SIEMENS' INTEGRATED CODE SYSTEM CASCADE-3D FOR CORE DESIGN AND SAFETY ANALYSIS

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ABSTRACT

This work presents the development, verification and application of the new program system CASCADE-3D which links Siemens' advanced code packages for in-core fuel management, core monitoring, and accident analysis. Direct coupling of these large code systems reduces the effort required to transfer information as well as eliminates the chance for errors to occur. The in-core fuel management code uses state-of-the-art technology including discontinuity factors, nodal expansion and pin power reconstruction. The core monitoring system utilizes the same core physics code and reports monitored data based upon the reactor measurements. The 3D transient code uses the same core physics models as the fuel management system incorporating the same highly optimized flux solver. Each of these systems has been extensively benchmarked against plant operating data as well as experimental results. Some of these comparisons are presented here as representative examples. By using the capabilities of CASCADE-3D the potential of modern fuel assemblies and in-core fuel management strategies can be much better utilized because safety limits which had been reduced due to conservative methods are now predicted more accurately. Due to the enhanced integration of the code packages used in core design and safety analysis a new extended code system is available for consistent analyses of the reactor core and the entire nuclear power plant.

1. INTRODUCTION

Advances in the fields of computer hardware and software technology - manifested most strikingly by ever-increasing calculation speeds and improvements of the user interfaces - now make it possible for large code systems to be directly coupled to each other. This means that previously separate areas of analysis - such as reactor physics, thermal hydraulics and systems dynamics - can now be integrated with the aim of considerably increasing simulation accuracy by eliminating conservative assumptions at code interfaces.

The new program system CASCADE-3D (**C**ore **A**nalysis & **S**afety **C**odes for **A**dvanced **D**esign **E**valuation) links some of Siemens advanced code packages for core design, core monitoring and safety analysis: SAV, POWERTRAX^{TM}, PANBOX, COBRA and RELAP5. Consequently by using CASCADE-3D, the potential of modern fuel assemblies and in-core fuel management strategies can be much better utilized because safety limits, which had been reduced due to conservative methods, are now predicted more accurately. In developing CASCADE-3D, the extension of the application range and the improvement of the prediction accuracy were of primary importance. Some examples are:

- analysis of all types of PWRs
- description of MOX fuel assemblies with increased plutonium content
- improvement of the user interfaces
- reload pattern optimization based on evolutionary algorithms supplemented by heuristics
- consistency between in-core fuel management and accident analysis
- simulation of complex accident scenarios

In this paper, we give an overview of the whole system (Figure 1), with emphasis on the new features and components and illustrate the capabilities of CASCADE-3D with typical results of steady-state and transient validation and application calculations.

2. IN-CORE FUEL MANAGEMENT WITH SAV

The code system SAV, an abbreviation based on the German for "standard design procedure", includes all the Siemens codes necessary for full scope in-core fuel management. From the very start, development of this code system has been based on the principle of ensuring that the mathematical and physical models contained in the codes reflect the most recent state of the art in science and technology. This has enabled major breakthroughs to be made in terms of accuracy and efficiency; e.g. through the development of advanced nodal techniques for solving the transport/diffusion equations, techniques which are now in widespread use throughout the world.

2.1 Main Components of SAV

Modern fuel assembly and core neutron-physics design require highly flexible, robust and reliable tools. The main components of SAV are the spectral code monitoring system FOXS, the reactor code PRISM and the post-processing code PINPOW (s. Fig. 1). These components interact in a robust and efficient way thanks to the automated information transfer between the codes.

FOXS – with the standard lattice code CASMO as central module – is a versatile tool for neutronic fuel assembly (FA) design. The utilized basic cross section library is well suited for the description of present and future uranium and MOX fuel assemblies. Main tasks of FOXS are providing data for FA and reflectors to the reactor codes PRISM and PANBOX, as well as describing repaired or reconstituted FAs. The provided data are essentially microscopic cross sections, assembly discontinuity factors and heterogeneous form functions; they result from multi-group 2-dimensional transport calculations.

PRISM, the reactor code, is based on a fast running steady-state flux solver. This state-of-the-art module combines the accuracy of advanced nodal methods with the efficiency gained by the application of multi-level techniques and the use of vectorization capabilities. It handles intra-nodal spatial variations of cross sections and uses discontinuity factors, which enables a more accurate treatment of asymmetric FAs in one node per assembly representations.

