DEVELOPMENT OF IN-VESSEL SOURCE TERM EVALUATION METHOD FOR ULOF EVENT IN SODIUM-COOLED FAST REACTORS

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Abstract

A Source term evaluation for sodium-cooled fast reactors is required to assess the environmental impact, just as for LWRs. From the source term evaluation perspective, ULOF event, a hypothetical leading to core disruptive accident (CDA), is significant because most FPs are released from fuel when it melts and vaporizes due to severe recriticality in an energetic scenario. FPs are expected to be retained within a large bubble comprising vapor of core materials (CDA bubble) and transferred to sodium through an interface between CDA bubble and sodium. Subsequently, FPs are leaked to containment with sodium via damaged reactor vessel head. However, very limited research exists to simulate FP transport phenomena for ULOF event worldwide. Under these circumstances, NRA initiated development of methodology for the source term evaluation focusing on ULOF event. The methodology is achieved by coupling in-house computer codes developed for severe accident analysis (AZORES), plant dynamics analysis (ADYTUM), FP transport analysis (ACTOR) and core disruptive analysis (ASTERIA-SFR). The paper describes the concept and progress of how methodology develops.

1. INTRODUCTION

Following the accident at Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Station (1F), the needs for research and development entailed by source term evaluation have been increasingly spotlighted [1]. To evaluate the amount and type of the chemical form of radionuclide materials released from Sodium-cooled Fast Reactors (SFRs), there is a need to estimate phenomena focusing on Fission Product (FP) transport behavior inside a reactor vessel (RV) such as release from fuel, chemical interaction of FPs, diffusion and transport between the bubble or cover gas and sodium, depending on how the accident progresses. Many research projects focusing on FP transport behavior have been conducted worldwide following the 1F accident. Most of this research, however, concerned Light Water Reactors (LWRs). Accordingly, this highlights the need to develop a methodology for source term evaluation considering inherent SFR features. The difference in coolant materials between LWRs and SFRs results in a range of FP release scenarios during severe accidents. For example, in LWRs, the coolant is totally vaporized inside reactor pressure vessel (RPV) when the core melt progression initiates. Eventually, volatile FPs are directly transported to the atmospheres of the RPV. For SFRs, FPs released from the fuel pin can be trapped by sodium of the liquid phase in a primary heat transfer system (PHTS) in Unprotected Loss of Flow (ULOF) event because of high boiling point of sodium, even under atmospheric pressure.

Historically, ULOF event has been studied as one of the Beyond Design Basis Accidents in SFRs in terms of short timescale to reach core damage and the consequences. In the energetic scenario of ULOF event, it is also well-known that an accident scenario transitions to the core expansion phase if exceedance of prompt criticality occurs during the initiating or transition phase. Things of interest to be evaluated from a safety perspective during the core expansion phase include the integrity of the reactor coolant system boundary such as the RV, primary circuit piping and shielding plug. Once the exceedance of prompt criticality occurs, core materials composed of fuel pellets retaining FPs and steel structure are possibly vaporized at the core region and expand into the upper plenum region. Subsequently, fuel-coolant interaction (FCI) occurs and a large Core Disruptive Accident bubble (CDA bubble) forms, with a radius in the order of meters, a high temperature of several thousand Kelvin and high pressure. Consequently, sodium at the upper plenum called sodium slug is driven by FCI pressure and may result in a leak path forming at the shielding plug due to the sodium slug impact. In case of a sodium leak from the RV to the containment via a damaged shielding plug, the containment could fail due to high temperature and pressure, since sodium interacts with oxygen and moisture on the operation floor of the containment.

The difference in coolant materials from LWRs should also be considered in terms of interaction with FPs. Here, the most dominant FPs are iodine and cesium, given their exposure and environmental impact, respectively. Both FPs are directly released from fuel pellets into the atmosphere of the RPV as a gas phase under LWR accident conditions, as already mentioned, but behave differently in SFRs because of interaction with sodium and the presence of coolant when the severe core damage occurs. In SFRs, most of the iodine is captured by sodium in the form of NaI, hence, iodine is rarely transported to cover gas [2, 3]. The behavior of cesium under SFR accident conditions, meanwhile, is complicated. When the fuel pin fails, cesium is released from the fuel pin with bubbles composed of noble gas and other volatile FPs. In energetic scenario, most of cesium can be released to CDA bubble due to melting and vaporization of fuel. Subsequently, the cesium could form aerosols or be transported to the sodium via a monoatomic or diatomic molecule depending on temperature, with the materials contained in the bubbles. Finally, cesium in the bubbles directly reaches the cover gas or cesium in sodium may be transported by evaporation from the sodium surface or diffusion caused by the gradient of concentration between the cover gas and upper plenum region. The retention factor (RF), generally defined as a fraction of the cesium quantity between the cover gas region and the initial inventory, should be a significant parameter to determine how much cesium could be transported to the environment.

