Profile SFR-50

PLANDTL

JAPAN

GENERAL INFORMATION

NAME OF THE FACILITY	Plant Dynamics Test Loop
ACRONYM	PLANDTL
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	In operation
Start of operation (date):	Choose an item.

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

The PLANDTL facility simulates the primary, the secondary and decay heat removal (DHR) circuits of sodium cooled fast reactors (SFRs). And those circuits are thermally connected through heat exchangers. The maximum sodium flow rates of primary, secondary and DHR

circuits are 1,200 (L/min), 700 (L/min) and 100 (L/min), respectively. The maximum temperature in the design is 625 ^{o}C and total amount of sodium is around 12 tons. The primary loop is connected to the test section which simulates the thermal-hydraulics behaviour in reactor vessel. Currently, the test section of PLANDTL is modified from the previous one (PLANDTL-DHX), the renewed models (PLANDTL-2) are simulated core and an upper plenum simplifying a loop type reactor. The simulated core consists of 30 heating channels for fuel sub-assemblies and 25 non-heating channels for neutron shields and control rods. Each channel is heated by an electric heater rod. The flow rate ratio between toward heating channels and toward non-heating channels is adjustable within the capacity of the flow control. The upper plenum contains the upper inner structure (UIS), a dipped type direct heat exchanger (DHX) and top entry type hot leg piping. The heat generated in the test core is transferred to the secondary circuit and/or the DHR circuit through the heat exchangers (HXs), and is cooled by air-coolers and released to atmosphere. Thus, tests on decay heat removal systems are applicable in PLANDTL-2. Measurements are made by thermocouples and electromagnetic flowmeter.

