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# LMFBR SYSTEM-WIDE TRANSIENT ANALYSIS: THE STATE OF THE ART AND U.S. VALIDATION NEEDS*

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## ABSTRACT

This paper summarizes the computational capabilities in the area of liquid metal fast breeder reactor (LMFBR) system-wide transient analysis in the United States, identifies various numerical and physical approximations, the degree of empiricism, range of applicability, model verification and experimental needs for a wide class of protected transients, in particular, natural circulation shutdown heat removal for both loop- and pool-type plants.

## INTRODUCTION

The design, operation and safety analysis of liquid-metal-cooled fast breeder reactors (LMFBRs) require 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 incidents whereas complete loss-of-forced cooling or a major rupture in the primary piping system with reactor scram are termed accidental transients.

Investigators have used mathematical models to predict transient behavior of LMFBRs, often using numerous simplifications to allow tractable analytical solutions. With the advent of modern high-speed digital computers, simulation of physically more realistic, and complex interacting phenomena covering a wider range of interest have become possible [1].

This paper reviews the computational capabilities in the area of LMFBR system-wide transient analysis in the U.S., and identifies various numerical and physical approximations, the degree of empiricism, range of applicability, model verification and experimental needs for a wide class of protected transients, in particular, natural circulation shutdown heat removal.

## LMFBR SYSTEM-WIDE TRANSIENT ANALYSIS



Large detailed systems simulation codes are needed for design and safety analysis. Specifically, the computer codes are required to predict the complete state of the power plant during normal operating and abnormal accident conditions. These system-wide transient codes are expected to predict three generic classes of thermal-hydraulic transients; namely, (1) operational, (2)

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incidental, and (3) accidental events. Table I identifies the required capabilities of LMFBR system simulation codes for various classes of transient events.

The required degree of physical detail and system complexity is highly dependent upon the transient to be analyzed. Specifically, for high flow - high power transients, where forced convection processes dominate, the in-vessel modeling must include sufficient detail in order to adequately predict the neutronic feedback effects; however, a coarser nodalization and representation of the heat transport systems and the steam generator system is often sufficient. The larger the degree of perturbation, the greater the need for physical and spatial detail in the simulation. Furthermore, in transients where free convection effects become significant, considerable care must be given to the adequate representation of the actual physical and spatial processes. For example, in many transients, the modeling of the entire reactor core by a single, average channel is highly questionable. Flow redistribution enhanced by thermal buoyancy in the reactor subassemblies is an important effect that must be taken into account in the modeling [2]. The sodium temperature and flow fields in the loop reactor upper plenum or the pot reactor hot pool region must be adequately modeled [1].

The effect of interassembly heat transfer can be a significant influence on the system behavior. The magnitude of this effect depends on several factors including the extent of radial power variation both within an individual assembly as well as across the reactor core, the power-to-flow ratios for each assembly, and the geometric characterization. Singer et al. [3] have attributed EBR-II radial temperature flattening effects to intersubassembly heat transfer with additional effects due to dynamic flow redistribution at low flow natural convection conditions. However, it seems that the interassembly heat transfer is accentuated in EBR-II due to the more pronounced radial power variation and the specific design of the EBR-II inlet. In large power reactors, however, the radial interassembly heat transfer is expected to be small because the radial temperature variations across the reactor core are minimized by design. In any case, the intersubassembly heat transfer tends to reduce the radial temperature gradient across the core thereby providing an extra safety margin. Ignoring interassembly heat transfer provides a degree of conservatism in the safety analysis [4].

A partial and/or complete flow reversal is possible during certain transient events, such as a major pipe rupture in the cold leg or a complete loss-of-heat-sink accident [1].

A summary of significant physical processes and model complexity is also given in Table I.

Another important consideration in LMFBR system-wide transient analysis is the numerical integration algorithms used in the solution of the steady state and time-dependent equations for unique and stable solution throughout the domain of interest. A simulator or a system code must therefore provide a considerable number of thermohydraulic parameters with engineering accuracy and reasonable computational efficiency. The attainable accuracy is, in part, dependent on the adequacy of the physical and mathematical models and their respective solutions.

