DEVELOPMENT OF PLANT LIFECYCLE OPTIMIZATION METHOD, ARKADIA FOR ADVANCED REACTORS

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Abstract

The paper describes the outline of the development of Advanced Reactor Knowledge- and AI-aided Design Integration Approach through the whole plant lifecycle (ARKADIA) which provides the best possible solutions for the design, safety measures, maintenance, and decommissioning of advanced nuclear reactors. ARKADIA consists of state-of-the-art numerical simulation technologies, artificial intelligence, and a knowledge base that includes experiences from the development and operation of fast reactors and related research and development. ARKADIA-Design and ARKADIA-Safety are currently being developed individually with as first target of sodium-cooled fast reactors. In a subsequent development phase, these two will be integrated into a single system, ARKADIA, that is applicable to advanced reactors with a variety of concepts, coolants, configurations, and output levels. ARKADIA-Design offers functions to support design optimization both in normal operating conditions and design basis events mainly during a conceptual design stage in the fields of core design, plant structure design including thermal-hydraulics analysis, and maintenance plan optimization. The main technology of the ARKADIA-Design is based on the multi-level simulation approach by the coupled analysis with numerical analysis codes according to user's requirements. For instance, the coupled simulation of neutronics, core deformation, core thermal hydraulics analyses could evaluate the negative feedback effect of the core deformation reactivity during the plant transient. ARKADIA-Safety is developed for automatic optimization of severe accident management and their feedback to a plant design. ARKADIA-Safety includes thermal-hydraulics and multi-physics code, SPECTRA, for in- and ex-vessel integrated severe accident simulation. The basic capability to evaluate severe accident progress was demonstrated through the analysis of a hypothetical loss of reactor level event in a sodium-cooled fast reactor.

1. INTRODUCTION

Japan Atomic Energy Agency (JAEA) has started to develop Advanced Reactor Knowledge- and AI-aided Design Integration Approach through the whole plant lifecycle (ARKADIA) to provide the best possible solutions for design, safety measures, maintenance, and decommissioning of the advanced nuclear reactors [1]. ARKADIA integrates artificial intelligence (AI) with state-of-the-art numerical simulations and a knowledge base obtained from past nuclear reactor development projects. This integration enables automatic optimization of a variety of conditions including safety, risk response, economics, and environmental compatibility. ARKADIA evolves autonomously due to its AI technology and ability to easily import newly obtained knowledge while being used by related organizations. Considering these advantageous features, ARKADIA would also be a powerful tool for educating and training young scientists and engineers.

ARKADIA will be developed in two phases. In the first phase, ARKADIA-Design for the optimization of reactor core design, reactor structure design, and maintenance plan and ARKADIA-Safety for evaluation of severe accidents (SAs) and feedback to a safety design are developed independently. The prime target at this point is a sodium-cooled fast reactor (SFR). In the following phase, they will then be integrated into a single system that will be applied to a wide range of reactor concepts, coolants, configurations, and output levels.

ARKADIA consists of three interconnected systems: Virtual plant Life System (VLS), a numerical simulator; Knowledge Management System (KMS), a knowledge base; and Enhanced and AI-aided optimization System (EAS), a tool to support the optimization process and even create new ideas beyond the reach of experienced engineers. ARKADIA has a platform to receive the requirement from the users through an interface and to make an appropriate evaluation flow depending on the purpose. The three systems work according to the evaluation flow constructed by the platform.

The state-of-the-art numerical simulation technologies for thermal-hydraulics phenomena appearing in advanced reactors are integrated into VLS. VLS of ARKADIA-Design (hereinafter, referred to as Design-VLS) is based on the multi-level simulation approach by the coupled analysis using necessary numerical simulation codes in accordance with user requirements. A plant dynamics analysis code is the base module that is connected to the codes for multi-dimensional thermal-hydraulics analysis, structural analysis, neutronics, seismic analysis, and so on. ARKADIA-Safety will realize automatic optimization of SA management and its feedback to a plant design. VLS of ARKADIA-Safety (hereinafter, referred to as Safety-VLS) includes the integrated SA analysis code, SPECTRA (Severe-accident PhEnomenological Computational tool for TRansient Assessment), which is originally developed by JAEA. To evaluate various SA scenarios in SFRs, this code will compute in- and ex-vessel phenomena by considering their interaction. The newly developed simulation technologies of Design- and Safety-VLS and their example applications are described in the later sections.

2. FUNCTION AND STRUCTURE

2.1. Common function

ARKADIA enables automatic optimization of a variety of conditions including safety, risk response, economics, and environmental compatibility. The automatic optimization flowchart which is common to ARKADIA-Design and -Safety is shown in Fig. 1. This flowchart includes a four-step process: (1) set objective, (2) obtain related information and set evaluation condition, (3) carry out the evaluation, and (4) confirm achievement of the objective. ARKADIA's autonomous evolution is aided by the knowledge gained during this evaluation. The knowledge gained from application to a real plant is saved in the knowledge base. The expanded knowledge base is applied to new designs and safety improvements. Details of each step are described below.

