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Technical note

Application of sub-channel modeling to BWR core analysis

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ABSTRACT

The current trends in nuclear science and engineering, related to modeling and simulation, are towards high-fidelity multi-physics multi-scale simulations to address industry challenge and high impact problems. These trends stimulate the utilization of sub-channel modeling approaches to nuclear reactor cores for both operation applications (core follow and cycle depletion evaluations) and safety applications (transient and accident analysis). While sub-channel modeling of Pressurized Water Reactor (PWR) cores has advanced significantly in last few years, the application of sub-channel modeling to Boiling Water Reactor (BWR) cores is under development. CTF is an improved version of COBRA-TF being developed and maintained by the North Carolina State University (NCSU) in cooperation with Oak Ridge National Laboratory (ORNL), and with support of the US Department of Energy (DOE) Consortium for Advanced Simulation of Light Water Reactors (CASL) as well as from the members of the CTF User's Group. CTF uses a two-fluid, three-field representation of the two-phase flow, which makes it capable of modeling the high-void flow conditions expected in BWR operation. This paper focuses on applications of CTF to mini- and whole-core BWR calculations on assembly/channel and pin-cell/sub-channel resolved levels as well as on demonstrating that CTF can properly model bypass flow. To increase the confidence in CTF's BWR modeling capabilities, simulations have been performed using the international Organization for Economic Cooperation and Development (OECD)/US Nuclear Regulatory Commission (NRC) Oskarshamn-2 benchmark, including modeling of a single assembly and a mini-core of 2×2 assemblies on a pin-by-pin/sub-channel level, and a full core model on an assembly/channel level. Each model is varied with an increasing amount of detail. Key parameters such as pressure losses and void fraction distribution were analyzed to determine the impact of different levels of detail within a thermal-hydraulic model on the simulation results. The results demonstrated that CTF is capable of modeling BWR core on different spatial resolution levels. The Oskarshamn-2 core simulations was used to further verify and demonstrate CTF's capabilities of modeling BWRs.

1. Introduction

The current trends in nuclear science and engineering, related to modeling and simulation, are towards high-fidelity multi-physics multi-scale simulations to address industry challenge and high impact problems. These trends stimulate the utilization of sub-channel modeling approaches to nuclear reactor cores for both operation applications (core follow and cycle depletion evaluations) and safety applications (transient and accident analysis). While sub-channel modeling of Pressurized Water Reactor (PWR) cores has advanced significantly in last few years, the application of sub-channel modeling to Boiling Water Reactor (BWR) cores is under development.

CTF is a Thermal-Hydraulic (TH) sub-channel code, which is an improved version of the computer code COBRA-TF (Thurgood et al., 1980) originally developed by the Pacific Northwest National Laboratory (PNNL) in the 1980s. The released public domain code version was further developed and improved by the Reactor Dynamics and Fuel Modeling Group (RDFMG) at the North Carolina State University (NCSU) as CTF for power and research reactor applications. The code is being maintained, improved, and distributed within the CTF User's Group activities to become a modern and robust TH reactor core simulator. The mandate of CTF User's Group, hosted by NCSU, is to leverage and combine all non-proprietary developments, improvements, modifications and error fixes; and the available verification and validation database and application experience of CTF from different organizations and activities. CTF is currently being jointly developed, improved, verified and validated by NCSU and Oak Ridge National Laboratory (ORNL) for

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applications in the US Department of Energy (DOE) program Consortium for Advanced Simulation of Light Water Reactors (CASL). The code provides a three-dimensional two-fluid three-field representation of the two-phase flow, where the liquid phase is subdivided into a continuous field and an entrained liquid drop field. Therefore, for each spatial dimension, CTF solves three momentum conservation equations, three mass conservation equations, and two energy conservation equations (Avramova et al., 2006). A thermal equilibrium is assumed between the continuous liquid field and the droplet field (Todd et al., 1999), leading to a set of eight equations solved for the fluid portion. Significant work has been done on verification and validation of CTF for BWRs (Salko et al., 2014, 2016). Especially important is the code validation using the Organization for Economic Cooperation and Development (OECD)/US Nuclear Regulatory Commission (NRC) Boiling Water Reactor Full-length Bundle Test (BFBT) benchmark.

