ANNEX 7: NEA NUCLEAR INNOVATION 2050

R&D COOPERATIVE PROJECT PROPOSAL

"Safety and Economy system of Gen IV DHR"

Experimental collaborations for evaluation methods of Decay heat removal in case of long term SBO and also in post accident heat removal for Advanced Reactors

Hideki Kamide

1. Justification of the selection

Decay heat removal (DHR) is one of the most significant issues for the safety of nuclear reactors especially in case of long term SBO and also in post accident heat removal (PAHR) as lessons learned from TEPCO Fukushima Dai-ichi NPP accidents, even for Generation-IV Advanced Reactors, e.g., Sodium cooled Fast Reactor (SFR), Lead cooled Fast Reactor (LFR), Gas cooled Fast Reactors (GFR), Very High Temperature Reactor (VHTR), and Molten Salt Reactor (MSR).

Natural circulation (NC) is one of most effective functions of DHR as a passive safety feature under such conditions of long term SBO and also in severe accidents. Further, NC has several merits to reduce main and auxiliary components (simple system) and also electric capacity of emergency power lines. It results in significant reduction of capital cost and also simple designs with higher safety.

The theme proposed here is linked with NI2050 Selection Criteria on following issues,

- Positive impact on Safety and also Economy as a Top Level Goal,
- Infrastructures of demonstration experiments are developed and used as a Necessary Tool.

2. The issue to tackle and objectives to reach

The themes to be resolved for the natural convection DHR are nearly common in most of reactor types, e.g., parallel channel flow redistribution, combination of convective heat transfer and heat conduction, thermal stratification due to low flow velocity, uncertainty of evaluation for the flow rate and the highest temperature in the core, and validation of analysis code or evaluation methods.

The natural circulation performance strongly depends on space arrangement of cooling heat exchangers and flow paths connecting the heat sources of core, heat exchangers, and heat sink. In a case of the PAHR, fuel debris will be dispersed in a rector vessel and they should be cooled so as to keep them safely inside the reactor vessel. Design of the heat exchanger of DHR systems might be required to cover such severe conditions.

Methodology of evaluation including uncertainties of input, physical models, computation methods, validation experiments, extrapolation from the experiments to a reactor will have some common perspectives across the reactor systems. Further, implementation of appropriate validation experiments is necessarily required in verification and validation (V&V) process of a simulation code especially for licensing of reactors.

The issues to be resolved in this theme and also the measures/tools to use for the R&D are matched with the NI2050 Grand Challenges and Opportunities on following points,

- Win-win approach on safety and economy with simple systems of decay heat removal,
- Passive safety of natural circulation is a significant and challenging issue on regulation and also public acceptance,
- Infrastructures, measurement technologies, and simulation methods used and developed in this theme have significant contributions to education and technology transfer on Gen-IV reactors.

3. What is done/exist already, who is doing what, what are the means (resources and infrastructures)

In the past many of the model experiments were carried out in the world, e.g., RAMONA, NEPTUN, ILONA in Germany, PLANDTL in Japan for SFR. However, it is still significant on the point of V&V to provide uncertainty evaluation of measurements in the experiments.

For LFR, experiments and present knowledge of natural circulation, particularly in a loop configuration, may be considered as satisfactory and this covers also prototypical DBA, DEC, and may be also cover long term operation situations in a passive mode. However, more knowledge is necessary to verify and further assess the natural convention behavior in full scale (or in representative scaled) configurations. Not only the scale but also multi-dimensional effect in a core and also in a large plenum of complex geometry will be of importance.

For GFR, decay heat removal in the mitigation phase of severe accidents is significant issue. The natural circulation is matter of interest for the decay heat removal.

For VHTR, DHR studies and installations do not have the purpose of protecting the core, but rather of demonstrating the protection of the investment (vessel, vessel head and concrete).

