Profile SFR-48

CCTL

JAPAN

GENERAL INFORMATION

NAME OF THE FACILITY	Core Component Thermal-hydraulic Test Loop
ACRONYM	CCTL
COOLANT(S) OF THE FACILITY	Sodium
LOCATION (address):	Oarai Research and Development Institute, Japan Atomic Energy Agency (JAEA), 4002 Narita, Oarai-machi, Ibaraki-ken, 311-1393, Japan
OPERATOR	JAEA
CONTACT PERSON (name, address, institute, function, telephone, email):	Shuji Ohno, Reactor Systems Design Department Sector of Fast Reactor and Advanced Reactor Research and Development Japan Atomic Energy Agency (JAEA) 4002 Narita, Oarai-machi, Ibaraki-ken, 311-1393, Japan Tel: +81 29 267 4141, email: ohno.shuji@jaea.go.jp

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

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

CCTL (Core Component Thermal-hydraulic Test Loop) is located at O-arai Research and Development Center in JAEA. Thermal-hydraulic characteristics of fuel subassemblies and components in the thermal transportation system have been tested in the sodium loop.

Acceptance of radioactive material

No

Scheme/diagram

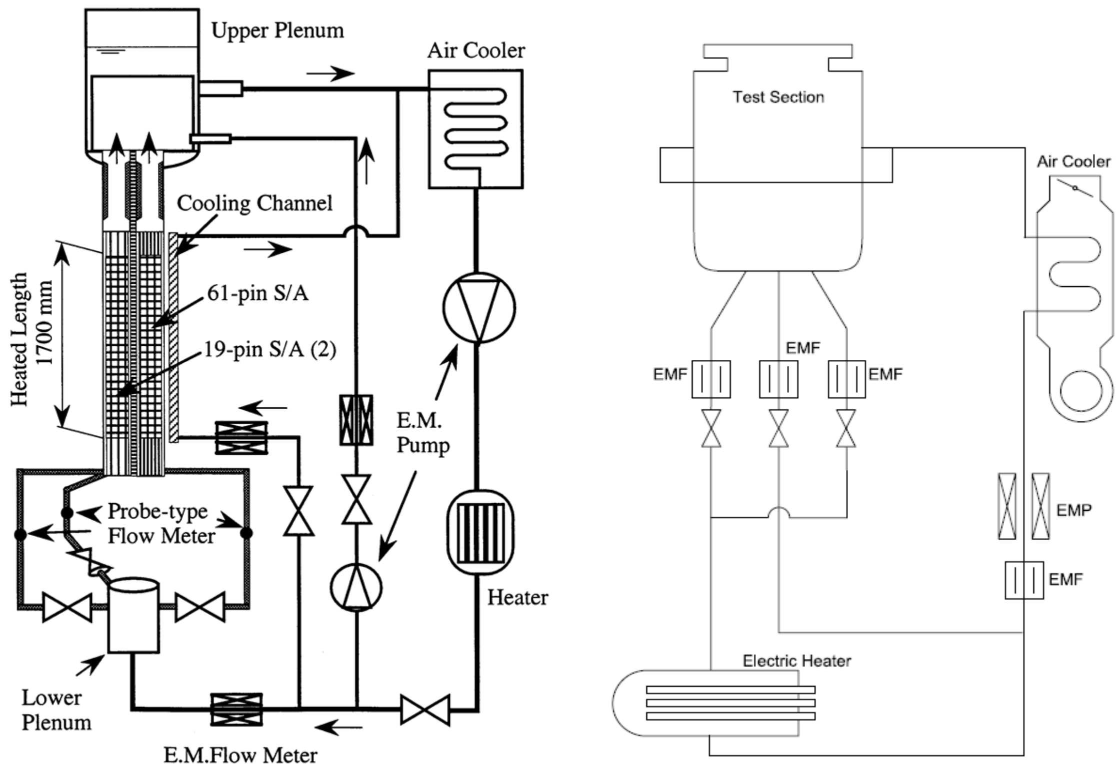

FIG. 1. Scheme of the CCTL facility in Multi-Assemblies test (left) and in in PLAJEST test

