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A numerical simulation method for molten material behavior in nuclear reactors

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HIGHLIGHTS

- A numerical simulation method based on computational fluid dynamics for calculating molten materials behavior is proposed.
- A new Poisson solver is developed which drastically improves the computational performance of the JUPITER algorithm.
- Verification and validation was performed by estimating fundamental problem characteristics and comparing with experimental data of molten material behavior. Projections obtained from JUPITER are in good agreement with these results.
- Our conclusion is that JUPITER has a potential to efficiently evaluate molten material behavior.

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ABSTRACT

In recent years, significant attention has been paid to the precise determination of relocation of molten materials in reactor pressure vessels of boiling water reactors (BWRs) during severe accidents. To address this problem, we have developed a computational fluid dynamics code JUPITER, based on thermal-hydraulic equations and multi-phase simulation models. Although the Poisson solver has previously been a performance bottleneck in the JUPITER code, this is resolved by a new hybrid parallel Poisson solver, whose strong scaling is extended up to ~200 k cores on the K-computer. As a result of the improved computational capability, the problem size and physical models are dramatically expanded. A series of verification and validation studies are enabled, which are in agreement with previous numerical simulations and experiments. These physical and computational capabilities of JUPITER enable us to investigate molten material behaviors in reactor relevant situations.

1. Introduction

During the accident at the Fukushima Daiichi Nuclear Power Station (1F), loss of backup power stopped operations of the emergency core cooling system, causing fuel rods to overheat due to uncontrolled heating from radioactive decay and oxidation of the fuel cladding. Apparently, core degradation occurred because fuel rods, control rods (CRs), and other components in the reactor vessel melted and relocated, interacting with the complex internal core in the reactor pressure vessel (RPV). The complex internal core consists of the core support plate and control rod guide tubes (CRGTs). As a part of the decommissioning processes of 1F, information about the current status of relocated molten materials is required in order to remove debris from reactors. However, because of high-level irradiation caused by debris and fission products (FPs) in the reactor building, on-site investigation becomes very difficult.

In such cases, it is considered that numerical simulations might be useful. After the 1F accident, the Modular Accident Analysis Program (MAAP) (Fauske and Associates, 1994–2005) and the molten core relocation analysis (MCRA) module, which is one of the modules of the severe accident (SA) analysis code SAMPSON (Ujita et al., 1999; Satoh et al., 2000), were used to understand the progress of the 1F accident, as well as the molten material relocation in the RPV. However, it is difficult to obtain information about the current status of debris and FPs by these existing SA analysis codes because of two main reasons.

After the accident of Three Mile Island unit 2, the development of existing SA analysis code has been initiated to evaluate the efficiency of countermeasures for SAs, and the impact of SAs on the environment. Therefore, the existing SA analysis codes have been developed for purposes of evaluation of the degrees of

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Nomenclature

a	absorption coefficient [1/m]
C_v	specific heat [J/kg/K]
D	representative length [m]
E	thickness of the solidified region [m]
$\boldsymbol{f}^{\text{IBM}}$	momentum forcing [N]
$\boldsymbol{F}$	external force [N]
$\boldsymbol{F}_S$	surface tension force [N]
$Fl_{i\pm1/2}, Fl_{j\pm1/2}, Fl_{k\pm1/2}$	VOF flux on i, j, k-th control volume surface for x, y, z direction [m]
$\boldsymbol{g}$	acceleration due to gravity [m/s^2]
h	heat transfer coefficient [W/m^2/K]
$H(\psi)$	heaviside function [–]
I	radiance [W/m^2/sr.]
L	latent heat [J/kg]
$\boldsymbol{n}$	normal vector [m]
$\boldsymbol{n}$	normal vector for radiance direction [–]
p	pressure [Pa]
Q	heat source [W/m^3]
Q_s	latent heat released from the solid phase to the liquid phase [W/m^3]
t	time [s]
T	temperature [K]
T_b	temperature of boundary at $x = 0$ [K]
T_f	phase-change temperature [K]
T_L	liquidus temperature [K]
$\boldsymbol{u}_f$	velocity $\lvert\psi_{\boldsymbol{f}}\rvert$ distance from body interface [m/s]
$\boldsymbol{u}_g$	velocity at the virtual point inside the body [m/s]
$\boldsymbol{u}$	velocity vector [m/s]
U	representative velocity [m/s]
V	volume of the control volume [m^3]
$\boldsymbol{x}$	position vector [m]

