4.3 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 Code of Federal Regulations (CFR) sections, such as 10 CFR 50 Appendix A [4-19] and the Standard Review Plan (SRP) [4-20], 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. As shown in Table 4.3-1, the regulatory safety requirements result from a hierarchy of requirements [4-21]. The most important requirement stipulated for the reactor system is "Protect public health and safety," per 10 CFR 1.11 [4-22], 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. To simplify and make the selection of FoM easier, a number of PIRTs select physical quantities that are directly related to the requirements above [4-21] as FoM.

The following are the characteristics required for FoM [4-21]:

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

In the selection of FoM for the 4S LF PIRT, the team began with "protect public health and safety," which is the highest level requirement in Table 4.3-1, and then, "propagation of subassembly-to-subassembly failure" is derived as the third requirement focusing on the LF event. The propagation of fuel subassembly-to-subassembly failure is caused by the "propagation of fuel pin failure." The propagation of fuel pin failure is dependent on the behavior of the failed fuel. Fission gas release, the presence of the sodium bond, and/or material produced by eutectic reaction would be the causes that affect the failure propagation to the neighboring fuel pins. For the event scenario selected for this LF PIRT, however, neither the subassembly failure propagation nor the fuel failure propagation occur as evaluated in subsections 4.1.1 and 4.1.2. In the case where fuel melts after pin failure, there is a possibility of failure propagation due to release of the melted fuel outside the fuel cladding. As a result, a total of three FoMs are selected, i.e., fuel and cladding temperature to judge the occurrence of fuel

melting and eutectic reaction, and coolant temperature to judge occurrence of coolant boiling, which would be an initiator of the fuel melting. Although fuel and cladding temperatures are not measured directly in the 4S, they can be analytically derived.

Table 4.3-1. Hierarchy of Regulatory Safety Requirement and Local Blockage Criteria

Level	Source	Criteria	Directly Related to Issue	Directly Related to Phenomena	Easily Comprehended	Explicit	Measurable
1	10 CFR 1.11 [4-22]	Protect public health and safety	Primary Regulatory Issue				
2	10 CFR 100.11 [4-23]	Limit fission product release	✓	✓	–	–	–
3	–	Possibility of subassembly-to-subassembly failure propagation	✓	✓	–	–	–
4	–	Possibility of fuel failure propagation	✓	✓	–	–	–
5	–	Deposition of melted fuel	✓	✓	–	–	–
	–	Fuel melting	✓	✓	✓	✓	✓
6	–	Fuel temperature, cladding temperature, and coolant temperature	✓	✓	✓	✓	–(1) ✓(2)

Notes:
1. Fuel and cladding temperature cannot be directly measured, but can be analytically derived.
2. Coolant temperature is measurable.

4.4 PHENOMENA IDENTIFICATION

Plausible phenomena are identified in this section. 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 4.4-1 describes the plausible phenomena that were identified in the LF PIRT. In this table, "Code" in the rightmost column is the index provided for each phenomenon, which is also used in Section 4.5.

The total number of plausible phenomena is 50.

Table 4.4-1. Descriptions of Plausible Phenomena

Phenomenon	Description	Code
Pressure loss in core region	• Pressure loss along the coolant flow path of the bundle in the core/fuel assemblies is caused by acceleration, friction, and form losses. They include losses from flow contraction and expansion depending on the inlet and outlet geometry, and losses from the wire 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. • Pressure loss would increase upon fission gas release by fuel pin failure. • Flow mode of the primary coolant is forced convection during long-term cooling after shutdown.	a01
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.	a02
Reactivity feedback before fuel failure	• Reactivity feedback in the core is caused by thermal expansion and contraction of the fuel, coolant, structural materials, and core support plate, which is caused by the decrease or increase of reactor temperature and by the Doppler effect. • Power increases when positive reactivity is added. • Not only the locally failed area but also the total core is affected by reactivity feedback. • Reactivity would be inserted as a result of gas passage. • Temperature change effects reactivity feedback. Temperature is uniform during rated operation, but it varies during the transient. • Scram reactivity is added at shutdown.	a03

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
Gap conductance between fuel and cladding	• There is a sodium bond in the fuel pin, so the gap between the fuel slug and cladding is filled with liquid sodium at the beginning of life. • Therefore, the gap conductance between the fuel slug and cladding is dependent on the thermal conductivity of sodium. • At the end of life, after 2% burnup, bond sodium moves to the gas plenum area, and cladding and fuel slug come into contact. • Gap conductance is dependent on whether the fuel is irradiated or unirradiated.	a04
Heat transfer between cladding and coolant	• Heat transfer between the cladding and coolant is dependent on coolant flow velocity near the cladding surface, the shape and diameter of spacer, and P (fuel pin pitch)/D (fuel pin diameter). • Heat transfer is affected when fission gas is released.	a05
Intra-assembly flow distribution	• Flow distribution in the core is adjusted by a flow orifice at the bottom of the fuel assembly. • Flow distribution inside a fuel assembly is controlled by bundle geometry. In the natural convection decay heat removal phase, it is also controlled by the temperature distribution inside the bundle. • The temperature distribution inside the bundle is controlled by the flow distribution inside the subassembly and by heat transfer between subassemblies. • Flow distribution inside the fuel assembly may deviate from the initial prediction due to uncertainties, e.g., manufacturing and fuel geometry changes in the 30 years of operation for the fluid resistance in fuel assembly. Then, these variations result in fuel, cladding, and coolant temperature fluctuation.	a06
Heat capacity of the 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.	a07

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
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 could occur due to local blockage that results from decrease of coolant flow followed by local temperature rise. • When coolant boiling occurs, fuel pin integrity is not likely maintained, and fission gas is released from the heated pins. Also, positive reactivity is added locally and core power increases, which causes a possibility of further disintegration of the fuel pin. • Coolant boiling will never happen under DBA conditions [4-16]. • Under very unlikely BDBA conditions, e.g., large-scale local blockage, eutectic reaction could occur, followed by fission gas release, which could result in local coolant boiling.	a08
Nominal core power	• There are two types of heating for core power, fission heating and fission product heating. The main heating for core power is fission heating. Rated core power is 30 MWt.	a09
Decay heat	• Decay heat is heat released by the decay of radioactive fission products. Even if the reactor shuts down, the radioactive isotopes 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. • The decay heat table for a fast reactor is similar to that of a LWR.	a10

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
Eutectic reaction between fuel and cladding	• Eutectic reaction is the phenomena in which alloy with a low melting point is produced at the contact boundaries of the two different kinds of metal. • 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 higher than 650°C and becomes more severe as temperature rises. • Rapid eutectic reaction occurs around 1100°C, which is higher than fuel melting point. • Eutectic reaction does not occur at rated operation (where the maximum temperature of the hottest pin is estimated to be 600°C, taking into account uncertainties). • Under an enrichment error condition during which fuel temperature reaches 650°C, fuel failure may occur at the end of life. • Under a local blockage event, fuel failure may occur because temperature increases locally, which may initiate a eutectic reaction. • After normal shutdown, fuel temperature is low enough that eutectic reaction does not occur.	a11
Temperature dependence of physical properties of material	• Physical properties such as specific heat, density, thermal conductivity, and creep characteristics of core components are temperature dependent.	a12
Fission gas transport from fuel slug into gas plenum	• The fission gas accumulating in the fuel slug results in pores in the fuel slug, increasing its pore density. Holes that lead to the outside of the slug are formed when these pores combine and link. Then, fission gas is released to the outside of the fuel slug through these holes and is transferred to the gas plenum region. • Sodium enters the fuel slug through the hole (sodium plugging). • Effective conductivity changes as a result of the variation of the pore density and the penetration of sodium. • Fission gas release rate from fuel slug is dependent on the fuel temperature.	a13

