Profile SFR-88

PRESCO

Republic of KOREA

GENERAL INFORMATION

NAME OF THE FACILITY	PRESCO
ACRONYM	Pressure and Core Flow Distribution for PGSFR
COOLANT(S) OF THE FACILITY	Water
LOCATION (address):	Thermal Hydraulics and Severe Accident Research Division, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea
OPERATOR	KAERI
CONTACT PERSON (name, address, institute, function, telephone, email):	Dong-Jin Euh, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea, KAERI, Thermal Hydraulics and Severe Accident Research Division, Tel. +82 42 868 2594, djeuh@kaeri.re.kr

STATUS OF THE FACILITY	In operation
Start of operation (date):	2017

MAIN RESEARCH FIELD(S)

☐ Zero power facility for V&V and licensing purposes
☐ Design Basis Accidents (DBA) and Design Extended Conditions (DEC)
☒ Thermal-hydraulics
☐ Coolant chemistry
☐ Materials
☒ Systems and components
☐ Instrumentation & ISI&R

TECHNICAL DESCRIPTION

Description of the facility

To simulate the hydrodynamic behaviour inside the reactor vessel of the PGSFR, a test facility, referred to here as PRESCO (Pressure and Core Flow Distribution for PGSFR), was constructed. The PRESCO test section was designed based on the concept of Euler number conservation. The Euler number refers to the ratio of the pressure drop to the dynamic force, which is also the pressure drop coefficient.

Table 1 summarizes the scaling ratio of major design parameters applied to the PRESCO test section. The reactor vessel and major in-vessel structures for the primary heat transport system of the PGSFR were linearly scaled down to a ratio of 1/5 in the PRESCO test section. The PRESCO facility is operated at 60℃ and under atmospheric pressure (under 3 bar) using water as a working fluid. Referring to the normal operation condition of the PGSFR, the ratios of the viscosity and density of the water flow to the sodium flow are estimated to be 1/0.54 and 1/0.85, respectively. To achieve a sufficiently large Reynolds number for the flow condition, the ratio of the velocity is determined to be 1/2, which corresponds to a Reynolds number ratio of 1/16. To simulate the pool behaviour, the Froude number, which is defined as the ratio of the flow inertia to the gravitational force, should be preserved. In the PRESCO test section, the hot pool level was considered as the characteristic length and was downscaled to a ratio of 1/5. From the Froude number conservation concept, the velocity ratio can be estimated to be $1/\sqrt{5}$, which is very close to 1/2. The flow path inside the fuel assemblies and IHXs (intermediate heat exchangers) were not directly preserved due to their complex flow geometry with tube bundles. The fuel assemblies and IHXs were restructured as simulators using a simplified flow path. The roles of the simulators are to preserve the pressure drop characteristics and to measure the mass flow rate through each simulator. The pressure drop ratio based on Euler number conservation can be estimated to be 1/3.42, which was considered in the design of the internal flow paths of the fuel assembly simulator and IHX simulator.

The reactor vessel and inner structures of the PRESCO test section are linearly scaled copies of those of the PGSFR reactor. The reactor simulator preserves the flow geometry along the major flow paths, except for the core and IHXs, which include the inlet plenum, core shroud, redan, pump inventory, IVTM, DHX, PSPS, and FTP with a length scale of 1/5 of the PGSFR. Because the present test focuses on the hydraulic behaviour of the plant, the facility was designed for low pressure and temperature conditions. Therefore, the thickness of the reactor vessel outer wall was not preserved, while the thickness of the internal structure is reduced by the scaling ratio applied to the overall design of the facility such that the internal volume and flow area can be preserved.