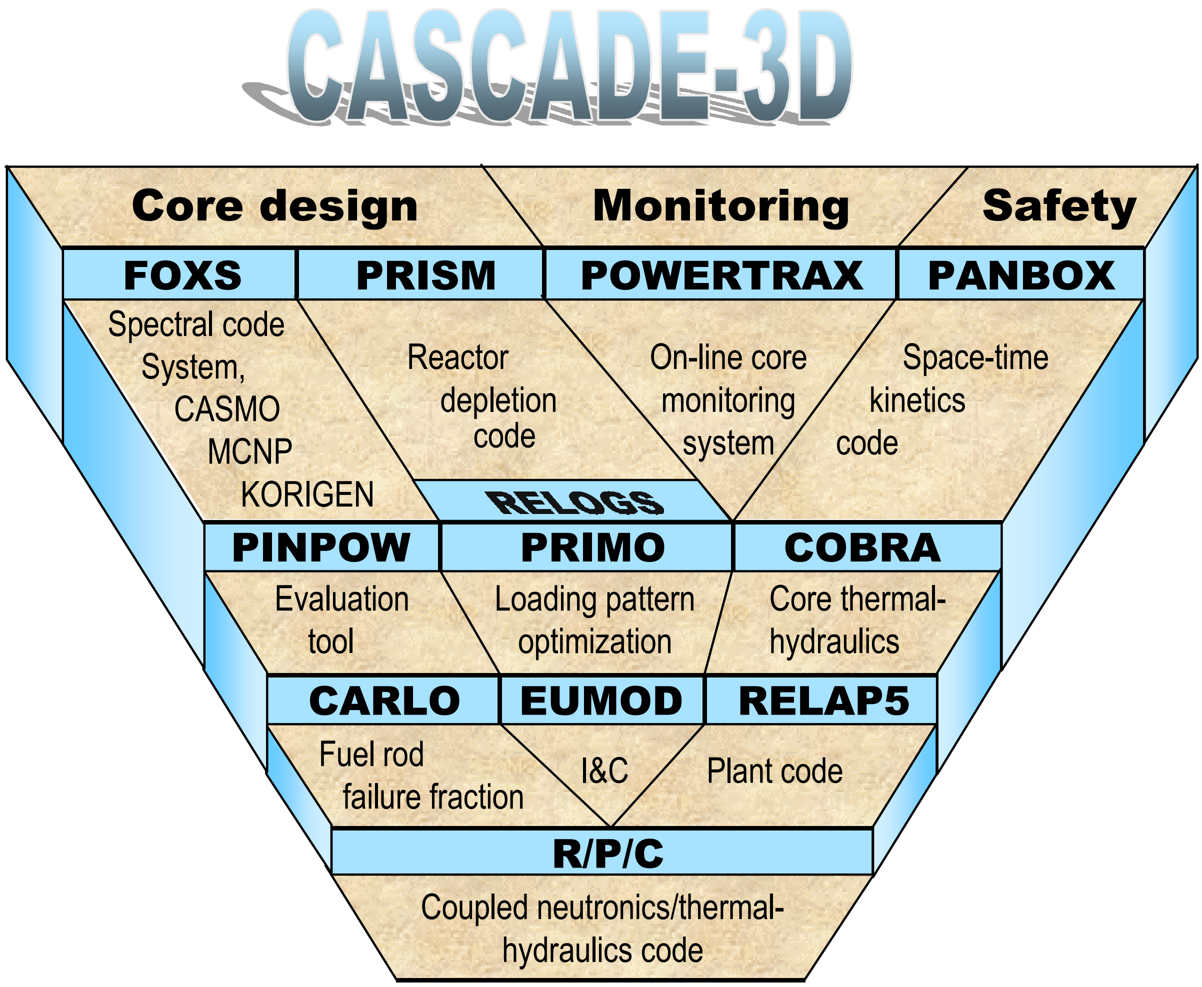

Figure 1. Integrated Code System CASCADE-3D for Advanced Core Design and Safety Analysis

Another important feature of PRISM is the continuous representation of the *microscopic* cross sections, which covers all possible combinations of thermal-hydraulic parameters in stationary reactor states. The standard nuclide chain consists of 16 heavy fuel nuclides, 3 burnable absorber nuclides and four explicit and one lumped fission product nuclide.
For an accurate and efficient determination of pinwise power, exposure, fluxes and detector signals, the full 3-D interpolation and modulation scheme is directly integrated into the core simulator.

Based on the 3-D pinwise solution provided by PRISM, the evaluation code PINPOW determines key safety parameters, such as local departure from nucleate boiling ratio (DNBR) and pinwise waterside clad corrosion data. Additional central capabilities of PINPOW are the generation of process computer data for various reactor types and the support of automated report generation.

