FAST REACTOR TECHNOLOGY R&D ACTIVITIES IN CHINA

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The basic research on fast reactor technology was started in the mid-1960's in China. The emphasis was put on fast reactor neutronics, thermohydraulics, sodium technology, materials, fuels, safety, sodium devices and instrumentation. In 1987, the research turned to applied basic research with the conceptual design of a 60 MW experimental fast reactor as a target. The Project of the China Experimental Fast Reactor (CEFR) with a thermal power 65 MW was launched in 1993. The R&D of fast reactor technology then carried out to serve a design demonstration connected with the different phases of the conceptual, preliminary and detailed design of the CEFR.

Recently, three directions of fast rector technology R&D activities have been considered, and some research programs have been developed. They are: (1) R&D related to the CEFR, i.e. experiments to be conducted on the CEFR for its safe operation, (2) R&D related to the projects of a prototype and the demonstration of fast reactors, and (3) advanced SFR technology within the framework of the international cooperation of INPRO and GIF.

KEYWORDS : Sodium-cooled Fast Reactor (SFR), Research and Development (R&D), Design Demonstration

1. INTRODUCTION

In China, the basic research on fast reactor technology was started in the mid-1960's and continued for approximately 20 years. During this period, there was no clear civilian nuclear power program in the country, thus little manpower and few resources were devoted to this research area. In 1986, the National High-Tech Program was launched and fast reactor technology development brought into line with this program. An applied basic research project was executed until 1993 with an engineering target of a 60 MW experimental fast reactor. It was considered to meet as fully as possible the requirements of the reactor design and safety analysis when working out the program outline.

The conceptual design of the China Experimental Fast Reactor (CEFR) of a 65 MW matched with a 20 MW turbine generator was finished in 1993, as a result of international cooperation with the Russia FBR Association for the technical design of the CEFR main systems and with France CEA for the R&D field. It was followed by the preliminary design and detailed design by a Chinese design team.

The third phase (1993-2006) of the R&D activities was concentrated on the CEFR design demonstration. Nearly 50 subjects were proposed from the conceptual, preliminary and even the detailed design.

For the recent and near-future R&D activities of fast reactor technology, the emphasis of their directions is as follows:

(1) Use the CEFR as a tool to verify the computer codes used for the CEFR design and to be used for the prototype and demonstration of a fast reactor, which have the same main technical selections.
(2) The CEFR is a model for studying CEFR safety properties.
(3) Research to support the safe operation of the CEFR.
(4) Applied research for the following up of the fast reactor prototype or demonstration, including the establishment of rigs and facilities for models that test key components and systems, as well as research related to the simplification of the systems.
(5) R&D for an advanced sodium fast reactor system mainly including an innovative SFR concept, reactor design and safety features, advanced fuel cycle strategy and technology and new materials.

2. PREVIOUS R&D ACTIVITIES FOR THE FAST REACTOR

2.1 Basic Researches (1968-1987)

The basic research for the fast reactor technology was started in the mid-1960's. The emphasis of this research

Table 1. Loops and Facilities for FBR Technology R&D (1968-1987)

No. Facilities	Parameters	Commissioning	Place
1. Fast Neutron Zero Power Facility	50kg U-235	1970.6	Ciae
2. Sodium Corrosion Test Loop	Temp. 600°C Sodium Velocity 12m/S $0<50 \times 10^{-6}$	1970.7	Ciae
3. Sodium Purification Loop	Sodium 150kg $0<20 \times 10^{-6}$	1970.9	Ciae
4. Sodium Thermohydraulic Loop	Sodium Flow Rate 20m^3/H Temp Max.550°C Power 50kw Pressure 0.55mpa	1970.10	Ciae
5. Sodium Convection Corrosion Loop	Temp. Max 700°C Velocity 6cm/S Volume 4 L $0<15 \times 10^{-6}$	1972	Ciae
6. Control Rod Driven Mechanism Parts Testing Facility	Medium: Water Flow Rate 1t/H Driven Length 800cm Eccentricity ±30cm	1979.10	Cnpi
7. Sodium Plugging Meter Testing Loop	Temp. 450°C Flow Rate 1m^3/H Sodium Volume 28 L.	1981.10	Ciae
8. Stress Corrosion Sodium Loop	Temp. Max. 700°C Load 600kg Sample Deformation 0-10mm	1981.12	Ciae
9. Alternating Magnetic Pump Sodium Testing Loop	Flow Rate 5t/H	1968	Shanghai
10. Alternating Magnetic Pump Sodium Testing Loop	Flow Rate 10t/H	1969	Shanghai
11. Direct Magnetic Pump Sodium Testing Loop	—	1968	Shanghai
12. Sodium Pump Testing Loop	Temp. Max 450°C Flow Rate 18m^{3}h	1984	Cnpi

