From Abroad

Reaching First Criticality and Physics Experiments in the CEFR

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The first criticality of the China Experimental Fast Reactor (CEFR) was reached on July 21, 2010. The CEFR is the first step in fast reactor engineering development in China, which was launched in the framework of The National High-Tech Program. The design and construction timetable, the main design characteristics and the parameters of the CEFR are briefly introduced in the paper. Emphasis is placed on describing the criticality calculation tools and results before its physics start-up and then the physics start-up testing process and results are described. Finally some important experiments completed at the reactor criticality state are listed.

KEYWORDS : *China Experimental Fast Reactor, first criticality, physics experiments*

I . Introduction

The first step in China for fast reactor engineering development is the China Experimental Fast Reactor (CEFR) which is a sodium cooled pool type reactor with the power 65 MWt/20 MWe. After its conceptual design completed, the technical consultancy and co-technical design were conducted with the Russian FBR Association from 1993 to 1996, and its preliminary and detailed designs were completed in 1997 and 2005 respectively. The ordering of components was started in 1997. Imported components came mainly from Russia, France and the UK, with all imports from abroad accounting for about 30% of the total systems and components budgets. The construction was carried out from 2000 to 2007, followed by pre-operating tests including cooled and hot states without and with sodium in sodium related components and systems. Its first criticality was realized on July 21, 2010, and this was followed by physics experiments on 16 items including control rod worths, reactivity coefficients, power distributions, capture- and fission-rate measurements, neutron spectrum indices *etc.*

The power start-up of pre-operation testing has been started. It is planned to incorporate the CEFR into the national grid with 40% full power in June, 2011.

II . China Experimental Fast Reactor (CEFR)

1. Basic characteristics and history of CEFR

After basic research studies (1967–1986) on sodium cooled fast reactor technology followed by applied basic research studies (1987–1993), the design and construction plan of the China Experimental Fast Reactor (CEFR) was launched. The purposes of the CEFR are the following :

- (1) to accumulate experiences on fast reactor design, construction and operation.
- (2) to irradiate and develop fuels and materials in its role as a fast neutron facility.
- (3) to test and demonstrate prototype equipment in its role as a platform for fast reactor parameters.

In order to decrease the technical-economic risks for following-ups of CEFR, the design requirements proposed for CEFR are as follows :

- (1) Main technical selections should be consistent with prototype and commercial fast reactors ;
- (2) Thermal parameters non-related to reactor size should be close to the commercial parameters.

Based on established pre-requirements, the conceptual design was finished in 1993, and in order to compensate for lack of fast reactor engineering experiences, from 1993 to 1996 work was done in cooperation with the Russia FBR Association for consultancy and co-technical design. Cooperative work with the French CEA for R&D on FBR technology has

been conducted since 1995. The preliminary and detailed designs were completed during 1995–2005. The CEFR construction started in 2000 and was completed in 2009. After about three years pre-operation testing, its first criticality was realized on July 21, 2010, and this has been followed by reactor core physics experiments.

2. CEFR Introduction

The CEFR is a pool type sodium cooled fast reactor, the fuel used for the first loading is UO_2 with enrichment of 64.4% U-235. The primary circuit including two main pumps, four intermediate heat exchangers and reactor core which is shown in **Fig.1** is in a reactor vessel 8 meters in diameter containing 260 t liquid sodium.

The secondary circuit is divided into two loops. Each one mainly composed of one sodium pump, two intermediate heat exchangers and one steam generator (evaporator and super-heater). The tertiary water steam circuit is also divided into two loops, but superheated steam 480°C and 14 MPa from each is combined to enter one turbine generator.

CEFR is a small sodium cooled fast reactor with very strong negative reactivity feedback due to the temperature and power changes and the sodium void reactivity coefficient is negative everywhere the core. So the CEFR core has self-stability features.

The reliable removal of decay heat is very important after reactor shut-down in the accident state if the electricity supply is lost and the main heat transfer system cannot be used. For this reason, two

independent passive decay heat removal systems (PDHRS) are provided. The decay heat is removed by natural convection and circulation of primary and secondary sodium of the PDHRS and natural convection by air as well in such an accident.

The probabilistic analysis showed that the core damage frequency is $4 \times 10^{-7}/\text{reactor} \cdot \text{a.}$ The safety analysis showed that the personal irradiation effective dose equivalent is less than 5 mSv/accident at the boundary of the CEFR site (153 m from CEFR reactor) by any beyond design base accident (BDBA). And the dose limit at the site boundary not requiring short-term off-site response satisfies the safety target proposed for advanced nuclear energy systems of Generation IV.

The main design parameters of CEFR are listed in **Table 1** and its outside view is given in **Fig 2**.