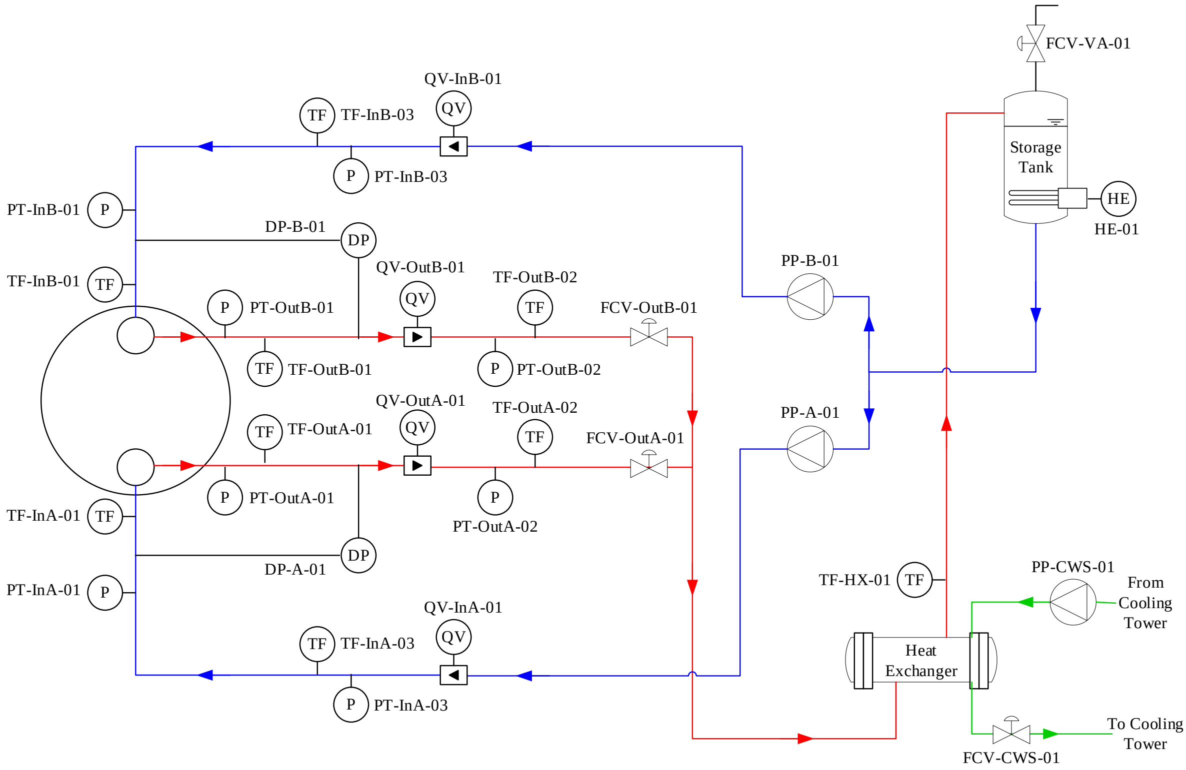
Table 1 Specifications of PRESCO Test Facility

	PGSFR	Test Model	Ratio (Test Model/PGSFR)	Remark
Coolant	Sodium	Water	-	
Temp. (℃)	467.5	60	-	
Press. (MPa)	0.1	0.1	-	
Density (kg/m3)	840.2	983.2	1.17	
Viscosity (Ns/m2)	2.48×10^{-4}	4.66×10^{-4}	1.88	
Length (Pool Dia., mm)	8654	1731	1/5	
Velocity (m/s)			1/2	Nominal Condition
Re (-)	-	-	1/16	
Eu (-)	-	-	1/1	-
Fr (-)	-	-	~1/1	Preserved if $V_R=1/\sqrt{5}$