Acceptance of radioactive material

No

Scheme/diagram

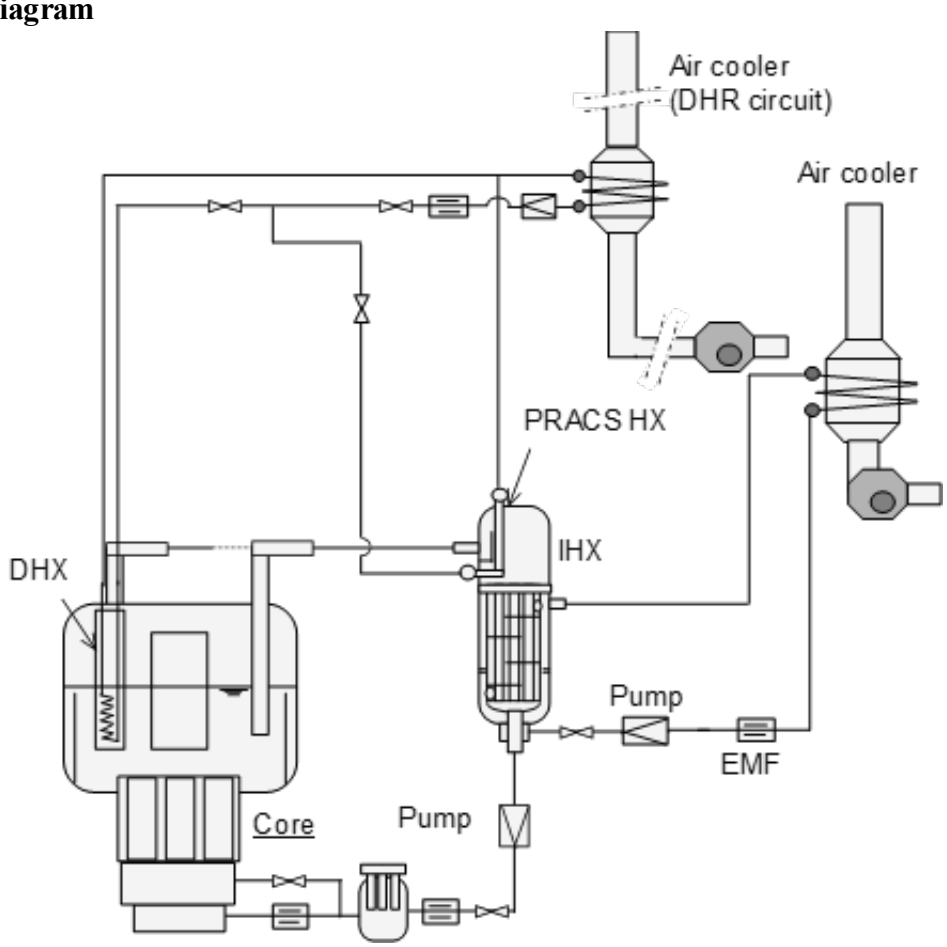

FIG. 1. Schematic view of components layout of PLANDTL facility

3D drawing/photo

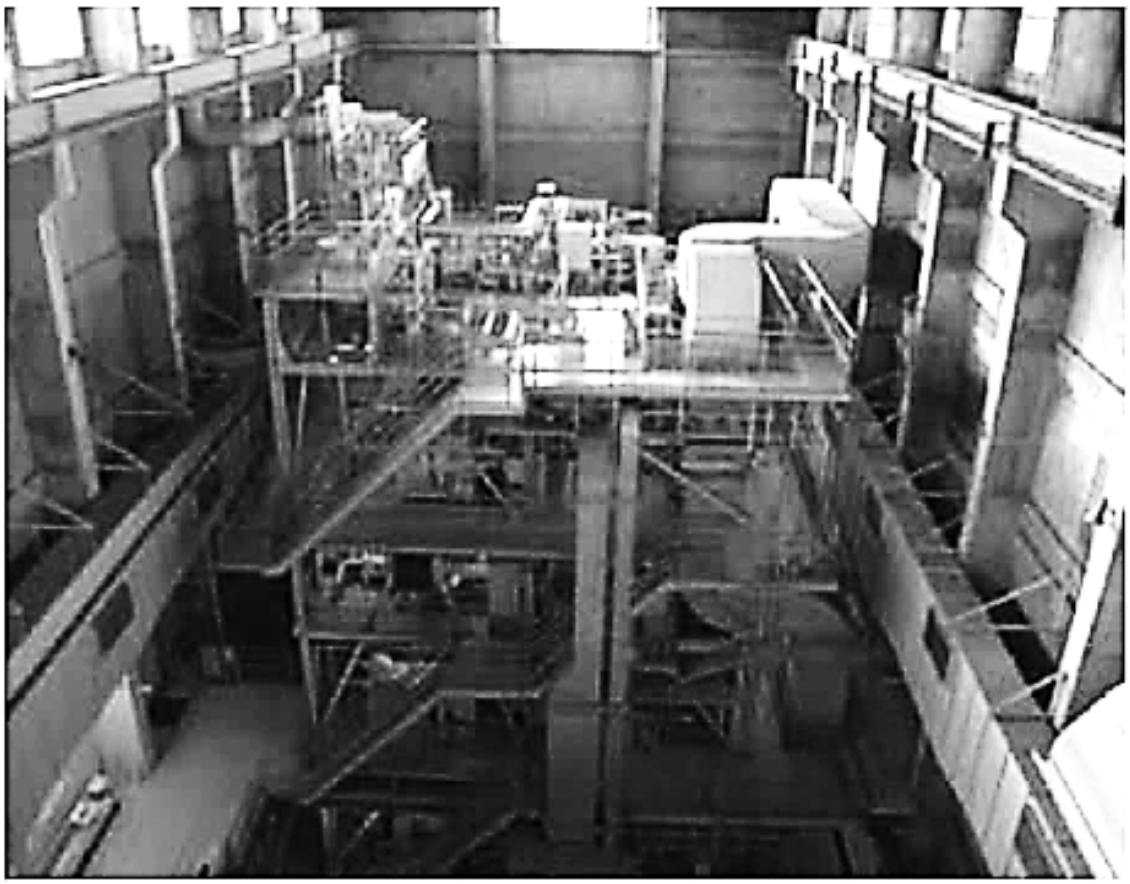

FIG. 2. Photo of panoramic view of PLANDTL apparatus

Parameters table

Coolant inventory	12 ton in loop (Capacity of dump tank: Approx. 25 tons)
Power	Approx. 1 MW
Test sections	
TS #1	*Characteristic dimensions* Depending on geometry of the test section (1mm ~ 1m) (at present) - Core length: 0.9 m (active heater: ~ 0.3 m) - Inner diameter of upper plenum: ~2m (Main pipe diameter: 0.1 m (4 inches))
	Static/dynamic experiment Both conditions
	Temperature range in the test section (Delta T) Maximum temperature: 625 °C Delta T is depending on the test conditions (at present) - Core: 150 °C (inlet/outlet) (Pre-heated temperature of piping is 250 °C)
	Operating pressure and design pressure Operating pressure: 0.03MPa Maximum pressure:0.8 MPa
	Flow range (mass, velocity, etc.) Maximum flow rate: 1,200 L/min in primary loop
Coolant chemistry measurement and control (active or not, measured parameters)	None
Instrumentation	Thermocouples, pressure transducer, flowmeter.

COMPLETED EXPERIMENTAL CAMPAIGNS: MAIN RESULTS AND ACHIEVEMENTS

The PLANDTL-DHX test facility has two decay heat removal systems of DRACS (Direct Reactor Auxiliary Cooling System) and PRACS (Primary Reactor Auxiliary Cooling System): the DRACS consists of direct heat exchangers (DHXs) in the upper plenum of the reactor vessel and the PRACS consists of primary heat exchangers (PHXs) in the upper plenum of the IHX. The core model has seven subassemblies consisting of the center assembly with 37 pins simulating a core fuel subassembly of a large scale SFRs and six outer subassemblies with 7 pins to set up the boundary conditions of the center assembly. The steady state experiment and also the transient experiment which simulates the transition of the heat transfer system in a reactor from forced circulation condition to natural circulation condition can be performed. When the cold sodium provided by the DHXs penetrates into inter-wrapper gaps in the core, the thermal-hydraulic phenomenon called as inter-wrapper flow (IWF) can occur and it may affect the thermal hydraulic behaviour in the core. In the