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
Fission gas transport from fuel to sodium bond, and sodium in primary system	• Fission gas released from the fuel into the sodium bond leaks into and is transported in the primary system sodium in case of a fuel failure. • Some of the released cesium and iodine would be dissolved in sodium.	a14
FP and fission gas transport from sodium in primary system to cover gas	• FPs and fission gases are transported through the sodium in the primary system and to the cover gas in case of a fuel failure. • Some of the released FPs would be dissolved in sodium.	a15
Flow induced vibration in a subassembly	• Fuel pin bundle vibrates from fluid and structure interaction when coolant flow velocity around the fuel pin bundle in the fuel assembly is high. • There is possibility of fuel pin failure due to fretting. • Vibration is small for a wire spacer compared to that of a grid spacer.	a16
Coolant flow between wrapper tubes	• Inter-subassembly heat transfer is enhanced by heat conduction and convection of sodium between the wrapper tubes during natural circulation decay heat removal. • Coolant flow between wrapper tubes influences inter- and intra-subassembly heat transfer phenomena during natural circulation decay heat removal.	a17
Maldistribution of the core flow in both intra- and inter-subassemblies (deviation of the mass flow rate in all the core subassemblies from ideal flow distribution)	• Coolant flow rate in an assembly may deviate from the predicted value due to manufacturing errors, pre-operation errors (detection failure for flaws in assembly orifice, fuel pin, wire spacer, and wrapper tube, and uncertainty and imperfection of analytical model (considered the cause of the deviation at natural circulation flow rate range).	a18
Radial power distribution	• Neutrons are not generated at the fixed absorber region and the center of the core where a shutdown rod is placed. 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.	a19

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
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 the 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.	a20
Cladding melting (w/o eutectic)	• The cladding material is HT-9, with a melting point of 1400°C. The melting point of fuel is 1100°C for metallic fuel, which melts earlier than cladding. The eutectic reaction starts at 650°C and erosion of the inner side of cladding occurs. The creep strength of the cladding reduces with its temperature increase, which results in cladding failure. • There is sufficient margin to cladding melting. • Rapid eutectic reaction occurs before the temperature reaches 1400°C when cladding melts.	a21
Transportation of molten cladding	• Molten cladding freezes in contact with liquid sodium and its debris/fragments are dispersed by sodium vapor or FP gas pressure and gravity, transported by liquid sodium flow in the coolant channels or flow out of the subassemblies.	a22
Relocation of molten cladding	• When molten cladding is redistributed in the subassembly, it solidifies and blocks the coolant subchannels. Fuel pin failure propagation may occur due to local temperature increase at the blocked area.	a23
In pin fuel melting	• Fuel melting occurs prior to the cladding melting for metallic fuel. The melting point of the U-10%Zr fuel alloy should be approximately 1100°C, depending on the composition of the alloy. • The power density of the 4S core is small and the core outlet becomes the maximum temperature region. The high thermal conductivity of metal fuel makes fuel melting occur around the top of the fuel slug.	a24

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
In pin fuel motion of the molten fuel	• Molten fuel can move inside the cladding before the cladding fails. • Negative reactivity is inserted when melted fuel is relocated in the gas plenum region.	a25
Eutectic formation	• Molten fuel reacts with inside wall of the cladding, formulating the liquid phase rapidly. The cladding fails due to erosion and creep.	a26
In pin fuel motion of the eutectic material	• Driven by gravity and pressure inside the pin, liquidus fuel relocates inside the cladding before cladding fails. • Negative reactivity is inserted when melted fuel is relocated in the gas plenum region.	a27
Outside fuel motion of the eutectic material	• After failure of the cladding, the liquidus fuel flows into the coolant path by fission gas pressure and expansion of the fission gas. Then, the liquidus fuel is dispersed by liquid sodium, sodium void, and gravity.	a28
Eutectic material freezing (both inside and outside pin)	• The eutectic material is dispersed and frozen in the short term after it flows into the coolant path. Negative reactivity is inserted when the solidified fuel flows out of the core. In the case of solidified fuel trapped in the gap of the wire spacer between the fuel pins, however, local blockage occurs and small positive reactivity due to coolant boiling may be inserted followed by fuel pin failure propagation. • Molten fuel would be produced when fuel temperature reaches 1100°C around which rapid eutectic reaction starts. Molten fuel is not produced in the 4S core during the selected scenario with temperature increase about 650°C or less.	a29
Failure of wrapper tube	• Wrapper tube failure is caused by erosion due to eutectic reaction of melted fuel and tube wall.	a30
Porosity of blocking substance	• The coolant flow in the subassembly would be maintained partially even if the coolant path is blocked locally if the blocking material has porosity.	a31
Size of the blocking substance	• If the size of the blocking material is small enough, it would flow out of the subassembly. If it is large, it would be trapped at the fuel assembly inlet.	a32

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
Blockage form	• The coolant flow rate is varied depending on the form of the blocked material.	a33
Location of blockage	• Local blockage could occur at core entrance, bundle region, and core outlet.	a34
Heat generation rate of blockage	• The blockage is heated during reactor operation. The heat generation rate of the blockage depends on the blocked material. The heat generation rate would be higher if the blockage consisted of fuel material.	a35
Blockage strength	• The behavior of local blockage would be changed in case blockage is fragile/firm. For example, in case the blockage is fragile, the blockage would be dispersed by coolant flow.	a36
Blockage materials	• The heat transfer behavior at the local blockage area would be changed dependent on the heat transfer rate of the blocked materials.	a37
Enrichment	• There are two types of subassemblies for the 4S core: 17% enrichment for the inner core, and 19% for the outer core. • A local temperature increase occurs when an overenriched fuel pin is loaded at the region intended for low flux, resulting in the possibility of fuel pin failure at the end of plant life.	a38
Thinning of the wire spacer due to corrosion and fretting	• The outer surface of the wire spacer would be more prone to corrosion due to chemical reaction [4-7]. The amount of the erosion is expected to be about 45 μm (design value), which is a trivial amount compared to its diameter (1.05 mm).	a39
Wear mark on the cladding	• A wear mark is a scratch on the pin where the pin wears against the wire. The subassemblies are designed to avoid this occurrence of the wear mark referencing the existing data of the wire spacer [4-9].	a40
Bowing of fuel pin	• Swelling of the fuel pin results in its bowing, which narrows the coolant passage, and in the worst case, point/line contact between adjacent pins occurs.	a41

