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Computer Analyses for the Design, Operation and Safety of New Isotope Production Reactors:

A Technology Status Review

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DISCLAIMER

FOREWORD

This review on nuclear reactor analysis methods has been prepared for the U.S. Department of Energy, Office of New Production Reactors, by the Reactor Analysis Division in the Department of Nuclear Energy at Brookhaven National Laboratory, Upton, NY, 11973. The review has been prepared for the purpose of selecting advanced computer analysis methods to support the design, development, safety and maintenance of new isotope production reactors.

CONTENTS

ABSTRACT

A review is presented on the currently available technologies for nuclear reactor analyses by computer. The important distinction is made between traditional computer calculation and advanced computer simulation. Simulation needs are defined to support the design, operation, maintenance and safety of isotope production reactors.

Existing methods of computer analyses are categorized in accordance with the type of computer involved in their execution: micro, mini, mainframe and supercomputers. Both general and special-purpose computers are discussed. Major computer codes are described, with regard for their use in analyzing isotope production reactors.

It has been determined in this review that conventional systems codes (TRAC, RELAP5, RETRAN, etc.) cannot meet four essential conditions for viable reactor simulation: simulation fidelity, on-line interactive operation with convenient graphics, high simulation speed, and at low cost. These conditions can be met by special-purpose computers (such as the AD100 of ADI), which are specifically designed for high-speed simulation of complex systems. They have an interactive simulation software system and a powerful simulation language.

The greatest shortcoming of existing systems codes (TRAC, RELAP5) is their mismatch between very high computational efforts and low simulation fidelity. They are based on complicated flow models that cannot be closed for lack of reliable correlations. The drift flux formulation (HIPA) is the viable alternative to the complicated two-fluid model (TRAC, RELAP5).

No existing computer code has the capability of accommodating all important processes in the core geometry of isotope production reactors. Experiments are needed (heat transfer measurements) to provide necessary correlations.

It is important for the nuclear community, both in government, industry and universities, to begin to take advantage of modern simulation technologies and equipment. Non-nuclear industries, such as aviation, spacecraft, automobile and pharmaceutical industries, as well as the armed forces, have utilized and promoted the technology of special-purpose, peripheral parallel processors for convenient and efficient high-speed simulations for many years and with outstanding success. Advanced simulation is particularly important for the nuclear industry, because the nuclear industry has to cope with extremely complex processes in very large and complicated systems, and it may have to respond quickly to unexpected emergencies. During the past decade, it has been shown that this is not possible with conventional standard methods and general-purpose computers. The most advanced technology should be used to render a powerful tool for a difficult simulation task.

1. INTRODUCTION

1.1 Background

The design, operation and maintenance of nuclear reactors requires the capability of performing computer analyses for the purpose of predicting reactor systems behavior under normal and accident conditions. This applies to nuclear reactors for power generation, for isotope production and for scientific experiments. The systems analyses are needed for:

(i) optimizing the design through parametric studies,
(ii) selecting and assessing technical specifications,
(iii) optimizing the control systems and selecting control set points,
(iv) evaluating operational error margins through parametric studies,
(v) safety analyses,
(vi) assessing and optimizing operating and emergency procedures,
(vii) training operators and technical advisory personnel, and
(viii) resolving technical issues concerning the plant response, as such issues arise during operation.

The systems analyses are plant-specific. A large number of analyses need to be performed for a particular reactor.

Many technical questions arising during the design and operation of a nuclear reactor either remain unanswered or will be resolved largely on the basis of guess work, unless a capability is available at low cost, which produces quickly reliable results. The technical staff associated with the design and operation of a nuclear reactor gains demonstrably the more in-depth experience about the plant behavior, the more convenient and the less expensive the analytical tools are at their disposal.

This report summarizes the status of available analytical methods for predicting reliable nuclear reactor system transients, namely existing computer codes, computers and computer simulation technologies.

1.2 Contrast Between Computer Calculation and Computer Simulation

Traditional computer analyses of nuclear reactor transients, proceeding both under normal and postulated accident conditions, have been performed by computer calculations. Computer calculations are carried out on general-purpose computers (super or minicomputers and micro processors), and by the use of standard FORTRAN computer codes, such as RETRAN [McFadden et al., 1981; Paulsen and Hughes, 1983; Hughes, Katsma and Paulsen, 1985], RELAP5 [Ransom et al., 1985; Feinauer et al., 1984; Trapp and Riemke, 1985; Ransom, 1983; Dimenna et al., 1988] or TRAC-PF1/MOD1 [1986; Liles et al., 1988], etc. The codes are executed in batch mode, and the results are post-processed with a graphics code to provide convenient and automated displays in graphical form.

Computer simulation contrasts the above-described computer calculation. It is carried out on a special-purpose simulation computer which provides through its systems software a simulation environment and utilizes a systems simulation

language for implementing specific physical models. Computer simulation is carried out on-line, interactively. It permits the analyst to influence the simulation as needed, either from a keyboard terminal or through suitable hardware components. Interactive operations are needed to analyze the plant response to operator actions, to introduce component or systems failures without pre-programming failure events, and in general for achieving efficiency in the conduct of systems analyses by observing and responding to the evolution of events, rather than by waiting for the completion of the calculation.

Special-purpose simulation computers have been specifically designed for a single purpose which is the efficient dynamic computer simulation of complex systems. They contain simulation task-specific processors which have their own built-in firmware (integration algorithms for integrating processors, as an example) and operate in parallel with each other in an architecture specifically designed for simulation without the need to vectorize or to separate the computer program (modeling software) into independent streams of instructions for parallel processing. This architecture renders the simulation computer superior to array processors which are not suitable for reactor systems simulation, because reactor simulation codes are predominantly scalar, i.e. they contain no large arrays. All reactor systems and thus their modeling equations are tightly coupled and cannot be easily separated and assigned to separate parallel processors.

The simulation environment of simulation computers is provided through a major operating systems software package and permits a large number of interactive operations from the keyboard. The development and implementation of such an operating system on a general-purpose computer, to achieve interactive operating capabilities comparable to those offered as part of a simulation system, would require resources equal to or greater than the needs for modeling and model implementations. Interactive operations afford on-line activities for pre-run preparations, for run-time control and monitoring, such as setting, changing or displaying parameters, initial conditions, setting and changing integration algorithms, specifying and displaying run conditions, integration speed-up factors and simulation time length, restarting, stopping, continuing and logging on-line activities. Interactive operations also afford the capabilities of plotting variables at the same rate as the problem solutions are being generated, of controlling post-run operations, of utilizing command files for setting up one or a sequence of simulations for the purpose of generating parameter envelopes for sensitivity studies. The simulation environment also provides a large set of on-line program diagnostics which are not available on general-purpose computers.

Simulation computers utilize a simulation language for implementing the mathematical model of the system to be simulated. The simulation language is state variable oriented and optimized for integrating large, complex system of nonlinear ordinary differential equations. While the analyst using the simulation system has no need to get involved with the programming language, it is extremely advantageous to use for program development, improvement and maintenance a simulation-oriented language which frees the programmer from concerns of numerical integration. Using a simulation language in the simulation environment (operating system) of a simulation computer, a program developer needs to know only what the characteristic features of the selected integration algorithms are, but he does not get involved with the implementation and testing

of such algorithms. Moreover, the simulation language, while providing all the needed FORTRAN capabilities, offers a large number of simulation-oriented constructs (kernel calls) and it provides the means needed to utilize the maximum computing speed to a much greater extent than is possible in general-purpose supercomputers.

In summary, computer simulation is superior to computer calculation, both in convenience and economy of code development and of applications in support of reactor design, operation and maintenance. According to G. A. Korn [Korn, 1989], "the convenience and speed of a direct-executing simulation system - which does not interrupt an analyst's train of thought with repeated compilation delays - must be experienced to be believed."

Simulation opens new avenues. It provides enhanced understanding of complex systems and offers new approaches which are unimaginable to those familiar with computer calculations, because simulation systems respond instantly and produce qualitative answers in short time and at low cost, often to questions that remain unanswered by traditional computer calculation, since their cost and time requirement are normally too high. The distinction between computer simulation and computer calculation is very important and will be made throughout this document.

1.3 Simulation Needs

The effective support of nuclear reactor design, operation and maintenance is possible only with a reactor plant-specific simulation facility which meets simultaneously these four criteria:

a. it must simulate realistically and accurately steady-state conditions, and normal and accidental transients, so as to meet or exceed ANSI/ANS-3.5 [ANS, 1985].

b. it must provide easy access through convenient on-line operations with on-line graphics display for efficient interpretation of predicted plant responses.

c. it must be capable of simulating much faster than with real-time process speeds.

d. it must be cost-effective; needed capital investment and operating expenses must be small relative to the costs incurred from training simulators, currently used mainframe computers or minicomputers.

These four characteristics encompass all that is required for predicting reactor systems responses reliably, on time and with reasonable manpower and financial expenditures. They imply the minimum of necessary preparations and no need for the involvement with computational issues, such as plant nodalization, numerical stability or error accumulations and convergence.

For isotope production reactors, there exists now no simulation facility which meets all four objectives listed above.[1] Several computer codes, executing on personal computers, meet objectives (b, c and d), but not (a) and are limited to a few applications, some of which resemble "back-of-the-envelope" type calculations. Failure to meet objective (b) leads to excessive requirements of manpower for training, for familiarization with, and upkeep of, the computer code. It also requires too much time for execution and interpretation. As pointed out above, a facility which does not meet objective (c) is inadequate for parametric studies and for optimization of procedures or design changes.

Most existing computer codes, such as TRAC, RELAP5, or RETRAN, used at this time, fail to meet objective (d). Consequently, they are used infrequently, only when extreme needs justify large expenditures and if sufficient funding is available. Time-consuming codes, requiring an expensive computer installation, are unsuitable for continuous, on-line access or for continuous stand-by preparedness. Continuous on-line access during all working hours is desirable for technical advisory personnel and, in fact, no more expensive than currently employed batch processing. Continuous stand-by preparedness would be invaluable for operator assistance in emergencies.

A newly developed and thoroughly proven technology for computer simulation of (BWR) reactor systems should be adapted for simulating isotope production reactors. This would meet the specific needs for designing, operating and maintaining new production reactors at low cost and greater efficiency than is possible with existing alternative analytical tools.

It has been stated above, that engineering analyses of nuclear reactor behavior are currently being carried out mainly through time-consuming and expensive batch processing of large, standard FORTRAN computer codes on mainframe or supercomputers. A large number of important and useful analyses are foregone because they are too expensive to carry out, or because easy access to a large enough computer is not available, or because it takes too much manpower effort to obtain the results.

It has been stated above, that four requirements must be met simultaneously for effective reactor simulation, namely, realistic and accurate simulation must be provided with easy and convenient, on-line interactive access, with high-speed simulation capability and at low cost.

In Chapter 2 below, we will show that no method or facility exists which meets the above four requirements, except the BNL Plant Analyzer which is now limited to BWR simulations.

Therefore, it is necessary to expand the newly developed BWR simulation capability and to apply it to isotope production reactors, taking into account the specific conditions and processes of that plant. This will provide the needed alternative to currently available, but inadequate, methods for reactor simulation. An isotope production reactor plant analyzer is needed which has

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[1]The BNL Engineering Plant Analyzer has these four characteristics but it is currently limited to BWR simulations [Wulff et al., 1984a].

all of the four characteristics stated above, primarily for the following reasons:

1.3.1 Modeling Fidelity

Detailed models, based on first principles and on available experimental data, are needed in conjunction with sufficiently fine nodalization for numerical integration to obtain engineering accuracy in the simulation of normal and abnormal plant transients. The short-cuts, coarse nodalizations and the large time steps for integration now used in all general-purpose computer simulations at real-time speeds must not be used. Instead, time and space intervals must be chosen to guarantee engineering accuracy. Consistent applications of the conservation laws are needed to assure reliable results for all simulated plant conditions.

The simulation fidelity must be greater than that of currently available micro and minicomputer codes or of training simulators. It must be equal to or better than that of systems codes with one-dimensional core flow and point kinetics simulation capability.

1.3.2 Interactive Operation

An efficient plant analyzer must produce an answer before its user has forgotten the question. This requires easy and convenient access to the plant analyzer. It requires the capability of changing input parameters interactively, and of receiving on-line responses by the plant analyzer. Results must be displayed on-line to provide control over the simulation.

Efficient testing of operating procedures and their optimization requires that the user of the plant analyzer can observe on-line displays of computed key parameters (coolant level position, pressures, fuel and coolant temperature, etc.) and that he can change valves, or trip pumps, etc. interactively and on-line, while the simulation is in progress.

On-line displays and interactive control over a simulation saves both time and resources, as many transients can be terminated after the needed results are obtained, or after it has been determined that the simulation does not follow the expected course. Batch processing cannot offer these savings.

Interactive capabilities encourage the user of the plant analyzer to explore power plant behavior, and to seek answers to "what if" questions, which would never be answered in normal batch mode operation. Interactive operation is the central feature of simulation. The benefits and the power of timely interactive simulation must be seen to be believed.

1.3.3 High-Speed and Low-Cost Simulation

Faster than real-time simulation speed is needed for the following reasons: firstly, all parametric studies require many repeated calculations with one (or rarely more) parameter(s) changed each time. Experience shows that standard computing speeds are inadequate and needed parametric studies are simply not carried out because they take too much time and cost too much. Statistical

studies have been performed in the past by the NRC to assess error margins of large computer codes, for example, but the sample size was inadequately small because of high computing costs.

