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# Modeling Requirements for VHTR Thermo-Fluid and Safety Analysis Codes

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## 1. Introduction

KAERI is carrying out a key technology development project for Nuclear Hydrogen System using a Very High Temperature gas cooled Reactor (VHTR). One of the major objectives of the project is to develop a thermo-fluid and safety analysis codes system for VHTR design applications. Even though there have been various codes developed over the past 30 years, most of them are based on 70's technology, thus, may not fulfill the desired accuracy and requirements of current licensing environment. In this context, many countries are developing their own VHTR code systems based on advanced modeling and programming techniques using existing water reactor (PWR) codes as backbones.

KAERI's codes system under development consists of; MARS-GCR [1] for system thermo-fluid and safety analysis, GAMMA [2] for reactor vessel thermo-fluid and air-ingress analysis, LILAC [3] for computational fluid dynamics, and MIDAS-GCR [4] for fission product and dust transport. All except for GAMMA use backbones originally developed for PWR applications, thus, they are improved and extended versions for the VHTR applications. These codes are being developed based on the code modeling requirements that were derived from the study on the VHTR phenomena identification and ranking tables (PIRTs) generated under KAERI/ANL/ INL I-NERI project [5]. This paper identifies the code modeling requirements and addresses their physical importance. It also introduces the current developmental status of VHTR codes system at KAERI.

## 2. Identification of Code Modeling Requirements

PIRT identifies the major phenomena by components for each time-phase of a nucleus set of events. Thus, PIRT provides a technical basis for a code development and validation, that is modeling requirements, as well as the construction of experimental database and needs. Under the I-NERI project, KAERI, ANL and INL identified major phenomena for the limiting transients and accidents of a VHTR; high pressure conduction cooling (loss of forced flow), low pressure conduction cooling (loss of coolant followed by air-ingress), load changes, anticipated transients without scram (ATWS, reactivity insertion), water-ingress and hydrogen-side upset events. From the PIRTs, we screened the major phenomena that were ranked as high and selected them as the code modeling requirements for VHTR analysis.

Following describes the major phenomena that should be modeled or approximated in the codes.

<u>Multi-dimensional fluid dynamics</u>

Not only during a steady state operation but also depending on the progress of an event, non-uniform and asymmetric flow, that is, multi-dimensional flow occurs in the core and plenums. It governs local hot spot in the core, and flow mixing, hot plume rise, stratification and striping in the plenums. Natural convection inside reactor cavity that partly contributes to core afterheat removal also requires a multi-dimensional fluid modeling. For such phenomena, multi-dimensional fluid dynamics model is a prerequisite in the codes.

<u>Flow distribution</u>

Reactor vessel internals consists of graphite reflector blocks and fuel blocks/pebbles. Graphite shrinks and swells by temperature and neutron fluence over life. Such dimensional changes of graphite internals affect the sizes of the gaps and bypass flow areas, that is, the effective core flow which determines the core power capability. In addition, during a high pressure conduction cooling event, fuel heat-up slows down by a natural circulation cooling inside the reactor vessel that is highly dependent on the core bypass configuration. Thus, it is important to accurately model the dimensional changes of graphite internals as a function of burn-up and temperature.

<u>Coolant Properties</u>

Nuclear hydrogen system incorporates helium as a primary coolant, while there is a potential of using helium, helium/nitrogen mixture or molten salt in the intermediate loop. Since the gas and molten salt properties vary widely within the temperature ranges expected to occur during transients, an accurate prediction of coolant properties is a basic requirement for analyzing the flow dynamics and heat transfer.

<u>Spatial Power Distribution and Reactivity Feedback</u>

Even during normal operations, large temperature rise in the core and a variation of local burn-up necessitate the modeling of a spatial reactivity feedback. During the progress of an event, the temperature variation in the heterogeneous core internals affects the spatial reactivity feedback and subsequent power distribution. Especially, during the ATWS event, local reactivity feedback by Xe decay and temperature redistribution affects the time to a recriticality. Thus, it is required to model the spatial kinetics with a detailed reactivity feedback, that is, to have a coupled analysis capability.

<u>Convection Heat Transfer</u>

During normal operations, core convection heat transfer falls into a forced convection. However, it is more likely to be in a mixed or free convection during accident conditions, in which a buoyancy effect plays an important role. Literature survey indicates that there is no sufficient database available for a complete set of heat transfer package for applicable ranges. It also puts an emphasis on the experimental needs.