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
Fission gas plenum volume	• If fuel fails, fission gas could impinge upon the neighboring pin, which would result in pin failure propagation due to loss of heat removal from the pin. • The amount of fission gas depends on the fission gas plenum volume. • The fission gas plenum volume of the 4S fuel pin is about 300 cm^3.	a42
Fission gas pressure	• As in a42, there is a possibility of pin failure propagation due to fission gas impingement after fuel failure. • Fission gas pressure influences the gas release rate.	a43
Location of gas impingement	• As in a42, there is a possibility of pin failure propagation resulting from deterioration of cooling capability due to fission gas impingement after fuel failure. • The location where such a gas impingement occurs has influence on the FoM because there are two types of flow regions in the 4S core and core power density has a distribution in both radial and axial directions.	a44
Fuel burnup	• The integrity of the fuel depends on the pressure inside the cladding. The pressure increases as fission gas is generated depending on the progression of fuel burnup. • The 4S average fuel burnup is 34,000 MWd/t.	a45
FP gas release duration	• If a fuel pin fails, the fission gas ejects from the failed pin toward a neighboring pin. The fission gas prevents coolant from cooling the neighboring pin. The longer the duration of the fission gas release, the larger the possibility of failure propagation.	a46
Heat transfer coefficient at fission gas release (gas blanketing)	• If a fuel pin fails, the fission gas ejects from the failed pin toward a neighboring pin. The heat transfer rate between the cladding and coolant is decreased by the fission gas, which results in the deterioration of cooling capability.	a47
Heat transfer deterioration due to flow stagnation	• The heat transfer rate of the coolant surface decreases when flow stagnation occurs, which results in a decrease in heat transfer from fuel to coolant. • The location where the flow stagnation occurs is above the blockage at rated operation, while it depends on the natural circulation flow rate during the shutdown phase.	a48

Table 4.4-1. Descriptions of Plausible Phenomena (cont.)

Phenomenon	Description	Code
Gas bubble breakup and buoyancy	• In case of fission gas release upon fuel pin failure, the gas would be expected to dissipate into smaller-size bubbles and flow upward due to buoyancy.	a49
Failed fuel detection (reactor cover gas radiation monitoring)	• If a fuel pin fails, fission gas is transported from the fuel pin through the primary coolant, and is released into the cover gas region. The fuel pin failure is detected by the monitor that measures the density of fission gas in the cover gas region.	a50

4.5 PIRT RANKING RESULTS

The process of establishing the ranking for relative importance of the phenomena seen in subsystems, and for their current state of knowledge (SoK), is the primary activity of the PIRT process [4-24].

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. 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 another fast breeder reactor (FBR) in Japan, namely Monju.

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

4.5.1 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.

4.5.1.1 Ranking Scale of Relative Importance of Phenomena

In this PIRT, 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 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.

4.5.1.2 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.
Not applicable (N/A):	Phenomenon classified as N/A at the ranking of their relative importance.

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. 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 of is judged as "Known."

4.5.2 Priority of Ranked Phenomena

In the PIRT report for DBAs [4-16], to identify the phenomena that require further investigation, all the phenomena in consideration were listed in order of priority. Following the same methodology, important phenomena in this LF PIRT are identified as follows.

The priorities of phenomena are determined using a five-level scale by combining the relative importance of the phenomena and the SoK, as shown in the matrix in Figure 4.5-1. The phenomena with importance ranking "N/A" are excluded because their effect on FoM is negligible.

The numbers assigned in the nine cells in the matrix in Figure 4.5-1 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. Those phenomena with higher priorities such as Priority 1, 2, and 3 are considered to have relatively high importance.

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)

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

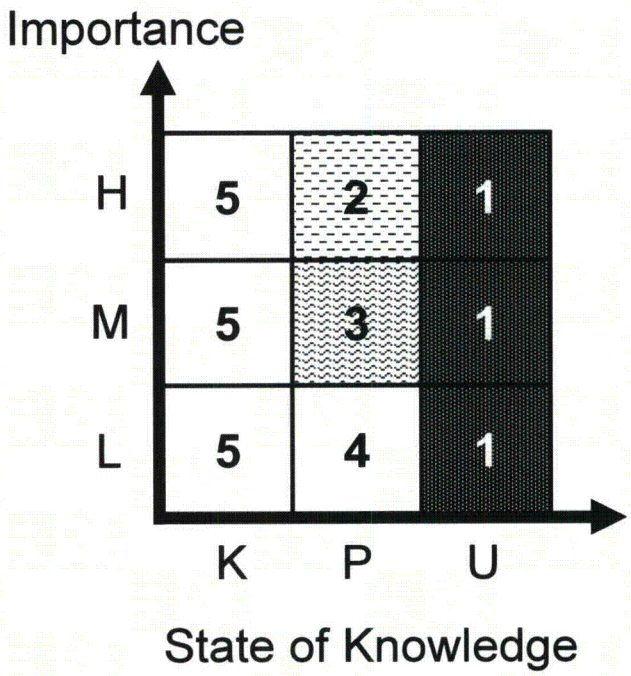

Figure 4.5-1. Priority for Further Investigation

4.5.3 Ranking Results

As described at the beginning of this section, the initial ranking of the PIRT phenomena was established by the Toshiba engineers based on their knowledge and experience. After that, the results are revised based on discussions with the IRAP.

In this section, final results of the rankings of the phenomena are shown, taking into account the 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 analysis are one factor used to determine the relative importance of the phenomena.

Table 4.5-1 shows the ranking results. This table contains not only the ranking results but also an explanation of the results with rationales and priority of the phenomena.

Table 4.5-1. PIRT Results for LF

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a01	Pressure loss in core region	H	H	K	5	– Pressure loss coefficient changes and flow decreases at locally blocked area, which results in power-to-flow (P/F) mismatch, and effects on FoM. Though it is a local behavior, its impact can be large depending on where the blockage occurs. – The impact on P/F can be large in the case of enrichment error where the overenriched subassembly is loaded at lower flow region in the core.	– Data on flow rate regarding pressure loss coefficient of core region at rated operation were obtained by tests [4-25] [4-26] [4-27] [4-28]. Data for 4S design have also been obtained [4-29] [4-30]. – The knowledge at low flow rate such as natural circulation or local blockage is obtained by analyses [4-31] [4-32] [4-33].
a02	Natural convection	N/A	H	K	5	– This phenomenon is not applicable at rated operation since the flow mode is forced convection. – For transient conditions, the electromagnetic pump trips after the scram, so this phenomenon has an impact on the FoM at the decay heat removal state after scram.	– Flow behavior in and around the core at normal operation can be evaluated analytically. – 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 [4-31].

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a03	Reactivity feedback	L	L	P	4	– There is a small amount of reactivity change at rated operation. – For a transient, there are small effects on FoM from reactivity feedback because of the insertion of scram reactivity.	– As a result of the criticality test, the analysis code is sufficiently verified from the organized nuclear data. Therefore, reactivity feedback can be estimated. – The knowledge of shape variation caused by temperature change at transient conditions is insufficient. Therefore, there is uncertainty in the evaluated value of reactivity feedback attributed to shape variation.
a04	Gap conductance between fuel and cladding	M	M	K	5	– The gap between fuel and cladding is about 100 μm. – Most of the bond sodium is transported to the gas plenum region when burnup exceeds 2%, which is 10 years after plant startup. – However, even at the end of plant life, bond sodium remains in the thinner gap between fuel slug and cladding, so there is gap conductance.	– Since the uncertainty of thermal conductivity for fuel, sodium, and cladding, which affects the evaluation of gap conductance, is small, uncertainty of the gap conductance is also small.
a05	Heat transfer between cladding and coolant	H	H	K	5	– In case of an LF, local flow rate and shape of the wire spacer change, so the heat transfer coefficient varies locally. – If the fuel fails, fission gas is released, which results in the deterioration of the heat transfer rate between cladding and coolant.	– The correlation equation of Nu number between cladding and coolant has been investigated. There is sufficient knowledge for many fuel pins, such as those of FFTF and CRBR [4-34][4-35]. – For the 4S, a modified Lyon's correlation is used. – Data for heat transfer at a local blockage condition have been obtained analytically [4-33].

