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Invited Article

A Preliminary Safety Analysis for the Prototype Gen IV Sodium-Cooled Fast Reactor

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ABSTRACT

Korea Atomic Energy Research Institute has been developing a pool-type sodium-cooled fast reactor of the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR). To assess the effectiveness of the inherent safety features of the PGSFR, the system transients during design basis accidents and design extended conditions are analyzed with MARS-LMR and the subchannel blockage events are analyzed with MATRA-LMR-FB. In addition, the in-vessel source term is calculated based on the super-safe, small, and simple reactor methodology. The results show that the PGSFR meets safety acceptance criteria with a sufficient margin during the events and keeps accidents from deteriorating into more severe accidents.

1. Introduction

Sodium-cooled fast reactor (SFR) design technologies have been developed in Korea since 1997 under a National Nuclear R&D Program to achieve an enhanced safety, an efficient utilization of uranium resources, and a reduction of a high-level waste volume. In 2015, the preliminary specific design of the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) was completed, which is a pool-type SFR with the thermal power of 392.2 MWt and uses metallic fuel of U–10%Zr for a core having inherent reactivity feedback mechanisms and high thermal conductivity.

Fig. 1 shows the overall configuration of the PGSFR, which consists of the primary heat transport system (PHTS), the intermediate heat transport system (IHTS), the steam generators (SGs) including balance of plant, and the decay heat removal system (DHRS).

The PHTS is placed in a large pool to make the system transients slower, thus giving a higher probability to terminate the abnormal events before they propagate into

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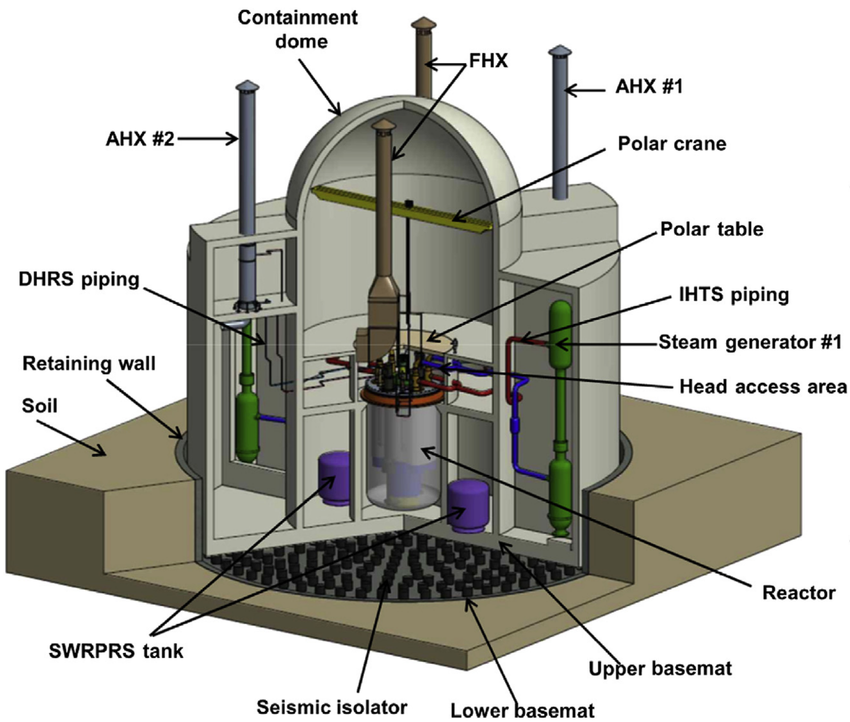

Fig. 1 – Overall configuration of the Prototype Gen-IV Sodium-cooled Fast Reactor. AHX, natural-draft sodium-to-air heat exchanger; DHRS, decay heat removal system; FHX, forced-draft sodium-to-air heat exchanger; IHTS, intermediate heat transport system, SWRPRS, sodium water reaction pressure relief system.

accidents. The IHTS loop is thermally coupled to the PHTS and the SGs. The IHTS transfers the reactor-generated heat from the intermediate heat exchanger (IHX) of the PHTS to the SG. The IHTS consists of two loops, and each loop has two IHXs, one electromagnetic (EM) pump, one expansion tank, and one SG. The SGs consist of two independent steam-generation loops and convert the subcooled water to a superheated steam. The DHRS with the heat transfer capability of 10 MWt is composed of two units of passive decay heat removal system (PDHRS) and two units of active decay heat removal system (ADHRS). In addition, a damper driven by the emergency diesel generator is attached to the natural-draft sodium-to-air heat exchanger (AHX) and the forced-draft sodium-to-air heat exchanger (FHX). The damper is designed with the concept of the passive fail-open type. The ADHRS has been designed to operate at half capacity by the natural circulation, even if the EM pump of ADHRS stops [1].

The fundamental approach to design a nuclear reactor with safety is defense-in-depth. The multiple, independent, and redundant means of the design assure the performance of safety functions in normal operation and in accident conditions. The cladding and end seals of fuel pin are the first barrier to protect the escape of radiological material to the environment. Table 1 shows the safety acceptance criteria of the fuel and cladding for each event category. An acceptance criterion for anticipated operational occurrences (AOOs) and design basis accident (DBA) Class 1 is established on the basis of cumulative damage function (CDF). CDF is introduced as a measure to protect against rupture due to thermal creep. A combination of temperature and duration limits is accepted as a design guideline in the sense that it is derived from the CDF equation, which is a function of time, temperature, and stress. CDF in MARS-LMR [2] can be defined by Eqs. (1–3).

$$\mathrm{CDF} = \int_{t=0}^{t=t} \frac{1}{t_r} dt \tag{1}$$

$$t_r = \theta \exp \frac{Q}{R} \cdot \frac{1}{T} \tag{2}$$

$$\ln\theta = -34.8 + tanh\ \frac{\sigma-200}{50} + \frac{12}{1.5 + 0.5\ \ tanh\ \frac{\sigma-200}{50}}\ \mathrm{In}\left[\mathrm{In}\frac{730}{\sigma}\right] - 0.5\left[1 + tanh\ \frac{\sigma-200}{50}\right] \cdot \left\{0.75\left[1 + tanh\ \left(\frac{\dot{T}-58}{50}\right)\right]\right\} \tag{3}$$

Table 1 – Safety acceptance criteria for event category.

Event category	AOO	DBA Class 1	DBA Class 2	DEC
Fuel/cladding	$CDF^*_{\sum AOO}$ < 0.05 Strain <1%	CDF_{event} < 0.05 Strain <1%	Fuel T <Solidus T Clad T <1,075°C Coolant T <Boiling T	Coolant T <Boiling T

AOO, anticipated operational occurrence; CDF, cumulative damage function; DBA, design basis accident; DEC, design extended condition.

where t_r is a rupture time (second), σ is a hoop stress (MPa), T is transient temperature (K), $\dot{T}$ is heating rate (K/s), activation energy Q is 70,170 (cal/mol), and gas constant R is 1.986 (cal/mol/K).

The acceptance criteria for DBA Class 2 and design extended condition (DEC) are established based on temperatures of pin melting and coolant boiling. In DEC events, massive fuel melting is allowed as long as the molten core is retained in vessel with a coolable geometry. In such a scenario, the coolant temperature is an important factor instead of the fuel or cladding temperatures, and it should be maintained below the sodium boiling temperature.

Based on the safety acceptance criteria as described in Table 1, system transients are carried out to assess the inherent safety features of the PGSFR. DBAs are analyzed with a conservative deterministic evaluation method (a best-estimate code and conservative inputs). DECs are analyzed with a best-estimate deterministic evaluation method (a best-estimate code and best-estimate inputs) supported by sensitivity analysis.

