Computer Code Abstract

Liquid-Metal-Cooled Reactor Thermal-Hydraulic Calculations in the United States

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Abstract – *A wide range of thermal-hydraulic computer codes has been developed by various organizations in the United States. These codes cover an extensive range of purposes from within-assembly-wise pin temperature calculations to plantwide transient analysis. The codes are used for static analysis, analysis of protected anticipated transients, and analysis of unprotected transients for the more recent liquid-metal-cooled reactor designs. Some of these codes are plant specific; others are more general and can be applied to a number of plants or designs.*

INTRODUCTION

Computer codes for within-assembly-wise coolant and pin temperature calculations, plantwide transient analysis, and multidimensional analysis of individual components have been developed in the United States. Some of these codes were developed for only a specific plant, whereas others were developed for more general usage. These codes, and the purposes for which they have been used, are described.

SASSYS-1

The SASSYS-1 liquid-metal-cooled reactor (LMR) systems analysis code[1] was developed at Argonne National Laboratory (ANL) primarily to analyze the consequences of failures in the shutdown heat removal system. However, the code is capable of handling a wide range of transients from minor operational transients to severe accidents leading to sodium boiling and disruption of the fuel pins. The code has mainly been used for analysis of operational transients and analysis of inherent response to unprotected transients in Integral Fast Reactor[2] (IFR), Sodium Advanced Fast Reactor[3] (SAFR), and Power Reactor Inherently Safe Module[4,5] (PRISM). The code uses a detailed multichannel core thermal-hydraulic treatment coupled with a point kinetics neutronics treatment; a general thermal-hydraulic treatment for the primary and intermediate heat transport loops and shutdown heat removal systems; a steam generator (SG) treatment; and a general control system treatment. It can handle any LMR configuration, pool, or loop. The code has been validated by comparison with Experimental Breeder Reactor II (EBR-II) natural circulation tests.[6]

SASSYS-1 uses a detailed multichannel core treatment from the SAS4A code.[7] A channel usually represents a subassembly or a group of similar subassemblies. Each channel contains a fuel pin, its associated coolant, and a structure representing wrapper wires and/or a share of the duct wall. Temperatures are calculated for fuel, cladding, coolant, and structure at a number of axial nodes. A channel can represent an average pin within a subassembly or it can represent a hot pin. For the analysis of severe accidents, SASSYS-1 couples directly with the SAS4A modules for phenomena associated with core disruption, including fuel and cladding relocation after melting or pin disruption. The coolant flow rate calculation for each channel is driven by pressure boundary conditions at the subassembly inlet and outlet, so the redistribution of flow between subassemblies at low flow rates is automatically accounted for.

The neutronics calculations in SASSYS-1 include the feedback effects necessary to analyze inherent safety. The reactivity feedbacks calculated include Doppler, sodium density or voiding, fuel thermal expansion, fuel and cladding relocation, control rod drive expansion, vessel wall expansion, radial expansion, and bowing. Decay heat is also accounted for using a treatment with up to six decay heat precursor groups to calculate transient decay heat power levels.

SASSYS-1 uses a general thermal-hydraulic treatment for the primary and intermediate heat transport loops. Liquid segments, gas segments, and compressible volumes are used to model an arbitrary arrangement of components. Liquid segments contain liquid flow elements and are characterized by incompressible flow. Elements represent pipes, pump impellers, intermediate heat exchanger (IHX) shell and tube sides, and the sodium side of an SG. Compressible volumes contain liquid and/or gas. They are characterized by pressure and mass. Compressible volumes represent inlet and outlet plenums, a pool, a pump bowl, or a pipe junction. Gas segments represent pipes between cover gas volumes. Gas segments are treated by isothermal gas flow.

SASSYS-1 models shutdown heat removal systems. These systems include a reactor vessel auxiliary cooling system[4] or reactor air cooling system[3] and a direct reactor auxiliary cooling system.[3]

The SASSYS-1 control system uses control blocks connected in an arbitrary manner to control pump motor torque, control rod reactivity, SG feedwater flow rate, and feedwater enthalpy. Almost any variable calculated by SASSYS-1 can be used as input to the control system.

SASSYS-1 can handle tasks that otherwise sometimes require coupling between two or three separate codes. For example, one code can be used for loop thermal hydraulics, a second code for core flow and coolant temperature calculations, and a third code for fuel pin temperatures. SASSYS-1 directly couples all of these calculations together in a single code.

