THERMOHYDRAULIC MODELING AND SIMULATION OF BREEDER REACTORS

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1. INTRODUCTION

An important phase in the design, operation and safety analysis of liquid-metal-cooled fast breeder reactors (LMFBRs) involves the prediction of plant response to various transients. Loss of power to a pump, turbine trip, or the uncontrolled withdrawal of a reactor control rod bank are examples of anticipated accidents. A complete loss of forced cooling or a major rupture in the primary piping system with reactor scram are termed accidental transients.

This paper deals with the modeling and simulation of system-wide transients in LMFBRs. "Unprotected" events (i.e., the presumption of failure of the plant protection system) leading to core-melt are not considered in this paper. The existing computational capabilities in the area of protected transients in the U.S. are noted. Various physical and numerical approximations that are made in these codes are discussed. Finally, the future direction in the area of model verification and improvements is discussed.

2. SIMULATION AND PHYSICAL MODELING2.1 Simulation Approaches

Dynamic simulation of complex physical systems, such as, nuclear power plants, submarines, and aircrafts requires identification of important physical processes and their mathematical modeling through conservation and constitutive principles. In general, the solution of these conservation and constitutive equations yields the desired state of the system under study. Regardless of the modeling details, the system may be simulated on an analog, digital, or hybrid computer [1].

There are a number of well-defined advantages and disadvantages of both analog and digital computers [2]. Perhaps the key distinction between analog and digital computers lies in the accuracy offered by them. In analog machines, the accuracy is limited by the quality of computer components and is rarely better than 0.1% of full-scale. The accuracy of data in digital computers is controlled by the number of bits in memory registers and numerical technique employed.

Hybrid computer techniques, as the name implies, represent an effort to combine in one system some of the characteristics associated with analog systems with those of digital systems. Thus, in principle, a hybrid

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computer can be utilized as purely analog or purely digital. In actual usage, however, hybridization involves the actual interconnecting of analog and digital portions within the system.

Analog computers are particularly useful in solving initial value problems in one independent variable. Partial differential equations involving two or more independent variables can be solved by permitting no more than one variable to vary in continuous form; the remaining variables will then have to be discretized by the application of finite difference approximations.

Digital computers are capable only of solving algebraic expressions. Therefore, all independent variables must be discretized. There are a number of numerical techniques that have been developed for, and particularly well suited to digital computers. It should, however, be added that the digital techniques are by far the most popular ones in use.

2.2 LMFBR System Simulation Capabilities

The primary aims of detailed system simulation codes are applications to design, training and safety analysis. Table I summarizes significant physical processes and the desired numerical efficiency for a number of representative transients in LMFBRs. It is, therefore, evident that a simulator or a system code is expected to yield a considerable number of thermohydraulic parameters with engineering accuracy and reasonable computational efficiency. The achieved accuracy is, in part, dependent on the realism of physical and mathematical models and their respective solutions.

Table II lists the LMFBR system simulation capabilities and limitations in the United States [3]. HALAP [4] was obtained by adopting the RELAP-3B [5] (BNL version of RELAP-3) computer code through substitution of sodium properties instead of those of water. Although this code is capable of providing a rudimentary flow decay for a loss of piping integrity accident in the primary system, many modeling as well as operating limitations exist.

The IANUS [6] code was developed for the Fast Flux Test Facility (FFTF) and is capable of modeling thermohydraulic transients in the primary, intermediate and the dump heat exchanger (tertiary) systems. The

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MO [7] code is similar to IANUS, and is designed for specific application to the Clinch River Breeder Reactor (CRBR). It models the primary and intermediate transport systems as well as the steam generating system. IANUS and DEMO are specifically designed for particular plants and lack many physical models as summarized in Table II. A version of DEMO was developed at BNL [8] for use in FFTF system transient analysis. More recently a version of DEMO has been adapted to a Pool-type LMFBR [9]. The dynamic simulation code NATCON [10], used for evaluation of external hydraulic behavior of the EBR-II primary system, has been linked to DEMO-IV computer code resulting in NATDEMO code [11]. Application of NATDEMO to EBR-II loss of power transients show excellent agreements with key plant variables.

The BRENDA [12] code developed at the University of Arizona consists of a number of digital simulators prepared and maintained as input files for the DARE-P software system (Digital Analyzer Replacement-Portable). The code is intended to provide convenient and economical simulations of transients for use in scoping studies.