2. Modeling methodology

Fig. 2 shows the MARS-LMR nodalization for the preliminary specific design of the PGSFR. The core is modeled by four parallel flow channels such as hottest subassembly, fuel assemblies, nonfuel assemblies, and leakage flow. Active fuel regions are axially divided into eight nodes. The PHTS is placed in a large pool, which is divided into two temperature zones. The four sodium-to-sodium decay heat exchangers (DHXs) and two pumps are located in the cold pool, whereas four IHXs are located in the hot pool to transfer the reactor-generated heat from the PHTS to the SG. The IHTS consists of the two IHXs tube side, piping, one EM pump, and one SG shell side. The SG tubes are divided into a total of 30 nodes. The SG inlet feed-water boundary region is adopted with a constant mass flow-rate condition. In addition, the SG outlet boundary region nearby high-pressure turbine is adopted with a constant pressure condition. Each DHRS is modeled by PDHRS and ADHRS, respectively. DHX is located and sub submerged in the cold pool region and the sodium-to-air heat exchanger is located in the upper region of the reactor building. The air boundary regions are imposed at the entrance and the exit of this part.

The reactor shutdown system requires a mandatory protection system to prevent the deterioration of the plant from all possible accidents. Table 2 lists the trip parameters and the set points with uncertainties in the reactor protection system.

3. Design basis events

To evaluate the capabilities of the components having safety-related functions under the accident situations, the analyses of design basis events are performed for seven representative events, which are a loss of flow (LOF), one-pump seizure (OPS), a loss of heat sink by sodium–water reaction (SWR), transient overpower (TOP), the station black-out (SBO), PHTS pipe break, and reactor vessel leak.

Conservative assumptions are applied to the analysis of the plant responses during the postulated DBAs, which are 102% of power condition with the ANS-79 decay power model [3], considering the 5-second delay in opening of AHX and FHX dampers and loss of offsite power (LOOP). In addition, one PDHRS and one ADHRS are available in accordance with a single failure criterion and maintenance.

3.1. *Loss of flow (bounding event of anticipated operational occurrence)*

LOF represents the loss of core cooling capability due to a pumping failure of both PHTS pumps, which is a bounding event of AOO. The imbalance between the reactor power and the primary flow rate is a main safety concern in the LOF event. To prevent the occurrence of the severe imbalance between power and flow, the PGSFR is designed as the reactor tripped by a high-power/flow parameter. Figs. 3 and 4 show the peak clad midwall temperature and CDF behavior during the LOF accident, respectively. The peak clad midwall temperature rapidly increases after the pump trip at 0.0 seconds, and then decreases nearly vertically after the reactor shutdown by trip signal of the power-to-flow rate ratio. The temperature rises due to both decreased mass flow rate by the PHTS pump coastdown and the diminution of the heat transfer to the IHTS by the isolation of the feed water due to LOOP. If the DHX heat removal exceeds the core decay heat power, the clad temperature can decrease continuously. In conclusion, the CDF during LOF is calculated as 2.06×10^{-5} by Eq. (1), which is much lower than 0.05 of an acceptance criterion for AOO.

3.2. *One-pump seizure (bounding event of DBA Class 1)*

OPS accident is a bounding event of DBA Class 1, which occurs due to one PHTS pump seizure caused by a failure of mechanical bearing or electric motor. Fig. 3 shows the peak clad midwall temperature. The temperature rapidly increases because of seizure of one PHTS pump and coastdown of the other PHTS pump due to LOOP. The peak clad temperature drops rapidly after the reactor trip by a trip signal of the power-to-flow rate ratio. Fig. 5 presents the comparison results of the core decay heat and heat removed by DHRS. The DHX heat removal exceeds the core decay heat power at about 5,420 seconds and the core outlet temperature decreases continuously. In conclusion, the CDF value is 1.13×10^{-3}, which is lower than 0.05 of a safety acceptance criterion for the DBA Class 1 as shown in Fig. 4.

3.3. *Transient overpower (bounding event of DBA Class 2)*

TOP accident is assumed to occur due to a single rod withdrawal with a control rod stop system failure, which is a bounding event of DBA Class 2. The event is initiated at 0.0 seconds, and a positive reactivity of 68.7¢ is inserted for 15 seconds. The core power rapidly increases due to the positive reactivity insertion. The reactor trips at 2.22 seconds by a high power-to-flow rate ratio parameter, and then the power drastically decreases by the reactor trip.

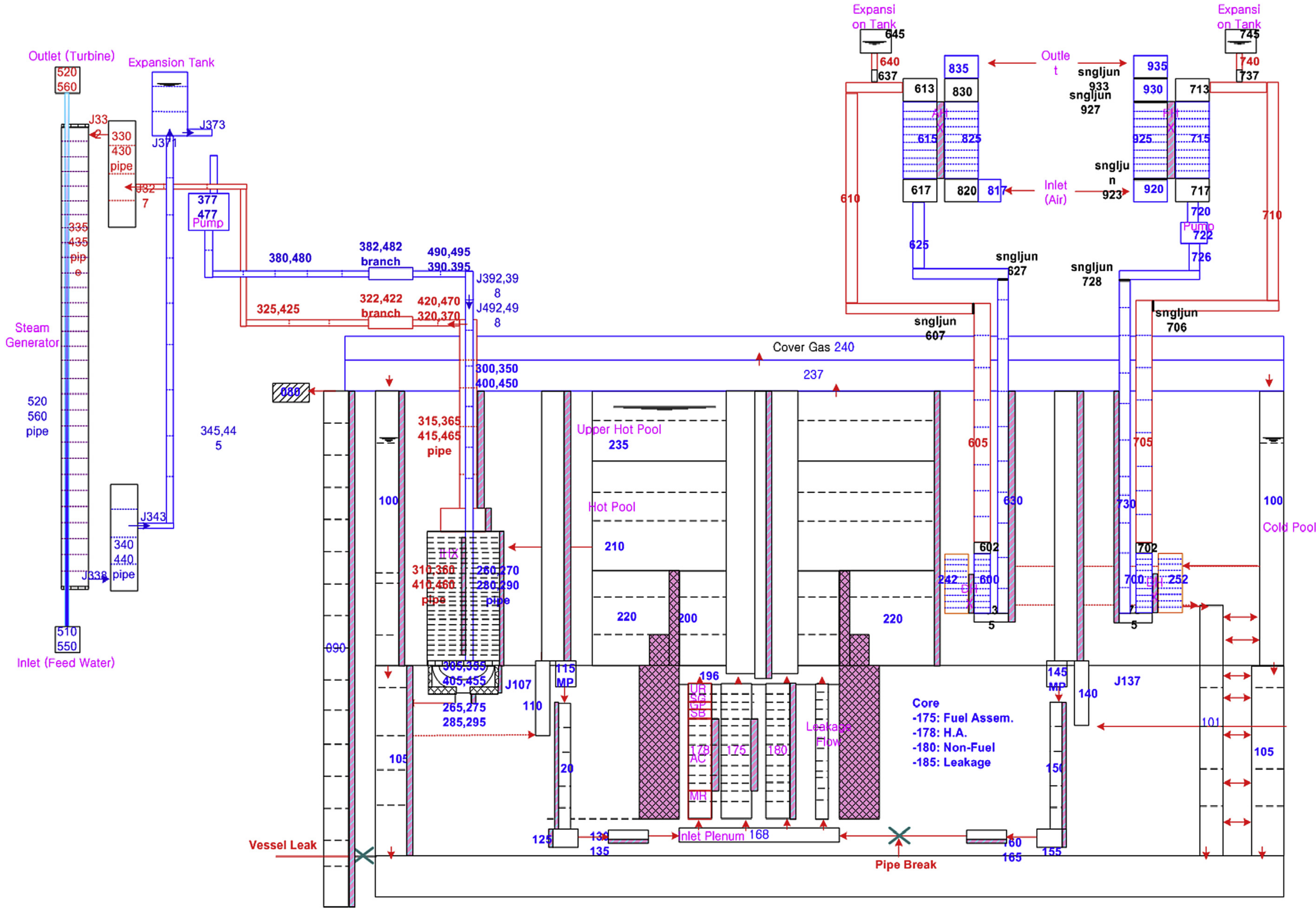

Fig. 2 — MARS-LMR nodalization for Prototype Gen-IV Sodium-cooled Fast Reactor.