Table II summarizes the status of the LMFBR system simulation capabilities and limitations in the U.S. NALAP [5] was obtained by adopting the RELAP-3B [6] (BNL version of RELAP-3) computer code by replacing the water properties

Table I Required Capabilities for System Simulation

| Transient Class | Example of Events | Significant Physical Process & System Complexity | Degree of Details and Nodalization | Importance of Numerical Efficiency |
|---|---|---|---|---|
| Operational | • Start-up<br>• Load Changes<br>• Shutdown | • Reactivity Feedback Effects<br>• Plant Protection and Control System (PPS-PCS) Behavior<br>• Decay Heating Effects | Detailed In-Vessel<br>Coarse Ex-vessel<br>Single loop Simulation | Medium |
| Incidental | • Reactivity Transients<br>• Single Pump Failure<br>• Turbine Trip<br>• Valve Failures<br>• Pony Motor Operation | • Reactivity Feedback Effects<br>• PPS-PCS Behavior<br>• Pump Behavior<br>• Assymmetric Multiloop Effects<br>• Flow Reversal and Stagnation<br>• Component Interactions | Detailed In-Vessel<br>Multiloop Simulation | High |
| Accidental | • Station Black-out<br>• Pipe Breaks<br>• Loss-of-Heat Sink<br>• Seismic Events | • Inter/Intra Assembly Heat and Flow Redistribution<br>• Upper Pool Plenum Stratification<br>• Inlet Plenum (Cold Pool) Mixing<br>• Partial Core Flow Reversal & Stagnation<br>• Sodium Boiling and Cladding Failure<br>• Piping Heat Transfer/Stratification<br>• Subassembly Pressure-drop<br>• Flow Orificing<br>• Core-bypass Interaction<br>• Auxiliary Cooling Paths<br>• Pump Dynamics and Stopped Rotor Resistance<br>• Axial Conduction Effects at Low Flow<br>• Thermal Instabilities<br>• Break Discharge Flow Dynamics<br>• Decay Heating Effects<br>• Steam Generator Dynamics<br>• IHX and Steam Generator Flow Maldistribution Effects<br>• Heat Losses to the Environment | Very Detailed In-Vessel<br>Very Detailed Ex-Vessel<br>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 | Scope | Unique Features | Limitation | Solution Technique | Unique Features | Computer Operable |
| NALAP BNL 1975 USNRC | Thermal-Hydraulic Overall Plant Simulation | General Code Adapted from RELAP | Lacks Steady State Capability<br>Limited to 40 Volumes and 50 Junctions<br>Lacks Sodium Boiling<br>Lacks Control Systems | Explicit | -- | CDC-7600 |
| IANUS W-Hanford 1976 Westinghouse | Thermal-Hydraulic Overall Plant Simulation for FFTF | Plant Protection and Control Systems<br>Primary Pipe Break Model<br>Natural Convection<br>Dump Heat Exchanger Model | Limited Nodalization<br>Single Core Channel<br>Lacks Sodium Boiling<br>Maximum of 2 Coolant Loops<br>Adiabatic Subassemblies | Explicit (Eulerian integration for energy & predictor corrector for flow) | -- | CDC-6600<br>CDC-7600 |
| DEMO W-ARD 1976 USDOE | Thermal-Hydraulic Overall Plant Simulation for CRBRP | Same as above, but Models Steam Generators | Limited Nodalization<br>Limited Core Channels<br>Lacks Flow Redistribution Model<br>Lacks Sodium Boiling<br>Lacks Models for Auxiliary Cooling Systems<br>Adiabatic Subassemblies | Explicit | Steady State Initialization | CDC-7600 |
| BRENDA U. of Arizona 1976 USNRC | Thermal-Hydraulic Overall Plant Simulation | Same as above<br>Turbo-generator Model<br>Steam Generator Auxiliary Systems | Limited Nodalization<br>Lacks Steady State Capability<br>Loop Version Only | Implicit | Uses the DARE Language | CDC-6600<br>CDC-7600 |
| EPRI-CURL Cornell Univ. 1977 EPRI | Thermal-Hydraulic Overall Plant Simulation | Plant Protection & Control Systems<br>Six Channel Reactor<br>Flow Redistribution & Reversal<br>Multiloop Simulation<br>Pipe Break Model<br>Natural Convection<br>Loop & Pool Versions | Lacks Sodium Boiling<br>Adiabatic Subassemblies<br>No Intra-assembly Effects<br>Recirculation-type Steam Generators<br>Lacks Auxiliary Systems<br>Limited Nodalization<br>Lacks Detailed Mixing Models for Plena (pools) | Explicit | Steady State Initialization<br>Multiple Time Step<br>Method for Process Time Constant Evaluation<br>Fast Running | IBM-370<br>CDC-7600 |
| SSC BNL 1977 USNRC | Thermal-Hydraulic Overall Plant Simulation | Plant Protection & Control Systems<br>General Multiloop Simulation<br>Multichannel Reactor<br>Flow Redistribution & Reversal<br>Pipe Break anywhere in Sodium Loops<br>Natural Convection<br>Nodalization Limited only by the Machine Size<br>Low Heat Flux Sodium Boiling Model<br>Detailed Steam Generator Model<br>Balance of Plant Models<br>Dump Heat Exchanger Models<br>Direct Reactor Auxiliary Cooling Systems<br>Loop & Pool Versions | Adiabatic Subassemblies<br>No Intra-assembly Effects<br>Lacks Detailed Mixing Models for Plena (pools) | Semi-implicit | Steady State Initialization<br>Variable Time Step<br>Very Fast Running | All |