As mentioned above, even though SASSYS-1 was developed to analyze shutdown heat removal with emphasis on natural circulation and decay heat removal, the code has actually been used for a wide range of cases. The main applications have been to operational transients and unprotected loss-of-flow (LOF), transient overpower, and loss-of-heat sink (LOHS) cases for inherently safe designs. For the unprotected accidents, the purpose of the calculations was to determine whether inherent feedbacks are sufficient to prevent unacceptable coolant and structural temperatures without control rod action.

SASSYS-1 runs relatively fast, usually faster than real time. This speed is necessary since some of the transients resulting from unprotected accidents in inherently safe reactors can run for hours or days of accident time before temperatures peak. For long, slow transients, time step sizes in the 1- to 20-s range can be used. Speeds of 10 to 30 times as fast as real time are obtained on an IBM-3033 computer. On a Cray-XMP computer, speeds of 180 times as fast as real time can be achieved.

NATDEMO

Although it was developed by ANL in a general form to analyze either a pool or loop reactor system, the NATDEMO code[8] has thus far been used exclusively on EBR-II. This code describes the thermal, hydraulic, and neutronic behavior of any LMR plant with a basic layout similar to EBR-II. The primary, secondary, and steam systems are described, although the condenser and feedwater train have not yet been included in the simulation model. The primary system model was developed independently of the balance-of-plant (BOP) model. The BOP simulation was taken from the DEMO-IV code, by adapting the secondary and steam system models to interface at the IHX.

Some improvements were made to the BOP model from DEMO-IV, notably the heat capacity/transport description of the secondary system piping, before incorporating it into NATDEMO. The code also includes a large number of plant control systems, but does not include phase changes of coolant or fuel, or component ruptures leading to loss-of-coolant accidents. The code can easily initialize all variables to any desired state that can be defined in terms of plant variables including reactor power, primary flow rate, reactor inlet temperature, turbine inlet pressure, turbine/bypass mode, and feedwater temperature.

Important features of the primary system model include detailed descriptions of power generation, parallel-channel thermal hydraulics with bouyancy, detailed reactivity feedback, reactor inlet and outlet piping, the IHX, the main primary pumps, the auxiliary pump, and the primary sodium pool. The superheaters, evaporators, the secondary electromagnetic pump, and the hundreds of metres of secondary sodium piping are included in the secondary system model. The steam system portion of NATDEMO includes a thermal-equilibrium drum model, the hydraulics for the natural convection recirculation flow through the evaporators, the steam pressure control system, the steam line and turbine, and the turbine bypass line to the condenser.

Because DEMO-IV was originally written for the CDC-7600 computer, the DEMO module was converted to the IBM-370 in double precision to approximate its computational accuracy with the Control Data Corporation (CDC) machine. In the overall NATDEMO code, the primary system module contains the main and controlling portion of the program, while the BOP module is treated as a dynamic subprogram. A special subroutine was developed to ensure proper time sequencing of the two schemes and to allow both modules to vary their computational step sizes independently during the solution. A typical problem on the IBM-3033 runs at 1.5 to 3.0 times real time, depending on the choice of options and convergence requirements.

Experimental data obtained since the original NATDEMO model was produced have validated the basic code and several improvements made later for a large variety of LOF with scram, LOF without scram, and LOHS without scram transients. Of particular importance in recent EBR-II tests and validation studies[9] has been the reactivity feedback model that describes nine effects, including subassembly bowing, grid expansion, and control rod bank expansion.

DEMO-IV

The DEMO-IV code[10] is the latest published version of the DEMO series to perform transient analyses of the Clinch River Breeder Reactor Plant (CRBRP). Although it was written specifically for CRBRP, an improved version has also been used—after some modification—to verify other code predictions of the natural circulation performance (following a scram) of the Fast Flux Test Facility (FFTF). The DEMO-IV code contains sufficient detail to handle typical CRBRP operational transients and failure events; it is written in FORTRAN-IV for the CDC-7600.

The DEMO program normally represents two plant loops. One program loop ("S-loop") represents a single plant loop used to model single-loop perturbations. The remaining two plant loops (or one for two-loop operation) are represented by the remaining program loop ("L-loop") in a lumped fashion. For special cases, hydraulics (only) for the third loop can be represented.