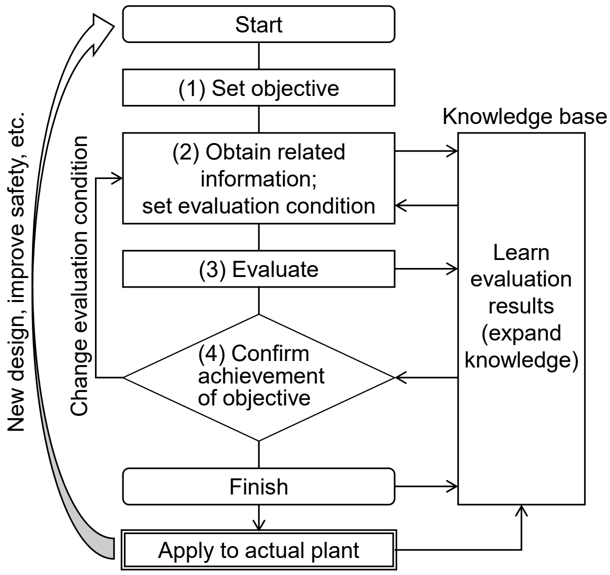


FIG. 1. Optimization flowchart of ARKADIA.

In step (1), an objective function is constructed. The purpose of the evaluation shown in Fig. 1 is to minimize this objective function. In step (2), ARKADIA extracts required data from the knowledge base and determines the evaluation condition. The required data is identified through the hierarchy of predetermined database tags in the knowledge base. In the future works, determination of the database tags will be examined based on the hierarchy method like a library classification. AI allows for the automatic identification of the required data and evaluation conditions. Input data for numerical analysis in the following step are generated automatically based on the selected condition. In step (3), numerical analysis is performed to evaluate the elemental functions included in the objective function. Numerical analysis is not required in some cases such as maintenance optimization. In step (4), ARKADIA evaluates the objective function to assess whether it satisfies a constraint condition and reaches a minimum value. According to this assessment, the process returns to step (2), and the evaluation condition is changed. To satisfy the objective, ARKADIA provides multiple solutions in which the objective function becomes minimum or close to a minimum value.

2.2. Common structure

To realize the automatic optimization described in the previous section, ARKADIA consists of a platform and the three interconnected systems: EAS, VLS, and KMS. The platform receives the user's requirements via a graphical user interface and controls the three systems to achieve purposes. This structure is common to the two ARKADIAs. The roles of the three systems are described below.

EAS identifies elements and parameters from given inputs at the start of the evaluation and then creates an objective function in step (1) in Fig. 1. In step (2), related information is extracted from the knowledge base using a database tag hierarchy. EAS also selects appropriate evaluation conditions and generates input data for numerical analysis which is performed by VLS. In step (4), EAS assesses the achievement of the objective from the value of the objective function. According to this assessment, EAS changes the evaluation conditions to find a better solution. AI technology is used in EAS for highly efficient optimal solution retrieval.

VLS, which is a main topic in this paper, evaluates all possible events including maintenance-related phenomena during a plant lifecycle. VLS will be applied to both design basis accidents and beyond design basis accidents. VLS includes a thermal-hydraulics computational code as a base module to evaluate plant status quantities such as pressure, temperature, and velocity. Liquid sodium single-phase flow is dominant in normal operation or abnormal transient conditions in SFRs. During SAs, on the other hand, compressible multiphase flow with phase change, transport of fission products, and chemical reaction appear. The SA phenomenon must be taken into account in the VLS's base module.

The application range of KMS is a whole plant lifecycle from design to decommissioning. KMS provides knowledge obtained from existing and future research and development. KMS consists of an information infrastructure, a database group, and a knowledge group.

3. NUMERICAL SIMULATION TECHNOLOGIES

3.1. ARKADIA-Design

3.1.1. Overview

ARKADIA-Design supports mainly the conceptual design of core, plant structure, and maintenance plan in normal operating conditions and design basis events. Figure 2 depicts a conceptual view of the functional elements in ARKADIA-Design. One of the main features of thermal-hydraulics analysis by Design-VLS is the multi-level simulation approach by the coupled analysis using necessary numerical simulation codes in accordance with user's requirement [2]. VLS also performs a coupled analysis with a plant dynamics analysis code as base module for plant dynamics analysis and multi-dimensional analysis code for analysis of local behaviours in detail. For core performance analysis, neutronics−core deformation−thermal hydraulics coupled analysis can be performed. The coupled analysis is carried out via an interface named PSSP (Programmable Synchronization Script by Python) developed by the authors. The PSSP is in charge of the execution of analysis codes, the interpretation of data to advance each calculation from one to the next, and the synchronization of data transferred between analysis codes. The PSSP is also in charge of the communication between the modules in EAS, VLS, and KMS.

3.1.2. Numerical methods for core reactivity evaluation

In conventional design procedures to find high-performance core specifications, experts in different fields such as neutronics, thermal hydraulics, and nuclear fuel have to determine individually the boundary conditions with a certain amount of the conservativeness due to the uncertainties in each evaluation to be transferred to the other field, e.g., thermal-hydraulics to neutronics, and then perform a series of analyses and evaluations. This approach requires many repetitions of analysis and evaluation by trial and error until the experts find specifications that satisfy the feasible conditions. This conventional procedure often takes long periods to reach a final solution and provides a conservative result.