III. Design and Scheme of Neutron Flux Detection

For CEFR source range neutron flux surveillance, four boron counters type ZJ 524 A : I A, I B, II A and II B are equipped outside the reactor guard vessel about 7.74 m from the core center as shown in **Fig. 3**. Three U-235 high temperature pulse fission chambers (type KHT 54-2) P 1, P 2 and P 3 which were imported from Russia, are arranged in the graphite column at the corner of the core shown as Fig. 3. A Cf-252 start-up neutron source with an intensity of $1 \times 10^9 \text{ n/s}$ is used in the core center.

The sensitivity and background values of detectors with start-up neutron source are given in **Table 2**.

Fig.1 The core of CEFR

Table 1 CEFR Main Design parameters

Parameter		Parameters
Thermal power	(MW)	65
Electric power, net	(MW)	20
Reactor core		
Height	(cm)	45.0
Diameter equivalent	(cm)	60.0
Fuel		(Pu, U) O ₂
Pu, total	(kg)	150.3
Pu-239	(kg)	97.7
U-235 (enrichment)	(kg)	42.6 (19.6%)
Linear power max.	(W/cm)	430
Neutron flux	(n/cm ² ·s)	3.7×10^{15}
Burn-up, target max.	(MWd/t)	100,000
Burn-up, first load max.	(MWd/t)	60,000
Inlet temp. of the core	(°C)	360
Outlet temp. of the core	(°C)	530
Diameter of main vessel (outside)	(m)	8.010
Primary circuit		
Number of loops		2
Quantity of sodium	(t)	260
Flow rate, total	(t/h)	1,328.4
Number of IHX per loop		2
Secondary circuit		
Number of loops		2
Quantity of sodium	(t)	48.2
Flow rate	(t/h)	986.4
Tertiary circuit		
Steam temperature	(°C)	480
Steam pressure	(MPa)	14
Flow rate	(t/h)	96.2
Plant life	(a (year))	30

Fig. 2 The outside view of CEFR

IV. Criticality Calculation

The physics start-up testing of CEFR is divided into two phases including loading up to first criticality and physics testing under low power. Three categories of tests are been conducted which are control rod reactivity measurements, reactivity coefficient measurements and capture- and fission-rate measurements using foil activation methods.

Fig. 3 Arrangement of neutron flux detectors**Table 2** Sensitivity and background values of detectors

Detectors	I A	I B	II A	II B	P 1	P 2	P 3
Sensitivity (c/n/cm ² ·s)	30.8				0.8		
Background (with Cf source)							
Control rods in core (c/s)	~0.5				57.1	13.9	14.8
Control rods out of core (c/s)	3.8	4.3	3.5	3.9	52.5	154.8	169.0

Before testing, the calculation analysis was carried out.

1. Description of Computer Codes and Related Nuclear Data

The three dimensional static analysis of each step of first loading was conducted with four code systems: ERANOS, CITATION¹⁾, HND and MCNP in which HND is a coarse mesh nodal diffusion code.

The Nvitamin-C file is used for the fine (171 groups) cross section data bank used in the core neutronics calculation by CITATION and HND codes, and incorporated with cross section processing code system PASC-1. ERANOS is an European fast reactor neutronics calculation code package matched with cross section data file JEF 2.0. MCNP has its own point cross section data bank. The cross section data used for physics criticality calculation are mainly from the ENDF/B-V bank.

The Nvitamin-C 171 group cross section bank was re-made up by the nuclear data center of the China

Institute of Atomic Energy, following the original Vitamin-C format and using rather new data (ENDF/B-VI, JEF-2, CENDL-2 and JENDL-3). In comparison to Vitamin-C Bank, besides having new data, the nuclide number has been increased from 66 to 105 in which single group false fission products of U-235 and Pu-239 are included. Based on approximately adopting the Bondarenko format for narrow resonance the resonance self-shielding is modified, to serve mainly for shielding and other calculations for the fast reactor or fusion reactor.

The micro cross section data bank of CITATION and HND is made up by PASC-1 code system with few groups bank. Even though that of CITATION could be made up by several codes, XSDRN²⁾ was designed specially for this purpose and it is also a one dimensional SN transport calculation module. PASC-1 code system, which has a function similar to AMPX, is a code package by which the main format cross section bank of neutron fine groups is combined to a wide or few groups format cross section bank for core diffusion calculation by CITATION. The PASC-1 code system flow is shown in Fig. 4.³⁾

2. Calculation Results

Prior to the actual loading of fuel subassembly, the critical calculation of each batch of loading by the above mentioned four code systems was completed to give effective multiplication factors and critical number of subassemblies. The calculation results are given in Table 3 in which all core effective multiplication factors have been corrected based on the calculation results of CEFR first loading benchmark facility BFS 831 (IPPE, Russia). The first critical subassembly number is predicted between 71 and 73 by the calculations.