The reactor model provides average channel representations in the core and radial blanket. Separate axial blanket sections are modeled. In the published DEMO-IV version, core radial blanket and bypass flow split assume a fixed ratio. The reactor core model in DEMO provides only for single-phase liquid sodium flow. The program calculates the sodium saturation temperature at the active core exit to allow the user to determine when two-phase conditions are reached. An upper plenum stratification model is provided for those cases in which a rapid decrease in flow occurs along with a decrease in core exit temperature. This model assumes that the upper plenum fills from the bottom with the colder sodium after the core exit jet height decays.

The DEMO SG model provides heat transfer based on subcooled, boiling, or superheat conditions. Perfect separation is assumed for fluid leaving the drum. Feedwater mixing takes place in a separate node just below the drum. This node thus represents the mixing zone as if it is a small mixed volume at the drum bottom. The SG model represents mass flow as the same through all nodes of a module model. Node-to-node fluid expansion, however, can be modeled in a special, more elaborate subroutine (S-loop only) to provide more accuracy for cases such as steam pipe ruptures. The recirculation loop contains a pump model, but it can also handle purely natural convection flow to the steam drum.

Several improvements were made to the code after the DEMO-IV version was published. As mentioned above, one modification/option was a dump heat exchanger (substituted for the steam system) to allow the FFTF BOP to be simulated. Other improvements include

1. heat capacity and heat transfer in secondary piping
2. flow redistribution in the reactor due to channel bouyancy and pressure drop characteristics
3. detailed homologous pump models
4. initializer to calculate any desired steady-state condition
5. feedwater train in the BOP
6. cover gas systems in the primary system
7. nonequilibrium drum model option for transients with rapid pressure changes.

A description of these modifications can be found elsewhere.[11]

DYNAMIC SIMULATION FOR NUCLEAR POWER PLANTS

The Dynamic Simulation for Nuclear Power Plants (DSNP) is a system[12] of programs and data sets by which a nuclear power plant or part thereof can be simulated at different levels of sophistication. The acronym DSNP is used interchangeably for the DSNP language, precompiler, libraries, and document generator. The DSNP language is a set of simple block-oriented statements, which together with the appropriate data, comprise a simulation of a nuclear power plant. The majority of the DSNP statements will result in the inclusion of a simulated physical module into the program. The DSNP precompiler is a FORTRAN program that will interpret the DSNP statements, rearrange the appropriate data sets, search the libraries for the requested modules, and produce an executable FORTRAN program—the plant simulation. The DSNP libraries include models of most of the components found in nuclear power plants. There are three libraries of modules each having a different level of sophistication. A level-four library is also available to include user-supplied modules.

A powerful feature of DSNP is the flexibility available to the user in arranging the configuration of components describing a reactor system. Each module or block used represents a component of the system. Combination of blocks allows the components to be assembled to form a connected system. Variables and definitions used in modeling each block are combined by the DSNP precompiler program. The modeler may rapidly build a system from the models available in DSNP by defining a small number of variables required for the input in each block and by equivalencing the variables used in connected blocks. Therefore, DSNP is a flexible user-oriented simulation language used to model neutronics and thermal-hydraulic effects in a reactor system.

The DSNP package can be used on several computer systems, including the IBM-370, APOLLO, and DEC/VAX (both the VMX and UNIX operating systems). Efforts are also under way to convert it to the CDC machines.

Although the code has been used worldwide to simulate light water reactor and LMR systems (as well as nonnuclear systems), most of its use in the United States has been at ANL. Here the code has principally been used to analyze EBR-II and the Argonne large pool conceptual LMR design. These analyses have included primary flow and inlet temperature transients, pump coastdown transients, and rod drop transients—all at power conditions. Because DSNP has successfully predicted the response to these reactor tests, many of the available libraries have been validated with EBR-II measured data.

SSC-L

The SSC-L code[13] was developed by Brookhaven National Laboratory (BNL) for analyzing LMR operational transients. The code has the flexibility to analyze any LMR loop-type plant, and a pool version, SSL-P, has been under development. SSC-L uses a detailed multichannel core treatment coupled with a thermal-hydraulic treatment for the primary and intermediate heat transport loops and the SGs.

MOMENTUM INTEGRAL NETWORK

The Momentum Integral Network (MINET) code[14] has been developed by BNL for transient analysis of flow and heat transfer networks. It is mainly used for analyzing the BOP in a power-generating facility. MINET can be coupled to SSC to extend the SSC analysis to the BOP beyond the SG. MINET has been validated with data from the EBR-II intermediate loop and SG system.