The EPRI-CURL [13] code developed at Cornell University is similar in scope to DEMO code, except for better physical and mathematical model representation, enhancing physical and numerical accuracy.

The SSC [14] code developed at Brookhaven National Laboratory is a highly general, state-of-the-art code for LMFBR system simulation studies. It provides rather detailed models for processes of interest in the reactor core as well as the heat transport and the steam generating system. Depending upon the transient to be analyzed, the user can choose, through appropriate input data cards, to emphasize either the reactor or the balance of plant or both. This code providing advanced system transient capability for the U.S. Nuclear Regulatory Commission's LMFBR licensing studies. This code has been in operation for more than four years and it is being used by a large number of organizations both at home and abroad. It is also being validated using the FFTF plant data as well as the existing test data from various other component tests.

2. Physical Models

In view of the limited space, it is not possible to detail all of the models that have been used in different codes. Only key features are listed below. For a more detailed presentation, the reader is referred to a review article by Agrawal and Habib-Rahbar [1].

The reactor heat generation rate is invariably presented by the point-kinetics. The entire reactor core, in most detailed system codes, represents at least one or more fuel, blanket or control assemblies. The thermal-hydraulics throughout the plant is represented by a suitably averaged one-dimensional (in axial direction) conservation equations. The multidimensional effects may be incorporated in an approximate way. For example, parallel reactor channels in the reactor core may be coupled via common pressure head points in the lower and upper mixing plenums. These regions themselves may be modeled either as a single or multiple lumped control volumes as done in the SSC code [14].

The pumps that are used in the primary and intermediate heat transport systems are free-surface,

variable speed, centrifugal units. The impeller behavior is best characterized by homologous head and torque relations. The transient speed and discharge rate are evaluated from impeller and coolant dynamic equations. The thermal performance of the intermediate heat exchanger, for most system-wide simulations, can be represented in terms of one equivalent heat transfer tube. Similarly, the steam generating system can be modeled in terms of a single heat transfer tube in each of the evaporator and superheater, if they are separate units, or by a single tube for once-through type steam generator designs. Due care must, of course, be given to the different flow regimes on the water side.

2.4 Numerical Techniques

There are two separate problems: one is to obtain the steady-state solution for the entire system which, then, is used as initial condition in integrating the time-dependent equations for all systems and subsystems. The steady-state solutions can best be obtained by solving time-independent conservation equations. This method is superior to the one in which the plant is initialized by numerically forcing time-derivative terms of the time-dependent equations to approach zero. The second part of the problem is considerably more involved and interesting since the entire system is comprised of processes which have characteristic time constants varying over a few orders of magnitude [15]. This results in a rather "stiff" set of equations. Grouping of variables that contribute to these small time constants and integrating them with small step size while using large step sizes for the rest of the variables has been one of the approaches to limit the computer time. This partitioning of variables has been done in the SSC code on the basis of physical arguments and has been found to be successful. The same underlying principle is used in the EPRI-CURL code. The experience to date shows savings in computing time by a factor of up to four. The method of separating the fast and slow variables has been studied mathematically [16] for linear systems and applied to non-linear systems by Girijashanker et al., [17] and has been successfully tested for a pressurized water reactor system. The utility of this scheme could be considerably enhanced if it could be made automatic.

3. FUTURE DIRECTIONS IN LMFBR SIMULATION

The future effort in the area of LMFBR simulation may be divided into four categories: model improvements, accident management, numerical methods, and computer code verification. In the first category, modeling improvements are needed for the hot-spot core temperatures to account for (1) key uncertainties in the fuel-clad gap conductance, (2) inter- and intra-assembly heat redistribution at low flows, (3) the mixing and stratification of sodium in large bodies such as the plenum in the reactor vessel and the intermediate heat exchanger, and (4) stratification in the pipings. A better understanding of the sodium boiling and post-boiling effects at low flows and low heat flux will also be desirable.

The area of accident management is beginning to get the attention it deserves. Much of the simulation studies have been directed towards finding the consequences of design basis accidents. In view of recent TMI-II experience, simulation for beyond design basis accidents, with emphasis on engineered safety systems, should be considered.

In the area of computation methods, there is always a desire for more efficient integration schemes. There is a growing interest in real-time (or faster) simulation. Invariably this would imply making some compromise on the modeling level.