The consistency between in-core fuel management and accident analysis is achieved on the one hand by using the same steady-state flux solver and pin reconstruction modules in both reactor codes PRISM and PANBOX. On the other hand, both codes are directly provided by the same input information in terms of cross section libraries, fuel assembly form functions as well as reactor data, geometry, and fuel particle number densities.

2.2 Validation and Application of SAV

The validation of SAV is based essentially on the comparison of calculations with measured data coming from critical experiments, and with startup physics and core follow measurements from commercial reactors. Based on more than 300 reactor cycles for Siemens, Westinghouse, Combustion Engineering and Framatome PWRs with power levels ranging from about 350 to 1450 MW, Siemens has accumulated a comprehensive experience in reload core design[1,2]. Benchmarking calculations complete the scope of validation. The validation of the system also includes many comparisons of calculated and measured pinwise power distributions from critical experiments (Babcock & Willcox, KRITZ facility and Plutonium Recycle Critical Facility). The extended data base for SAV leads to a corresponding wide range of application. Table 1 illustrates the prediction capability of SAV for a typical Westinghouse reactor operated in long cycles for which the FA inventory is characterized by high initial U-235 enrichments – i.e. up to 4.95 w/o U-235 – and high numbers of Gd-poisoned fuel rods – i.e. up to 24 rods per FA with up to 8 w/o Gd_2O_3 concentration. Calculation and measurement results are compared for typical startup physics test data.

Table I: C/M Comparison for a Westinghouse Reactor

Parameter	Measurement	Difference	Difference %
HZP critical boron concentration [ppm]	2206	8	N/A
ARO HZP moderator temperature coefficient [pcm/°F]	-1.66	.32	N/A
Bank B worth [pcm]	1109	-26	-2.3
Bank D worth [pcm]	1074	5	0.5
Bank C worth [pcm]	948	23	2.4
Bank A worth [pcm]	244	22	9.0
Bank SC worth [pcm]	299	11	3.7
Bank SB worth [pcm]	996	-22	2.2
Bank SA worth [pcm]	842	8	1.0

Another example of the prediction capability of SAV is given for a typical Siemens reactor in the frame of a recent measurement campaign at end-of-cycle. Table 2, shows the differences between measurement and calculation results for boron equivalents of control rod (CR) worths.

Table II: C/M Comparison for a Siemens reactor

Parameter	Moderator temperature [°C]	Meas.-Calc. [ppm B_{nat}]
CR worth for Nettobank *)	53	+6
CR worth for Nettobank *)	123	+27
CR worth for DI-bank sequence	53	+3
*) Nettobank=ARI minus CR-Quartet		

A Combustion Engineering plant with cruciform control blades has also been validated against startup physics test data. Table 3 presents the comparisons for critical boron and control bank worths.

Table III: C/M Comparison for a Combustion Engineering Reactor

Parameter	Measurement	Difference	Difference %
ARO HZP critical boron concentration [ppm]	1801	-7	N/A
ARO HZP isothermal temperature coefficient [pcm/°F]	-0.660	-0.48	N/A
ARO HZP moderator temperature coefficient [pcm/°F]	0.840	-0.48	N/A
Differential Boron Worth [pcm]	7.2	-0.18	2.5
Reference Bank in Boron end point [ppm]	1602	-4	N/A
Bank 1 worth [pcm]	688	-33	-4.8
Bank 2 worth [pcm]	600	-33	-5.5
Bank 3 worth [pcm]	613	-39	-6.4
Bank 4 worth [pcm]	564	30	5.3
Bank A worth [pcm]	703	35	5.0
Bank B worth [pcm]	1433	46	3.2
Total Bank Worth [pcm]	4601	6	0.1

A unique challenge was provided for a CE reactor that was shutdown in mid-cycle for about three and one half years. A special extended fission product model was developed for describing the isotopic decay during shutdown. Figure 2 presents the boron letdown curve for this reactor indicating excellent agreement both before and after the extended shutdown.