with sodium properties. Although this code could provide a rudimentary analysis of flow decay for a loss of piping integrity accident in the primary system, it has many severe modeling as well as operating limitations.

The IANUS [7] 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 DEMO [8] code is similar to IANUS, and is designed for specific application to the Clinch River Breeder Reactor (CRBR). It models the primary and intermediate heat transport systems as well as the steam generating system. Since IANUS and DEMO are designed for specific plants, they lack many physical models as summarized in Table II. 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 thermal-hydraulic behavior of the EBR-II primary system, has been coupled to the DEMO-IV computer code resulting in the NATDEMO code [11]. Application of NATDEMO to EBR-II loss of power transients shows excellent agreement 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 [2] code developed at Cornell University is similar in scope to the DEMO code, with the exception of better physical and mathematical representations, which enhance physical and numerical accuracy.

The SSC [13] 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 (including sodium boiling) as well as in the heat transport and the steam generating system. Depending upon the transient to be analyzed, the user can choose, through appropriate input, to emphasize either the in-reactor or the balance of plant or both. This code provides an 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 and existing test data from various other component tests.

Detailed subassembly thermohydraulic modeling is not available in all of the above-mentioned computer codes. The maximum assembly temperatures (hot spot and/or hot channel) are obtained by applying appropriate hot spot/hot channel factors to the radial-wide average temperatures. This method can lead to very conservative estimates of hot channel coolant and hot spot cladding temperatures. Consequently, more detailed rod bundle thermal-hydraulic analysis must be used for best-estimate predictions.

## MODEL ASSESSMENT AND DATA BASE

The simulation of the entire plant to include the various processes as well as the interaction between different processes and/or components for the plant is obviously a complex problem. Experimental verification of such simulations is equally if not more complex. Clearly, the optimum data from a plant can be obtained by subjecting the plant under investigation to a variety of transients. This procedure is not feasible for two reasons: (1) the data base is often needed prior to building the plant, and (2) it may not be prudent to simulate all off-normal transients. Therefore, alternative procedures are employed [1].

A system simulation code, which incorporates simulation for individual processes and components, can be verified either on a modular or component basis or it can be checked against integral data for similar thermohydraulic conditions obtained from another plant. In just about all situations, areas are likely to be found where improvements would be needed. These improvements can be in the form of updating a physical model or developing an improved correlation. The resulting improvement is then incorporated into the system code and the entire process is repeated. The process of code verification is a continuing one but not an endless one. It is to be expected that as the data base is enlarged, there will be a continual decrement in improving models or correlations.

Experiments may be performed in existing plants as a part of the preoperational testing prior to full power operation. By and large, these integral or in-plant experiments will provide a very important data base since these data are prototypical. Alternately, tests may also be carried out for a single process or a component in a simulated environment. The laboratory tests are generally inexpensive and readily performed. Perhaps one of the most significant drawbacks of the laboratory tests is a potential for nonprototypical testing conditions. Nevertheless, it should be added that both in-plant as well as laboratory tests should be performed.

