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Review of Sodium-cooled Fast Reactor Severe Accident Experiments

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1. Introduction

Immediately after the bombing of Pearl Harbor in 1941, research on plutonium production for atomic weapons was consolidated at the University of Chicago under the Nobel Laureate Arthur H. Compton. The “metallurgical Laboratory” (later to become Argonne National Laboratory) was the code name given to Compton’s facility. It was here that a small group of scientists led by Enrico Fermi built the world‘s first reactor Chicago Pile-1(CP-1), which achieved initial criticality on December 2, 1942. This was became the cornerstone of the nuclear researches which is being performed until now. Four years later, the world’s first fast-neutron reactor ‘Clementine’ mercury cooled experimental reactor was built. After that, researches about fast-neutron reactor, especially sodium-cooled fast reactor (SFR), are performing in U.S., U.K., Germany, Russia, France, Japan, India, and Korea until now [1]. All experimental and prototypical SFR are listed in Table 1 with information about operation period, reactor type, and fuel type.

In Korea PGSFR (Prototype Gen-IV Sodium-cooled Fast Reactor), metal fueled pool type SFR, is developing by KAERI (Korean Atomic Energy Research Institute) and many kind of researches for the PGSFR are performed to obtain specific design approval from Korean regulatory body until 2020. In the present paper, status of experimental researches for SFR severe accident is reviewed and necessary experiments to obtain the specific design approval are proposed. General information and necessary reason of SFR severe accident research are reviewed in section 2. Important experimental researches for SFR severe accident performed by U.S. are investigated in section 3. And also further experimental works for SFR severe accident are proposed in section 4. In section 5, this present study is finalized by summarizing this literature survey.

2. SFR Severe Accident

2.1 General information of SFR severe accident

In the domain of safety, the concerns are expressed since the 1950s particularly referring to the risk of an uncontrolled power excursion in case of large-sized fast reactor systems and positive sodium void effects in case of SFRs. In the context of advanced reactor concepts, for example, Gen IV systems, treatment of severe accidents in the design is one of the key issues of R&D plans [2]. This requires complete understanding of various scenarios and the associated phenomena in the allied domains of science, engineering, and technology that can be hypothesized for robust safety demonstration. The thermal reactor core is optimized with respect to the fuel to moderator ratio for just optimum moderation, due to which any motion of the fuel material will lead to negative reactivity under loss of moderation. In the fast reactor, the core is not in optimum reactivity configuration. This means that any motion of fuel, depending upon core compaction and core expansion or dispersion, could introduce either positive or negative reactivity, respectively. If there is fuel melting, there will be core compaction due to the downward motion of molten fuel or fuel slumping, which will lead to large positive reactivity addition. This in turn will result in a super-prompt critical excursion and release of a lot of thermal energy followed by mechanical consequences. The severe accident scenario in the SFR is defined based on this physics.

For this accident to take place, at least two or more low probability failures must take place in sequence, for example, a large reactivity insertion event coupled with complete failure of the plant protection system. Those accidents, which can cause degradation or melting of whole core, are called severe accident or CDA (Core Disruptive Accident) in SFR. Probability of occurring CDA is extremely low, and hence, it is referred to as a hypothetical accident and called HCDA [3]. ULOF (Unprotected Loss of Flow), UTOP (Unprotected Transient Overpower), and ULOHS (Unprotected Loss of Heat Sink) are considered as HCDA in SFR. The reason of selecting these events as HCDA in SFR is related to reactivity feedback mechanism of SFR [4].

2.2 necessity of SFR severe accident research

Researches for preventing and mitigating HCDA should be performed even if there is no severe accident which result in whole core degradation or melting in the SFR operation history. This is because there is probability of occurring HCDA although the probability is extremely low. Actually, melting and relocation of some fuel rods in a sub-assembly were experienced in Fermi-1 and EBR-II [5, 6]. It indicates HCDA can’t be eliminated totally in SFR safety concerns.

In addition, U.S. NRC recommended that PRISM has to show safety against severe accident in PSER (Pre-application Safety Evaluation Report) [7] although HCDA can be eliminated due to inherent safety feature of SFR SMR (Small Modular Reactor) in PRISM PSID (Preliminary Safety Information Document) [8].

And also SNL (Sandia National Laboratory) listed severe accident phenomena as knowledge gaps in their report [9]. They insist that these knowledge gaps are must be solved for safe and stable operation of SFR. According to above reports, SFR severe accident research is very important in point of reactor safety and scientific achievement.