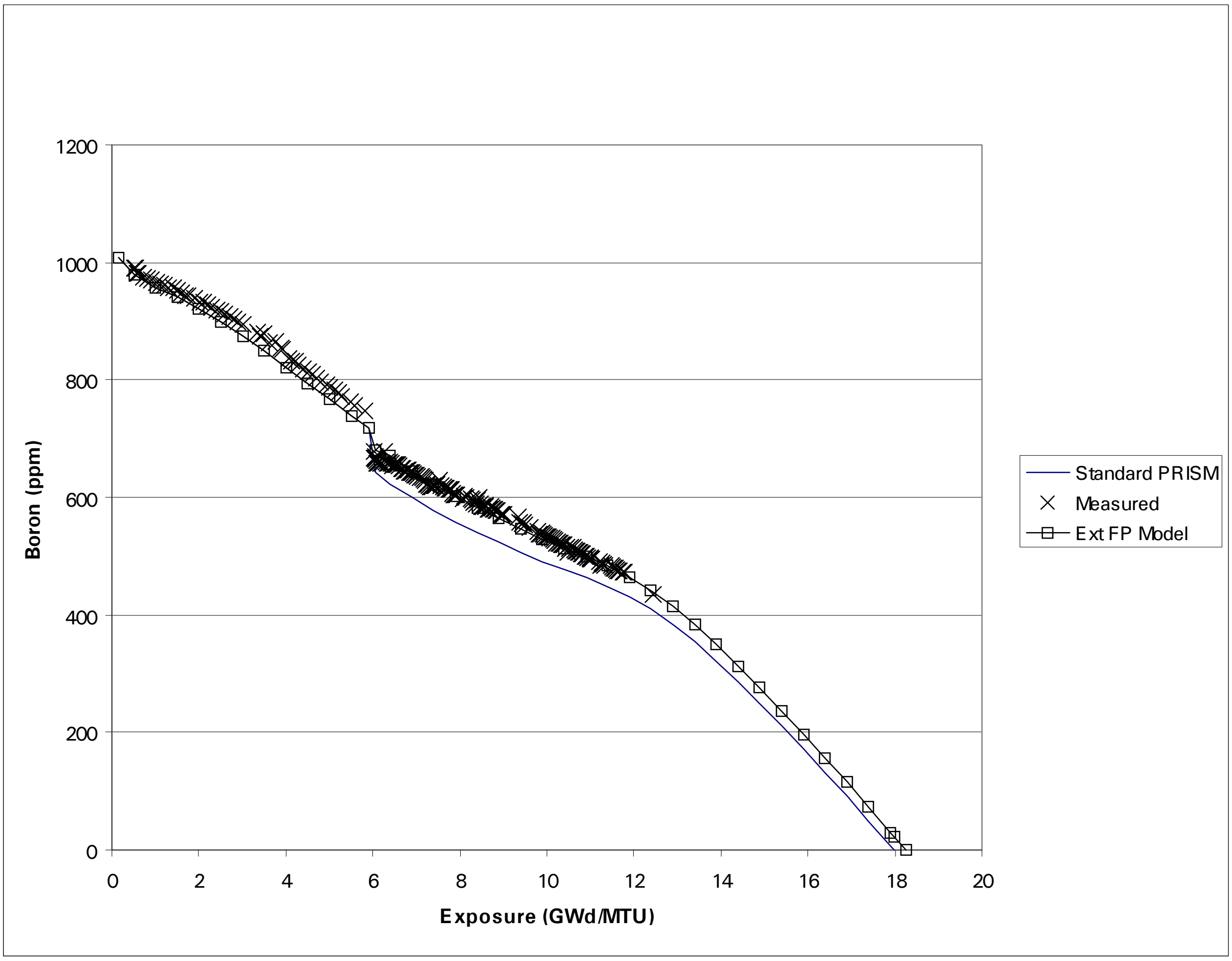

Figure 2. Critical Boron Concentration for Operation
Before and After Extended Shutdown

The extended data base for the SAV validation leads to the following very wide range of qualified application:

<u>Fuel designs:</u>

- UO_2 fuel without and with burnable absorbers (Gd or B) of Siemens, Framatome, Combustion Engineering and Westinghouse designs
- MOX fuel of Siemens, Framatome, and Westinghouse designs

<u>Core designs:</u>

- Siemens, Framatome, CE and Westinghouse core designs with core size ranging from 121 FA up to 217 FA and FA lattices ranging from 14x14 to 18x18 design
- Initial cores with boron glass absorber assemblies
- Control elements in the form of spiders as well as cruciform

Core loading strategies:

- out-in, part low leakage and full low leakage loading patterns (with and without Gd)
- 6 to 20 months cycle lengths
- UO_2 FA: U235 enrichments ranging up to 5.0 w/o U235
- MOX FA: Pu-fiss contents ranging up to 5.0 w/o Pu-fiss
- UO_2/Gd FA: Gd concentrations up to 10 w/o Gd_2O_3
- Average fuel assembly burnups ranging up to 60 MWd/kg HM

3. PRIMO: EVOLUTIONARY ALGORITHMS TO OPTIMIZE LOADING PATTERNS

PRIMO – a new component of CASCADE-3D – is a powerful tool for automated loading pattern (LP) optimization[3]. This new code is based on Evolutionary Algorithms (EA) which have proved to be robust optimization methods for complex engineering tasks. Rather than searching from one solution to the next one, EA search from one *generation* of trial solutions to another one, according to the well-known principle of the biological evolution – *survival of the fittest*. By means of an appropriate coding of the decision variables, an LP on the engineering side corresponds to a *chromosome* which defines an *individual* on the biology side. The basic mechanisms of *mutation* and *selection* can then be translated from biology to engineering to perform FA shuffling operations towards an optimized LP.

