be evaluated by design requiement.	d03

Note:

H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-1. Final PIRT Results for FCC (cont.)

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	(No Phase Partition)	SoK	Ranking Rationale for Importance	Ranking Rationale for SoK	Code
45	-	Heat transfer between upper plenum and intermediate coolant external to IHX	L	P	– Since the amount of coolant contained in this part is small, it has a small effect on cladding temperature.	– There is sufficient knowledge about heat transfer coefficient at normal operation [8-9] [8-10] [8-14]. – For low Re number phases, such as natural circulation, it is difficult to measure the coolant flow velocity, and data contain uncertainty.	d04
	D1	Intermediate EMP					
	E	Residual Heat Removal System					
	F	Instrumentation and Control System					
	F1	Instrumentation and Control Equipment					
	F11	Plant Protection Sensors					
46	-	Delay of scram signal of neutron flux	M	K	– Same as a15, b22, and b23.	– Response time is specifically set according to the design requirement.	f03
47	-	Delay of scram signal of IHX primary outlet temperature	L	K	– This phenomenon has little to do with shutdown in FCC.	– Same as f01.	f04
	F12	Others					
48	-	Delay of interlock signal of SG outlet temperature	L	K	– This phenomenon has little to do with shutdown in FCC.	– Same as f01.	f05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially known, U: Unknown

Table 8.4.3-2. Phenomena with Difference Between Initial and Final Ranking Results for FCC

	Event: Failure of a Cavity Can (FCC)			Change from Initial Ranking
	Figures of Merit (FoM): Cladding Temperature			Importance (No Phase Partition)
No.	Subsystem/Component		Phenomenon	
-	A	Core and Fuel Assemblies		
5	-		Radial heat transfer between subassemblies (S/A<-->sodium<-->S/A)	L → M
6	-		Heat capacity of core assemblies	M → L
8	-		Core power	M → L
9	-		Decay heat	M → L
11	-		Rate of scram reactivity insertion	H → M
12	-		Delay of scram reactivity insertion	H → M
-	B	Reactor System		
-	B2	Reactor Internal Structures		
-	B20	General		
21	-		Coolant mixing effect in upper plenum including thermal stratification	M → L
-	B26	Reactivity Control Drive Mechanism		
35	-		Shutdown velocity of reflector	H → M
36	-		Shutdown velocity of shutdown rod	H → M
-	C	Primary Heat Transport System		
-	C0	General		
38	-		Heat capacity of coolant	M → L
-	F	Instrumentation and Control System		
-	F1	Instrumentation and Control Equipment		
-	F11	Plant Protection Sensors		
46	-		Delay of scram signal of neutron flux	H → M
-	F12	Others		

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9 FURTHER INVESTIGATION

The purpose of this chapter is to indicate how to select the phenomena that have high or medium importance but lack sufficient knowledge based on the results of the 4S PIRT, and how to expand knowledge of these phenomena in the future.

Regarding further investigation to enhance knowledge of the phenomena, both testing and theoretical evaluation are planned. The theoretical evaluation will include detailed analysis using computational fluid dynamics (CFD) codes.

9.1 PROCEDURE FOR FURTHER INVESTIGATION

In this section, the PIRT results stated in Chapter 8 are summarized, and priorities for further investigation of each phenomenon are established. Priorities are determined using a five-level scale by combining the relative importance of the phenomenon obtained from the PIRT results and the SoK, as shown in the matrix in Figure 9.1-1. In this figure, phenomena with importance ranking "N/A" are excluded because their effect on FoM is negligible. These priorities were established based on advice and suggestions from the PIRT IRAP. The procedure for further investigation is chosen beased on each priority.

The numbers assigned in the nine cells in the matrix in Figure 9.1-1 indicate the order of priority. A smaller number indicates higher priority.

The combinations of the importance of the phenomena and the SoK for each priority are as follows.

Priority 1:	Unknown (regardless of importance)
Priority 2:	High importance/partially known
Priority 3:	Medium importance/partially known
Priority 4:	Low importance/partially known
Priority 5:	Known (regardless of importance)

Tables 9.1-1, 9.1-2, and 9.1-3 show the phenomena with priority numbers, updating the Tables 8.4.1-1, 8.4.2-1, and 8.4.3-1, which show the PIRT result for each event.

Table 9.1-4 summarizes the plan for further investigation based on the priorities listed in Tables 9.1-1, 9.1-2, and 9.1-3. The rationales associated with each of the priorities are as follows.

(1) Priority 1

For the phenomena categorized as Priority 1, the importance ranking result itself may contain large uncertainty because the phenomena are unknown. Although improvement of knowledge is required to minimize the uncertainty of the importance ranking, since knowledge is insufficient, the result obtained from the theory and analysis has a large uncertainty. Therefore, even though it requires greater work, compared with such analysis, testing is required as an investigational tool to expand the knowledge of the phenomenon. In the actual testing, however, some preliminary prior evaluations by analysis will be needed such as for the design of the testing device.

(2) Priorities 2 and 3

The phenomena categorized as Priorities 2 and 3 are all partially known regardless of importance ranking. Therefore, the uncertainty of the analytical model in the detailed analysis code is small and the range of uncertainty is properly recognized. As a first approach, investigation by the theoretical evaluation such as the analysis using the detailed analysis code shall be conducted, because there is a high possibility that those results will help to reduce the range of uncertainty quantitatively. If those investigations do not contribute to the improvement of knowledge sufficiently, however, testing shall be performed as a second approach. Judgment will be made for whether the knowledge is sufficiently expanded by an expert's evaluation of the analysis results.

(3) Priority 4

Although knowledge of the phenomena in Priority 4 is partial, since its importance is low, further knowledge improvement is judged to be not required immediately.

(4) Priority 5

Knowledge of the phenomena in Priority 5 is sufficient, regardless of importance ranking. Therefore, further knowledge improvement is not required for this evaluation.

Table 9.1-1. Final PIRT with Priority for Further Investigation of LOSP

	Event: Loss of Offsite Power (LOSP)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
	A	Core and Fuel Assemblies					
1	-	Pressure loss in core region	M	L	P	3	a01
2	-	Pressure loss in reflector region	L	L	P	4	a02
3	-	Natural convection	L	M	P	3	a03
4	-	Reactivity feedback	L	N/A	P	4	a04
5	-	Gap conductance between fuel and cladding	L	L	K	5	a05
6	-	Heat transfer between cladding and coolant	L	L	K	5	a06
7	-	Intra- and inter-assembly flow distribution	H	L	P	2	a07
8	-	Radial heat transfer between subassemblies (S/A<-->sodium<-->S/A)	M	M	P	3	a08
9	-	Heat transfer between reflector and coolant	L	L	P	4	a09
10	-	Heat capacity of core assemblies	L	L	K	5	a10
11	-	Coolant boiling	N/A	N/A	K	N/A	a11
12	-	Core power	L	L	K	5	a12
13	-	Decay heat	L	L	K	5	a13
14	-	Heat transfer between core support plate and sodium	L	L	P	4	a14
15	-	Rate of scram reactivity insertion	L	L	K	5	a15
16	-	Delay of scram reactivity insertion	L	L	K	5	a16
17	-	Eutectic reaction between fuel and cladding	L	L	P	4	a17
18	-	Temperature dependence of physical properties of materials	L	L	K	5	a18
19	-	FP release from fuel slug into gas plenum	N/A	N/A	K	N/A	a19
20	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	P	N/A	a20
21	-	FP transport from sodium in primary system to cover gas	N/A	N/A	P	N/A	a21

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-1. Final PIRT with Priority for Further Investigation of LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
22	-	Flow-induced vibration in a subassembly	L	L	P	4	a22
23	-	Inter-wrapper flow between wrapper tubes	L	M	P	3	a23
24	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	2	a24
25	-	Radial power distribution	L	L	K	5	a25
26	-	Axial power distribution	L	L	K	5	a26
	B	Reactor System					
	B1	Reactor Vessel					
27	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	K	N/A	b01
	B2	Reactor Internal Structures					
	B20	General					
28	-	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	3	b02
29	-	Temperature dependence of physical properties of structural materials	L	L	K	5	b03
30	-	Natural convection	L	M	P	3	b04
31	-	Flow-induced vibration	N/A	N/A	P	N/A	b05
	B21	Reflector					
32	-	Deformation due to thermal effect and irradiation	L	L	P	4	b06
33	-	Flow in reflector region	L	L	P	4	b07
34	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	4	b08
	B22	Lower Plenum					
35	-	Pressure loss	L	L	P	4	b09
36	-	Heat capacity	L	L	K	5	b10
37	-	Coolant mixing and thermal stratification	L	L	P	4	b11
38	-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	L	P	4	b12

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-1. Final PIRT with Priority for Further Investigation of LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	**Subsystem/ Component**	**Phenomenon**	**1st Phase**	**2nd Phase**	**SoK**	**Priority**	**Code**
	B23	Upper Plenum					
39	-	Pressure loss	L	L	P	4	b13
40	-	Heat capacity	L	L	K	5	b14
41	-	Heat transfer between shielding plug and sodium	L	L	K	5	b15
42	-	Coolant mixing, thermal stratification, and thermal striping	L	L	P	4	b16
	B24	Vertical Shroud					
43	-	Radial heat transfer between inside and outside coolant through vertical shroud	L	L	P	4	b17
	B25	Radial Shield					
44	-	Flow in radial shield region	L	L	P	4	b18
45	-	Heat capacity	L	L	K	5	b19
46	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	4	b20
47	-	Radial heat transfer in radial shield region	L	L	P	4	b21
	B26	Reactivity Control Drive Mechanism					
48	-	Shutdown velocity of reflector	L	L	K	5	b22
49	-	Shutdown velocity of shutdown rod	L	L	K	5	b23
	C	Primary Heat Transport System					
	C0	General					
50	-	Natural circulation	L	M	P	3	c01
51	-	Sodium inventory	L	L	K	5	c02
52	-	Heat capacity of coolant	L	L	K	5	c03
	C1	IHX					
53	-	Pressure loss	L	L	P	4	c04
54	-	Heat transfer from primary coolant to intermediate coolant	L	L	P	4	c05
55	-	Primary flow rate	L	L	P	4	c06
56	-	Intermediate flow rate	L	L	P	4	c07

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-1. Final PIRT with Priority for Further Investigation of LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature			Importance				
No.	Subsystem/ Component		Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
57	-		Heat capacity	L	L	K	5	c08
58	-		Spatial distribution effect of intermediate flow path in IHX annulus shape	L	L	P	4	c09
	C2	Primary EMP						
59	-		Flow coastdown performance	L	L	P	4	c10
60	-		Pressure loss	L	L	P	4	c11
61	-		Pump head	L	L	K	5	c12
62	-		Heat capacity and joule heat at flow coastdown	L	N/A	P	4	c13
	D	Intermediate Heat Transport System						
	D0	General						
63	-		Pressure loss	L	L	P	4	d01
64	-		Natural circulation	L	M	P	3	d02
65	-		Heat removal from SG	L	N/A	K	5	d03
66	-		Heat transfer between upper plenum and intermediate coolant external to IHX	L	L	P	4	d04
	D1	Intermediate EMP						
67	-		Flow coastdown performance	L	L	K	5	d05
68	-		Pressure loss	L	L	P	4	d06
69	-		Pump head	L	L	K	5	d07
	D2	Steam Generator System						
70	-		Heat capacity of structure, sodium, water, and steam	L	L	K	5	d08
	E	Residual Heat Removal System						
	E1	Air cooler of IRACS						
71	-		Pressure loss of sodium side	L	L	P	4	e01
72	-		Pressure loss of air side	L	L	P	4	e02
73	-		Heat transfer between tube and air	L	M	P	3	e03
74	-		Heat transfer between tube and sodium	L	L	P	4	e04
75	-		Inlet air temperature range	L	M	K	5	e05
76	-		Heat capacity	L	L	K	5	e06

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-1. Final PIRT with Priority for Further Investigation of LOSP (cont.)

	Event: Loss of Offsite Power (LOSP)							
	Figures of Merit (FoM): Cladding Temperature			Importance				
No.	Subsystem/ Component		Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
	E2	RVACS						
77	-		Pressure loss in air flow path	L	L	K	5	e07
78	-		Heat transfer between GV wall and air	L	L	K	5	e08
79	-		Heat transfer between collector wall and air	L	L	K	5	e09
80	-		Heat transfer between concrete wall and air	L	L	K	5	e10
81	-		Thermal radiation between RV wall and GV wall	L	L	P	4	e11
82	-		Thermal radiation between GV wall and heat collector wall	L	L	P	4	e12
83	-		Thermal radiation between heat collector wall and concrete wall	L	L	P	4	e13
84	-		Asymmetric airflow	L	L	P	4	e14
85	-		Inlet air temperature range	L	L	K	5	e15
	F	Instrumentation and Control System						
	F1	Instrumentation and Control Equipment						
	F11	Plant Protection Sensors						
86	-		Delay of scram signal of primary EMP voltage and current	L	L	K	5	f01
87	-		Delay of scram signal of power line voltage	L	L	K	5	f02
88	-		Delay of scram signal of neutron flux	L	L	K	5	f03
89	-		Delay of scram signal of IHX primary outlet temperature	L	L	K	5	f04
	F12	Others						
90	-		Delay of interlock signal of SG outlet temperature	L	L	K	5	f05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-2. Final PIRT with Priority for Further Investigation of SLIP

	Event: Sodium Leakage from Intermediate Piping (SLIP)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	**Subsystem/ Component**	**Phenomenon**	**1st Phase**	**2nd Phase**	**SoK**	**Priority**	**Code**
	A	Core and Fuel Assemblies					
1	-	Pressure loss in core region	M	L	P	3	a01
2	-	Pressure loss in reflector region	L	L	P	4	a02
3	-	Natural convection	L	M	P	3	a03
4	-	Reactivity feedback	L	N/A	P	4	a04
5	-	Gap conductance between fuel and cladding	L	L	K	5	a05
6	-	Heat transfer between cladding and coolant	L	L	K	5	a06
7	-	Intra- and inter-assembly flow distribution	H	L	P	2	a07
8	-	Radial heat transfer between subassemblies (S/A<-->sodium <-->S/A)	M	M	P	3	a08
9	-	Heat transfer between reflector and coolant	L	L	P	4	a09
10	-	Heat capacity of core assemblies	L	L	K	5	a10
11	-	Coolant boiling	N/A	N/A	K	N/A	a11
12	-	Core power	L	L	K	5	a12
13	-	Decay heat	L	L	K	5	a13
14	-	Heat transfer between core support plate and sodium	L	L	P	4	a14
15	-	Rate of scram reactivity insertion	L	L	K	5	a15
16	-	Delay of scram reactivity insertion	L	L	K	5	a16
17	-	Eutectic reaction between fuel and cladding	L	L	P	4	a17
18	-	Temperature dependence of physical properties of materials	L	L	K	5	a18
19	-	FP release from fuel slug into gas plenum	N/A	N/A	K	N/A	a19
20	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	P	N/A	a20
21	-	FP transport from sodium in primary system to cover gas	N/A	N/A	P	N/A	a21

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-2. Final PIRT with Priority for Further Investigation of SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)						
	Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
22	-	Flow-induced vibration in a subassembly	L	L	P	4	a22
23	-	Inter-wrapper flow between wrapper tubes	L	M	P	3	a23
24	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	2	a24
25	-	Radial power distribution	L	L	K	5	a25
26	-	Axial power distribution	L	L	K	5	a26
	B	Reactor System					
	B1	Reactor Vessel					
27	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	K	N/A	b01
	B2	Reactor Internal Structures					
	B20	General					
28	-	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	3	b02
29	-	Temperature dependence of physical properties of structural materials	L	L	K	5	b03
30	-	Natural convection	L	M	P	3	b04
31	-	Flow-induced vibration	N/A	N/A	P	N/A	b05
	B21	Reflector					
32	-	Deformation due to thermal effect and irradiation	L	L	P	4	b06
33	-	Flow in reflector region	L	L	P	4	b07
34	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	4	b08
	B22	Lower Plenum					
35	-	Pressure loss	L	L	P	4	b09
36	-	Heat capacity	L	L	K	5	b10
37	-	Coolant mixing and thermal stratification	L	L	P	4	b11
38	-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	L	P	4	b12

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-2. Final PIRT with Priority for Further Investigation of SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
	B23	Upper Plenum					
39	-	Pressure loss	L	L	P	4	b13
40	-	Heat capacity	L	L	K	5	b14
41	-	Heat transfer between shielding plug and sodium	L	L	K	5	b15
42	-	Coolant mixing, thermal stratification, and thermal striping	L	L	P	4	b16
	B24	Vertical Shroud					
43	-	Radial heat transfer between inside and outside coolant through vertical shroud	L	L	P	4	b17
	B25	Radial Shield					
44	-	Flow in radial shield region	L	L	P	4	b18
45	-	Heat capacity	L	L	K	5	b19
46	-	Effect of generated heat by neutron capture and gamma rays	L	L	P	4	b20
47	-	Radial heat transfer in radial shield region	L	L	P	4	b21
	B26	Reactivity Control Drive Mechanism					
48	-	Shutdown velocity of reflector	L	L	K	5	b22
49	-	Shutdown velocity of shutdown rod	L	L	K	5	b23
	C	Primary Heat Transport System					
	C0	General					
50	-	Natural circulation	L	M	P	3	c01
51	-	Sodium inventory	L	L	K	5	c02
52	-	Heat capacity of coolant	L	L	K	5	c03

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-2. Final PIRT with Priority for Further Investigation of SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
	C1	IHX					
53	-	Pressure loss	L	L	P	4	c04
54	-	Heat transfer from primary coolant to intermediate coolant	N/A	N/A	P	N/A	c05
55	-	Primary flow rate	L	L	P	4	c06
56	-	Intermediate flow rate	N/A	N/A	P	N/A	c07
57	-	Heat capacity	L	L	K	5	c08
58	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	N/A	N/A	P	N/A	c09
	C2	Primary EMP					
59	-	Flow coastdown performance	L	L	P	4	c10
60	-	Pressure loss	L	L	P	4	c11
61	-	Pump head	L	L	K	5	c12
62	-	Heat capacity and joule heat at flow coastdown	L	N/A	P	4	c13
	E	Residual Heat Removal System					
	E2	RVACS					
63	-	Pressure loss in air flow path	L	L	K	5	e07
64	-	Heat transfer between GV wall and air	L	L	K	5	e08
65	-	Heat transfer between collector wall and air	L	L	K	5	e09
66	-	Heat transfer between concrete wall and air	L	L	K	5	e10
67	-	Thermal radiation between RV wall and GV wall	L	M	P	3	e11
68	-	Thermal radiation between GV wall and heat collector wall	L	M	P	3	e12
69	-	Thermal radiation between heat collector wall and concrete wall	L	M	P	3	e13
70	-	Asymmetric airflow	L	M	P	3	e14
71	-	Inlet air temperature range	L	L	K	5	e15

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-2. Final PIRT with Priority for Further Investigation of SLIP (cont.)

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature		Importance					
No.	Subsystem/ Component	Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
	F	Instrumentation and Control System					
	F1	Instrumentation and Control Equipment					
	F11	Plant Protection Sensors					
72	-	Delay of scram signal of primary EMP voltage and current	L	L	K	5	f01
73	-	Delay of scram signal of power line voltage	L	L	K	5	f02
74	-	Delay of scram signal of neutron flux	L	L	K	5	f03
75	-	Delay of scram signal of IHX primary outlet temperature	L	L	K	5	f04
	F12	Others					
76	-	Delay of interlock signal of SG outlet temperature	L	L	K	5	f05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-3. Final PIRT with Priority for Further Investigation of FCC

	Event: Failure of a Cavity Can (FCC)					
	Figures of Merit (FoM): Cladding Temperature					
No.	Subsystem/ Component	Phenomenon	Importance	SoK	Priority	Code
	A	Core and Fuel Assemblies				
1	-	Reactivity feedback	L	P	4	a04
2	-	Gap conductance between fuel and cladding	L	K	5	a05
3	-	Heat transfer between cladding and coolant	L	K	5	a06
4	-	Intra- and inter-assembly flow distribution	H	P	2	a07
5	-	Radial heat transfer between subassemblies (S/A<-->sodium<-->S/A)	M	P	3	a08
6	-	Heat capacity of core assemblies	L	K	5	a10
7	-	Coolant boiling	N/A	K	N/A	a11
8	-	Core power	L	K	5	a12
9	-	Decay heat	L	K	5	a13
10	-	Heat transfer between core support plate and sodium	L	P	4	a14
11	-	Rate of scram reactivity insertion	M	K	5	a15
12	-	Delay of scram reactivity insertion	M	K	5	a16
13	-	Eutectic reaction between fuel and cladding	L	P	4	a17
14	-	Temperature dependence of physical properties of materials	L	K	5	a18
15	-	Flow-induced vibration in a subassembly	L	P	4	a22
16	-	Inter-wrapper flow between wrapper tubes	L	P	4	a23
17	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	P	4	a24
18	-	Radial power distribution	L	K	5	a25
19	-	Axial power distribution	L	K	5	a26
20	-	Reactivity insertion by cavity can failure	M	P	3	a27

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-3. Final PIRT with Priority for Further Investigation of FCC (cont.)