Secondly, a large number of transients may take as long as one to four hours or even days. Such transients must be simulated in much less time, particularly when a number of variations need to be computed. Typical examples are the prediction of peak clad temperatures, containment pressure and coolant inventories (in storage tanks, sumps, suppression pool, etc.) during long-term recovery maneuvers. Often an envelope of a key parameter (peak clad temperature) is sought as a function of another parameter (over its uncertainty range), requiring one transient simulation for each point on the envelope. Many transients must then be simulated to determine the envelope.

It has already been demonstrated that much faster than real-time simulation speeds are advantageous even for short transients, lasting as little as a few minutes. Real-time simulation speed is desirable only for some plant-specific training; faster than real-time simulation, coupled with interrupt capabilities, is preferred for generic training. Slower than real-time simulation has no advantages over faster than real-time simulation.

It has been said that the user could not follow the on-line displays from a simulation which proceeds faster than with real-time speed. One should notice, however, that hardly anybody takes as much as a few seconds to read a graph from a one-hour-long transient. Most transients are analyzed and re-analyzed after the computer simulation is completed. It is the plant analyzer's obligation to provide the answers as quickly as possible and to have the flexibility of computing at low speeds only when user intervention is needed in the simulation. Even then, the simulation speed should not be less than real-time speed, because realistic interventions during the simulation must be achieved in real-time by the operator.

It is self-evident that simulation costs need to be reduced below the costs now incurred from standard batch processing on mainframe or supercomputers. "Costs" include here not only computing expenses comprising capital investment and operational costs, but also the manpower expenses from initiating, executing and evaluating the simulation. Therefore, user convenience and computing speed are both needed for cost-effective simulation. Both must be achieved, together with high simulation fidelity and at low cost.

1.4 Purpose of Report

It is the objective of this document to summarize the capabilities and limitations of state-of-the-art candidate computer codes, computers and simulation technologies suitable to support the design, optimization, operation and maintenance of new isotope production reactors. Excluded from this summary are proprietary codes and facilities which are not accessible to peer review (as required for example in the Code Scalability, Applicability and Uncertainty (CSAU) Evaluation Methodology [Compendium, 1987, see Section 4.3.4]).

2. EXISTING TECHNOLOGIES FOR COMPUTER ANALYSES OF NUCLEAR REACTOR SYSTEMS

Below are presented currently existing techniques, originally developed to analyze nuclear power plant behavior and now being considered to analyze design concepts, operational characteristics of, and accident sequences in, isotope production reactors. Excluded from this presentation are proprietary computer codes which are not accessible to peer review. Also excluded are codes which serve to analyze severe accidents with core degradation. The state-of-the-art code for core degradation and pressure vessel failure analyses is MELPROG [Camp et al., 1987], which has the same type of field equations as the thermohydraulics code TRAC-PF1 [Liles et al., 1988], contains the same constitutive equations for energy and momentum transfer as TRAC-PF1 (see Camp et al., 1987, pp. 15 and 16) and employs the same stability enhancing two-step (SETS) method for numerical integrations as used in TRAC-PF1 (see Camp et al., 1987, p. 17). Consequently, existing techniques for analyzing core degradation consequences in the reactor vessel are deemed to be covered in this document as part of the summary on thermohydraulics codes for mainframe computers (see Section 2.3 below).

We present available techniques in three categories, namely the techniques associated with microcomputers, minicomputers including array processors, and mainframe computers, including supercomputers.

2.1 Microcomputer-Related Analysis Methods

The computing speed and memory capacity of microprocessors (286, 386) in lap top, personal and desk top computers has been and will be increasing in the future and so will be their capability to analyze reactor transients. As a consequence, the distinction between micro and minicomputer capabilities may disappear in the future. At this time and in the near future, however, microprocessor-based reactor analysis methods cannot be expected to meet all four criteria listed above in Section 1.3.

Microprocessor-based reactor analysis technologies fall into two categories, namely in reactor simulation and reactor calculation methods (cf. Section 1.2).

Examples for the microprocessor-based reactor simulation methods are the proprietary codes RETACT of Singer Link-Miles, CETRAN of Combustion Engineering and PCTRAN by Clifford Po, and others. The codes provide in principle the kind of simulation environment discussed in Section 1.2, they meet Criteria (b) and (d) as specified above in Section 1.3. Criteria (a) and (c), however, are not met simultaneously, because microprocessors set up for simulation do not have the necessary detail of modeling nor the computational accuracy for realistic and accurate reactor simulation. The main reason is that real-time simulation speed cannot be achieved with current microprocessor simulation technology, while providing the nodalization detail required for accurate simulation. Neither the RETACT nor the PCTRAN versions for BWR simulations demonstrated the capability of analyzing the power and flow oscillations, as they occurred on March 9, 1988 at the LaSalle County No. 2 BWR power plant.

In the public domain is the Modular Modeling System (MMS)-based PWR Compact Analyzer of the Electric Power Research Institute [Ipakchi, Khadem and Peng, 1987]. The reference delineates on p. 6-1 the basic factors limiting this technology in its use for interactive real-time simulation. Figure 6-2 on p. 6-7 shows that the EPRI PWR Compact Analyzer simulates between seven and thirty times slower than with real-time simulation speed, yet, as shown on p. 5-5, the simulation is limited to "operating conditions ranging from full power to stand-by, and natural circulation"..., when "the reactor coolant system remains subcooled." Clearly the MMS-based simulation facility of EPRI, utilized on microcomputers, does not meet Criteria (a) and (c); it is not clear whether it meets Criterion (d). For MMS implementation on minicomputers, see Section 2.2 below.

Microprocessor-based calculational methods have been developed at several institutions by implementing existing FORTRAN codes, originally programmed for general-purpose mainframe computers, on personal computers (PCs) or on desk top computers. The result is a calculational facility which fails to satisfy Criteria (b), (c) and (d) in Section 1.3. The facility requires remarkably low capital investment for acquisition, but it fails to meet Criterion (d) because of the high manpower cost needed afterwards for applications. It fails Criteria (c) and (d) because it is impossible to produce calculational results on time and at low cost.

The codes implemented for computer calculation on PCs and desk top computers are represented by RELAP5 [Ransom et al., 1985], RETRAN [McFadden et al., 1981], and possibly by TRAC-PF1/MOD1 [1986]. The reader is therefore referred to Sections 2.3.1-3 below for a discussion on code capabilities and limitations affecting the degree to which existing calculational methods on microprocessors meet Criterion (a) in Section 1.3.

It has been claimed that large numbers of microprocessors can be combined to work in parallel, while exchanging data through a hypercube interconnection (Westinghouse, through Paralex, a venture company) or through a ring bus connection (Control Data). While it is correct that computing power expressed in million floating point operations per second can be increased at low cost to enormous proportions, the manageability of programming and maintaining a reactor systems simulation for efficiently utilizing this raw computing power has never been demonstrated, even in principle. It should be recognized that computing speed improvements through the use of parallel processing in general-purpose computers cannot be expected to exceed the improvements achieved by implementing the vectorization of mainframe computer codes on supercomputers (CRAY-XMP). These improvements are a disappointingly modest increase in computing speed, by a factor of 2.4 to 2.8 [Ishiguro, Harda, Shinozawa and Naracka, 1985]. The reason for such disappointments is that for both programming a set of array processors and for vectorizing a code, one needs to have extensive regularity in the set of computer instructions. One must be able to separate the instruction set into independent instruction streams, one for each processor, or one needs large arrays on which to operate in parallel. Reactor systems simulation codes do not have the above regularity, as can be seen from the many disappointing efforts aimed to speed up their execution [Liles et al., 1983, etc.].

In conclusion, minicomputer-based reactor systems analysis methods cannot at this time and in the near future meet all the four criteria which are specified in Section 1.3 and required for realistic and cost-effective support of reactor systems design, operation and maintenance. It would require a major effort in computer systems software development to utilize a combination of microprocessors in an interactive simulation environment as described in Section 1.2. It is far less expensive to utilize existing simulation computers which already provide a genuine simulation environment. One such simulation system is described below in Section 2.2.3.

2.2 Analysis Methods Related to Minicomputers and Array Processors

2.2.1 Minicomputer-Related Technologies

General purpose minicomputers are used in full-scope training and in some compact simulators. Full-scope training simulators are plant-specific replicas of control room consoles and contain a minicomputer replacing the power plant. Singer-Link, Electronic Associates, Westinghouse and Combustion Engineering have built almost all existing training simulators. The S.E.L. (Gould) minicomputer is the most frequently used minicomputer in simulators.

The capabilities and limitations of training simulators have been reviewed in detail in 1980 and 1981 [Wulff, 1980, 1981; Cheng and Wulff, 1981]. It has been found that training simulators built prior to 1981 are limited to the simulation of steady-state conditions and quasi-steady transients within the parameter range of normal operations. Specifically, PWR simulators could not simulate two-phase flow conditions in the primary coolant loops nor the motion of two-phase mixture levels beyond the narrow control range in the steam generator secondary side. Significant discrepancies were found in a PWR simulator and RETRAN code comparison [Wulff and Cheng, 1981].

Training simulators are designed to reproduce, at real-time simulation speed, accurate start-up, load-following and shut-down transients and the plant responses to a very large number of malfunctions. All full-scope training simulators have the capability of introducing arbitrary combinations of a large number of malfunctions, but they lack the ability of reliably simulating the consequences from multiple malfunctions, primarily because of their inability to simulate the dynamics of nonhomogeneous, nonequilibrium two-phase flows under abnormal reactor conditions. Even though model improvements have been introduced in simulators built and delivered after the publication of the above referenced assessments, almost all the training simulators are still limited to the simulation of normal operations.

Full-scope training simulators employ standard FORTRAN computer programs and general-purpose computers. This is not an optimum combination for efficient simulation. Real-time simulation speeds are often achieved only through modeling compromises. Faster than real-time simulations are impossible on full-scope training simulators but they are required for efficient parametric studies.

The most recently built training simulators from Singer-Link have significantly improved simulation capabilities, based on the newly developed RETACT coolant dynamics code. This proprietary code executes on general-purpose

minicomputers, but cannot achieve the simulation speed needed for parametric studies or for the support of accident management (cf. Section 2.1).

Training simulators are in general unsuitable for use as engineering plant analyzers. Most existing training simulators do not have the necessary range of realistic simulation capabilities, they are expensive to operate and, most frequently, committed full-time to training and requalification. They achieve at most real-time simulation speed and are therefore unsuitable for parametric studies (see also Section 1.3.3).

A general package for power plant analyses with the use of minicomputers has been developed under the auspices of the Electric Power Research Institute (EPRI). It is called the Modular Modeling System (MMS) [Modular Modeling System, 1983; Divakaruni and Wong, 1984]. For MMS's more recent use in microcomputers, see Section 2.1 above. MMS was originally designed for fossil power plants, and it has been extended recently to simulate two-phase flows in nuclear reactor systems. It utilizes two continuous system simulation languages, either the Advanced Continuous Simulation Language (ACSL) from Mitchell and Gauthier Associates, Inc., or the Engineering Analysis System 5 (EASY5) from Boeing Computer Services Co.

The major emphasis of the MMS development is modularity, which is important when the same code is to be used for many very different object systems, but it is not useful for simulating a <u>particular</u> nuclear reactor plant. In fact, modularity is detrimental from the point of simulation efficiency because it necessitates computational overhead.

It should also be noted that the two-phase coolant dynamics simulation capabilities in MMS are not well established. The careful reader of the reference by Divakaruni and Wong will notice, for example, in the contributions by S. J. Oh and J. P. Sursock, that conservation laws are written for control volumes and fixed boundaries, but then applied to control volumes with moving boundaries. Also, the equation for the speeds of moving boundaries is incorrect because of errors in differentiation. The user of MMS can, at best, utilize some selected plant component models and the overall simulation system, but must develop his own two-phase flow modules for the nuclear reactor of interest.

New general-purpose minicomputers are being offered for real-time simulation by Modcomp (Classic II), Control Data (AFP), EAI (EAI 2000) and others. These minicomputers offer high computing speeds and large memory capacities. However, as general-purpose computers they come only with standard FORTRAN compilers and lack the systems software which (a) is needed to utilize efficiently the theoretical computing speed capacity in reactor systems simulations, and (b) is required to provide the simulation environment described in Section 1.2 above. Experience has shown, that power reactor thermohydraulics codes execute on advanced minicomputers at a small fraction of full computing speed, that the turnaround is slow and the timely resolution of technical issues is difficult or impossible to achieve, particularly when many parametric variations are to be analyzed. Parametric studies accompany most analyses of reactor systems behavior.

Large FORTRAN codes, such as TRAC-PF1, RELAP5 or RETRAN, have also been implemented on minicomputers. The execution of these codes on minicomputers is far slower than on mainframe or supercomputers, where it is already inadequate (cf. Section 2.3 below).

In summary, general-purpose minicomputers do not meet the four requirements specified in Section 1.3 above. They are inadequate for realistic, cost-effective and timely (on-line, interactive) simulation of nuclear reactor systems, because they have either severe modeling limitations, or they execute too slowly, or they are not available most of the time (training simulators), or they are not open to peer review (proprietary codes, such as Westinghouse's TREAT, Singer-Link-Miles' RETACT, etc.).