Loading patterns of good qualities correspond to individuals with a high fitness, a crucial issue in the selection process. It is essential to have an appropriate mathematical representation of the quality, according to the various optimization criteria and their associated constraints given by the reload designer. A suitable definition of the quality, i.e. of the fitness, leads to the convergence of the process towards an optimum. In PRIMO the optimization goals and constraints are gathered in an *objective function*. Criteria like BOC boron concentration, cycle length, number of fresh FA and local power peaking factors can be combined in a weighted form. Unacceptable LPs in terms of violated constraints are not merely rejected but assigned with *penalty terms* added to the objective function.

The objective function parameters are calculated on the basis of *standard* core simulations with the design code PRISM which is coupled to PRIMO in a modular way. This is a key feature of the system with respect to the acceptance of the optimized solution. It represents a decisive advantage against optimization methods using simplified models for the core cycle simulations.

The PRISM calculations represent the most time consuming part of the whole optimization process. Therefore we look on the one hand at the core simulator and on the other hand at the optimization module to speed up the convergence towards an optimized solution. The needed CPU time for the core simulations can be reduced by

(1) performing BOC calculations only at the beginning and switching automatically to full cycle calculations once it is worthwhile in terms of LP quality,
(2) using a less tight convergence criterion for the flux iterations so long as fine optimization is not required,

(3) performing the overhead operations once and for all and repeating only the LP-dependent flux iterations.

Additionally, EA are particularly suitable for coarse grain parallel computing on workstation clusters and/or multiprocessor systems. To increase the search speed of the EA, reload expert knowledge in form of heuristic rules has been implemented into the code, too. A key feature of the system is the use of variation probabilities in order to find the right balance between pure EA and heuristics. Furthermore we are combining synchronous and asynchronous optimization modules to better escape from local optima and consequently increase the robustness of the optimization process. All these speed-up measures lead to turn around computing times of a few hours on a cluster of 4 Intel Pentium III 600 MHz processors.

Last but not least PRIMO is fully integrated in RELOGS, the standard SAV graphical user interface for reload optimization support. The required additional input preparation effort is negligible so that PRIMO can be routinely started and comfortably combined with human optimization.

4. POWERTRAX™: ON-LINE CORE MONITORING SYSTEM FOR PWR CORES

The CASCADE-3D code system links the Siemens on-line core monitoring system POWERTRAX™ and the 3-D steady-state core simulator PRISM. This yields a consistent methodology between nuclear core design and online core monitoring.

The POWERTRAX™ software package[4] is a comprehensive advanced on-line power distribution monitoring and operations support system for PWRs. The system uses on-line plant computer data as well as measured reaction rates obtained with the in-core detector system to update the 3-D neutronic core model PRISM which is used to perform the monitoring calculations and the operation support calculations.

POWERTRAX™ uses a graphical menu driven interface for performing the many calculations necessary to support reactor operations. Examples of these support calculations include processing flux trace files, performing the ex-core to in-core axial flux difference (AFD) calibration calculations, determining shutdown boron concentrations, determining start-up estimated critical conditions, developing operating strategies for maneuvering through anticipated power changes and displaying precalculated start-up and operational data.

Included within POWERTRAX™ is an Operating Strategy Generator (OSG) to model the reactor behavior through some future anticipated time intervals. The OSG can be run with the optimum strategy algorithm activated, in which case it attempts to determine the best way to operate the plant through the planned maneuver. The “best operating strategy” is defined as the one that accomplishes the planned maneuver with the maximum capacity factor, minimum coolant treatment and within applicable operating limits. The OSG can also be run with the optimum strategy algorithm turned off. In this case the OSG simply runs through a set of calculations that define the planned maneuver. This option can be used to provide an initial starting point for other POWERTRAX™ predictive functions, such as the estimated critical conditions and the shutdown boron concentration calculations.