experiments by using PLANDTL-DHX test facility, the thermal-hydraulic behaviour and the mutual interaction in the primary and the secondary heat transfer systems and the decay heat removal systems can be well investigated in heat removal operation including natural circulation condition of SFRs. Also, influence of decay heat removal systems on the thermal hydraulics in the core can be investigated.

PLANNED EXPERIMENTS (including time schedule)

Experiments relating to the sever accident and DHRS are on-going in PLANDTL-2 geometry.

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*)

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2) K. Hayashi, "Investigation of interaction between heat transport systems during the natural circulation decay heat removal in FBRs; Influence of decay heat removal system type and the Secondary heat transport system", PNC-TN9410-97-045 (1997). [in Japanese]

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4) M. Nishimura, et al., "Transient experiments on fast reactor core thermal-hydraulics and its numerical analysis - inter-subassembly heat transfer and inter-wrapper flow under natural circulation conditions", Nuclear Engineering and Design, 200 (2000), pp.157 - 175.

5) H. Kamide, et al., "Investigation of Core Thermo-hydraulics in Fast Reactors - Inter-Wrapper Flow During Natural Circulation -", Nuclear Technology, 133 (2001) pp. 77 - 91.

6) H. Kamide, et al., "Sodium experiments of buoyancy-driven penetration flow into low-power subassemblies in a sodium-cooled fast reactor during natural circulation decay heat removal", Nuclear Technology, 175(3) (2011), pp.628 - 640

7) A. Ono, et al., "An experimental study on natural circulation decay heat removal system for a loop type fast reactor", Journal of Nuclear Science and Technology, Vol. 53, No. 9 (2016), pp. 1385-1396.

8) T.Ezure, et al., "Study on Multi-Dimensional Core Cooling Behavior of Sodium Cooled Fast Reactor under DRACS Operating Condition", Proceedings of 18th Topical Meeting

on Nuclear Reactor Thermal Hydraulics (NURETH18), Portland, USA, October 18-22 (2019), 450-28311. [USB Flash Drive]