But now SFR severe accident research becomes indispensable thing in Korea because Korean government legislates for which all NPPs in Korea must manage severe accident in form of SAMG after the Fukushima accident. Therefore, severe accident research for PGSFR is inevitable in point of specific design approval and important in point of escalating scientific level.

3. Experiments for SFR Severe Accident

Severe accident researches of oxide fueled SFR such as sodium thermo-hydraulics, Fuel motion, FCI (Fuel Coolant Interaction), sodium fire, and source term have being performed by mainly France and Japan. Unfortunately, results and data of these researches can't be applied directly to the PGSFR because the PGSFR is metallic fueled SFR. Experimental research results about same type of reactors such as EBR-II and TREAT should be applied to the PGSFR. EBR-II and TREAT experiments are selected in this study because typical HCDAs (ULOF, UTOP, and ULOHS) are covered in these experiments. Applicable results of these two experiments are followed.

3.1 EBR(Experimental Breeder Reactor)-II

EBR-II, metallic fueled pool type SFR, had contributed to SFR research for about 30 years after it reaches first criticality in 1964. EBR-II operation can be divided into three phase. In first phase, research was focused on transforming U-238 to plutonium for nuclear weapons. Second phase started in 1969 and studies about irradiated fuel inspection, fuel material, and fuel burnup effects are performed. Research purpose was changed to SFR safety in third phase, 1983 due to the TMI-2 accident. Many SFR safety studies such as SHRT (Shutdown Heat Removal Test) are performed in EBR-II during the third phase. Especially residual heat removal using the natural circulation and inherent feedback effect are addressed in this study.

First of all, we will review demonstration of passive decay heat removal system in EBR-II. Reactor was stopped abruptly during 100% full power and 70% power condition without any active coolant cooling system such as forced circulation cooling by using primary pump and SG. In this experiment, EBR-II could cool down the coolant by using the passive decay heat removal system. This implies the metallic fueled pool type SFR like as EBR-II also has ability to cool down the reactor by using residual heat removal system without any active cooling system. In addition, ability of passive reactivity control of EBR-II is verified by perturbing primary coolant and secondary feedwater flow rate. As a result of this experiment, EBR-II can maintain well stable and safe condition in abnormal operation by its inherent reactivity feedback ability. It means metallic fueled pool type SFR has proven ability to control reactivity inherently in abnormal condition [10]. And also inherent reactivity feedback was tested in conditions of ULOF and ULOHS [11, 12]. As a result of these two experiments, negative reactivity was inserted by inherent reactivity feedback and reactor was tripped safely. Validation of NATDEMO code also performed in these experiments. Above experimental researches results using the EBR-II shown that the metallic fueled pool type SFR has these abilities;

- Passive decay heat removal (natural circulation),
- Passive reactivity control in abnormal condition,
- Inherent reactivity feedback in condition of ULOF and ULOHS.

3.2 TREAT M series

HCDA experimental research results in condition of ULOF and ULOHS are addressed in former section. In this section, HCDA experimental study results in UTOP condition will be reviewed by reviewing the TREAT M series experiment result. TREAT R, L, H, E, M series experiments had been performed in US. Among these experiments, M series [13], especially M7 test using U-10Zr fuel and HT9 cladding, can be applicable into the PGSFR. M series tests were performed by reactivity insertion during 8 seconds to reproduce the UTOP condition. Purpose of these tests is as follows; 1) determination of margin to failure and identification of underlying mechanisms, 2) assessment of prefailure axial expansion as a potentially significant prefailure reactivity removal mechanism 3) preliminary assessment of postfailure events, i.e., behavior of disrupted fuel and coolant.

As results of M series experiments, cladding damage induced in M series tests was strongly weighted to high cladding temperature of ~1350 K. Typically, at M series heating rates, nearly total eutectic penetration would be required to fail cladding at low burnup, partial penetration would be required at midrange burnup, and almost no penetration would be required at high burnup. When cladding failed, similar postfailure events characterized the behavior of all fuel types tested. In each case, about half of the fuel inventory, corresponding roughly to the fuel melt fraction, was ejected rapidly through a small cladding breach at the fuel top. Cladding failure was always accompanied by a sudden, temporary reversal of inlet coolant flow and rapid coolant voiding. Ejected fuel dispersed rapidly, combing with cladding and structural materials into a highly mobile low melting point eutectic and traveling upward with the coolant to locations well downstream of the original fuel zone. Once ejected, molten fuel was highly mobile in the coolant channel, showing little tendency to cause blockages.