5. DETERMINATION OF THE LOCA FUEL FAILURE FRACTION

Another decisive improvement has been achieved by applying statistical methods to fuel rod design calculations. This approach allows, for example, the determination of the LOCA fuel failure fraction with a reasonable degree of conservatism by explicitly taking into account the individual power histories of all fuel rods in the core. For this purpose the fuel rod design code CARO was coupled to our LOCA code BETHY, leading to the coupled system CARLO. The core damage extent, especially for PWRs in Germany, thus can directly be calculated following the fuel management calculations for a given loading pattern.

A key feature of this new methodology is that the individual power histories for every fuel rod in the core are taken into account, leading to a higher prediction accuracy. This has become possible thanks to the integration of the CARLO code into CASCADE-3D.

6. ACCIDENT ANALYSIS WITH RELAP5/PANBOX

The linking of the best-estimate plant simulation code RELAP5 and of the three-dimensional core transient program PANBOX – R/P/C in short – enables the simulation and analysis of complex postulated accident sequences (e.g., rod ejection, steam system piping failure, ATWS, boron dilution transients, etc.), which are characterized by a significant change in the power distribution in the reactor core, and by interaction between core and plant.

6.1 General Features of the Coupling

The coupling of PANBOX to RELAP5 has been performed via the general Siemens RELAP5 interface package EUMOD in an explicit way. This means that PANBOX is called once at the end of every RELAP5 time step, without iterating between the solutions of both codes. For all problems analyzed so far, this coupling procedure has proven to be stable. External code modules (e.g. I&C) may also be linked through EUMOD to R/P/C. Core thermal-hydraulics can be calculated using RELAP5 (internal integration) or COBRA, the thermal-hydraulic module of the standalone nodal space-time kinetics code PANBOX (external integration, with a one-channel-per-fuel-assembly grid).

R/P/C has the capabilities of RELAP5 with added ability to calculate space-time kinetics with PANBOX and thermal margins with its COBRA module (with an embedded subchannel grid). Plant transients and postulated accident scenarios may now be calculated realistically using a 3-D neutron-kinetics model. In order to save computing time, one-dimensional or point kinetics models may also be activated in an adaptive algorithm, using parameters generated automatically from the 3-D solution and cross section database[5].

6.2 Validation and Application of RELAP5/PANBOX

The extensive verification program also includes the recalculation of the transient 'inadvertent opening of a pressurizer relief valve' with the coupled system. In the past an iterative process between the plant system code and the core kinetics code has been used to determine safety related parameters like maximum fuel centerline temperature or DNB ratio. In this iterative procedure core inlet enthalpy and mass flow rate and core outlet pressure were written on file by the plant system code. With this information as input, a standalone PANBOX calculation has been performed. The resulting core power distribution has been used as input for a second thermalhydraulic calculation, followed by the next PANBOX calculation. This procedure has been repeated until the relevant solution variables remained unchanged.

Nowadays, this scenario can be calculated in one step using coupled code systems. As part of the verification program this case has been recalculated with the RELAP5/PANBOX code system using the same initial and boundary conditions as in the iterative procedure. The main result can be summarized by comparing the reactor power of the coupled and iterative solution. A very good agreement between the two methods has been achieved. (Figure 3). This demonstrates that data transfer within the coupled code system is performed correctly.

