Source Term Estimation for Radioactivity Release under Severe Accident Scenarios in Sodium cooled Fast Reactors: Technical Specification and Approach

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Abstract. Due to the inherent characteristics and robust design of Sodium cooled Fast Reactors (SFR), the core disruptive accident (CDA) is considered a very unlikely event. Nevertheless, one of the hypothetical scenarios that is used to confirm the safety of the reactor and to serve as a basis for containment design and severe accident management measures is the Unprotected Loss of Flow (ULOF) accident. Determining radioactive source term released into the containment and subsequently to the environment, are required steps in the assessment of the adequacy of the containment and the radiological impact to the site after and accident. In general, the estimation of the source term for the sodium cooled fast reactors requires computational tools similar to those developed for light water reactors. However, the state of the art for sodium fast reactor safety analysis is not as advanced as it is for light water reactors, and there are a number of different areas where totally different methodologies have to be used. With the goal of improving the current state of the art for modeling the in-vessel and in-containment source terms, the IAEA launched a coordinated research project, with participants from nine countries, to do benchmark simulations for the source term estimation with different models and tools. The paper presents the problem definition and approach. The resulting models are expected to provide a more realistic than existing conservative estimates and would further help to identify areas for experimental investigations through sensitivity and uncertainty analysis of the improved integrated models.

Key Words: source term, radio-nuclide transport, core damage

1. Introduction

In Sodium cooled Fast Reactors (SFR), the hypothetical Core Disruptive Accident (CDA) is a Beyond Design Basis Accident (BDBA) where a postulated initiating event combined with total failure of shut down systems results in a power increase or surge in the core. The initiating events postulated is typically either a loss of flow or uncontrolled absorber rod withdrawal event. The reactor shuts down subsequently due to inherent feedback or core disassembly depending on the initiating transient and reactor feedback coefficients, including sodium void coefficient. The consequent thermal energy release has an equivalent mechanical

work potential, usually in the range of few to hundreds of mega joules during which high temperatures and high pressure are reached. Though the accident is a BDBA, the Reactor Containment Building (RCB) is designed to be able to handle the consequences of CDA and to ensure that the dose rate at the site boundary is within the prescribed limit.

The International Atomic Energy Agency launched, in 2016, a coordinated research project (CRP) to help study the consequences of this type event, named "Radioactive Release from the Prototype Fast Breeder Reactor under Severe Accident Conditions". Participants from nine different countries are contributing to this project, and the technical aspects to be addressed are subdivided into three main parts. First is the in-vessel source term estimation, consisting of risk important fission product distribution in the fuel pins, their release mechanisms into the coolant and subsequent reaction and transport in the coolant and release to the cover gas. This part will involve modeling the physical disassembly of the core and its dynamics in the vessel. Second is the primary system/containment interface source term estimation consisting of models for the cover gas, sodium ejection and radionuclide chemical composition and distribution in the containment. The third part is the estimation of the fission product evolution within the containment, aerosol behavior, and physical boundary conditions.

The IAEA CRP is still in progress, but this paper describes work that was done leading up to the start of the project in order to define reference cases and define the boundary condition for different parts of the study. Towards this, an SFR model has been defined and developed. The input data required for the simulations have been calculated and boundary conditions identified and specified. The paper presents the problem definition, approach and results obtained from the preliminary modeling. The resulting models are expected to provide a more realistic than existing conservative estimates and would further help to identify areas for experimental investigations through sensitivity and uncertainty analysis of the improved integrated models.

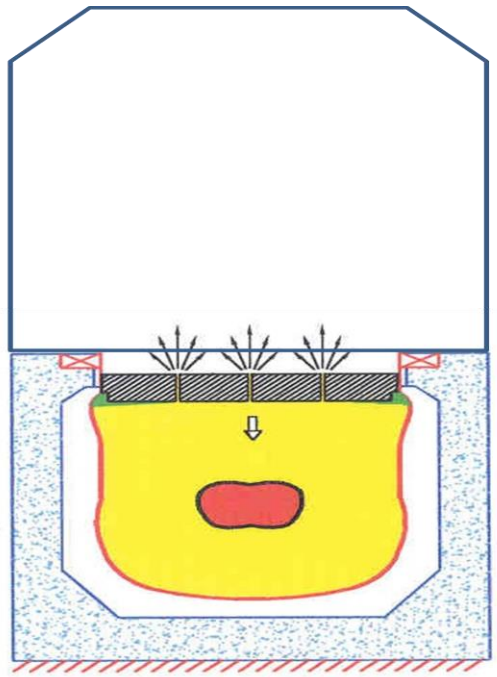

FIG. 1. Schematic of SFR undergoing a CDA

The consequences of CDA in terms of radioactivity release to outside the containment system which may affect the environment and the public is of paramount importance from public acceptance point of view, especially after the Fukushima event. Even though structural integrity of the primary vessel can be ensured by way of demonstrating its capability to withstand high mechanical energy, the pressure developed within the vessel during a CDA can lead to sodium release to RCB through several potential leak paths in the top shield structure of the reactor as shown in the schematic Fig.1. The ejected sodium can burn inside the RCB, leading to an increase in temperature and pressure. The ejection of sodium into RCB would also be accompanied by radioactive fission products and fuel that have come out of the core. Fission products can leak to the environment through the leak paths in the RCB.