In recent years, several research projects on source term evaluation for SFRs were launched in response to active SFR development projects worldwide [4, 5, 6]. The research applying source term evaluation to reactor scale analysis, particularly calculating FP transport behavior directly coupled with thermal-dynamics depending on accident progression for ULOF event involving core disruptive condition, remains limited.

NRA has been developing several computer codes to obtain technical knowledge on SFR safety. ACTOR [7] is an in-house computer code to calculate the FP transport details in PHTS during Loss of Heat Sink (LOHS) event. AZORES [8] is a computer code to analyze accident progression and FP transport, including ex-vessel phenomena during LOHS and Loss of Reactor Level (LORL) event. Conversely, there are no source-term evaluation tools applicable to ULOF event particularly involving the severe recriticality and violent material relocation. Therefore, NRA has been developing a comprehensive methodology, achieved by coupling those computer codes with ASTERIA-SFR [9], which is a code for mechanistic CDA analysis also developed in NRA. The paper describes overview of the methodology on source term evaluation during ULOF event and its development progress.

2. OVERVIEW OF COMPUTER CODES

This section describes the computer codes which will comprise the methodology of source term evaluation.

2.1. ACTOR

ACTOR is the analysis code used to calculate the transport behavior of FPs, which are released from the gas plenum or fuel pellets due to a cladding breach caused by a rise in temperature during an accident and into the coolant sodium and cover gas of the SFR plant. To analyze FP transport behavior as shown in Fig. 1, ACTOR encompasses several analytical models corresponding to the following phenomena:

(a) FP release from fuel;
(b) FP transport in sodium;
(c) Adsorption of FP to structure surfaces from sodium;
(d) Entrained FP transport by rare gas;
(e) FP heating in cover gas; and
(f) Adsorption of FP to structure surfaces in cover gas.

The principal analytical models adopted in ACTOR are developed based on the following experimental study results obtained:

(a) Experiment about FP release from irradiated FBR MOX fuel [10];
(b) Experiment about FP adsorption on the surface of sodium piping [11];
(c) Experiment about gas bubble behavior in the upper plenum of a RV [12]; and
(d) Experiment about entrained FP transport from rare gas to sodium [13].

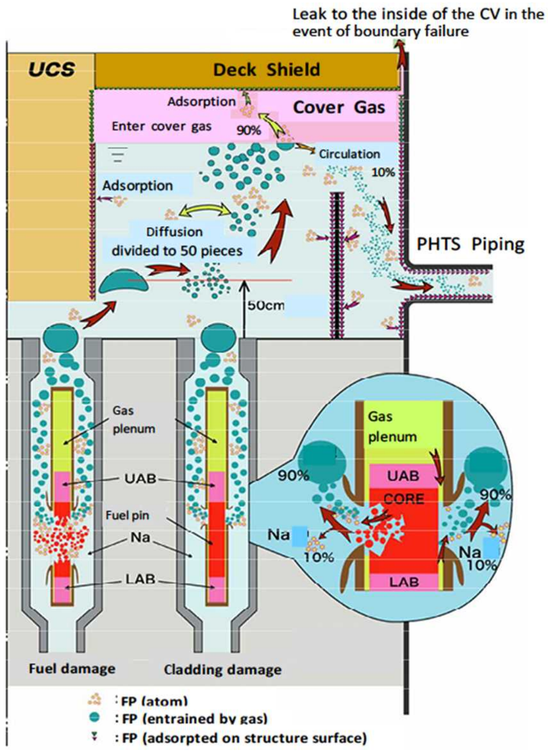

FIG. 1. Overview of ACTOR [7].

2.2. AZORES

AZORES was developed to analyze severe accident phenomena leading to the release of molten core debris and sodium to the containment and radio nuclide behavior in the containment system. In the present study, AZORES was applied to the core material relocation and ex-vessel source term analysis for the typical SFR accident sequence.

Fig. 2 shows the following phenomena affecting accident progression after core damage initiation in SFRs:

(1) The sodium-concrete interaction between the sodium released from the primary coolant system and building floor concrete in the containment;
(2) The melted debris-concrete interaction between the molten core materials released following the RV failure and the floor concrete under the RV;
(3) Sodium combustion at the containment dome of an oxygen environment in the event of sodium leak from the primary cooling system to the containment dome;
(4) Hydrogen burning in various rooms with an oxygen environment in the event hydrogen is generated by sodium-concrete interaction; and
(5) Aerosol formation and growth with sodium oxide and radionuclide materials, aerosol deposition and transportation.

AZORES can address source term analysis based on thermal hydraulics behavior for (1) through (5) under severe accident conditions.