4. Necessary Research

Representative HCDAs (ULOF, ULOHS, and UTOP) are studied experimentally in EBR-II and TREAT, and these tests are reviewed in former section. However, many scenarios exist in SFR severe accident and these all scenarios are not covered by above experiments. That is, still needs for verification of inherent safety features of SFR are remained. One of these scenarios is as follow. Cladding will be failed and then molten fuel ejected into the coolant channel after the ULOF is occurred. If the ULOF occurred in low burnup fueled SFR then a little FP (Fission Product) exists in fuel pin and pressure in intact fuel is relatively low. So ejected fuel can’t be fragmented and may be poured to the bottom of core. Theses molten poured fuel can form a molten fuel pool in center of subassembly and then this pool may block the coolant channel. If coolant channel blockage is occurred, core cooling is not possible. This molten fuel can propagate to the hex can surface and duct can be penetrated by relocated molten fuel. If it is, molten fuel can pass through between ducts and huge molten fuel pool can be formed in whole core. It causes recriticality and then enormous energy can be released from the core. Finally whole core may be disrupted. Therefore fuel motion and flow blockage possibility when the ULOF take place in low burnup fueled reactor must be studied.

5. Conclusion

Literature survey about experimental research for the SFR severe accidents is performed to apply these results into the PGSFR. Inherent safety features were tested in condition of HCDAs such as ULOF, ULOHS, and UTOP by EBR-II and TREAT tests. But all HCDA scenarios are not considered by above experiments. So other type of experiments to verify the inherent safety features of SFR should be performed for other HCDA scenario, especially ULOF with partial/total channel blockage. Safety issues such as inherent reactivity feedback, fuel motion, and channel blockage must be solved by experimental research to obtain the specific design approval from the Korean regulatory body.

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Table 1. History of SFR development and operation

Country	Name	Operational Period*	Reactor Type	Fuel Type
U.S.	Clementine	1946 - 1952	Mercury-cooled Experimental Reactor	Plutonium Metal
	EBR-I	1951 - 1963	Loop Type NaK-cooled Fast Reactor	Metallic Fuel**
	LAMPRE 1	1961 - 1963	Sodium-cooled Fast Reactor	Molten Plutonium
	Fermi 1	1963 - 1972	Loop type SFR	Metallic Fuel (U-10%Mo)
	EBR-II	1964 - 1994	Pool type SFR	Metallic HEU*** fuel
	SEFOR	1969 - 1972	SFR	MOX (Mixed-Oxide Fuel)
	FFTF	1980 - 1992	Loop type SFR	MOX
	CRBR	1981 - 1983	Loop type SFR	MOX
France	Rapsodie	1967 - 1983	Loop type SFR	MOX
	Phenix	1973 - 2009	Pool type SFR	MOX
	Superphenix 1	1985 - 1996	Pool type SFR	MOX
	ASTRID	Developing		
Japan	Joyo	1977 - Present	Loop type SFR	MOX
	Monju	1994 - 2010	Loop type SFR	MOX
Russia	BR-5 / BR-10	1958 - 2002	Loop type SFR (BR-5: NaK, BR-10: Na)	MOX
	BOR-60	1968 - present	Loop type SFR	MOX
	BN-350****	1972 - 1994	Loop type SFR	
	BN-600	1980 - present	Pool type SFR	MOX
	BN-800	under construction	Pool type SFR	MOX
	BN-1200	Developing		
U.K.	DFR	1959 - 1977	Loop type SFR (NaK)	Metallic fuel → MOX
	PFR	1974 - 1994	Pool type SFR	MOX
Germany	KNK-II	1977-1991	Loop type SFR	MOX
	SNR-300	N/A	Loop type SFR	N/A
India	FBTR	1985 - Present	Loop type SFR	MOX
	PFBR	under construction	Pool type SFR	MOX
	CFBR	Developing		

China	CEFR	2010 - Present	Pool type SFR	MOX
	CDFR	Developing		
Italy	PEC	Never went critical		
International Collaboration	GIF◆	2001 - Present		
	INPRO◆◆	2001 - Present		
	IFNEC◆◆◆	2006 - Present		

* 1st Criticality ~ last shutdown
** Various type of metallic fuel was used
*** Highly Enriched Uranium
**** This is because BN-350 was developed by the Soviet Union even if it is located in Kazakhstan.
◆ GIF is abbreviation of 'Generation IV International Forum'.
◆◆ Collaborative research group of 40 IAEA members
◆◆◆ It was formerly GNEP (Global Nuclear Energy Partnership). The name of IFNEC (International Framework for Nuclear Energy Cooperation) is used from 2010.