As shown in Fig. 2, ARKADIA-Design enables the evaluation of the reactor characteristics by combining neutronics, fuel behaviour, core deformation, and plant dynamics. The conventional complicated design process is expected to be rationalized and streamlined. The Core Evaluation and Design Support Subsystem included in the EAS controls this evaluation process. In this subsystem, the optimization process will be automatically implemented not only in execution control of the analysis codes and handling of their input and output data but also in the process of evaluation of core fuel specifications. The Core Performance Analysis Subsystem in the VLS is also being developed to predict performance in the core by coupling the analysis codes for the neutronics, the thermal hydraulics, and the core deformation. This subsystem can predict reactivity feedback by considering the core deformation depending on temperature distribution during the transient in AOO (anticipated operational occurrences) and ATWS (anticipated transient without scram) events, and the core material arrangement variations during control rod withdrawal events. This system enables quantitative evaluation of the negative reactivity feedback due to core deformation during transient events, which is not considered in conventional core design and safety evaluation due to their large uncertainty. This phenomenon well known as core deformation reactivity feedback is to be considered in the core design of the innovative SFR to improve excessive conservativeness in safety evaluation and to optimize the core design.

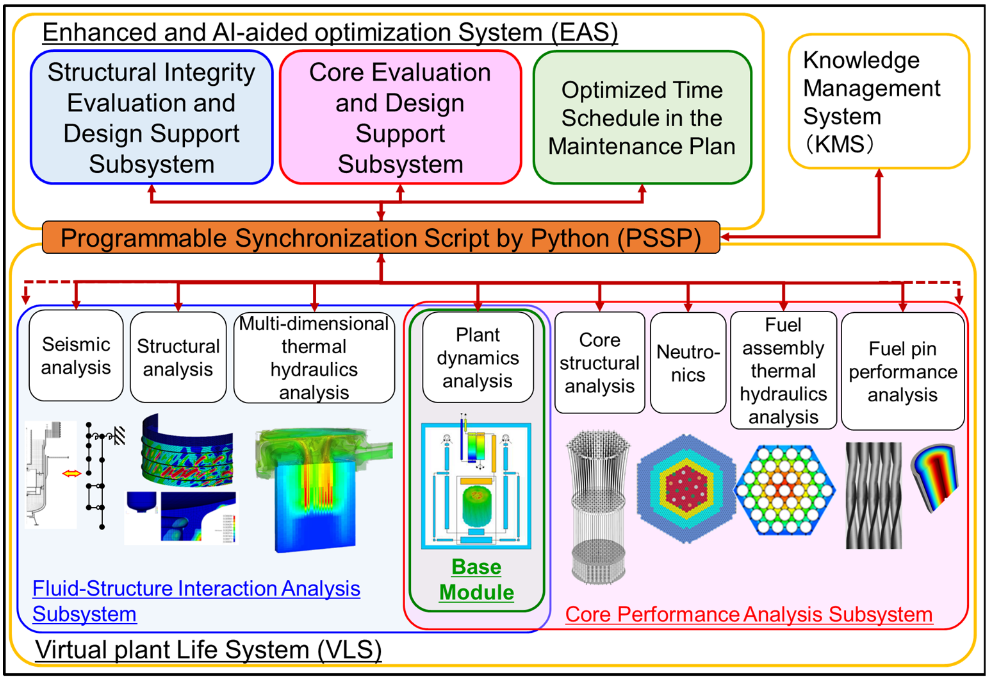


FIG. 2. Conceptual view of module structure in ARKADIA-Design.

3.1.3. Coupled analysis for core numerical methods for core design

Figure 3 shows the outline of the coupling method based on the sequential two-way coupling technique with demonstrative results in a negative feedback effect evaluation in application to an unprotected loss of flow condition in EBR-II. The coupled analysis is implemented using a neutronics calculation program named MARBLE, a plant dynamics analysis code named Super-COPD, and a finite element analysis code named FINAS [3]. The temperature distribution in the core is estimated by the thermal-hydraulics model of the core in the Super-COPD and the deformation of assemblies in the core is estimated by the FINAS. The computed temperature and deformation are used to analyse the core deformation reactivity in the neutronics calculation by the MARBLE. This code-to-code coupled analysis was realized by transferring the boundary conditions computed at a last time step (Fig. 3a). As shown in Figs. 3b to 3d, a negative feedback effect of the core deformation reactivity during the plant transient is evaluated [3]. Additionally, another analysis system for fuel assembly design by coupling the analysis codes for the fuel assembly deformation, the thermal hydraulics, and the fuel pin behaviour has been developed [4] and is to be connected to the VLS.