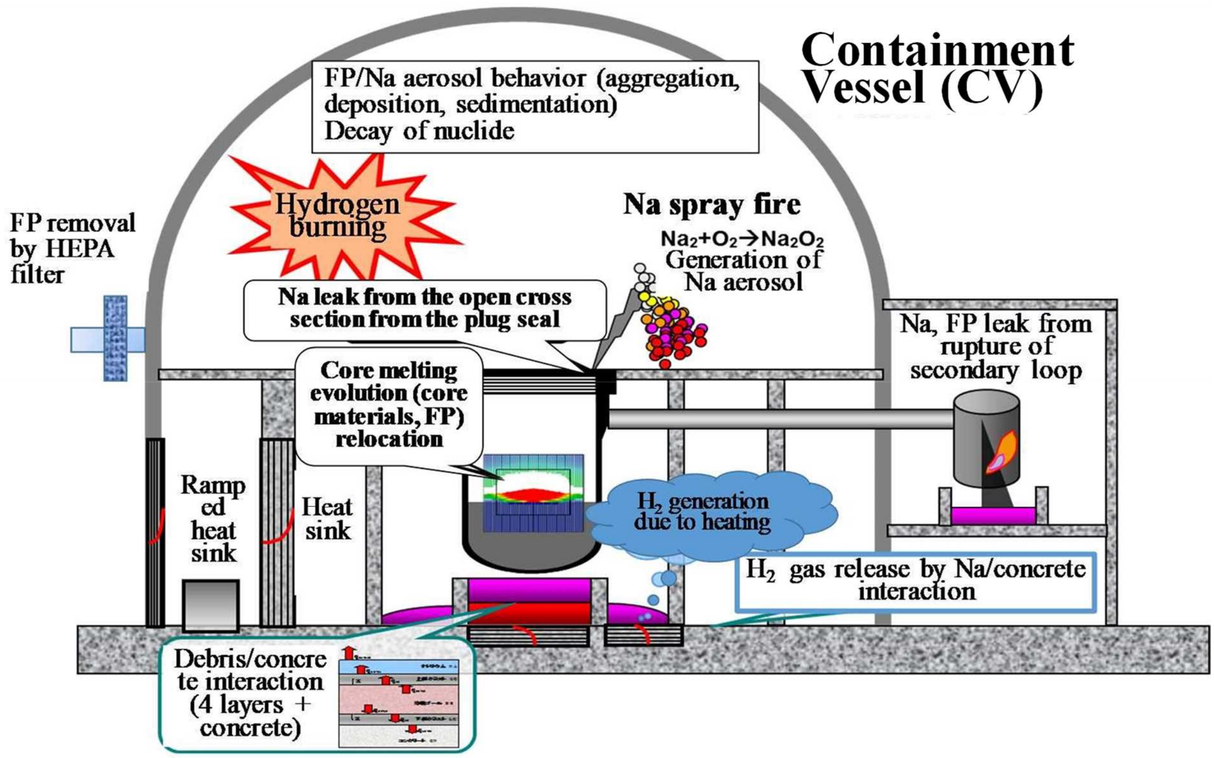

FIG. 2. Overview of AZORES [8].

2.3. ADYTUM **[14]**

ADYTUM is a plant dynamic analysis code for SFRs based on the one-dimensional flow network model for primary and secondary sodium loop systems.

The flow network model calculates the two-phase and incompressible sodium flow, heat transport along the flow path and heat transfer between sodium and structures and includes highly flexible scope to treat transient thermal-hydraulics.

The ADYTUM concept diagram is shown in Fig. 3. This code comprises several packages of a physical model named "module" which are reactor point kinetics, a mechanical pump, heat exchanger and matrix solver. All modules are connected to the flow network model to transfer the variables calculated in each module. Accordingly, ADYTUM is designed to be applicable to any SFR plant design by the flow network model representing the plant design appropriately. In addition, each module can be flexibly connected to the flow network model where a specific component is installed at an actual SFR plant, such as fuels loaded in the core region, pumps and heat exchangers, since each module calculates independently from others.