The spread of the activity and dose rates at the site boundary and habitability of the control room need to be evaluated so as to provide sufficient

design measures to protect the public from the consequences and to ensure that the dose rate exposure is within the design basis limits, which is important in the domain of safety.

This requires a good understanding of the whole phenomena under a severe accident scenario. Towards this, it is essential to precisely estimate the radioactivity source term, by modeling the phenomena of the mechanism of transport of fission products involving core bubble expansion characteristics, heat transfer interaction among core materials, sodium & cover gas, chemical interaction between fission products and sodium etc and to characterize the parameters influencing the fission product retention and ejection.

2. Description of the Reference Sodium Cooled Fast Reactor

2.1 Primary System

In order to benchmark the development of mechanistic models for the assessment of in vessel and in containment source term, a reference SFR has been defined. The reference reactor is a generic model loosely based on the Prototype Fast Breeder Reactor, currently under commissioning in India. The reference reactor consists of a primary sodium circuit, two secondary sodium circuits and a steam water circuit [1]. Primary sodium circuit consists of core, control plug, hot pool, cold pool, two primary sodium pumps, four Intermediate Heat Exchangers (IHX), and Safety Grade Decay Heat Removal System (SGDHRS) etc. Primary sodium pumps (PSP) pump sodium from the cold pool into the grid plate. Grid plate (GP) supports all the subassemblies (SA) and distributes sodium to them. When sodium passes through the SA, heat transfers from the fuel pins to the coolant and increases primary sodium temperature. The hot sodium coming out of the SA outlets mixes in the hot pool and enters the IHX. The hot sodium while passing through the IHX transfer its heat to the secondary sodium and becomes cold. The cold primary sodium coming out of the IHX mixes in the cold pool. Finally the secondary sodium transfers heat to the steam-water system.Schematic of reactor assembly section through pump and intermediate heat exchanger along with top view is shown in Fig. 2a and 2b. The reactor core layout is shown in Fig. 3 and important core parameters are given in Table I.