	Event: Failure of a Cavity Can (FCC)					
	Figures of Merit (FoM): Cladding Temperature					
No.	Subsystem/ Component	Phenomenon	Importance	SoK	Priority	Code
	B	Reactor System				
	B1	Reactor Vessel				
	B2	Reactor Internal Structures				
	B20	General				
21	-	Coolant mixing effect in upper plenum including thermal stratification	L	P	4	b02
	B21	Reflector				
22	-	Flow in reflector region	L	P	4	b07
23	-	Effect of generated heat by neutron capture and gamma rays	L	P	4	b08
	B22	Lower Plenum				
24	-	Heat capacity	L	K	5	b10
25	-	Coolant mixing and thermal stratification	L	P	4	b11
26	-	Heat release from half-ellipse-shaped plate at lower end of reactor vessel	L	P	4	b12
	B23	Upper Plenum				
27	-	Heat capacity	L	K	5	b14
28	-	Heat transfer between shielding plug and sodium	L	K	5	b15
29	-	Coolant mixing, thermal stratification, and thermal striping	L	P	4	b16
	B24	Vertical Shroud				
30	-	Radial heat transfer between inside and outside coolant through vertical shroud	L	P	4	b17
	B25	Radial Shield				
31	-	Flow in radial shield region	L	P	4	b18
32	-	Heat capacity	L	K	5	b19
33	-	Effect of generated heat by neutron capture and gamma rays	L	P	4	b20
34	-	Radial heat transfer in radial shield region	L	P	4	b21

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-3. Final PIRT with Priority for Further Investigation of FCC (cont.)

	Event: Failure of a Cavity Can (FCC)					
	Figures of Merit (FoM): Cladding Temperature					
No.	Subsystem/ Component	Phenomenon	Importance	SoK	Priority	Code
	B26	Reactivity Control Drive Mechanism				
35	-	Shutdown velocity of reflector	M	K	5	b22
36	-	Shutdown velocity of shutdown rod	M	K	5	b23
	C	Primary Heat Transport System				
	C0	General				
37	-	Sodium inventory	L	K	5	c02
38	-	Heat capacity of coolant	L	K	5	c03
	C1	IHX				
39	-	Heat transfer from primary coolant to intermediate coolant	L	P	4	c05
40	-	Primary flow rate	L	P	4	c06
41	-	Intermediate flow rate	L	P	4	c07
42	-	Heat capacity	L	K	5	c08
43	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	L	P	4	c09
	D	Intermediate Heat Transport System				
	D0	General				
44	-	Heat removal from SG	L	K	5	d03
45	-	Heat transfer between upper plenum and intermediate coolant external to IHX	L	P	4	d04
	D1	Intermediate EMP				
	E	Residual Heat Removal System				

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-3. Final PIRT with Priority for Further Investigation of FCC (cont.)

	Event: Failure of a Cavity Can (FCC)					
	Figures of Merit (FoM): Cladding Temperature					
No.	Subsystem/ Component	Phenomenon	Importance	SoK	Priority	Code
	F	Instrumentation and Control System				
	F1	Instrumentation and Control Equipment				
	F11	Plant Protection Sensors				
46	-	Delay of scram signal of neutron flux	M	K	5	f03
47	-	Delay of scram signal of IHX primary outlet temperature	L	K	5	f04
	F12	Others				
48	-	Delay of interlock signal of SG outlet temperature	L	K	5	f05

Note:
H: High, M: Medium, L: Low, N/A: Not applicable, K: Known, P: Partially Known, U: Unknown

Table 9.1-4. Procedure for Further Investigation

	Theoretical Evaluation	Test
Priority 1 (None currently identified)	Test planning ➡	✓
Priority 2 and Priority 3	✓ ➡	Depends on results of theoretical evaluation
Priority 4 and Priority 5	None	None

Priority 1: Unknown

Priority 2: High importance and partially known

Priority 3: Medium importance and partially known

Priority 4: Low importance and partially known

Priority 5: Known

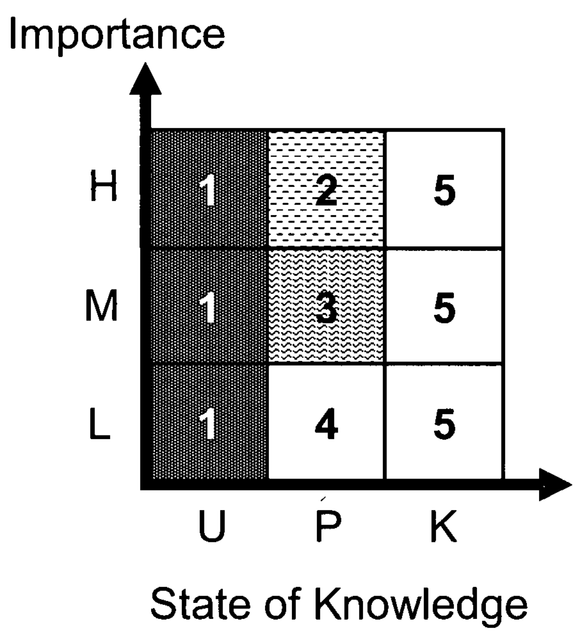

Figure 9.1-1. Priority for Further Investigation

9.2 LIST OF PHENOMENA REQUIRING FURTHER INVESTIGATION

Tables 9.2-1, 9.2-2, and 9.2-3 show only the phenomena with Priorities of 1 through 3 for each event. These tables are constructed by rearranging Tables 9.1-1, 9.1-2, and 9.1-3 to show the phenomena in order of priority.

Also, Tables 9.2-4, 9.2-5, and 9.2-6 show how phenomena selected are dealt in further investigation in the format of Table 9.1-4. Testing is required for phenomena with Priority 1. For phenomena with Priorities 2 and 3, as a first approach, investigation by the theoretical evaluation such as the analysis using the detailed analysis code shall be conducted. If thos investigations do not contribute to the improvement of knowledge sufficiency, however, testing shall be performed as a second approach. For phenomena with Priorities 4 and 5, firther knowledge improvement is judged to be not required.

The phenomena selected from each event for further investigation, and the contents of each investigation, are described below.

9.2.1 Loss of Offsite Power

In this event, no phenomena are assigned Priority 1. That means that there are no phenomena whose SoK is unknown for this event.

The phenomena assigned Priority 2 are discussed below. See Tables 9.1-1, 9.2-1, 9.2-4. The index to the left of the name of each phenomenon identifies the code used in the PIRT table.

(a07) Intra- and inter-assembly flow distribution (subsystem/component: core/fuel assemblies)

If the intra- and inter-assembly flow distribution deviates from the design specification, the coolant flow rate in each assembly varies from the expected value and this affects the cladding temperature.

The pressure loss coefficient of the intra- and inter-assemblies whose flow rate ranges from the rated flow rate to 30 percent of the rated flow rate can be estimated by the data obtained from tests [8-60] [8-61] using the same scale facility as 4S.

However, test data for the pressure loss coefficient in the core at 30 percent or less of the rated flow rate, using the same scale facility as 4S, have not been obtained.

In the 1st phase, because the flow rate decreased to 30 percent or less of the rated value, there is an uncertain region for the pressure loss coefficient.

Therefore, the characteristics of the pressure loss coefficient at low flow rates need to be clarified by further investigation. In addition to that, the characteristics of the flow distribution of inter-assembly at low flow rates also need to be clarified by further investigation.

(a24) Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies (subsystem/component: core/fuel assemblies)

This phenomenon considers flow distribution and heat removal with very low Re numbers. As described in "(a07) Intra- and inter-assembly flow distribution," if the flow distribution of the intra- and inter-assembly deviates from the design specification, the coolant flow rate in each assembly varies from the expected value and this affects the cladding temperature.

As described in "(a07) Intra- and inter-assembly flow distribution," the pressure loss coefficient of the intra- and inter-assemblies whose flow rate ranges from the rated flow rate to 30 percent of the rated flow rate can be estimated only by the data obtained from tests using the same scale facility as 4S.

Therefore, in the same way as (a07), the characteristics of the pressure loss coefficient at low flow rates need to be clarified by further investigation. In addition, the characteristics of inter-assembly flow distribution at low flow rates also need to be clarified by further investigation.

The phenomena assigned Priority 3 are discussed below. See Tables 9.1-1, 9.2-1, 9.2-4.

(a01) Pressure loss in core region (subsystem/component: core/fuel assemblies)

(Same as for "Intra- and inter-assembly flow distribution" of (a07))

(a03) Natural convection (subsystem/component: core/fuel assemblies)

This phenomenon considers the local natural convection occurring in the core. In the 1st phase, local natural convection does not occur because the effect of the EMP flow coastdown remains. In the 2nd phase, pressure loss may occur that affects the core flow rate by local natural convection of the core because the core flow rate is very small.

Flow behavior in and around the core, the behavior of local eddies, and the pressure loss attributed to the eddies at rated and natural circulation conditions can be evaluated analytically. However, for the low Re number regime, as in natural circulation, it is difficult to measure flow distribution and temperature distribution in and around the core in an experiment, so there are limited data to verify the analysis results.

This phenomenon is significantly affected by the pressure loss and radial and axial temperature distribution of the core.

Therefore, the characteristics and behavior of local eddies around the core, "Intra- and inter-assembly flow distribution" and "Radial heat transfer between subassemblies" need to be clarified by further investigation.

(a08) Radial heat transfer between subassemblies (subsystem/component: core/fuel assemblies)

This phenomenon considers the heat transfers from a high-temperature assembly through the sodium coolant into a low-temperature assembly.

Thermal conduction is the dominant effect because the gaps between assemblies are narrow and the effect of convection is small, as shown by the result of the sensitivity analysis (Figure 8.3.3.1-1).

However, the calculation model used in the sensitivity analysis is not so detailed that uncertainty of the result is reduced. For example, there might be the effect of natural convection as in (a03).

Therefore, the effect of "Radial heat transfer between subassemblies" at low flow rates needs to be clarified by further investigation.

(a23) Inter-wrapper flow between wrapper tubes (subsystem/component: core/fuel assemblies)

This phenomenon considers flow in the gaps between assemblies.

Although data exist for sodium flow behavior in the gap between fuel assemblies at natural circulation and their flow resistance, the data are highly dependent on configuration of the test system. Therefore, it is difficult to apply these test data to the 4S system from the aspect of scaling and changes of geometric shape.

Therefore, the characteristics of coolant flow behavior in the fuel assembly gaps in the natural circulation state, and the flow resistance, need to be clarified by further investigation.

(b02) Coolant mixing effect in upper plenum including thermal stratification (subsystem/ component: reactor internal structures/general)

This phenomenon considers the mixing effect of coolant in the upper plenum including thermal stratification.

After the EMP trips and the reactor is scrammed, low-temperature coolant flows into the upper plenum where high-temperature coolant remains and thermal stratification occurs. In this period, the degree to which coolant is mixed by natural circulation is the important factor that determines coolant temperature, and has a large effect on cladding temperature.

Although sufficient data exist for mixing effect by test using water as a coolant, there are no test data using sodium, and to apply water data to systems using sodium is difficult from the aspects of scaling for different geometric shapes and consideration of the different physical properties.

Also, in the region of low Re numbers, the measurement data in the water tests have an uncertainty because it is difficult to measure the mixing effect of coolant. The Re number in the natural circulation condition for 4S is extremely small. Therefore, it is difficult to apply the water test results to 4S. Furthermore, there is little knowledge for the 4S system regarding thermal stratification during natural circulation.

Therefore, the mixing characteristics of coolant in the upper plenum need to be clarified by further investigation.

(b04) Natural convection (subsystem/component: reactor internal structures/general)

This phenomenon considers the local natural convection around the reactor internal structures. In the 1st phase, local natural convection is unlikely to occur, because the effect of the EMP flow coastdown remains, as the natural convection in the core described in (a03). In the 2nd phase, since the coolant flow rate is very small, local natural convection around the reactor internal structures may cause a pressure loss that affects the core flow rate.

Flow behavior around the reactor internal structures, the behavior of local eddies, and the pressure loss caused by local eddies at rated conditions and the natural circulation state can be evaluated analytically. However, in low Re number regimes such as natural circulation, it is difficult to measure in an experiment the flow distribution and temperature distribution around the reactor internal structures, and there are limited data to verify the analysis results.

There is a high possibility that this phenomenon occurs, for example, at the reflector and the radial shield where heat capacity is high, because this event is significantly affected by the geometric shape and radial and axial temperature distribution.

Therefore, the temperature distribution of the reactor internal structures at low flow rates, and the effect of local eddies attributed to the temperature distribution, need to be clarified by further investigation.

(c01) Natural circulation (subsystem/component: primary heat transport system/general)

This phenomenon considers natural circulation of the coolant in the primary system and closely relates to the phenomena described in (a07), (a24), (a01), (a03), (a08), (a23), (b02), and (b04).

In the 1st phase, since the effect of the EMP flow coastdown remains, natural circulation behavior in the primary system is not noticeable.

In the 2nd phase, natural circulation flow rate in the primary system is the dominant factor that determines the amount of heat removal from the cladding. Many tests for natural circulation have been conducted in the past and there is sufficient knowledge. Since natural circulation has a strong dependency on geometric shape of the flow path, however, application of the measurement data to the 4S contains uncertainty.

The natural circulation flow rate can be analytically evaluated because it is determined by the heat balance between the core, which is the heat source, and the IHX, which is the heat sink. As described in (b04), however, the pressure loss and radial and axial temperature distribution for the natural circulation regimes have uncertainty. Therefore, it is difficult to properly evaluate the pressure loss caused by local eddies in and around each reactor internal structure or in the core at natural circulation. In addition, the reactor internal structures have a role of both heat source and heat sink due to its high heat capacity. Therefore, this effect should be considered as well.

Based on the above, it is difficult to quantitatively evaluate the flow rate of natural circulation by simple methods.

Therefore, the characteristics of natural circulation in the primary system at low flow rates need to be clarified by the further investigation.

(d02) Natural circulation (subsystem/component: intermediate heat transport system/general)

This phenomenon considers the natural circulation of intermediate system coolant. The natural circulation flow rate in the intermediate system is determined by the heat balance between the IHX, which is the heat source, and IRACS, which is the heat sink.

As described for natural circulation in (c01), although there is sufficient knowledge, the application of the measurement data to 4S contains some uncertainties. Also, as described in the following (e03), the characteristics for heat transfer at the air side of AC has uncertainty as well, and the effect is large.

Therefore, the characteristics of natural circulation of intermediate system natural circulation at low flow rates need to be clarified by further investigation, along with characteristics of the heat transfer at the air side of AC.

(e03) Heat transfer between tube and air (subsystem/component: residual heat removal system/Air cooler of IRACS)

This phenomenon considers the heat transfer between heat transfer tube in the IRACS and air.

There is considerable knowledge regarding the heat transfer coefficient at rated conditions. However, the flow behavior at the air side that affects the heat transfer has a large uncertainty because it is determined by the open/close status of the dampers and clogging of the filter placed at the air inlet.

Therefore, the characteristics of heat transfer at the air side of IRACS need to be clarified by further investigation.

Based on the above, in this event, the characteristics of natural circulation in the primary system and phenomena related to the characteristics of heat transfer at the air side of the IRACS were selected as objects for further investigation.

9.2.2 Sodium Leakage from Intermediate Piping

In this event, no phenomena are assigned to Priority 1, meaning that there are no phenomena whose SoK is unknown.

The phenomena assigned Priority 2 are discussed below. See Tables 9.1-2, 9.2-2, 9.2-5.

The details of the following phenomena were explained in Section 9.2.1 for the LOSP and are not repeated.

(a07) Intra- and inter-assembly flow distribution (subsystem/component: core/fuel assemblies)

(a24) Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies (subsystem/component: core/fuel assemblies)

The phenomena assigned Priority 3 are discussed below. See Tables 9.1-2, 9.2-2, 9.2-5.

The details of the following phenomena (a01), (a03), (a08), (a23), (b02), (b04), and (c01) were explained in Section 9.2.1 for the LOSP and are not repeated.

(a01) Pressure loss in core region (subsystem/component: core/fuel assemblies)

(a03) Natural convection (subsystem/component: core/fuel assemblies)

(a08) Radial heat transfer between subassemblies (subsystem/component: core/fuel assemblies)

(a23) Inter-wrapper flow between wrapper tubes (subsystem/component: core/fuel assemblies)

(b02) Coolant mixing effect in upper plenum including thermal stratification (subsystem/ component: reactor internal structures/general)

(b04) Natural convection (subsystem/component: reactor internal structures/general)

(c01) Natural circulation (subsystem/component: primary heat transport system/general)

(e11) Thermal radiation between RV wall and GV wall (subsystem/component: RHRS/RVACS)

This phenomenon considers radiation as it contributes to the heat removal of the RVACS, which is one of the RHRS components that actively utilizes the effect of radiation. However, there is uncertainty in the emissivity of the structural materials such as the RV, GV, and heat collector that affects the efficiency of radiation. Specifically, there are no data considering aging effects for a long period such as the 30-year operating period for the 4S.

Therefore, the characteristics for aging deterioration of emissivity need to be clarified by further investigation.

(e12) Thermal radiation between GV wall and heat collector wall (subsystem/component: RHRS/RVACS)

(Same as for "Thermal radiation between RV wall and GV wall," (j))

(e13) Thermal radiation between heat collector wall and concrete wall (subsystem/ component: RHRS/RVACS)

(Same as for "Thermal radiation between RV wall and GV wall," (j))

(e14) Asymmetric airflow (subsystem/component: RHRS/RVACS)

This phenomenon considers asymmetric airflow in the annulus flow path of the RVACS. The geometry of the RVACS airflow path itself is point-symmetric to the center point of a horizontal cross-section of the reactor vessel. However, there is the possibility of generation of asymmetric airflow in the annulus flow path attributed to the misalignment of RVACS position at installation and the clogging of the filter at the air inlet. As a result, this asymmetric flow might cause generation of local eddies and an increase in pressure loss.

Therefore, the airflow characteristics of the RVACS with consideration of sodium flow in the RV and characteristics of heat removal from the RV by RVACS shall be clarified by further investigation.

Based on the above, although many of the phenomena selected as objects selected for further investigation in this event overlap with those for the LOSP, radiation of RVACS and asymmetry of the airflow path were selected for further investigation.

9.2.3 Failure of a Cavity Can

In this event, no phenomena are assigned Priority 1. That means that there are no phenomena whose SoK is unknown for this event.

The phenomena assigned Priority 2 are discussed below. See Tables 9.1-3, 9.2-3, 9.2-6.

The details of both the following phenomena were explained in Section 9.2.1 for the LOSP and are not repeated.

(a07) Intra- and inter-assembly flow distribution (subsystem/component: core/fuel assemblies)

The phenomena assigned Priority 3 are discussed below. See Tables 9.1-3, 9.2-3, 9.2-6.

(a08) Radial heat transfer between subassemblies (subsystem/component: core/fuel assemblies)

(a27) Reactivity insertion by cavity can failure (subsystem/component: core/fuel assemblies)

This phenomenon considers the reactivity insertion by the cavity can failure and is specific to this event. The amount of reactivity insertion is limited by the volume of one cavity can as the event assumes the failure of one out of 36 cavity cans. However, uncertainty of the reactivity insertion rate is large because several potential failure patterns can be assumed because of manufacturing defects and other factors.