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a06	Intra-assembly flow distribution	M	H	P	2	– Mass flow rate in bundle is maintained even if some subchannels are blocked, so the effect on FoM is not so large.	– Flow distribution among assemblies at rated operation can be evaluated by pressure loss coefficient of each assembly [4-34] [4-35]. – Data of the pressure loss coefficient of core region can be obtained from testing. – For low Re number phases, such as natural circulation, it is difficult to measure the pressure loss (coefficient), so the data contain uncertainty.
a07	Heat capacity of the assemblies	L	H	K	5	– This phenomenon largely effects FoM during the transient phase, when flow rate varies.	– Materials used and weights of component structures can be evaluated from the design data. – There are sufficient data for specific heat values.
a08	Coolant boiling	N/A	N/A	N/A	N/A	– Boiling does not occur in the range of temperature increase described in subsection 4.1.1.	– N/A
a09	Nominal core power	H	M	K	5	– For the enrichment error condition, core power would be changed from 30 MW thermal.	– Since the core nuclear data are well organized, power can be calculated [4-36] [4-37].
a10	Decay heat	N/A	M	K	5	– This phenomenon is not applicable at rated operation. – This phenomenon affects the FoM during a transient when the core is heated mainly by decay heat.	– Nuclear data are well-organized [4-38]. – Since the decay data of FPs are also well-organized, decay heat can be calculated.

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a11	Eutectic reaction between fuel and cladding	H	H	K	5	– Eutectic reaction would be expected to occur at the LF region during rated operation or transient conditions, which has a large effect on FoM.	– There are data obtained from existing reactors pertinent to the starting temperature and the growth rate of the eutectic reaction [4-39] [4-40] [4-41]. – Data were obtained from heating tests using a fuel pin after irradiation. – Burnup composition of fuel affects the starting temperature and the growth rate of eutectic reaction, but the data are limited.
a12	Temperature dependence of physical properties of materials	M	M	K	5	– Temperature varies at LF events during rated operation and transient conditions, which results in a change of heat transport behavior, so there is some effect on FoM.	– There are sufficient data for metallic fuel, structural materials, and coolant. [4-40]
a13	Fission gas transport from fuel slug into gas plenum	H	M	K	5	– This phenomenon has a large impact on the fuel failure detection system during rated operation. – However, this phenomenon is a long-term effect, so there is small impact for transient conditions that are a short-term event.	– There are sufficient data for gas release rate in the fuel pin [4-28] [4-41].
a14	FP transport from fuel to sodium bond, and sodium in primary system	L	L	P	4	– Cover gas monitoring system detects noble gases that are not dissolved in the sodium, so the effects on FoM are small.	– There is sufficient knowledge of nuclides dissolved in sodium [4-42] [4-43] [4-44] [4-45] [4-46]. – Behavior of transportation is known, but there is uncertainty about that of dissolution into sodium that is mixed with noble gases and aerosol.

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a15	FP and fission gas transport from sodium in primary system to cover gas	H	M	P	2	– Same as No.13.	– Same as No.14.
a16	Flow-induced vibration in a subassembly	M	L	K	5	– In case of a wire spacer failure, the unwrapped fuel pin would be expected to vibrate and develop wear. – However, the magnitude of vibration would be small because the 4S fuel pin has a large diameter and the flow rate is low. – The effect of this phenomenon on FoM is less than thermal behavior, so the importance rank would be relatively smaller. – The effect on FoM is low for transient conditions since flow rate becomes low as the EM pumps trip.	– The evaluation of the flow-induced vibration has been generalized from existing test data using water as a fluid. – There are enough test data using sodium for fuel assemblies with wire spacers [4-47] [4-48].
a17	Coolant flow between wrapper tubes	L	L	P	4	– Heat transfer between subassemblies has a small impact on FoM in the selected LF events.	– In the natural circulation phase, there are test data of sodium flow behavior in gaps between fuel assemblies and their flow resistance [4-49]. – However, the data are highly relative to test scale and geometry, and there is insufficient knowledge regarding scaling. – Therefore, there would enough knowledge at rated operation while it is limited for the transient phase.

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a18	Maldistribution of the core flow both intra- and inter-subassemblies (deviation of the mass flow rate in all the core subassemblies from ideal flow distribution)	H	M	K	5	– This phenomenon has a large uncertainty associated with the designed flow distribution, and it would be expected to impact the FoM at rated operation. – If the total flow of a subassembly exceeds 5%, the outlet temperature of the subassembly would increase by several percent. – The flow rate is relatively low for the transient phase.	– Although the cause of maldistribution in the core flow contains uncertainties [4-27] [4-34] [4-50] [4-51] [4-52] [4-53] [4-54] [4-55] [4-56] [4-57] [4-58] [4-59] [4-60] [4-61] [4-62] [4-63] [4-64] [4-65] [4-66] [4-67] [4-68] [4-69] [4-70] [4-71] [4-72], there is sufficient knowledge for the behavior resulting from the maldistribution.
a19	Radial power distribution	H	M	K	5	– There is a large impact on FoM in the case of enrichment error where the radial power distribution changes at the over-enriched subassembly during rated operation. – Power is relatively small for transient conditions.	– Ample power distribution data have been obtained by extensive criticality testing and experimental reactors. – There are also validated calculation codes.
a20	Axial power distribution	H	M	K	5	– Same as a19.	– Same as a19.
a21	Cladding melting (w/o eutectic)	N/A	N/A	N/A	N/A	– The melting point of cladding is higher than that of the fuel slug. – Eutectic reaction precedes cladding melting.	– N/A
a22	Transportation of molten cladding	N/A	N/A	N/A	N/A	– Same as a21.	– N/A
a23	Relocation of molten cladding	N/A	N/A	N/A	N/A	– Same as a21.	– N/A

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a24	In-pin fuel melting	M	H	K	5	– The melting point of the fuel decreases as burnup proceeds. It is about 1100°C initially, but it decreases to 800°C over core life. If coolant temperature increases to 800°C, for a certain period of time, iron from cladding mixes with the fuel slug . Iron does not mix with fuel when coolant temperature is under 600°C. – This phenomenon occurs earlier than cladding melting. – There is enough margin to the fuel melting point at rated operation (coolant temperature < 600°C); however, the margin would become smaller for transient conditions.	– Sufficient data were obtained from the TREAT tests [4-73] [4-74] [4-75] [4-76] and analysis [4-77] [4-78] [4-79].
a25	In-pin fuel motion of the molten fuel	M	H	K	5	– Same as a24.	– Same as a24.
a26	Eutectic formation	M	M	K	5	– During the scenarios considered in this LF PIRT, there is enough margin to the rapid eutectic start temperature (around 800°C), so even if the eutectic reaction starts around 650°C, it progresses at a very low speed.	– Same as a24.
a27	In-pin fuel motion of the eutectic material	L	L	K	5	– There is no space for melted fuel to relocate inside the fuel pin.	– Same as a24.
a28	Outside-fuel motion of the eutectic material	H	M	P	2	– Coolant flow is prevented by this phenomenon. – Eutectic reaction with neighboring pin would occur. – The effect on FoM is larger at rated operation because the power density is high.	– There is sufficient knowledge obtained from the TREAT tests [4-73] [4-74] [4-75] [4-76] [4-77], analysis [4-78] [4-79], and the SLSF test [4-80].