The area of computer code verification really has two parts: one to ascertain that the code is doing what it was intended to do, and the second to ascertain whether the code is simulating realistically. The first part is relatively easier to verify via inter-code comparisons and "isolated" analytical solutions. The second part is more involved—it can be done through code applications to the whole plant data as well as phenomenological tests. This phase of code verification is a continuing process and it must be carried out in order to provide confidence in the analytical tools.

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Table I Required Capabilities for System Simulation

Transient Class	Example of Events	Significant Physical Processes & System Complexity	Degree of Details and Modelization	Importance of Non-Efficient
Operation	<ul style="list-style-type: none"> Start-up Load Changes Shutdown 	<ul style="list-style-type: none"> Reactivity Feedback Effects Plant Protection and Control Systems (PPS-PCS) Decay Heating Effects 	Detailed in-vessel Coarse ex-vessel Single loop simulation	Medium
Incidental	<ul style="list-style-type: none"> Reactivity Transients Single Pump Failure Turbine Trip Valve Failures Pump Motor Operation 	<ul style="list-style-type: none"> Reactivity Feedback Effects PPS-PCS Behavior Pump Behavior Asymmetric Multiloop Effects Flow Reversal and Stagnation Component Interactions 	Detailed in-vessel Multiloop simulation	High
Accidental	<ul style="list-style-type: none"> Station Black-out Pipe Breaks Loss-of-Heat Sink Seismic Events 	<ul style="list-style-type: none"> Inter/intra assembly Heat and Flow Redistribution Upper (Pool) Plenum Stratification Initial Plenum (Cold Pool) Mixing Partial Core Flow Reversal & Stagnation Sodium Boiling and Cladding Failure Piping Heat Transfer/Stratification Subassembly Pressure-Drop Flow Oscillations Core/Steam Interaction Auxiliary Cooling Paths Pump Dynamics and Stopped Motor Resistance Axial Convection Effects at Low Flow Thermal Instabilities Steam Discharge Flow Dynamics Decay Heating Effects Steam Generator Dynamics HX and Steam Generator Flow Maldistribution Heat Losses to the Environment 	Very detailed in-vessel Very detailed ex-vessel Multiloop simulation	Very High

Table II Summary of System Simulation Capability in the U.S.

Code	Physical Modeling Capability	Mathematical and Computational Aspects
Name Organization Year Support	Issues Unique Features Limitation	Solution Techniques Unique Features Computer Operable
KALAP SRI 1975 USNRC	Thermal-Hydraulic Overall Plant Simulation General Code Adapted from KALAP	Explicit — CDC-7600
LANS University 1976 Westinghouse	Thermal-Hydraulic Overall Plant Simulation for FFT Plant Protection and Control Systems Primary Pipe Break Model Natural Convection Duct Heat Exchanger Model	Explicit (Eulerian integration for energy & predictor corrector for flow) — CDC-6400 CDC-7600
SEMO M-40 1976	Thermal-Hydraulic Overall Plant Simulation for FFT Same as above, but models Steam Generator	Explicit Steady state capability CDC-7600
SKENCA U. of Arizona 1976 USNRC	Thermal-Hydraulic Overall Plant Simulation Same as above LANS-Gen- erator Model Steam Generator Auxiliary Systems	Explicit or Explicit Uses the SAS language CDC-6400 CDC-7600
SPRI-ORNL General 1977 SPRI	Thermal-Hydraulic Overall Plant Simulation Plant Protection & Control SAS Channel Reactor Flow Redistribution & Reversal Multiloop Simulation Pipe Break Model Natural Convection Loop & Pool Junctions	Explicit Steady state capability Multiple time scales Methods for process time constant evaluation Fast running IBM-370 CDC-7600
SIC SRI 1977 USNRC	Thermal-Hydraulic Overall Plant Simulation Plant Protection & Control Systems General Multiloop Simulation Multichannel Reactor Flow Redistribution & Reversal Pipe Break anywhere in Sodium Loop Natural Convection Modelization limited only to the machine size Low heat flux sodium boiling Model Detailed Steam Generator Model Balance of Plant Model Duct Heat Exchanger Model Direct Reactor Auxiliary Cooling Systems Loop & Pool Junctions	Semi-implicit Steady state capability Variable time scales Very fast running All