You are required to sign up for the technical tour at the time of on-site registration on July 17, 2023 at the conference venue. Seats are limited. First come, first served.

西安交通大学建成的快堆蒸汽发生器的综合试验装置 PUSA，是我国首个且是唯一能够从事钠冷快堆蒸汽发生器综合性能考核验证的大型试验装置，采用电加热模拟核反应堆热源，总功率 30 MW，技术参数国际领先。该装置填补了国内钠冷快堆大型自主化涉钠设备综合试验平台的空白，全面验证了示范快堆蒸汽发生器性能满足设计及运行需求，为我国首台完全自主研发的钠-水蒸汽发生器按期完成制造出厂并应用于示范快堆提供了重要参考依据，对我国发展先进核电和核动力，提高核威慑能力具有重大意义。

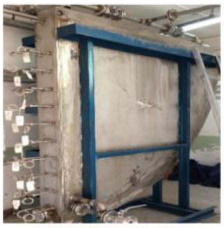
The comprehensive test facility for fast reactor steam generator, PUSA, built by Xi 'an Jiaotong University, is the first and only large-scale test facility capable of comprehensive performance assessment and validation of sodium-cooled fast reactor steam generators in China. It adopts electric heating to simulate nuclear reactor heat source, with a total power of 30 MW and international leading technical parameters. The facility fills the gap in the comprehensive test platform of autonomous large-scale sodium-related devices of SFRs in China, successfully verifies that the performance of the demonstration fast reactor steam generator meets the design and operation requirements, and provides a significant reference for China's first fully self-developed sodium-water steam generator to complete the manufacturing and application in the demonstration fast reactor on schedule. It is of great significance for the further development of advanced nuclear power in China.

钠-水蒸发器（全尺寸等温等压，高 22 m）

西安交通大学核反应堆热工水力研究室（NuTHeL）自主设计并搭建了大型先进压水堆复杂系统集成效应、核心设备分离效应、单元部件现象效应全系列测试验证装置，包括非能动余热排出系统机理试验平台、大型先进压水堆 ADS-4 喷放泄压系统平台、堆芯下管座异物过滤特性测试平台、ECC 安注特性系统平台、压力容器下封头熔融池验证装置、稳压器水封验证装置、稳压器波动管 CCFL 特性试验平台、蒸汽发生器管束流动换热试验平台、安全壳穹顶冷凝验证平台、安全壳通风冷凝验证平台、安全壳喷淋现象验证平台、高温高压单棒 CHF 试验平台、螺旋管流动换热特性试验平台等。

Nuclear Thermal-hydraulic Laboratory (NuTHeL) of Xi'an Jiaotong University has independently designed and built a full range of testing and validation devices of large advanced pressurized water reactors for the integration effect of complex systems, the separation effect of core equipment, and the phenomenon effect of unit components. It includes the mechanism test platform of passive residual heat removal system, the ADS-4 injection pressure relief system platform of large advanced pressurized water reactor, the foreign material filtration characteristics test platform of the lower core header, the system platform of ECC safety injection characteristic, the validation device of the corium pool in the lower head of pressure vessel, the validation device of Voltage regulator water seal, the test platform of CCFL characteristic in the pressurizer surge-line, the test platform of flow heat transfer in steam generator tube bundles, the validation platform of containment dome condensation, the validation platform of containment ventilation condensation, the validation platform of containment spray phenomenon, the test platform on CHF of single rod at high temperature and high pressure and the test platform of flow heat transfer characteristics of helical tube, etc.

大型先进压水堆ADS-4喷放泄压系统平台



下封头熔融池验证装置

安全壳穹顶冷凝验证平台

安全壳通风冷凝验证平台

压水堆非能动余热排出系统机理试验平台

稳压器水封验证装置

安全壳喷淋现象验证平台

西安交通大学核反应堆热工水力研究室（NuTHeL）面向先进核动力系统热工流体设计和安全分析验证的重大亟需，建成了核反应堆严重事故综合试验基地、钠冷快堆、铅铋堆、熔盐堆、海洋小堆等先进核动力系统热工安全验证平台，包括六自由度高温高压对称双环路自然循环系统平台、氟盐冷却高温堆螺旋十字型燃料组件热工水力特性试验平台、大型铅铋流动换热综合试验平台、全尺寸快堆组件外套管变形特性试验平台、铅铋冷却剂流动换热及流动凝固机理试验平台、铅基蒸发器传热管破裂事故热工水力试验平台、铅基反应堆严重事故下燃料熔化及熔融物迁徙试验平台、铅基堆堆芯熔融物与冷却剂相互作用试验平台等。

核反应堆严重事故综合试验基地

钠冷快堆、铅铋堆、熔盐堆等先进核动力系统热工安全验证平台

NuTHeL has built a comprehensive test base for severe accidents of nuclear reactors, as well as sodium-cooled fast reactors, lead-bismuth reactors, molten salt reactors, marine small reactors and other advanced nuclear power system thermal safety validation platforms, to meet the urgent need for thermal fluid design and safety analysis and validation of advanced nuclear power systems. It includes the six-degree-of-freedom platform of symmetrical double-loop natural circulation system under high-temperature and high-pressure condition, the test platform of thermal and hydraulic characteristics of spiral-cross fuel in fluorine-salt-cooled high-temperature reactors, the large-scale comprehensive test platform of lead-bismuth flow heat transfer, the test platform for deformation characteristics of full-scale fast reactor assembly outer casing, the mechanism test platform on flow heat transfer and flow solidification of lead-bismuth coolant, the thermal hydraulic test platform of lead-based generator tube rupture accident, the test platform of fuel melting and molten relocation in lead-based reactor under severe accident, the test platform of interaction between molten and coolant in lead-based reactor core etc.