Therefore, the characteristics of the failure patterns need to be clarified by further investigation.

Based on the above, although most of the phenomena selected for further investigation in this event overlap with phenomena selected for the LOSP, reactivity insertion by cavity can failure, the phenomenon specific to this event, was selected for further investigation.

Table 9.2-1. Rearranged Final PIRT Results for LOSP

		Event: Loss of Offsite Power (LOSP)						
		Figures of Merit (FoM): Cladding Temperature		Importance				
No.	Subsystem/Component		Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
-	Highly ranked phenomena with partially known SoK							
7	A	Core/Fuel Assemblies	Intra- and inter-assembly flow distribution	H	L	P	2	a07
24	A	Core/Fuel Assemblies	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	2	a24
-	Moderately ranked phenomena with partially known SoK							
1	A	Core/Fuel Assemblies	Pressure loss in core region	M	L	P	3	a01
3	A	Core/Fuel Assemblies	Natural convection	L	M	P	3	a03
8	A	Core/Fuel Assemblies	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	M	P	3	a08
23	A	Core/Fuel Assemblies	Inter-wrapper flow between wrapper tubes	L	M	P	3	a23
28	B	Reactor System	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	3	b02
30	B	Reactor System	Natural convection	L	M	P	3	b04
50	C	Primary Heat Transport System	Natural circulation	L	M	P	3	c01
64	D	Intermediate Heat Transport System	Natural circulation	L	M	P	3	d02
73	E	Residual Heat Removal System	Heat transfer between tube and air	L	M	P	3	e03

Table 9.2-2. Rearranged Final PIRT Results for SLIP

	Event: Sodium Leakage from Intermediate Piping (SLIP)							
	Figures of Merit (FoM): Cladding Temperature			Importance				
No.	Subsystem/Component		Phenomenon	1st Phase	2nd Phase	SoK	Priority	Code
-	Highly ranked phenomena with partially known SoK							
7	A	Core/Fuel Assemblies	Intra- and inter-assembly flow distribution	H	L	P	2	a07
24	A	Core/Fuel Assemblies	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	L	H	P	2	a24
-	Moderately ranked phenomena with partially known SoK							
1	A	Core/Fuel Assemblies	Pressure loss in core region	M	L	P	3	a01
3	A	Core/Fuel Assemblies	Natural convection	L	M	P	3	a03
8	A	Core/Fuel Assemblies	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	M	P	3	a08
23	A	Core/Fuel Assemblies	Inter-wrapper flow between wrapper tubes	L	M	P	3	a23
28	B	Reactor System	Coolant mixing effect in upper plenum including thermal stratification	L	M	P	3	b02
30	B	Reactor System	Natural convection	L	M	P	3	b04
50	C	Primary Heat Transport System	Natural circulation	L	M	P	3	c01
67	E	Residual Heat Removal System	Thermal radiation between RV wall and GV wall	L	M	P	3	e11
68	E	Residual Heat Removal System	Thermal radiation between GV wall and heat collector wall	L	M	P	3	e12
69	E	Residual Heat Removal System	Thermal radiation between heat collector wall and concrete wall	L	M	P	3	e13
70	E	Residual Heat Removal System	Asymmetric airflow	L	M	P	3	e14

Table 9.2-3. Rearranged Final PIRT Results for FCC

	Event: Failure of a Cavity Can (FCC)						
	Figures of Merit (FoM): Cladding Temperature			Importance	SoK	Priority	Code
No.	Subsystem/Component		Phenomenon				
-	Highly ranked phenomena with partially known SoK						
4	A	Core/Fuel Assemblies	Intra- and inter-assembly flow distribution	H	P	2	a07
-	Moderately ranked phenomena with partially known SoK						
5	A	Core/Fuel Assemblies	Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A)	M	P	3	a08
20	A	Core/Fuel Assemblies	Reactivity insertion by cavity can failure	M	P	3	a27

Table 9.2-4. Further Investigation for Phenomena with Higher Priority for LOSP

	Theoretical Evaluation	Test
Priority 1	None currently identified	
Priority 2 and Priority 3	• Intra- and inter-assembly flow distribution • Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies • Pressure loss in core region • Natural convection (in core/fuel assemblies) • Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A) • Inter-wrapper flow between wrapper tubes • Coolant mixing effect in upper plenum including thermal stratification • Natural convection (in reactor system) • Natural circulation (in primary heat transport system) • Natural circulation (in intermediate heat transport system) • Heat transfer between tube and air	Depends on results of theoretical evaluation
Priority 4 and Priority 5	79 phenomena currently are identified. However, further investigation is not planned.	

Table 9.2-5. Further Investigation for Phenomena with Higher Priority for SLIP

	Theoretical Evaluation	**Test**
Priority 1	None currently identified	
Priority 2 and Priority 3	• Intra- and inter-assembly flow distribution • Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A) • Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies • Pressure loss in core region • Natural convection (in core/fuel assemblies) • Inter-wrapper flow between wrapper tubes • Coolant mixing effect in upper plenum including thermal stratification • Natural convection (in reactor system) • Natural circulation (in primary heat transport system) • Thermal radiation between RV wall and GV wall • Thermal radiation between GV wall and heat collector wall • Thermal radiation between heat collector wall and concrete wall • Asymmetric airflow	Depends on results of theoretical evaluation
Priority 4 and Priority 5	63 phenomena currently are identified. However, further investigation is not planned.	

Table 9.2-6. Further Investigation for Phenomena with Higher Priority for FCC

	Theoretical Evaluation	**Test**
Priority 1	None currently identified	
Priority 2 and Priority 3	• Intra- and inter-assembly flow distribution • Radial heat transfer between subassemblies (S/A <--> sodium <--> S/A) • Reactivity insertion by cavity can failure	Depends on results of theoretical evaluation
Priority 4 and Priority 5	45 phenomena currently are identified. However, further investigation is not planned.	

10 SUMMARY

In this report, the PIRT process was applied to planning of the further investigation of the 4S.

The purposes of this application were (1) to classify the expected phenomena for the 4S according to the importance and SoK, (2) to set the priority for further investigation to improve the knowledge level, and (3) to clarify the contents of the further investigation to be conducted according to those priorities.

The 4S PIRT has focused on confirming the importance and current knowledge for the performance of safety-related structures, systems, and components (SSCs) of the 4S.

The important findings obtained by application of the PIRT process are as follows.

- In the 4S PIRT, events related to the function of the RPS were defined from the set of AOOs and DBAs focusing on safety-related systems. The following three events were selected: loss of offsite power (LOSP), sodium leakage from intermediate piping (SLIP), and failure of a cavity can (FCC).

- To identify the relevant phenomena for the events above, the 4S plant system was classified into the following five groups from the viewpoint of thermal-hydraulic behavior:

  - Core and fuel assemblies, reactor system, primary heat transport system (PHTS), intermediate heat transport system (IHTS), and residual heat removal system (RHRS).

    Also, one additional subsystem was included from the viewpoint of RPS performance assurance: instrumentation and control (I&C) system.

    Subsequently, those subsystems were partitioned into components and subcomponents.

- Cladding temperature was defined as the Figure of Merit (FoM) to establish the rankings for importance of phenomena. This FoM was selected from the viewpoint of integrity of the fuel pin and integrity of the primary cooling boundary. This is common for all three events that were selected.

- Phenomena were identified for each subsystem and component. The numbers of phenomena identified for each subsystem are as follows.

Core/fuel assemblies:	27 phenomena
Reactor system:	23 phenomena
PHTS:	13 phenomena
IHTS:	8 phenomena
RHRS:	15 phenomena
Instrumentation and control system:	5 phenomena

- The rankings for the importance level of phenomena and the SoK were established through the following two steps.

  (1) Preliminary PIRT rankings were established based on information that was currently available, including opinions of specialists.

  (2) The rankings were considered and re-evaluated in reference to the results of sensitivity analyses conducted using the safety analysis code ARGO.

- The phenomena were ranked for relative importance using the following four levels according to how much the phenomena affected the FoM.

  High (H): High impact on FoM
  Medium (M): Moderate impact on FoM
  Low (L): Low impact on FoM
  Not applicable (N/A): No or insignificant impact on FoM

  In addition, the phenomena were ranked for knowledge using three levels.

  Known (K): Small uncertainty in test data and analytical modeling
  Partially Known (P): Moderate uncertainty in test data and analytical modeling
  Unknown (U): Very limited or no knowledge and large uncertainty in test data and analytical modeling

- By using the PIRT results, the priority for further investigation to be implemented for each phenomenon is specified according to a combination of the levels of importance and knowledge. Five levels of priority are specified as shown in Figure 9.1-1.

  Priority 1: Unknown (regardless of importance)
  Priority 2: High importance/partially known
  Priority 3: Medium importance/partially known
  Priority 4: Low importance/partially known
  Priority 5: Known (regardless of importance)

  Here, 16 phenomena were ranked in either Priority 1, 2, or 3, and were selected for further investigation. The total number of 0, 2, and 14 is assigned to Priorities 1, 2, and 3, respectively.

- The 16 phenomena selected are roughly classified as follows:

  - Coolant flow in core in very low Re number (pressure loss, flow distribution, local natural convection, radial heat transfer, and inter-wrapper flow)

  - Coolant mixing effect in upper plenum in very low Re number

  - Natural convection around reactor internal structure in very low Re number

- Natural circulation in primary and intermediate system in very low Re number
- Heat transfer in air side of IRACS
- Radiation heat transfer in RVACS
- Asymmetric airflow in RVACS
- Reactivity insertion by cavity can failure

- As regards the further investigation, analyses by a detailed analysis code such as a CFD code or testing are scheduled to be implemented. Measures for the phenomena subject to further investigation (Priority 1, 2, and 3) are as follows.
  - Priority 1: No knowledge of the phenomena with this priority exists and the ranking result of the importance itself might contain a large uncertainty. Therefore, theoretical evaluation is not conducted and knowledge improvement shall be attempted only by testing. However, some preliminary evaluations by analysis shall be conducted for the design of the test facility.
  - Priorities 2 and 3: The uncertainty of the analysis model is not so large because the phenomenon is partly known, and it can be said that these phenomena are generally recognized. Therefore, detailed investigation using an analysis code is deemed more appropriate and easier for improving knowledge, and shall be attempted as the initial approach. If adequate results cannot be achieved by this approach, however, implementation of testing as a second approach shall be considered.

As described above, the PIRT described in this report is established to be used when making a concrete plan for future testing and theoretical evaluation. If new knowledge is obtained in the further investigation, the contents of the 4S PIRT will be improved and expanded. Also, the 4S PIRT will aid in development and validation of the analysis code.

APPENDIX A
SUFFICIENCY OF SELECTED PHENOMENA: EVENT AND PHENOMENA MATRIX

The objectives of the 4S PIRT project are as follows:

1. Classify the phenomena expected in the 4S by the level of importance and SoK.

2. For the categories in item 1, set the priority for further investigation to be implemented to expand the SoK.

3. Based on the priority determined in item 2, clarify the content of the test or analyses to be implemented in the near future.

This PIRT focuses on identifying the relative importance and associated SoK of phenomena related to the performance of safety-related systems in the 4S. In the 4S PIRT, the important safety systems are the RPS. In this chapter, plausible phenomena are identified using tables similar to that used in Chapter 7 for the event whose sequence is different from LOSP, FCC, and SLIP events. The phenomena are compared with the phenomena identified for those three events. It is confirmed that the identified phenomena in the three events are sufficient as the phenomena that indicate the validity for the design of RPS. The ATWS events do not affect the RPS because they are assumed to be inoperative during these events; these additional phenomena will be addressed in a future PIRT report.

A.1 EVENT SELECTION TO INVESTIGATE THE SUFFICIENCY OF PHENOMENA

For the 4S, 17 types of events below are selected as AOOs and DBAs by using the failure modes and effects analysis (FMEA) and master logic diagram (MLD) for the SSCs.

1. Reactivity insertion by uncontrollable motion of segments of reflector at full-power operation

2. Reactivity insertion by uncontrolled motion of segments of reflector at startup

3. Decrease of primary coolant flow

4. Decrease of intermediate coolant flow

5. Increase of primary coolant flow

6. Increase of intermediate coolant flow

7. Inner or outer tube failure of steam generator

8. Increase of feedwater flow

9. Decrease of feedwater flow

10. Loss of offsite power (LOSP)

11. Rapid motion of segments of reflector at startup

12. Failure of a cavity can (FCC)

13. Reactor vessel leakage

14. One primary EMP failure (instantaneous loss of power to one pump, equivalent to a mechanical pump seizure)

15. Sodium leak from intermediate piping (SLIP)

16. Local faults in a fuel assembly

17. Primary cover gas boundary failure

In the list above, 1, 2, and 11 are events involving reactivity insertion, where the reactor scrams after the increase in power and primary coolant temperature, and the event sequence after the scram is almost the same as an FCC. Therefore, FCC (Event 12) is selected as the representative event.

Events 3 through 9 and 14 are the events in which the reactor scrams after the increase of primary and intermediate system temperature and decrease of primary and intermediate flow rate, and the event sequence after the scram is almost the same as LOSP. Therefore, LOSP (Event 10) is selected as the representative event.

Event 15, SLIP, is also selected.

The remaining events, which follow a sequence different from the three selected events, are 13 (reactor vessel leakage), 16 (local faults in a fuel assembly), and 17 (primary cover gas boundary failure). The initiator and the main event sequences of these remaining three events are described in this section. Also, the plausible phenomena of the three events are compared with plausible phenomena of LOSP, SLIP, and FCC chosen for the 4S PIRT.

In addition to the remaining three events, ATWS events and the sodium-water reaction event are selected to investigate the sufficiency of plausible phenomena that are identified for LOSP, SLIP, and FCC. The ATWS event is a BDBA as described in SRP 15.0, and the occurrence frequency of sodium-water reaction event is very low because double-wall tubes are adopted as heat transfer tube in the SG. These BDBAs are selected in this Appendix, however, because they have received much attention in the fast reactor field.

The additional BDBAs are as follows.

- ATWS

  - ✓ Unprotected loss of flow (ULOF)
  - ✓ Unprotected transient of overpower (UTOP)
  - ✓ Unprotected loss of heat sink (ULOHS)

- Sodium/water reaction due to steam generator tube failure

A.2 REACTOR VESSEL LEAKAGE

A.2.1 Initiators and Sequence of Reactor Vessel Leakage

(1) Initiators

A sodium leak from the reactor vessel is caused by a coolant boundary failure of the vessel. A leak caused by chemical corrosion of reactor vessel has an estimated frequency of occurrence less than that of an extremely unlikely event because the chemical composition of reactor vessel material is compatible with the primary coolant.

Also, since the primary coolant is contained in a sealed reactor vessel during operation, it is very unlikely that impurity incorporation into sodium occurs after the pre-start up sodium purification. Therefore, chemical corrosion of the reactor vessel material caused by a compound of sodium and oxygen has an estimated frequency of occurrences less than that of an extremely unlikely event. Thus, the occurrence frequency of a reactor vessel leak is considered to be very low, but nevertheless the leak is assumed as a passive component failure.

(2) Event Sequence

When the reactor vessel fails below the sodium level and sodium leaks, the primary sodium leak is detected by the sodium leak detector inside the guard vessel or by the low level signal of the reactor vessel sodium level gauge. When the decrease of sodium level is detected by primary sodium leak detection, the reactor is caused to scram at once.

When the scram signal of low sodium level in the reactor vessel is transmitted, the reactor shuts down automatically. The scram signal causes the reflector and shutdown rod to lower and the primary EMPs, intermediate EMP, and feedwater pumps trip. The flow coastdown system gradually decreases primary and intermediate coolant flow after pump trip. Primary flow decreases to 50 percent of rated flow after 30 seconds from pump trip and reaches natural circulation condition after 60 seconds.

At some stage, the sodium leak will stop because the gas pressures in the reactor vessel and guard vessel will be equalized. Also, even if the guard vessel is filled with sodium, primary coolant circulation is maintained because the sodium level will still be higher than the IHX inlet.

Residual heat can be removed by natural circulation in the RVACS and IRACS. Also, since the space between the reactor vessel and guard vessel is filled with sodium, thermal conduction between the RV and GV is improved, and the heat removal performance of the RVACS is also improved.

Leaked sodium is retained in the GV. Sodium combustion is controlled because the space between reactor vessel and guard vessel is filled with inert nitrogen gas. Although sodium reacts with residual oxygen and residual water vapor, the GV is designed to retain its structural integrity even if the GV temperature and pressure are raised by this reaction.

In the RV leakage event, the source terms are normally primary coolant and primary cover gas. The largest source term condition is that RV leakage occurs when the reactor maintains operation with failed fuel until the failure limit is reached. In this case, volatile fission products (FPs) are transferred from the coolant that flows into the GV and to the gas space in the GV. These volatile FPs transfer to the top dome by passing through the seal of the GV. From the inside of the top dome, the volatile FPs are then leaked to the reactor building with activated primary sodium vapor. Calculating the amount of radioactive material released outside based on the design leak ratio of the seal and the design leak ratio of the top dome, even if 1 percent of fuel failure is assumed, exposure investigation at the site boundary for 30 days is 7×10^{-4} rem or less. This is significantly less than the criterion value of 25 rem [A-2] [A-3].

A.2.2 Phenomena Relating to the Reactor Vessel Leakage

Table A.2.2-1 shows phenomena relating to this event. Table A.2.2-2 shows the hierarchy of regulatory requirements and examples of criteria. The boundary that contains the radioactive material is described in level 3 and 4. The reactor vessel leakage is the event in which the primary coolant boundary is failed and radioactive material is released. Therefore, the FoM is applied for exposure. If the FoM is applied to exposure, phenomena relating to FP migration behavior at primary coolant leakage become important. The phenomena relating to this event are described below.

- FP release from fuel slug into gas plenum

  This phenomenon relates to the behavior of FPs transferring from the fuel slug to the gas plenum in the fuel pin. In this event, the amount of FPs in the primary coolant affects the exposure because primary sodium leaks into the guard vessel followed by the FPs being released into the reactor building though the top dome. The amount of FPs in the primary coolant depends on the amount of migration of both the FPs in the fuel gas plenum and FPs remaining in the fuel slug into the primary coolant. The migration behavior of FPs in the fuel gas plenum after cladding failure differs from that of FPs retained in the fuel slug. This phenomenon affects the amount of FP migration into the primary coolant because the behavior of the FPs in the fuel gas plenum differs from the behavior of the FPs remaining in the fuel slug if the cladding fails.

- FP transport from fuel to sodium bond and sodium in primary system

  This phenomenon relates to the behavior of the FPs that transfer from the fuel pin into the sodium bond and the primary coolant by fuel pin failure. In this event, primary coolant sodium leaks into the guard vessel, and the amount of FPs in the primary coolant affects the exposure. Therefore, this phenomenon affects the amount of FPs in the primary coolant when fuel failure occurs.

- FP transport from sodium in primary system to cover gas

  This phenomenon relates to the behavior of the FPs that transfer from the coolant into the primary cover gas. The migration behavior to the cover gas of FPs in the primary coolant is affected by the amount of FPs in the primary coolant, and this phenomenon affects exposure.

- Sodium inventory in primary boundary

  This phenomenon relates to the amount of primary sodium or primary sodium level. The sodium level affects the amount of leakage, depending on the location of the reactor vessel failure. Therefore, a difference in the leak level affects exposure. Sodium level in the primary system is determined by the amount, temperature, and density of primary sodium.

- Decrease of cover gas pressure in reactor vessel

  Variation of cover gas pressure affects the rate of sodium leakage.

- Pressure loss in airflow path (RVACS)

  The RVACS pressure loss is determined by the geometry and wall friction in the natural circulation flow path at outside of the GV. After sodium leakage from RV to GV, heat transfer between the RV and GV improves due to sodium between the RV and GV. Therefore, GV temperature rises, the amount of heat removal increases, and natural circulation flow rate changes.