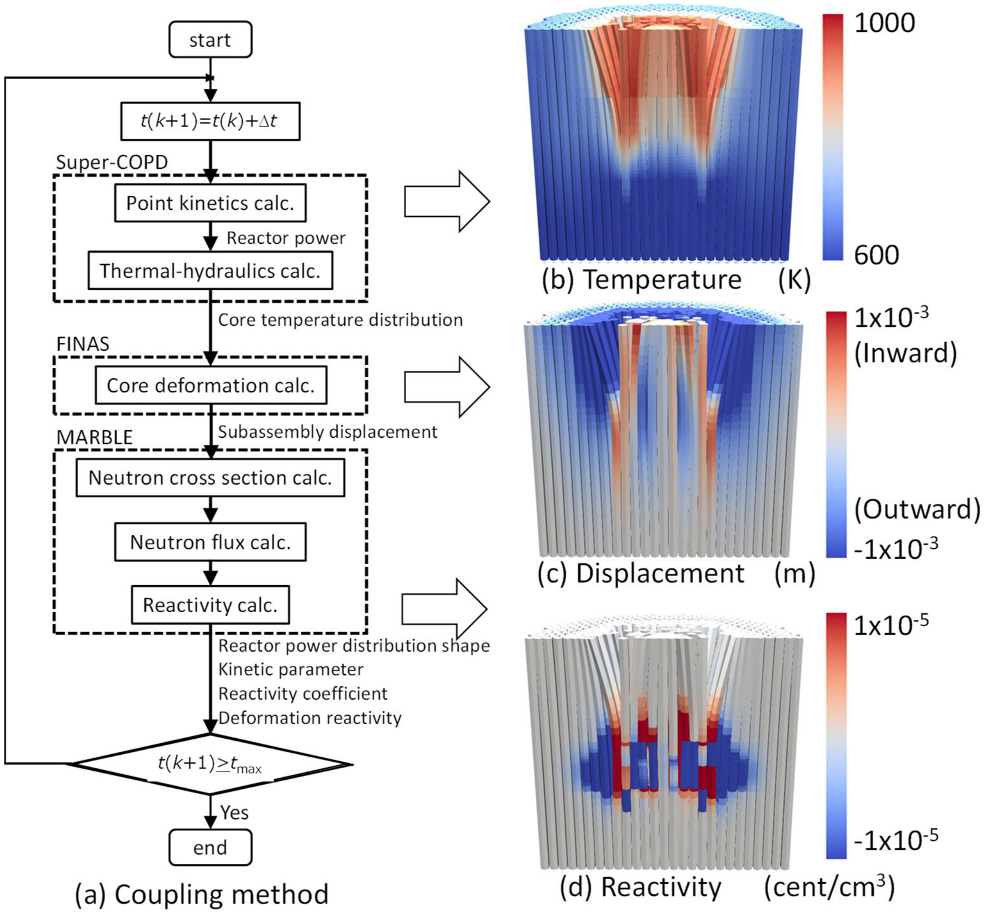


FIG. 3. Coupled analysis for core reactivity evaluation in application to an unprotected loss of flow condition in EBR-II.

3.2. ARKADIA-Safety

3.2.1. Overview

ARKADIA-Safety is developed for automatic optimization of SA management and its feedback to the plant design. As one example SA scenario in SFR, sodium coolant leaks from a failed pipe in a primary cooling loop and it causes sodium fire in an ex-vessel compartment. In case a reactor core disrupts due to significant lowering of a liquid level and a lower head of a reactor vessel (RV) is failed, the concrete floor is ablated by the released sodium and debris. VLS of ARKADIA-Safety evaluates the in- and ex-vessel phenomena which progress while being mutually influenced.

In most of the existing evaluation methods [5-7], users need to connect the different codes by transferring temporal and spatial boundary conditions to realize entire evaluation from in- to ex-vessel phenomena. This approach provides excessive conservativeness in some cases and makes many parametric analyses difficult to achieve. Using a single code like the LWR safety evaluation codes [8-10] is a best way to realize in- and ex-vessel entire evaluation. The authors have started to develop a new computational code, SPECTRA (Severe-accident PhEnomenological computational Code for TRansient Assessment) to compute in- and ex-vessel phenomena simultaneously and to evaluate various SA scenarios in SFRs. Figure 4 shows elemental phenomena to be considered in SPECTRA, which was identified from an importance analysis. SPECTRA is a base module of VLS in ARKADIA-Safety. The following sections explain newly developed computational models for in-vessel thermal hydraulics phenomena including molten core relocation and ex-vessel phenomena including sodium–debris–concrete interaction and sodium fire as a part of SPECTRA. Application to the above-mentioned SA scenario which is called the Loss Of Reactor Level (LORL) event is also presented.

