A Review of Various Stages of the Severe Accident and Different Physical Phenomena for the Sodium-cooled Fast Reactor

Chul-Kyu Lim[a], Bong-Jin Ko[a], Sang-Gil Park[a], Si-Won Seo[a] and Sang-Ji Kim[b*]
[a]*Atomic Creative Technology Co., Ltd., #204, IT Venture Town, 35, Techno 9 Ro, Yuseong-gu, Daejeon 34027, Korea*
[b] *Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseoung-gu, Daejeon, 34057, Korea*
[*]*Corresponding author: sjkim3@kaeri.re.kr*

1. Introduction

So far, Design Based Accident (DBA) and Design Extended Condition (DEC) events scenarios of Prototype Gen-Ⅳ Sodium-cooled Fast Reactor (PGSFR) developed by Korea Atomic Energy Research Institute (KAERI) have been mostly studied [1, 2], and the study of severe accident such as Hypothetical Core Disruptive Accident (HCDA) have been considered only at the pre-disassembly phase. In addition, the development of the thermal fluid model and the reaction model of MARS-LMR code has been carried out [3], and the transient analysis methodology of PGSFR system has been established based on it [4]. However, the interpretation of severe accident is not comprehensive and the description of severe accident scenarios and phenomena seems to be somewhat insufficiency. Therefore, this paper examines the major severe accident scenarios and phenomena applicable to PGSFR of pool type with metal fuels.

2. Severe Accident Scenarios

In the case of a hypothetical multiple accident that does not cause a reactor shutdown due to failure of the protection system, the Core Disruptive Accident (CDA), in which the core is heated and the fuel and the cladding melt, may occur in SFRs. An accident that could cause fuel disruption is the Anticipated Transient without Scram (ATWS) events. Table 1 summarizes the types of accidents and initial events of ATWS.

Table 1: Accident types and initial events of ATWS

Types	Initial events	Note
Unprotected Transient of Over Power, UTOP	An inserted control rod is accidentally withdrawn	It is assumed that an abrupt amount of positive response is added to the core as a control rod is withdrawn and the Reactor Protection System (RPS) is not working properly.
Unprotected Loss of Flow, ULOF	A power to the coolant pumps is lost	It is assumed that all the power supplied to the primary pump is lost, or the forced circulation is stopped due to a common failure of other factors, and the reactor trip is failure.
Unprotected Loss of Heat Sink, ULOHS	The IHTS is trip.	It is assumed that only Passive Decay Heat Removal System (PDHRS) is the way to eliminate core generation heat as the normal heat removal capability through Intermediate Heat Transport System (IHTS) and Steam Generator (SG) is completely lost, and so on.
	A power to the IHTS is lost	
	A feed-water supply to the steam generators is lost	

It is very unlikely to occur, but a severe accident in which the entire core is collapsed under hypothetical conditions is called the HCDA. In general, the studies on severe accidents for SFRs are based on HCDA. In order to cope with severe accidents of the SFRs, the understanding the potential scenarios of HCDA should be improved. The process of HCDA can be classified as shown in Fig. 1.

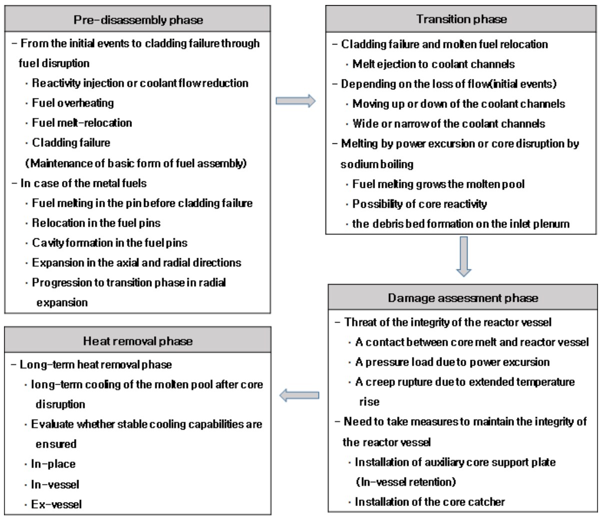

Fig. 1. The various stages of HCDA.

2.1 Pre-disassembly Phase (Core Damage and Cladding Failure Stage)

In the case of metal fuels used in PGSFR, the melting of the fuel occurs first before a cladding failure due to the low melting temperature, and the molten fuel is to be relocated within the fuel pin. When the inside of the fuel pin begins to melt, leading to the formation of an internal cavity as shown in Fig. 2 [5]. Since the cavity that is filled with a mixture of molten fuel and fission gas is at a higher pressure than the surrounding environment, it can be expanded in the axial and radial directions to break the cladding or to release it to the upper gas plenum as shown in Fig. 3 [5].