The ability to correctly predict pressure losses in two phase-flow is vital to modeling of BWRs. This includes effects from wall drag, interfacial drag, and form losses due to the presence of spacer grids. CTF includes a two-phase pressure drop model based on the work of Wallis, and grids are treated using simple velocity head losses (Wallis, 1969). Spacer grids also have an impact on rod heat transfer due to enhanced turbulence and boundary layer disruption within and downstream of the grid (Magedanz et al., 2012). In very high void conditions, entrained droplets are broken up into smaller drops, which increases the droplet surface area and interfacial heat transfer. Additionally, rewetting of the spacer grid in accident conditions leads to cooling of the superheated vapor flowing through the core (Park et al., 1986). An analysis of steady state and transient void distribution predictions for Phase I of the OECD/NRC BFBT benchmark using CTF is presented in Avramova et al. (2007). The CTF validation to the OECD/NRC BFBT benchmark single- and two-phase pressure drop exercises has proven the code capabilities of predicting pressure losses in a BWR environment (Avramova et al., 2011).

This paper discusses the results obtained from three different BWR CTF models: single assembly on a pin-cell (sub-channel) resolved level (Model 1), 2×2 array on a pin-cell (sub-channel) resolved level (Model 2), and full core on an assembly-cell (channel) resolved level (Model 3). Each model had three levels of detail that investigated the effects of internal flow, external flow, and flow inside water rods. All tests were modeled at steady state conditions and follow the OECD/NRC Oskarshamn-2 BWR Benchmark Specifications (Javier and Perin, 2014).

2. CTF models

The developed three CTF models with different spatial resolution are presented in Table 1 along with the operating conditions. Table 2 shows the fidelity and detail of each model. The internal bypass was defined as a bypass region between assemblies, while the external bypass was defined as a bypass region that surrounds the assembly. The water channel was defined as a bypass region that is located at the center of the assembly. This modeling approach acts similar to a water rod, but is inherently a bypass region. Finally, Table 3 lists all assumptions used for each model.

Table 2
Models Variation.

	Internal bypass	External bypass	Water channel bypass
Base case	Not included	Not included	Not included
Bypass case	Included	Included	Not included
Water Channel	Included	Included	Included

2.1. Single assembly on a pin-cell resolved level

The first and simplest model was a single assembly consisting of 91 fuel rods (8 partial and 83 full rods), shown in Fig. 1. The rods colored red indicate a partial rod, which is a fuel rod that is not the full rod length. The figure shows the difference between the rod types. An ATRIUM-10 assembly from Oskarshamn-2 specifications was used for all input values (Magedanz et al., 2012). The model had three variations, shown in Fig. 2. The initial model consisted of just the assembly, with the area the water rod acting as a solid adiabatic surface. Each subsequent variation added detail to the previous model. The first addition was a bypass that surrounds the assembly. The next addition was a water rod, which was modeled as a water channel bypass, at roughly the center of the assembly that took the place of 9 fuel rods.

CTF was able to model partial rods in two different structures. Since BWRs typically contain partial rods that are paramount to the design, it was important that simulations are created to model them as close to realistically possible. This single ATRIUM-10 assembly model had two axial sections: one containing all 91 rods, and the second upper section containing only the tops of the full-length rods.

2.2. 2×2 Array assemblies on a pin-cell resolved level

The second model was an expansion of the previous model. The single assembly was expanded into a 2×2 array following the same path of detail as shown below in Fig. 3. Since the model contained multiple assemblies, the bypass was split up into internal and external sections. Each bypass section was created the same way as was done for the single assembly case. However, instead of splitting the rods into two axial sections, the partial length rods were captured using the axial geometry variation feature of CTF. By changing the size of the channels and gaps around the partial rods at a specific height (in this case the point at which they end), it effectively creates a new area without splitting the rods into two sections. The expansion modeling can be seen in Fig. 4, note that this method will show little change in the overall pressure loss, but local flow and enthalpy distributions may be slightly different. Fig. 4 depicts the two different possible modeling techniques used for this simulation.

2.3. Full core on assembly-cell resolved level

The CTF full core model was built based on the specifications in the Oskarshamn-2 benchmark package and represents the core on an assembly (channel)-level rather than in a pin-by-pin (sub-channel) level. Therefore, the fuel pins for each assembly were

Table 1
Operating Conditions.

	Pressure (bar)	Linear Heat Rate (kw/m)	Number of Assemblies	Resolution	Partial Rod
Model 1	70.2	14.8946	1	Pin-cell level	Two sections*
Model 2	70.2	14.8946	4	Pin-cell level	Geometry variation
Model 3	71.66	15.8319	444	Assembly-cell level	Two sections

* See Fig. 4, Section 2.2.