In order to resolve such themes of decay heat removal including a degraded core in PAHR, there are several experimental facilities and also experimental plans in the world, as follows of examples:

Reactor types	Facility name *: planning or construction phase	Targeted region	Fluid	Country
SFR	PLANDTL-2	Whole Core	Sodium	Japan
	PHEASANT	Reactor vessel	Water	Japan
	AtheNa-DHRS*	Reactor Vessel	Sodium	Japan
	STELLA-1	Heat exchangers	Sodium	Korea
	STELLA-2*	RV + DHRS	Sodium	Korea

	CHEOPS	Core, Power Conv. System	Sodium	France
	EAGLE-3 (IGR in Kazakhstan)	Core and Inlet Plenum	Molten Fuel and Sodium	Japan
	MELT	Core and Inlet Plenum	Molten steel and Sodium	Japan
SFR and others	NSTF	RV wall cooling and air duct	Air	USA
	PLINIUS-2*	Core Catcher	Molten Fuel	France
GFR/ VHTR	STU Helium Loop	Core and Primary System	He	Slovak
	S-ALLEGRO Loop	Core and Primary System	He	Czech
VHTR	HTTF	Vessel, Stratification of He and Air	He, Nitrogen, Air	USA
MSR	High pressure and temperature loop	Primary system	Water	China
LFR	E-SCAPE	DBA and DEC Thermal-hydraulics	Lead-Bismuth Eutectic	Belgium
	CIRCE	Large pool Thermal-hydraulics	Lead-Bismuth Eutectic	Italy
	NACIE-UP	DBA and DEC Thermal-hydraulics	Lead-Bismuth Eutectic	Italy
	TALL	DBA and DEC Thermal-hydraulics	Lead-Bismuth Eutectic	Sweden
	TELEMAT	Systems and components	Lead-Bismuth Eutectic	Germany
	HELIOS	Thermal hydraulics	Lead-Bismuth Eutectic	Korea
	CLEAR-S	DBA and DEC Thermal-hydraulics	Lead-Bismuth Eutectic	China

There are also several reactor experiments of natural circulation. Examples of SFR, VHTR and LFR are as follows.

Reactor types	Reactor name	Targeted region	Fluid	Country
SFR	EBR-II		Sodium	USA
	FFTF		Sodium	USA
	Phenix		Sodium	France
	Joyo		Sodium	Japan
	Monju (Partial Power)		Sodium	Japan
	CEFR		Sodium	China
VHTR	HTTR	ULOF	He	Japan

LFR	HELIOS	Primary side	LBE	Korea
	NACIE	Natural and gas-injection-assisted circulation	Lead-Bismuth Eutectic	Italy

However, there are limited experiments of NC for PAHR in advanced reactors.

4. What can be done to improve/accelerate through cooperation

Significant issues to be done in this field are

- Development of simulation methods with uncertainty evaluation where local and complex phenomena interact system behaviour via flow and temperature distributions along multiple flow paths of NC,
- Wide range of validation experiments on separate effect and integral effect with uncertainty evaluations of measurements, and also scale effect to simulate reactors. In this frame it is strongly suggested to define a basic set of reliable experimental data for validation of the computer codes. Note that the chosen experiments needs to be complemented with all information needed for the simulations.
- These validation data obtained from the experiments will be useful not only for developers but also regulators. It is welcome to share the experimental program with researchers in the regulation side. Collaboration work with such regulation people on the test program and data to be obtained will contribute high quality data and also common understanding of phenomena between developers and regulators.
- Effective and robust DHR measures by NC for PAHR

Network of experimental facilities can contribute wide range of phenomena to be understood for variety of DHR systems, geometry, and conditions. New test facility plans based on worldwide requirements of code validations and plant designs in each country can provide higher results if it stands on the experimental network. Such experimental network will be able to overcome the difficulty of wide spectrum of phenomena and common idea of uncertainty treatment.

Examples of existing and planned NC experiments for reactor safety are shown below.

Scope of test	Test Facility	Status	Country
Diversity of DHRS including PAHR	PHEASANT	Start of experiments	Japan
Mutual interaction among DHRS, plenum and core	PLANDTL-2	Start of experiments	Japan
Integrated test with various DHRS	AtheNa-DHRS	Planning	Japan
Core, Power Conversion System	CHEOPS	Planning	France
Coolability of ex-vessel cooling system	NSTF	On going	USA
Heat Exchangers	STELLA-1	On going	Korea

Primary system	STELLA-2	Planning	Korea
Sodium Boiling	AR1	On going	Russia
Core channel cooling with gas	HTTF	On going	USA
Core material relocation and cooling in CDA	MELT	On going	Japan
	EAGLE-3	On going	Japan
Debris on Core Catcher	PLINIUS-2	Planning	France

5. Action plan and necessary means (resources and infrastructures)

Validations need simple separate effect experiments, and integral experiments which cover wide range of phenomena, reactor designs of decay heat removal systems, core, components, and reactor vessel. Further, reactor experiments are of significance on reactivity feedback, scale effect, and integral effects, even though the measurements are limited.