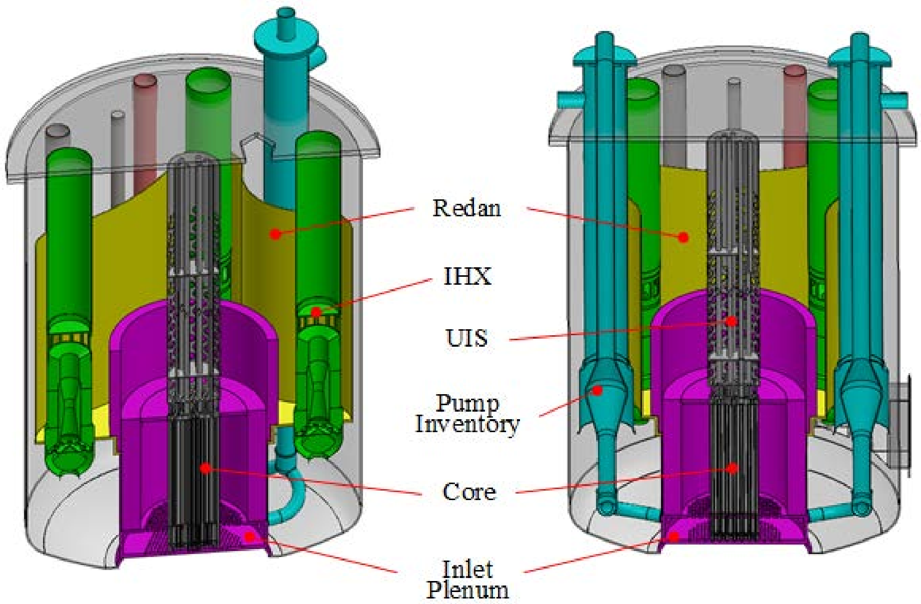
Acceptance of radioactive material

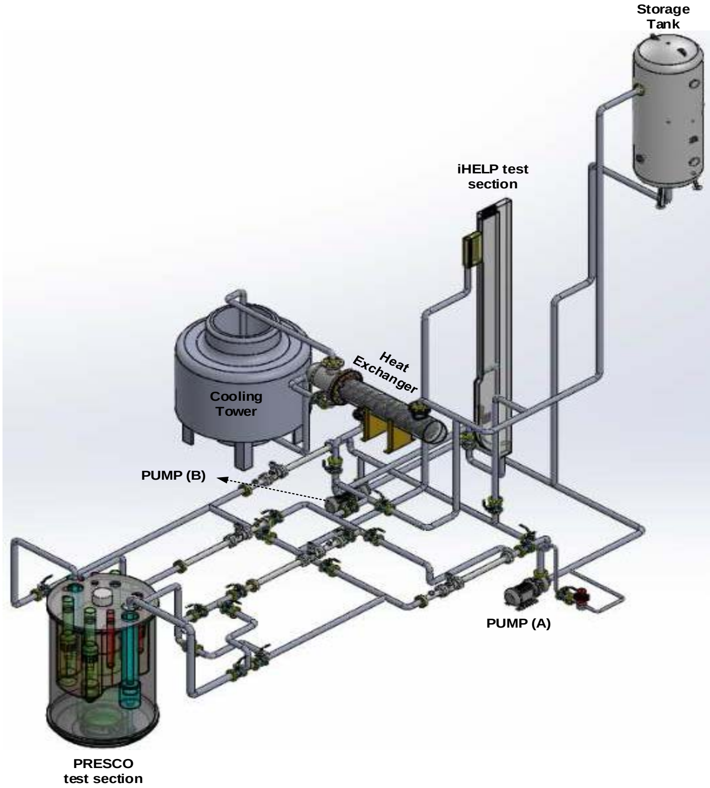
No

Scheme/diagram



3D drawing/photo





Parameters table

Coolant inventory	9 m^3 at maximum
Power	120 kW
Test sections	
TS #1	*Characteristic dimensions* Approximately, 2 m (diameter), 3 m (height)
	Static/dynamic experiment Static
	Temperature range in the test section (Delta T) 60 ℃
	Operating pressure and design pressure Operating Pressure: 101.3 kPa ~ 400 kPa Design pressure: ~0.5 MPa
	Flow range (mass, velocity, etc.) 13.9 ~ 49.5 kg/s
Coolant chemistry measurement and control (active or not, measured parameters)	Not measured
Instrumentation	RTD, Vortex flow meter, Pressure transducer

COMPLETED EXPERIMENTAL CAMPAIGNS: MAIN RESULTS AND ACHIEVEMENTS

The current test was performed to measure the parameters as summarized below.