2.2.2 Array Processor Related Technologies

Array processors are widely promoted for complex system simulation at high-speed and low cost, even though array processors are not especially suited for dynamic (reactor) systems simulation.

Array processors consist of arrays of two or more central processing units, complete with memory and I/O processing. They were originally designed for signal processing (CAT scan), and are therefore very powerful for operating in parallel on independent parallel streams of instructions and data. A simulation, however, cannot be divided into independent parallel streams of instructions because of the physical interdependence between components and processes in the plant. With array processors, one has the choice between lagging a number of parameters, or idling some of the processors. Lagging is the use of parameter values from the previous time step when they are needed for the current time step before they can be computed. The resulting time skewing increases the truncation error accumulation and causes numerical damping. Idling of some of the processors, on the other hand, leads to the wide-spread observation that the computing speed increases much less than proportionally to the number of parallel processors.

Array processors have the potential of providing great computing economy in three-dimensional transport simulations (fluid dynamics, neutron kinetics) which involve operations on large arrays and a large number of independent operations. Array processors are also very powerful in processing a large number of signals from independent sensors. But array processors are not suitable for the largely scalar structure of nuclear reactor systems simulations. It is extremely difficult to program arrays of processors, both during initial code development and later for code improvements, such that the processors are effectively utilized.

2.2.3 Special-Purpose Simulation Components

BNL has developed a new technology for high-speed, low-cost, interactive simulation of transients in nuclear reactors. The new technology has been implemented for BWR-4 power plants in BNL's Engineering Plant Analyzer (EPA). It has been developed under the auspices of the USNRC and consists chiefly of the High-Speed Interactive Plant Analyzer code HIPA-BWR4/I for BWR-4 plants with Mark I containment, and of the special-purpose peripheral processor, the AD10

of Applied Dynamics International (ADI) in Ann Arbor, Michigan. A detailed description of the Plant Analyzer can be found in the reference by Wulff et al. [1984a]. BWR-5 and BWR-6 versions have also been developed. The BNL Plant Analyzer represents the most advanced technology for nuclear power plant simulation in minicomputers and is suitable for supporting plant operations. It is based on an integrated concept for cost-efficient simulation, a concept in which modeling, numerical methods and computer architecture are coherently optimized for efficient simulation of BWR power plants. The techniques developed, however, are general and can be applied to any other type of reactor as well. In fact, the new technology is now being extended in the advanced AD100 simulation system from ADI, for the implementation of PWR simulation capabilities.

The uniqueness and superior performance capability of the BNL Engineering Plant Analyzer is evident from the fact that it is the only available simulation facility that was capable of reproducing[2] within less than three days the actual power and flow oscillations which occurred on March 9, 1988 at the LaSalle-2 station. Over sixty detailed and complete plant simulations have been carried out after more definitive plant data became available and before any other NRC-sponsored BWR systems code could produce its first LaSalle-2 simulation.

2.2.3.1 The AD100 Hardware and Software Systems

The AD100 Simulation System is currently the only facility available for simulating large complex systems which meets all of the four criteria introduced in Section 1.3 and needed for viable reactor systems analyses; it provides for realistic and accurate simulations of normal and postulated accidental transients through convenient, on-line interactive operation with on-line graphic displays, at higher than real-time speed and at costs far below that of other existing analysis methods.

As a special-purpose computer, the AD100 system is specifically designed for the single purpose of complex system simulation. It is unquestionably better suited for simulating complicated nuclear reactor systems, than are the general-purpose computers. The AD100 embodies inherently a unique combination of hardware architecture and systems software that has been demonstrated to outperform supercomputers at costs much below that of supercomputer operations.

The AD100 Simulation System consists of nine simulation task-specific microprocessors, e.g. for numerical integration, nonlinear function generation, etc. The microprocessors work in parallel, have pipeline architectures for high-speed execution and contain built in algorithms defined through systems firmware. This means that the programmer has no need to develop, implement and debug such simulation tasks as numerical integration, etc. Instead, he is free to concentrate on the formulation of the mathematical models describing the system of interest.

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[2]Initial calculations, based on preliminary information on plant conditions.

The AD100 Simulation System employs 64-bit floating point arithmetic. It requires no scaling (as did the AD10), has a dynamic range of 300 decades (100 times greater than the AD10) and is two to five times faster than the AD10, which achieved simulation speeds more than 100 times faster than the CDC-7600 mainframe computer and eight times faster than real-time speeds, while simulating the BWR plant, complete with Nuclear Steam Supply System, Balance of Plant, Control Systems, Engineered Safety Systems and Containment.

The AD100 Simulation System has up to 256 analog to digital and/or digital to analog converters for interfacing with instrumentation, control systems, analog input devices or analog display systems. The AD100 also has four dual-ported memory processors for digital interfacing through 19 kword (16 bit each) buffers at the rate of 10 million words per second in each channel. A fiber optics interface is also available with the transfer rate of 200 Mbits/sec in each direction. The AD100 can be linked to DEC, GOULD (SEL), Harris and Perkins Elmer Computers. The AD100 is controlled through any VAX computer, such as the Micro-VAX-II workstation.

The AD100 provides, through its systems software, a fully interactive on-line simulation environment, complete with graphics display.

2.2.3.2 The Systems Simulation Language and Program Transportability

It has long been recognized, particularly by the large membership of the Society for Computer Simulation (SCS), that FORTRAN is not the most suitable language for simulation computer programs. Continuous system simulation languages have been developed and standardized.

A number of continuous system simulation languages have been written for FORTRAN compilers. Typical examples are EASY5 and ACSL (cf. Section 2.2.1). Such programs offer the convenience and clarity of the simulation language but since they are machine-translated into FORTRAN, they do not provide the execution efficiency afforded by the direct translation from the high-level system simulation to the machine language.

The BNL Plant Analyzer for BWR simulation accepts the Modular Programming System (MPS-10) language. The most important features of MPS-10 are its simplicity and compactness. In MPS-10 one needs to satisfy only one rule in order to utilize fully the parallelism of the AD10. Thirty-six thousand executable FORTRAN statements reduce to less than six thousand MPS-10 statements, even though MPS-10 has no DO-loop equivalent. One of the main benefits from MPS-10 is the fact that only 5,660 MPS-10 statements instead of approximately 60,000 FORTRAN statements need to be executed to advance the simulation from one time level to the next.

The more advanced AD100 computer utilizes the AD100 language ADSIM, which is superior to MPS-10, because it is more compact, more readable and more flexible. ADSIM is easy to learn. The AD100 firmware translates ADSIM directly into machine instructions for more efficient execution. ADSIM is being maintained by ADI.

As an example, the ADSIM language needs only six short program lines to represent an attitude controller, consisting of a lead-lag compensator, a lag compensator, a hysteresis switch, a summer and a second-order actuator. This example demonstrates the superiority of ADSIM over FORTRAN. ADSIM has been developed to the point where it qualifies as a standard of simulation languages.

Use of ADSIM as the programming language would greatly facilitate the initial program development and, more importantly, it would simplify later program modifications. It will be easier and less expensive to change the program under ADSIM than under FORTRAN. However, the AD100 is also capable of executing FORTRAN programs. FORTRAN codes are less efficient than the ADSIM code, even on the AD100 system.

Software Transportability permits the execution of a code on different computers without reprogramming. In the past, the nuclear industry has insisted on transportability for obvious reasons of economy. As it turned out, however, all large codes need to be converted before they execute on a computer other than the one on which they have been developed, often even when both computers are of the same type but operate under different operating systems. Conversion takes upwards of six man-months of effort.

It must be pointed out that transportability and simulation efficiency are mutually exclusive. One program can fully utilize the particular features for computing efficiency of only one computer. Experience has proven, for example, that standard FORTRAN, when executed on a CRAY supercomputer, produces disappointingly small improvements over standard serial machines.

Transportability of programs is less important, when relatively inexpensive minicomputers are involved, than it is for costly mainframe computers. The cost of an entire AD100 can be recovered within a relatively short time from savings in both computing and manpower expenses. This is true even more so, because the ADSIM language is easy to adopt.

It must be stressed, that the ultimate use of any plant analyzer requires no familiarity whatsoever with the programming language. Only program changes require ADSIM programming, and it has been shown, that learning the simulation language is a negligible effort when compared with the modeling effort itself. Moreover, the interactive capabilities of the AD100 under ADSIM are unsurpassed for checking out new coding.

Should it be desirable to execute the simulation software of the plant analyzer in FORTRAN on a general-purpose computer, then it would be possible to translate by machine the ADSIM code into FORTRAN. However, it must be recognized that the resulting FORTRAN code would not execute nearly as fast as the ADSIM code.

Remote Access Capability is provided by the AD100 system through on-line interactive response at the locations of engineering offices where issues of reactor plant behavior must be resolved within a reasonable time, even though the plant analyzer may be located elsewhere.

Remote access can be provided through standard telephone linkage, via a standard modem. The remote user is able to access the plant analyzer from a DEC Micro-VAX-II workstation.

From the workstation, the user is able to control the plant analyzer, to display plant analyzer results graphically and in tabular form. The user is also able to assemble his plant-specific input data file, containing geometric data, operating parameters, control parameters, setpoints, etc. Before or during the simulation, the user can enter on-line any operator action or malfunction and observe instantly the resulting plant response. This remote access capability has been demonstrated by using the BNL Plant Analyzer for BWR simulations.

2.2.3.3 Advanced Modeling Principles in BNL Engineering Plant Analyzer (EPA)

The simulation efficiencies achieved for BWR (and PWR) power plant simulations are due in part to the application of six advanced modeling principles which are unique to the BNL EPA, as no other computer code for nuclear reactor analyses employs these principles. They are equally applicable to isotope production reactors. The six principles [Wulff, Cheng, Lekach, Mallen and Stritar, 1984] constitute the most important differences between the modeling in the BNL EPA and in other systems codes. Detailed explanations of these principles are found in the cited reference (see Appendix A), their important implications are discussed below.

The six principles address (i) the selection of models, particularly for two-phase flow coolant dynamics, (ii) the priority ranking of processors, (iii) the use of analytical integration, (iv) the elimination of iterative procedures, (v) the use of pretabulation for complex functions, and (vi) the choice between implicit and explicit integration methods, based on permissible integration step sizes (accuracy, stability) and frame times (time required by the computer to advance from one to the next time step).

The application of the first modeling principle reveals that the six-equation two-fluid model is inferior to the four-equation drift flux model because the former cannot be closed for lack of instrumentation which is needed to develop two-fluid model transfer laws for mass, momentum and energy between phases and for momentum and energy between each phase and the flow channel structures. The second modeling principle is called for by prudence in general and leads to justifiable model simplifications. The third principle reduces difficulties arising from numerical instability, and it reduces the computational errors due to numerical diffusion. Modeling principles number four and five reduce the number of necessary arithmetic and logical operations by orders of magnitude, thereby increasing the simulation speed. Application of the sixth principle reveals that explicit numerical integration is superior to implicit integration for all but the slowest or quasi-steady transients.

These principles evolved in part from the efforts to utilize to the fullest extent possible the high-speed simulation capabilities of the ADI special-purpose parallel processors, but they would be equally advantageous in other computers as well. Unfortunately, they have not been employed in other large-scale nuclear reactor analyses.

2.2.3.4 Summary on Special-Purpose Simulation Computer Technology

The features of the AD100 Simulation System described above clearly show the superiority of computer simulation over the traditional computer calculation on general-purpose computers. The simulation technology developed at BNL is the only proven method for efficient simulation, as it combines advanced simulation modeling principles with simulation computer architecture and simulation system software, specifically designed by ADI for the efficient simulation of complex systems, such as an isotope production reactor.

2.3 Analysis Methods Related to Mainframe and Supercomputers

The greatest efforts toward accurate and detailed computer analysis of transients in nuclear reactors for power generation have been made in the last fifteen years under the auspices of the U.S. Nuclear Regulatory Commission (USNRC) and of the Electric Power Research Institute (EPRI). These computer analyses employ mainframe and supercomputers, they constitute computer calculations, rather than simulations. They exceed in scope and computational detail the computer-aided analysis tools available now for research reactors (HFBR, HFIR, etc.) and for isotope production reactors.

The USNRC and EPRI-sponsored efforts toward nuclear reactor thermohydraulic analyses led to six systems codes, namely to RELAP3B and RELAP4, now superseded by TRAC [TRAC-PF1/MOD1, 1986], RELAP5 [Ransom et al., 1985], to RETRAN [McFadden et al., 1981] and to RAMONA-3B [Wulff et al., 1984b] and TRACB [Taylor, et al., 1984].

The original development of these codes, except RAMONA-3B, was dominated by Large Break Loss of Coolant Accident (LBLOCA) simulations. Once the greater importance of Small Break Loss of Coolant Accidents (SBLOCA) was recognized after the Three Mile Island accident, work was started in 1979, and is still continuing, on also including phenomena associated with SBLOCA and with similarly frequent events. The systems codes are now claimed to simulate all events short of core degradation.

These systems codes are written in FORTRAN and executed chiefly on large mainframe computers (CDC-7600, Cyber 176, etc.). The size and computational effort of these codes, particularly of TRAC, increasingly necessitates their execution on supercomputers (CRAY or Cyber 205). Even though efforts have been made to implement some of these codes (RELAP5) on mini and advanced micro-computers, none of these implementations could possibly meet the analysis requirements customarily imposed on national laboratories for the timely resolution of safety issues. Therefore, only large mainframe and supercomputer implementations are considered here to be relevant to this review.