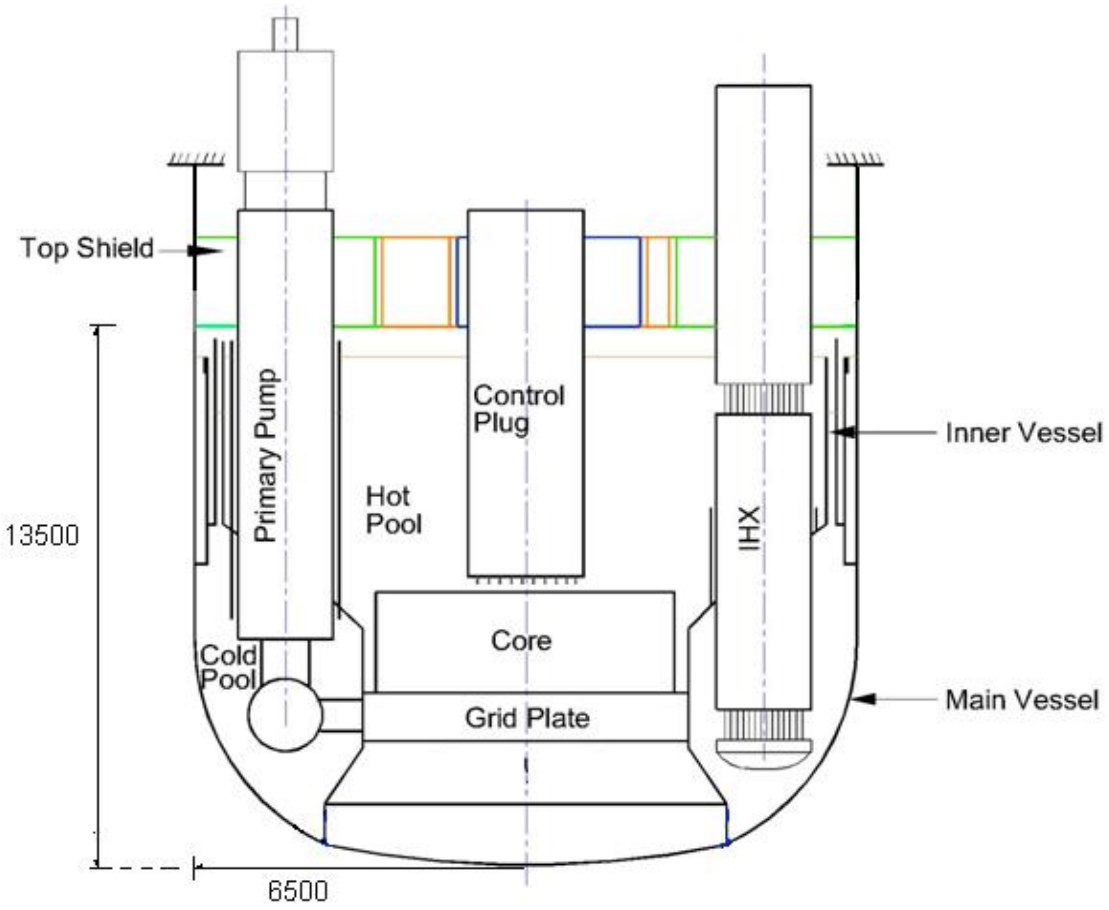

FIG.2a. Schematic of Reactor Assembly Section through IHX and PUMP

FIG.2b. Schematic of Reactor Assembly Top View

TABLE I: CORE PARAMETERS

Reactor power	1250MWt/500 MWe
Fuel	PuO_2-UO_2
Coolant	Sodium
No. of absorber rods	9 CSR and 3 DSR
Max. fuel burnup	100 GWd/t
Blanket material	Depleted UO_2
No. of SA	180
Re fuelling interval	180 EFPD
Fuel smeared density	82.5%
Concept of primary Na circuit	Pool
Coolant inlet temperature	670 K
Coolant outlet temperature	820 K

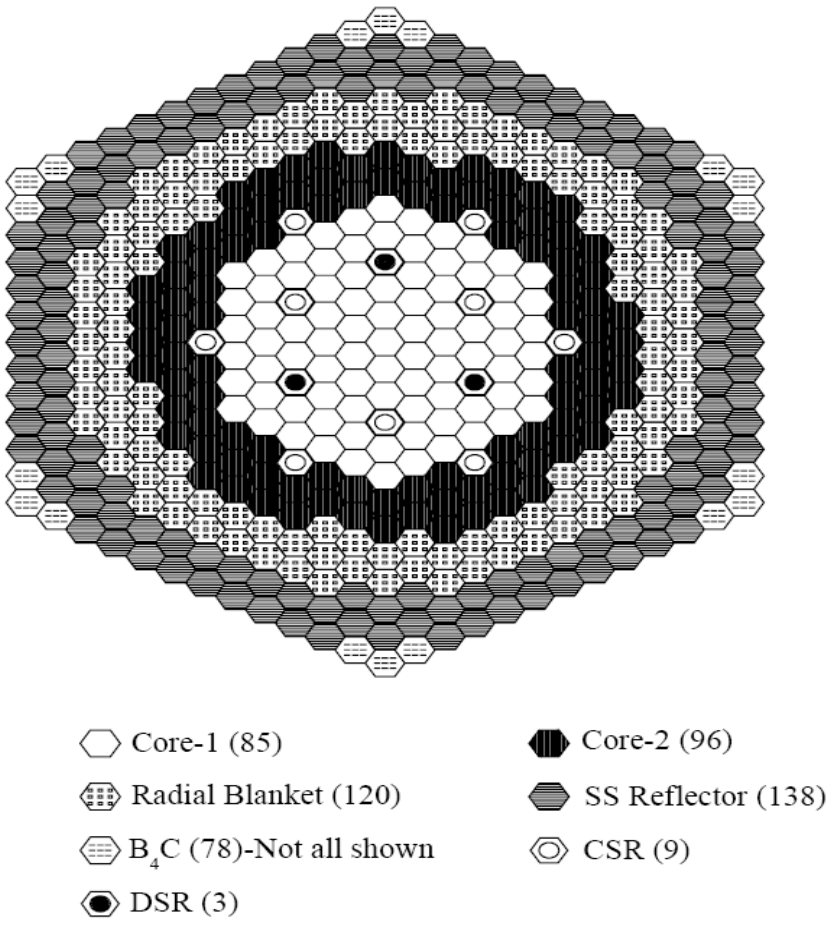

FIG.3.Core configuration

2.2 Reactor Containment Building

The plan view of the cells within the RCB above roof slab level (floor level) is shown in Fig. 4. The cells mainly contain cover gas systems. Above roof slab a working platform is provided at a height of 4 m. This platform is not leak tight around pipe and equipment penetrations. Effective area of opening in the platform is 5 m^2. The RCB volumes below roof slab level are not (conservatively) considered for expansion and deposition of radioactivity although they are connected by openings for stair ways and cable/piping ducts.