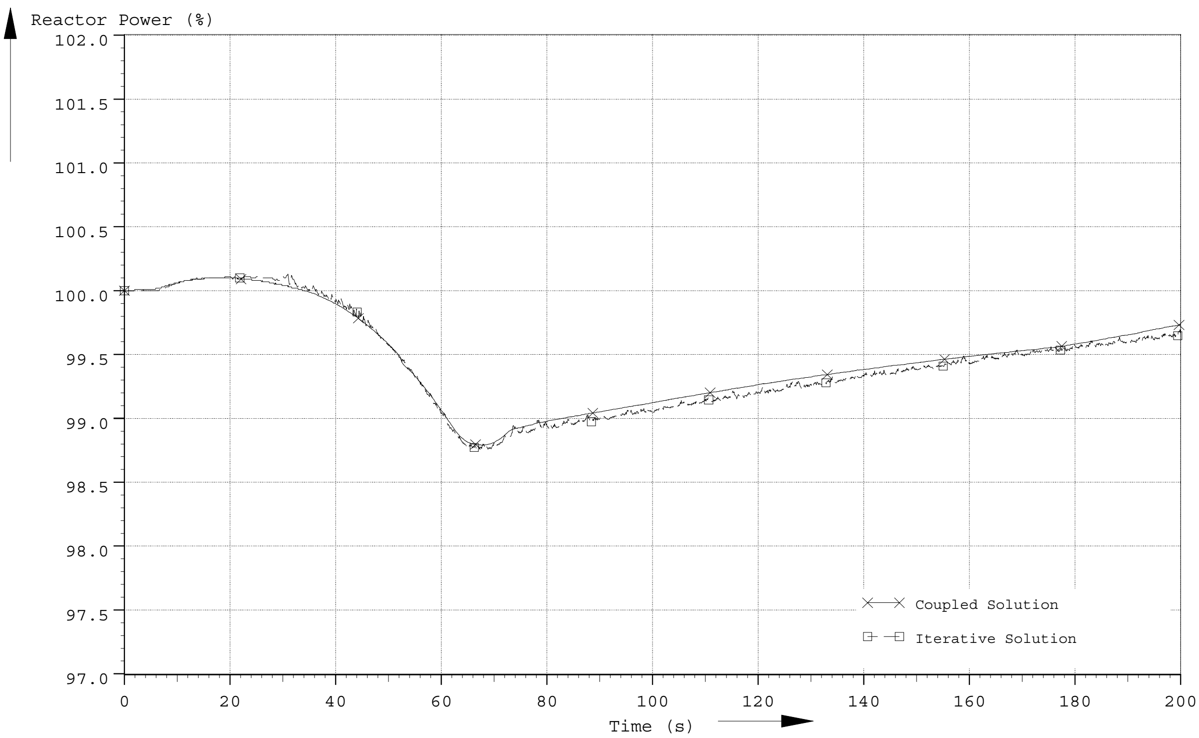

Figure 3. Inadvertent Opening of a Pressurizer Relief Valve:
Comparison between Iterative (4 iterations) and Coupled Solution

The ongoing program to validate the R/P/C code system includes the calculation of international benchmarks. For example, the R/P/C results for the LWR "rod ejection" benchmarks[6,7] and the PWR "uncontrolled rods withdrawal" benchmarks at zero power[8] compare favorably with the reference solutions obtained by using finer meshes in space and time. Furthermore we are currently participating in the NEA main steam line break benchmark[9].

The current scope of validation is successfully completed by comparisons with several measurements, e.g. single rod drops and reactor trips, main coolant pump trips, switch-on and -off of coolant pumps, a pump shaft break event and load rejection to in-house power[10]. Though the experience with coupled systems like R/P/C is still limited, in comparison with the extensive use of the standalone codes RELAP5 and PANBOX, it could already be demonstrated that the application of coupled codes can reduce uncertainties in the analysis of, e.g., boron dilution accidents and specific cool-down transients with strongly negative moderator temperature coefficients. R/P/C was successfully used for both these types of accidents.

6.3 Benefits of Coupled Analyses

The general motivation for integrating 3-D spatial kinetics in system codes has two major aspects. On the one hand, time consuming and costly iterative procedures can be avoided. This in turn minimizes manpower cost through increased efficiency. On the other hand, the coupled code system increases the accuracy of simulation for a variety of applications in licensing safety analysis by eliminating conservative assumptions at code interfaces.

As an example of R/P/C application we consider a multiple steam system piping leak following an earthquake. Due to feedback effects resulting from the strong temperature decrease a return to power occurs although reactor scram has been actuated. The main goal of the analysis is to demonstrate that long-term subcriticality is ensured and the plant is being brought to a safe and stable state.

Assuming that the volume control system is unavailable, boron injection is restricted to the extra borating system. However, borating system actuation is limited to conditions where primary pressures are below the pump shut-off head. The amount of injected borated water and, consequently, the final reactivity are closely correlated with the volume contraction in the primary system: the lower the reactor power, the higher the volume contraction, the higher the injected boron mass, and vice versa.

Using a traditional point kinetics approach with preserved axial and radial core power distributions and moderator feedback based on the coolant temperature reactivity coefficient over a wide temperature range would inevitably result in an early recriticality and higher peak powers. As a consequence, primary pressures would be lower. Especially for present and future MOX fuel assemblies with even stronger negative moderator temperature feedback the demonstration of long-term subcriticality is endangered.

The advantage of the coupled code system is demonstrated by using R/P/C for the given multiple steam line leak scenario. Results are presented in Figures 4-5. Shortly after the initiating event scram is actuated and reactor power is decreasing. However, due to feedback effects at about 200 s a return to power occurs and reactor power increases (Figure 4). In this state the reactor is critical and can be made subcritical only by an addition of boronated water with the extra borating system. Inline with the borating system injection the boron concentration in the primary system is increasing and reactor power is gradually decreasing. Core reactivity becomes negative at about 1900 s.