Table 2 – Trip parameters and set points.

Parameter	Set point (uncertainty)
High core outlet temperature	565°C (±6°C)
High core inlet temperature	410°C (±6°C)
High power-to-primary heat transport system flow ratio	119% (±2.4%)
Steam generator shell outlet temperature	359°C (±6°C)
Low hot-pool level	20 cm below 100% operating level (±10 cm)

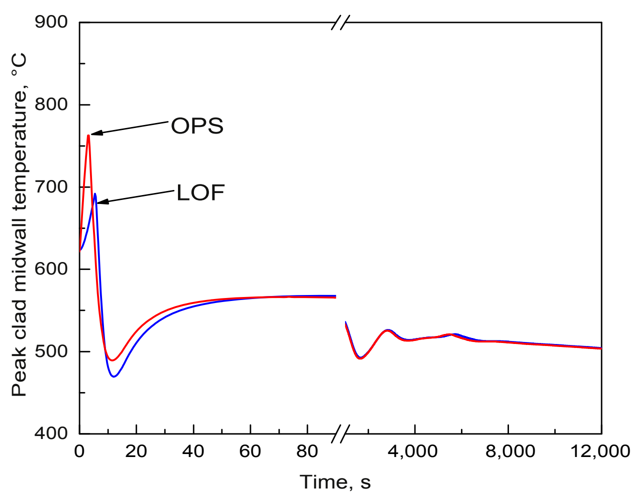

Fig. 3 – Peak clad midwall temperatures at loss of flow and one-pump seizure. LOF, loss of flow; OPS, one-pump seizure.

Figs. 6 and 7 show the peak clad temperature and peak fuel temperature during the TOP, pipe break, SBO, and vessel leak accidents. In case of TOP, the peak temperatures of fuel, clad, and coolant are calculated as 795°C, 741°C, and 724°C, which meet the safety acceptance criteria of DBA Class 2. The

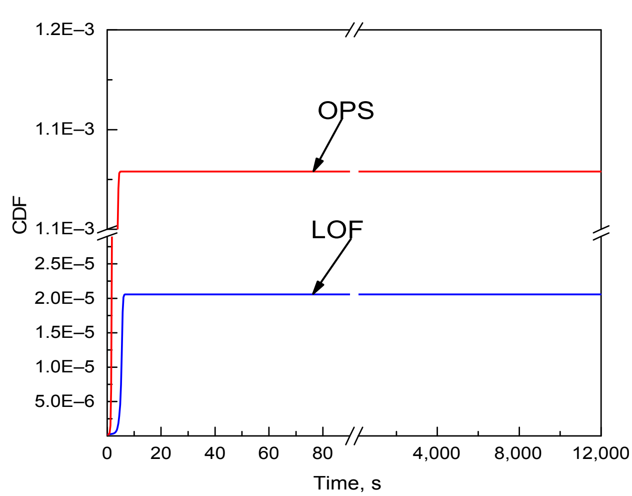

Fig. 4 – Cumulative damage function (CDF) at loss of flow and one-pump seizure. LOF, loss of flow; OPS, one-pump seizure.

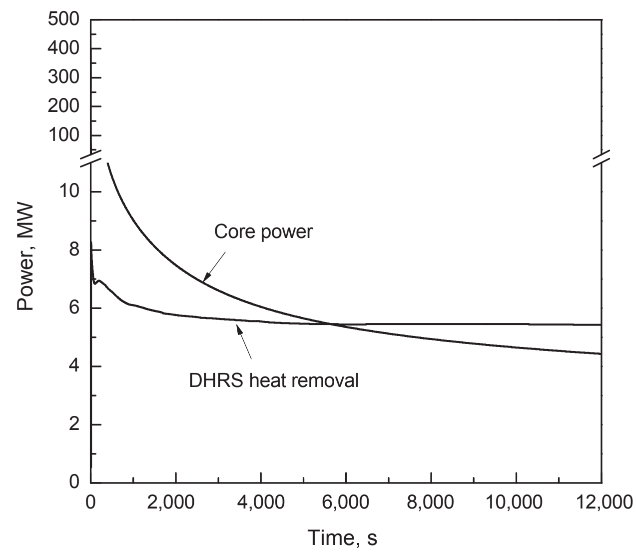

Fig. 5 – Heat removal of decay heat removal system (DHRS) at one-pump seizure.

coolant temperature has a sufficient safety margin against the boiling temperature as shown in Fig. 8.

3.4. *Station black-out*

SBO is initiated by a simultaneous loss of both offsite power sources and on-site power sources including emergency diesel generator. The on-site emergency power supplies for operation of EM pumps, blowers, and dampers of the DHRS. ADHRS has at least 50% of heat removal capacity against a complete loss of power. For this reason, the total heat removal capacity of DHRS is about 3.75 MWt during the SBO accident. Fig. 9 shows a comparison between decay heat removal rate of the DHRS and a reactor power. Because total heat removal

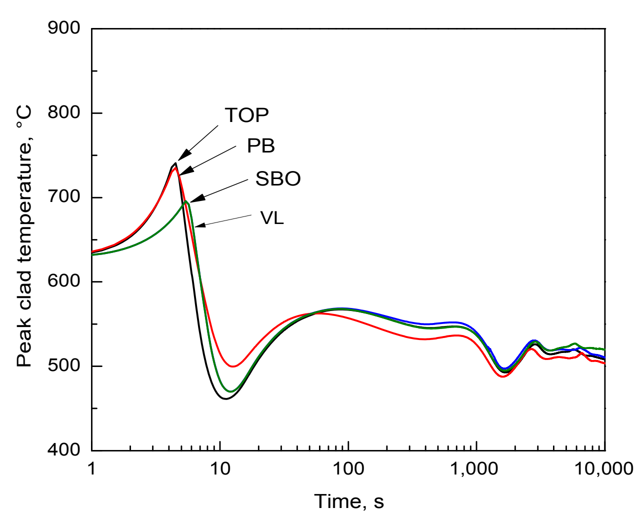

Fig. 6 – Peak clad temperatures at design basis accident-II. PB, pipe break; SBO, station black-out; TOP, transient overpower; VL, vessel leak.

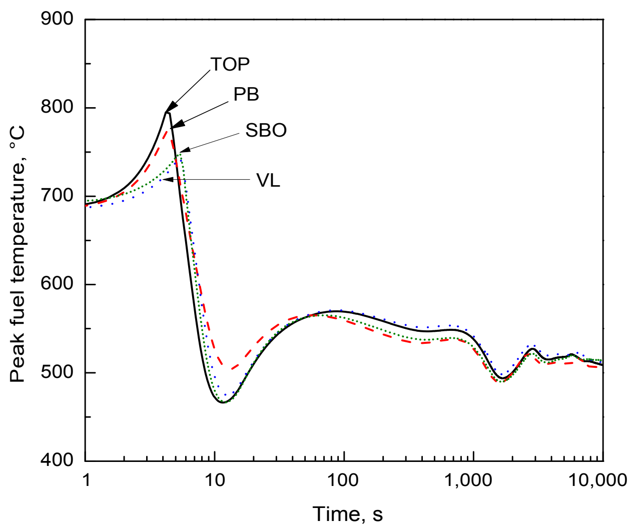

Fig. 7 – Peak fuel temperatures at design basis accident-II. PB, pipe break; SBO, station black-out; TOP, transient overpower; VL, vessel leak.