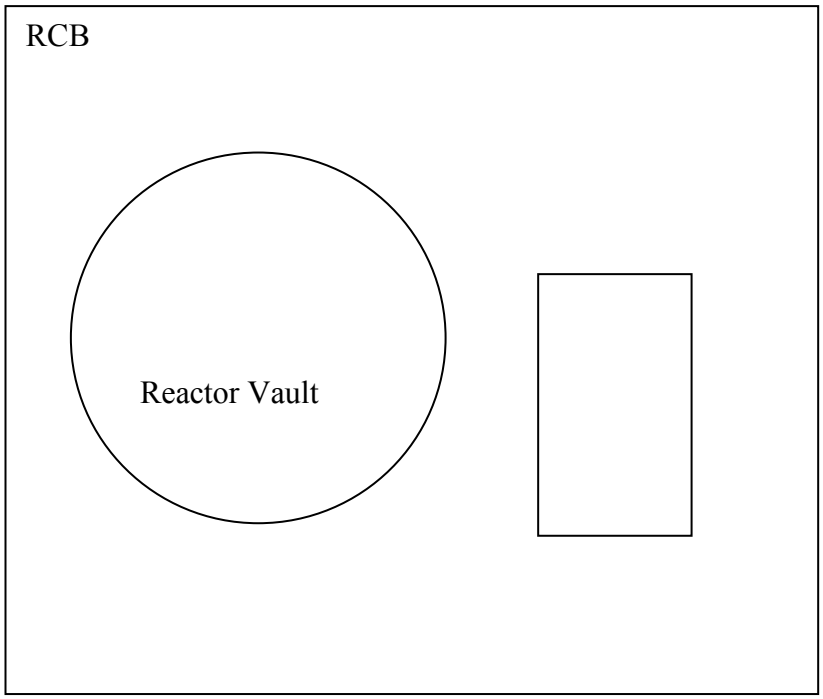

FIG. 4. RCB plan View at 18m height Showing Reactor Vault and Major Cell Area

The cover gas systems are housed inside the RCB cells and finally paths exist for the exhaust of purified and delayed effluents to be vented through stack. The effective area available for leak through the top shield is comparatively much larger than the flow path available through cover gas argon circuit (pipe diameter 10 cm and many filters and tanks in the flow path), the cover gas circuit need not to be considered for the first phase of the analysis.

RCB Data

Length = 40 m

Breadth = 35 m

Height above roof slab level 55m

Height below roof slab level 18 m

Rectangular Cell RCB above roof slab level

Length =7 m

Breadth=18 m

Height =22 m

Platform above roof slab at 4.0 m height, diameter 18m

RCB total volume = 77000 m^3

RCB available volume = 74000 m^3

2.3 Radio Nuclide Inventory

The radionuclide inventory (in Bq) of important fission product isotopes and actinides is given in Table II. The data corresponding to three cases, viz., MOEC, EOEC and 100GWd/t burnup cases are given in the tables. Since there are 180 fuel assemblies, the activities for the three cases are obtained as follows.

$A_{100GWd/t}$ (Bq)= 180 x $A_{100GWd/t}$

A_{EOEC}(Bq) = 60 x A_3 (540) + 60 x A_2 (360)+60 x A_1 (180)

A_{MOC} (Bq)= 60 x $A_{2.5}$ + 60 x $A_{1.5}$ +60 x $A_{0.5}$

TABLE: II CORE INVENTORY

Radionuclide	Half life	Activity in Bq for MOEC	Activity in Bq for Peak Burnup (100 MWd/t)
I-131	8.02 d	1.15E+18	1.95E+18
I-132	2.30 h	1.82E+18	2.59E+18
I-133	20.80 h	2.43E+18	3.35E+18
I-134	52.50 m	2.63E+18	3.34E+18
I-135	6.57 h	2.32E+18	2.94E+18
Cs-134	754.50 d	2.13E+16	1.27E+17
Cs-137	30.07 y	4.74E+16	1.32E+17
Rb-88	17.78 m	4.91E+17	6.89E+17
Ru-103	39.26 d	2.31E+18	3.18E+18
Ru-106	373.59 d	1.94E+18	1.40E+18
Sr-89	50.53 d	5.93E+17	9.40E+17
Sr-90	28.79 y	1.68E+16	3.80E+16
Ce-141	32.50 d	1.78E+18	2.79E+18
Ce-144	284.89 d	6.90E+17	1.39E+18
Te-131m	30.00 h	3.42E+17	2.15E+17
Te-132	3.20 d	1.78E+18	2.48E+18
Ba-140	12.75 d	1.91E+18	2.55E+18
Zr-95	64.02 d	1.50E+18	2.32E+18
La-140	1.68 d	1.94E+18	2.59E+18
Kr-83m	1.85 h	1.06E+17	1.47E+17
Kr-85	10.70 y	2.65E+15	6.19E+15

Kr-85m	4.48 h	2.03E+17	2.98E+17
Kr-87	76.30 m	3.70E+17	5.39E+17
Kr-88	2.84 h	4.74E+17	6.52E+17
Kr-89	3.15 m	5.58E+17	7.68E+17
Xe-131m	11.84 d	1.28E+16	1.58E+16
Xe-133	5.24 d	2.39E+18	3.37E+18
Xe-133m	2.2 d	7.97E+16	1.10E+17
Xe-135	9.14 h	2.64E+18	3.51E+18
Xe-135m	15.29 m	6.02E+17	8.34E+17
Xe-137	3.82 m	2.14E+18	2.96E+18
Xe-138	14.08 m	1.87E+18	2.59E+18
U-237	6.75 d	1.24E+17	1.65E+17
U-239	23.45 m	2.34E+19	2.84E+19
Np-239	2.35 d	2.32E+19	2.84E+19
Pu-238	87.7 y	2.51E+14	5.77E+14
Pu-239	2.4×10^{4} y	3.22E+15	3.52E+15
Pu-240	6564 y	4.75E+15	5.77E+15
Pu-241	14 y	4.11E+17	4.75E+17
Pu-242	3.7×10^{5} y	4.61E+12	5.79E+12
Cm-242	0.44 y	3.25E+16	6.81E+16
Cm-243	28.5 y	8.74E+12	2.49E+13
Cm-244	18.1 y	6.01E+14	1.66E+15