Collaboration scheme can be described as a 3D matrix of Field by two axes with depth of Layer. Schematic of scheme is shown in Figure. Field and Layer are listed below.

Field

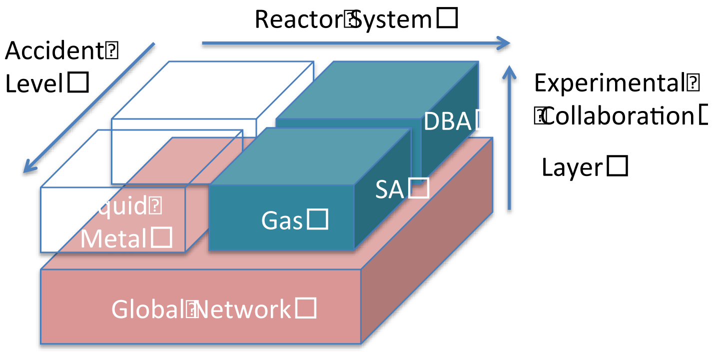
- Reactor systems
  - Liquid Metal: SFR, LFR, ADS, Sodium cooled SMR
  - Gas: HTR, VHTR, GFR
  - Others

- Accident Levels
  - Design base accident and DEC in normal geometry of core
  - Post Accident Heat Removal with degraded core and debris

Layer

- Global network across the reactor systems
  - Measurement technique
  - Measurement accuracy and uncertainty estimation
  - Exchange of PIRT and mutual review
  - Global view of PIRT based on accumulation of PIRT files

- Experimental collaboration in each field
  - Existing Facility and Test section
    - Proposal of New Test Series
  - New Facility or Test Section
    - Proposal of New Facility or Test section and experimental plan



Collaboration Scheme

The Global Network will make progress of crosscut issues across the reactor systems. However, Experimental Collaborations of several reactor systems should be established first and they will have further progress on the global network later. In this sense, a specific proposal of Experimental Collaboration in SFR is shown here as the first step.

1. Specific Program on SFR:

Experimental collaborations for evaluation methods of decay heat removal in case of long term SBO and also in post accident heat removal

- Development of various types of decay heat removal system (DHRS), which can remove the decay heat in case of long term SBO and even in severe accident conditions of damaged core
- Evaluation methods to be applied these issues
- Experiments and code V&V on these issues
  - ✓ Performance of single and combinations of several DHRS under natural circulation, thermal stratification, complex flow paths in reactor vessel, core, core catcher
  - ✓ Demonstration data of the DHRS

Points to be better program

- Wide variations of contributions for easy to join: In-kind: Experimental data, New test section.
- A benchmark on existing test data as the first step of the project

2. Action Plan

Scope:

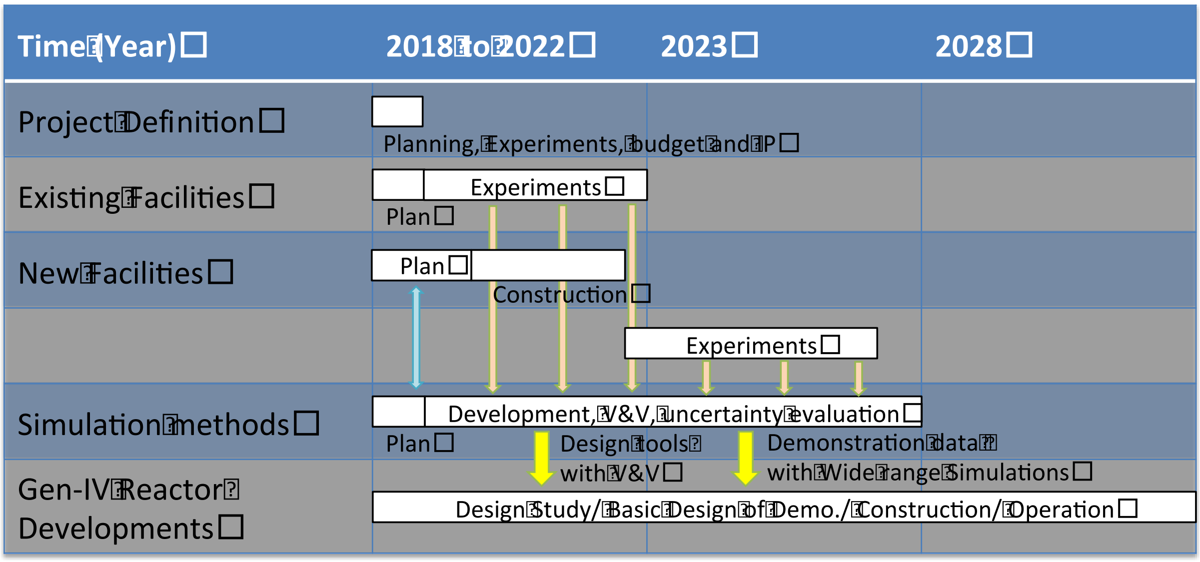
Collaboration of experiments for the code development and validations