IANUS, ARIES, AND TAP

Three other one-dimensional system codes that have been used extensively are the IANUS code,[15] a plant-specific code used by the FFTF project; the ARIES-P code[16] used by General Electric Company; and TAP (Ref. 17), utilized by Rockwell International.

COMMIX-1A

The COMMIX-1A code,[18] developed at ANL, is a general-purpose, porous-medium, three-dimensional, transient thermal-hydraulic code. Its original objective was to analyze component mixing problems, such as pipe or plenum flow, with some applications to rodded bundle analysis. More recently, the code has been expanded to include models that allow it to calculate the entire within-vessel flow and temperature fields of a pool-type LMR. While such calculations are expensive, they are the only way to address the inherently three-dimensional effects such as are found in pool LMR calculations. Therefore, the COMMIX analysis complements that of the cheaper one-dimensional codes, e.g., SASSYS. The COMMIX-1A version has been the major calculational tool to date and is single phase only: There also exists a two-phase version (COMMIX-2). The 1A version has recently been enhanced to include radiative heat transfer. This code is designed to accommodate forced and natural circulation calculations and uses an effective viscosity turbulence model that provides a simplified but reasonably adequate approximation of turbulent transport phenomena. The standard governing equations of conservation of mass, energy, and momentum are solved as an initial value problem in time and as a boundary value problem in space.

The solution of the system of equations is based on the replacement of an explicit flow path representation by continuum or quasi-continuum geometries. The continuum methodology is designed for application with components such as piping or inlet plenums; the quasi-continuum description is required for such components as rod bundles or heat exchangers and utilizes a "porous-body" model. In the quasi-continuum formulation, the presence of solid objects within the flow field has two effects. The first is to reduce the flow area, thus affecting flow velocities and related quantities. The second effect is that the solid objects alter the energy and momentum transfer. The former effect is modeled by inclusion of the proper volume porosity and surface permeabilities. The latter is taken into account by allowing the region to contain a distributed heat source/sink for heat transfer and/or a distributed resistance for momentum transfer. The distributed resistance is an effective means by which to achieve the desired pressure drop across the region of interest, e.g., the core assemblies.

The quasi-continuum calculation, therefore, is predicated on the assumption that a real system containing solid objects can be replaced by a porous-body calculational model having uniformly distributed solid objects such that the system and the model both have (a) the same volumetric porosities, (b) the same surface permeabilities, and (c) the same interactions between fluid and solid surfaces; i.e., the same rates of heat and momentum transfer. The calculational model and the physical system are therefore rendered thermal-hydraulically equivalent. The application of the quasi-continuum methodology to the vessel internals of a pool-type liquid-metal fast breeder reactor thus transforms an almost insurmountable calculational problem into a manageable state. The vast amount of detailed modeling, which normally would be required to sufficiently describe the flow paths, is replaced with a less cumbersome yet adequate model.

This code has been used to model the steady-state and transient thermal-hydraulic response of EBR-II, FFTF, and prototype fast reactor, including natural circulation decay heat removal, with good agreement with measured data (see, for example, Refs. 19 through 23). In addition, numerous validation studies have been conducted involving special purpose laboratory experiments modeling IHXs, piping systems, fluid plena, and related geometries with generally good results.

TEMPEST

A code that addresses similar problems as COMMIX-1A is TEMPEST (Ref. 24), which also has three-dimensional transient capability. Developed at Pacific Northwest Laboratory, TEMPEST is a transient, three-dimensional, hydrothermal computer program that solves the full three-dimensional, time-dependent equations of motion, continuity, and heat transport for either laminar or turbulent fluid flow, including heat diffusion and generation in both solid and liquid materials. The equations governing mass, momentum, and energy conservation for incompressible flows with small density variations (i.e., Boussinesq approximation) are solved using finite difference techniques. Turbulence is treated using a two-equation k model. The finite difference approach for the fluid-flow solution is based on a semi-implicit procedure whereby the momentum equations are solved explicitly, and the continuity/pressure solution is obtained implicitly.

COBRA

The family of COBRA codes has been developed to rigorously analyze thermal-hydraulic conditions within a fuel rod bundle. The space between adjacent fuel rods (three rods in triangular and four rods in square pitch lattices) is identified as a coolant subchannel; adjacent subchannels exchange mass and energy along their axial length. The solution of the energy equation in the fuel rod, which includes a spatially dependent heat source and conduction, is coupled to the solution for the fluid flow and temperature fields in the surrounding coolant subchannels. The effects of a spiraling wire wrap fuel rod spacer may be explicitly modeled. The COBRA-WC code[25,26] version was developed from COBRA-IV-I (Ref. 27); it allows a coupled analysis of multiple assemblies to be performed for steady-state and transient situations that cover forced and natural circulation conditions in single-phase flow.