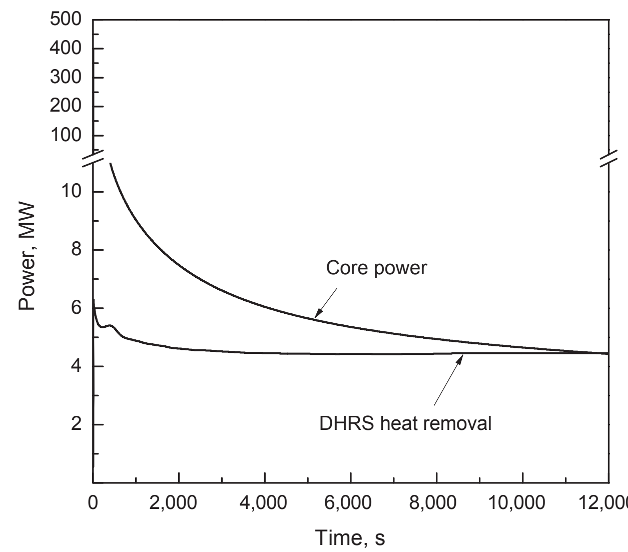

Fig. 9 – Heat removal of decay heat removal system (DHRS) at station black-out.

capacity of DHRS in SBO accident is smaller than that of the others, it needs more time as the DHRS heat removal capacity exceeds the core decay heat power at 11,740 seconds, and the core outlet temperature is gradually decreased.

3.5. *Pipe break*

The PHTS pipe break event is similar to the LOF accident. This event indicates one pipe break connected between an inlet plenum and PHTS pump. The flow through the broken pipe is discharged into the cold pool, and some of the sodium of an intact pipe is also released into the cold pool. The event is initiated at 0.0 seconds, and the PHTS and IHTS pumps stop according to the assumption of LOOP at the same time. Therefore, the residual heat removal is achieved only by SGs and DHRS. The peak clad temperature and fuel temperature are shown in Figs. 6 and 7, which satisfy the safety criteria of DBA Class 2. Fig. 10 shows a comparison between the decay heat removal rate of the DHRS and a reactor power. After 5,561 seconds, the amount of heat removed by the DHX is higher than core residual heat production, and the core outlet temperature decreases continuously.

3.6. *Reactor vessel leak*

The reactor vessel leak is a typical accident of a sodium leak at the PHTS boundary, which is assumed to occur at the bottom of the reactor vessel conservatively. The size of leakage area is

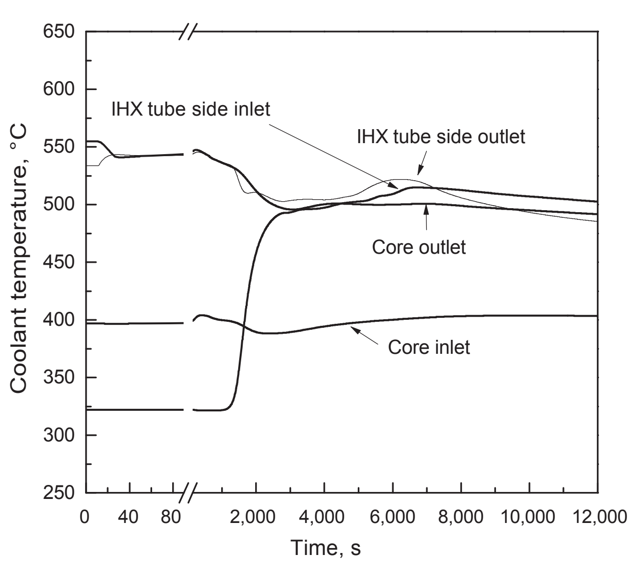

Fig. 8 – Coolant temperatures at transient overpower. IHX, intermediate heat exchanger.

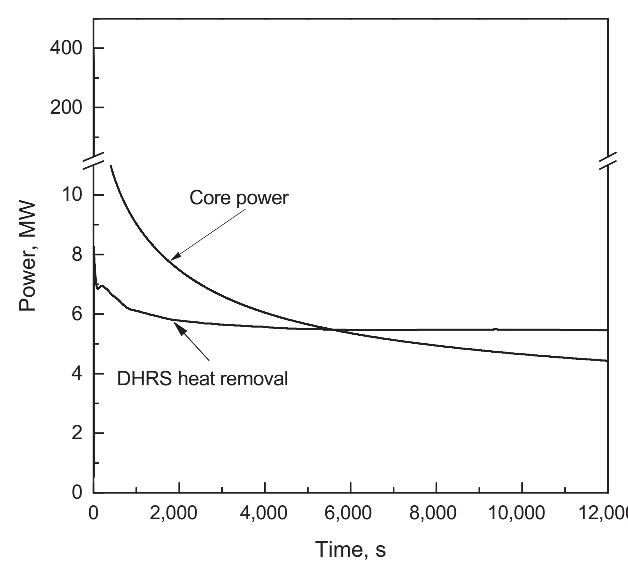

Fig. 10 – Heat removal of decay heat removal system (DHRS) at pipe break.

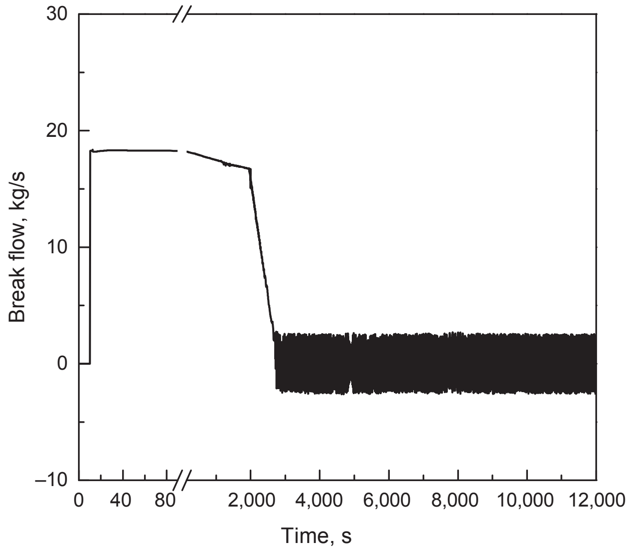

Fig. 11 – Sodium leak rate at vessel leak.

assumed to be 10.0 cm^2 to analyze the system transient. The accident is assumed to occur at 0.0 seconds, and the PHTS and IHTS pumps stop according to the assumption of LOOP at the same time. This accident is detected by a low-level parameter due to the leakage flow through the reactor vessel. The leakage flow rate during the accident is shown in Fig. 11. This event mainly affects the level of sodium in the PHTS. The sodium levels in the hot and cold pools are maintained at levels above the inlets of IHX and DHX during transients as shown in Fig. 12.