3. Accident Scenario

In pool type SFR, by design loss of coolant accident is very unlikely event. Typically, sodium cooled fast reactors provide a guard vessel to prevent core uncovery in the event of leak developing in the primary vessel. Extended loss of heat removal capabilities at a plant could lead to evaporation of the sodium coolant and core uncovery and core melt. Since this scenario would occur after a very long time because of the large heat capacity of the primary system inventory of coolant, this scenario would not be case for large early release of radioactivity. However, loss of flow event is a possibility and is protected by shutdown systems. Loss of flow followed by unavailability of shutdown system, referred to as Unprotected Loss of Flow Accident (ULOFA), could result in rapid core meltdown. Hence ULOFA is considered for the benchmark investigation. ULOFA transient can occur due to loss of power to both the primary pumps. This leads to the rise of coolant temperature but also leads to an initial decrease in power and fuel temperature due to negative core expansion feedback. However, since the power to flow ratio becomes high, this ultimately results in coolant temperature rise, and boiling in the upper part of the highly rated channel. As void spreads radially outward and axially inward towards core centre large positive reactivity is introduced. It may lead to power excursion and finally to clad dry out that results in rapid increase in cladding and fuel temperatures and, later, in melting of cladding and fuel. Due to inherent uncertainties in modelling of this phase, a conservative approach is followed in the simulations performed in IGCAR with dedicated in-house CDA code. Once one third part of fissile zone is molten fuel slumping is initiated as follows. The middle one third core slumps and occupies the bottom one third coolant. The top one third slumps and occupies the middle one third. The transient moves to the disassembly phase when the peak fuel temperature reaches boiling point. The analysis is continued in the disassembly phase till reactor becomes sub-critical.

3.1 Degraded core data

Preliminary test simulations of the core disruptive accident (CDA) were performed with dedicated CDA analysis code developed in IGCAR. The core bubble data pertaining to the LOFA scenario are presented for the nominal (non-energetic) and conservative (highly energetic) cases of energy release in Table III. The conservative case corresponds to a mechanical energy release of 100MJ. The nominal case corresponds to <0.1MJ.

TABLE III: CORE BUBBLE DATA

No.	Parameter	Nominal Case (Non-Energetic)	Conservative Case (Highly –Energetic)
1	Reactor Thermal Power	1250MW	1250 MW
2	Fuel Melting Point	2750 °C	2750 °C
3	Fuel Boiling Point	3387 °C	3387 °C
4	Clad Melting Point	1427 °C	1427 °C
5	Clad Boiling Point	2750 °C	2750 °C
6	Total Core Volume	3 m^3	3m^3
7	Transient + Disassembly Phase	80s+42 ms	80s+11ms
8	Peak Temp.	3460 °C	4945 °C
9	Peak Pressure	0.23 MPa	9.7 Mpa
10	Thermal Energy Released	300MJ (0.1MJ, Mech. Work)	5000MJ (100MJ, Mech. Work)
11	Melt Fraction	46 %	54 %
12	Vapor Fraction	0.2 %	40 %
13	Peak cover gas pressure	-	1.6 MPa
14	Quasi static pressure of core bubble	-	0.2 MPa

4. Radionuclide Release and Transport

Towards the calculation of the in containment source term, the important stages of calculation can be arranged in terms of the significant volumes participating in the transport process and time scales. The important volumes are the fuel (active core region excluding SA parts), sodium coolant, cover gas and RCB. The corresponding transfer paths for RN as depicted in Fig.5 are,

i) Fission gas present in the gap to coolant.
ii) Fission gas from coolant to cover gas and then to RCB
iii) Lesser volatiles from core to coolant
iv) Lesser volatiles from coolant to RCB due to coolant ejection
v) Lesser volatiles from coolant to cover gas and then to RCB