was placed on fast reactor neutronics and safety, liquid sodium thermohydraulics, uranium dioxide fuels, materials and sodium technology including sodium purification, impurities analysis, material-sodium compatibility, sodium instrumentations and devices in small scale. Up to 1987, approximately 12 sodium loops and testing facilities were established, as shown in Table 1.

The Small Fast Neutron Zero Power Facility, as shown in Fig. 1, containing 50kg U-235 was commissioned in 1970, after numerous criticality tests, reactivity measurements, micro-parameter and integral parameter measurements were carried out. Currently, it serves to train the operating team of CEFR, to demonstrate methods of reactor neutronics testing to be used in the first physical start-up of CEFR, and to validate the neutron detectors and instrumentations.

2.2 Applied Basic Research (1988-1993)

In the framework of the National High-Tech Program involved applied basic research with a 60 MW (approximate value) experimental fast reactor as a target. The main emphases were aimed at fast reactor design study, sodium technology, materials, fuels, and reactor safety. More than 20 testing facilities and sodium loops were installed. The main facilities are listed in Table 2.

Table 2. Loops and Facilities for FBR Technology R&D (1988-1993)

No. Facilities	Parameters	Commissioning	Place
1. Sodium Thermohydraulic Loop	Temp. Max 550°C Flow Rate 20m³/H Power 300kw	1990.12	Ciae
2. Sodium Purification Loop	Volume 85 L Temp. Max 520°C Flow Rate 1m³/H	1990.12	Ciae
3. Sodium Boiling Testing Loop	Temp. Max 1000°C Flow Rate 20m³/H	1991.3	Xian Jiaotong University
4. Mass Transfer Sodium Loop	Temp. Max 550°C Temp. Min 450°C Flow Rate 2m³/H $O<20 \times 10^{-6}$ $C<(0.1\sim5) \times 10^{-6}$	1990.10	Ciae
5. Material Corrosion Sodium Testing Loop	Temp. Max 600°C Flow Rate 12 M³/H $O<10 \times 10^{-6}$ $C<1 \times 10^{-6}$	1992.10	Ciae
6. Fission Products-Cladding Testing Facility	Temp. Max 550~700°C O/M:1.96-2.00 Bu Simulation 5-10%	1990.12	Ciae
7. Creep-Fatigue Sodium Testing Loop	Temp. Max 600°C Velocity 1~3m/S	1992	Ciae
8. Bi-Axial Creep Testing Facility	Temp. Max 900°C Pressure Max. 10.0mpa	1990.10	Ciae
9. U-Zr Induction Heating Molten And Pressure Casting Testing Facility	U-Zr 300g	1991.3	Ciae
10. Alloy Fuel Molten And Pressure Casting Facility		1991.12	Cnpi
11. Na-H_2o Reaction Testing Loop	H_2o Leakage Rate<1g/S Temp. 300-500°C Pressure. 1mpa	1991.3	Ciae
12. Hydrogen Detection Sodium Loop	Temp. Max 500°C Flow Rate 5 M³/H	1991	Tsinghua University
13. Carbon Analysis Facility		1991	Ciae

As the first subject of the fast reactor design study, the main research works were as follows: (1) adaptability evaluation of computer codes related to reactor neutronics, shielding, thermohydraulics, mechanics and safety analysis; (2) development of new computer codes and software including multi-group cross-section files for the design of a fast reactor, and compilation of design criteria for key components and systems of fast reactor, among other works.