3.7. *Sodium–water reaction*

Sodium can rigorously react with water or steam. This chemical reaction generates high-pressure waves and high-

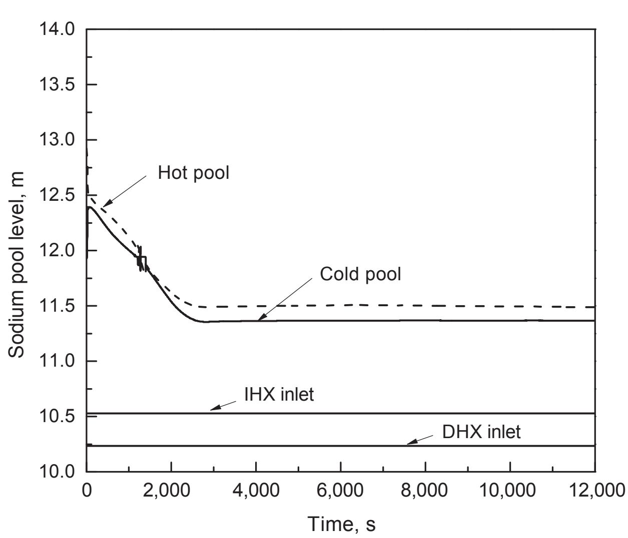

Fig. 12 – Sodium levels at vessel leak. DHX, decay heat exchanger; IHX, intermediate heat exchanger.

temperature reaction heat. In the PGSFR, the SWR event can occur due to the rupture of SG tubes. This event threatens the integrity of the PHTS. Fig. 13 presents the MARS-LMR nodalization to model the failure of heat removal function of one IHTS loop due to the SWR event. The flow and pressure boundary conditions are applied to the cold and hot legs, respectively. To model the failure of the IHTS function, the mass flow rate at the TMDP junctions (C391, C396, C491, and C496) are set to zero. The total discharged time of the sodium of the affected IHTS loop is conservatively assumed to be 5.0 seconds.

Fig. 14 presents the mass flow rate at the two IHTS loops. The mass flow rate of the affected IHTS loop is linearly decreased during the 5.0 seconds. Fig. 15 presents the comparison results of the core decay heat and heat removed by DHRS. After 5,000 seconds, the heat removal exceeds the decay heat. Considering the long-term cooling, the reactor is normally cooled by the DHRS. The peak temperatures of the fuel and cladding reach 744.2°C and 691.6°C at 5.4 seconds, respectively, as shown in Fig. 16. It has a sufficient margin to the safety criteria.

4. Design extended condition

DEC events contain the anticipated transient without scram (ATWS) in which the safety of PHTS is achieved by the inherent reactivity feedback. The key phenomena in the DEC event are the inherent safety characteristics, which should maintain a balance between the reactor power and the decay heat removal rate. In this study, ATWS events are analyzed with a best-estimation approach using the MARS-LMR code. The core is divided into individual flow groups, and nominal values for design parameters are used with the ANS-94 model for decay heat. Moreover, all four DHRS are available in DEC modeling. To evaluate the inherent reactivity feedback mechanisms, five reactivity feedback models, namely, fuel Doppler, sodium density, fuel pin axial expansion, core radial expansion, and control rod driveline and reactor vessel (CRDL/RV) expansion, are taken into account. In addition, the diverse protection system, which should be activated for ATWS events, is neglected to demonstrate inherent safety characteristics.

4.1. *Unprotected transient overpower*

The reactivity insertion event could occur due to control rod withdrawal or steam-line break. In this work, the unprotected TOP event is assumed to occur by a single control rod withdrawal with a reactivity insertion amount of US$0.3 for 15.0 seconds, considering the limit of the maximum insertion amount by the control rod stop system. The change of the core power is determined by reactivity feedback, and the CRDL/RV expansion reactivity feedback has the highest negative reactivity during this event as shown in Fig. 17. Although the peak clad temperature in the hot subassembly rises to 674.6°C near the eutectic temperature, there is no clad penetration. The peak coolant temperature in the hot subassembly has a sufficient safety margin against the boiling temperature as shown in Fig. 18. Therefore, the unprotected TOP event

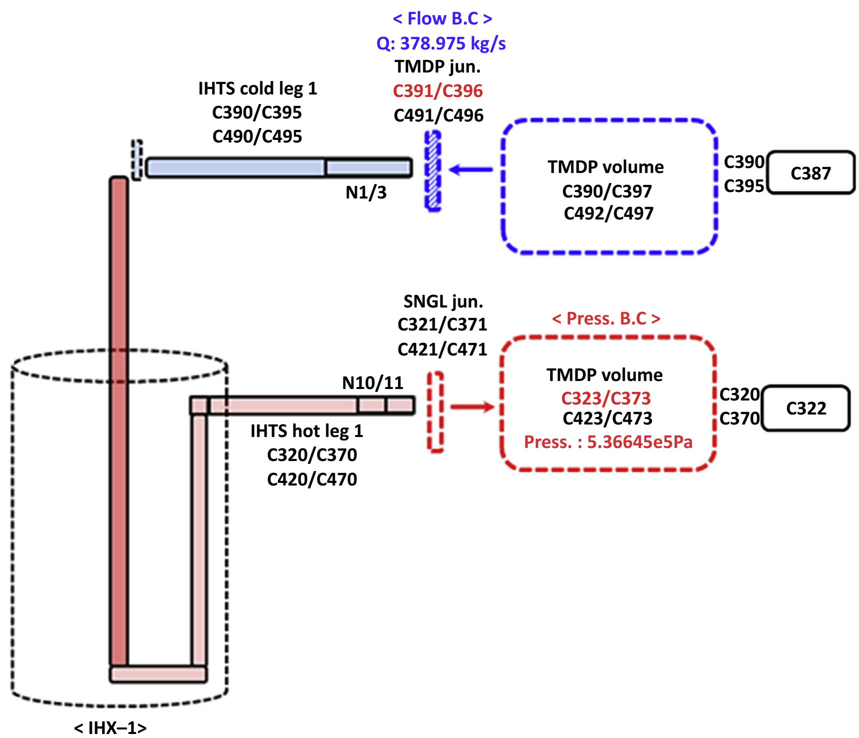

Fig. 13 – Schematic diagram to model the affected intermediate heat transport system (IHTS).

satisfies the criteria for DEC, and thus there is no propagation to a severe accident.

4.2. *Unprotected loss of flow*

The unprotected loss of flow event is initiated with PHTS pump failures. Based on the peak coolant temperature, the two PHTS pump failures event is a bounding event. During the event, the radial expansion reactivity acts as the highest negative reactivity as shown in Fig. 19. Because the peak clad temperature is maintained higher than the eutectic temperature, a clad penetration occurs. However, the clad thinning is about 10.0 μm. The peak coolant temperature is 882.1°C as shown in Fig. 20. In addition, the peak coolant temperature has a safety margin against the sodium boiling temperature as shown in Fig. 20.

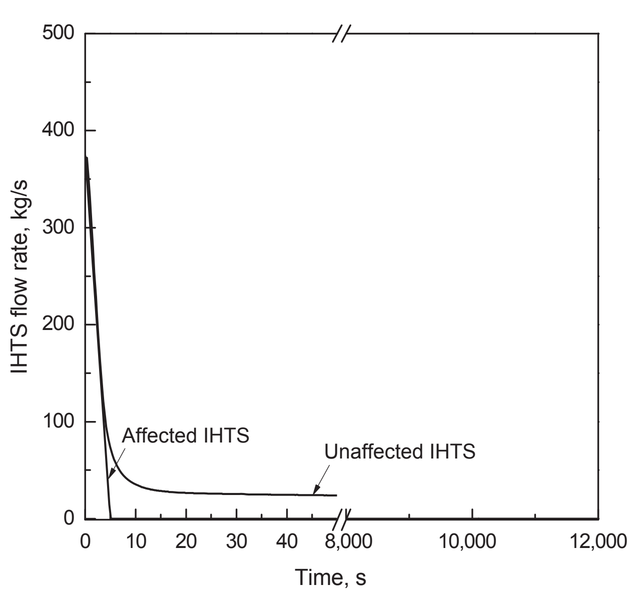

Fig. 14 – Mass flow rate of intermediate heat transport system (IHTS) loops at sodium–water reaction.