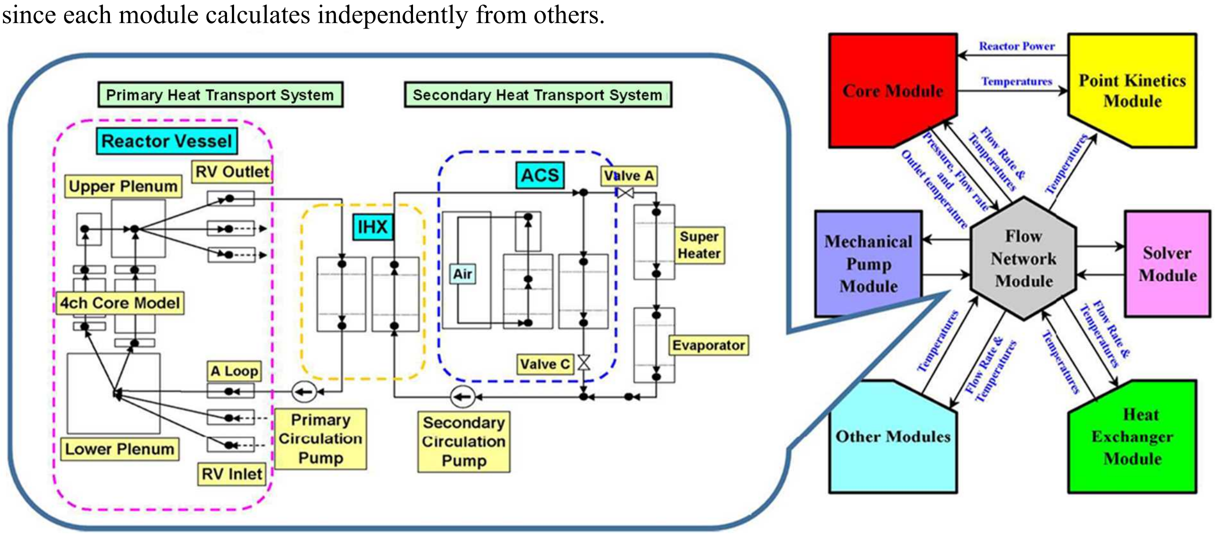

FIG. 3. Overview of ADYTUM [14].

2.4. ASTERIA-SFR

As shown in Fig. 4, ASTERIA-SFR consists of three major modules, namely a thermo-fluid dynamics calculation module, CONCORD, a fuel behavior calculation module, FEMAXI-FBR and a space-time neutronics calculation module, PARTISN/RKIN. The data transfer in each time-step is conducted between these modules via ASTERIA-CNTL which deals with not only performing data transfer but also a time step control for each module. The thermo-fluid dynamics calculation module is used for the source term evaluation. In Chapter 3, the outline of CONCORD is described in more detail.

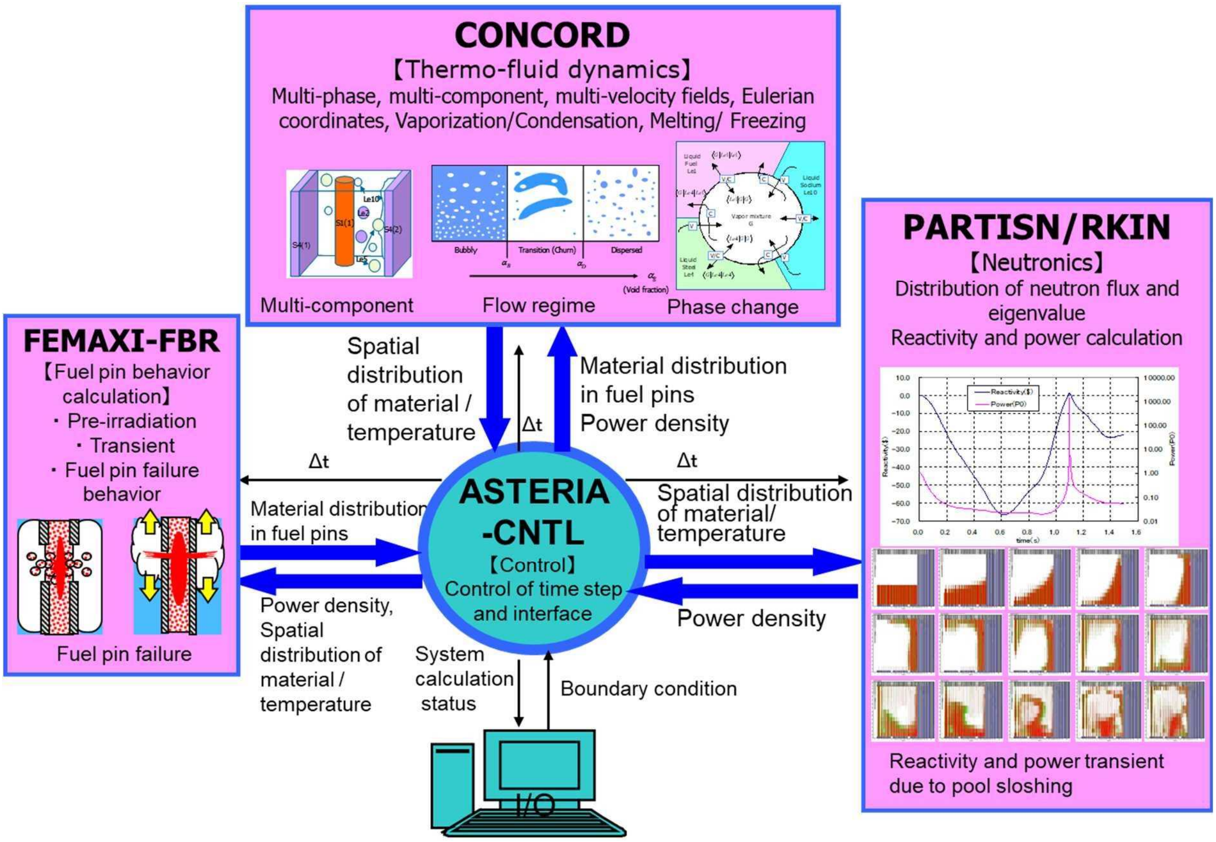

FIG. 4. Overview of ASTERIA-SFR [9].