1) Each fuel assembly simulator flow rate (core flow distribution)

2) Each fuel assembly simulator outlet pressure (core outlet pressure distribution)

3) Inlet plenum pressure distribution

4) Each IHX simulator flow rate

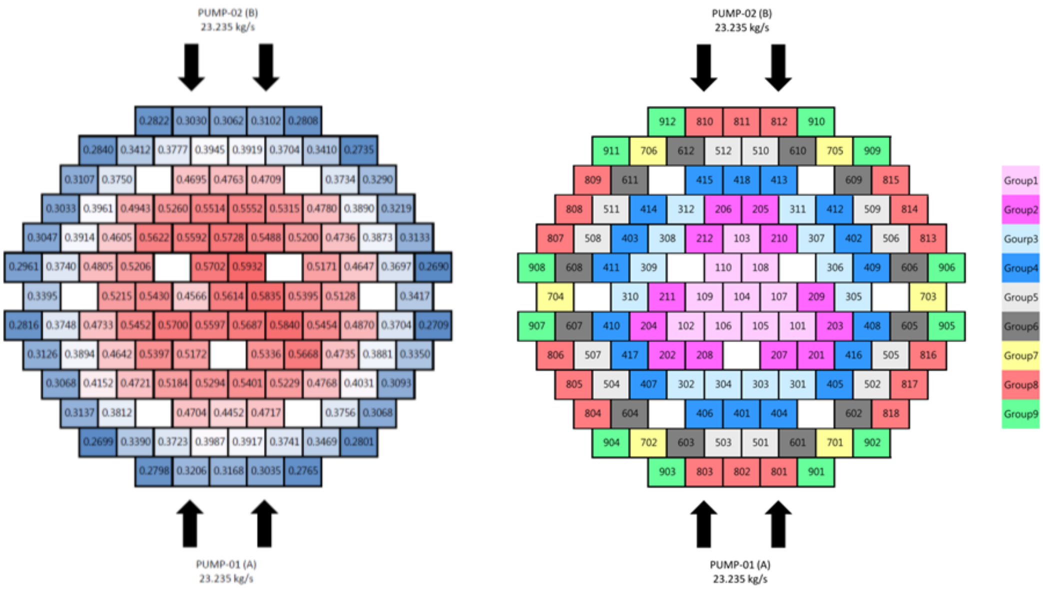
5) Sectional pressure drops between the inlet and outlet

6) Reactor inlet flow

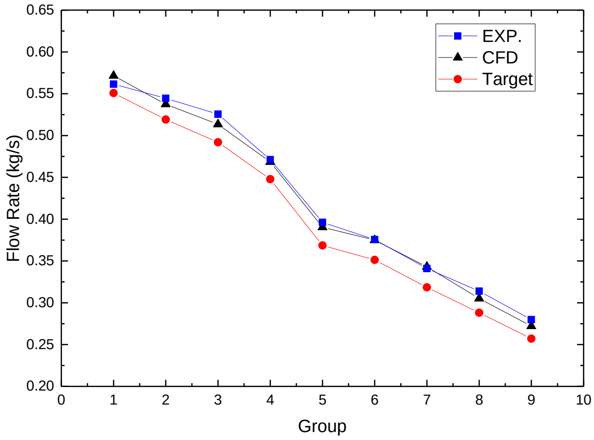
7) Reactor outlet flow

The required test matrix consists of (1) the normal operation (100% flow rate) condition, (2) the 30% flow rate condition, (3) the 50% flow rate condition, (4) the 80% flow rate condition, (5) the 106.5% flow rate condition, (6) the 1 pump nominal condition and 1 pump 90% operation condition, and (7) the 1 pump fail and 1 pump normal operation condition.