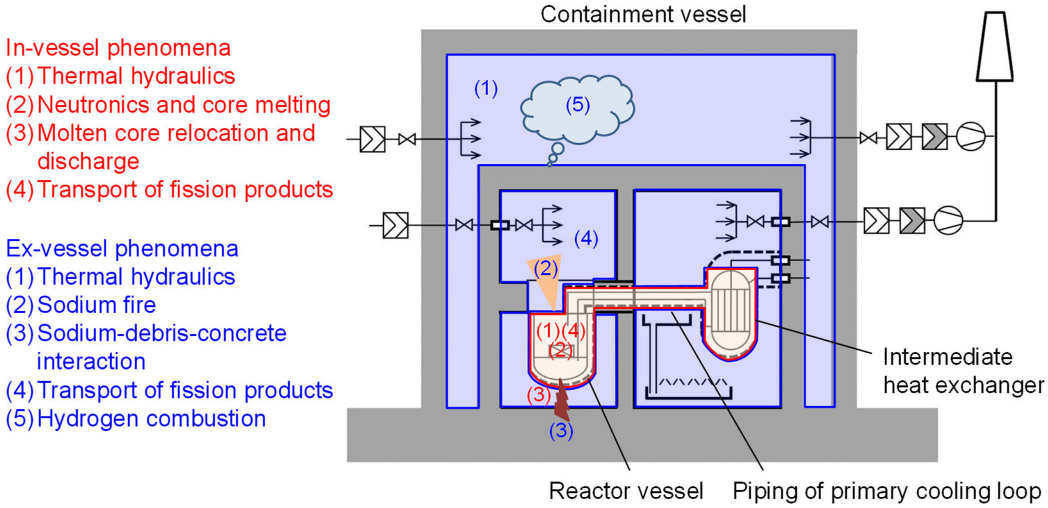


FIG. 4. Phenomena to be evaluated by SPECTRA (adapted from FR22 paper [11]).

3.2.2. Numerical methods for integrated safety evaluation

SPECTRA consists of an in- and ex-vessel module as illustrated in Fig. 5. Each module has a thermal-hydraulics model as a base part. The in- and ex-vessel modules are coupled by exchanging their boundary parameters at every time step. For example, the amount of sodium that leaks from a failed pipe is computed from a pressure difference between the inside and outside of the pipe. Several models for elemental physical phenomena are integrated into the thermal-hydraulics model. Each module will be expanded by improving the current models and integrating other models.

In-vessel coolant flow during a SA includes cover gas and vaporized sodium in some cases. A possibility of core meltdown due to loss of a cooling capability should also be considered in the SA evaluation. The in-vessel module computes behaviors of liquid sodium, gas, and molten core. A multidimensional multifluid model is adopted for this analysis. A fully implicit scheme is applied to this model to enable stable computation with a large time step width. In-vessel relocation of a molten core is computed by a Lagrange-based particle method. A Dissipative Particle Dynamics (DPD) method is chosen from the viewpoint of its low computational load compared with other particle methods. The method can compute the behavior of high-viscosity fluid by using empirical parameters. The molten core analysis model is coupled with the multifluid model by considering porosity and permeability. Momentum and heat are exchanged between the two models.

The ex-vessel thermal-hydraulics model evaluates pressure, temperature, and concentration in each cell and velocity between the cells by a lumped mass model. This model consists of mass, momentum, and energy conservation equations for a multicomponent atmosphere including aerosols. These equations were discretized with a fully implicit method for fast computation as with the in-vessel thermal-hydraulics model. A pressure equation is solved by a successive over relaxation method. Thermal-hydraulics model can consider any number of a wall and compute convection and radiation heat transfer between the atmosphere and the wall. In case a lower head of RV fails, a concrete floor is ablated by chemical reaction with a sodium pool and debris. This sodium–concrete, and debris–concrete interaction is one of the key phenomena under core melt-through events. Sodium–concrete and debris–concrete interaction model was constructed based on models in CONTAIN-LMR [12]. Aoyagi, et al [12] integrated sodium fire models into SPECTRA. The outline of the model is described briefly below. The sodium fire models consist of a spray fire model and a pool fire model. The spray fire model in SPECTRA is based on a model of a multidimensional computational fluid dynamics code, AQUA-SF [14]. The implemented model has some modifications for a lumped mass model, such as eliminations of spray-droplets spreading and the effect of gas-phase velocity on the spray-droplets motion. On the other hand, the pool fire model is based on a model in SPHINCS [15]. SPHINCS has a multidimensional sodium pool model. The pool and floor structure are divided into two-dimensional ring-shaped regions in radial and vertical directions. This ring model is applicable to a pool spreading. The combustion rate of pool fire is determined by a flame sheet model. This model employs four conservation equations in terms of mass and energy transfer on the flame sheet layer with zero thickness: molar flux of sodium vapor and oxygen (or water vapor), heat transfer between pool and flame sheet, and atmosphere and flame sheet. Atmospheric reactions are computed by considering the chemical equilibrium of gas and aerosol components in the atmosphere. Aoyagi, et al [13] demonstrated the capability of the implemented sodium fire model.