- Heat transfer (between the GV wall and air, between collector wall and air, between concrete wall and air)

  This phenomenon relates to heat transfer between the GV and collector, or between concrete and air and is dependent on the geometry and flow velocity of the air. After sodium leakage, heat transfer between the RV and GV improves. Therefore, GV temperature increases and the amount of residual heat removal increases. As a result, the heat transfer rate between GV and air, collector and air, and concrete and air also will be increased.

- Thermal radiation (between the GV wall and collector wall, between the collector wall and concrete)

  This phenomenon relates to the heat transfer by radiation between the GV and collector, and between the collector and concrete, and it is dependent on geometry, radiation coefficient (emissivity), and temperature. After sodium leakage, heat transfer between the RV and GV improves. Therefore, GV temperature increases, and the heat transfer rate by radiation between GV and collectors and collector and concrete increases. Thermal radiation between the RV wall and GV wall is not applicable because leaked sodium fills the space between RV and GV.

- Asymmetric airflow

  This phenomenon relates to asymmetric flow caused by the temperature distribution in the reactor and by geometry. After sodium leakage, heat transfer between the RV and GV improves, and therefore GV temperature rises. Asymmetric flow is dependent on the condition of the leaked sodium as well.

- Delay of scram signal of reactor sodium level meter

  The reactor sodium level instrumentation measures sodium level in the reactor vessel and, therefore, the occurrence of primary sodium leakage is detected. If the scram signal of sodium level is delayed, reactor shutdown is also delayed. However, even if reactor shutdown is delayed, the delay does not have a major impact on fuel integrity since it does not degrade the reactor cooling performance. Delay of scram might affect the initial leak rate because it may affect the pressure of primary coolant.

- Radiation monitor

  This phenomenon relates to the detection of radioactive material. After primary sodium leakage, the increase of radioactivity in the top dome and in the reactor building is measured by the radiation monitors.

- Sodium leakage monitor

  This phenomenon relates to the detection of sodium leakage. Primary sodium leakage in the guard vessel is detected by the sodium monitor.

- Cover gas pressure gauge

  This phenomenon relates to detection of changes in the cover gas pressure of the primary system. In the event of primary sodium leakage, the cover gas pressure gauge detects the decrease in cover gas pressure.

- Failed fuel detection (reactor cover gas radiation monitoring)

  This phenomenon relates to detection of a fuel failure by measuring the FPs that transfer from the fuel into the cover gas space when a fuel pin is failed. When sodium is leaked, exposure is affected depending on the occurrence of fuel failure and the number of failures.

- Functional integrity of equipment for preventing sodium fire

  This phenomenon relates to the function of equipment that prevents sodium combustion. For reactor vessel leakage, sodium combustion is prevented by inerting the space between the RV and the GV with a nitrogen gas. Integrity of this function affects the sodium burning capacity.

- Sealing integrity, overall leakage testability (top dome)

  This phenomenon relates to the sealing performance of the top dome. When sodium leaks from the reactor vessel, radioactive material is transported from the space between the GV and the RV through a seal into the top dome. Therefore, exposure is affected by the sealing performance of the top dome.

- Integrity of isolation function of containment system

  This phenomenon relates to the isolation function of the top dome. When sodium leaks from the reactor vessel, radioactive material is transported from the space between the GV and RV through a seal into the top dome. Therefore, exposure is affected by the isolation function of the top dome.

- Irradiation dose of GV

  This phenomenon relates to the neutron irradiation of GV during operation. This phenomenon affects the retention capability of leaked sodium after a leak. The range of neutron irradiation dose for the GV is investigated using proven shield calculations. The GV is designed to ensure its structural integrity based on the investigation results of the neutron irradiation dose.

- Sealing integrity of GV

  This phenomenon relates to the integrity of the seal of GV. When sodium leaks from the RV, radioactive material is released from the GV to the top dome. Therefore, the sealing performance of the GV affects exposure.

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events

				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
A	Core/Fuel Assemblies			—	—	—	—	—	—
-	-	-	Pressure loss in core region	✓	✓	✓	N/A	✓	N/A
-	-	-	Pressure loss in reflector region	✓	✓	✓	N/A	N/A	N/A
-	-	-	Natural convection	✓	✓	✓	N/A	✓	N/A
-	-	-	Reactivity feedback	✓	✓	✓	N/A	✓	N/A
-	-	-	Gap conductance between fuel and cladding	✓	✓	✓	N/A	✓	N/A
-	-	-	Heat transfer between cladding and coolant	✓	✓	✓	N/A	✓	N/A
-	-	-	Intra- and inter-assembly flow distribution	✓	✓	✓	N/A	✓	N/A
-	-	-	Radial heat transfer between subassemblies (S/A <--> sodium <-->S/A)	✓	✓	✓	N/A	✓	N/A
-	-	-	Heat transfer between reflector and coolant	✓	✓	✓	N/A	✓	N/A
-	-	-	Heat capacity of the core assemblies	✓	✓	✓	N/A	✓	N/A
-	-	-	Coolant boiling	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Core power	✓	✓	✓	N/A	✓	N/A
-	-	-	Decay heat	✓	✓	✓	N/A	✓	N/A
-	-	-	Heat transfer between core support plate and sodium	✓	✓	✓	N/A	N/A	N/A
-	-	-	Rate and delay of scram reactivity insertion	✓	✓	✓	N/A	✓	N/A
-	-	-	Eutectic reaction between fuel and cladding	✓	✓	✓	N/A	✓	N/A
-	-	-	Temperature dependence of physical properties of materials	✓	✓	✓	N/A	✓	N/A
-	-	-	FP release from fuel slug into gas plenum	N/A	N/A	N/A	✓	✓	✓
-	-	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	N/A	✓	✓	✓
-	-	-	FP transport from sodium in primary system to cover gas	N/A	N/A	N/A	✓	✓	✓
-	-	-	Flow-induced vibration in a subassembly	✓	✓	✓	N/A	✓	N/A
-	-	-	Inter-wrapper flow between wrapper tubes	✓	✓	✓	N/A	✓	N/A
-	-	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	✓	✓	✓	N/A	✓	N/A

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
-	-		Radial power distribution	✓	✓	✓	N/A	N/A	N/A
-	-		Axial power distribution	✓	✓	✓	N/A	✓	N/A
B	Reactor System			—	—	—	—	—	—
-	B1	-		—	—	—	—	—	—
-	-	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	N/A	N/A	N/A	N/A
-	B2	Reactor Cover Plug		—	—	—	—	—	—
-	-	-	Increase of dose rate in top dome (insufficient shielding from neutron, gamma-ray streaming, etc.)	N/A	N/A	N/A	N/A	N/A	✓
-	B3	Reactor Internal Structures		—	—	—	—	—	—
-	B30	General		—	—	—	—	—	—
-	-	-	Coolant mixing effect in upper plenum including thermal stratification	✓	✓	✓	N/A	N/A	N/A
-	-	-	Temperature dependence of physical properties of structural materials	✓	✓	✓	N/A	N/A	N/A
-	-	-	Natural convection	✓	✓	✓	N/A	N/A	N/A
-	-	-	Flow-induced vibration	✓	✓	✓	N/A	N/A	N/A
-	B31	Reflector (Cavity)							
-	-	-	Deformation due to thermal effect and irradiation	✓	✓	✓	N/A	N/A	N/A
-	-	-	Flow in reflector region	✓	✓	✓	N/A	N/A	N/A
-	-	-	Effect of generated heat by neutron capture and gamma rays	✓	✓	✓	N/A	N/A	N/A
-	B32	Lower Plenum		—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat capacity	✓	✓	✓	N/A	N/A	N/A
-	-	-	Mixing effect of coolant (including thermal stratification)	✓	✓	✓	N/A	N/A	N/A

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
-	-	-	Heat radiation from lower-end half-ellipse-shaped plate of reactor vessel	✓	✓	✓	N/A	N/A	N/A
-	B33	Upper Plenum		—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat capacity	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat transfer between shielding plug and sodium	✓	✓	✓	N/A	N/A	N/A
-	-	-	Coolant mixing effect at core outlet	✓	✓	✓	N/A	N/A	N/A
-	B34	Vertical Shroud		—	—	—	—	—	—
-	-	-	Radial heat transfer between inside and outside coolant through vertical shroud	✓	✓	✓	N/A	N/A	N/A
-	B36	Radial Shield		—	—	—	—	—	—
-	-	-	Flow in radial shield region	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat capacity	✓	✓	✓	N/A	N/A	N/A
-	-	-	Effect of generated heat by neutron capture and gamma rays	✓	✓	✓	N/A	N/A	N/A
-	-	-	Radial heat transfer in radial shield region	✓	✓	✓	N/A	N/A	N/A
-	B37	Reactivity Control Drive Mechanism		—	—	—	—	—	—
-	-	-	Shutdown velocity of reflector	✓	✓	✓	N/A	N/A	N/A
-	-	-	Shutdown velocity of shutdown rod	✓	✓	✓	N/A	N/A	N/A
C	Primary Heat Transport System			—	—	—	—	—	—
-	C0	General		—	—	—	—	—	—
-	-	-	Natural circulation	✓	✓	✓	N/A	N/A	N/A
-	-	-	Sodium inventory in primary boundary	✓	✓	✓	✓	N/A	N/A
-	-	-	Heat capacity of coolant	✓	✓	✓	N/A	N/A	N/A

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
-	C1	IHX		—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat transfer from primary coolant to intermediate coolant	✓	✓	✓	N/A	N/A	N/A
-	-	-	Primary flow rate	✓	✓	✓	N/A	N/A	N/A
-	-	-	Intermediate flow rate	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat capacity	✓	✓	✓	N/A	N/A	N/A
-	-	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	✓	✓	✓	N/A	N/A	N/A
-	C2	Primary EMP		—	—	—	—	—	—
-	-	-	Flow coastdown performance	✓	✓	✓	N/A	N/A	N/A
-	-	-	Pressure loss	✓	✓	✓	N/A	N/A	N/A
-	-	-	Pump head	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat capacity and joule heat at flow coastdown	✓	✓	✓	N/A	N/A	N/A
-	C4	Primary Argon Cover Gas System		—	—	—	—	—	—
-	-	-	Increase of cover gas pressure in reactor vessel	N/A	N/A	N/A	N/A	✓	N/A
-	-	-	Decrease of cover gas pressure in reactor vessel	N/A	N/A	N/A	✓	N/A	✓
D	Intermediate Heat Transport System			—	—	—	—	—	—
-	D0	General		—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	N/A	N/A	N/A
-	-	-	Natural circulation	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat removal from SG	✓	✓	✓	N/A	N/A	N/A
-	-	-	Sodium drain	N/A	N/A	✓	N/A	N/A	N/A
-	-	-	Heat transfer between upper plenum and intermediate coolant external to IHX	✓	✓	✓	N/A	N/A	N/A

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
-	D1	Intermediate EMP		—	—	—	—	—	—
-	-	-	Flow coastdown performance	✓	✓	✓	N/A	N/A	N/A
-	-	-	Pressure loss	✓	✓	✓	N/A	N/A	N/A
-	-	-	Pump head	✓	✓	✓	N/A	N/A	N/A
-	D2	Steam Generator System		—	—	—	—	—	—
-	D20	General		—	—	—	—	—	—
-	-	-	Integrity of redundant function on SG isolation and blowdown	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Functional integrity of rupture disk, steam-water reaction (SWR) release system, SWR product storage tank, and cyclone separator	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Heat capacity of structure, sodium, water, and steam	✓	✓	✓	N/A	N/A	N/A
-	-	-	Transport of sodium-water interaction products	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Sodium-water reaction	N/A	N/A	N/A	N/A	N/A	N/A
-	D21	SG Tube Leak Detection System		—	—	—	—	—	—
-	-	-	Failure of inner tube	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Failure of outer tube	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Penetration failure of double-wall tube	N/A	N/A	N/A	N/A	N/A	N/A
-	D3	Sodium Dump Tank		—	—	—	—	—	—
-	D4	Intermediate Cold Trap		—	—	—	—	—	—
-	D5	Intermediate Argon Cover Gas System		—	—	—	—	—	—
-	-	-	Functional integrity of cover gas pressure control	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Intermediate cover gas pressure and volume	N/A	N/A	N/A	N/A	N/A	N/A

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

-				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
E	Residual Heat Removal System			—	—	—	—	—	—
-	E1	IRACS (Air Cooler)		—	—	—	—	—	—
-	-	-	Pressure loss of sodium side	✓	✓	✓	N/A	N/A	N/A
-	-	-	Pressure loss of air side	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat transfer between tube and air	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat transfer between tube and sodium	✓	✓	✓	N/A	N/A	N/A
-	-	-	Inlet air temperature range	✓	✓	✓	N/A	N/A	N/A
-	-	-	Heat capacity	✓	✓	✓	N/A	N/A	N/A
-	E11	Stack		—	—	—	—	—	—
-	E2	RVACS		—	—	—	—	—	—
-	-	-	Pressure loss in airflow path	✓	✓	✓	✓	N/A	N/A
-	-	-	Heat transfer between GV wall and air	✓	✓	✓	✓	N/A	N/A
-	-	-	Heat transfer between collector wall and air	✓	✓	✓	✓	N/A	N/A
-	-	-	Heat transfer between concrete wall and air	✓	✓	✓	✓	N/A	N/A
-	-	-	Thermal radiation between RV wall and GV wall	✓	✓	✓	N/A	N/A	N/A
-	-	-	Thermal radiation between GV wall and collector wall	✓	✓	✓	✓	N/A	N/A
-	-	-	Thermal radiation between collector wall and concrete	✓	✓	✓	✓	N/A	N/A
-	-	-	Asymmetric airflow	✓	✓	✓	✓	N/A	N/A
-	-	-	Inlet air temperature range	✓	✓	✓	✓	N/A	N/A
-	E21	Stack		—	—	—	—	—	—
F	Water Steam System			—	—	—	—	—	—
-	F1	Piping		—	—	—	—	—	—
-	F2	Turbine Generator		—	—	—	—	—	—

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

-				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
G	Instrumentation and Control System			—	—	—	—	—	—
-	G1	Instrumentation and Control Equipment		—	—	—	—	—	—
-	G11	Plant Protection Sensors		—	—	—	—	—	—
-	-	-	Delay of scram signal of primary EMP voltage and current	✓	N/A	✓	N/A	N/A	N/A
-	-	-	Delay of scram signal of low power line voltage	✓	N/A	N/A	N/A	N/A	N/A
-	-	-	Delay of scram signal of neutron flux	✓	✓	✓	N/A	N/A	N/A
-	-	-	Delay of scram signal of IHX primary outlet temperature	✓	✓	✓	N/A	N/A	N/A
-	-	-	Delay of scram signal of reactor sodium level meter	N/A	N/A	N/A	✓	N/A	N/A
-	G12	Others		—	—	—	—	—	—
-	-	-	Radiation monitor	N/A	N/A	N/A	✓	✓	✓
-	-	-	Sodium leakage monitor	N/A	N/A	✓	✓	N/A	N/A
-	-	-	Cover gas pressure gauge	N/A	N/A	N/A	✓	✓	✓
-	-	-	Failed fuel detection (reactor cover gas radiation monitoring)	N/A	N/A	N/A	✓	✓	✓
-	-	-	IHX primary inlet temperature	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Delay of interlock signal of SG inlet and/or outlet temperature	✓	✓	✓	N/A	N/A	N/A
-	G2	Electric Installation		—	—	—	—	—	—
H	Reactor Auxiliary Systems			—	—	—	—	—	—
-	H1	Reactor Auxiliary Cooling System		—	—	—	—	—	—

Table A.2.2-1. Phenomena Regarding Representative Design Basis Events (cont.)

-				AOO and DBA					
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	Reactor Vessel Leakage	Local Faults (local blockage in fuel assembly)	Primary Cover Gas Boundary Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	Cladding temperature	Cladding temperature	Cladding temperature	Radiation exposure evaluation (Dose)	- Fuel temperature - Cladding temperature	Radiation exposure evaluation (Dose)
-	H2	Waste System		—	—	—	—	—	—
-	H3	Ventilation and Air Conditioning System		—	—	—	—	—	—
-	H5	Leaked Sodium Treatment System		—	—	—	—	—	—
-	-	-	Functional integrity of equipment for preventing sodium fire	N/A	N/A	N/A	✓	N/A	N/A
K	-	Containment System		—	—	—	—	—	—
-	K1	Top Dome		—	—	—	—	—	—
-	-	-	Sealing integrity	N/A	N/A	N/A	✓	N/A	✓
-	-	-	Integrity of isolation function of containment system	N/A	N/A	N/A	✓	N/A	✓
-	K2	Guard Vessel		—	—	—	—	—	—
-	-	-	Irradiation dose of guard vessel	N/A	N/A	N/A	✓	N/A	N/A
-	-	-	Sealing integrity	N/A	N/A	N/A	✓	N/A	N/A
M	-	Building		—	—	—	—	—	—
-	M1	Reactor Building		—	—	—	—	—	—
-	M2	Turbine Building		—	—	—	—	—	—
N	-	Seismic Isolator		—	—	—	—	—	—

Table A.2.2-2. Hierarchy of Regulatory Safety Requirement and Example of Criteria

Level	Source	Criteria	Directly Related to Issue	Directly Related to Phenomena	Easily Comprehended	Explicit	Measurable
1	10 CFR 1.11	Protect public health and safety	Primary Regulatory Issue				
2	10 CFR 100	Limit fission product release	✓	✓	-	-	-
3	10 CFR 50 Appendix A	Limit fuel failure and containment breach	✓	✓	-	-	-
4	SRP 6.2 Containments	Limit containment pressure, temperature, etc.	✓	✓	✓	-	-
	SRP 15.1.4 to 15.6.1 Non-LOCA	Fuel limits, energy deposition, fuel temperature, etc.	✓	✓	✓	✓	✓
	10 CFR 50.46 and SRP 15.6.5 LCOA	Peak cladding temperature, hydrogen generation, etc.	✓	✓	✓	✓	✓
5	AP600: NUREG/CR-6541, INEL-94/0061 Rev. 2	Vessel inventory	✓	✓	✓	✓	✓
	SBWR: NUREG/CR-6472. BNL-NUREG-52501	Vessel inventory	✓	✓	✓	✓	✓

A.3 LOCAL FAULTS IN A FUEL ASSEMBLY

A.3.1 Initiator and Sequence of Local Faults in a Fuel Assembly

Regarding a local fault in a fuel assembly, the following initiators are identified from past experience [6-8] (PRISM PSER).

- Mispositioning of fuel assemblies
- Enrichment error
- Failure of the fuel assembly grid spacer
- Fuel deformation (distortion, swelling, etc.)
- Local blockage in the assembly by foreign substance
- Blockage at assembly inlet by foreign substance
- Passing of gas bubble

The factors and event sequences of these local faults are described below.

(1) Mispositioning of Fuel Assemblies

Each fuel assembly is loaded once at the initial stage of operation. There are 18 assemblies in the 4S core, with the enrichment of the IC different from that of the MC and OC. Mispositioning of an assembly means loading of the assembly into the incorrect region. Each assembly is equipped with a notch for identification, and has a structure preventing the mispositioning mechanically. In the event that the mispositioning of an assembly occurred, fuel temperature would change by 20 to 30°C or less, and cladding temperature would change by about 20°C. The misposition between inner and outer fuel would cause an increase of power. The enrichment of inner/outer fuel is 17 to 19 percent. Reactor power increases by 11 percent in a mispositioned fuel assembly. The hottest cladding temperature is 612°C and inlet temperature is 355°C. Increase of cladding temperature is as follows: $(612\text{-}355)\times 0.11=28$ --> about 30°C. As a result, part of the fuel in the assembly that was loaded incorrectly might be failed by the end of core lifetime, and would be detected due to fission product release to the sodium and then to the cover gas.

(2) Enrichment Error

The 4S is equipped with fuel that has two levels of enrichment. An enrichment error is caused when fuel intended for an outer fuel assembly is placed into an inner fuel assembly or vice versa. Enrichment errors can be prevented by quality control at the fuel manufacturing plant.

Even if an enrichment error occurred with the 4S metallic fuel, which has good thermal conductivity and low power density, fuel temperature would change by about 30°C or less by mispositioning of fuel assemblies, and cladding temperature would change less than 30°C, so that fuel failure would not occur.