Concerning sodium technology, a small-scale sodium purification loop was built with a production capacity of 1.5 kg nuclear grade sodium per day, a sodium impurities on-line detection system (plugging meter and immersed sodium flow-meter), sodium-contaminated equipment cleaning system, sodium-fire extinguishing equipment and systems were all established during this period.

In terms of the materials, the studies were performed mainly for 316 Ti modified 20% cold worked stainless steel for fuel cladding and hexagonal tube material. The

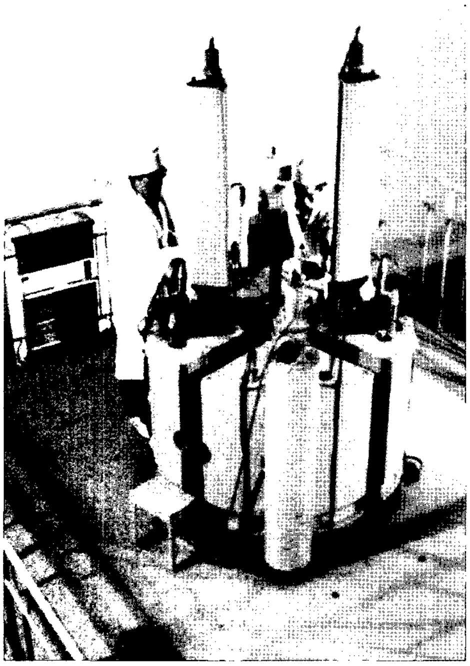

Fig. 1. Fast Neutron Zero Power Facility

mechanics properties, interaction with fission products (out of pile) and bi-axial stress testing based on the materials with stabilized fabrication technology were studied. The trial-fabrication of MOX and U-Zr as well as in-core restructuring tests of UO_2 were completed.

Concerning the safety study, the testing and analysis of electrical heated fuel-pin bundle blockages, sodium boiling, and sodium water reactions were carried out.

Though limited by time and budget constraints, the aforementioned research was neither systematical nor comprehensive. As a result, these projects laid a very good basis for the China Experimental Fast Reactor Project (CEFR).

2.3 Design Demonstration (1993-2006)

In 1993, the China Experimental Fast Reactor Project (CEFR) entered the design stage. By the end of the detailed design, nearly 50 items of design demonstration, of which approximately 10% by Russia through the bilateral cooperation were completed, with the main purpose of verifying the design and accumulating operational experience.

Among these design demonstrations, two rounds of core-neutronics mockup experiments and two rounds of reactor vessel water natural convection tests to verify the passive decay heat-removal capability were done, serving the conceptual design and preliminary design of the CEFR, respectively.

Generally, the CEFR design related components and systems were proved by these demonstrations. However, the performance of the sodium-siphon destruction device for the auxiliary sodium loop was found unacceptable by the experiments on the sodium-siphon testing facility shown in Fig. 2. The device was modified after the experiments to maintain a proper performance level.

Based on the experience gained from the installation of the sodium purification loop set up in 1990, a mid-sized sodium purification system was built in 1999 with a nuclear-grade sodium production capability of 300 kg per day during this design demonstration period. Continuously based on this technology, a nuclear-grade sodium purification plant with a producing rate of 1.5t per day was commissioned in 2005 in Inner Mongolia, north of China and 250t nuclear-grade sodium was safely transported to five storage tanks within the CEFR site.

Various main loops and facilities used for the CEFR design demonstration are shown in Table 3.