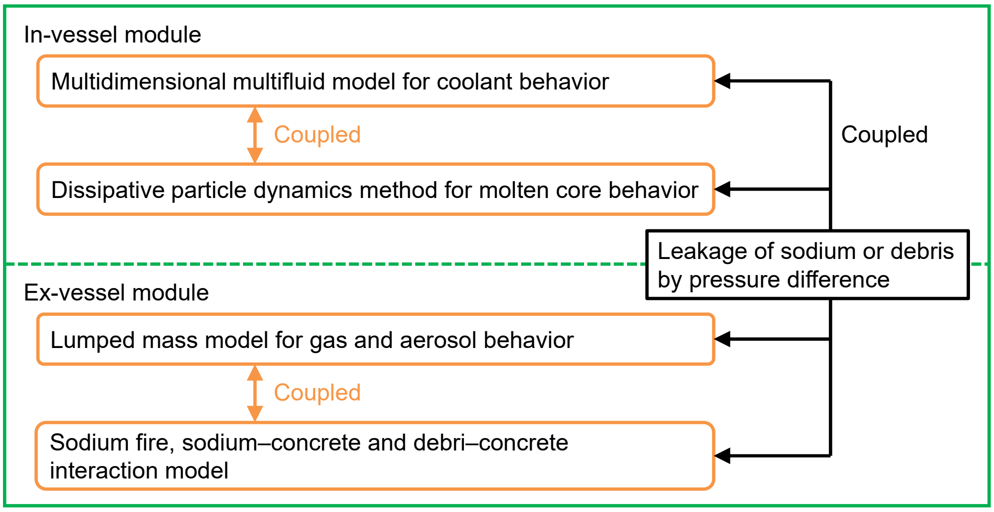
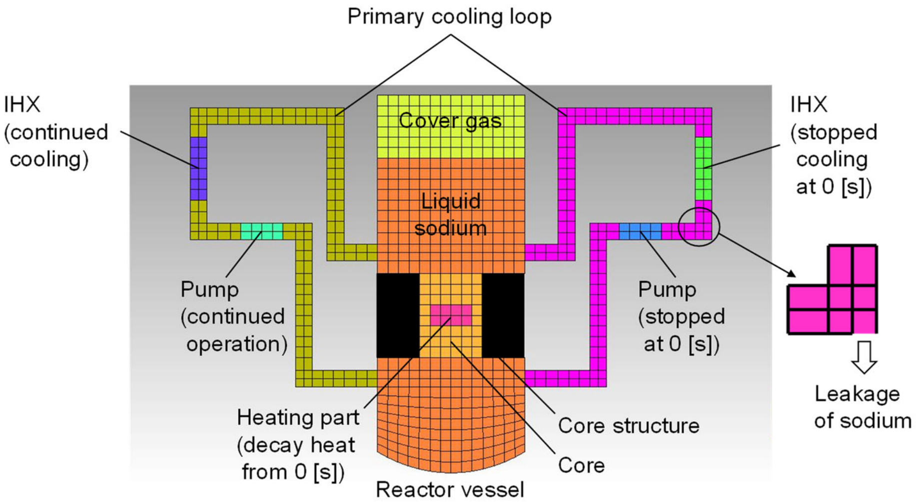


FIG. 5. Current structure of modules and models in SPECTRA (adapted from FR22 paper [11]).