Sequence:

- Planning of experimental series and steps using existing and new facilities
- Preliminary analyses and evaluation of DHR performance.
- Construction of test section and loop
- Experiments and Benchmark Analyses
- Dataset and V&V of evaluation methods

Time line

- Planning of experimental series and steps ~ 2019
- Construction of test facilities ~ 2022



3. Necessary Means

Resources

Budget: Depending the experimental plan and new facility

Competence: Available

Infrastructure

Existing:

- NSTF for RV wall cooling by natural draft (Air)
- PLANDTL-2 for core thermal hydraulics with DRACS and NC (Sodium)
- PHEASANT for DHR of damaged core and core catcher (Water)
- Others

Planning:

- STELLA-2 for DHRS of natural circulation and ex-vessel cooling (Sodium)
- AtheNa-RV/DHRS for integrated thermal hydraulics test with various DHRSs (Sodium)
- Others

Proposals of experimental plan and international discussion table can be managed for a wide range of collaboration. It is needed to discuss a collaboration framework to provide infrastructure, to share the experimental results, and to discuss IP rights with financial contributions.

An example of experimental plan for natural circulation and severe accident measure of DHRS is shown in the Annex.

ANNEX: AtheNa-DHRS experimental program

1. Outline of AtheNa-DHRS experimental program

1.1 Objectives
Decay heat removal is one of the most significant functions in reactor safety. The severe accident in Fukushima Dai-ichi nuclear power plants revealed this point. SFR Safety design criteria (SDC) was proposed for sodium cooled fast reactors taking the lesson learned from the SA into account. In the SFR SDC, independency and diversity are emphasized as the key features of the decay heat removal system.

The objectives of this AtheNa-DHRS experimental program are to provide experimental dataset for design and also evaluation methods of such required decay heat removal systems (DHRSs) and to enhance safety feature of SFR.

1.2 Countermeasures for DECs required in SDC
GIF sets safety goals for Generation IV reactor as follows in 2002

SR-1; Operations will excel in safety and reliability
SR-2; A very low likelihood and degree of reactor core damage
SR-3; Eliminate the need for offsite emergency response
*SR: Safety and Reliability and its related goals

A focal point of these safety goals is "Eliminate the need for offsite emergency response." To achieve this goal, Generation IV reactor shall take countermeasures even under Design Extension Conditions (DECs) "to prevent significant core damage" and "to mitigate consequence of core damage to practicably prevent radioactive materials release which calls on off-site emergency response". These requirements will be explicitly described in SDC.

DECs and related countermeasures against external events such as earthquake and tsunami should be taken since Fukushima Dai-ichi NPPs accident led to core damage due to loss of cooling, which was caused by earthquake and severe tsunami with subsequent long term total black out.
SDC for Generation IV reactor will require enhancement of countermeasures for DECs and in order to justify SDC, it is required to show effectiveness of prevention and mitigation of core damage.

1.3 Candidates of countermeasure for LOHRS
Several concepts concerning decay heat removal under severe conditions including degradation of core are shown in Figure 1. The cooling systems can be categorized into 1) in-vessel cooling system, 2) ex-vessel cooling system, and 3) main cooling system. A steam generator (SG) cooling using air flow along outer wall, nitrogen gas flow inside heat transfer tube, etc., can be categorized into 3) Main cooling system. However this kind of system needs the main heat transport systems.

Recent designs of SFR in the world tend to have direct reactor auxiliary cooling system (DRACS), which has dipped heat exchangers (DHXs) in reactor upper plenum. Some ideas to extend variety and functions of DRACS can be considered in the category of 1) In-vessel cooling system.