SUPERENERGY-2

The ENERGY series of codes,[28] culminating in SUPERENERGY-2 (Ref. 29), also provides steady-state subchannel flow and temperature estimates within the assemblies. In contrast with a more rigorous COBRA subchannel-type solution, the ENERGY formulation is very cheap to run and thus provides an ideal design code: For this reason it has become the most widely used within-assembly design tool over the past several years. SUPERENERGY-2 provides this excellent design capability due to the fact that the intraassembly thermal-hydraulic behavior is approximated expediently via two approximations. First, the rod bundle is divided into two distinct flow regimes—central and wall regions. Second, each region treats its rod bundle as a porous-body continuum. An enhanced eddy diffusivity accounts for the thermal mixing between subchannels due to the wire wrap. A second parameter, the swirl flow parameter in the bundle periphery, is the ratio of the transverse swirl velocity to the axial velocity in the region. Both of these parameters are empirically determined as functions of the bundle geometry and Reynolds number. This model allows whole-core calculations on the subchannel level to be accomplished in minutes rather than hours of CPU time. These codes were developed at Massachusetts Institute of Technology by Todreas et al. and have been verified versus FFTF transients and laboratory experiments.

FORE-2M/COBRA-WC/DEMO

Westinghouse design methodology employs a sequence of three codes. The one-dimensional transient plantwide analysis is done via DEMO (Ref. 10). Whole-core analysis is accomplished with COBRA-WC (Ref. 25), which models the subassemblies of interest at a level that achieves the detail desired at the time. For example, the assembly calculation can model each pin explicitly or, more typically, utilize a lumped subregion model. However, it should be noted that calculations are extremely expensive in the more detailed mode. Given the COBRA-WC assembly flow rates and inter/intraassembly heat transfer effects, the FORE-2M code, which is a variant of FORE-II (Ref. 30), then provides the detailed pin and subchannel temperature calculation, as well as the hot channel pin analysis.

OVERVIEW

As noted above, some of the codes are plant specific, whereas others are applicable to a wide range of plants. In general, the plant-specific codes were developed first in response to the needs of specific projects. The more general codes came later to meet the needs of innovative new designs and to allow the validation of a code with tests on existing reactors and the application of the same code to new designs. The plant-specific codes tend to be complementary: Each is applicable to a different plant. On the other hand, some of the more general codes are somewhat redundant, partly because both the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) developed codes. The DOE developed SASSYS-1 and TEMPEST, whereas the NRC developed SSC-L and COMMIX. The NRC wanted to develop independent assessment capabilities rather than depending exclusively on codes supplied by the projects.

A number of foreign codes are similar to those used in the United States. The SABRE (Ref. 31), VELASCO (Ref. 32), and BACCHUS (Ref. 33) codes are similar to COBRA, and all are used for rod bundle analysis. FLOW-3D (Ref. 34), PHOENICS (Ref. 35), ASTEC (Ref. 36), ESTET (Ref. 37), and TRIO (Ref. 38) are three-dimensional codes similar to COMMIX and TEMPEST. The Japanese use PTAR (Ref. 39) and THAUPR-III (Ref. 40) for two-dimensional analysis of outlet plenums and reactor components.

CONCLUSIONS

Various organizations in the United States have developed a wide range of computer codes. Within-assembly-wise coolant and pin temperature calculations are done by COBRA, FORE-II, SUPERENERGY-2, and to some extent by SASSYS-1 and COMMIX-1A. Plantwide transient analysis is done by SASSYS-1, NATDEMO, DEMO, IANUS, SSC-L, DSNP, ARIES-P, TAP, and to some extent by COMMIX-1A. Multidimensional analysis of individual components is done by COMMIX-1A and TEMPEST. Some codes are plant specific: NATDEMO for EBR-II, IANUS for FFTF, DEMO for CRBR, TAP for SAFR, and ARIES-P for PRISM. Other codes have been applied more generally: SASSYS-1, COMMIX-1A, TEMPEST, COBRA, SUPERENERGY-2 and DSNP for any LMR, and SSC-L for any loop-type plant.

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