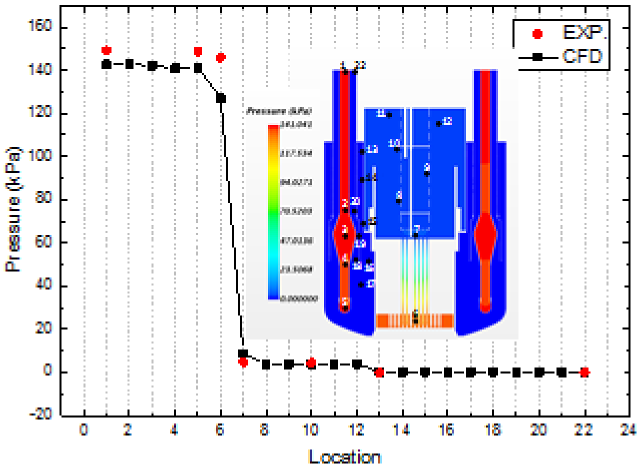
The flow distribution of the PGSFR was identified by an ensemble-averaging process using 15 or 9 independent data tests. The data was produced based on the data requirement for the test parameters, the required uncertainty, and the allowable degrees of data scatter. The results showed a good balance between the mass flow rates and the pressure drops. The core flow distribution showed reasonable results compared to the desired values. The core exit pressure was found to have an even distribution. The flow distribution and hydraulic resistance can be utilized as key input data for safety analyses of the PGSFR core thermal margin and plant transient conditions and for evaluations of the soundness and hydraulic performance of the reactor design. These outcomes can also serve as benchmarking data for a CFD analysis and during the physical model development process considering the complex geometry.



Core flow rate distribution (normal operation condition)



Group averaged mass flow rate (normal operation condition)



Pressure distribution along major flow path

PLANNED EXPERIMENTS (including time schedule)

Visualization of flow inside the reactor vessel using PIV technique is planned to be completed in December, 2018.

TRAINING ACTIVITIES

REFERENCES (*specification of availability and language*)

- Kim, W.S., Hong, S.H., Choi, H.S., Chang, S.K., Euh, D.J., "Experimental Study on Core Flow Distribution inside PGSFR Reactor Vessel under Asymmetric Inlet Flow Conditions", The Eleventh Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety, Busan, Korea, November 18-21, 2018 (En)
- Kim, W.S., Chang, S.K., Euh, D.J., "CFD Calculation for Preliminary Analysis of SFR Reactor Flow Distribution Test", CFD4NRS-7, Shanghai, China, September 4-6, 2018 (En)
- Kim, W.S., Hong, S.H., Seol, H.S., Kim, J.W., Choi, H.S., Chu, I.C., Chang, S.K., Euh, D.J., "Design of Fuel Assembly Simulator for Reactor Flow Distribution Test Facility for Prototype Gen-IV Sodium Cooled Fast Reactor", The Ninth JSME-KSME Thermal and Fluids Engineering Conference, Okinawa, Japan, October 28-30, 2017 (En)

- Euh, D.J., Kim, W.S., Choi, H.S., Chang, S.K., Chung, H.J., “Experimental Methods for Flow Identification of the PGSFR Primary Heat Transfer System”, The Twelfth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Operation and Safety, Qingdao, China, October 14-18, 2018 (En)
- Kim, W.S., Chang, S.K., Youn, Y.J., Park, J.K., Choi, H.S., Kim, J.U., Hong, S.H., Seol, H.S., Euh, D.J., “Construction Report of SFR Reactor Flow Distribution Test Facility,” Korea Atomic Energy Research Institute, KAERI/TR-6950/2017 (Kr)
- Kim, W.S., Chang, S.K., Youn, Y.J., Park, J.K., Choi, H.S., Kim, J.U., Hong, S.H., Seol, H.S., Euh, D.J., “Report of SFR Reactor Flow Distribution Test,” Korea Atomic Energy Research Institute, KAERI/TR-6952/2017 (Kr)