3. DEVELOPMENT OF METHODOLOGY

3.1. Overview

Fig. 5 outlines the methodology to be developed and cesium transport phenomena addressed in the same. It is noted that the red-bordered box indicates the phenomena calculated by each of the computer codes. ACTOR and AZORES are capable of performing FP transport calculations in PHTS and the containment respectively. Source term evaluations can be categorized into two scenarios, namely instantaneous and delayed source term respectively, depending on the accident scenario [15]. Those codes already show capability for the latter delayed source term scenario in which fuel and FPs are released from the combustion of a sodium pool or evaporation, because those codes originally target LOHS and LORL event, in which a reactor is successfully shutdown but means to remove decay heat are lost and FPs are released from the fuel slowly after the core is revealed. On the other hand, the computer codes mentioned above are not applicable to the former instantaneous source term scenario in which the transport of radioactive materials, mostly aerosols, in a single large bubble or several smaller bubbles under energetic condition, to the cover gas and the containment through leaks occurs because only gravity-driven fuel relocation is treated in AZORES. Hence, this underlines the need to develop methodology applicable to the FP transport in an energetic scenario.

ASTERIA-SFR plays a significant role in the methodology for in-vessel source term evaluation in ULOF event particularly the FP transport involving CDA bubble behavior inside the RV as described in Section 3.2. The subsequent scenario is needed to be simulated consistently, based on the calculation results of the core materials relocation caused by the severe recriticality including from initiating and transition phase. To conduct the scheme mentioned above systematically, those computer codes will be incorporated to the single integrated severe accident analysis code system described in Section 3.3. As a first development step, modeling of FP material in ASTERIA-SFR and coupling between ACTOR and ADYTUM have already been initiated.

Fig. 6 shows role of each computer codes and data transferred between them in the methodology. Blue and red arrows are expressed data transfer way via Input/Output files and shared memory respectively. Evaluation begins from top of Fig. 6. Data transferred between computer codes is indicated at next to each arrow. First, ASTERIA-SFR calculates core disruptive behavior such as heat and mass transfer, fuel pin behavior, and power transient from initiation of ULOF event to when exceedance of prompt criticality occurs or deep sub-critical is achieved.

Calculation flow for the core expansion phase is shown right side of the Fig. 6. In the phase, ASTERIA-SFR is used again but only CONCORD is performed in order to estimate amount of leaked sodium, cover gas and cesium to CV in the methodology. Mass and temperature distribution of the core materials are given as an input data from result of first calculation of ASTERIA-SFR. After that, result of second ASTERIA-SFR calculation is transferred to AZORES via output files. Quantities from ASTERIA-SFR are given as boundary conditions of calculation by AZORES for FP transport from CV to environment. AZORES calculates pressure and temperature of atmosphere and FP transport at CV dome considering Na combustion. Finally, amount of FPs released to environment is estimated in the core expansion phase since AZORES is capable of calculation on FP leakage from CV to environment as well.

On the other hand, calculation scheme for PAMR/PAHR phase is shown at left side of Fig. 6. AZORES, ADYTUM and ACTOR are utilized to evaluation of the phase and are communicated data via shared memory since three codes are worked as a single code system. Quantities for FP transport calculation by ACTOR such as fuel temperature, coolant temperature, coolant flow rate and so on are provided by AZORES and ADYTUM. Since AZORES and ADYTUM are capable of calculation regarding melt progression and integrity of PHTS including RV, ACTOR calculates FP distribution at PHTS and FP release to CV depending on accident progression. AZORES is used to calculate FP release from CV to environment in PAMR/PAHR phase as well as a part of the code system. Thus, once calculation result of ASTERIA-SFR at beginning of this scheme is given to the code system, FP transport behavior including release to environment is obtained.