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a29	Eutectic material freezing (both inside and outside of the pin)	H	M	K	5	– FPs in the fuel slug would be released outside the fuel pin because metallic fuel has smaller margin to the eutectic start temperature at the top of the fuel slug where the temperature is higher. – The frozen material can be swept out from the subassembly or become a blockage in the subassembly. In the former, fission gas plays the role of a driving force. The fission gas pressure at the end of plant life is expected to be 3 MPa, which is enough to sweep the material.	– Same as a24.
a30	Failure of wrapper tube	L	L	P	4	– This phenomenon is unlikely in the selected events.	– There are test data for wrapper tube failure obtained from the CABRI, SCARABEE, and EAGLE tests [4-81] [4-82] [4-83].
a31	Porosity of blocking substance	H	H	P	2	– This phenomenon is important beyond a certain limit where it causes reduction of the flow.	– There are some experimental data regarding local blockage in the subassembly [4-32] [4-73] [4-84] [4-85] [4-86] [4-87] [4-88] [4-89] [4-90] [4-91] [4-92] [4-93] [4-94] [4-95] [4-96] [4-97] [4-98]; however, the data focused on metallic fuel as well as the 4S fuel specifications are limited.
a32	Size of blocking substance	H	H	P	2	– This phenomenon is important beyond a certain limit where it significantly reduces the flow.	– Same as a31.
a33	Blockage form	H	H	P	2	– If the whole fuel pin is surrounded by blockage, the impact on FoM is large at rated operation and transient conditions.	– Same as a31.
a34	Location of blockage	H	H	P	2	Because there is power distribution in the axial and radial directions, the location of blockage has a large influence on FoM.	Same as a31.

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance		SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
Code	Phenomenon	Rated Operation	Transient				
a35	Heat generation rate of blockage	H	H	P	2	– In addition to a31 through a34, the FoM is affected if the blockage generates heat, e.g., frozen eutectic material.	– Same as a31.
a36	Blockage strength	H	H	P	2	– The FoM is largely affected by this item when the material is strong enough against coolant flow to block the coolant path.	– Same as a31.
a37	Blockage materials	H	H	P	2	– The lower the heat transfer rate of the blocked material, the larger the increase of the fuel and cladding temperature would become.	– Same as a31.
a38	Enrichment	H	H	K	5	– The deviation of designed enrichment directly causes FoM at rated operation and transient conditions.	– There is enough experience regarding manufacturing of the fuel pin.
a39	Thinning of the wire spacer due to corrosion by chemical reaction	L	L	K	5	– The amount of thinning from corrosion is small.	– There are sufficient test data for corrosion of the wire spacer [4-7].
a40	Wear mark on fuel cladding	M	L	K	5	– Same as a38.	– Same as a38.
a41	Bowing of fuel pin	L	L	P	4	– Referring to the results of the analysis for pin-to-pin contact of the neighboring pins (see subsection 4.1.2.3), the impact on FoM is small.	– There are data on fuel pin bowing [4-97], but the 4S fuel pin diameter is larger than that evaluated. – Some analysis has been conducted under conditions narrower than the 4S coolant flow path [4-99] [4-100].
a42	Fission gas plenum volume	M	M	K	5	– Fission gas volume affects duration of fission gas release (gas blanketing).	– Fission gas plenum volume is clearly specified for the fuel design.

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance		SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
Code	Phenomenon	Rated Operation	Transient				
a43	Fission gas pressure	M	M	K	5	– Fission gas pressure influences the duration of gas blanketing and the area where the gas impinges.	– Fission gas pressure depends on the fuel specification, burnup duration, ratio of generation of noble gases from nuclear fission, and the ratio of noble gases transported from the slug to the gas plenum. – There are existing research and radiation data regarding fuel specification, burnup duration, and generation ratio of noble gases. Also, data related to the behavior of gas transportation from slug to gas plenum were obtained from EBR-II [4-101] [4-102].
a44	Location of gas impingement	M	M	P	3	– Gas impingement influences the neighboring pin as external pressure on the pin.	– Some data exist regarding the effect of fission gas release [4-4] [4-34] [4-103] [4-104]. – Existing data, however, cannot be directly applied to 4S because of the difference in fuel geometry. – Although enough margin to the fuel failure due to gas impingement would be expected for the 4S fuel since it utilizes metallic fuel with low power density, tests for 4S equivalent to CABRI or SCARABEE could be profitable to acquire data.

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance		SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
Code	Phenomenon	Rated Operation	Transient				
a45	Fuel burnup	H	H	K	5	– This phenomenon affects the amount of fission gas generated that will be released in the cover gas region, so this item is important for the fuel failure detection system.	– Same as a9.
a46	Fission gas release duration	H	H	K	5	– The fission gas release duration is equal to the duration of decrease of heat removal at the neighboring pin. If the duration is longer, there is a possibility of failure of the neighboring pin. Therefore, the effect on FoM is large.	– The duration of fission gas release can be derived from fission gas plenum volume and pressure, size of the pin failure, and coolant pressure. – Also, there are some data regarding the fission gas release duration [4-90] [4-103] [4-104].
a47	Heat transfer coefficient at fission gas release (gas blanketing)	H	H	P	2	– Although the pressure in the fission gas plenum of 4S is small, the amount of fission gas would be large since the fuel pin diameter is relatively large compared to other fast reactors. Therefore, the duration of the fission gas release would be longer, which may result in prevention of heat removal of the adjacent pin followed by failure propagation.	– Same as a43. – Although there are plenty of data regarding fission gas impingement, the behavior of heat removal of the adjacent pin is dependent on the size of the gas leak hole and gas release rate.
a48	Heat transfer deterioration due to flow stagnation	H	H	K	5	– Flow stagnation occurs when the flow path is not intact as designed, e.g., under local blockage or fission gas leakage, etc. On this occasion, the heat removal performance is deteriorated, which affects the FoM.	– There are sufficient data regarding the behavior of coolant flow under a deteriorated flow path [4-91] [4-100] [4-104].
a49	Gas bubble breakup and buoyancy	L	L	P	4	– The time scale of the buoyancy-induced transport rate of gas bubbles is on the order of seconds, but that of the fission gas release duration is on the order of ms. Therefore, the effect of the blanketed area on FoM is small. – Buoyancy effects impact the fuel detection system.	– The data regarding metal fuel are limited [4-105].

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

Table 4.5-1. PIRT Results for LF (cont.)