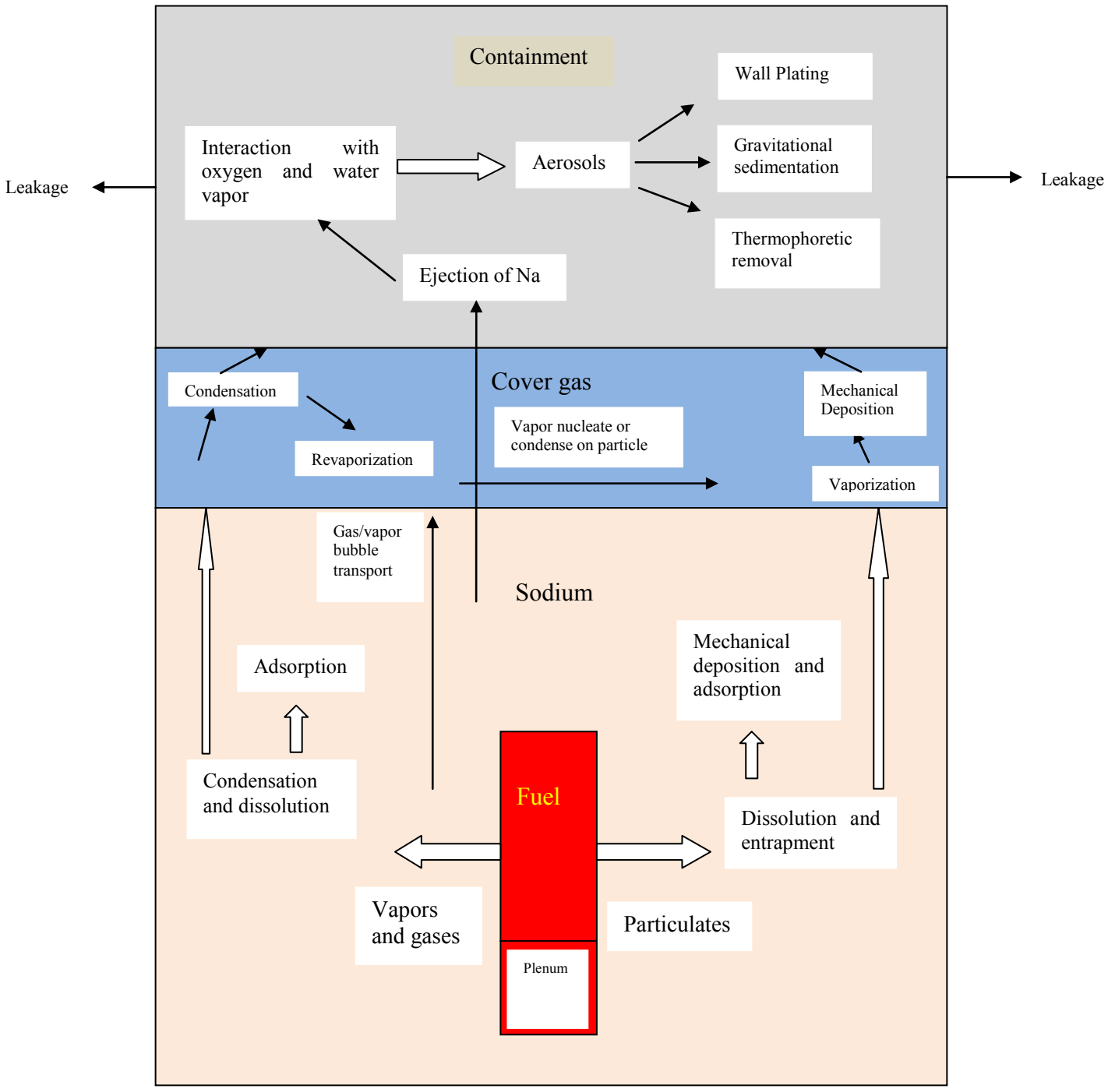

Fig. 5. Various processes for the release or retention of radio-nuclides [3,5,6]

Equivalently, if the retention fraction fractions in these volumes as a function of time are known, then the release fraction to RCB can be calculated.The RN amount will be a function of time in each of these volumes. For fast reactors the phenomena in the core and coolant regions are very different from that in PWR. However in containment phenomena for SFR and LWR are similar but for the pressure, temperature and chemical environment, the LWR models can be adapted for in containment phenomena. The RN can bedivided into the following groups as shown in Table IV based on their chemical similarity [2].

TABLE IV: RADIONUCLIDE GROUPS

No.	**Title**	**Elements in group**
1	Noble gases	Xe, Kr
2	Halogens	I, Br
3	Alkali Metals	Cs, Rb
4	Tellurium group	Te, Sb, Se
5	Barium, strontium	Ba, Sr
6	Noble metals	Ru, Rh, Pd, Mo, Tc, Co
7	Cerium group Rare earths	Ce, Y
8	Lanthanides &Actinides	La, Pu, Np

No detailed modelling effort is required for xenon and krypton as their release fractions are close to 1. The core inventory is given in Becquerels for all the isotopes. It is equal to their production rate for short lived nuclei and for long lived nuclei the activity will be less than their production rate.