(3) Failure of the Fuel Assembly Grid Spacer

Failure of the fuel assembly grid spacer is caused by either a manufacturing defect, vibration, or irradiation. Grid spacer failure is prevented by quality-controlled design and manufacturing. The integrity of the grid spacer is confirmed by the inspection of the fuel outside the reactor before fuel loading and by the flow rate measurement after loading. If the grid spacer fails, the heat of fuel assembly cannot be normally removed. Although fuel temperature and cladding temperature increase as a result, the fuel does not melt because the 4S metallic fuel has high thermal conductivity and low linear power. However, the cladding might be failed at the end of life by the increase in temperature. If it is failed, the number of failed fuel pins is detected by the cover gas radiation monitor, and operation would continue until 1 percent of total fuel pins failed, which is the operation limit.

(4) Fuel Deformation (distortion, swelling, etc.)

Fuel deformation is caused by fuel swelling and creep. The amount of deformation by swelling and creep is evaluated. The grid spacer is designed to keep the fuel pins from coming in contact with each other. Moreover, the location where deformation occurs is close to the axial center of the core at which local temperature slightly rises. The possibility of fuel failure due to deformation by distortion or swelling is low because deformation does not occur at a maximum temperature point of the fuel pin. If it is failed, the number of failed fuel pins is detected by the cover gas radiation monitor, and operation would continued until 1 percent of total fuel pins failed, which is the operation limit.

(5) Local Blockage in the Assembly by Foreign Substance

This event results when local blockage in the assembly interferes with coolant flow locally. Causes of blockage can include the following:

- Loose parts such as bolts
- Incorporation of a foreign substance

In the case that a part such as a bolt becomes loose, the item is lodged in the IHX heat transfer tube, the primary flow path of the EMP, a support plate (upper support plate, middle upper support plate, middle lower support plate) of the shield region, or the core support plate. Also, an orifice hole is placed at the fuel assembly inlet and any item larger than the orifice hole cannot enter the assembly. Furthermore, quality-controlled design and manufacturing prevent the occurrence of loose parts. In addition, since an EMP is used, there is no possibility of incorporating lubricating oil as a foreign substance. The possibility of incorporating any other foreign substance is lower than that of an existing fast reactor because 4S keeps the cover gas contained without refueling. Also, inspection before transport, tests outside the reactor before fuel loading, and flow rate measurements after reactor loading confirm that the reactor internal structures and components are in normal condition. In the event that a foreign substance smaller than the orifice hole entered an assembly and was restrained by the grid, a local flow path blockage would occur in the assembly. The blockage would interfere with coolant flow and heat removal performance would decrease.

Because the 4S metallic fuel has good thermal conductivity and low power density, even if a local blockage occurred in some part of the assembly and temperature of some fuel increased and cladding temperature increased, the fuel would not melt and sodium would not boil [A-4].

(6) Blockage at Inlet by Foreign Substance

This event results when some part of the inlet of an assembly is blocked by a foreign substance. The fuel assembly inlet is equipped with 30 orifice holes, and even if four orifice holes out of 30 are blocked, the flow rate of the assembly remains at 85 percent or more of that of the rated operation. Fuel failure can be prevented if 85 percent of flow rate is assured. It is possible (but unlikely) that a sheet of metal would block many orifice holes. However, the loose parts such as sheets would be lodged in an IHX heat transfer tube, the primary flow path of the EMP, a support plate (upper support plate, middle upper support plate, middle lower support plate) of the shield region, or the core support plate. However, in the event that five or six holes were blocked, cladding might fail at the end of life.

(7) Gas Bubble Passage

This event results from gas accumulated in the primary coolant. The gas could be incorporated in the primary coolant when filling the coolant or be entrained during operation. Gas might be entrained and a gas bubble passed into the core. A flow test is conducted before and after fuel loading to discharge the residual gas that might be accumulated in the RV when filling it with sodium. Also, the velocity of sodium flow is about 10 cm/s at the free surface and therefore, there is low probability of gas entrainment at the liquid surface.

In the event that a gas bubble entered the core, negative reactivity would be inserted in the lower and upper parts of the assembly, and positive reactivity would be inserted in the middle part of the assembly. Void reactivity in the assembly is small, and the period of void passage is about several seconds. Therefore, even if a gas bubble with the same size as one assembly passed the core, reactivity would be inserted only for several seconds, and the reactivity insertion and power increase would be small. Even if the heat generated by the fuel assembly could not be removed during several seconds of gas bubble passage, the increase in cladding temperature would be small, because power density is low and the difference between fuel temperature and cladding temperature is small.

A.3.2 Phenomena Relating to the Local Faults in a Fuel Assembly

For the event of local fault in a fuel assembly, the mispositioning of a fuel assembly (item 1) is unlikely to occur, cladding temperature would be only slightly affected by enrichment error (item 2), fuel deformation (item 4), or gas bubble passage accident (item 7). Therefore, failure of the fuel assembly grid spacer (item 3), local blockage in the assembly by a foreign substance (item 5), and blockage at assembly inlet by a foreign substance (item 6) that might cause fuel failure at the end of life are included in the items to be investigated in a subsequent PIRT report. These three initiators do not have a large difference in event sequence. Since the fuel assembly is dense, the event considered most credible is local blockage in the assembly by a foreign substance, which has been examined for previous fast reactors. Table A.2.2-1 shows the phenomena relating to this event. Table A.2.2-2 shows the hierarchy of regulatory requirements and examples of criteria. For this event, the fuel might fail at end of life as a result of the blockage. Therefore, the FoMs to be applied to this event are the fuel temperature and cladding temperature in level 4, shown in Table A.2.2-2.

Among the plausible phenomena, those that are specific to this event are described here. The plausible phenomena that have been described in Section A.2.2 are not repeated.

- Pressure loss in core region

  If local blockage occurred in the assembly, the pressure loss would increase locally and the flow rate decrease locally in the assembly.

- Natural convection

  If a local blockage occurred in the assembly, a change of flow behavior and coolant temperature distribution would result, followed by changes in the behavior of the natural circulation.

- Reactivity feedback

  If a local blockage occurred in the assembly, the temperature distribution for the fuel and coolant in the assembly would change. Even if the fuel temperature partially increased, the amount of reactivity insertion would be 10 cents or less and have little effect on the fuel and cladding temperatures.

- Heat transfer between cladding and coolant

  If a local blockage occurred in the assembly, the heat transfer behavior would change at the blockage due to the change of flow behavior.

- Intra- and inter-assembly flow distribution

  If a local blockage occurred in the assembly, the flow rate distribution would change at the blockage and the intra-assembly flow rate distribution would change due to the change in pressure loss in the blocked assembly.

- Eutectic reaction between fuel and cladding

  If the cladding temperature increased by about 30°C, it is possible (but unlikely) that the cladding would fail at the end of life. In that case, statistically there is a possibility of occurrence of a fuel pin with higher temperature at inside the cladding that is around the eutectic starting temperature. The fuel might fail by cladding thinning continuously progressing at a very slow rate during operation. If the fuel composition and cladding thickness were changed by eutectic reaction phase erosion, the fuel temperature would be affected, although it is a small effect.

- Increase of cover gas pressure in reactor vessel

  This phenomenon considers an increase of cover gas pressure in the reactor vessel. Cover gas pressure changes by a very small degree according to the sodium volume expansion and contraction when coolant temperature is changed at the transient event. Furthermore, if a fuel pin failed at the end of life, FP gas would be released from the failed fuel and transfer to the cover gas. As a result, cover gas pressure would increase. However, this is not assumed to have a large effect on FoM.

This event does not result in reactor scram. Therefore, phenomena related to RPS are not applicable.

A.4 PRIMARY COVER GAS BOUNDARY FAILURE

A.4.1 Initiators and Sequence of Primary Cover Gas Boundary Failure

The primary cover gas leak accident is caused by a failure of the seal of the primary cover gas boundary or of a welded part. Primary argon gas, which is a radioactive material, and FPs that might exist in the primary argon gas, leak due to the failure of primary cover gas boundary, and transfer to the containment facility, i.e., inside the top dome. Subsequently, this radioactive material leaks from the top dome into the reactor building.

In this case, the leak of primary cover gas is detected by either a radioactive concentration high signal in the containment facility or the primary cover gas pressure low signal, and the reactor is caused to shut down manually. Decay heat is removed by the RHRS after the reactor shutdown. Also, the containment facility is isolated.

A.4.2 Phenomena Relating to Primary Cover Gas Boundary Failure

Table A.2.2-1 shows the phenomena relating to primary cover gas leak accident. In this event, radioactive material is released. Table A.2.2-2 shows hierarchy of regulatory requirements and examples of criteria. The fuel cladding and containment system boundary that contains the radioactive material is described in levels 3 and 4. The leak of primary cover gas is the event in which the boundary is damaged and radioactive materials are released. Therefore, the FoM of this event is release of the radioactive material. The phenomena specific to this event are described below. Phenomena already described in Sections A.2.2 and A.3.2 are not repeated.

- Increase of dose rate in top dome (insufficient neutron shielding, gamma ray streaming, etc.)

  This phenomenon considers the increases in the amount of radiation in the top dome caused by the deterioration of shielding performance. An increase in the amount of radiation in the top dome affects the performance of detection of the activated argon gas when it leaks.

- Decrease of cover gas pressure in reactor vessel

  The primary cover gas pressure decreases by a primary cover gas leak. It is easy to measure the pressure since the primary cover gas is sealed.

The phenomena related to shut down systems and RHRS are not applicable because they do not affect the behavior of activated primary gas release.

A.5 SUFFICIENCY OF SELECTED PHENOMENA IN REACTOR VESSEL LEAKAGE, LOCAL FAULTS OF FUEL ASSEMBLY, AND PRIMARY COVER GAS BOUNDARY FAILURE

In the events of (1) reactor vessel leakage, (2) local faults of fuel assembly, and (3) leakage of primary cover gas, the plausible phenomena additional to the phenomena identified in the existing three events, i.e., LOSP, SLIP, and FCC are as follows.

- FP release from fuel slug into gas plenum
- FP transport from fuel to sodium bond, and sodium in primary system
- FP transport from sodium in primary system to cover gas
- Increase of dose rate in top dome
- Increase of cover gas pressure in reactor vessel
- Decrease of cover gas pressure in reactor vessel
- Delay of scram signal of reactor sodium level meter
- Radiation monitor
- Failed fuel detection (reactor cover gas radiation monitoring)
- Cover gas pressure gage
- Functional integrity of equipment for preventing sodium fire
- Sealing integrity of top dome

- Integrity of isolation function of containment system
- Irradiation dose of guard vessel
- Sealing integrity of GV

The only phenomenon above that relates to the shutdown system and RHRS is "delay of scram signal of reactor sodium level meter," which deals with an RPS signal. Details of the sensors for the protection system signal are described in the 4S Safety Analysis Report (SAR), which will be submitted.

A.6 UNPROTECTED LOSS OF FLOW

A.6.1 Initiators and Sequence of Unprotected Loss of Flow (ULOF)

This event results from a failure to scram when the primary coolant flow rate decreases. Initiators include offsite power loss and primary pump fault. Event sequences in this event are as follows. Figure A.6-1 shows the power and flow rate change, Figure A.6-2 shows the assembly outlet temperature variation, Figure A.6-3 shows the reactivity change, and Figure A.6-4 shows long-term core inlet and outlet temperature variation. These analysis results were obtained using the ARGO code.

The amount of heat removal in the core decreases by the decrease of flow rate, and the temperatures of the fuel, core coolant, and structural material rise (Figures A.6-1 and A.6-2).

Negative reactivity such as Doppler reactivity, fuel density reactivity, coolant density reactivity, structural material density reactivity, axial core and reflector relative displacement reactivity, and radial core expansion reactivity is inserted negatively into core by the increase of core temperature, and reactor power decreases (Figure A.6-3).

Axial core and reflector relative displacement reactivity results from the relative position changes by thermal expansion of the core, reflector driveline, RV, reflector and core support structures, which are axially expanded by temperature increase during ULOF. Also, axial core and reflector relative displacement reactivity is inserted. If, due to thermal expansion, the reflector moves lower than initial position, the reflector effect is decreased. Thus, the reflector moves relatively downward and negative reactivity is inserted. After that, the reactor vessel is expanded axially by heat if RV temperature rises. Therefore, the core moves relatively downward and positive reactivity is inserted. The reactivity considering these effects is the axial core and reflector relative displacement reactivity. Even if the shutdown rod is expanded, there is no effect on reactivity change because the shutdown rod is on standby away from the top of the core where reactivity is not affected.

Radial core expansion reactivity is inserted when the core temperature rises. The pad located in the middle of the assembly expands due to its temperature increase, and this results in the thermal core expansion in the radial direction. Core radial expansion could be inserted by assembly bowing due to the higher temperature and irradiation; therefore, its behavior is complicated. These effects are investigated using a fuel assembly bowing code [A-5].

The coolant flow rate decreases is followed by the pump coastdown system, and transitions to the natural circulation condition (Figure A.6-1) after 60 seconds from pump trip.

Although the core temperature increases momentarily, when primary coolant flow rate reaches natural circulation conditions, the core temperature starts to decrease (Figure A.6-2).

Beyond that point, the heat of the core is stably removed under low-power conditions by natural circulation (Figure A.6-2).

At the initial stage of the event, in the case that the scram signal fails to be transmitted, the AC system dampers are not opened, so core heat is removed only by the RVACS (Figure A.6-4). If manual startup of the AC system is achieved by an operator, natural circulation heat removal by IRACS can be achieved after opening of the dampers in the air path of the AC. In case of shutdown failure, despite the transmission of the scram signal, heat is removed by the IRACS and RVACS.

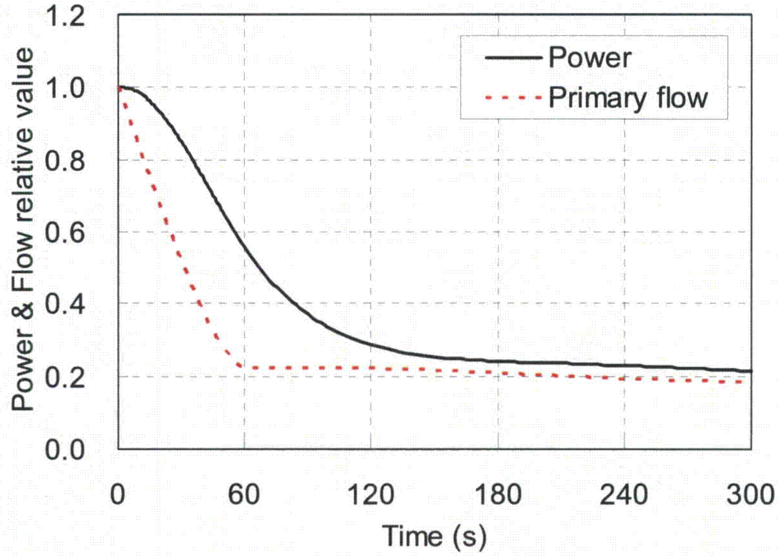

Figure A.6-1. Power and Primary Flow (ULOF)

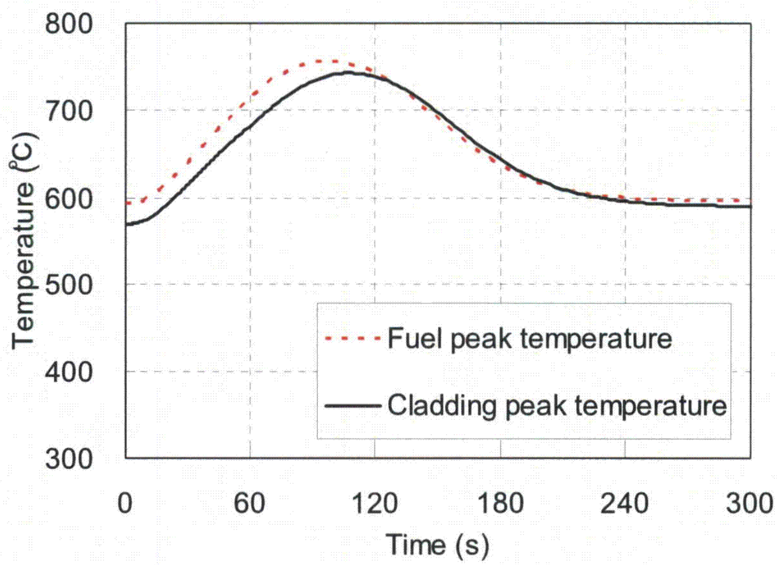

Figure A.6-2. Fuel and Cladding Peak Temperature (ULOF)

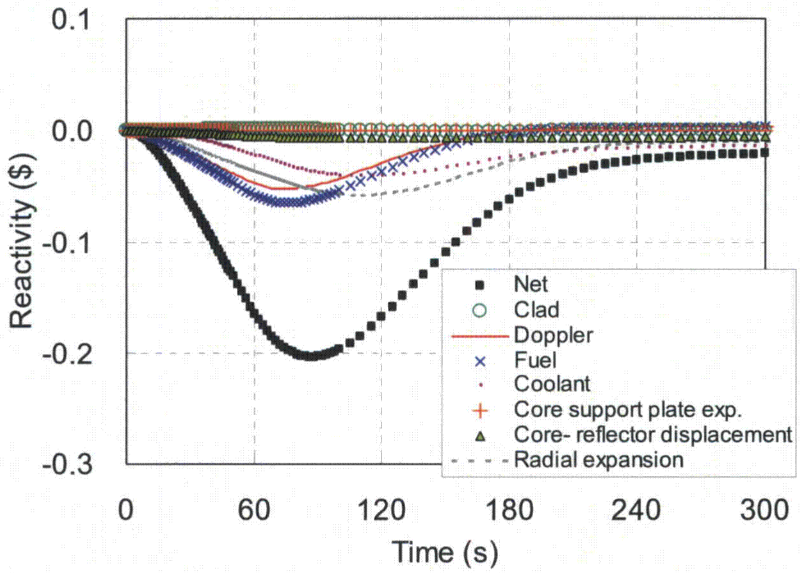

Figure A.6-3. Reactivity (ULOF)

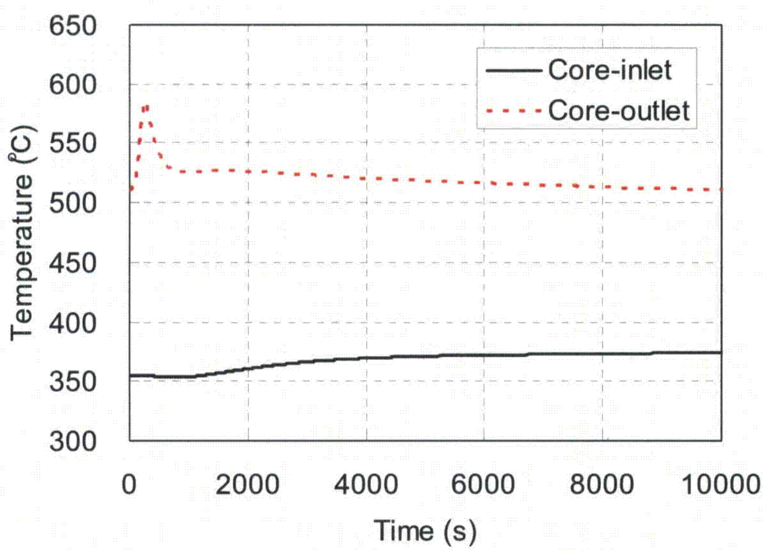

Figure A.6-4. Core Inlet and Outlet Temperature (ULOF)

A.6.2 Phenomena Relating to the Unprotected Loss of Flow

Table A.6-1 shows the phenomena relating to the ULOF event. Table A.2.2-2 shows the hierarchy of regulatory requirements and examples of criteria. For this event, the cladding might fail, and the temperature of the primary coolant boundary increase. Therefore, the FoM is applied for cladding temperature and temperature of primary coolant boundary [4-3]. The plausible phenomena specific to this event are described below. The phenomena already described in Sections A.2.2, A.3.2, and A.4.2 are not repeated.