	Event: Local Fault						
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance					
Code	Phenomenon	Rated Operation	Transient	SoK	Priority	Ranking Rationale for Importance	Ranking Rationale for SoK
a50	Failed fuel detection (reactor cover gas radiation monitoring)	H	H	K	5	– Important to monitor to detect the occurrence of LF. – In the absence of high fission gas plenum pressure, clad failure could be caused by manufacturing defects, random causes, etc. Notice that eutectic formation will occur only after fuel swelling removes the sodium bond from the clad, which will happen only after a significant amount of fission gas is released to the gas plenum, which can then be used to detect clad failure.	– There is enough experience for cover gas radiation monitoring methods.

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

4.5.4 Organizing the Results

The purpose of this section is to organize the results obtained in subsection 4.5.3. Table 4.5-2 shows only the phenomena with Priorities of 2 and 3. In this LF PIRT, no phenomena are assigned a Priority of 1. That means that there are no phenomena whose SoK is unknown for this event.

As a result, 12 phenomena were identified to be of relative importance for further investigation; 11 phenomena with Priority 2 and 1 phenomenon with Priority 3.

Table 4.5-2. Rearranged Final LF PIRT Results

	Event: Local Fault				
	Figures of Merit (FoM): Fuel Temperature, Cladding Temperature, and Coolant Temperature	Importance			
Code	Phenomenon	Rated Operation	Transient	SoK	Priority
–	Highly ranked phenomena with partially known SoK				
a06	Intra-assembly flow distribution	M	H	P	2
a15	FP and fission gas transport from sodium in primary system to cover gas	H	M	P	2
a28	Outside fuel motion of the eutectic material	H	M	P	2
a31	Porosity of blocking substance	H	H	P	2
a32	Size of blocking substance	H	H	P	2
a33	Blockage form	H	H	P	2
a34	Location of blockage	H	H	P	2
a35	Heat generation rate of blockage	H	H	P	2
a36	Blockage strength	H	H	P	2
a37	Blockage materials	H	H	P	2
a47	Heat transfer coefficient at fission gas release (gas blanketing)	H	H	P	2
–	Moderately ranked phenomena with partially known SoK				
a44	Location of gas impingement	M	M	P	3

4.6 SUMMARY OF PIRT FOR LOCAL FAULT EVENT

In Section 4.5, the phenomena were selected that have high or medium importance but lack sufficient knowledge based on the results of the LF PIRT. Table 4.6-1 shows the phenomena classified by priority. The uncertainties of those phenomena ranked in higher priority can be reduced by investigation by theoretical evaluation such as using detailed analysis code. Moreover, the effects associated with the local fault events resulting from the accidents such as a long-term station blackout (SBO) shall be evaluated for the future plan.

Table 4.6-1. Identified Important Phenomena of LF

Priority 1	None currently identified
Priority 2 and Priority 3	• Intra-assembly flow distribution • FP and fission gas transport from sodium in primary system to cover gas • Outside-fuel motion of the eutectic material • Porosity of blocking substance • Size of blocking substance • Blockage form • Blockage materials • Location of blockage • Heat generation rate of blockage • Blockage strength • Location of gas impingement • Heat transfer coefficient at fission gas release (gas blanketing)
Priority 4 and Priority 5	34 phenomena currently are identified.
N/A	4 phenomena currently are identified.

4.7 REFERENCES

[4-1] NUREG-1368, "Pre-application Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," U.S. NRC, 1994.

[4-2] NUREG-0968, "Safety Evaluation Report related to the Construction of the Clinch River Breeder Reactor Plant," U.S. NRC, 1983.

[4-3] "Installation Permit Application Document for the Atomic Reactor of the Prototype Fast Breeder Reactor Monju," JAEA, 2007 (in Japanese).

[4-4] Hans K. Fauske, "Some Aspects of Fuel-Pin-Failure Propagation in Sodium-Cooled Fast Reactors," ANL, Nucl. Science and Engineering, 54, 10-17, 1974.

[4-5] American National Standards Institute/American Nuclear Society Standard – ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Power Plants."

[4-6] American National Standards Institute/American Nuclear Society Standard – ANSI/ANS-52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Power Plants."

[4-7] A.W. Thoriey, "Corrosion Behavior of Steel and Nickel Alloys in High Temperature Sodium," *Alkali Met. Coolants, Proc. Symp.*, pp. 97-118, 1967.

[4-8] ADAMS: ML0814407675, "4S Design Description," Toshiba Corp., 2008.

[4-9] A. Otsubo, et. al., "The Occurrence of Wear Marks on Fast Reactor Fuel Pin Cladding," *Journal of Nuclear Science and Technology*, Vol. 36, No.6, p. 522.

[4-10] ADAMS: ML092170507, "4S Safety Analysis," July 2009.

[4-11] K. Haga, et al., "The Effect of Bowing Distortions of Heat Transfer in Seven Pin Bundle," ASME Winter Annual Meeting, New York, 74-WA/HT-50, November 1974.

[4-12] "Evaluation of the methodology of accident analysis for fast reactor – Local fault in subassembly," Japan Nuclear Energy Safety Organization (JNES), August 2008 (in Japanese).

[4-13] A. M.Judd, "Leakage of Pump Bearing Oil into the PFR Primary Sodium," IWGFR/89, O-Arai, Japan, June 1994.

[4-14] CRBRP-1 Appendix III, p.III-82, "III.9.1.8 Initiator 1.11."

[4-15] Sodium Technology Handbook, JNC TN9410 2005-011, 2005 (in Japanese).

[4-16] ADAMS: ML101400662, "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," Toshiba Corp., 2010.

[4-17] M. K. Booker, et al., "Comparison of the Mechanical Strength Properties of Several high-Chromium Ferritic Steels," Proc. of an ASM International Conference on Production, Fabrication, Properties and Application of Ferritic Steels for High-Temperature Applications, 1981.

[4-18] ADAMS: ML092170507, "4S Safety Analysis," Toshiba Corp., 2009.

[4-19] 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."

[4-20] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

[4-21] 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.

[4-22] 10 CFR 1.11, "The Commission."

[4-23] 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

[4-24] 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.

[4-25] S. K. Cheng and N. E. Todreas, "Hydrodynamic Models and Correlations for Bare and Wire-wrapped Hexagonal Rod Bundles – Bundle Friction Factors, Subchannel Friction Factors and Mixing Parameters," *Nucl. Engrg. Des.*, 92, 1986.

[4-26] K. Rehme, "Pressure Drop Correlations for Fuel Element Spacers," Nuclear Technology, 17, 1973.

[4-27] E. U. Khan, T. L. George, and D. R. Rector, "COBRA-WC pretest predictions and post-test analysis of the FOTA temperature distribution during FFTF natural circulation transients," *Trans. Am. Nucl. Soc.* (USA), 43, 1982.

[4-28] A. E. Walter and A. B. Reynolds, "Fast Breeder Reactors," Elsevier, 1981.

[4-29] T. Koga, "Development of an advanced fuel subassembly for non-refueling reactor core: Water test using full scale fuel subassembly model with both high volume-fraction and low pressure loss," 2004 Annual Meeting of the Atomic Energy Society of Japan, September 2004.

[4-30] T. Koga, "Development of an advanced fuel subassembly for non-refueling core," ANS 2005 Winter Meeting, 2005.

[4-31] A. Khakim, "Tight Lattice Fuel Pin Bundle Thermal hydraulics Analysis of Sodium Cooled Fast Reactors," 2011.

[4-32] H. Ninokata, "Summary of LMFBR subassembly blockage studies at PNC," PNC, PNC TN9420 90-004, 1990.