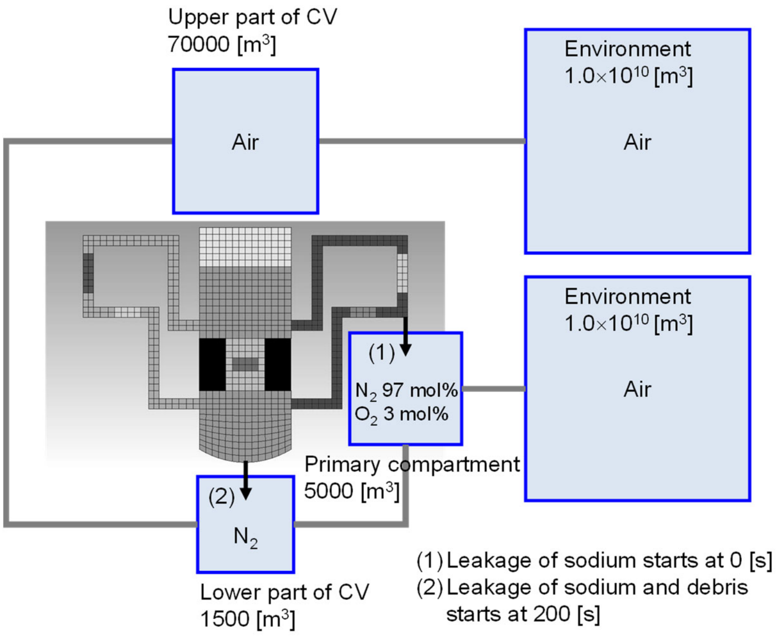
3.2.3. Overall test analysis

Analysis of the hypothetical LORL event was performed for an overall functional test of the SPECTRA code which includes the models on ex-vessel potential phenomena such as sodium fire, sodium-concrete, and debris-concrete interaction. This analysis considers a two-dimensional in-vessel part and a five-cells ex-vessel part. The simplified configuration and coarse mesh are used for the in-vessel part as shown in Fig. 6. The in-vessel part consists of a reactor vessel (RV) and two primary cooling loops. RV includes a core, a core structure, liquid sodium, and cover gas. In this two-dimensional region, porosity was considered in each cell of the CFD analysis to reflect the real volume difference between the components, such as the RV and primary cooling loop. There are a pump region and an Intermediate Heat Exchanger (IHX) region in the primary cooling loop. A momentum corresponding to a pump head was given in the cells of the pump region. Heat is removed in the cells of the IHX region. Before starting the LORL analysis, an analysis only for the in-vessel part was done under a normal operation condition until the coolant flow reached a steady state. This steady state was used as an initial condition of the LORL analysis. At 0 second of the LORL analysis, liquid sodium starts to leak from a cell as shown in Fig. 6. Pump drive and heat removal by IHX stop simultaneously. Decay heat is given in the several cells at the center of the core region. The core is assumed to melt at 800 °C of its temperature for simplicity. The sodium and debris were assumed to start to leak from a lower head of RV at 200 seconds for simplicity. This assumption is based on the mean temperature of the debris particle accumulated at the lower head of RV. Mass of the leaked debris is calculated from the number of debris particles which go out from the opened boundary of the bottom cells of RV. On the other hand, the ex-vessel part consists of an upper and a lower part of a containment vessel (CV), a compartment in the primary cooling system, and the two environment cells outside of the CV. The primary compartment was initially filled with 97 mol% Nitrogen gas and 3 mol% Oxygen. The leaked sodium reacts with oxygen in this cell. Pressure and temperature will increase due to sodium fire. The sodium leak rate is calculated from the pressure difference between the pipe internal and the primary compartment. The sodium-concrete and debris-concrete interaction start in this cell from 200 seconds.

To make an initial condition of this LORL analysis, CFD analysis by in-vessel module was carried out under a normal operating condition. The computed coolant temperature and velocity are shown in Fig. 7. This analysis starts from the condition of a uniform temperature of 500 °C. The circulation flow is formed 10 seconds after starting of analysis. The sodium is heated by the reactor core and goes out from the RV. The heat is removed by IHX. The liquid surface fluctuates due to this circulation flow during this analysis. At 450 seconds, the temperature, velocity, and shape of the liquid surface reached to almost steady state. At this steady state, the inlet and outlet temperature of RV is about 370 and 500 °C, respectively. This realistic steady state condition was used as an initial condition of the LORL analysis.



(a) In-vessel conditions



(b) Ex-vessel conditions

FIG. 6. Computational setup for LORL analysis.

Figure 8 shows computed in-vessel coolant temperature and ex-vessel atmosphere temperature at four different times. The amount of leaked debris is shown by a bar indicator below the lower part of the CV. Due to sodium leakage, the liquid level drops to the level of the outlet pipe at 90 seconds. Sodium fire in the primary compartment causes increase of the atmosphere temperature. Coolant temperature increases locally in the core heating part and the particles expressing a molten core fall toward the lower head of RV. The cooling path from RV and primary loop completely breaks at 218 seconds. Debris started to go out from RV at 200 seconds. At 218 and 350 seconds, atmosphere temperature in the lower part of CV continues to increase due to the occurrence of sodium-concrete and debris-concrete interaction. As a functional test for in- and ex-vessel coupled models, hypothetical severe conditions were simulated successfully. Although the courant number in the in-vessel region always exceeded 1 and reached to about 10 according to flow situation, the calculation proceeded stably. This integrated analysis with a low computational cost will be effective for an exhaustive evaluation of SA scenarios and dynamic probabilistic risk assessment.

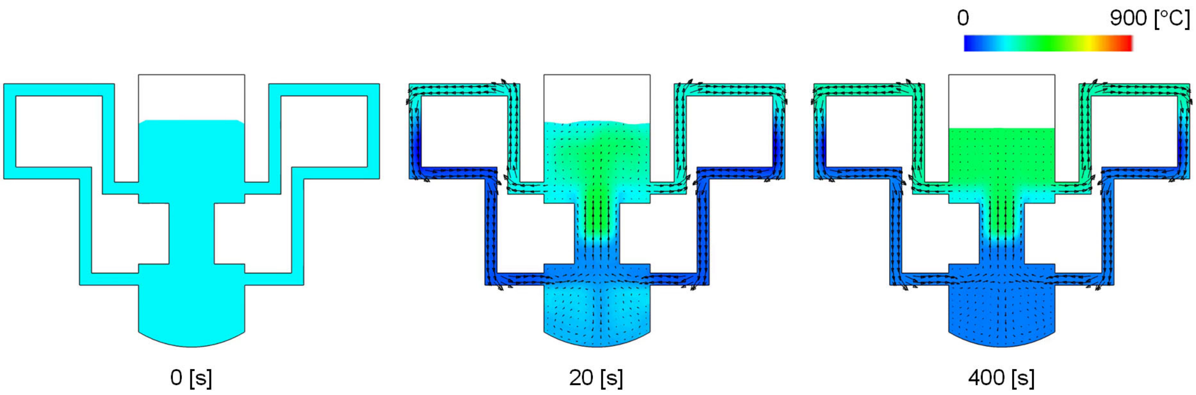


FIG. 7. In-vessel CFD analysis to make a steady state.