- Pressure loss in core region

  In this event, core outlet temperature becomes higher than that of rated power. The primary coolant flow rate becomes that of natural circulation conditions based on the difference in temperature between the core inlet and outlet. Therefore, pressure loss in the natural circulation state differs from that in the event scram was successful.

- Reactivity feedback

  In this event, because the core temperature increases more than that at rated power, negative reactivity such as Doppler reactivity, fuel density reactivity, coolant density reactivity, core radial expansion reactivity, and core-reflector axial relative displacement reactivity are inserted into the core. Then, reactor power decreases due to the negative reactivity. Fuel and cladding temperatures are determined by the balance of reactor power and flow rate. Therefore, reactivity feedback affects fuel and cladding temperatures.

- Intra- and inter-assembly flow distribution, radial heat transfer between subassemblies (S/A ⇔ sodium ⇔ S/A), and heat transfer between reflector and coolant

  In this event, the core temperature increases more than that at rated power, and the primary coolant flow rate becomes that of natural circulation conditions based on the difference in temperature between the core inlet and outlet. Therefore, the temperature condition is different from that in the event scram was successful. The flow distribution of the intra- and inter-assembly at natural circulation, radial heat transfer between subassemblies, and heat transfer between reflector and coolant differ as well.

- Core power

  In this event, since reactor power changes due to the negative reactivity inserted by Doppler, fuel density, and coolant density reactivity, the factor of reactivity insertion differs from that in the event scram was successful, as does the transient behavior of reactor power.

- Heat transfer between core support plate and sodium

  In this event, the inlet temperature of the core rises after approximately 1000 seconds because it is assumed that the heat removal of the water-steam system is lost. The temperature of the core support plate also rises due to heat transfer associated with the increase of the inlet temperature of the core. If the temperature of the core support plate rises, negative reactivity caused by the core support plate expansion is inserted, because radial expansion causes the fuel assemblies to spread, thereby increasing neutron leakage and decreasing fuel density. However, this reactivity does not affect the initial peak temperature because it takes time (1000 seconds or more) until the core inlet temperature rises.

- Natural circulation

  In this event, the temperature conditions are different from rated condition because core outlet temperature increases to more than that at rated condition. Also, if the scram failure is caused by the failure of IRACS, only the RVACS is available for heat removal because the startup signal to open the AC dampers is not transmitted. Therefore, natural circulation is different from the event succeeded in scram.

- Delay of interlock signal of SG inlet and/or outlet temperature

  In this event, the intermediate pump and feedwater pumps trip by the interlock signal transmitted from the temperature instrumentation at SG inlet and outlet, even if the reactor scram system fails to transmit the signal to trip the pumps. Therefore, this phenomenan affects the FOMs.

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
A	Core/Fuel Assemblies			—	—	—	—	—	—	—
-	G20	-	Pressure loss in core region	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Pressure loss in reflector region	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Natural convection	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Reactivity feedback	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Gap conductance between fuel and cladding	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat transfer between cladding and coolant	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Intra- and inter-assembly flow distribution	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Radial heat transfer between subassemblies (S/A ⇔ sodium ⇔ S/A)	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat transfer between reflector and coolant	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity of core assemblies	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Coolant boiling	N/A	N/A	N/A				N/A
-	-	-	Core power	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat transfer between core support plate and sodium	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Decay heat	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Rate and delay of scram reactivity insertion	✓	✓	✓	✓			N/A
-	-	-	Eutectic reaction between fuel and cladding	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Temperature dependence of physical properties of materials	✓	✓	✓	✓	✓	✓	N/A
-	-	-	FP release from fuel slug into gas plenum	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	FP transport from fuel to sodium bond, and sodium in primary system	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	FP transport from sodium in primary system to cover gas	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-		Event	Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
-	-	-	Flow-induced vibration in a subassembly	✓	✓	✓	N/A	N/A	N/A	N/A
-	-	-	Inter-wrapper flow between wrapper tubes	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Maldistribution of the core flow: redistribution of the mass flow in all core subassemblies	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Radial power distribution	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Axial power distribution	✓	✓	✓	✓	✓	✓	N/A
B	Reactor System			—	—	—	—	—	—	—
-	B1	Reactor Vessel		—	—	—	—	—	—	—
-	-	-	Temperature fluctuation of reactor vessel by change of liquid level	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	B2	Reactor Cover Plug		—	—	—	—	—	—	—
-	-	-	Increase of dose rate in top dome (insufficient shielding from neutron, gamma-ray streaming, etc.)	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	B3	Reactor Internal Structures		—	—	—	—	—	—	—
-	B30	General		—	—	—	—	—	—	—
-	-	-	Coolant mixing effect in upper plenum including thermal stratification	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Temperature dependence of physical properties of structural materials	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Natural convection	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Flow-induced vibration	✓	✓	✓	N/A	N/A	N/A	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
-	B31	Reflector (Cavity)		—	—	—	—	—	—	—
-	-	-	Deformation due to thermal effect and irradiation	✓	✓	✓	N/A	✓	✓	N/A
-	-	-	Flow in reflector region	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Effect of generated heat by neutron capture and gamma rays	✓	✓	✓	✓	✓	✓	N/A
-	B32	Lower Plenum		—	—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Mixing effect of coolant including thermal stratification	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat radiation from lower-end half-ellipse-shaped plate of reactor vessel	✓	✓	✓	✓	✓	✓	N/A
-	B33	Upper Plenum		—	—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat transfer between shielding plug and sodium	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Radial coolant temperature distribution at core outlet	✓	✓	✓	✓	✓	✓	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
-	B34	Vertical Shroud		—	—	—	—	—	—	—
-	-	-	Radial heat transfer between inside and outside coolant through vertical shroud	✓	✓	✓	✓	✓	✓	N/A
-	B36	Radial Shield		—	—	—	—	—	—	—
-	-	-	Local flow behavior in radial shield region	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Effect of generated heat by neutron capture and gamma rays	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Radial heat transfer in radial shield region	✓	✓	✓	✓	✓	✓	N/A
-	B37	Reactivity Control Drive Mechanism		—	—	—	—	—	—	—
-	-	-	Shutdown velocity of reflector	✓	✓	✓	N/A	N/A	N/A	N/A
-	-	-	Shutdown velocity of shutdown rod	✓	✓	✓	N/A	N/A	N/A	N/A
C	Primary Heat Transport System			—	—	—	—	—	—	—
-	C0	General		—	—	—	—	—	—	—
-	-	-	Natural circulation	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Sodium inventory in primary boundary	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity of coolant	✓	✓	✓	✓	✓	✓	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed Pressure in Intermediate System (IHX Boundary)
-	C1	IHX		—	—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat transfer from primary coolant to intermediate coolant	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Primary flow rate	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Intermediate flow rate	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Spatial distribution effect of intermediate flow path in IHX annulus shape	✓	✓	✓	✓	✓	✓	N/A
-	C2	Primary EMP		—	—	—	—	—	—	—
-	-	-	Flow coastdown performance	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Pressure loss	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Pump head	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity and joule heat at flow coastdown	✓	✓	✓	✓	✓	✓	N/A
-	C4	Primary Argon Cover Gas System		—	—	—	—	—	—	—
-	-	-	Increase of cover gas pressure in reactor vessel	N/A	N/A	N/A	N/A	✓	✓	N/A
-	-	-	Decrease of cover gas pressure in reactor vessel	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
D	Intermediate Heat Transport System			—	—	—	—	—	—	—
-	D0	General		—	—	—	—	—	—	—
-	-	-	Pressure loss	✓	✓	✓	✓	✓	✓	✓
-	-	-	Natural circulation	✓	✓	✓	✓	✓	✓	✓
-	-	-	Heat removal from SG	✓	✓	✓	✓	✓	✓	✓
-	-	-	Sodium drain	N/A	N/A	✓	N/A	N/A	N/A	✓
-	-	-	Heat transfer between upper plenum and intermediate coolant external to IHX	✓	✓	✓	✓	✓	✓	N/A
-	D1	Intermediate EMP		—	—	—	—	—	—	—
-	-	-	Flow coastdown performance	✓	✓	✓	✓	✓	✓	✓
-	-	-	Pressure loss	✓	✓	✓	✓	✓	✓	✓
-	-	-	Pump head	✓	✓	✓	✓	✓	✓	✓
-	D2	Steam Generator System		—	—	—	—	—	—	—
-	D2 0	General		—	—	—	—	—	—	—
-	-	-	Integrity of redundant function on SG isolation and blowdown	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	-	-	Functional integrity of rupture disk, SWR release system, SWR product storage tank, and cyclone separator	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	-	-	Heat capacity	✓	✓	✓	✓	✓	✓	✓

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
-	-	-	Transport of sodium-water interaction products	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	-	-	Sodium-water reaction	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	D21	SG Tube Leak Detection System		—	—	—	—	—	—	—
-	-	-	Failure of inner tube	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	-	-	Failure of outer tube	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	-	-	Penetration failure of double-wall tube	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	D3	Sodium Dump Tank		—	—	—	—	—	—	—
-	D4	Intermediate Cold Trap		—	—	—	—	—	—	—
-	D5	Intermediate Argon Cover Gas System		—	—	—	—	—	—	—
-	-	-	Functional integrity of cover gas pressure control	N/A	N/A	N/A	N/A	N/A	N/A	✓
-	-	-	Intermediate cover gas pressure and volume	N/A	N/A	N/A	N/A	N/A	N/A	✓
E	Residual Heat Removal System			—	—	—	—	—	—	—
-	E1	IRACS (Air Cooler)		—	—	—	—	—	—	—
-	-	-	Pressure loss of sodium side	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Pressure loss of air side	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat transfer between tube and air	✓	✓	✓	✓	✓	✓	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
-	-	-	Heat transfer between tube and sodium	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Inlet air temperature range	✓	✓	✓	✓	✓	✓	N/A
-	-	-	Heat capacity	✓	✓	✓	✓	✓	✓	N/A
-	E11	Stack		—	—	—	—	—	—	—
-	E2	RVACS		—	—	—	—	—	—	—
-	-	-	Pressure loss in airflow path	✓	✓	✓	✓	✓	✓	✓
-	-	-	Heat transfer between GV wall and air	✓	✓	✓	✓	✓	✓	✓
-	-	-	Heat transfer between collector wall and air	✓	✓	✓	✓	✓	✓	✓
-	-	-	Heat transfer between concrete wall and air	✓	✓	✓	✓	✓	✓	✓
-	-	-	Thermal radiation between RV wall and GV wall	✓	✓	✓	✓	✓	✓	✓
-	-	-	Thermal radiation between GV wall and collector wall	✓	✓	✓	✓	✓	✓	✓
-	-	-	Thermal radiation between collector wall and concrete	✓	✓	✓	✓	✓	✓	✓
-	-	-	Asymmetric airflow	✓	✓	✓	✓	✓	✓	✓
-	-	-	Inlet air temperature range	✓	✓	✓	✓	✓	✓	✓
-	E21	Stack		—	—	—	—	—	—	—
F	Water-Steam System			—	—	—	—	—	—	—
	F1	Piping		—	—	—	—	—	—	—
	F2	Turbine Generator		—	—	—	—	—	—	—

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and cladding temperature -Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
G	Instrumentation and Control System			—	—	—	—	—	—	—
-	G1	Instrumentation and Control Equipment		—	—	—	—	—	—	—
-	G11	Plant Protection Sensors		—	—	—	—	—	—	—
-	-	-	Delay of scram signal of primary EMP voltage and current	✓	N/A	✓	N/A	N/A	N/A	N/A
-	-	-	Delay of scram signal of low power line voltage	✓	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Delay of scram signal of neutron flux	✓	✓	✓	N/A	N/A	N/A	N/A
-	-	-	Delay of scram signal of IHX primary outlet temperature	✓	✓	✓	N/A	N/A	N/A	N/A
-	-	-	Delay of scram signal of reactor sodium level meter	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	G13	Others		—	—	—	—	—	—	—
-	-	-	Radiation monitor	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Sodium leakage monitor	N/A	N/A	✓	N/A	N/A	N/A	N/A
-	-	-	Cover gas pressure gauge	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and Cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
-	-	-	Failed fuel detection (reactor cover gas radiation monitoring)	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	IHX primary inlet temperature	N/A	N/A	N/A	N/A	✓	✓	N/A
-	-	-	Delay of interlock signal of SG inlet and/or outlet temperature	✓	✓	✓	✓	✓	✓	✓
-	G2	Electric Installation		—	—	—	—	—	—	—
H	Reactor Auxiliary Systems			—	—	—	—	—	—	—
-	H1	Reactor Auxiliary Cooling System		—	—	—	—	—	—	—
-	H2	Waste System		—	—	—	—	—	—	—
-	H3	Ventilation and Air Conditioning System		—	—	—	—	—	—	—
-	H5	Leaked Sodium Treatment System		—	—	—	—	—	—	—
-	-	-	Functional integrity of equipment for preventing sodium fire	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table A.6-1. Phenomena Regarding Representative Beyond-Design-Basis Accidents (cont.)

-				AOO and DBA			ATWS			BDBA
-	-	Event		Loss of Offsite Power	Cavity Can Failure	Sodium Fire due to Leak from Intermediate Piping	ULOF	UTOP	ULOHS	Sodium-Water Reaction due to SG Tube Failure
-	-	Equipment	Phenomenon Indicator (Subsystem)	- Cladding temperature	- Cladding temperature	- Cladding temperature	- Cladding temperature - Primary coolant boundary temperature	- Fuel and Cladding temperature - Primary coolant boundary temperature	- Cladding temperature - Primary coolant boundary temperature	- Developed pressure in intermediate system (IHX boundary)
K	Containment System			—	—	—	—	—	—	—
-	K1	Top Dome		—	—	—	—	—	—	—
-	-	-	Sealing integrity	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Integrity of isolation function of containment system	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	K2	Guard Vessel		—	—	—	—	—	—	—
-	-	-	Irradiation dose of guard vessel	N/A	N/A	N/A	N/A	N/A	N/A	N/A
-	-	-	Sealing integrity	N/A	N/A	N/A	N/A	N/A	N/A	N/A
M	Building			—	—	—	—	—	—	—
-	M1	Reactor Building		—	—	—	—	—	—	—
-	M2	Turbine Building		—	—	—	—	—	—	—
N	Seismic Isolator			—	—	—	—	—	—	—

A.7 UNPROTECTED TRANSIENT OF OVERPOWER

A.7.1 Initiators and Sequence of Unprotected Transient of Overpower

This event results from a failure to scram when reactor power increases after some positive reactivity insertion. Initiators include an uncontrolled withdrawal of the reflector and injection of cold sodium into the core. A scram signal is transmitted by the power-range neutron monitor or wide-range neutron monitor. In addition, the 4S is designed to limit the amount of positive reactivity that can be incorrectly inserted and only an ultra-slow-speed drive mechanism is running for burnup compensation during rated operation [1-3] [3-1]. Therefore, even if the velocity rises to the upper limit of the drive mechanism when the reflector drive mechanism fails, the insertion velocity is slow and operators have sufficient time of several days to recover the drive mechanism or controller. At startup, the drive mechanism gross reactivity insertion is limited by mechanical stops. Transients of reactivity insertion event by uncontrolled motion of segments of reflector using the drive mechanism for startup are investigated at full-power operation as a UTOP event. Figure A.7-1 shows the power and flow rate change, Figure A.7-2 shows assembly outlet temperature, Figure A.7-3 shows the reactivity change, and Figure A.7-4 shows the long-term core inlet and outlet temperature variation.

Core power rises due to the positive reactivity insertion and the temperature of the fuel, core coolant, and structural material rise (see Figures A.7-1 and A.7-2).

Negative reactivity is inserted by temperature increases of the fuel, core coolant, and structural materials. Therefore, even though positive reactivity is inserted by uncontrolled reflector, reactor power stays at the same level. As a result, system temperatures rise, and they are stabilized in the high-temperature condition (see Figures A.7-2, A.7-3, and A.7-4).

After this, the operator causes the reactor to shut down.

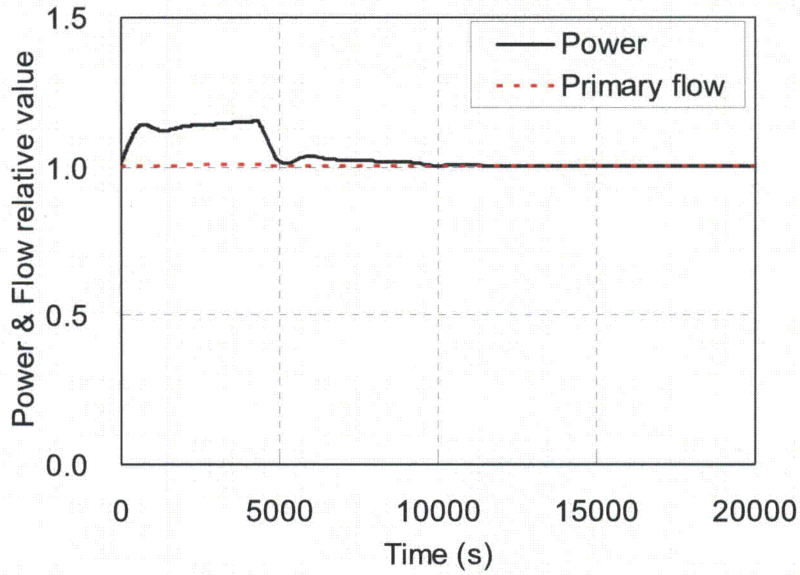

Figure A.7-1. Power and Primary Flow (UTOP)

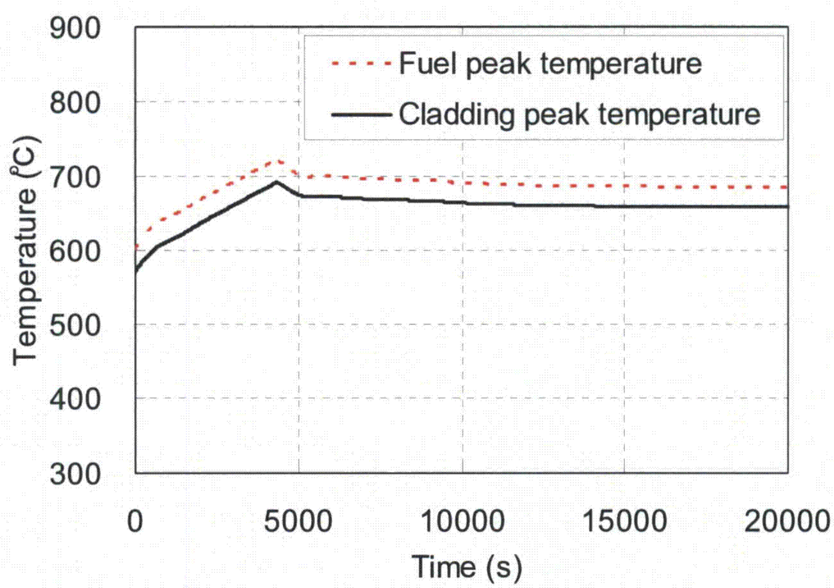

Figure A.7-2. Fuel and Cladding Peak Temperature (UTOP)

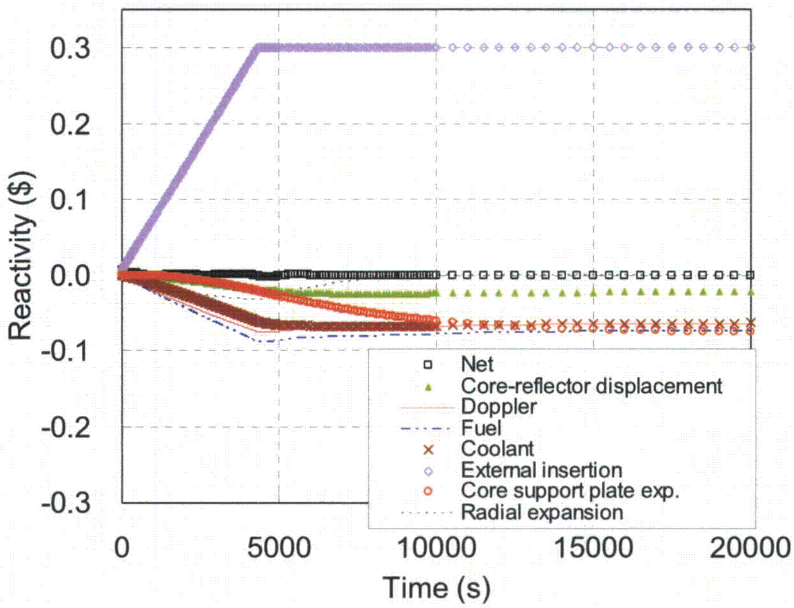

Figure A.7-3. Reactivity (UTOP)

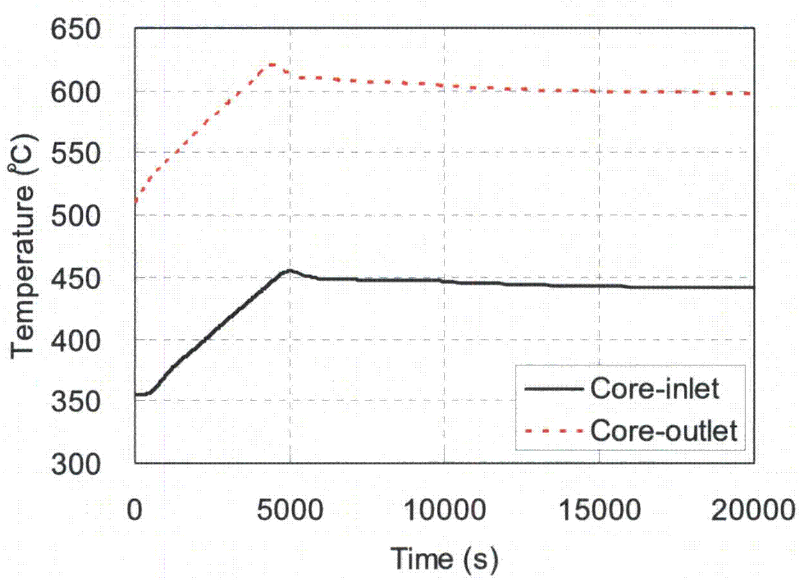

Figure A.7-4. Core Inlet and Outlet Temperature (UTOP)

A.7.2 Phenomena Relating to the Unprotected Transient of Overpower

Table A.6-1 shows phenomena relating to this event. Table A.2.2-2 shows the hierarchy of regulatory requirements and examples of criteria. For this event, the fuel and cladding might fail, and the temperature of primary coolant boundary increases. Therefore, the FoM is applied to the fuel temperature, cladding temperature, and temperature of primary coolant boundary. The phenomena that are specific to this event are described below. Phenomena associated with the ULOF event are not repeated.