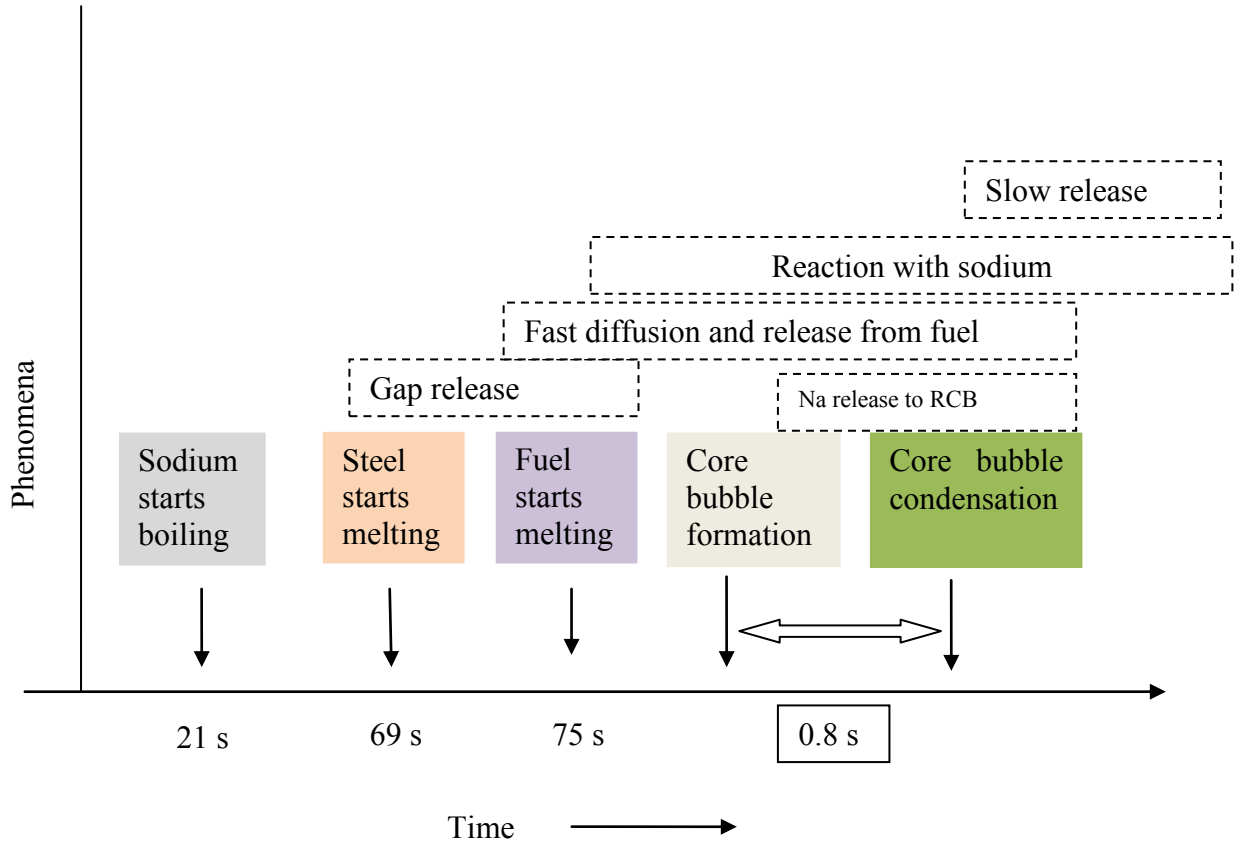

FIG. 6: Time sequence of major events during the CDA

Figure 6, presents the time sequence of the melting points of structure and fuel. It also presents the core bubble formation and condensation time interval. This figure helps to illustrate the release phenomena involved during the course of the accident.

4.1 Radionuclide release from fuel and transport in sodium

At present it appears that there is no modeling capability to address this question. However, the general understanding is follows [3-5]. There are many chemical and thermodynamic considerations that must be taken into account to predict the retention of the radionuclides in fuel and sodium. Vapors released from the fuel would have different retention depending on the vapor pressure and solubility in sodium. Vapors with high vapor pressure and low sodium solubility will be directly transported through the primary sodium and to the cover gas region. Some vapors may condense completely to the liquid phase once they come in contact with colder sodium and could dissolve, or they may nucleate within a bubble and be transported as aerosols. Other vapors will directly dissolve in the primary sodium from the gas phase due to high solubility of the element or compound in sodium. Adsorption of the dissolved vapors on the primary system structure is also possible, but this phenomenon is dependent on properties of the individual element or compound and the primary system structural materials.

Particulates that are released from the fuel may become entrapped within a vapor or gas bubble. These particulates may settle on or migrate to the surface of the bubble and interact with the sodium; the same outcome will occur if the bubble collapses as the vapor condenses when colder sodium is encountered. The particulates may dissolve in the primary sodium or become entrained in the moving sodium stream. The specific behaviour of the particulate depends on whether compounds are formed and its corresponding solubility in sodium. Subsequently, adsorption on structure may occur, especially in lower temperature regions of the primary system where dissolved radionuclides may precipitate. Mechanical deposition within the primary system is alsoa possibility, especially for entrained particles. An approximate estimate of the RN release from fuel to coolant can be obtained as a function of the fuel temperature and surface to volume ratio of the molten fuel mass.

4.2Radionuclide transport to cover gas

The transport of RN from sodium to cover gas can occur along any of the two major pathways. The vapors that are directly transported through the primary sodium in bubbles due to their high vapor pressure and low solubility will be released to the cover gas region upon reaching the sodium surface. If any entrained particles are present within these bubbles, they may also be initially released to the cover gas region when the bubble bursts at the surface of the sodium pool.