3.2. Methodology for the core expansion phase

Understanding the phenomena regarding FP transport behavior is key to estimate the amount of FPs released from the RV appropriately. However, no method exists to calculate the FP transport behavior for instantaneous source term scenario such as the energetic scenario of ULOF event considering CDA bubble behavior. Hence, we launched the computer codes development for this scenario. CONCORD, a thermal fluid module of ASTERIA-SFR, is a mechanistic analysis method, which can calculate heat and mass transfer such as vaporization/condensation (V/C), melting/freezing (M/F) and advection/diffusion (A/D) in detail. Therefore, CONCORD is adopted because FP transport behavior, including leak to the containment, can be simulated appropriately based on mechanistic analysis method during very short timescale (<1s) of the energetic scenario of ULOF event such as the core expansion phase, provided precise physical and thermal properties of FP are given to CONCORD. Cesium is selected as a representative FP modeled in CONCORD. To identify the phenomena in terms of cesium transport shown in Fig. 5, analytical models regarding concentration diffusion, gas-liquid equilibrium partition coefficient and the physical and thermal properties of cesium are developed.

There are two ways to deal with cesium in CONCORD in this study. One is herein named the “Cs component model”, in which cesium is modeled by Equation-of-State (EOS) model of CONCORD. The EOS model provides adequate physical and thermal properties alongside wide-ranging pressure and temperature based on thermodynamic relationships. This model is also capable of directly simulating the thermal hydraulic behavior of cesium such as heat and mass transfer including V/C and M/F during ULOF event.

The other is herein named the “Cs concentration model” and aims to address the diffusion effect between the liquid and liquid/gas phases, although the heat and mass transfer of cesium is simpler than in the Cs component model. In this model, cesium is not defined as an independent component and the cesium amounts of the liquid and gas phases are represented as molar concentration fractions to liquid sodium and FP gas components respectively. Thus, the liquid and gas cesium are advected by the velocities of liquid Sodium and FP gas respectively. When the temperature of liquid sodium containing liquid cesium exceeds the boiling point of cesium, the liquid cesium evaporates and the evaporated cesium transports to the FP gas component. On the other hand, when the temperature of the FP gas containing gaseous cesium goes below the condensation point of the cesium, cesium condensate transports to the liquid sodium component. The diffusion model is imported from that of ACTOR and the following three diffusion paths are addressed in CONCORD: (1) Diffusion of cesium dissolved in the liquid phase of sodium.

(2) Diffusion between liquid sodium and the gas phase, which are continuous phase of flow regime model. (3) Diffusion between liquid sodium and the gas bubble, which are continuous and dispersion phases of the flow regime model respectively. The diffusion of cesium is driven by concentration difference and the interfacial area between each of the phases.

On interface between liquid and gas phase, it is said to occur that liquid on the surface forms droplets (called entrainment jet) and enters gas region. To make the source term evaluation more realistic, entrainment model based on Rayleigh-Taylor instability [16] is introduced to CONCORD and applied to calculations involving both the Cs models, because the interfacial area between the liquid and gas phases is expected to change as a result of the entrainment. The effect of entrainment on cesium transport will be verified.

To validate the Cs models of CONCORD, a comparison of results is scheduled between the calculation and FAUST test [17], which was conducted to investigate FP transport behavior, particularly the aerosol transport into the cover gas under the core expansion phase condition. In the test series, FAUST-II/B test was served to investigate several species such as Cs, I_2, CsI and so on under a large sodium pool system. The CDA condition was simulated by injecting a relevant specimen with pressurized Ar gas from the bottom of the pool. The portion of species not retained in the sodium pool was taken from the cover gas and measured at sampling pots. Subsequently, based on the amount of material measured at the sampling pots and injected into the sodium pool, RFs were evaluated in the test. The Cs models will be compared with RFs of No. 314 of the FAUST-II/B test, which employs cesium for the specimen.

3.3. Methodology for PAMR and PAHR phases

After the core reaches neutronic shutdown by discharge of fuel from the core, ULOF event transitions into post-accident material relocation (PAMR) and post-accident heat removal (PAHR) phases. Then, the integrity of the RV is discussed in terms of coolability of degraded core materials, particularly in the scenario having transitioned from initiating and transition phases. If the integrity of RV is lost, degraded core materials and sodium are leaked from the damaged vessel head or lower head with FPs. FPs would however behave independently from the molten fuel, since most of the FPs have already been released from the fuel when the fuel is melted or vaporized due to the severe recriticality. In particular, during ULOF event, the coolant keeps circulating in PHTS under the controlled flow rate, despite the core experiencing severe core damage, and FPs are transported by sodium to somewhere in PHTS, including inside the RV. This is why an evaluation method calculating FP transport in PHTS is needed for the in-vessel source term evaluation of ULOF event. Simulation of the in-vessel source term is conducted by ACTOR, which calculates the transport phenomena from the time the violent phenomena are ended and the core reaches deep subcriticality. On the other hand, the computer code served in the thermal hydraulics calculation during the PAMR/PAHR phase is transferred from ASTERIA-SFR to the code system mentioned in Section 3.1, which is originally designed to evaluate the integrity of reactor cooling system boundary taking into account melt progression. Calculation results on FP distribution and thermal hydraulics in the RV involving behavior of CDA bubble by CONCORD are provided to ACTOR as input data. At this time, ACTOR is performed as a code system module, hence FP transport can be solved based on thermal hydraulics calculations in PAMR/PAHR phase as well. ACTOR begins subsequent calculation of FP transport such as advection, adsorption and diffusion until FPs are leaked from the boundary.