- Reactivity feedback

  In this event, core negative reactivity such as Doppler reactivity and fuel density reactivity is inserted because core temperature increases more than the temperature at rated conditions. The increase in reactor power is reduced by the negative reactivity.

- Heat transfer between reflector and coolant

  In this event, the core inlet and outlet temperatures rise higher than those at rated conditions. The primary coolant flow rate is almost the same as at rated conditions. Since fuel and coolant temperatures at the core inlet rise, the behavior of radial heat transfer of the assemblies differs from that at rated conditions.

- Core power

  In this event, reactor power remains constant after negative reactivity such as Doppler reactivity and fuel density reactivity is inserted, and the assembly temperatures differ from those at rated conditions.

- Heat transfer between core support plate and sodium

  In this event, temperature rises slowly. The inlet temperature of the core rises after about 350 seconds (Figure A.6-4). If the inlet temperature of the core rises, the temperature of the core support plate also rises by heat transfer. If the temperature of the core support plate rises, negative reactivity due to core support plate expansion is inserted. Radial expansion of the core support plate causes the fuel assemblies to spread, thereby increasing neutron leakage and decreasing fuel density.

- Eutectic reaction between fuel and cladding

  In this event, temperature rises slowly. When cladding temperature reaches the eutectic starting temperature, cladding thinning is initiated at a slow rate ($< 3\times10^{-2}$ μm/s at 650-700°C) [1-2] [8-2].

- Deformation due to thermal effect and irradiation

  In this event, core inlet temperature rises, and the temperature of the reflector cavity rises higher than that at rated conditions.

- Sodium inventory in primary boundary

  In this event, sodium level rises because of thermal expansion of the sodium as the system temperature rises.

- Increase of cover gas pressure in reactor vessel

  In this event, the contained primary cover gas pressure rises because system temperature rises.

- Heat removal to SG

  In this event, because of the system temperature increase, reactor power is stabilized in the condition higher than the rated condition, depending on the increase in the amount of heat removal.

- Thermal radiation between the RV wall and GV wall

  In this event, the heat transfer coefficient of the radiation between the RV and GV becomes higher because RV temperature becomes higher.

- IHX primary inlet temperature and delay of interlock signal of SG inlet and/or outlet temperature instrumentation

  In this event, the primary pumps, intermediate pump, and feedwater pumps trip by the interlock signal transmitted from the temperature instrumentation at IHX and SG, even if the reactor scram system fails to transmit the signal to trip the pumps. After the pump trip, this event will follow the event sequence of the ULOF event. Therefore, these phenomena have strong influence on the event sequence.

A.8 UNPROTECTED LOSS OF HEAT SINK

A.8.1 Initiators and Sequence of Unprotected Loss of Heat Sink

This event results from a scram failure after the heat removal function of the water-steam system is lost. Initiators are the heat removal function loss by malfunction of the water-steam system. Transients of this event are shown in Figures A.8-1 to A.8-4. Figure A.8-1 shows the power and flow rate transients, Figure A.8-2 shows the assembly outlet temperature, Figure A.8-3 shows the reactivity transients, and Figure A.8-4 shows the long-term variation of core inlet and outlet temperatures.

The intermediate SG outlet temperature rises due to loss of heat removal by the water-steam system, and the inlet temperature of the core rises (Figures A.8-1 and A.8-4).

The temperatures of the fuel, core coolant, and structural material rise (Figure A.8-2) when the inlet temperature of the core rises.

Negative reactivity is inserted (Figure A.8-3) by the temperature increase of each core region. Negative reactivity is inserted from Doppler reactivity, fuel density reactivity, structural material density reactivity, core radial expansion reactivity, and support plate expansion reactivity. In this event, since heat removal is lost, the coolant temperature of the intermediate system and the primary outlet temperature of the IHX rises at first. When negative reactivity is inserted, reactor power decreases (Figures A.8-1 and A.8-4).

Reactor power decreases to the level of decay power, and system temperature will be stabilized in the state near the average temperature at rated conditions.

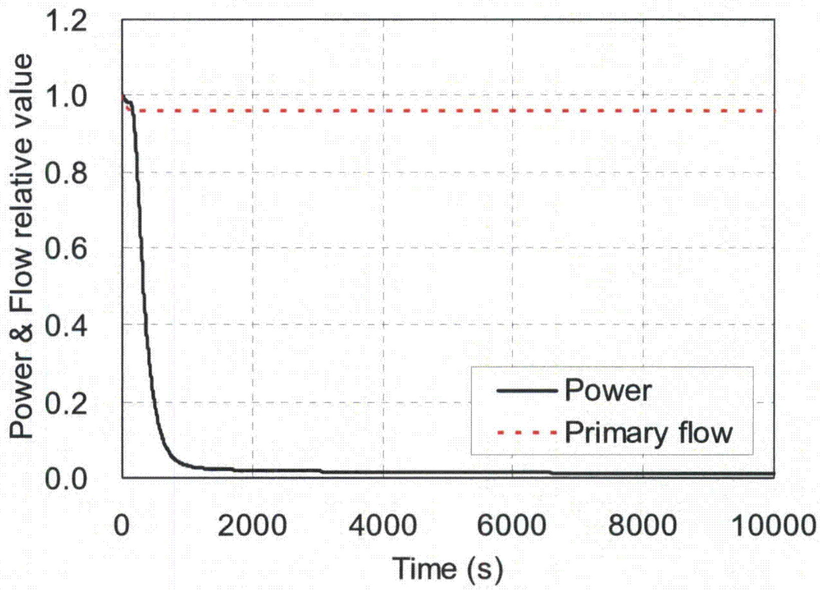

Figure A.8-1. Power and Primary Flow (ULOHS)

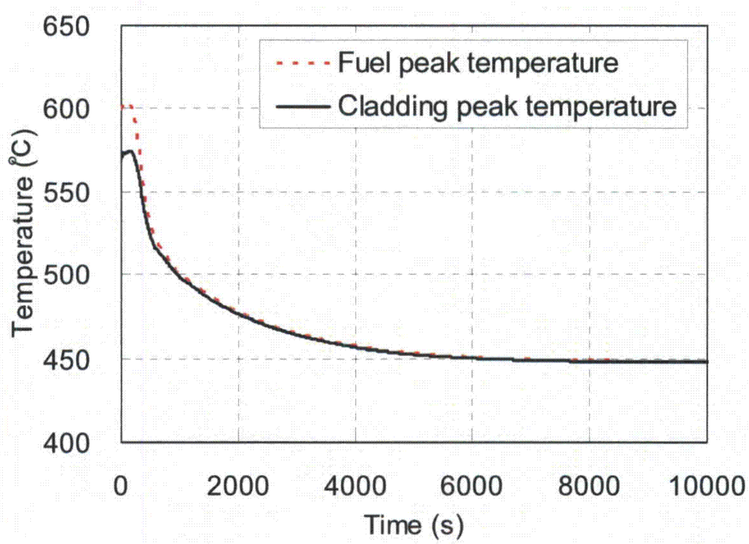

Figure A.8-2. Fuel and Cladding Peak Temperatures (ULOHS)

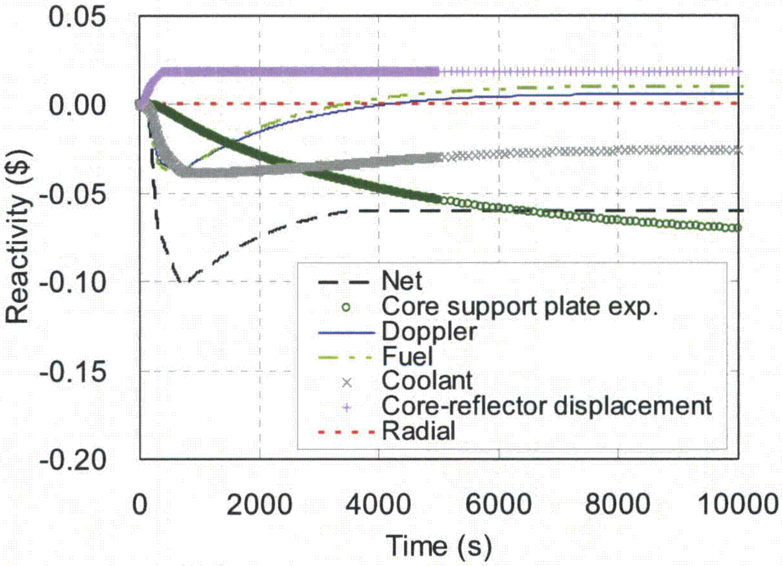

Figure A.8-3. Reactivity (ULOHS)

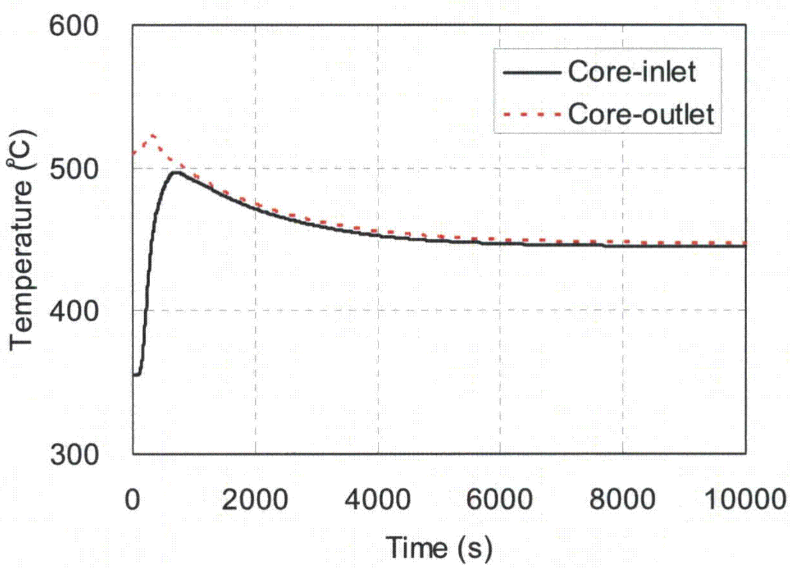

Figure A.8-4. Core Inlet and Outlet Temperatures (ULOHS)

A.8.2 Phenomena Relating to the ULOHS

Table A.6-1 shows phenomena relating to the ULOHS event. Table A.2.2-2 shows hierarchy of regulatory requirements and examples of criteria. For this event, the cladding temperature increases, then the temperature of the primary coolant boundary increases. Therefore, the FoM is applied to the cladding temperature and temperature of the primary coolant boundary. The cladding temperature is selected because this event affects the integrity of the cladding. The phenomena that are specific to this event are described below. The phenomena that are described in the ULOF and UTOP events are not repeated.

- Reactivity feedback

  In this event, the inlet temperature of the core initially rises. And the cladding and fuel temperature rise due to the inlet temperature increase. Then, negative reactivity caused by Doppler reactivity, fuel density reactivity, and core support plate expansion reactivity is inserted into the core. As a result, reactor power decreases.

  The main reactivity feedbacks that have a large impact on the FoM are as follows.

  - Doppler reactivity
  - Fuel temperature reactivity
  - Structure temperature reactivity
  - Core support plate structure
  - Core radial expansion
  - Core – reflector relative displacement

- Heat transfer between reflector and coolant

  In this event, the coolant flow rate is almost the same as the rated flow rate. Concurrently, reactor power decreases to less than rated power due to negative reactivity. Therefore, the reactor enters a condition of rated flow rate and high core inlet temperature with low power and the heat transfer conditions of the reflector and coolant differ from those of other events.

- Core power

  In this event, reactor power decreases by negative reactivity such as support plate expansion reactivity, Doppler reactivity, and fuel density reactivity. In the event of loss of the heat removal function, the amount of each reactivity insertion differs because coolant temperature at the core inlet rises at first. Therefore, the reactivity contributions differ from those of the ULOF/UTOP events.

- Heat transfer between core support plate and sodium

  In this event, after the loss heat removal by the water/steam system, if the core inlet temperature rises, the temperature of the core support plate rises by heat transfer. If the temperature of the core support plate rises, negative reactivity is inserted due to core support plate expansion, because radial expansion causes the fuel assemblies to spread, thereby increasing neutron leakage and decreasing fuel density.

- Deformation due to thermal effect and irradiation

  In this event, the core inlet temperature rises after the loss of heat removal. Therefore, the reactor enters a condition of rated flow rate and high temperature of the core inlet with low power. Therefore, the temperature condition of the reflector cavity differs from that in the event scram was successful. The high temperature of the core inlet causes the deformation.

- Sodium inventory in primary boundary

  In this event, sodium level rises because of thermal expansion as the temperature rises.

- Increase of cover gas pressure in reactor vessel

  In this event, the sodium volume expands because the temperature increases, and the contained primary cover gas pressure rises.

- Thermal radiation between the RV wall and GV wall

  In this event, since the RV temperature becomes high, the heat transfer rate by radiation also becomes high.

- Delay of interlock signal of SG inlet and outlet temperature

  In this event, the intermediate pump trips by the interlock signal transmitted from the temperature instrumentation at SG inlet and outlet, even if the reactor scram system fails to transmit the signal to trip the pumps. After that, the primary pumps trip by interlock signal. As a result, this event, this event will follow the event sequence of the ULOF event. Therefore, this phenomenan has a strong influence on the event sequence.

A.9 SUFFICIENCY OF SELECTED PHENOMENA IN ATWS

The specific plausible phenomena related to the three events for which scram fails are as follows.

- IHX primary inlet temperature and delay of interlock signal of SG inlet and outlet temperature

The following reactivity feedbacks have a large effect on FoM because the reactor power change in the scram failure event is determined by reactivity.

- Doppler reactivity

- Fuel temperature reactivity
- Coolant temperature reactivity
- Structure temperature reactivity
- Core support plate structure
- Core radial expansion
- Axial core – reflector relative displacement

A.10 SODIUM-WATER REACTION CAUSED BY SG TUBE FAILURE

A.10.1 Initiators and Sequence of Sodium-Water Reaction Caused by SG Tube Failure

The occurrence frequency of a sodium-water reaction is reduced by the application of a double-wall heat transfer tube for the SG [4-3]. This subsection assumes that the inside of the heat transfer tube fails at the initial stage, and then the outside of that tube fails at almost the same position as inside.

If the inside of the heat transfer tube fails, water and/or steam enter the space between inner and outer heat transfer tube. Inner tube failure is revealed by moisture detection.

If both the inner and outer tubes fail and operation continues, water and steam transfer into the intermediate sodium and a sodium-water reaction occurs.

The intermediate pump trips by the interlock signal of the intermediate pressure gauge placed in the intermediate system cover gas region because intermediate system pressure increases if a sodium-water reaction occurs, and finally the reactor shuts down.

A.10.2 Phenomena Relating to the Sodium-Water Reaction Caused by SG Tube Failure

Table A.6-1 shows the phenomena relating to this event.

Intermediate system pressure (spike pressure at the initial stage of the sodium-water reaction and quasi-steady pressure during which pressure continues to increase due to the reaction) is selected as the FoM. Plausible phenomena specific to this event are described below.

- Sodium drain

  In this event, when the single heat transfer tube failure is detected, the operator causes the reactor to shut down, and the water-steam system is blown down. The intermediate system coolant is drained after residual heat is removed.

- Integrity of redundant function on SG isolation and blowdown

  In this event, when the single heat transfer tube failure is detected, the operator causes the reactor to shut down, and the water-steam system is blown down.

- Functional integrity of rupture disk, steam-water reaction release system, steam-water reaction product storage tank, and cyclone separator

  This phenomenon relates to the function of the equipment used to decrease intermediate system pressure by destroying the rupture disk in case of a shutdown failure when the intermediate system pressure rises as a result of a sodium-water reaction.

- Transport of sodium-water reaction products

  This phenomenon relates to transport of sodium-water interaction products. Sodium-water reaction products are discharged from steam generator to storage tank through the discharge duct after sodium-water interaction.

- Sodium-water reaction

  This phenomenon relates to influence of the sodium-water reaction. When sodium and water react due to failure of double-wall tubes, temperature and pressure around the leak position increases. The intermediate system pressure increase is detected, intermediate and feedwater pumps are tripped, and the water-steam system is blown down. As a result, the sodium-water reaction is terminated.

- Failure of inner tube: SG tube leak detection system

  This phenomenon relates to the failure detection of the inner tube. When the failure of the inner heat transfer tube is detected, the operator causes the reactor to shut down and the water-steam system is blown down.

- Failure of outer tube: SG tube leak detection system

  This phenomenon relates to failure detection of the outer tube. When failure of the outer heat transfer tube is detected, the operator causes the reactor to shut down and the water-steam system is blown down.

- Penetration failure of double-wall tube: SG tube leak detection system

  This phenomenon relates to double-wall tube failure. When the double-wall tube failure is detected, the operator causes the reactor to shut down and the water-steam system is blown down.

- Functional integrity of cover gas pressure control

  This phenomenon relates to the integrity of the cover gas control system in the intermediate system. In this event, if both sides of the double-wall tube fail, intermediate system pressure rises by the sodium-water reaction. Therefore, the integrity of cover gas control in the intermediate system affects detection of the sodium-water reaction by the pressure increase.

- Intermediate argon cover gas pressure and volume

  This phenomenon relates to cover gas pressure and volume in the intermediate system. In this event, the initial spike in pressure is affected by pressure at the start of the sodium-water reaction. Quasi-steady pressure is affected by both cover gas volume and initial pressure because the cover gas space is compressed by the pressure increase.

Identified phenomena in the sodium-water reaction accident caused by failure of the SG heat transfer tube are as follows.