Below are discussed those characteristics of the systems codes which are relevant to the selection of future analysis tools. Particularly, it is shown that these large systems codes are very complex and expensive to use, so that, in part, they do not achieve the objectives for which their complexity is responsible and that they are unsuitable for efficient interactive simulation of nuclear reactor systems, especially for day-to-day production analyses in support of plant operations.

2.3.1 The TRAC Code has been developed for the USNRC in two versions. The first versions are TRAC-PD[3] and -PF for PWR plants. Both have been developed by Los Alamos National Laboratory (LANL). The second version is TRAC-BD for BWR plants and was developed from TRAC-PD by INEL. The TRAC series was started, together with RELAP5 discussed below, as an advanced thermohydraulic code suited to supersede RELAP4.

The TRAC codes are considered by many today to be the most reliable and detailed thermohydraulic codes for reactor systems, and they were once proposed to become the standard codes of the USNRC. However, consultants to the Advisory Committee on Reactor Safeguards' Subcommittee on Thermal Hydraulic Phenomena, who reviewed the TRAC-PF1 code, recommended against further use of this code beyond LBLOCA analyses, because it is "too complicated" and "not a useful tool" [pp. 316-319, ACRS 1988a], because there is "overkill in some of the great detail" and "not enough physics" [p. 326, ACRS 1988a], because "there's too many cases where the physics is not right" [p. 351, ACRS 1988a].

The status of the TRAC codes is based mainly on these four claims:

(a) TRAC used a six-equation two-phase flow model,
(b) TRAC predicts three-dimensional two-phase flow in the pressure vessel,
(c) TRAC is flexible and accommodates any test facility as well as full-scale power plants, and
(d) TRAC has been assessed.

Claims (a), (c) and (d) apply equally well to RELAP5. All claims are relevant to the nuclear reactor analyses because they are said to be needed to achieve the stated objectives in realistic and accurate predictions. Therefore, we review the claims below.

Claims (a) and (b) are related to the modeling of the two-phase flow dynamics of the reactor coolant. This modeling is the central problem of realistic and accurate reactor system simulation.

Two-phase flow models are based on three, four, five or six conservation equations for mass, momentum and energy [Wulff, 1981a]. The simplest model is the three-equation model of homogeneous equilibrium flow. It hardly ever applies to either light or heavy water-moderated reactor conditions, because it implies that equal temperatures and equal velocities exist in vapor and liquid phases or, equivalently, that extremely slow processes occur simultaneously with strong mixing. The most rigorous and detailed model has six equations, three conservation laws for each of the liquid and the vapor phases. It implies no significant restrictions. But, most importantly, it requires, in addition to all the intrinsic constitutive laws which are always needed (thermodynamic properties), also seven extrinsic constitutive laws which describe the transfers of mass, momentum and energy between the phases and of momentum and energy

[3]Letters B,D,F and P designate "BWR", "detailed", "fast", and "PWR", respectively.

between each phase and the wall [Wulff, 1981]. These laws are not known for reactor conditions except in separated flows (annular flow or stratified flow in horizontal ducts) [p. 348, ACRS, 1988a, Wulff, 1981]. These fundamental laws will not be available until the necessary basic instrumentation has been developed, and not until the appropriate models have been first derived from experiments and then verified.

In the meantime, TRAC uses mixture correlations as its data base (reported in TRAC-PD1, 1981; TRAC-PF1, 1984; Taylor et al., 1984; Adessio, 1985], just as the four-equation drift flux model [Zuber and Findlay, 1965]. TRAC depends upon unverified hypotheses to complete its six-equation models. Claim (a) is invalid for TRAC (and for RELAP5).

The six-equation two-fluid model implies a time consuming computing effort and a tremendous requirement on computer memory capacity. Yet, the two-fluid model cannot produce more information than is contained in its data base. It should be used only in separated flows, where the interphase geometry and, with it, the transfer laws are known and where the simple mixture model does not apply. The indiscriminate application of the six-equation model in TRAC is in part responsible for TRAC's extremely high computing expenses.

Concerning Claim (b), TRAC integrates for the reactor vessel three momentum balances each for vapor and liquid, to compute three-dimensional flow fields. However, the momentum balances lack the internal fluid shear term, and for this reason TRAC failed to reproduce the only available experimental results obtained to verify its capability of computing multi-dimensional flow distributions [Saha et al., 1982].

TRAC contains field equations [TRAC-PD2, 1981] which apply to two-phase flow domains not penetrated by solid structures. These equations are used in TRAC for reactor vessel regions containing fuel rod arrays. The equations are used in TRAC without having been suitably volume averaged over a domain containing a two-phase coolant mixture and solid structures. The appropriate averaging can be found in standard text books [Slattery, 1972].

This well-established volume averaging for heterogeneous systems would have produced the unambiguous relation between measured and predicted velocity components, which is indispensable for code verification, and it would have produced both the wall shear and the internal shear, quite contrary to a recent claim [Adessio et al., 1985] that some (unpublished) averaging process could eliminate internal shear. It is well known from elementary fluid dynamics that global integrations, which lead from internal shear to shear at solid boundaries, also eliminate the capability of computing velocity distributions between such boundaries. This capability is the stated aim of TRAC. TRAC is obviously incapable of computing the simplest velocity distributions known: the two-dimensional velocity distribution of single-phase flow, laminar or turbulent, in a common pipe, but it is expected to predict three-dimensional two-phase flows in complex geometries (downcomer or plena).

The lack of internal shear is compensated for by "tuning" of wall shear, which is possible only if yet to be predicted results are known a priori. Therefore, TRAC is in principle a one-dimensional hydraulics code.

Concerning claim (c), TRAC has indeed the capability of accommodating, via input data specifications, both small-scale experimental facilities and full scale power plants. This feature is absolutely essential (although not sufficient) for "extrapolating" from small-scale test results to the unknown behavior of full-scale power plants. Moreover, this important capability provides a possible link between plant-specific full-scope training simulators and plant analyzers on the one hand, and well-controlled laboratory experiments related to severe off-normal conditions on the other.

It must be pointed out, however, that this flexibility cannot be utilized in the application of a verified simulation facility (such as a training simulator or plant analyzer) to a specific nuclear reactor. Moreover, TRAC (and RELAP5) are far too flexible, permitting for example physically impossible flow connections (e.g., from core interior to interior turbine stages) at the expense not only from additional overhead in computational efforts, but also from the requirement for specifying unnecessary input data. The indiscriminate flexibility for system assembly in TRAC contributes to its computing expense.

Finally, concerning claim (d), TRAC has indeed undergone intensive developmental and independent code assessment, which has been supported by a large international effort on experiments. The most complete TRAC-PF1/MOD1 assessment was performed through the use of the Code Scaling, Applicability and Uncertainty Evaluation Methodology [CSAU TPG, 1989]. TRAC was assessed regarding its ability to predict the maximum cladding temperature rise during a Large Break LOCA, for which the code had been originally developed. Table 40 on Page 99 of the cited report reveals[4] that TRAC-PF1/MOD1 predicts for the first (blowdown) peak the maximum cladding temperature rise of 509°F with the uncertainty of 395°F (78%), part of which (22%) is attributed to the net effect from systematic modeling errors. The uncertainty for the second and third (reflood) peaks is larger (154%), as expected, because of error propagation and accumulation. This shows clearly that the uncertainty of TRAC predictions is larger than previously reported and that it must be expected to be larger for Small Break LOCA analyses. In fact, code uncertainties must be determined for each specific transient. Uncertainty estimates, if available from previous developmental and independent code assessments, cannot be generalized to other applications, particularly if code modifications were also made in the past. Once a code such as TRAC had been perfected, it could serve for selected benchmark calculations, but it would be far too expensive for production analyses for day-to-day resolution of reactor related issues.

TRAC has been combined with a user-friendly graphics facility to form a Nuclear Plant Analyzer (NPA). The resulting software package is very large, so as to run only on a CRAY supercomputer. The operation of the TRAC-NPA is expensive, because it executes too slowly on a very costly computer and it requires of the user that he be familiar not only with the plant of interest but also with the complicated TRAC code.

In summary, the TRAC code and its NPA version are unsuitable for efficient, on-line interactive reactor system simulations. TRAC is expensive to execute

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[4]the initial clad temperature is 650°F.

because it is programmed only with standard methods in standard FORTRAN for general-purpose computers to simulate complicated three-dimensional two-phase flow processes with the rigor of the two-fluid model. TRAC is inefficient because it is expensive to execute yet unable to reliably simulate three-dimensional flow processes with the rigor of a true two-fluid model.

2.3.2 The RELAP5 Code has been developed for the USNRC at INEL [Ransom et al., 1985; Trapp, 1985]. It started out with a five-equation model and was later expanded to contain six equations, as TRAC. For the purpose of this review, the differences between TRAC and RELAP5 are insignificant. RELAP5 has been widely accepted because it appealed to the many users which are familiar with RELAP4 input data formats.

RELAP5 started out as a five-equation thermohydraulics code, with a strong emphasis on the need for a balance between mathematical model formulation and the data base available for supporting the model. Later, a sixth equation was introduced. Thus, the principal difference between RELAP5 and TRAC disappeared. RELAP5 simulates only one-dimensional coolant flows (cf. discussion on TRAC claim (b) in Section 2.3.1). It uses point kinetics as TRAC-PF1, but it employs a numerical integration scheme which differs from that of TRAC. RELAP5's integration scheme has never been carefully documented, and it is claimed to involve the solution of nonlinear equations without iteration. As there is no basis for such a solution scheme in the open literature, it is not clear whether RELAP5 produces the solution to the original conservation equations. Moreover, the numerical methods in RELAP5 contain also ad hoc time averaging which was declared to be wrong during the ACRS review of RELAP5 [pp. 100-101, 117-118, ACRS, 1988b]. Thus, the numerical integration methods in RELAP5 remain an open issue.

RELAP5 contains the six-equation two-fluid model which cannot be closed for lack of needed closure relations as is clearly evident from [Dimenna et al. (1988)]. This reference states (p. 4-1) that some correlations are based on "engineering judgement", because of the "incompleteness in science". It shows on pp. 4-5 through 4-59 that thirty-seven ad hoc multipliers are used just to "tune" interfacial heat transfer in RELAP5, largely in order to justify the code developer's intuition. Chaotically deforming interface geometries are assumed to be spherical (pp. 4-5 and 7), "ad hoc correlations" had to be made up for lack of bases in physics, theory or experiment (pp. 4-20, 24, 25 and 28). It is obvious from this that the two-fluid model, implemented in RELAP5 to provide computational detail, cannot provide that detail but requires the computing resources needed to integrate the six equations of the two-fluid model.

RELAP5 also has a user-friendly graphics package, and its use with RELAP5 constitutes INEL's Nuclear Power Plant Analyzer (NPA). The INEL NPA was scheduled to support the USNRC Technical Training Center for both on-line and off-line simulation of nuclear power plants, using previously computed results for off-line operation. The INEL NPA, however, failed to function in either mode, its remote access and graphics terminal has been removed from the center.

Kmetyk et al. [1984] shows on p. 8 in Table III that RELAP5 calculates transients in small-scale test facilities on CDC-7600 and CYBER 76 mainframe computers 40.0 $\pm$ 60.1 (standard deviation) slower than with real-time speed, on

CRAY-1 it executes 21.3 ± 40.7 times slower than with real-time speed. Summers [1983] shows on p. 180 in Table 5.3.1 that RELAP5 takes 9.0 ± 4.7 times longer than real-time to simulate the semiscale test facility on the CYBER-76 computer. It should be recognized that complete full-scale reactor simulations take even longer than the above computing times.

In summary, what has been concluded for TRAC applies also for RELAP5. RELAP5, even in its NPA operating mode, is unsuitable for efficient, on-line interactive reactor simulations at high speed or low cost. RELAP5 does not meet the four requirements of viable reactor systems simulation specified in Section 1.3 above.

2.3.3 The RETRAN Code has been developed, with RELAP4 used as a starting point, by Energy Incorporated (EI) of Idaho Falls, Idaho for the Electric Power Research Institute (EPRI) [Energy Incorporated, 1981]. RETRAN is the only code for reactor systems simulation that has undergone an official Quality Assurance (QA) program. This guarantees, among other things, agreement between its FORTRAN coding and the RETRAN documentation.

RETRAN employs five field equations for the coolant dynamics; the three-equation model of homogeneous equilibrium flow is expanded by adding the vapor mass balance and the dynamic equation for phasic velocity difference. Thus, RETRAN simulates one-dimensional nonhomogeneous, nonequilibrium two-phase flow and either point kinetics or space-time kinetics (using a space-time factorization method). RETRAN is used for simulating both PWR and BWR transients.

The RETRAN code has a detailed control systems simulation capability which allows the assembly of any control system through input data specifications.

The RETRAN code is a versatile code and is suitable for a large class of transients where thermal disequilibria in the vapor phase are unimportant. Its chief weaknesses are the highly hypothetical formulation of its dynamic slip model for phasic velocity differences, a formulation that is not related to appropriate experiments, then the fact that there is no documentation on consistency of its numerical solution method [Wulff, 1989, p. 207] and further its program size (necessitating large mainframe computers) and its slow execution speed. RETRAN is, consequently, also unsuited for efficient interactive simulation of reactor transients.