A cooling system to remove heat through reactor vessel wall is categorized into 2) Ex-vessel cooling system. The reactor vessel auxiliary cooling system (RVACS) for small power reactors is categorized into this system. If we can increase the heat removal capacity, this system has large variety of component and heat transport system in comparison with those in other categories. There is also direct cooling path for fuel debris on in-vessel core catcher.

We can select the categories 1) and 2) as the first candidates of additional DHRS from the point of independency on main heat transport systems, when we consider severe conditions of internal and/or external events which can cause core degradations.

1.4 Overall experimental plan

1.4.1 Phenomena to be investigated

Figure 2 shows thermal hydraulic phenomena to be investigated in each of two categories. Natural convection in gap region between fuel subassembly (Inter-wrapper Flow: IWF) and reverse flow through heated fuel subassemblies are significant issues to remove heat from degraded core in both categories. Natural convection in upper plenum is also basic phenomena to provide cold fluid into IWF and core and also main heat transport path into lower plenum.

Enhancement of heat transfer between reactor vessel (RV) wall and containment vessel (CV) wall is a key issue in the category of ex-vessel cooling. Natural convection and heat removal in lower plenum including debris on core catcher is also significant for in-vessel retention.

1.4.2 Combinations of sodium and water experiments

In order to clarify these phenomena and provide experimental database for system and component designs and also validation of the simulation methods, set of sodium and water experiments are planned.

A water experiment named "PHEASANT" is available to see natural convections in reactor hot and cold plena with DRACS and Ex-vessel cooling system. Interaction between DRACS and Ex-vessel cooling system is also objective of this experiment.
A middle-scale sodium experiment using existing sodium loop named "PLANDTL-2" is available to investigate core thermal hydraulics, e.g., IWF and reverse flow in core, for In-vessel cooling system.

A large-scale sodium experiment using AtheNa facility, "AtheNa-DHRS," is planned to investigate thermal hydraulics in a reactor vessel with several types of DHRS and also the ex-vessel cooling system (future option), as an integrated test. The experimental plan can be discussed with international partners of the collaboration.

2. Specific plan of experiments

Specific plans using three experimental facilities can be provided later. The outlines of test sections of PHEASANT, PLANDTL-2, and AtheNa-DHRS are respectively shown in Figs. 3 to 5.

3. Code validation

Recently validation of simulation codes is significant, especially for safety evaluation. The data obtained by this experimental program can provide database for such code validations in wide area of phenomena and wide variations of parameters. Especially large sodium test of AtheNa-DHRS is very unique and strong tool to provide high quality data of sodium experiment under high Pe number condition (Pe number distortion will be 1/3 to 1/10).

In this AtheNa-DHRS experimental program, benchmark problems will be provided with detailed data of measurement with precise boundary conditions and geometry data.