V. Physics Start-up Experiments

As shown in Table 4, CEFR physics start-up experiments include four categories: loading and critical tests, control rod worth measurements,

Fig. 4 PASC-1 code system flow

reactivity effect measurements, and activation measurements. Three criticality states are considered: net core criticality (minimum criticality), cold criticality and hot criticality for operation loading.

The first criticality involves just the net core in which the temperature is stabilized at $250^{\circ}\text{C} \pm 5^{\circ}\text{C}$ and almost all control rods are out of the reactor core. Fuel subassemblies are replaced from dummy ones in a batch wise manner to reach criticality using the extrapolated method with the reciprocal of the neutron flux counts to obtain critical mass as shown Fig. 5.

When loading up to 72 fuel subassemblies, the reactor reaches criticality and the excess reactivity of the core is determined by the calibrated differential worth of the control rods. The criticality tests for operation loading showed that the number of core loading is 79 fuel subassemblies for the rated operations including power ascension from cold to hot states. Related parameters are listed in Table 5.

Table 3 CEFR first loading criticality calculations

Load batch	Core S.A. number	Effective multiplication factors				Extrapolated critical number of subassemblies			
		ERANOS	CITATION	HND	MCNP	ERANOS	CITATION	HND	MCNP
1	24	0.65133	0.66858	0.67167	0.69139	—	—	—	—
2	40	0.81749	0.82594	0.82991	0.84316	36.8	57.7	57.2	56.5
3	48	0.88044	0.88227	0.88635	0.89820	57.6	64.7	64.1	62.6
4	56	0.92831	0.92669	0.93066	0.94085	63.2	69.2	68.5	67.3
5	62	0.95644	0.95327	0.95718	0.96494	68.0	72.5	71.7	70.5
6	66	0.97466	0.96994	0.97390	0.98024	71.3	73.2	72.2	71.5
7	69	0.99002	0.98422	0.98822	0.99305	71.6	72.3	71.5	70.3
8	70	0.99501	0.98883	0.99281	0.99784	70.9	72.4	71.6	70.5
9	71	0.99985	0.99333	0.99726	1.00190	71.0	72.5	71.6	70.5
10	72	1.00495	0.99803	1.00199	1.00684	71.0	72.4	71.6	70.6
11	73	1.01010	1.00281	1.00676	1.01151	71.0	72.4	71.6	70.5

Table 4 CEFR physics start up experiments

Loading and criticality	Control rod worth	Reactivity coefficient	Activation of foils
1. Loading and first criticality	1. Scaling at first criticality	1. Pressure reactivity coefficient	1. Nuclear reaction rate relative distribution profiles
2. Initial core : Operation loading cold criticality	2. Operation loading cold state	2. Operation loading cold state	2. Flowrate reactivity coefficient
3. Nuclear heat starting	3. Operation loading hot state	3. Sodium void reactivity coefficient	3. Neutron spectrum measurements
4. Operation loading hot criticality		4. Fuel subassembly exchange reactivity	4. Absolute nuclear power determined by reaction rates with foils in relation to the neutron count rates of detectors
		5. Temperature reactivity coefficient

Fig. 5 Extrapolated method for first criticality**Table 5** Criticality testing parameters

Parameter	Net core (Minimum critical core)	Initial core Operation cold	Initial core Operation hot
Core state			
Fuel S.A./No.	72	79	79
Primary Na (°C)	245	245	302
Primary Ar pressure (kPa)	5.2	5.3	49.7
Primary Na level (mm)	4,344	4,345	4,421
Primary pump speed (rpm)	151	151	901
Critical rod elevation (mm)			
1# compensation (SH 1)	542	283	299
2# compensation (SH 2)	542	282	299
3# compensation (SH 3)	543	284	299
1# regulation (RE 1)	542	272	255
2# regulation (RE 2)	122	273	256
Double period (s)	>500	>500	>500

VI. Summary

The first criticality of the CEFR has been reached and results were consistent with the calculated results obtained by four different neutronics code systems which are being used now for the design of the China demonstration fast reactor CFR-1000. Some important safety related parameters, *e.g.* control rod worth, temperature coefficient, sodium void effect, fuel

subassembly exchange reactivity, inertia whirl properties of primary and secondary pumps, etc. have been demonstrated in experiments at the initial core following the first criticality.

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References

- 1) T. B. Fowler, D. R. Vondy, G. W. Cunningham, *Nuclear Reactor Core Analysis Code: CITATION*, ORNL-TM-2469, rev. 2, Oak Ridge National Lab., (1971).
- 2) N. M. Greene, *XSDRN: A discrete ordinates spectral averaging code*, ORNL-TM-250, (1969).
- 3) Y. Wang, J. Oppe, J. B. M. de Haas, *et al.*, *The Petten AMPX/SCALE code system PASC-1 for reactor neutronics calculation*, ECN-89-005, (1988).

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