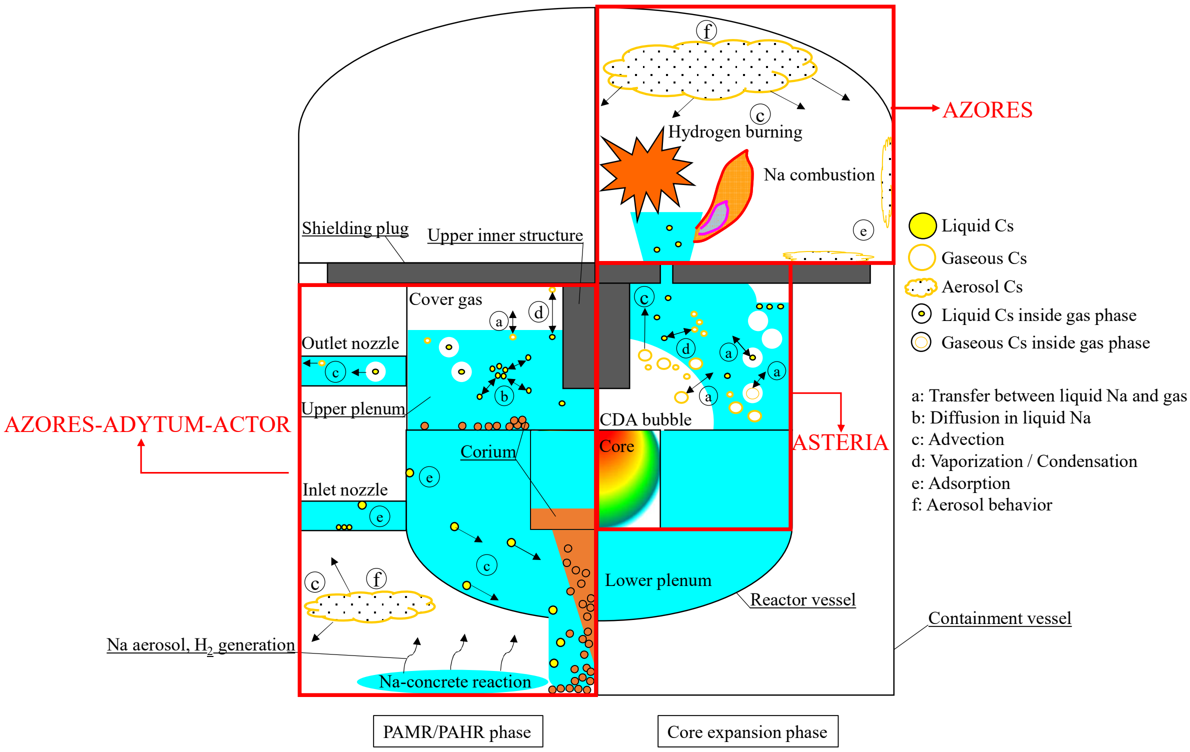

FIG. 5. Phenomena on in-vessel Cs transport during ULOF considered in our methodology.