4.3. *Unprotected loss of heat sink*

The unprotected loss of heat sink event is initiated with failures of heat removal by SGs. Based on the peak coolant temperature, the feed-water isolation event is a bounding event due to a total loss of the SGs. During the event, the core radial expansion affects the dominant negative reactivity, whereas CRDL/RV is the dominant positive as shown in Fig. 21. In this event, there is no clad thinning, because the peak clad temperature is maintained below the eutectic temperature. In addition, the peak coolant temperature is calculated as 623.5°C, which has an enough margin against the boiling temperature as shown in Fig. 22.

4.4. *Internal blockage*

A subchannel blockage could occur by the collection of fragments or erosion substances formed in PHTS. The blockage induces a large pressure drop, which decreases the mass flow rate. Thus, the outlet temperature could be significantly increased, and the accidents could lead to a failure of the fuel cladding or local boiling. The large blockages are extremely unlikely, because the inlet flow modules of the PGSFR are designed to prevent large particles

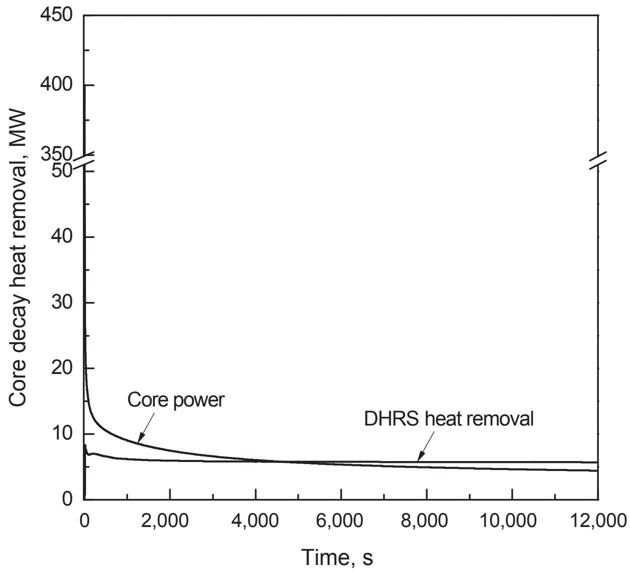

Fig. 15 – Heat removal of decay heat removal system (DHRS) at sodium–water reaction.

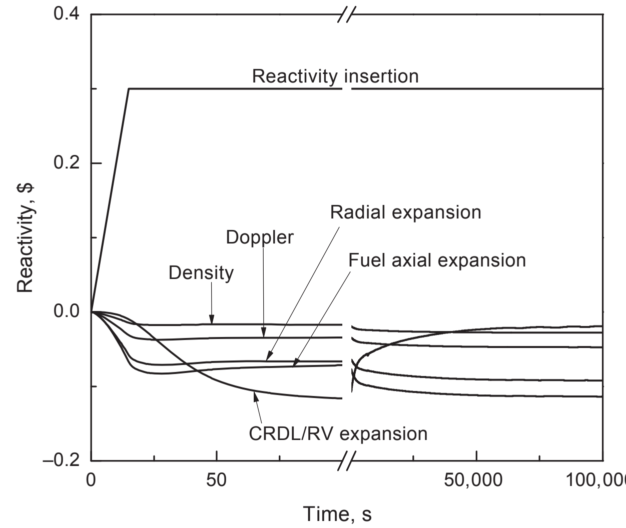

Fig. 17 – Reactivity behaviors during the unprotected transient overpower. CRDL/RV, control rod driveline and reactor vessel.

from entering into the fuel assemblies. In addition, the PGSFR design adopts wire-wrap spacers, which can minimize the possibility of trapping debris in the fuel assembly region. Therefore, this study is focused on the number of blocked orifice hole. The consequence of the blockage accident in a fuel assembly is analyzed with a subchannel analysis code MATRA-LMR/FB [4]. Fig. 23 shows the clad temperature according to the number of blocked orifice hole. The maximum temperature appears at the end of about 1,000 mm in an effective core. When two holes are blocked among the six holes of the inlet orifice, the highest temperature is calculated as 621°C. If three orifice holes are blocked, the maximum clad temperature is calculated as 664°C. If four or more holes are blocked, a reactor protection signal provides the reactor trip by detecting the coolant temperature. There is thus a very low possibility of the inlet blockage accident proceeding to a severe accident.

5. Source term evaluation

The source term (ST) is defined as the release of radionuclides from the fuel and coolant into the containment, and subsequently to the environment. Because there are not much experimental data or many years of experience about the ST of metal fuel in SFR, the super-safe, small, and simple (4S) reactor methodology [5] is applied on the preliminarily evaluation of the in-vessel ST in the PGSFR. This section presents

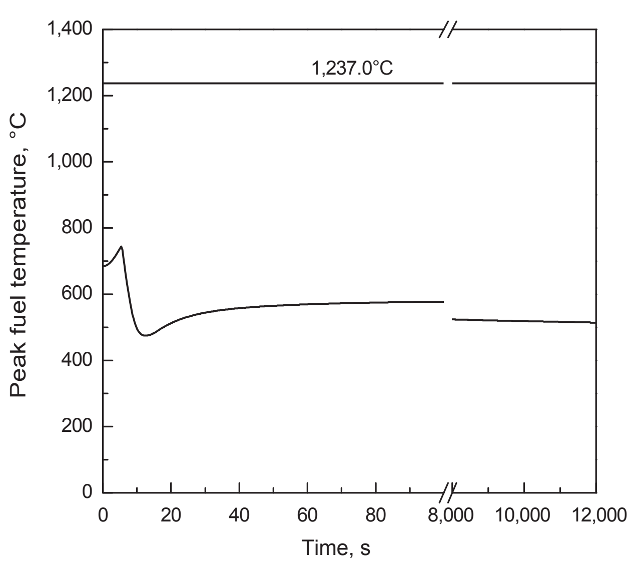

Fig. 16 – Peak fuel temperatures at sodium–water reaction.

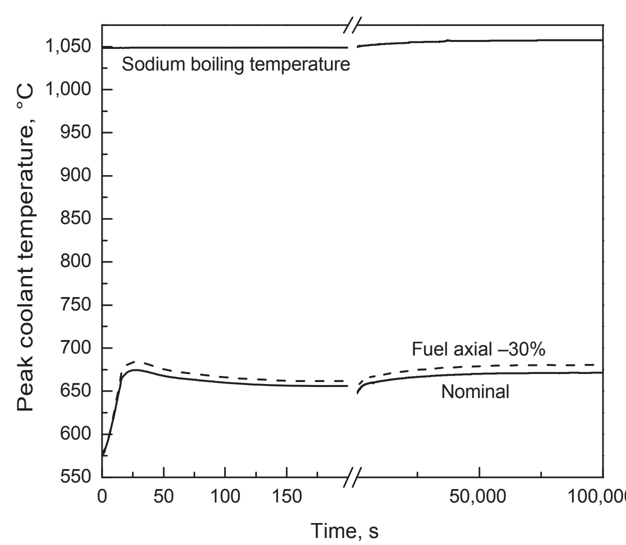

Fig. 18 – Sodium temperatures during the unprotected transient overpower.