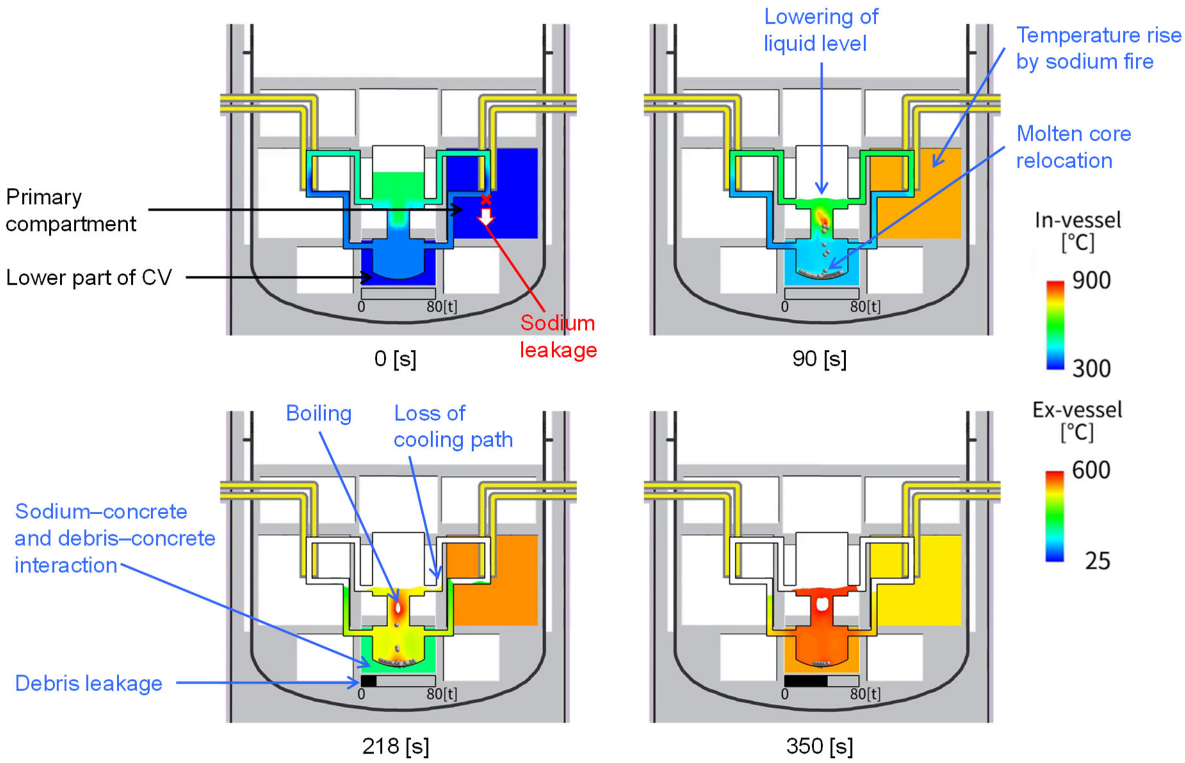


FIG. 8. Ex-vessel phenomena simulated by the SPECTRA code for hypothetical severe conditions.

3.2.4. Future works

Some new models are now developed for more realistic analysis and applicability expansion. The in-vessel coolant flow is currently computed by CFD. For fast computation, a part of the in-vessel region should be calculated by a lumped-mass model. Pressure equations of CFD and the lumped model will be combined in SPECTRA for rational and refined analysis. The core disruption model is being developed. The model is targeted for both mixed oxide fuel and metal fuel. The sodium–concrete interaction, sodium–debris–concrete interaction, and sodium fire described below are computed in the different mesh systems. A new integrated model for these ex-vessel phenomena is being developed [16].

ARKADIA-Safety will be applied to various types of advanced reactors. One of the application targets is a Power Reactor Innovative Small Module (PRISM) reactor which is a pool-type, metal-fueled, small modular sodium-cooled fast reactor. The PRISM-type reactor has the Vessel Auxiliary Cooling System (RVACS) which maintains reactor temperatures well below design limits using air natural circulation to remove heat from the reactor module. To expand the applicability of SPECTRA, a new model considering air natural circulation, heat conduction, heat transfer, and radiation is developed.

4. CONCLUSIONS

ARKADIA currently consists of ARKADIA-Design and -Safety. Design- and Safety-VLS has state-of-the-art numerical simulation technologies. The main technology of the Design-VLS is a multi-level simulation. A neutronics–core deformation–thermal hydraulics coupled simulation was introduced as a representative function in Design-VLS. Application of this VLS to the unprotected loss of flow condition in EBR-II showed a negative feedback effect of the core deformation reactivity. Safety-VLS is based on thermal-hydraulics and multi-physics code, SPECTRA, for in- and ex-vessel integrated numerical analysis of SAs. The basic capability to evaluate transient plant status during SA was demonstrated through the analysis of a LORL event. Some new models for SPECTRA are now developed for more realistic analysis and applicability expansion.

ACKNOWLEDGEMENTS

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