Table III indicates various parts of the LMFBR system that require verification. Included in this table are the key parameters and their importance. This list is not intended to be exhaustive, rather it shows some of the most crucial parameters. The importance rating is subjective in that it strongly depends upon the plant transient under investigation. The importance rating included in Table III is perhaps more representative of a natural circulation transient.

Table IV summarizes some of the current tests and experiments either completed, underway or in progress which are directly applicable to model/code validation.

As part of the SSC Code Validation Program, comparisons of integral FFTF plant data have been made with code calculations. Four natural circulation tests have been conducted between November 1980 and March 1981, at pre-scram power levels of 5, 35, 75 and 100%, respectively [14]. After a prescribed length of time at power, the tests were initiated by scramming the control rods into the reactor. Immediately thereafter, main coolant pumps were turned off, simulating loss of all normal and emergency power. Extensive temperature and flow measurements were taken.

Figure 1 shows the measured Fuel Open Test Assembly (FOTA) temperature following scram to natural circulation from 100% power.

Also shown in Fig. 1 are the predicted FOTA temperatures calculated by the IANUS/CORA [14] and SSC-L computer codes. The agreement is shown to be quite good; however, it must be noted that the IANUS/CORA model allows individual intersubassembly redistribution of heat and flow; the SSC-L code only models the flow redistribution. The peak temperature predicted by SSC-L is therefore higher than both the data and the IANUS/CORA model.

Steady power-natural circulation tests have been performed at the U.S. EBR-II, the French PHENIX, and the British PFR reactors [15]. A similar analysis

Table III. LMFBR System Validation Needs Relevant to Natural Circulation

| Component | Parameter/Phenomena | Importance | Data Availability |
|---|---|---|---|
| I. Reactor Vessel | | | |
| 1. Inlet Module | Pressure Drop | High | Limited |
| | Mixing During Partial Flow Reversal | Medium | None |
| 2. Plena & Pools (Hot & Cold) | Mixing and Stratification | High | Limited |
| 3. Assemblies | Pressure Drop Laminar/Critical Transition/ Turbulent | High | Limited/Limited/ Sufficient |
| | Heat Transfer Coefficients | High | Sufficient |
| | Intra-Assembly Redistribution Heat/Flow | High | Limited |
| | Inter-Assembly Redistribution Heat/Flow | High | Limited |
| | Low-Heat Flux Boiling Dryout Correlations | Medium | Limited |
| | Decay Heat | High | Limited |
| 4. Structural Material | Shutdown Heat/Heat Losses | Medium | None |
| II. Heat Transport System | | | |
| 1. Piping | Pressure Drops | Low | Sufficient |
| | Stratification | Medium | Limited |
| 2. Pumps | Frictional Torque | High | None |
| | Locked Rotor Resistance | High | Limited |
| 3. Check Valve | Pressure Drop | High | Limited |
| 4. IHX | Shell Side Pressure Drop | Medium | Limited |
| | Flow Maldistribution | Medium | Limited |
| III. Steam Generator System | | | |
| 1. SG | Pressure Drops | High | Sufficient |
| | Dryout Correlations | Medium | Sufficient |

Table IV. Natural Circulation Test Data

| Facility | Test | Potential Applicability |
|---|---|---|
| FFTF | Acceptance Tests | Model Validation on System-Wide Basis<br>Adequacy of Core Modeling |
| EBR-II | XX07, XX08 Series | Assessment of Inter-Assembly Flow Redistribution and Heat Transfer |
| ETEC | CRBR Components | |
| | SG Module | Model Validation |
| | Pump | Assessment of Frictional Torque and Locked-Rotor Resistance |
| | PACC | Model Validation |
| GE | Flow Redistribution Experiments | Assessment or Inter-Assembly Flow Redistribution Model<br>Investigation or Parallel Channel Instability Effect<br>Lower Plenum Mixing Effect |
| ANL | Flow Mixing and Stratification | Assessment of IHX and Pipe Models |
| ORNL | Boiling Experiments | Dryout Prediciton |