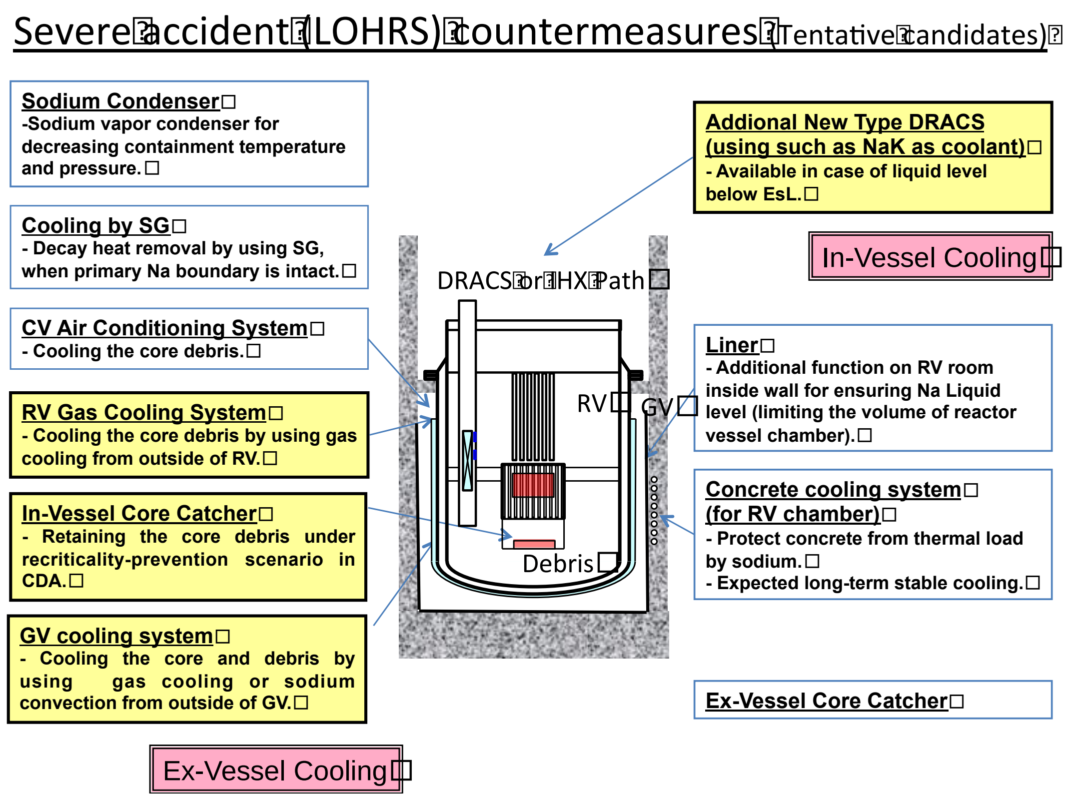


Fig. 1 Candidates of countermeasure for LOHRS

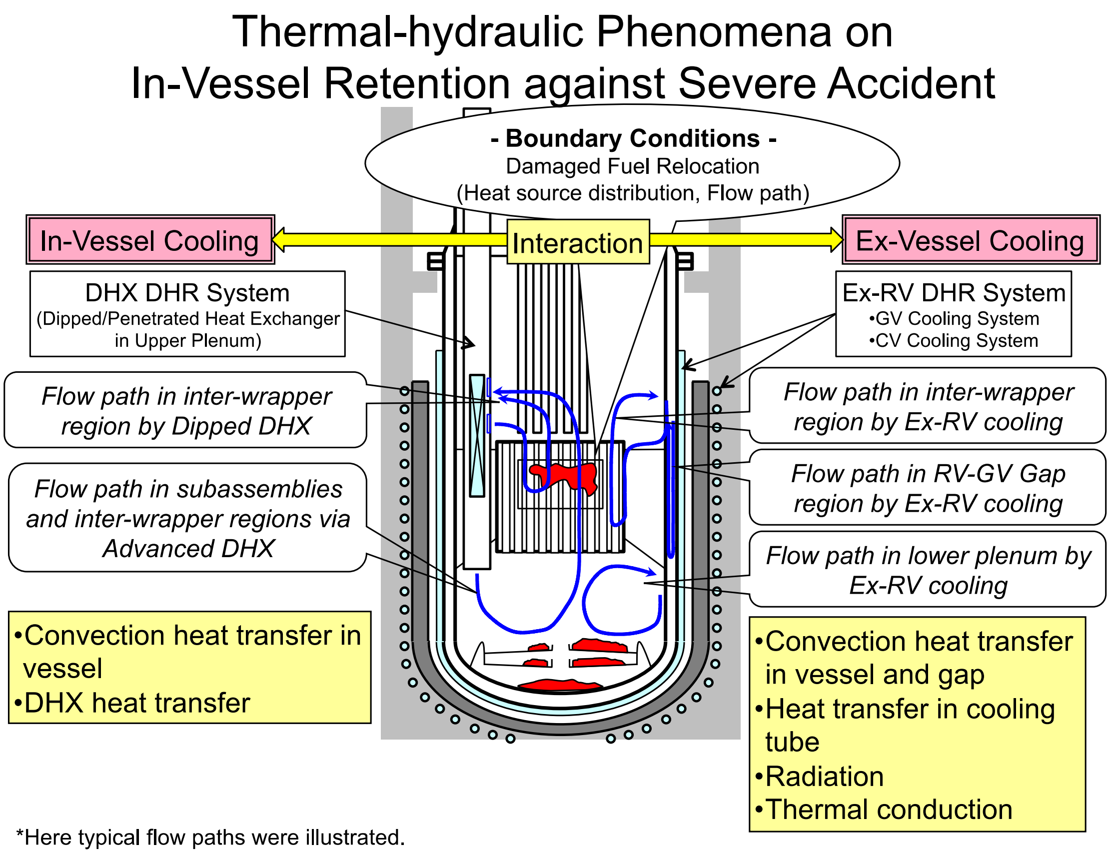


Fig 2 Thermal Hydraulic Phenomena during Decay Heat Removal in PAHR

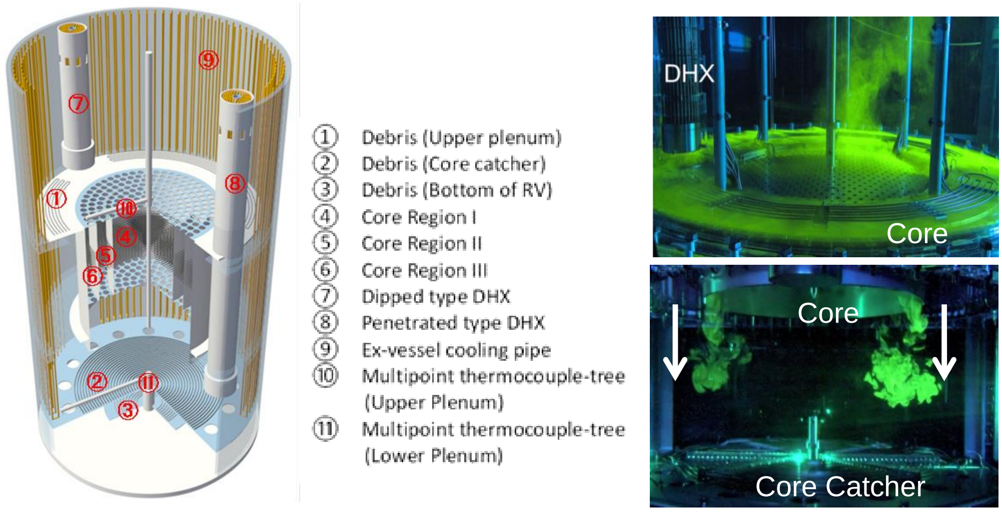