2.4 Summary of Existing Computer Analysis Methods

The spectrum of currently available technologies for analyzing nuclear reactor transients by computer has been discussed. Covered are the methods utilizing micro, mini, mainframe and supercomputers.

It is shown that most technologies fall into the traditional category of computer calculation and do not meet the requirements of viable computer analysis suitable for complex nuclear reactor systems design, optimization and maintenance. Computer calculation cannot provide realistic and accurate reactor analysis conveniently, at low cost and in short time.

In contrast, efficient computer simulation has been demonstrated successfully through the use of special-purpose parallel processors in the category of minicomputers, and of advanced simulation principles.

No existing computer code is suitable to accommodate the two-phase flow processors in the core geometry of isotope production reactors, without considerable modifications in key correlations. These modifications require, in part, new experiments.

3. CONCLUSIONS

It is important for the nuclear community, both in government, industry and universities, to begin to take advantage of modern simulation technologies and equipment. Non-nuclear industries, as well as the armed forces, have utilized and promoted the technology of special-purpose, peripheral parallel processors for convenient and efficient high-speed simulations for many years and with outstanding success. It is particularly important for the nuclear industry, because the nuclear industry has to cope with extremely complex processes in very large and complicated systems and it may have to respond quickly to unexpected emergencies. During the past decade, it has been shown that this is not possible with standard methods and general-purpose computers. The most advanced technology should be used to render a powerful tool for a difficult simulation task.

For consistency, the same simulation facility should be used for conceptual design, design optimization, control system development and optimization, for developing normal operating and emergency operating procedures and for safety analysis.

It has been shown in this review that traditional computer codes, written in standard FORTRAN and executed on general-purpose micro, mini, mainframe or supercomputers, cannot meet the requirements of viable reactor simulation as needed in support of isotope production reactors. The chief reason for this inability is that they cannot offer an efficient simulation environment for on-line interactive operation.

It has also been shown that special-purpose computers, such as the AD100 of applied Dynamics International, can provide a complete simulation environment and meet all four requirements for viable nuclear reactor simulation as specified in Section 1.3 above. The AD100 can outperform alternative computers, including supercomputers in carrying out parametric studies, optimizations of design concepts, control systems and operating procedures, in supporting human reliability studies and in training. The AD100 can also be used to test control system components in closed feedback loops, since it has high-speed, high-fidelity analog to digital and digital to analog converters. Above all, it provides the means to resolve issues at such low cost and great ease as cannot be imagined by anyone not familiar with the AD100 Simulation System performance.

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APPENDIX

HIGH-SPEED LWR TRANSIENTS SIMULATION FOR OPTIMIZING EMERGENCY RESPONSE (PAPER REPRINT)

HIGH-SPEED LWR TRANSIENTS SIMULATION FOR OPTIMIZING EMERGENCY RESPONSE*

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ABSTRACT

The purpose of computer-assisted emergency response in nuclear power plants, and the requirements for achieving such a response, are presented. An important requirement is the attainment of realistic high-speed plant simulations at the reactor site. Currently pursued development programs for plant simulations are reviewed. Five modeling principles are established and a criterion is presented for selecting numerical procedures and efficient computer hardware to achieve high-speed simulations. A newly developed technology for high-speed power plant simulation is described and results are presented.

It is shown that simulation speeds ten times greater than real-time process-speeds are possible, and that plant instrumentation can be made part of the computational loop in a small, on-site minicomputer.

Additional technical issues are presented which must still be resolved before the newly developed technology can be implemented in a nuclear power plant.

INTRODUCTION

The Aim of Computer-Aided Emergency Response

Operators of nuclear power plants respond to emergencies by following documented emergency procedures. Emergency procedures are carefully developed sequences of operator actions, designed to mitigate a large number of anticipated emergency scenarios. Even though the scope of emergencies covered by procedures is large and growing, the emergency potential is still larger, and an operator will always face the challenge of having to synthesize a response to unanticipated events. In such an emergency, the operator should be assisted by a computer.

The objectives of computer assistance in a power plant are (i) to monitor the plant's performance, (ii) to diagnose failures in instruments, components and systems, and (iii) to predict quickly the plant responses to several remedial operator actions after an accident. This must be achieved fast enough, so that the operator can select the optimum strategy before committing himself to a recovery maneuver. The latter two objectives are the aims of emergency assistance by computer.

Requirements for Computer-Assisted Emergency Response

Local Expertise. Emergency response is the responsibility of the operating supervisor at the plant. Only the technical staff at the power plant site has the in-depth knowledge about the plant which is essential for an emergency response. Therefore, the computing facility for emergency response must be operated with plant-specific expertise by the technical staff at the site of the plant.

On-Site Simulation Facility. It is impossible to prepare a remote central simulation facility to an emergency response at a particular plant, because invaluable time would be lost first in loading the respective instructional program and input data set, and then in synchronizing the simulations with the transient conditions in the power plant. Instead, a low-cost dedicated minicomputer must be available at the site of the power plant.

The minicomputer must be loaded with its plant-specific instructional software package and with all plant-specific data other than operating conditions. It must be continuously linked with the plant control, instrumentation and protection systems such that the dynamic simulation can be locked in step with the actual transient in the power plant, until the plant simulator or analyzer is needed for emergency response. The plant analyzer can also be made to ignore the signals from the plant and then serve for a variety of plant transient analyses and for operator training.

It would not be wise to maintain this readiness for several power plants in a single, large computer facility with a supercomputer, because all plants would be left in despair during the failure of a single computer, and remote program maintenance and upgrading for plant-specific program changes are not practical.

Fast Simulation Speeds. While plant performance monitoring can be done at real-time simulation speed, the prediction of plant responses to contemplated remedial actions in emergency situations requires much faster simulation speeds. Simulation speeds of ten times faster than real-time speeds or higher must be achieved without loss in frequency response. High computing speeds are routinely obtained for slow, quasi-static events. They are also needed for rapid transients in severe accidents.

On-Line Link with Reactor. As discussed above, the plant analyzer must be linked with plant control and protection systems via one-way optical data transfer channels and with the control room instrumentation so that the transient in the power plant can be reproduced in the analyzer until a prediction is needed in an emergency.

This requirement should not be confused with the task of establishing the steady-state conditions in the plant from a few operating conditions because a transient starting condition is defined only by the

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

complete set of all state variables for the system. It is therefore necessary to isolate from the entire plant that portion which must be dynamically simulated for an emergency in the nuclear steam supply system. Then one must define the boundary interfaces between the simulated portion and the remainder of the system. These interfaces are invariable control actuator positions for valves, relays, etc., and must be communicated on demand to the plant analyzer.

The communication channels must be scanned during every computational cycle, to assure computational continuity and fidelity. The computational cycling frequency must be approximately an order of magnitude greater than the expected simulation response frequency.

The plant analyzer must offer a large number of computed parameters from which the operator can select any one for convenient graphical display.

Scope of Paper

From all the above requirements imposed on a nuclear power plant analyzer or simulator for emergency response, the requirements of high-speed realistic simulations of severe transients in a low-cost minicomputer are the most difficult ones to achieve. We present first a summary of previously used and currently contemplated approaches to nuclear power plant simulation, then we report on a newly developed technology and present an assessment of its capabilities.

CURRENT APPROACHES TO HIGH SIMULATION SPEED

Real-time simulation speed has been sought by manufacturers of training simulators, by vendors for the design of nuclear power plants and, on behalf of the Nuclear Regulatory Commission, by the developers of codes for safety analyses in national laboratories. Real-time simulation speed can be achieved only by matching modeling and programming techniques with computer capabilities.

Modeling

The mathematical models for training simulators were simplified in the past to achieve real-time simulation speed with available computing power. Rates of change (for fission power, for example [1]) have been artificially limited and coolant inertia has been ignored [1] to maintain computational stability. Such modeling restrictions are now being eliminated as faster minicomputers become available.

The need for increased computing speed has had almost no impact on the modeling in the major systems codes TRAC [2] and RELAP-5 [3], developed by Los Alamos National Laboratory and Idaho National Engineering Laboratory, respectively, for the U.S. Nuclear Regulatory Commission. The models are based on standard finite differencing of partial differential equations for the two-phase coolant mixture. Increases in computing speed have been sought by remodeling the power plant with fewer computational cells. Finite difference methods, however, converge to the correct solution only as the mesh size and the time step decrease toward zero. As expected, any significant reduction of the cell number leads to significant losses in accuracy [4]. Analytical methods have rarely been used in the past to increase simulation efficiently [5].

Numerical Integration

Major efforts have been made to improve the integration algorithms for greater execution speed, particularly in codes which use matrix inversions for solving large systems of nonlinear equations, such as TRAC and RELAP-5. A linearization technique has been used to replace the matrix of rank (MN), representing M field equations in N computational cells, before its inversion by a matrix of rank N for the pressure field. This reduces drastically the computational effort for integrating all field equations. The method fails, however, when partial derivatives of coolant properties vanish or tend toward infinity. With this method one cannot realize the computational savings arising from the behavior of incompressible fluids.

Predictor-corrector methods, traditionally used for integrating ordinary differential equations, have also been used for integrating partial differential equations faster in the *fast* version TRAC-PF1 [2].

PROGRAMMING LANGUAGE

One must always choose between computing efficiency and program transportability. Computer codes written in standard FORTRAN can readily be made to execute on any large computer, yet they can never utilize fully the architecture of a particular computer. FORTRAN compilers have been developed for several computers with parallel and/or pipeline execution, but they do not produce efficient machine codes unless the programmer adheres to very detailed and machine-dependent constraints which render the code machine-specific. The Cray-1 supercomputer executes for a dense linear system [6] from two to six million floating point operations when programmed in standard FORTRAN. Specialized Vector FORTRAN for the Cray machine executes up to six times faster, Vector Assembly Language up to twenty times faster, but only on the Cray machine. One can have either an efficient code, or a transportable code.

Program transportability has always been given a higher priority than execution efficiency by vendors of training simulators, by power plant designers and in government-sponsored code developments for reactor safety analyses. There is widespread apprehension against unfamiliar, non-FORTRAN programming languages in the nuclear industry which is unparalleled in non-nuclear simulation activities.

Computers

There are two trends in the selection of computers for nuclear power plant simulations. Vendors of training simulators seek to employ minicomputers with increasing computing power, accomplished by parallel processing in pipelined computer architectures. In contrast, for almost all major systems codes, developed on behalf of the Nuclear Regulatory Commission, the trend is from large mainframe computers such as the CDC-7600 to even larger super computers such as the Cray computer. Simulations are largely carried out on general-purpose computers with standard FORTRAN programs. However, this paper deals with a power plant simulation on a modern special-purpose peripheral processor which was specifically designed for systems simulation.

Graphic Display Systems. The advantage of multicolor graphics for input and output processing has been widely accepted because it has the potential of

reducing sharply the first and last of the three time-consuming operations in simulation, which are input data preparations, program execution and evaluation of output data.

NEWLY DEVELOPED HIGH-SPEED SIMULATION TECHNOLOGY

The careful selection of modeling and programming techniques and the deliberate choice of a particularly suitable, special-purpose minicomputer produced a new plant analyzer for realistic simulations of normal and severe abnormal transients in nuclear power plants. The most prominent distinctions of the plant analyzer are its great simulation speed, its low capital and operating costs, its outstanding user conveniences and its unsurpassed ability to accommodate control and instrumentation signals in analog or digital form.

The plant analyzer was developed for BWR power plant simulations. The simulation principles employed, however, are general and apply to all power plant simulations.

Modeling Techniques

The most demanding part of light water reactor plant simulation is the modeling of the two-phase flow thermohydraulics in the reactor vessel and, if appropriate, in the steam generators. The coolant dynamics, therefore, dominate the selection of modeling strategy, mathematical methods and computer hardware.

The balance of plant simulation, while taxing computing capacity and input/output processing, is effectively achieved with familiar mathematical models. The simulation of neutron kinetics may require special attention. The plant analyzer described herein employs point kinetics, requiring a relatively small simulation effort. Multidimensional kinetics simulation, however, can also be achieved with standard models and a suitable peripheral processor. Here, the focus of power plant simulation is on the coolant thermohydraulics.

Modeling Principles. Modeling efficiency is achieved when the greatest possible simulation fidelity is produced with the smallest possible number of arithmetic operations. It is achieved by:

(i) selecting the least complicated thermohydraulics model for two-phase flow which accommodates all the available experimental information on two-phase flow,

(ii) eliminating from the models all irrelevant phenomena, while accounting for all possible flow patterns and important processess,

(iii) executing as many integrations as possible in analytical form and evaluating the analytical solutions dynamically during the simulation,

(iv) executing in advance all iterative procedures required for the solution of implicit sets of nonlinear equations, then tabulating the results in terms of explicitly known variables and interpolating the tables during the simulation.

(v) combining analytically in every equation all constitutive relations (material properties, correlations, etc.) into the smallest possible number of expressions and tabulating the expressions for interpolation during the simulation.

The first of these five principles suggests to use only field equations with known mass, momentum and energy interphase and wall to fluid transfer terms that can be modeled and validated with currently available instrumentation [7], since otherwise such field equations only burden the computations, without producing reliable information. By adhering to the fourth and fifth principles one takes advantage of low-cost central core memory now available even in modern minicomputers, and one reduces many computations of any complexity to evaluations of linear expressions. All five principles require some engineering judgement and developmental efforts but, together with proper selections of computing methods and processors, they contribute significantly to efficient high-speed simulation. Below we demonstrate how these principles have been applied in the plant analyzer. For a complete description of the models see Reference [8].