- Integrity of redundant function on SG isolation and blowdown
- Functional integrity of rupture disk, sodium-water reaction release system, sodium-water reaction product storage tank, and cyclone separator
- Failure of inner tube: SG tube leak detection system
- Failure of outer tube: SG tube leak detection system
- Penetration failure of double-wall tube: SG tube leak detection system
- Functional integrity of cover gas pressure control
- Intermediate cover gas pressure and volume

A.11 CONCLUSIONS

The purpose of this Appendix is to investigate whether the selected phenomena for three events (LOSP, SLIP, and FCC) are sufficient as phenomena to show the validity of the RPS. The plausible phenomena were investigated for the AOO, DBA, ATWS, and sodium-water reaction events. Plausible phenomena that are selected in this Appendix but are not selected for the three events evaluated in the 4S PIRT report are listed below. Reactivity feedback is selected for the three events such as LOSP. Importance levels differ in the ATWS events from the three events. However, the phenomenon is not listed again here because it was already described in Chapter 7 as a plausible phenomenon. The following are the phenomena not relating directly to the shutdown system and RHRS that were initially defined as objects of this 4S PIRT.

- Core/fuel assemblies
  - FP release from fuel slug into gas plenum
  - FP transport from fuel to sodium bond, and sodium in primary system
  - FP transport from sodium in primary system to cover gas

- Primary argon cover gas system
  - Decrease of cover gas pressure in reactor vessel
  - Increase of cover gas pressure in reactor vessel
- Plant protection sensors
  - Delay of scram signal of reactor sodium level meter
- Others
  - Radiation monitor
  - Failed fuel detection (reactor cover gas radiation monitoring)
  - IHX primary inlet temperature
  - Delay of interlock signal of SG inlet and/or outlet temperature instrumentation
  - Cover gas pressure gauge
- Leaked sodium treatment system
  - Functional integrity of equipment for preventing sodium fire
- Top dome
  - Sealing integrity, overall leakage testability
  - Integrity of isolation function of containment system
- Guard vessel
  - Irradiation dose of guard vessel
  - Sealing integrity
- Steam generator system
  - Integrity of redundant function on SG isolation and blowdown
  - Functional integrity of rupture disk, sodium-water reaction release system, sodium-water reaction product storage tank, and cyclone separator
- SG tube leak detection system
  - Failure of inner tube: SG tube leak detection system
  - Failure of outer tube: SG tube leak detection system
  - Penetration failure of double-wall tube: SG tube leak detection system

- Intermediate argon cover gas system
  - Functional integrity of cover gas pressure control
  - Intermediate cover gas pressure and volume

Among the preceding phenomena, the factor that relates to the shutdown system and RHRS phenomena is the RPS signal, which is the delay of the scram signal of the reactor sodium level meter. Details of the sensors for each protection system signal are provided in the 4S Safety Analysis [4-1]. For these events discussed in Appendix A, additional PIRT reports are being developed.

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APPENDIX B
BRIEF EXPLANATION OF ARGO CODE AND SOME RESULTS OF CODE VERIFICATION AND VALIDATION

The ARGO code is used in the sensitivity analysis performed in the PIRT. This Appendix provides a brief explanation of the ARGO code and discusses the present status of code verification and validation.

The ARGO code has been used for design for several FRs, including the 4S. The paper [B-1] is an example of the validation for ARGO that has been performed so far. This compares the ARGO results with the results using a CFD code regarding natural circulation behavior in the 4S reactor vessel, and shows that the ARGO results are in good agreement with the CFD code results.

In the reference, the list of ARGO study papers is shown [B-2] [B-3] [B-4] [B-5] [B-6] [B-7].

The main analysis functions that the ARGO code includes are as follows

- Plant thermal dynamics
- Variation of reactor power and decay heat
- Variation of the reactor fuel pin temperature
- The calculation formula for cumulative damage fraction (CDF) of the cladding
- Variation of the primary coolant boundary temperature
- Thermal-hydraulic characteristics of the intermediate heat exchanger and the steam generator
- Hydraulic characteristics of the circulation pump
- Heat removal by residual heat removal systems (e.g., IRACS, RVACS)
- Actuation of the safety and interlock systems

The ARGO code is currently undergoing validation [B-1] and verification (V&V) following RG 1.203 [B-8] and NUREG-1737 [B-9]. The document summarizing the results of the validity of the analysis results for the ARGO code will be submitted as part of the Design Approval application.

B.1 EXPLANATION OF PLANT DYNAMICS ANALYSIS CODE (ARGO)

The objective of the analysis using the ARGO code is to evaluate the integrity of fuel pins and the temperature of the primary coolant boundary during transients, accidents, and ATWS. The main transients and accidents to be considered are an increase or decrease in heat removal by the balance of plant (BOP), decrease or increase of flow by the primary- or intermediate-loop coolant, an anticipated reactivity insertion at full-power operation or startup, and loss of offsite power (LOSP). Regarding these events, to calculate fuel cladding temperature and primary coolant boundary temperature with sufficient precision, the plant transient states themselves must be modeled with similar precision.

The ARGO code models the movement of a one-dimensional incompressible fluid, and expresses the fluid flow according to the one-dimensional flow network model, including the intermediate heat exchanger, steam generator, and core. The ARGO code analyzes the transient starting from an initial steady thermal-hydraulic balance.

The core is modeled with multiple channels, with each channel representing one pin enclosed in the associated wrapper tube. The one-dimensional flow in the channel absorbs heat from the fuel pins and flows out from the channel; the flows of each channel join at the channel exit. The fuel, cladding, and coolant in the core are divided in the axial direction. The fuel pellet or slug is further divided in the radial direction into cells. The main flow network model is shown in Figure B-1.

The ARGO code describes the fuel pin under a specific thermal condition, such as the hottest pin in the core, and evaluates the integrity of fuel and cladding. The temperature distribution of the specific fuel pin is evaluated by dividing the pin in detail. The initial stress of the cladding is assumed in the detailed pin analysis, such as the hottest pin analysis, and the CDF of the cladding is evaluated by stress-rapture correlation, i.e., the Larson-Miller parameter (LMP). Because there is a eutectic formation due to interaction between the metal fuel and cladding, an inside temperature of the cladding is calculated semi-statically; the amount of the thinning is calculated according to the inside temperature of the cladding, and is reflected in a transitional increase in stress.

Nuclear dynamic characteristics (kinetics) are calculated as the reactor power transient with a one-point approximation model (point kinetics) and six group-delayed neutron precursors. The reactivity contributions include the insertion reactivity, scram reactivity, Doppler feedback, and reactivity feedback by the temperature change of the fuel, cladding, and wrapper tube. Reactivity contributions can, furthermore, consider the reactivity change by thermal expansion of reactivity caused by the relative displacement of the core and the reflector drive extension rod, etc., or the reactivity change by radial direction expansion of the reactor core if necessary.

The flow network models the equation of fluid motion in the one dimension by balancing the pump head, natural convection head, and pressure loss. The friction pressure loss is expressed in a form depending on the Reynolds number. This form also includes shape pressure loss. The motion equation of the flow network is systematically solved by considering flow distribution in the reactor core, flow of the heat transport system, and flow in the heat exchanger. The SG models a heat exchange of the intermediate-loop fluid (sodium) and the water of the water-steam system. The water-steam system side of the SG treats various phases of water, ranging from the liquid single-phase flow of water, a liquid-vapor two-phase flow, to superheated steam.

The code models heat exchange as a counterflow between sodium-flowing tubes or between a sodium-flowing tube and a water-flowing tube. The Nusselt number used in the heat exchanger model is modeled in the form depending on both the Reynolds number and the Prandtl number. The heat exchanger is divided in the axial direction into cells, and models the heat transfer between the primary side fluid and intermediate side fluid through the heat transfer tube. The steel material of the shroud is considered as thermal capacity if necessary.

Heat transfer by forced convection, natural convection, and radiation can be modeled. The RVACS is modeled by combining natural convection and the heat transfer models. In this code, the heat transfer can be defined between the structure and fluid, in addition to the heat transfer model. Such heat transfer is selected from a choice of forced convection, natural convection, radiation, or the thermal conduction type. The fluid can take into account the movement of one dimension at the same time as the heat transfer; hence, the RVACS is modeled by a combination of this heat transfer and the movement of the fluid. In other words, the radiation heat transfers from RV to GV and from GV to collector, the convective heat transfer from GV to air and from collector to air, and the air motion (i.e., the process in which the air takes in the atmospheric air, receives the released heat, rises due to natural circulation head, and is dissipated into the atmospheric air) are all described.

The ARGO code models general events and interlocks such as the initiator and scram signal, e.g., a pump trip, scram, and flow path failure (leakage and opening or closing of a valve) in transition.

The transient trigger is expressed by the general interlock model: namely, pump trip, scram, occurrence and the loss of flow path, opening and closing of the valve, etc., actuated by a general signal that is, in other words, the specified delayed time in the event that the temperature, pressure, fluid level, head, flow rate, etc., decrease below or increase above the specified setpoint.

B.2 EXPERIENCE OF APPLICATION

The ARGO code has been developed by TOSHIBA and has been used in the area of safety analysis for Fast Reactor since 1990, such as an analysis of ULOF for "Bottom supported reactor vessel" an evaluation of passive safety for Japanese Demonstration Fast Reactor which was developed by Japan Atomic Power Company (JAPC) or an evaluation for comparison of an analysis model, which ARGO has, with an simple model. For 4S, this code has been applied for the safety analysis since 1990.

B.2.1 Verification and Validation of the ARGO Code

This section describes the methods used to verify the ARGO safety analysis code and to confirm its validity.

(i) Evaluation Model Development and Assessment Process (EMDAP)

The ARGO flow network code is used in the 4S safety analysis. Verification and validation of the application methodology of the ARGO code to the safety analysis of the 4S is made in accordance with the EMDAP method described in Regulatory Guide 1.203 and issued by the U.S. NRC.

This EMDAP method serves as guidance for those developing or evaluating analytical codes and provides details of the development process used to evaluate transients and accidents involving nuclear power plants. Analytical codes that are developed properly in accordance with the EMDAP method can be considered as acceptable codes by the U.S. NRC.

As shown in Figure B-2, the EMDAP methodology [B-8] consists of four elements and 20 steps. The four elements are as follows:

- Element 1: Defines the target event to be analyzed, and the phenomenon or process appearing in such event, as well as the Figure of Merit (FoM). It also selects the important phenomenon or process through the use of phenomena identification and ranking table (PIRT) and other tools. Furthermore, it clarifies the requirements for an evaluation model that evaluates the selected phenomenon and process.

- Element 2: Constructs a database for verifying a model, which evaluates the important phenomenon and process selected in Element 1. While the test data and plant data are included in this database, the scalability, similarity, and other aspects of such data must be evaluated.

- Element 3: Based on the requirements resulting from Element 1, the methodology produces a model and incorporates it into analytical code.

- Element 4: Verifies the EM developed in Element 3, using the database created in Element 2. Its evaluation process is classified into the following two evaluations:

  - Bottom-up evaluation: This evaluation focuses on evaluation of the individual basic models of analytical code. In this evaluation, the pedigree, applicability, and fidelity to the separate effects test (SET) data and scalability of the expression used in the models are evaluated.

  - Top-down evaluation: This evaluation focuses on the ability and performance evaluation of the analytical code. In this evaluation, the overall performance (governing equation, numerics, applicability, fidelity to integral effects test [IET] data and scalability) are evaluated.

(ii) Verification of the Safety Analysis Code

The verification of ARGO has been almost completed. The report for the work will be submitted in the near future.

(iii) Validation of the Safety Analysis Code

As described in subsection (i), the validation process of the safety analysis code, which is described in steps 14 and 18 of EMDAP, involves not only a comparison with theory or a calculation result by other analysis codes but also conducting a test analysis. Validation of the safety analysis code ARGO will be also conducted, using data obtained from the SETs, component performance tests (CPTs), and IETs.

SET is a test that provides information concerning a basic model and/or one phenomenon, and its test analysis is performed to demonstrate the sufficiency of fundamental and constitutive equations. It needs to be performed for the phenomena selected in the PIRT in step 4 (Figure B-2).

CPT provides performance data concerning the individual components of the 4S, and is performed to demonstrate the sufficiency of the component modeling in the ARGO code. It must be performed for the specific components in actual use.

IET is a test used to simulate the scaled 4S (1/2 or 1/3 scale), which is performed to evaluate the integrated system and the interaction between various components and processes in the system. In addition, it is provided to demonstrate accuracy in simulating a reaction inside the integrated system and offer information on the noding of a complicated system.

The validation test plan is shown in Table B-1.

Table B-1. Validation Test Plan (ARGO Qualification Database)

Test Type	Phenomena	Qualification Database
SET	Heat transfer between cladding and coolant	Modified Lyon's equation
	Radial heat transfer between subassemblies	Subbotin's equation
	Heat transfer from primary coolant to intermediate coolant	Heat tube test (IHX)
	Heat transfer between tube and air	Heat tube test (air cooler)
	Heat transfer between tube and sodium	Heat tube test (air cooler)
	Thermal radiation RV wall and GV wall	Radiation (theory)
	Thermal radiation GV wall and heat collector wall	
	Thermal radiation heat collector wall and concrete wall	
	Pressure loss in core region	Pressure loss test
	Pressure loss	Heat tube test (IHX)
	Pressure loss of sodium side	Heat tube test (air cooler)
	Pressure loss of air side	Heat tube test (air cooler)
	Reactivity feedback	One-point kinetics
	Fuel behavior	CDF test
Test Type	**Component**	**Qualification Database**
CPT	IHX	Test for IHX
	RVACS	Test for RVACS
	Air cooler	Test for air cooler
	PHTS	Natural circulation test
Test Type	**Integral Test / Plant Data**	**Qualification Database**
IET	Benchmark	Results of CFD and other transients and accidents codes
	Monju	Natural circulation test and others
	Integral test for 4S	Natural circulation test and others

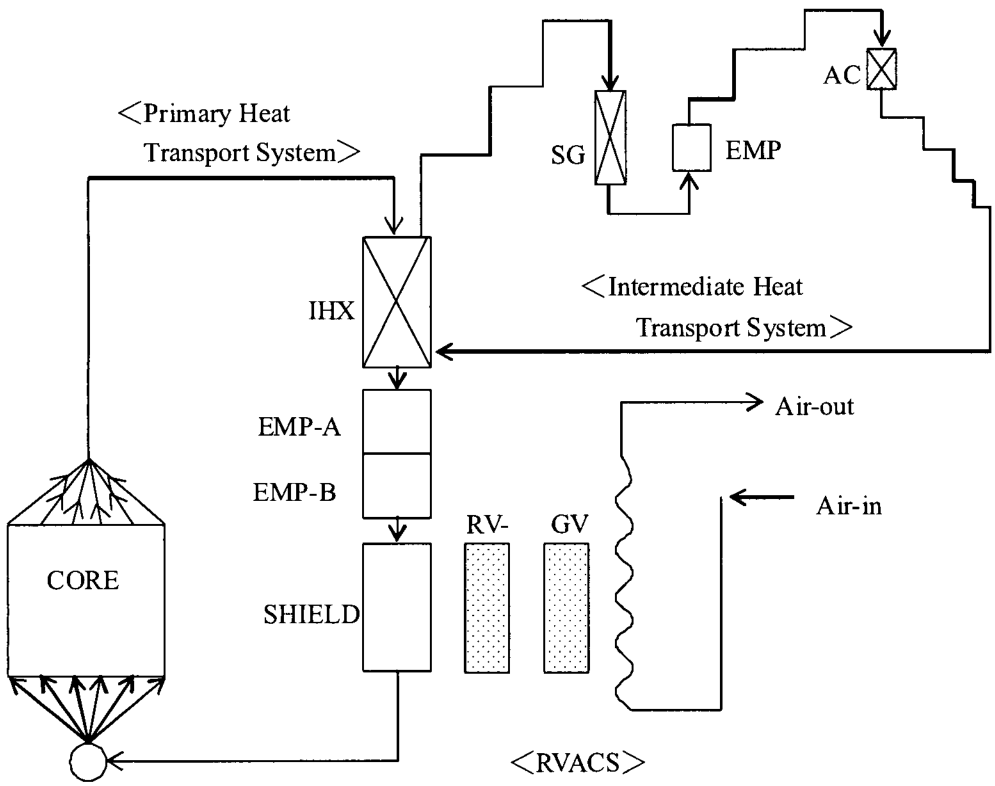

Figure B-1. Flow Network Model of 4S for ARGO Code Calculations

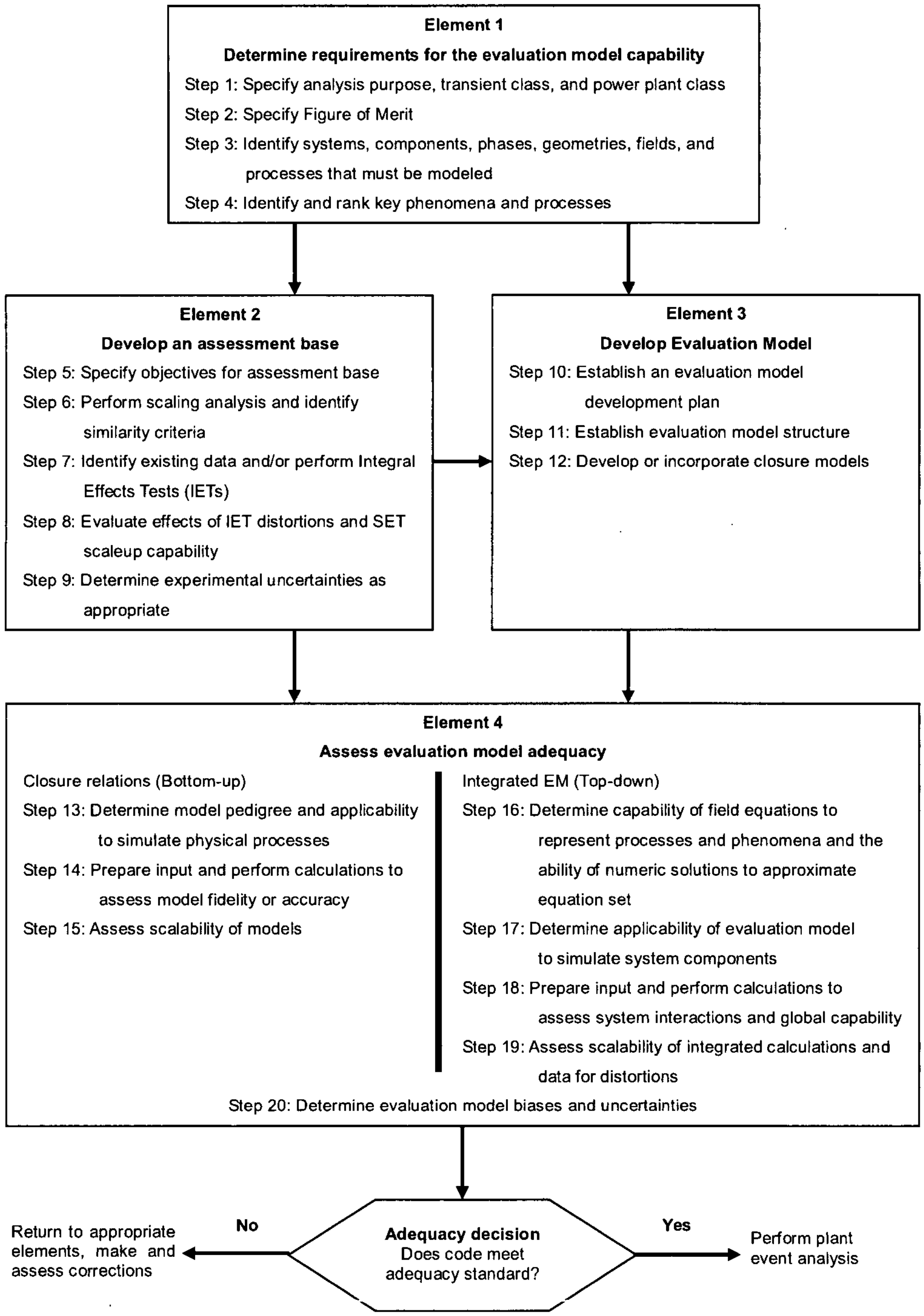

Figure B-2. Elements of Evaluation Model Development and Assessment Process (EMDAP) [B-8]

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