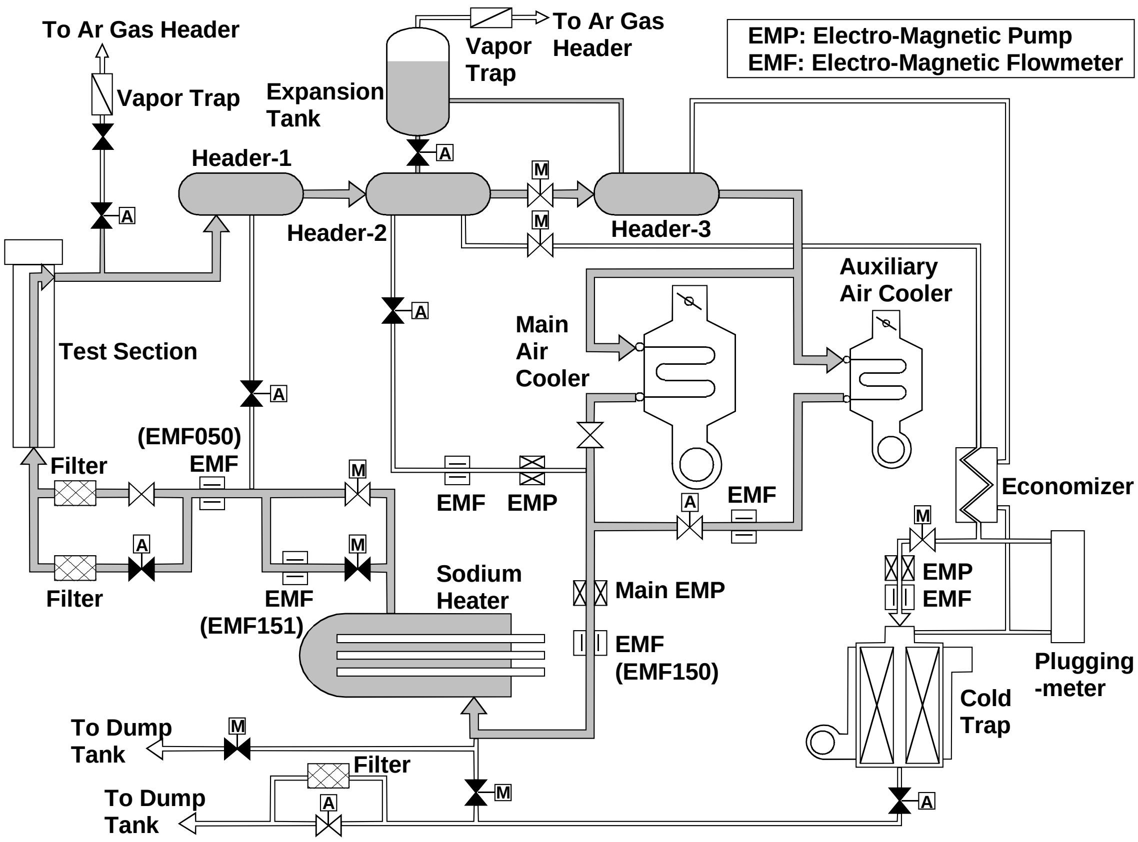

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FIG. 2. Flow Diagrams of Core Component Teat Loop (CCTL). Component layout in 37 pin-bundle test.

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3D drawing/photo

FIG. 3. Panoramic view of CCTL apparatus

Parameters table

Coolant inventory	Depending on volume of the test section (Capacity of dump tank: Approx. 25 tons)
Power	Electric heater, Max. 1MW
Test sections	
TS #1	*Characteristic dimensions* Depending on geometry of the test section (1mm ~ 1m) (Inner diameter of main pipe: 0.1 m)
	Static/dynamic experiment Mainly static experiment
	Temperature range in the test section (Delta T) Maximum temperature: 625 (^{o}C) Max. power of sodium heater: 1 (MW) Delta T is depending on the test conditions (250 (^{o}C) is pre-heated temperature of piping)
	Operating pressure and design pressure Maximum pressure:0.8 MPa
	Flow range (mass, velocity, etc.) Max. 600 (L/min)
Coolant chemistry measurement and control (active or not, measured parameters)	None
Instrumentation	Thermocouples, pressure transducer, flow meter

COMPLETED EXPERIMENTAL CAMPAIGNS: MAIN RESULTS AND ACHIEVEMENTS

1. Inter-Subassembly Heat Transfer during Natural Circulation

Temperature distributions in fuel subassemblies of fast reactors interactively affect heat transfer from center to outer region of the core (inter-subassembly heat transfer) and cooling capability of an inter-wrapper flow, as well as maximum cladding temperature. The prediction of temperature distribution in the subassembly is, therefore one of the important issues for the reactor safety assessment. In the experiment, the simulated core consists of three subassemblies: a 61 pin bundle and two 19-pin bundles. The fuel pins are simulated by electrical heater rods. The inter-subassembly heat transfer occurs particularly in the radial blanket regions of the core due to a steep radial temperature distribution across the driver core and radial reflector regions. Hence the 61-pin bundle simulates a blanket subassembly. Thus, steady-state sodium experiments were carried out using a three-subassembly model. The transverse temperature distributions in the subassemblies were measured under conditions wherein inter-subassembly heat transfer occurred. Experimental analyses were also carried out using a three-dimensional analysis code that modelled the multi-bundle system (a1, a2, a3).