Coolant Hydraulics. For the BWR plant analyzer development described herein, it was recognized [8] that phase separation, coolant mixture level tracking, nonequilibrium boiling, flashing and condensing, and particularly the tight coupling between fission power and vapor void fraction in the reactor core are the most important aspects of coolant dynamics modeling. Acoustical effects in the steam lines are deemed to be important, while acoustical effects in the liquid phase of the coolant mixture are unimportant.

Following the first modeling principle, we selected the four-equation drift flux model which consists of the vapor mass balance and the three balance equations for mixture mass, momentum and energy. Following the second principle and recognizing the irrelevance of acoustical effects in the liquid, we decided to use the volume-averaged vessel pressure*

$$\langle p \rangle = \frac{1}{V_v} \int_{V_v} p dV \qquad (1)$$

to compute all thermophysical coolant properties. The vessel pressure is computed by combining the mixture mass balance, expressed as the volumetric flux divergence equation,

$$\nabla \cdot \vec{j}_m = \frac{\rho_\ell - \rho_g}{\rho_\ell \rho_g} \Gamma_v - \left[\frac{\alpha}{\rho_g} \frac{D_g \rho_g}{D\tau} + \frac{(1-\alpha)}{\rho_\ell} \frac{D_\ell \rho_\ell}{D\tau}\right], \qquad (2)$$

with the caloric equations of state for liquid and vapor, and with the phasic energy balance for the liquid, and by integrating the resulting ordinary differential equation

$$\dot{\langle p \rangle} = \{(Aj)_{fw} - (Aj)_{s\ell} + \sum (Aj)_{inj} + \int_{V_2} \frac{\rho_\ell - \rho_g}{\rho_\ell \rho_g} \Gamma_v dV -$$

$$\frac{1}{\rho_\ell^2} \left(\frac{\partial \rho}{\partial u}\right)_p \left[\int_{A_1} q''_w dA + \int_{V_1} q'''_\ell dV\right]\Big] /$$

$$\left\{\int_{V_2} \alpha \frac{\rho'_g}{\rho_g} dV + \int_{V_1} \frac{1}{\rho_\ell} \left(\frac{\partial \rho_\ell}{\partial p}\right)_u dV\right\} . \qquad (3)$$

Following the third principle, we integrated Equation 2 analytically to replace the partial differential equation of mixture mass conservation by this quadrature in space

*All symbols are defined in the Nomenclature at the end of the paper.

$$j_m(z) = j_m(o) + \frac{\rho_\ell - \rho_g}{\rho_\ell \rho_g} \int_o^z \Gamma_v d\tilde{z} - \frac{\rho'_g}{\rho_g} \langle p \rangle \int_o^z \alpha d\tilde{z}$$

$$- \frac{1}{\rho_\ell^2} \left(\frac{\partial \rho_\ell}{\partial u}\right)_p \left\{ \int_o^z (1-\alpha) q''_w \frac{C}{A} d\tilde{z} - h_{fg} \int_o^z \Gamma_v d\tilde{z} \right.$$

$$\left. + \int_o^z (\alpha - 1) q'''_\ell dz \right\} . \qquad (4)$$

The time-dependent starting value $j_m(o)$ is computed from the momentum equation. Satisfying again the third principle, we integrated analytically the momentum balance

$$\frac{\partial G_m}{\partial \tau} + \frac{\partial}{\partial z} \left[\alpha \rho_g w_g^2 + (1-\alpha) \rho_\ell w_\ell^2 \right] + \frac{\partial p}{\partial z} + g_z \rho_m$$

$$+ f_{\ell o} \phi_{\ell o}^2 \frac{G_m |G_m|}{2 d_h \rho_\ell} = 0 \qquad (5)$$

along every straight stream-tube segment of every closed flow contour in the vessel and recirculation loops, linked the results to eliminate the pressures at segment terminals and thereby replaced Equation 5 in every computational cell by three loop momentum balances

$$\frac{d\mathcal{M}_j}{d\tau} = \Delta p_{JTP} - \sum_{i=1}^{N_{s,j}} \left\{ g_z \langle \rho_m \rangle + \frac{1}{2 d_h} \int_o^L \frac{f_{\ell o} \phi_{\ell o}^2}{\rho_\ell} G_m |G_m| dz \right\}_{ij} +$$

$$\frac{1}{2} \sum_{\substack{i=1 \\ i \neq JTP}}^{N_{s,i}} \left\{ (w_\ell G_\ell + w_g G_g)^+ - (w_\ell G_\ell + w_g G_g)^- \cdot \right.$$

$$\left. \left[1 + \left(\frac{A^-}{A_{min}} \right)^2 \zeta \right] \right\}_{ij} , \quad j = 1, \ldots, 3 \qquad (6)$$

for the vessel and a similar one for the recirculation loop. The loop momentum in Equation 6 is defined by

$$\mathcal{M}_j = \sum_{i=1}^{N_{s,i}} \int_o^{L_{ij}} G_m(z)\, dz, \qquad (7)$$

and the mixture, vapor and liquid mass fluxes are related to the mixture volume flux in Equation 4 by, respectively,

$$\left.\begin{aligned} G_g &= w_g \rho_g = \rho_g \left(f_1 \langle j_m \rangle + f_2 \right) \\ G_\ell &= w_\ell \rho_\ell = \rho_\ell \left[(1-f_1) \langle j_m \rangle - f_2 \right] \\ G_m &= \left[\rho_\ell - f_1 (\rho_\ell - \rho_g) \right] \langle j_m \rangle - f_2 (\rho_\ell - \rho_g), \end{aligned}\right\} \qquad (8)$$

where $$f_1 = C_o \langle \alpha \rangle, \quad f_2 = \langle \alpha \rangle \langle\langle V_{gj} \rangle\rangle . \qquad (9)$$

Notice that Equation 6 accounts fully for gravity effects, wall shear, momentum flux and form losses. It predicts natural circulation but not the unimportant propagation of acoustical waves. Equation 6 not only replaces the expensive task of integrating, for example, as many as fifty-four momentum equations for fifty-four vessel cells by the simpler task of integrating three momentum equations for three loops, covering all fifty-four cells, but it also reduces drastically the computational stiffness of the mathematical model. Only in the steam line does the plant analyzer integrate the momentum balance in finite-difference form for each one of ten computational cells [8].

The use of Equations 3, 4 and 6 reduces the need for integrating the 216 differential equations of the four-equation model for fifty-four computational cells, to the much simpler task of integrating only 111 differential equations. These are the three loop momentum balances (Equations 6), and fifty-four each of the vapor mass balances

$$\left(\frac{dm_g}{d\tau} \right)_j = V_j \langle \Gamma_v \rangle_j + (AG_g)_{j-1} - (AG_g)_j , \qquad (10)$$

$$j = 1, \ldots, 54$$

and of the mixture energy balances

$$V_j \left(\frac{d \langle u_m \rho_m \rangle}{d\tau} \right)_j = \left[A(G_\ell h_\ell + G_g h_g) \right]_{j-1} - \left[A(G_\ell h_\ell + G_g h_g) \right]_j$$

$$+ L_j \left[\langle q'_w \rangle + A \langle (1-\alpha) q'''_\ell \rangle \right]_j ,$$

$$i = 1, \ldots, 54 . \qquad (11)$$

Further details, specifically on constitutive relations for vapor generation and heat transfer, are found in Reference [8].

Conduction in Fuel Elements. Following once again the third modeling principle, we integrated the transient conduction equation for radial conduction of heat, over the cross sections of fuel pellet, gas gap and fuel cladding to obtain an ordinary differential equation [8,9] for the rate of fuel temperature change:

$$\frac{d \langle T \rangle}{d\tau} = \frac{2}{s R_w} \frac{\tilde{k}_{c\ell}}{(\tilde{\rho c})_{f\ell}} \frac{N_{Bi} (T_w - \langle T \rangle)}{1 + N_{Bi} (C_g + F_{pr})} + \left(\frac{R_1}{R_w} \right)^2 \frac{\langle q''' \rangle_f}{(\tilde{\rho c})_{f\ell}} \qquad (12)$$

where $N_{Bi} = h_c s / k_{c\ell}$ is the cladding Biot number, C_g is a constant depending on geometry and F_{pr} is a function of thermal properties [8]. Equation 12 is integrated for every axial fuel element (twelve in each channel) and then used to compute for each axial element the radial temperature distributions. In the fuel pellet the distribution is, with $\zeta = r/R_1$

$$T_{f\ell} - T_w = \left[1 + N_{Bi} \left[\frac{R_w}{R_m} + \frac{R_w}{R_1} \frac{\tilde{k}_{c\ell}}{s} \left(\frac{\delta}{k} \right)_g + \frac{\tilde{k}_{c\ell}}{2 k_f} \frac{R_w}{s} (1 - \zeta^2) \right] \right]$$

$$\cdot (T_w - T_\infty) , \qquad (13)$$

and in the cladding the distribution is, with $n = (r - R_1)/s$

$$T_{c\ell} - T_\infty = \left[1 + N_{Bi} \frac{R_w}{R_m} (1 - n) \right] (T_w - T_\infty) . \qquad (14)$$

In Equations 13 and 14

$$T_w - T_\infty = (\langle T \rangle - T_\infty) / \left[1 + N_{Bi} (C_g + F_{pr}) \right] \qquad (15)$$

is the excess wall temperature above the coolant temperature T_m.

Equation 15 is implicit in T_w because the heat transfer coefficient h_c in the Biot number and the material properties in F_{pr} depend on T_w. Therefore, the calculation of the right-hand side of Equation 12 and of Equations 13 through 15 requires an iterative procedure, involving the selection of the appropriate heat transfer regime and the evaluation of the corresponding heat transfer correlation. Following the fourth modeling principle, the excess wall temperature is computed in advance by Newton-Raphson iteration, over the entire range of possible fuel temperatures and flow conditions. The result is stored in a multi-dimensional table, with known variables as table entries. The evaluation of transcendental expressions is thereby reduced to a time-saving linear interpolation of a nonlinear table. A single table accounts, without loss in accuracy, for all the material properties and heat transfer correlations [8], regardless of their complexity.

Turbine Models. In order to predict the turbine power and the steam exit enthalpy from inlet conditions, turbine speed and exit pressure, one must compute the isentropic enthalpy drop and the thermodynamic turbine efficiency [8]. Following the third modeling principle, we integrate Gibbs' equation

$$Tds_m = dh_m - dp/\rho_m \tag{16}$$

along an isentrope, from the inlet condition $\{(h_m)_i, p_i\}$ to the exit pressure p_e and find

$$\Delta h_{is} = (h_m)_i \left[\frac{T_s(p_e)}{T_s(p_i)} - 1 \right] + T_s(p_e) \left[\int_{p_i}^{p_e} \frac{v_f}{T_s} dp - \int_{T_s(p_i)}^{T_s(p_e)} h_f \frac{dT_s}{T_s^2} \right] . \tag{17}$$

Equation 17 is evaluated by closed-form integration, with known polynomials for the saturation properties. Following the fifth modeling principle, we tabulated the term in square brackets and the last term in Equation 17, each as a two-place function. Thus, the isentropic enthalpy drop in the turbine is computed rigorously by evaluating linear expressions in $(h_m)_i$, p_i and p_e.

Adhering again to the fifth principle, the turbine speed-dependent thermodynamic efficiency is evaluated for a set of nine fixed coefficients of a particular stage, specifying the theoretical degree of reaction, the rotor entrance flow angle, the mean radius ratio, the meridian velocity ratio, the losses in rotor and stator, the reheat factor and the kinetic energy recoveries [8], and as a function of the ratio $\nu = U_1/\sqrt{2\Delta h_{is}}$ of rotor speed over maximum steam velocity. Here, a single linear interpolation produces the result of two square root extractions, the evaluation of a trigonometric function and over fifty additions, subtractions and multiplications.

Feedwater Preheaters. The feedwater temperature rise in the combination of a main heat exchanger and a drain cooler is computed from analytical integrations of the energy balances for the fluids on the shell and tube sides. The result is dynamically evaluated during the simulation, as called for by the third modeling principle:

$$(T_{fw})_e - (T_{fw})_e = [(T_s)_c - (T_{fw})_i] \cdot f_{FWH} , \tag{18}$$

where

$$f_{FWH} = \frac{e^m (u-1)}{ue^m - 1} (1-e^n) , \tag{19}$$

$$m = (AU)_D [1/W_{SHS} - 1/W_{fw}]/c_p ; \; n = (AU)_M/(c_p W_{fw}). \tag{20}$$

Effects of thermal storage and transport times are modeled as a first-order time lag. Individual heater failures are also modeled.

Other Components. The five modeling principles listed above have been employed for the nuclear steam supply system, the balance of plant components, the control systems and the plant protection systems. While ordinary differential equations are used to predict angular speeds in pumps, turbines, electric motors and generators, control functions and valve positions, all nonlinear characteristics for induction motors, pumps and valves are precomputed from first principles or generally applicable empirical correlations and tabulated for linear interpolation in the plant analyzer [8].