[4-33] S. Nishimura and N. Ueda, "Analytical Evaluation of Local Fault in Sodium Cooled Small Fast Reactor (4S) – Preliminary Evaluation of Partial Blockage in Coolant Channel –," The Seventh JSME-KSME Thermal and Fluids Engineering Conference, Oct. 2008.

[4-34] K. Haga, et al., "Temperature rise due to fission gas release in locally blocked LMFBR fuel subassembly simulators," PRN, *Proc. of the LMFBR Safety Topical Meeting*, 1982.

[4-35] I. E. Idelchik, "Handbook of Hydraulic Resistance," Jaico Publishing House, 2005.

[4-36] N. Ueda, et al., "Development of the 4S and related technologies (7): An application of physics benchmark experiment results to safety analyses of small fast reactors – An analysis of delayed neutron fraction benchmark results using nuclear design methodology –," Proceedings of ICAPP 09, 2009.

[4-37] Y. Tsuboi, et al., "Development of the 4S and related technologies (8): An application of physics benchmark experiment results to safety analyses of small fast reactors – An analysis of delayed neutron fraction benchmark results using nuclear design methodology –," Proceedings of ICAPP 09, 2009.

[4-38] Y. Hermann, et al., "Technical support for a proposed decay heat guide using SAS2H/ORIGEN-S data," NUREG/CR-5625 / ORNL-6698, 1994.

[4-39] H. Tsai, et al., "Potential eutectic failure mechanism for stainless steel cans containing plutonium metal," *Proc. of Int. Symp. on PATRAM*, 2007.

[4-40] ADAMS: ML082050556: A. M. Yacout, "Long Life Metallic Fuel for the Super Safe, Small and Simple (4S) Reactor," 2008.

[4-41] ADAMS: ML080510370, "4S reactor Second Meeting with NRC Pre-application review," February 2008.

[4-42] ADAMS: ML081400095, "4S reactor Third pre-application review meeting with NRC," May 2008.

[4-43] E. L. Gluekler and L. Baker Jr., "Post Accident Heat Removal in LMFBR's," *Proc. of Winter Annual Meeting of American Society of Mechanical Engineering*, 1977.

[4-44] C. G. Allan, et al., "Solubility and Deposition Behavior of Sodium Bromide and Sodium Iodine in Sodium/Stainless Steel Systems," TRG Report 2458(D), 1973.

[4-45] B. D. Pollock, et al., "Vaporization of Fission Products from Sodium," ANL-7520 Part 1, Argonne National Laboratory, 1969.

[4-46] J. K. Fink, "Tables of thermodynamic properties of sodium," ANL-CEN-RSD-82-4, 1982.

[4-47] Y. Iwamoto, et al., "Study on flow-induced-vibration evaluation of large-diameter pipings in a sodium-cooled fast reactor, 4; Experiments on the 1/10-scale hot leg test facility in reynolds number of 50000 and 320000," *Proceedings of 6th Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS-6)*, 2008.

[4-48] B. Damiano, et al., "Summary of ORNL long-term surveillance at the FFTF," ORNL/TM-10767, 1992.

[4-49] M. Nishimura, et al., "Transient experiments on fast reactor core thermal-hydraulics and its numerical analysis: Inter-subassembly heat transfer and inter-wrapper flow under natural circulation conditions," *Nucl. Eng. and Des.*, Vol. 200, 157, 2000.

[4-50] M. Sawada, H. Arikawa, and N. Mizoo, "Experiment and analysis on natural convection characteristics in the experimental fast reactor JOYO," *Nucl. Eng. Des.* (Netherlands), 120, 1990.

[4-51] K. Shiba, "Transient thermohydraulics research activities," *Journal of Nucl. Science and Tech.*, 26, 1989.

[4-52] D. Tenchine and D. Grand, "Onset of natural circulation in a sodium loop," Fourth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, NURETH-4, 1, 1989.

[4-53] V. S. Bhise and C. F. Bonilla, "The Experimental Pressure and Critical Point of Sodium," *Proc. International Conf. Liquid Metal Technology in Energy Production*, Seven Springs, 1977.

[4-54] S. Das Gupta, "Experimental High Temperature Coefficients of Compressibility and Expansivity of Liquid Sodium and Other Related Properties," Dr. Engineering Science Dissertation with C. F. Bonilla, Dept. of Chemical Engineering and Applied Chemistry, Columbia University, Xerox-University Microfilms, 1977.

[4-55] W. A. Ragland, et al., "EBR-II axial temperature distributions measured during in-vessel natural circulation experiment," ANL-RA-PP--69287, 1994.

[4-56] D. Mohr, et al., "Natural-Convection Behavior of EBR-II:A Comparison of CONVECT Analysis with Test Results," *Trans. Am. Nucl. Soc.*, 1975.

[4-57] R. M. Singer, et al., "Steady State Natural Circulation Performance of the Experimental Breeder Reactor II Primary Heat Transfer Circuit," *Nuclear Science and Engineering*, 1977.

[4-58] W. L. Baumann, et al., "EBR-II In-Vessel Natural-Circulation Analysis," NUREG/CR-2821, ANL-82-66, 1982.

[4-59] H. Ninokata and A. Izumi, "Decay heat removal system of the MONJU reactor plant and studies related to the passive actuation and performances," *Proceedings of the 1990 International Fast Reactor Safety Meeting 2*, 1990.

[4-60] T. Ishizu, et al., "An evaluation of Passive Safety Failures for the Japanese Prototype LMFBR Monju," *Proceedings of NURETH-11*, 2005.

[4-61] M. Tanabe, et al., "Monju function test," PNC Technical Report, 93, 18-31, 1995 (in Japanese).

[4-62] B. Valentin, et al., "Natural convection tests in Phenix coltemp experiments," *Proceedings of Int. Fast Reactor Safety Mtg.*, Snowbird, 1990.

[4-63] B. Braquilanges, "Natural circulation test in SPX1," *Proceedings of Int. Fast Reactor Safety Mtg.*, Snowbird, 1990.

[4-64] F. Namekawa, et al., "Experimental study for in-vessel thermal hydraulics in a loop-type LMFBR during natural circulation decay heat removal," *Trans. Am. Nucl. Soc.* (USA) 44, 1983.

[4-65] H. Kamide, et al., "An experimental study of inter-subassembly heat transfer during natural circulation decay heat removal in fast breeder reactors," *Nuclear Engineering and Design*, 183, 1998.

[4-66] W. C. Horak, et al., "Long-term post-test simulation of the FFTF natural circulation tests using SSC," BNL-NUREG-34245, 1984.

[4-67] R. E. Peterson, et al., "Safety related experience in FFTF startup and operation," *Proceedings of the LMFBR. Safety Topical Meeting* (France), 2, 1982.

[4-68] T. R. Beaver, et al., "FFTF natural circulation tests," *Trans. Am. Nucl. Soc.* (USA), 39, 1981.

[4-69] H. P. Planchon, "The experimental breeder reactor II inherent shutdown and heat removal tests – Test results and analysis," Proceedings of the International Topical Meeting on Fast Reactor Safety, 1987.

[4-70] H. Hoffmann, et al., "Thermohydraulic investigations of decay heat removal systems by natural convection for liquid-metal fast breeder reactors," *Nucl. Tech.*, 88, 1989.