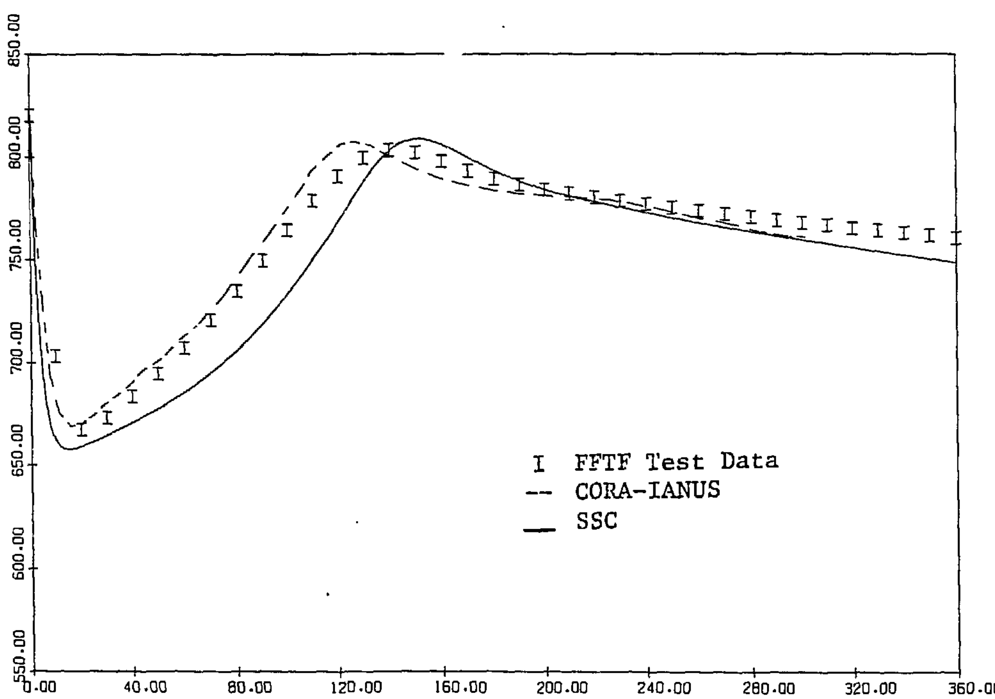


Fig. 1: FFTF Row 2 FOTA Average Coolant Exit Temperature at Top of Fuel Location for the 100% Power Test.

using the SSC-P computer code shows predictions to be within 5% of the measured data [15].

Special effects tests summarized in Table IV are being performed at Argonne National Laboratory (ANL), Oak Ridge National Laboratory (ORNL), General Electric (GE), and the Energy Technology Engineering Center (ETEC) which are very conducive to evaluation of physical models.

The ANL data [16] can be used to determine the adequacy of one-dimensional models during low flow natural convection, where the thermal buoyancy and stratification effects can influence the pressure drop and heat transfer in the plant components (Pipings, IHX, etc.).

The ORNL data [17] can be used to validate low and high heat flux sodium boiling models. These data can also be utilized to assess the intraassembly incoherency phenomenon associated with the bundle size.

Natural convection water simulation of interassembly flow redistribution, and upper plenum mixing and stratification phenomena are being conducted at GE [18]. The test model is not prototypic of the reactor. However, valuable data regarding the coolant flow dynamics and interchannel instability effects as a function of channel geometry and heat flux can be obtained.

Component performance testing is underway at the ETEC with both water and liquid metal coolant to verify the characteristics of LMFBR coolant pumps, valves and the steam generator units in support of the Clinch River Breeder Reactor Project (CRBRP).

In the area of experimental data, there is a vital need for more prototypic data on sodium mixing and stratification effects at low flow free convection conditions, especially in the plena and large pools. Experimental data from prototypic facilities for multiple bundle geometry to quantify and assess the effects of inter and intraassembly heat and flow redistribution are also needed.

Data are needed on the operation of pumps, especially at or near locked rotor and cavitation and its impact on the system response.

Finally, benchmark transient problems should be established to be used by various model developers and code users so that various models and assumptions as well as use of plant data can be compared.

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