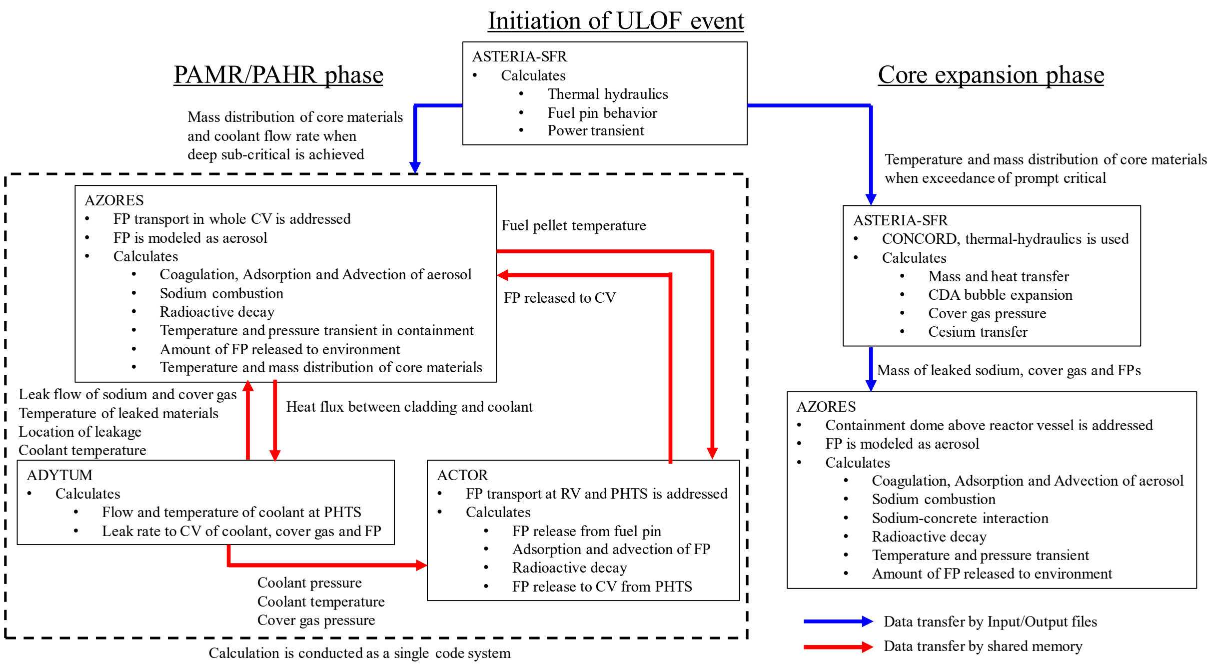

FIG. 6 Calculation scheme of Source Term Evaluation for ULOF event by this methodology.

4. CONCLUSION

NRA initiated the development of methodology for the source term evaluation focusing on ULOF event. This methodology is achieved by coupling in-house computer codes, namely AZORES, ADYTUM, ACTOR and ASTERIA-SFR.

ADYTUM is capable of simulating thermal hydraulics in reactor cooling system such as coolant flow velocity and temperature response depending on accident progression, which affect FP transport by modeling the flow path inside the RV, piping and heat exchangers appropriately. ACTOR deals with phenomena on FP transport which are released from fuel, adsorption on structures and transport delivered by coolant flow. Coolant flow and temperature, however, must be given from thermal hydraulics codes. AZORES has the advantage of extended analysis of core melt progression over time and ex-vessel phenomena, including FP transport in PAMR/PAHR phases. On the other hand, AZORES is not applicable to ULOF event in which core expansion due to the severe recriticality occurs within a very short period, since only fuel relocation driven by gravity is treated. CONCORD, the thermal hydraulic module of ASTERIA-SFR, can conduct a detailed heat and mass transfer simulation based on mechanistic analysis method, provided precise physical and thermal properties of FP are given.

In this context, ASTERIA-SFR plays a significant role to simulate FP transport behavior with violent thermal hydraulics within short timescale of ULOF event in this methodology. To evaluate ULOF event, cesium is selected as a representative FP to confirm the applicability of CONCORD to the source term evaluation. The development of analytical models regarding concentration diffusion, gas-liquid equilibrium partition coefficient and the physical and thermal properties of cesium for CONCORD is initiated. For the subsequent calculation from core expansion phase, AZORES, ADYTUM and ACTOR will be integrated to simulate FP transport during PAMR/PAHR phases. The result of CONCORD is given as input data of the code system described above, to simulate FP transport of an extended period in the reactor cooling system and the ex-vessel phenomena.

In future, the codes will be verified by conducting whole plant analysis, while significant phenomena affecting cesium transport and further analytical models will be investigated by comparison with experiments.

Acronyms list

1F	Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Station
ACS	Auxiliary cooling system
ACTOR	Radioactive materials transport code in Primary heat transport system
ADYTUM	Plant dynamics code
ASTERIA-CNTL	Control module of ASTERIA-SFR
ASTERIA-SFR	Core disurptive behavior code
AZORES	Integreted severe accident code
CDA	Core disurptive accident
CONCORD	Thermo-fluid dynamics module of ASTERIA-SFR
CV	Containment vessel
EOS	Equation of state
FAUST-II/B	Experiment on aerosol behavior
FBR	Fast breeder reactor
FCI	Fuel-coolant interaction
FEMAXI-FBR	Fuel pin behavior calculation module of ASTERIA-SFR
FP	Fission product
IHX	Intermidiate heat exchanger
LOHS	Loss-Of-Heat sink
LORL	Loss-Of-Reactor level
LWR	Light water reactor
M/F	Melting and freezing
MOX	Mixed-oxide
NRA	Nulear Regulation Authority, Japan
PAHR	Post accident heat removal
PAMR	Post accident material relocation
PARTISON/RKIN	Neutronics module of ASTERIA-SFR
PHTS	Primay heat transport system
RF	Retention factor
RPV	Reactor pressure vessel
RV	Reactor vessel
SFR	Sodium-cooled reactor
ULOF	Unprotected Loss-Of-Flow
V/C	Varporization and Condensation

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