[4-71] J. L. Gillette, et al., "Experimental study of the transition from to natural circulation in EBR-II at low power and flow," *Proc. of 18th National Heat Transfer Conference*, 1979.

[4-72] H. Hoffmann, et al., "Thermohydraulic model experiments and calculations on the transition from forced to natural circulation for pool-type fast reactors," *Trans. Am. Nucl. Soc.* (USA), 62, 1990.

[4-73] T. H. Bauer, et al., "Behavior of Modern Metallic Fuel in TREAT Transient Overpower Test," *Nuclear Technology*, Vol. 92, p.325, 1990.

[4-74] A. E. Wright, et al., "Recent Metal Fuel Safety Tests in TREAT," Proc. of Science and Technology of Fast Reactor Safety, Guernsey, Vol.1, p.59, 1986.

[4-75] E. A. Rhodes, et al., "Fuel Motion in Overpower Test of Metallic Fast Reactor Fuel," *Nuclear Technology*, Vol. 98, p. 91, 1992.

[4-76] W. R. Robinson, et al., "Integral Fast Reactor Tests M2 and M3 in TREAT," *Trans. Am. Nucl. Soc.*, 50, 352, 1985.

[4-77] A. M. Tentner, et al., "Analysis of Metal Fuel Transient Overpower Experiments with the SAS4A Accident Analysis Code," Proc. of the 1990 International Fast Reactor Safety Meeting, Snowbird, Vol. I, 1990.

[4-78] A. M. Tentner, et al., "PINCALE; A Mechanistic Model for the Analysis of In-pin Fuel Relocation under LOF and TOP conditions for SAS4A," *Trans. Am. Nucl. Soc.*, 49:275, Boston, MA , 1985.

[4-79] J. E. Cahalan, et al., "Advanced LMR Safety Analysis Capabilities in the SASSYS-1 and SAS4A Computer Codes," *Proc. ARS'94 Int'l Topical Mtg. on Advanced Reactors Safety*, Pittsburgh, Pennsylvania, 1994.

[4-80] D. H. Thompson, et al., "Summary and conclusions – SLSF local fault safety experiment P4," ANL, *Proc. Int. Top. Mtg. on Fast Reactor Safety / CONF-850410-14*, 1985.

[4-81] Y. Onda, et al., "Three-Pin Cluster CABRI Tests Simulating the Unprotected Loss-of-Flow Accident in Sodium-Cooled Fast Reactors," *Journal of Nucl. Science and Technology*, Vol. 48, No.2, p.188, 2011.

[4-82] G. Kayser, et al., "Summary of the SCARABEE-N Subassembly Melting and Propagation Tests with an Application to a Hypothetical Total Instantaneous Blockage in a Reactor," NSE Vol.128 144-185, 1998.

[4-83] K. Konishi, et al., "The result of a wall failure in-pile experiment under EAGLE project," *Nucl. Engineering and Design*, Vol. 237, Issue 22, p. 2165, 2007.

[4-84] Bando, et al., "Research on Local Blockage in Fuel by Foreign Material Test," Fall Meeting of the Atomic Energy Society of Japan, Published Abstracts, F45 (in Japanese).

[4-85] M. H. Fontana, et al., "Effect of partial blockages in simulated LMFBR fuel assemblies," CONF-740401-13, ORNL, 1974.

[4-86] M. Uotani, et al., "Local flow blockage experiments in 37-pin sodium cooled bundles with grid spacers," PNC, PNC TN941 78-141, 1978.

[4-87] F. Huber and W. Peppler, "Boiling and Dryout behind Local Blockages in Sodium Cooled Rod Bundles," KfK, NED 82 341, 1984.

[4-88] M. Uotani and K. Haga, "Experimental investigation of sodium boiling in partially blocked fuel subassemblies," PNC, NED 82 319, 1984.

[4-89] Gaston Kayser, et al., "Summary of the SCARABEE-N Subassembly Melting and Propagation Tests with an Application to a Hypothetical Total Instantaneous Blockage in a Reactor," *Nuclear Science and Engineering*, Vol. 128, 1997.

[4-90] H. Nakamura, et al., "Hydraulic Simulation of Local Blockage in a LMFBR Fuel Subassembly," OEC, *Nuclear Engineering and Design*, 1980.

[4-91] K. Yamaguchi, et al., "Boiling and Dryout Conditions in Disturbed Cluster Geometry and Their Application to the Liquid-Metal Fast Breeder Reactor Local Fault Assessment," PNC (Power Reactor and Nuclear Fuel Development Corporation) KfK, *Nuclear Science and Engineering*, 1984.

[4-92] J. Olive and P. Jolas, "Internal blockage in a fissile Super-Phenix type subassembly: the Scarlet experiments and their interpretation by the Cafca-NA3 code," CEA, EDF, *Nucl. Energy*, 29, No. 4, 1990.

[4-93] T. Honda and H. Oshima, "Analysis of local blockage of peripheral coolant pass in fuel subassembly (Japanese)," PNC/OEC, JNC TN9400 2001-019, 2000.

[4-94] A. Alfred, et al., "Partial Flow Blockage Effects within a (Liquid Metal Cooled Fast Reactor) LMFBR Fuel Assembly," University of Pittsburgh, 1975.

[4-95] T. Honda and H. Oshima, "Analysis of local blockage of peripheral coolant pass in fuel subassembly (Japanese)," JNC, PNC TN9400 2001-019, 2000.

[4-96] T. Cadiou and J. Louvet, "Evaluation of the Accident Scenario Initiated by a Total Instantaneous Blockage in a PHENIX Subassembly," CEA, *Nuclear Technology*, Vol. 153, 2006.

[4-97] K. Haga, "Current status and development of research on local failure (Japanese)," PNC, PNC TN2410 87-002, 1987.

[4-98] J. T. Han and M. H. Fontana, "Blockages in LMFBR fuel assemblies – A review," ORNL, CONF-771120-14, 1977.

[4-99] A. Khakim, "Tight Lattice Fuel Pin Bundle Thermal hydraulics Analysis of Sodium Cooled Fast Reactors," 2011.

[4-100] E. Baglietto, "Anisotropic Turbulence Modeling for Accurate Rod Bundle Simulations," Proceedings of ICONE14, 2006.

[4-101] R. G. Pahl, et al., "Steady-state Irradiation Testing of U-Pu-Zr Fuel to >18 at% Burnup," *Proc. Int. Fast Reactor Safety Meeting*, Snowbird, 1990, Vol. 4, American Nuclear Society, 1990.

[4-102] R. G. Pahl, et al., "The Characterization and Monitoring of Metallic Fuel Breaches in EBR-II," International Conference on Fast Reactor Systems and Fuel Cycles, Kyoto (Japan), 1991.

[4-103] K. Haga, et al., "Temperature Rise due to Fission Gas Release in Locally Blocked LMFBR Fuel Subassembly Simulators," PNC, 1985.

[4-104] J. B. Van Erp, et al., "Potential fuel failure propagation due to fission-gas release in LMFBR subassemblies," ANL, ASME Paper no. 72-WA/NE-14, 1972.

[4-105] J. D. Gabor, et al., "Breakup and Quench of Molten Metal Fuel in Sodium," International Topical Meeting on Safety of Next Generation Power Reactors, Seattle, Washington, 1988.