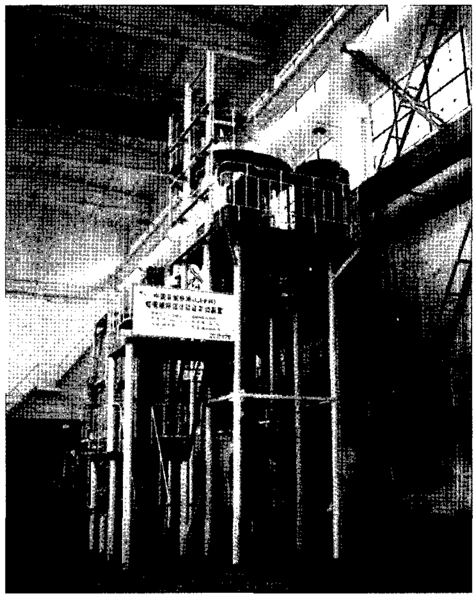

Fig. 2. Siphon Destruction Testing Facility

3. R&D ACTIVITIES PLANNED FOR FAST REACTORS

3.1 R&D Activities on CEFR (2007-2015)

During the pre-operation testing and trial-operation of CEFR, it was planned to collect as much data as possible by the internal CEFR data digital network to validate and verify the designed computer codes and safety analysis codes related to neutronics, shielding, thermohydraulics, mechanics, and transient analyses.

To identify the safety regulations, design criteria, safety guidelines, standards and safety classification of components and systems for CEFR, but which were mainly transferred from PWRs, the conducting of carefully CEFR safety experiments was considered that included UTOP, ULOF,

Table 3. CEFR Design Demonstration Facilities

No. Facilities	Main Parameters	Commissioning	Place
1. Sodium Component Cleaning	Water Spray Rate 1.2kg/H Water Flow Rate 8 L/H N_2 Pressure 0.4mpa N_2 Flow Rate $2m^3/H$	1995.10	Ciae
2. Sodium Purification System (Mid-Size)	Temp Max 350°C Purification Flow Rate 300kg/D $O<20\times10^{-6}$ $Ca<10\times10^{-6}$	1999.7	Ciae
3. Sodium Fire Testing Lab.	Room Size $3\times4\times5m$ Sodium Pressure 0.2mpa Sodium Flow Rate $0.25m^3/H$ Sodium Quantity 250kg	1998.10	Ciae
4. Fixed Expansion Graphite Extinguishing Facility	N2 Pressure 1.6mpa Graphite Volume 40 L	1999.10	Ciae
5. Sodium Fire Detection System	Smoke Sensors Temperature Sensors	1999.2	Ciae
6. Sodium Sol-Gel Purification And Filtration Facility	Volume 200 L Ventilation 1700 M^3/H Filter Pressure Drop 0.25mpa	1999.11	Ciae
7. Sodium Testing Loop (Espresso)	Sodium Flow Rote 110 M^3/H Pressure Max. 1.07mpa Temp. Max 600°C Max. Thermo-Shock 200°C/S	1997	Ciae
8. Sodium Testing Loop (Cedi)	Sodium Flow Rate 320 M^3/H Pressure Max. 1.4mpa Temp. Max 650°C Max. Thermo-Shock 50°C/S	1997	Ciae
9. Water Testing Loop	Flow Rate 100 M^3/H Pressure 9mpa Temp. Max 150°C	1997	Ciae
10. Reactor Vessel Water Natural Convection Simulation (For Cefr Conceptual Design)	Diameter 1.6m Power 30kw	1995	Ciae
11. Sodium Siphon Testing Facility	Cefr-Sized	2002	Ciae
12. Over-Pressure Protection Test Facility	Cefr-Sized	2003	Ciae
13. Sodium Valve Testing Loop	Corresponding To Dg.86	2002	Ciae
14. Fuel Handling System Summing-Up Demonstration	Cefr-Sized	2006	Ciae

and ULOHS starting from low power with small addition of the reactivity value in a step-wise advance. For all of the safety experiments, each step was to be accompanied with calculations in advance and the computer codes and input data were to be checked by the results of the former steps.