Fig. 3 PHEASANT Test Section and Examples of Flow Visualization

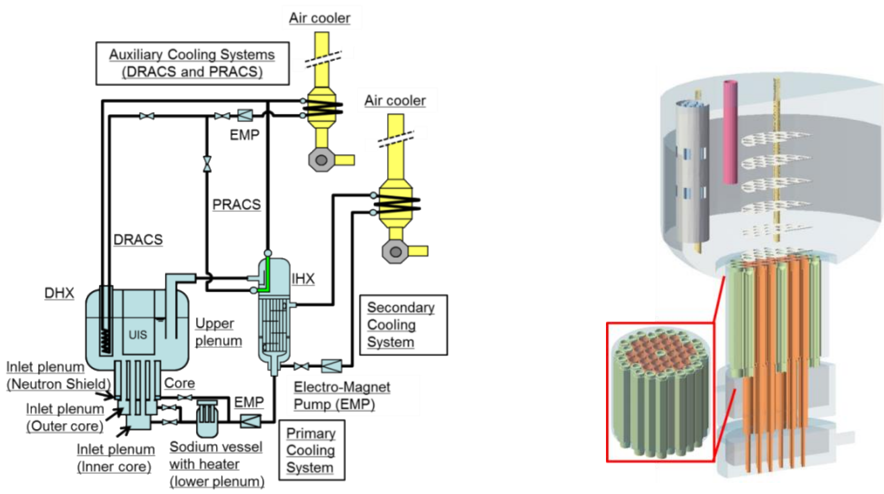


Fig. 4 Schematic view of PLANDTL-2 Sodium Loop and Test Section

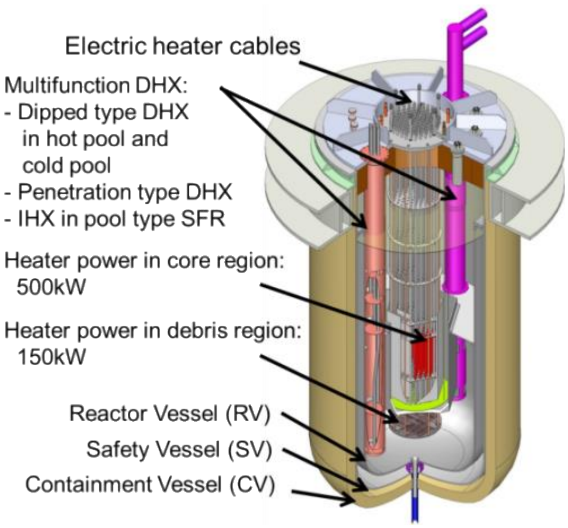


Fig. 5 Tentative image of planned Test Section of AtheNa-DHRS