If the cavity is extended axially, the pressurized molten fuel in the cavity is connected to the upper plenum with lower pressure, and then the molten fuel can relocate suddenly. It is leading to shutdown of the reactor due to a large insertion of negative reactivity [5]. However, in case of the radial extension of the cavity, the cladding can be melting which can cause fuel pin failure. Then the accident is progressed to the transition phase.

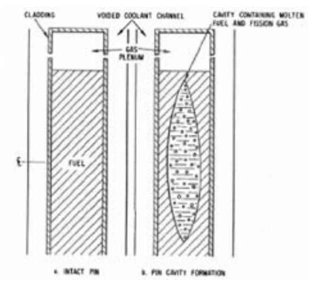

Fig. 2. The cavity formation during initial accident phase [5].

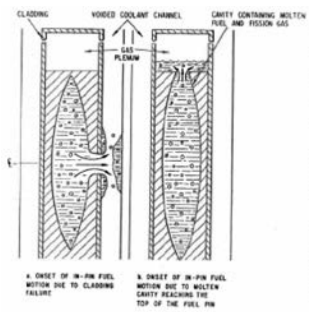

Fig. 3. The molten fuel relocation [5].

2.2 Transition Phase
(Relocation of Molten Fuel Stage)

The scenarios at the transition phase may depend on whether the released molten fuel exits the core through the coolant channels. First, the accidents where the coolant flow rates are maintained consider such as UTOP or ULOHS.

In these initial events, the released molten fuel can easily escape from the coolant channel due to forced convection in channels as shown in Fig. 4. Particularly, in the case of metal fuels, the location of the cladding failure will occur at the top of the core. So, the molten fuel will be easily swept away to the fuel assembly outlet.

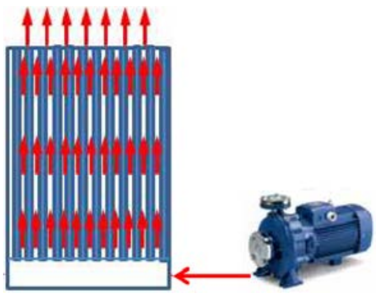

Fig. 4. Conceptual design for the release of the molten fuel by forced flow in channels

Second, the accidents where the coolant flow rates are maintained consider as in ULOF. In this accident, the released molten fuel is dispersed to the upward and downward at the cladding rupture position as shown in Fig. 5. The molten fuels moving upwardly are expected to escape the fuel assembly easily and the freezing of the molten fuel discharged onto the core, which is a relatively high temperature environment, is decreased.

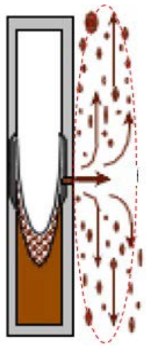

Fig. 5. Fragmentation mechanism in the coolant channel [6].

If a large amount of fission gas was contained in the fuel pin before the cladding rupture, the fission gas would provide a momentum that the melt can easily escape to the top of the core. In the case of the downwardly directed molten fuel, it is expected that the molten fuel will become more easily solidified and plugging or blocking will occur due to the lower temperature compared to the top of the fuel assembly. However, the progress of the accident can be changed depending on the shape of the channel under the core.

When the flow area of the coolant channel is wide and the sodium vapor instead of liquid sodium is filled at the location of the cladding failure, the core melt will quickly fall to the point where the liquid sodium is filled. On the other hand, when the flow area of the coolant channel is narrow, the coolant channel will tend to be blocked due to solidification of the core melt, and the possibility of melting at the wall surface of the fuel assembly duct will increase. Also, the core melt escaping through the fuel assembly ducts can damage the walls of neighboring fuel assembly ducts, resulting in melt propagation in fuel assembly units.

2.3 Damage Assessment Phase
(Evaluation Stage of Reactor Vessel Integrity)

If the accident continues unabated over the transition phase, it continues up to core expansion and core collapse. This threatens the integrity of the reactor vessel. There are three major factors that can cause reactor vessel damage in accident process of HCDA.

- Melting due to the direct contact between the high temperature core melt and reactor vessel
- Pressure load due to power excursion
- Creep rupture due to extended temperature rise

Existing analysis and experimental results predict that even if the cladding rupture before the relocation in the fuel fin and the melt diffusion between the fuel assemblies occurs, there will be no re-critical situation and the power excursion will not occur. However, depending on the situation, it is not possible to exclude the possibility that a hot molten pool may be formed due to the occurrence of reorganization in the process of relocation of the core melt. If the bottom of the reactor vessel is damaged due to the erosion due to melting of the reactor substructure due to the direct contact between the molten pool and the reactor substructure, the melt can flow out of the reactor vessel. Once the reactor vessel is destroyed, the possibility of radioactive material release to the atmosphere increases.