3.2 R&D Activities for CEFR Operation (2007-2010)

Research was considered to support the safe operation of the CEFR, including a full-scale CEFR simulator for the training of the operating team, complicated equipment maintenance simulators for the training of the maintenance team, design and construction of a sodium loop for a demonstration of the domestic production of key components, reform and modification of the hot cells to adapt to the examination of irradiated fuels with high contents of TRU, a MOX laboratory for pre-operation testing, and a MA extraction process combined with a PUREX process..

In addition, there is a plan to prepare and place samples of materials 316(Ti), 304, 316SS and graphite shielding materials containing boron used for CEFR in the center or in a reflector for long-term irradiation performance testing.

Establishment of expert systems concerning the operator support system, transient state and operation optimization system and the core-fuel management optimization system are to be arranged.

3.3 Applied Research for Follow Up (2011-2015)

As the second step of the fast reactor engineering development in China, the design and construction of the China Prototype or Demonstration Fast Reactor (CPFR/CDFR) with power of 600MW or more is ongoing. The CEFR provides a sound basis to develop the demonstration reactor as its successor, due to the fact that continuity up to a commercialized fast reactor of the main technical selections; as an example, the operation temperatures in the main heat transfer systems are considered. However, the following applied research is also considered.

First, three important design studies, specifically three design selections that the program is currently facing, should be conducted related to the following: the supporting style of main vessel, the top hung or bottom supported; the number of loops for the secondary circuit; and reactor building seismic isolation system, in case the site maximum seismic intensity reaches 7 degrees in the MSK-64 classification.

Second, several new pieces of equipment and their instrumentation should be development from the viewpoint of safety or domestic production. These include a passive shut-down system, a more efficient delay neutron-detection system, visual detection under sodium, a sodium boiling detection system, the design and manufacture of a sodium pump, a steam generator and an advanced used sodium treatment system.

Finally, the loops and facilities should be built, mainly including a fast neutron zero power facility corresponding to 600MW - 1500MW water Mock-up to simulate a CPFR/CDFR reactor block for passive-decay heat-removal system study, and a multi-purpose sodium engineering loop mainly for sodium pump and steam generator testing

3.4 R&D for Advanced SFR Technology (2007-2015)

Comparing the development targets of the Chinese fast reactor strategy with the goals of the Generation IV nuclear energy systems, they appear to be nearly identical. However, the China fast reactor technical development is only in the stage of experimental reactor construction. It is realized that the advanced sodium fast reactor (SFR) technology proposed to be developed by the G IV technical roadmap is nearly impossible for use in a China Prototype or Demonstration Fast Reactor, which is intended to be built before 2020. However, for further development, this research will be very valuable. Therefore, the following R&D acclivities are considered for the development targets of the China fast reactor.

(1) An innovative concept of the SFR, for example a simplified primary sodium purification system, and a decrease in the risk of sodium leakage as well as a new steam generator with a low risk of sodium-water reaction.
(2) The design and construction of a U-Pu-Zr Alloy fuel fabrication and pyro-processing complex for fast reactor.
(3) Remote fabrication technology of $(Pu\text{-}MA\text{-}U)O_2$ fuel.
(4) Testing, collection and compilation of data file of new prospective materials for an advanced SFR, such as ODS, 15-15Ti, HT9, 9Cr1Mo, or 12%Cr ferric steel.

4. SUMMARY

The strategy and consideration of the near-term R&D Activities for SFR are worked out, but the implement programs remain in the application phase.

Although thus far nearly 50 years of experience with this type of reactor has been accumulated in total for all fast reactor countries,

the realization of the goals of G IV remains relatively costly, with much R&D work to be done. It is a common understanding that the best way is through international cooperation to share both the cost and experience. Presently, China has taken part in INPRO and GIF officially; correspondingly, cooperation programs are in gradual progress.

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REFERENCES

[1] Zhao Rankei et al. Research work progress in Energy Technology Area of "863 Program" (1986-2000), Atomic Energy Publishing House 2001. P.3 (in Chinese).
[2] A Technology Roadmap for Generation IV Nuclear Energy Systems, GIF-002-00 Issued By the US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, December 2002, P.6.