2. Sodium experiment of parallel triple jets

Sodium experiments of parallel triple jets configuration was performed in order to evaluate the temperature fluctuation characteristics in fluid and the characteristics of temperature fluctuation transferred from fluid to structure. As for the experimental geometry, a cold jet on center and hot jets on both sides flowed vertically and along the wall. The temperature fluctuation intensity near the wall decreased as the discharged velocity of the jet decreased. The power spectrum density of the temperature fluctuation could be evaluated using Strouhal number based on the discharged velocities of the jets. In this experiment, the temperature fluctuation characteristics related to the thermal striping phenomena were well investigated. (b1, b2, b3, b4)

3. Sodium experiments on a porous blockage in a fuel subassembly.

The 37-pin bundle sodium experiment for porous local blockage was conducted to investigate the thermal-hydraulic field in the pin bundle with porous blockage and to obtain a verification data for numerical analysis. The test section simulated a fuel sub-assembly of 37-pin bundle with pin diameter and pin pitch, which was equivalent to the large-scale reactor. The test section was included the porous blockage in 14 sub-channels of 2 rows near the wrapper tube wall. Through the experiment, knowledge of spatial distribution of sodium temperature and the characteristics of the maximum temperature in the porous blockage were obtained. (c1, c2, c3).

PLANNED EXPERIMENTS (including time schedule)

Five jets-configured thermal mixing tests (near future)

TRAINING ACTIVITIES

Training activities have been carried out before the operation of experimental campaign based on a quality management system of JAEA.

REFERENCES (*specification of availability and language*)

a1) H. Kamide, et al., "An Experimental Study of Inter-Subassembly Heat Transfer during Natural Circulation Decay Heat Removal in Fast Breeder Reactors", Nuclear Engineering and Design, 183 (1998), pp. 97-106.

a2) M. Nishimura, et al., "Development of Multi-Dimensional Thermal Hydraulic Modelling Using Mixing Factors for Wire Wrapped Fuel Pin Bundles with Inter-Subassembly Heat Transfer in Fast Reactors", PNC-TN9410 96-289 (1996). [in Japanese]

a3) H. Kamide, et al., "Calculational Method for Inter-Subassembly Heat Transfer during Natural Circulation in Fast Breeder Reactors - Verification based on CCTL and PLANDTL sodium tests -", PNC-TN9410 96-268 (1996). [in Japanese] b1) N. Kimura, et al., "Experimental study on thermal striping; Sodium experiment of parallel triple jets", JNC-TN9400 2001-063 (2001) [in Japanese]

b2) N. Kimura, et al., "Experimental Study on Thermal Striping Phenomena - Evaluation on Transfer Characteristics of Temperature Fluctuation from Fluid to Structure", JNC-TN9400 2002-059 (2002). [in Japanese]

b3) N. Kimura, et al., "Experimental Investigation on Transfer Characteristics of Temperature Fluctuation from Liquid Sodium to Wall in Parallel Triple-Jet", International Journal of Heat and Mass Transfer, 50(9-10) (2007), pp. 2024-2036.

b4) N. Kimura, et al., "Study on Thermal-Hydraulics of Thermal Striping Phenomena - Evaluation of Transfer Characteristics of Temperature Fluctuation based on Conjugated Numerical Simulation in Triple-Parallel Jet Geometry", JAEA-Research 2012-017 (2012). [in Japanese]

c1) J. Kobayashi, et al., "Study on temperature field in porous blockage in a fuel subassembly; 37-pin bundle sodium experiment", JNC-TN9400 2000-025 (2000) [in Japanese]

c2) M. Tanaka, et al., "Study on Temperature Field in Porous Blockage in A Fuel Subassembly (II); 37-Pin Bundle Sodium Experiment", JNC-TN9400 2001-062 (2001). [in Japanese]

c3) M. Tanaka, et al., "Study on Thermal-Hydraulics In A Porous Blockage in A Fuel Subassembly of LMFBR", Proc. of 11th International Conference on Nuclear Engineering (ICONE), Tokyo, JAPAN, April 20-23 (2003) ICONE11-36362.