In general, the melting point of steel is 1700 K. The melting point of the metal fuel is 1400 K, which is lower than the melting point of steel, but it is known that the melting temperature point is about 1400 K due to the rapid eutectic reaction between core melt and steel. Therefore, the local temperature of the reactor vessel should be kept below 1400 K, but it is not easy. However, if the particulate bed or molten pool can be cooling, there will be no damage to the reactor vessel.

As an alternative to maintain the integrity of the reactor vessel, a backup core support plate is considered. This is to suppress the core melt in the reactor vessel. Another alternative is to install a core catcher. The core catcher may be installed internal vessel, between two vessels, external vessels.

2.4 Heat Removal Phase
(Assessment of Long-term Cooling Stage)

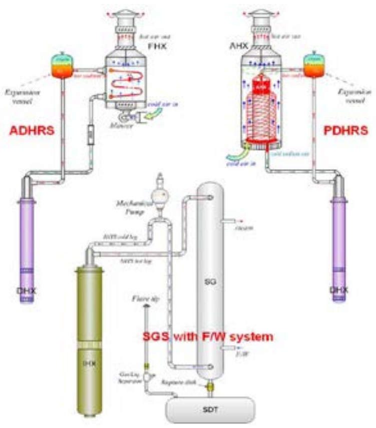

Fig. 6. Schematic diagram of the DHRSs in PGSFR [7]

The final stage of HCDA is to perform long term heat removal after an accident. After the core collapse, the melting core must be cooled from a long-term perspective. The thermo-hydrodynamic characteristics of the core melt should be analyzed to determine whether long-term stable cooling performance is achieved. Long-term heat removal is possible through the following three pathways. The method of removing heat normally through the IHTS and the SG in the reactor core (in-place) (Fig. 6), using the residual heat removal system in the reactor vessel (in-vessel) such as PDHRS (Fig. 6), and the method of cooling by introducing the IVR-ERVC concept can be considered (Fig. 7).

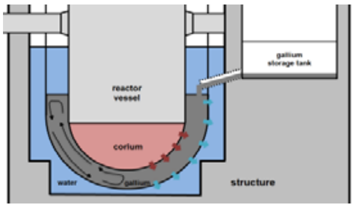

Fig. 7. Schematic diagram for IVR-ERVC [8]

3. Conclusions

As a preliminary review of the study for the severe accidents of PGSFR in Korea, the scenarios for HCDA events in SFRs have reviewed. Since the severe accident scenarios of PGSFR of pool type using metal fuel may be different according to actual design characteristics and operating conditions, it is necessary to study the development of the specific scenarios that reflect it and the concrete definition of HCDA events. Also, future studies on the development of severe accident codes that can analyze the phenomena at each step should be carried out.

REFERENCES

[1] K. L. Lee et. al., A Preliminary Safety Analysis for the Prototype Gen Ⅳ Sodium-Cooled Fast Reactor, NET, Vol. 48, p.1071-1082, 2016.
[2] C. W. Choi et. al., Analyses of UTOP events for the design of control rod stop system in PGSFR using MARS-LMR, Annals of Nuclear Energy, Vol. 96, p.422-431, 2016
[3] J. B. Kim et. al., On the Safety and Performance Demonstration Tests of Prototype Gen Ⅳ Sodium-Cooled Fast Reactor and Validation and Verification of Computational Codes, NET, Vol. 48, p.1083-1095, 2016.
[4] C. W. Choi et. al., Safety Analysis of Anticipated Transient without Scram (ATWS) events for the Prototype GEN-IV SFR (PGSFR) using MARS-LMR, Transactions of the KNS Spring Meeting, Gwangju, Korea, May 30-31, 2013.
[5] A. M. Tentner et. al., SAS4A/SASSYS-1 Safety Analysis Code System_Chapter 15, ANL/NE-12/4, Argonne National Laboratory, 2012.
[6] Ueda et. al., Analytical Evaluation of Metallic Fuel Element Failure Initiated by ATWS, Denryoku Chuo Kenkyusho Hokoku, Vol. 31, 1999.
[7] Jaewoon Yoo, Safety Design Approach for the Development of Prototype Gen-Ⅳ SFR, 4th Joint IAEA-GIF TM/WS on Safety of SFR, IAEA, Vienna, June 10-11, 2014.
[8] S. D. Park and I. C. Bang, Advanced In-Vessel Retention (IVR) Strategy by External Reactor Vessel Cooling (ERVC) with Liquid Metal, Transactions of the KNS Spring Meeting, Gyeongju, Korea, October 24-25, 2013.