Greek letters

α	thermal diffusivity [m^2/s]
β	interface thickness parameter for in the THINC method [–]
$\Delta(\psi)$	delta function [m^{-1}]
ε	half width of thickness of the interface [m]
$\Delta\phi$	amount of the phase change [–]
Δx	grid width for x-direction [m]
Δy	grid width for y-direction [m]
Δz	grid width for z-direction [m]
κ	curvature [m^{-1}]
λ	thermal conductivity [W/m/K]
μ	viscosity [Pa s]
ρ	density [kg/m^3]
σ	surface tension coefficient [N/m]
$\tilde{\sigma}$	Stefan-Boltzmann constant (5.67037×10^{-8}) [W/m^2/K^4]
ϕ	volume of fluid function [–]
ψ	level-set function [m]
$\lvert\psi_g\rvert$	distance of normal direction between $\boldsymbol{u}_g$ and the body interface [m]
$\lvert\psi_f\rvert$	distance of normal direction between $\boldsymbol{u}_f$ and the body interface [m]
Ω	steradian sr.

Subscripts

i	i-th grid point
j	j-th grid point
k	k-th grid point
l	l-th component

Superscripts

G	physical property of gas phase
L	physical property of liquid phase
n	n-th time step
S	physical property of solid phase

development of SAs, not including detailed information about the relocation of melted debris, as in flow pattern of the melt relocation, and the complicated interfaces of the melted debris. Furthermore, the Three Mile Island unit 2 was a pressurized water reactor (PWR). Consequently, prior to the 1F accident, the existing SA codes mainly focused on PWRs, while experimental and numerical information for SAs in BWRs were limited. The BWRs consist of a lot of core internals with complicated structures, such as channel boxes, core support structures, and CRGTs. In the numerical simulations constructed to evaluate the current status of debris in the 1F, differences between PWRs and BWRs, including effects of these complicated structures, must be taken into account.

Therefore, to obtain precise estimates of molten material behavior in the RPVs and to improve the accuracy of the SA analysis code, we have constructed a computational fluid dynamics (CFD) code that can predict molten material relocation in the RPVs. We call this code the JUPITER (JAEA Utility Program with Immersed boundary Technic and Equations of multi-phase flow analysis to simulate Relocation behavior of molten materials). JUPITER can predict molten material behavior, including solidification and melt relocation, based on three-dimensional, multi-phase and thermal-hydraulic simulation models. These simulation models provide unified and detailed analysis of molten material relocation from the core to the lower head. It is noted that a part of these simulation models have been already implemented in existing commercial CFD codes, e.g., for casting simulations. However, the computational performance of such CFD codes is limited by the scalability of a Poisson solver, and extreme scale simulations of molten material relocation in long time scale SA processes covering the whole RPVs are prohibitive. To our knowledge, unified and detailed analysis of molten material behavior on the entire RPVs scale based on CFD approaches has not been conducted to date. Although application of the CFD approach to SA phenomena may be cumbersome, it could provide valuable and important knowledge for unrevealed behavior of molten material. In order to resolve this critical issue, in this paper, we develop a new hybrid parallel Poisson solver for JUPITER, and address systematic verification and validation studies against former numerical simulations and experiments.

To determine detailed and unified molten material behavior of the reactor core containing complex structures from fuel assemblies to the lower plenum, we consider the following features and hypotheses in JUPITER.

Features;

- Complicated structures in the RPV, core support plates, fuel assemblies, and CRs are implemented using a simple numerical model for solids.
- Interfaces between each fluid component are captured by advanced numerical interface-capturing schemes. This approach reduces computational costs and can capture interfaces sharply and efficiently.