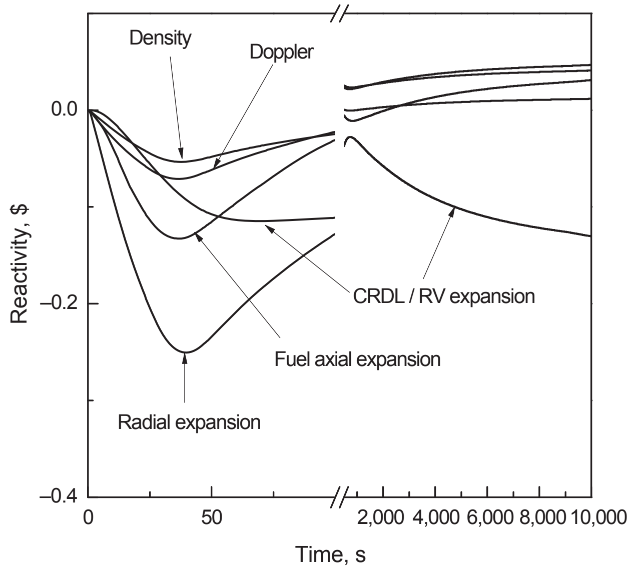

Fig. 19 – Reactivity behaviors during the unprotected loss of flow. CRDL/RV, control rod driveline and reactor vessel.

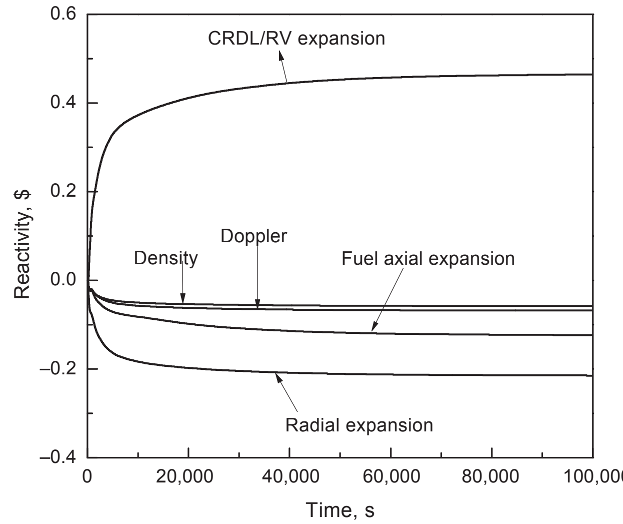

Fig. 21 – Reactivity behaviors during the unprotected loss of heat sink. CRDL/RV, control rod driveline and reactor vessel.

the matters of progress of the preliminary evaluation on the in-vessel ST.

The radionuclide groups are specified based on NUREG-1465 ST [6] and Regulatory Guide 1.183 [7]. Radionuclides with a half-life of more than 1 minute are considered. The radionuclide groups and the elements are as follows:

1. Nobles gases: Xe and Kr
2. Halogens: I and Br
3. Alkali metals: Cs and Rb
4. Tellurium group: Te, Sb, and Se
5. Barium, strontium: Ba and Sr
6. Noble metals: Ru, Rh, Pd, Mo, Tc, and Co
7. Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, and Am
8. Cerium group: Ce, Pu, Np, and U
9. Coolant: Na

The inventory of each radionuclide is calculated by the ORIGEN-2 code using the realistic burn-up conditions. The nominal value of the radiological inventory is multiplied by a factor of 1.1 as an uncertainty margin to give the radiological inventory.

5.1. *Release from the core to the primary sodium*

ST during the release from the core to the primary sodium is calculated based on the 4S methodology. In the noble gases

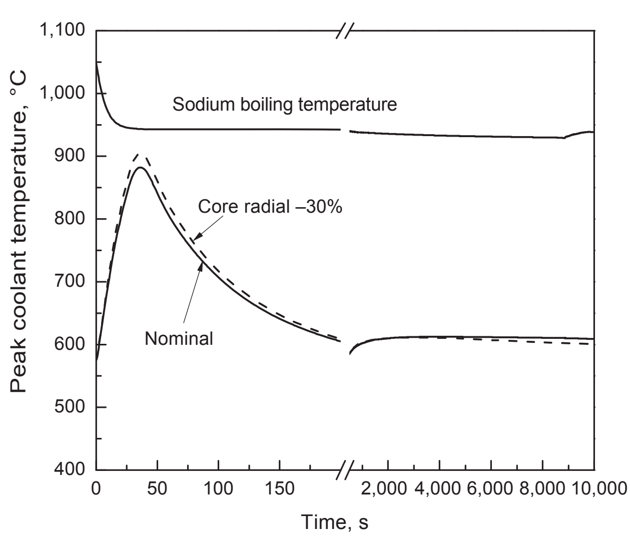

Fig. 20 – Sodium temperatures during the unprotected loss of flow.

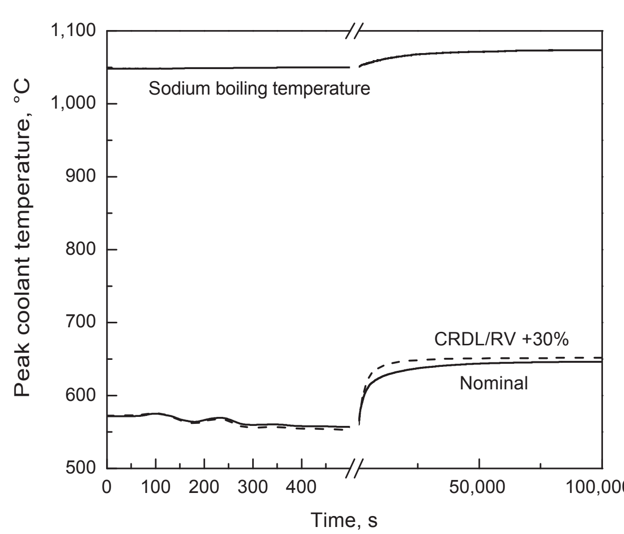

Fig. 22 – Sodium temperatures during the unprotected loss of heat sink. CRDL/RV, control rod driveline and reactor vessel.

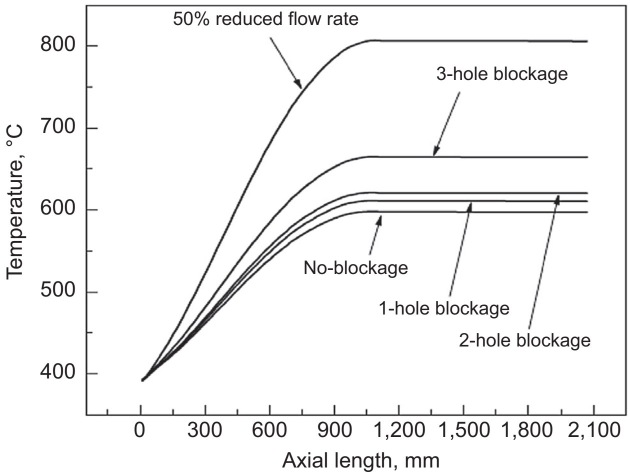

Fig. 23 – Peak clad temperatures according to the number of orifice hole blocked.

group, the fission gas is retained mostly in the fuel with burn ups less than 1–2%; 100% of the fission gas is assumed to be released instantaneously to the primary sodium on clad failure.

In the halogens and alkali metals groups, although the formation of CsI is possible for both types of fuel, the possibility of retaining elemental I in the PGSFR fuel is made extremely remote by the presence of uranium metal and sodium (to form UI_3 and NaI). In this calculation, no I is retained in the fuel as UI_3, 100% of the Cs inventory is released from the fuel to the primary sodium as elemental Cs (no CsI is formed), and 100% of I is released from the fuel to the primary sodium.