Selection of Computing Methods and Computer

The computing method and the special-purpose processor have been chosen for the plant analyzer to achieve the greatest possible simulation speed in a minicomputer. The choice of suitable algorithms for the numerical simulation and of the corresponding processor are, however, intimately related to the form of the mathematical models. As explained above, physics dictated the selection of the four-equation mixture model for coolant dynamics. This four-equation model was then cast in the form of ordinary differential equations (cf Eqs. 3, 6, 12 and particularly Eqs. 10 and 11). This form, and also the fourth and fifth modeling principles established earlier, suggest strongly simulation procedures and computer characteristics, the selection of which is discussed below.

First, we decided whether to integrate explicitly or implicitly, and then we selected a suitable digital computer. A numerical integration procedure is called explicit if the values of all state variables for a future time level are computed explicitly in terms of state variable values only from the past and present time levels. Such a procedure requires no iteration on state variables.* All other numerical integration procedures are called implicit, lead to nonlinear difference equations and require iterations for every time step.

The rational choice between explicit or implicit integration is based upon estimates of these three measures:

(i) the most predominantly encountered high-frequency (f_ν) content of input data and computed parameters,

(ii) the permissible integration step size $\Delta\tau_{int}$ and

(iii) the frame time $\Delta\tau_{frm}$ needed by the computer to advance the simulation from one time level to the next.

*Minor iterations may be necessary to calculate subsidiary variables (cf. Eq. 15).

Frequency Content. Of all the processes driving the system transients, the one with the highest natural frequency f_ν must be accurately simulated and therefore imposes an upper limit $\Delta\tau_\nu = 1/(5f_\nu)$* on the integration step size $\Delta\tau_{int}$. This limit must be imposed over and above applicable limits arising from numerical stability or form truncation errors.

Neutron kinetics has the highest frequency, but it is not a driving mechanism in a BWR power plant. Instead, prompt fission follows the relevant driving mechanism of acoustics in the steam line. In a BWR-4 plant the pressure oscillations reach the frequency of approximately $f_\nu = 10$ Hz.

Permissible Integration Step Size. The highest possible simulation speed is achieved with the largest permissible integration step size

$$\Delta\tau_{int} = \text{Min}\ \{\Delta\tau_a, \Delta\tau_s, \Delta\tau_\nu\}\ , \qquad (21)$$

where $\Delta\tau_a$ is the limit required to maintain accuracy by controlling the truncation errors inherent in all finite difference analogues for derivatives, $\Delta\tau_s$ is the stability limit imposed to avoid exponential growth of round-off errors, and $\Delta\tau_\nu$ is the limit imposed to simulate accurately the high frequency content of driving processes or of boundary conditions. The first two limits characterize the integrating algorithm and depend on its time-dependent eigenvalues. The third limit characterizes the system to be simulated. The first and last limits are related to each other via the same high-order time derivatives of the mathematical model.

For unconditionally stable integrating algorithms $\Delta\tau_s$ is obviously unbounded. Most implicit algorithms have this desirable feature, while explicit integrations are conditionally stable with a bounded $\Delta\tau_s$.

Frame Time. $\Delta\tau_{frm}$ is the clock time that the computer needs to execute all computations for the advancement of the simulation from one time level at τ to the next level at $\tau + \Delta\tau_{int}$. The frame time for implicit integration is always much larger than for explicit integration because of time-consuming iterations.

If the model is formulated, the number of arithmetic operations is established and the frame time can be predicted from given processing speeds. Most computer vendors specify maximum execution speeds (in million floating point operations per seconds, for example) but such specifications are useless because of the uncertainty in the utilization of maximum processing rates. Only Applied Dynamics International of Ann Arbor, Michigan, specifies the time requirement for each operation in sufficient detail for reliable frame time estimates.

Computing Speed. Once the above three measures have been estimated one can easily compute the simulation speed as the multiple S of real-time process speed:

$$S = \Delta\tau_{int}/\Delta\tau_{frm}\ . \qquad (22)$$

*The factor 5 applies here to a third-order Adams-Bashford integration routine and may be greater for lower-order algorithms.

It is clear that if $\Delta\tau_\nu$ is small, explicit integration with its small $\Delta\tau_{frm}$ produces the larger ratio S, while unconditionally stable, implicit algorithms simulate quasi-steady transients faster.

Computing Method for Plant Analyzer. Computing experience with TRAC and RELAP-5 codes lead to an estimate of the frame time $\Delta\tau_{frm}$. An implicit integration of four field equations for approximately one hundred computational cells would require a frame time of 180 milliseconds on a large mainframe, general-purpose compucer (CDC-7600) or 160 to 700 milliseconds on an array processor programmed in FORTRAN. An explicit integration of the same problem requires approximately 30 milliseconds on a general-purpose computer and 6 milliseconds on available array processors designed specifically for explicit integrations.

Since $\Delta\tau_{int}$ in Equation 22 is limited, according to Equation 21, to $\Delta\tau_\nu = 50$ milliseconds for both explicit and implicit integrations, it is clear that explicit integration produces the higher speed-up factor S in Equation 22. Therefore, explicit integration was chosen for the plant analyzer.

Implicit integration is of interest only when $\Delta\tau_\nu$ and $\Delta\tau_a$ in Equation 21 are much larger than 180 milliseconds. To achieve ten times real-time simulation speed (S = 10) with implicit integration, one would have to limit the range of simulated transients to those whose frequency content is below 0.1 Hz, and one would have to use a large mainframe computer.

Computer Selection for Plant Analyzer. Implicit integration involves matrix inversions which are best accomplished in array processors with two or more central processsing units, because the elementary row operations of matrix inversion involve parallel column operations, similar to signal processing. Explicit integration, on the other hand, is best accomplished in a special-purpose peripheral processor which has the inherent characteristics of an analog computer and the integrating capacity and stability of a digital minicomputer.

With the explicit integration method selected, a nationwide search was launched for the most suitable minicomputer to execute high-speed integration of large systems of nonlinear explicit first-order ordinary differential equations. The AD10 of Applied Dynamics International in Ann Arbor, Michigan emerged as the most suitable minicomputer available in 1981. Two AD10 units have been installed at BNL and are operational since March 1982.

Major Characteristics of Plant Analyzer

A detailed description of the AD10 architecture is beyond the scope of this paper but can be found elsewhere [8,10]. In brief, the AD10 is a special-purpose peripheral array processor, designed for high-speed simulation of large complex systems by integration of nonlinear ordinary differential equations. The AD10 is programmed via a host computer, a PDP or VAX minicomputer.

The AD10 contains six distinct, task-specific processors which operate in parallel and are synchronized at the computing cycle frequency of 10 MHz. The six processors serve (i) to link the AD10 with the host computer, (ii) to time and control the other five processors, (iii) to execute logical decisions and binary searches for table interpolation (iv) to execute additions, subtractions and multiplications in integer or

fractional arithmetic, (v) to carry out numerical integrations, and (vi) to address memory. Two additions and one multiplication can be carried out in one cycle, resulting in thirty million fractional operations per second. Twenty million words can be transferred each second between memory and processors. Internally generated digital data can be issued as digital data at the rate of 3 million words per second or converted to analog signals in the range from -10 volt to +10 volt. Input signals can be accepted as digital (3 million words per second) or analog signals (± 10 volt).

Capabilities. The two AD10 processors installed at BNL can integrate as many as 1,950 state equations, with any combination of seventeen built-in algorithms, such as first through fourth orders Adams-Bashford or Adams-Moulton and second through fourth orders Runge-Kutta procedures. The mix of algorithms can be altered with a single keyboard command, on-line and without reloading the program.

The two processors can generate as many as 18 nonlinear functions of one variable, plus 34 functions of two variables, plus 12 functions of three variables in as little as 98 microseconds, regardless of the functions complexity. This feature is utilized by Modeling Principles iv and v.

Up to 256 input and output analog channels can be scanned for every computing frame. This makes the outside world (instrumentation and controls in a power plant) part of central core memory.

The BWR plant simulation presented here entails 320 integrations with 4,000 subroutine or module calls, including the interpolation of over 200 nonlinear multidimensional tables, many as often as 54 times during every computing frame. All of this is achieved in the frame time $\Delta\tau_{frm} = 5.4$ ms. The maximum integration step size is $\Delta\tau_{int} = 54$ ms, producing a ten times greater than real-time simulation speed (cf. Eq. 22). Twenty-eight analog channels are scanned 200 times per second to introduce operator actions and malfunctions any time before or during the simulation. Sixteen output channels are currently updated 200 times per second for graphic display and storage of computed results. This capability is indispensible for computer-aided emergency response.

Limitations. The AD10 processors installed at BNL* employ 16-bit integer arithmetic in all but the integrating processors (which have a 48-bit mantissa in pseudo floating point representation). The 16-bit integer arithmetic not only requires scaling of all variables, but also limits the dynamic range to two decades if the relative error is kept below 1/2%. All variables are scaled to fall in the range [-1, +1]. The least significant bit is $2^{-15} = 3 \times 10^{-5}$.

Scaling means additional analytical work during program implementation. One is rewarded, however, from this scaling work by gaining in-depth understanding of the mathematical models and by obtaining a solid basis for program validation.

*A new floating-point processor FX is now available with greater computing speed and capacity, at the same cost.

We have employed dynamic scaling for one parameter only, to expand its dynamic range from two to five decades. Assume that in the scaled equation

$$\bar{y} = C\ \bar{f}(\bar{x}) \quad , -1 \le \bar{x} \le +1 \qquad (23)$$
$$0 \le |C| \le 512, \text{ constant}$$

$|\bar{f}|$ reaches the noise level for $\bar{x} \in [a,b]$. Then

$$\bar{y} = C\cdot\bar{f}(\bar{x})\cdot\bar{p}(\bar{x}) + \frac{C}{1000}\,[1000\cdot\bar{f}(\bar{x})]\,[1-\bar{p}(x)],$$

$$\text{where} \quad \bar{p}(\bar{x}) = 1 \text{ for } \bar{x} \in [-1,a) \text{ and } \bar{x} \in (b,1] \qquad (24)$$

$$\bar{p}(\bar{x}) = 0 \text{ for } \bar{x} \in [a,b],$$

has a dynamic range which is three decades larger than that of Equation 23.

Programming Language. The AD10 is not a general-purpose computer, programmable in general-purpose FORTRAN. The AD10's own high-level systems simulation language MPS-10 consists of subroutine or module calls which reflect analog computer operations. The major advantage of MPS-10 is that it contains a single rule with which to utilize fully the parallel and pipelined processing in the AD10. Module calls need only to be packaged in groups, called Data Areas, until such groups are filled up. Then all overhead operations are minimized, pipe lines are fully utilized and the maximum simulation speed is achieved, unless one cares to program in assembly language. MPS-10 is relatively easy to learn.

Graphic Display. Currently we transmit any set of fifteen parameters and time to an expanded IBM Personal Computer (PC) for storage and subsequent display in labelled diagrams on a four color CRT monitor. Any two of the fifteen parameters can be selected for on-line display during the simulation. A dot matrix printer produces black and white hard copies of the monitor image.

Any variable can be displayed also on a Type 4012 Tektronix storage oscilloscope with thermal printer. The oscilloscope is the primary display device for program development and interactive system analysis.

ACHIEVEMENTS OF HIGH-SPEED SIMULATION

Scope

Figure 1 shows schematically the BWR-4 plant configuration simulated for this paper. Shown are the reactor, the balance-of-plant components and the control system, i.e., the pressure regulator (P), the feedwater regulator (FW) and the recirculation flow controller (FW). The arrangement of computational cells in the vessel is shown in Figure 2.

The simulation encompasses neutron kinetics, thermal conduction in fuel structures, nonhomogeneous, nonequilibrium coolant dynamics with level tracking, boron transport, the dynamics of steam line flow, turbines, condensers, feedwater preheaters, turbine-driven feedwater pumps, pump speed-controlled recirculation flows, suppression pool, plant protection systems and the control system described above [8].

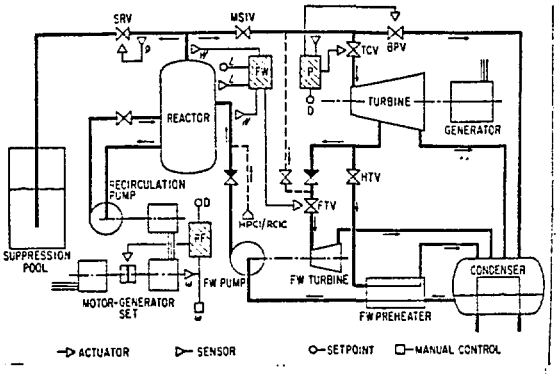

Figure 1 BWR Plant Schematic

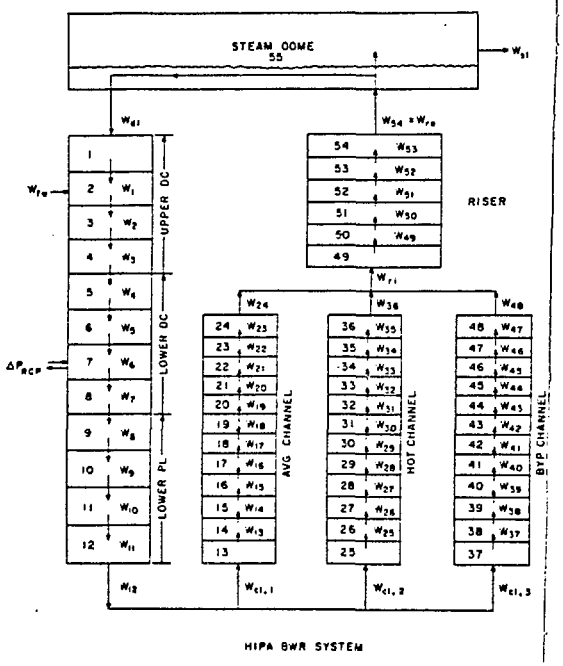

Figure 2 Vessel Flow Schematic and Arrangement of Computational Cells in Pressure Vessel

Thirty-seven different transients have been simulated as part of the developmental assessment for the plant analyzer, including sixteen multiple failure events. Geometric parameters, operating conditions and control parameters (delay times, amplifier gains etc.) were entered through keyboard commands. Failures were entered from a control panel shown in Figure 3.