The vapors that are dissolved in the primary sodium, along with dissolved particulates, must vaporize in order to escape the sodium. The radionuclides that vaporize from the pool and reach the cover gas region will encounter a temperature decrease. This can lead to the condensation of vapors on structural surfaces or onto particles. Mechanical deposition can again remove particulates from the cover gas region, as particles impact the various structures present, or settle back onto the sodium pool due to gravity. Lastly, re-suspension or re-vaporization of some radionuclides is again possible if temperature changes occur in the cover gas region, as volatility typically increases with increasing temperature. Mechanically deposited particles could resuspend if a mechanical shock occurs on the structure where the particles are located, or if vapor flow increases adjacent to the structure.

4.3 Radionuclide release to RCB

Transport of RN to the RCB from cover gas and sodium can occur in the following possible ways. Since the core bubble during the quasi-static stage of expansion and condensation causes ejection of sodium on to the roof slab, during this period RN can be released into the containment along with the ejected sodium and cover gas. Much colder temperatures than the cover gas region is likely be found in containment, which will encourage additional condensation of vapors either onto the surface of structures, or onto aerosols. Mechanical deposition will once again reduce any particles that managed to successfully transport from the cover gas region into containment. Leakage from the cover gas region means that sodium vapor in the cover gas region may also enter containment. The sodium vapor will react with the oxygen and water vapor to form aerosol particles. These particles will agglomerate and may remove vapors or particulates that have condensed or mechanically deposited on their surfaces. Other radionuclides may also react with the oxygen and water vapor and decompose to form new compounds.

4.4 In containment phenomena

The source term in RCB depends on the extent of core damage, in-core phenomena and the mass of sodium ejected into the containment along with the contained radio nuclides, their radioactive decay and various removal processes in the RCB. Data from various analytical experiments and theoretical studies indicate that the fraction of noble gas release is nearly total. The burning sodium will form aerosols in the containment, whose size distribution is an important factor in the determination of removal rates through gravitational sedimentation, wall plating and thermophoresis. The radio nuclides may form their own aerosols and may coagulate on to the sodium aerosols [6,7]. For volatile fission products, iodine and cesium, vapor contribution to the source term is very small 2-4% compared to the iodine and cesium combined with liquid sodium aerosol contribution. In a simplified model, the assumption is made that within minutes of the release, the agglomeration process would proceed to reach a stage, which could be used to describe the evolution of the aerosol concentration of an average size through a rate equation with a rate constant which depends only on the radioactive isotope.

The pressure, temperature distribution inside the containment, moisture and the temperature gradient near the containment walls is important for the determination of the aerosol removal rates. Since the 'in containment' behaviour of the radio-nuclides is mainly governed by aerosol dynamics and the modelling experience available in LWR could be utilized here to a large extent. Some additional amount of sodium and radioactive particles released from the sodium pool will also eventually emerge into the reactor containment in the later stages. This fraction will depend on the nature of the flow path from the reactor coolant system to the containment after CDA.

5. IAEA Coordinated Research Project

The International Atomic Energy Agency launched, in 2016, a coordinated research project (CRP) to help study the source term due to CDA, named "Radioactive Release from the Prototype Fast Breeder Reactor under Severe Accident Conditions". Participants from nine different countries are contributing to this project, and the technical aspects to be addressed are subdivided into three work packages as follows.

WP-1: Expansion Phase: In-vessel Source Term
WP-2: Post-Accident Material Relocation (PAMR) Phase: Primary system/containment system interface source term
WP-3: Post Accident Heat Removal (PAHR) Phase: In-containment phenomenology

First is the in-vessel source term estimation, consisting of risk important fission product distribution in the fuel pins, their release mechanisms into the coolant and subsequent reaction and transport in the coolant and release to the cover gas. Second is the primary system/containment interface source term estimation consisting of models for the cover gas, sodium ejection and radionuclide chemical composition and distribution in the containment. The third part is the estimation of the fission product evolution within the containment, aerosol behavior, and physical boundary conditions. The countries participating in the coordinated research project (according to IAEA agreement) work packages is given below in Table V:

TABLE V: PARTICIPATING COUNTRIES AND WORK PACKAGES

Country	**Organization**	**WP1 Expansion Phase**	**WP2 PARM**	**WP3 PAHR**
Canada	UOIT			
China	CIAE			X
China	NCEPU		X	
China	XJTU			X
France	CEA	X		X
France	IRSN	X	X	X
Germany	KIT			X
India	IGCAR	X	X	X
Korea	KAERI	X		
Russia	IBRAE	X	X	X
Spain	CIEMAT			X
USA	TerraPower			X

6. Conclusion

Towards development of realistic mechanistic models for the assessment of SFR source term, detailed technical specifications have been developed for a reference SFR of 500MWe design. Various time scales and phenomena required for the modelling are brought out and discussed. The models required for the improved source term estimations are proposed to be developed based on existing experimental data from published literature, thermodynamic properties, simulation of underlying basic physical and chemical process and limited additional experiments.

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