*Radiation Heat Transfer*

Outstanding safety feature of VHTR is that core afterheat can be removed by natural phenomena even without a forced cooling and operator action. Core afterheat is transferred to the reactor vessel wall by a conduction and radiation inside the reactor vessel. It is, then, transferred to the reactor cavity cooling system mainly by the radiation from the wall. Thus, radiation heat transfer is the major heat removal mechanism that has to be modeled.

*Multi-dimensional Heat Conduction*

Accurate prediction of local temperature distribution in a complex geometry of the reactor vessel internals is to ensure a local hot spot, reactivity feedback and core afterheat removal. Prismatic fuel conducts heat from the fuel compact to the flow channel multi-dimensionally, whereas pebble fuels contact each other to conduct heat. Piled-up fuel and reflector blocks conduct heat axially by a contact heat transfer. Radial heat transfer through gaps between blocks is by a radiation whose amount is determined by the surface temperature. If a radial contact of blocks occurs by their dimensional changes, a contact heat transfer should be taken care. Moreover, graphite material properties vary with the burn-up.

*Graphite Chemical Reaction and Multi-Species Gas*

In case of a break in the primary boundary, air or water may ingress into the reactor vessel. Graphite fuels and internals chemically react with air or water, then, produce various gases which react again with bulk air or product gases. Some exothermic reactions add heat to the core, which may result in an undesired core heat-up. Thus, in order to predict air or water ingress events, specific models for a graphite oxidation and hydrolysis and models for a multi-species gas flow are required.

*System Component Models*

Nuclear hydrogen system layout consists of a primary coolant loop, intermediate loop and thermo-chemical hydrogen production process. Alternative layout option is to produce hydrogen by using high temperature electrolysis using electricity generated by gas turbo-machinery. In order to accurately predict overall system performance and transient response, system components models such as circulators, heat exchangers, turbo-machinery components are necessary.

*Fission Product and Dust Transport*

During normal operations, fission products and graphite dusts released from the fuel and core internals plate-out at various locations of a system components. It affects the radiation level for a controlled maintenance. In case of a primary boundary break, the lift-off of them determines the source terms for the fission product and aerosol transport inside the reactor building. Since VHTR adopts a confinement concept which admits the controlled vent, it is very important to accurately model the source terms and their transport inside the confinement.

*Tritium Production and Permeation*

Tritium is generated by various nuclear reactions such as He-3(n,p)T, ternary fission, Li-6(n,$\alpha$)T, B-10(n,2$\alpha$)T, B-10(n,$\alpha$)T, Li-7(n,n$\alpha$)T. Some of Tritium accumulates in the primary helium coolant and a fraction of Tritium in the primary coolant permeates through the heat exchanger walls and may contaminate the environment and the product hydrogen. A Tritium contamination may raise a potential radiation hazard to plant workers and consumers and must be controlled to an acceptable level. Thus, it is required to accurately model a tritium production and permeation.

## 3. Conclusions

Modeling requirements for VHTR thermo-fluid and safety analysis codes have been identified and discussed for their physical importance. Based on the code modeling requirements, KAERI has been developing a codes system as shown in the following table. We will further develop and validate the codes for application to the design of a VHTR and nuclear hydrogen system.

Table. Developmental Status of KAERI Codes System

| Modeling Requirements | MARS | GAMMA | LILAC | MIDAS |
|---|---|---|---|---|
| Multi-D Fluid Dynamics | O | O | O | Δ |
| Flow Distribution (Bypass) | Δ | Δ | Δ | Δ |
| Coolant Properties* | Δ | Δ | Δ | Δ |
| Coupled Spatial Kinetics | O | X | n/a | n/a |
| Convection Heat Transfer | O | O | O | X |
| Radiation Heat Transfer | O | O | O | n/a |
| Multi-D Heat Conduction | Δ | Δ | O | Δ |
| Graphite Chemical Reaction | n/a | O | n/a | n/a |
| System Component Model | Δ | X | n/a | X |
| FP and Dust Transport | n/a | n/a | n/a | Δ |
| Tritium Transport** | n/a | n/a | n/a | X |

* Molten Salt properties need to be implemented

** TRITGO is currently used

O: completed, Δ: partially implemented, X: planned

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