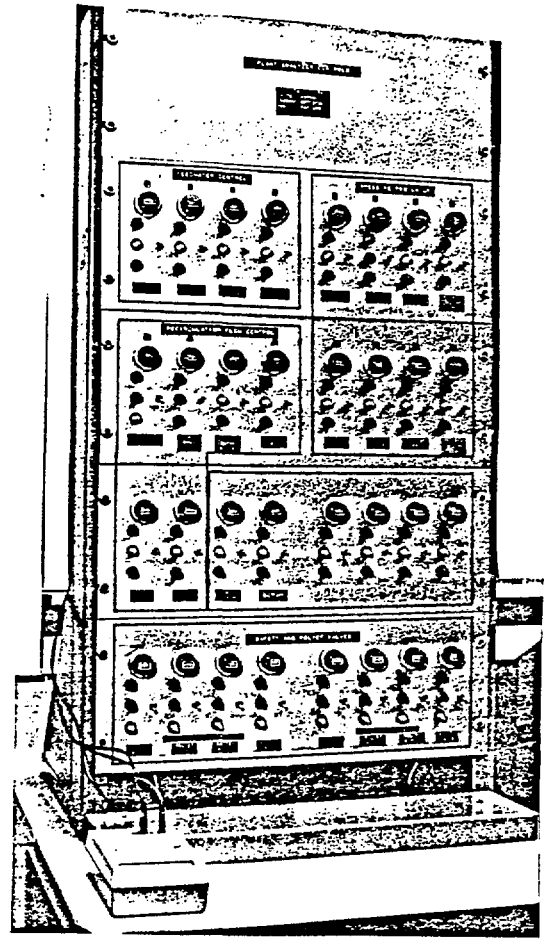

Figure 3 Control Panel

Simulation Speed

All transients can be simulated at speeds up to ten times faster than real process speed. It is easy to simulate as many as ten different transients in less than two hours and display the results on the IBM PC.

This simulation capability results from the plant analyzer's outstanding convenience for changing input data, from its ability to respond instantly to input changes, from its high simulation speed and from its graphic display capability which still can be further improved.

Simulation Accuracy

Good simulation accuracy can be achieved only with complete and exact plant-specific input data for geometric parameters, set point data for the plant protection system, control parameters and characteristics for pumps, valves, etc. A preliminary assessment of the plant analyzer has been performed with most input

data as available for the Peach Bottom II BWR power plant. Unspecified data have been inferred from normal steady-state operating conditions and from data published in the Final Safety Analysis Report for Peach Bottom II.

Plant analyzer results have been compared [8] with results from CDC-7600 calculations using the same original equations, from the Final Safety Analysis Report, from GE calculations for ten different Anticipated Transients Without Scram in a generic BWR-4 plant, with results from TRAC-BD1, RELAP-5 and RAMONA-3B. The comparisons show relatively good agreement even though the transients were simulated with slightly different input data. Figures 4 through 10 show typical results of the comparisons between plant analyzer results and results from major systems codes. Figure 4 shows two pressure curves from TRAC-BD1 because the authors [11] observed an error in TRAC-BD1's kinetics calculations. Other differences are explained in more detail in reference [8].

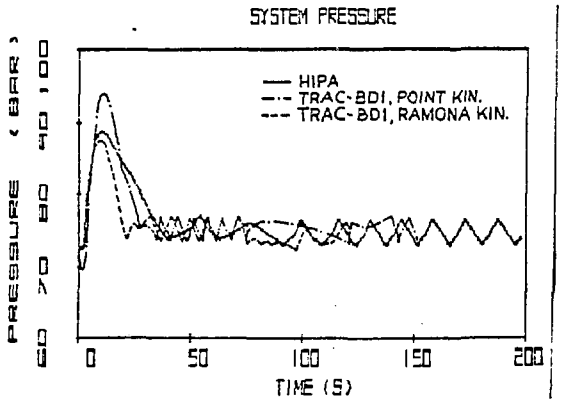

Figure 4 Comparison of System Pressure Predictions by Plant Analyzer and TRAC-BD1 Code

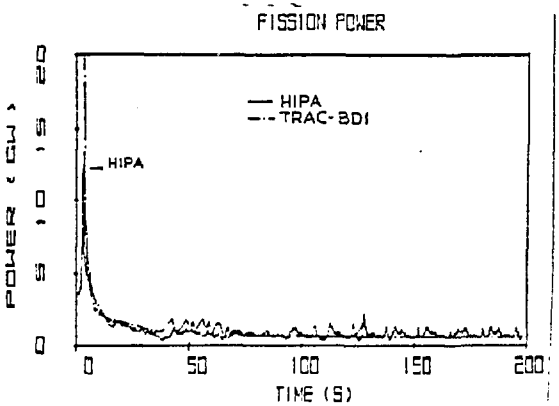

Figure 5 Comparison of System Pressure Predictions by Plant Analyzer and TRAC-BD1 Code

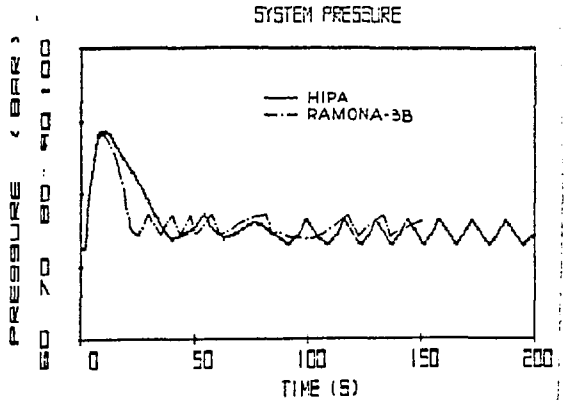

Figure 6 Comparison of Vessel Pressure Predictions from Plant Analyzer and RAMONA-3B Code

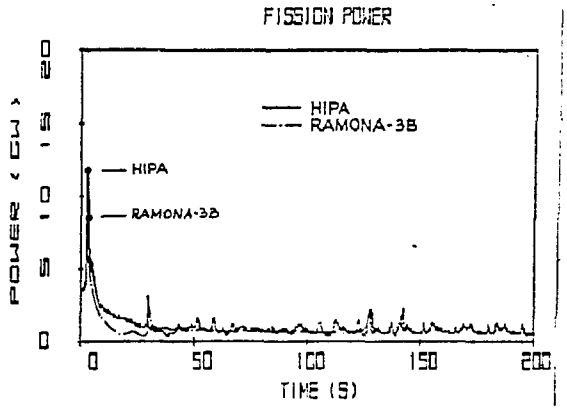

Figure 7 Comparison of Fission Power Predictions from Plant Analyzer and RAMONA-3B Code

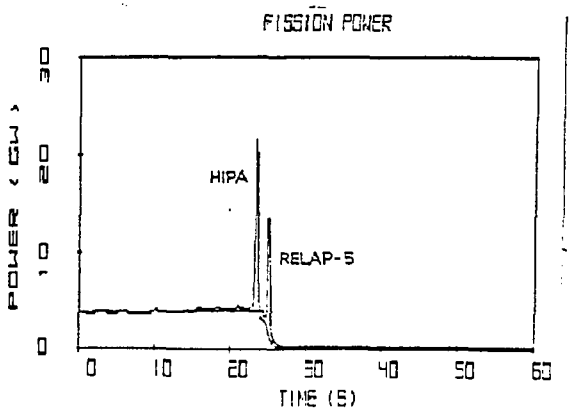

Figure 8 Comparison of Plant Analyzer with RELAP-5 Calculations for Failure of Feedwater Controller at Maximum Demand, Fission Power

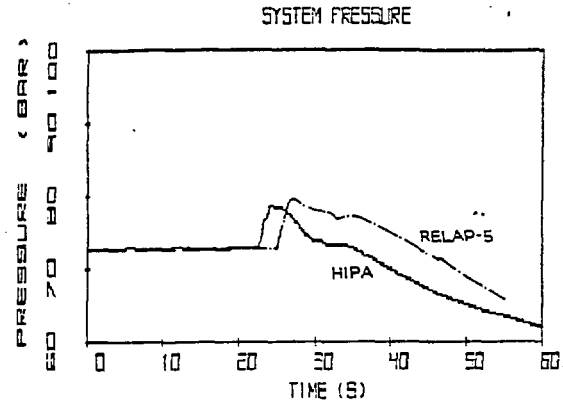

Figure 9. Comparison of Plant Analyzer Results with RELAP-5 Calculations for Failure of Feedwater Controller at Maximum Demand, System Pressure

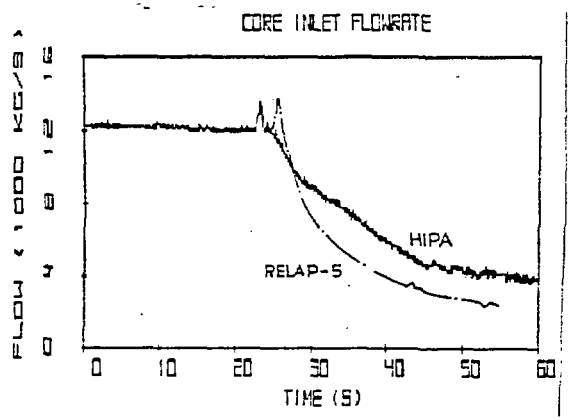

Figure 10 Comparison of Plant Analyzer Results with RELAP-5 Calculations for Failure of Feedwater Controller at Maximum Demand, Core Inlet Mixture Mass Flow Rate

CONCLUSIONS

BNL Plant Analyzer

It has been demonstrated that a combination of advanced modeling techniques and modern special-purpose array processor technology can yield a nuclear power plant simulation capability which is superior to that obtained on large mainframe computers, with standard FORTRAN systems codes.

Realistic and accurate simulations can be achieved with the plant analyzer at much faster than real-time simulation speeds, at low cost and with unsurpassed user convenience, for both normal and severe abnormal events.

The plant analyzer can accommodate hardware (controls and instruments) on-line into its computational loop and make the outside world part of its central core memory.

Computer-Aided Emergency Response

The new technology presented here produces the necessary simulation speed for plant monitoring, for failure diagnosis and for on-line predictions of the plant response to operator actions which might be contemplated for the mitigation of the consequences from reactor accidents.

A dedicated facility at the plant site, programmed for, and synchronized with, the specific power plant is required, in principle, for timely and effective emergency response. New research is required to accomplish the synchronization.

NOMENCLATURE

A	cross-sectional area
A_{min}	$\mathrm{Min}\{A^+,A^-\}$
c	specific heat
C	heated perimeter
C_0	Zuber-Findley distribution parameter
C_g	geometric parameter
d_h	hydraulic diameter
$D_k/D\tau$	substantial derivative for phase k, k=g,ℓ
f_{FWH}	Eq. 19
f_1, f_2	Eq. 9
$f_{\ell o}$	single-phase friction factor
F_{pr}	property parameter
g_z	component of gravity in z-direction
G	mass flow rate
h	enthalpy
h_c	convective heat transfer coefficient
j	volumetric flux
k	thermal conductivity
L	length
m	Eq. 20
m_g	vapor mass
M_j	Eq. 7
n	Eq. 20
$N_{s,i}$	number of segments in j-th flow contour
p	pressure
q'_w	linear heating rate
q''_w	wall heat flux
q'''_γ	gamma heat absorption rate
r	radius
R_1	fuel pellet radius
R_m	mean cladding radius
R_w	outer cladding radius
s	cladding thickness
s_m	mixture entropy
T	temperature
u	internal energy
U	overall heat transfer coefficient
U_1	rotor entrance speed
v	specific volume
$\langle\langle V_{gj}\rangle\rangle$	void fraction-weighted, area-averaged drift velocity
V_v	vessel volume, V_1+V_2
V_1	volume of pure liquid in vessel
V_2	V_v-V_1
w	axial velocity component
W	mass flow rate
z	axial coordinate
α	void fraction
Γ_v	vapor generation rate per unit of volume
ζ	form loss coefficient
μ	W_{fw}/W_{SHS}
ν	$U_1/\sqrt{2gh_{ts}}$
ρ	density
τ	time
$\phi^2_{\ell o}$	two-phase multiplier for wall shear

Subscripts

c	condenser
cl	cladding
D	drain cooler
e	exit
f	saturated liquid
fl	fuel
fw	feedwater
g	saturated vapor
i	inlet
ij	loop and segment indices
inj	injection
is	isentropic
j	loop index
JTP	jet pump
l	saturated liquid
m	two-phase mixture
s	saturated
sl	steam line
SHS	shell side of heat exchanger
v	vapor
∞	coolant (subcooled, saturated or superheated)

Special Symbols

<>	average
·	time derivative
'	derivative with respect to pressure, along saturation line
+ -	up and downstream of expansion or contraction

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