In the Te group, the elements interact with the fuel bond sodium to form Na_2X compounds (e.g., Na_2Te for Te). In this calculation, 100% of the inventory in this group is involved in this reaction with the fuel bond sodium. In the Ba and Sr group, the melting points of Ba and Sr are higher than the peak fuel temperature estimated for DBAs and most of the inventory will be likely retained in the fuel. In this calculation, 100% of the inventory of this group is dissolved in the bond sodium and released.

In the noble metals group, noble metals have melting points that are significantly higher than that of the metallic fuel. In this calculation, a release fraction of 0.1% is assumed. In the Ce and lanthanides groups, the elements hardly react with sodium and have low solubility. In this calculation, a release fraction at 1% pin failure is estimated to be less than 0.001% at the fuel end of life.

5.2. *Release from the primary sodium to cover gas space*

ST during the release from the primary sodium to the cover gas space is calculated using the assumption of the 4S methodology. Fig. 24 shows the release fraction with sodium temperature. For the assumed primary sodium temperature of 650°C, the release fractions from the primary sodium to the cover gas space are 3.7×10^{-6}, 7.9×10^{-6}, and 1.4×10^{-4} for sodium, NaI, and Cs. In Te, Ba, and Sr; noble metals; and in the Ce and lanthanides groups, the elements have very small saturated vapor pressure compared with sodium, halogens, and alkali metals. In this calculation, the release fraction of these elements should be as low as that of sodium or lower (3.7×10^{-6}).

Table 3 summarizes the results of the in-vessel ST in the PGSFR by applying the assumption of the 4S methodology. The leakage from the cover gas to the environment through the containment can be calculated using the in-vessel ST and the design leakage rate.

6. Conclusions

The consequences of the system transient during DBAs and DECs for the PGSFR are analyzed with the MARS-LMR and the local faults such as a flow blockage are analyzed with the MATRA-LMR-FB to evaluate the integrity of fuel, cladding, and coolant during the accidents. In addition, the in-vessel ST is calculated based on the 4S methodology.

Conservative and best-estimated methodologies are applied to DBA and DEC, respectively. The results of the DBAs show that the PGSFR design meets safety acceptance criteria with a sufficient margin and maintains its safety functions required to mitigate the accidents. Through the analysis result of DECs, the inherent safety characteristics of the PGSFR from a negative reactivity feedback are identified, and the ATWS results indicate an enough margin of sodium boiling that can cause the core melt or disruptive accident. Moreover, the result of a subchannel blockage event shows that there is much low possibility for the event to extend to a severe accident in the PGSFR design. Furthermore, the progress of the preliminary evaluation on the in-vessel ST is developed based on the 4S methodology. The final ST from the cover gas to the environment through the containment will be obtained using this in-vessel ST and the design leakage rate in further work.

In conclusion, the preliminary specific design of the PGSFR ensures safety margins for DBAs and it also accommodates the DECs of ATWSs without further proceeding to a more severe condition.

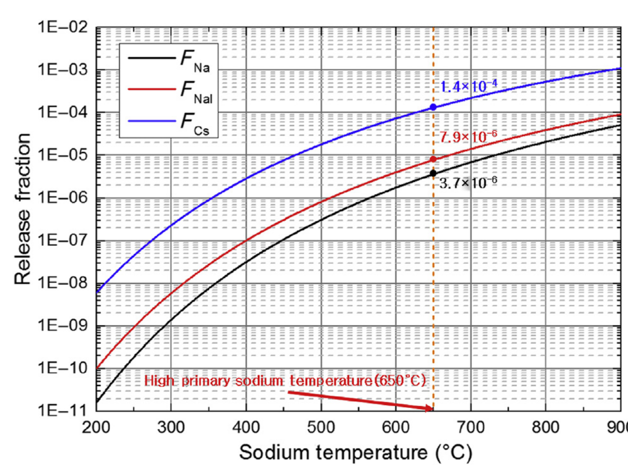

Fig. 24 – Release fractions of sodium, NaI, and Cs from coolant to cover gas.

Table 3 – Results of the in-vessel source term in Prototype Gen-IV Sodium-cooled Fast Reactor.

Radionuclide group	Elements	In-vessel source term (g)		
		1% Failed fuel pins	1 Melted fuel assembly	Whole core melt
Noble gases	Xe	4.39496×10^{2}	3.92470×10^{2}	4.39496×10^{4}
	Kr	4.62520×10^{1}	4.13030×10^{1}	4.62520×10^{3}
Halogens	I	2.39730×10^{-4}	2.14079×10^{-4}	2.39730×10^{-2}
	Br	2.07996×10^{-5}	1.85741×10^{-5}	2.07996×10^{-3}
Alkali metals	Cs	5.25273×10^{-2}	4.69069×10^{-2}	5.25273×10^{00}
	Rb	5.90118×10^{-3}	5.26975×10^{-3}	5.90118×10^{-1}
Tellurium group	Te	2.07916×10^{-4}	1.85669×10^{-4}	2.07916×10^{-2}
	Sb	2.35175×10^{-5}	2.10012×10^{-5}	2.35175×10^{-3}
	Se	2.61019×10^{-5}	2.33090×10^{-5}	2.61019×10^{-3}
Barium, strontium	Ba	5.24106×10^{-4}	4.68027×10^{-4}	5.24106×10^{-2}
	Sr	4.23429×10^{-4}	3.78122×10^{-4}	4.23429×10^{-2}
Noble metals	Ru	8.03944×10^{-5}	7.17922×10^{-4}	8.03944×10^{-2}
	Rh	1.89944×10^{-5}	1.69620×10^{-4}	1.89944×10^{-2}
	Pd	2.25723×10^{-5}	2.01571×10^{-4}	2.25723×10^{-2}
	Mo	1.25720×10^{-4}	1.12268×10^{-3}	1.25720×10^{-1}
	Tc	3.03901×10^{-5}	2.71383×10^{-4}	3.03901×10^{-2}
	Co	0.00000×10^{00}	0.00000×10^{00}	0.00000×10^{00}
Lanthanides	La	4.77959×10^{-7}	4.26817×10^{-4}	4.77959×10^{-2}
	Zr	3.21490×10^{-5}	2.87090×10^{-2}	3.21490×10^{00}
	Nd	1.40441×10^{-6}	1.25413×10^{-3}	1.40441×10^{-1}
	Eu	2.56746×10^{-8}	2.29274×10^{-5}	2.56746×10^{-3}
	Nb	2.06224×10^{-8}	1.84158×10^{-5}	2.06224×10^{-3}
	Pm	1.14719×10^{-7}	1.02444×10^{-4}	1.14719×10^{-2}
	Pr	4.33195×10^{-7}	3.86844×10^{-4}	4.33195×10^{-2}
	Sm	2.53532×10^{-7}	2.26405×10^{-4}	2.53532×10^{-2}
	Y	2.15475×10^{-7}	1.92419×10^{-4}	2.15475×10^{-2}
	Cm	3.07875×10^{-12}	2.74933×10^{-9}	3.07875×10^{-7}
	Am	1.13605×10^{-10}	1.01450×10^{-7}	1.13605×10^{-5}
Cerium group	Ce	1.05652×10^{-6}	9.43475×10^{-4}	1.05652×10^{-1}
	Pu	5.78650×10^{-6}	5.16735×10^{-3}	5.78650×10^{-1}
	Np	1.38801×10^{-7}	1.23949×10^{-4}	1.38801×10^{-2}
	U	2.87778×10^{-4}	2.56986×10^{-1}	2.87778×10^{1}
Coolant	Na	2.67460×10^{-6}	2.67460×10^{-6}	2.67460×10^{-6}

Conflicts of interest

The authors have no conflicts of interest to declare.

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