As the volume control system is not available primary pressure is increasing (due to primary coolant volume increase). At about 3300 s, when primary pressure rises above 14 MPa (Figure 5), a bypass valve in the recirculation line of the extra borating system starts opening, thereby reducing the injection rate. Further on, boron injection will be stopped and the boron concentration will remain constant. Further reduction of core reactivity will be impossible. However, in this phase of the transient the reactor is already subcritical. Due to the realistic feedback treatment using a 3-D neutron kinetics model adequate boron injection is enabled to guarantee a sufficient shutdown margin. The calculation illustrates that safety requirements and specific acceptance criteria can be fulfilled without cost intensive hardware modifications.

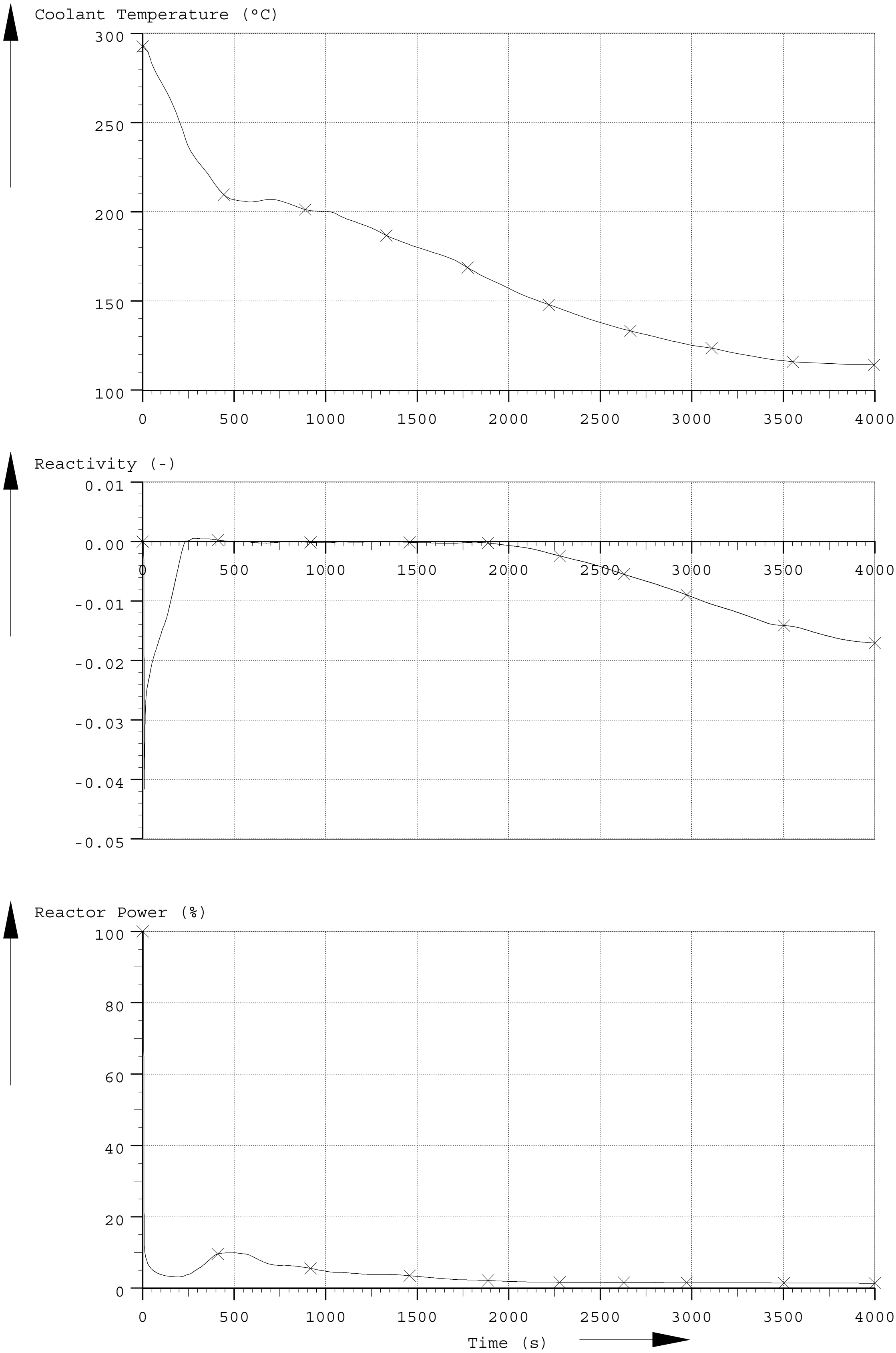

Figure 4. R/P/C Results for a Multiple Steam Line Leak

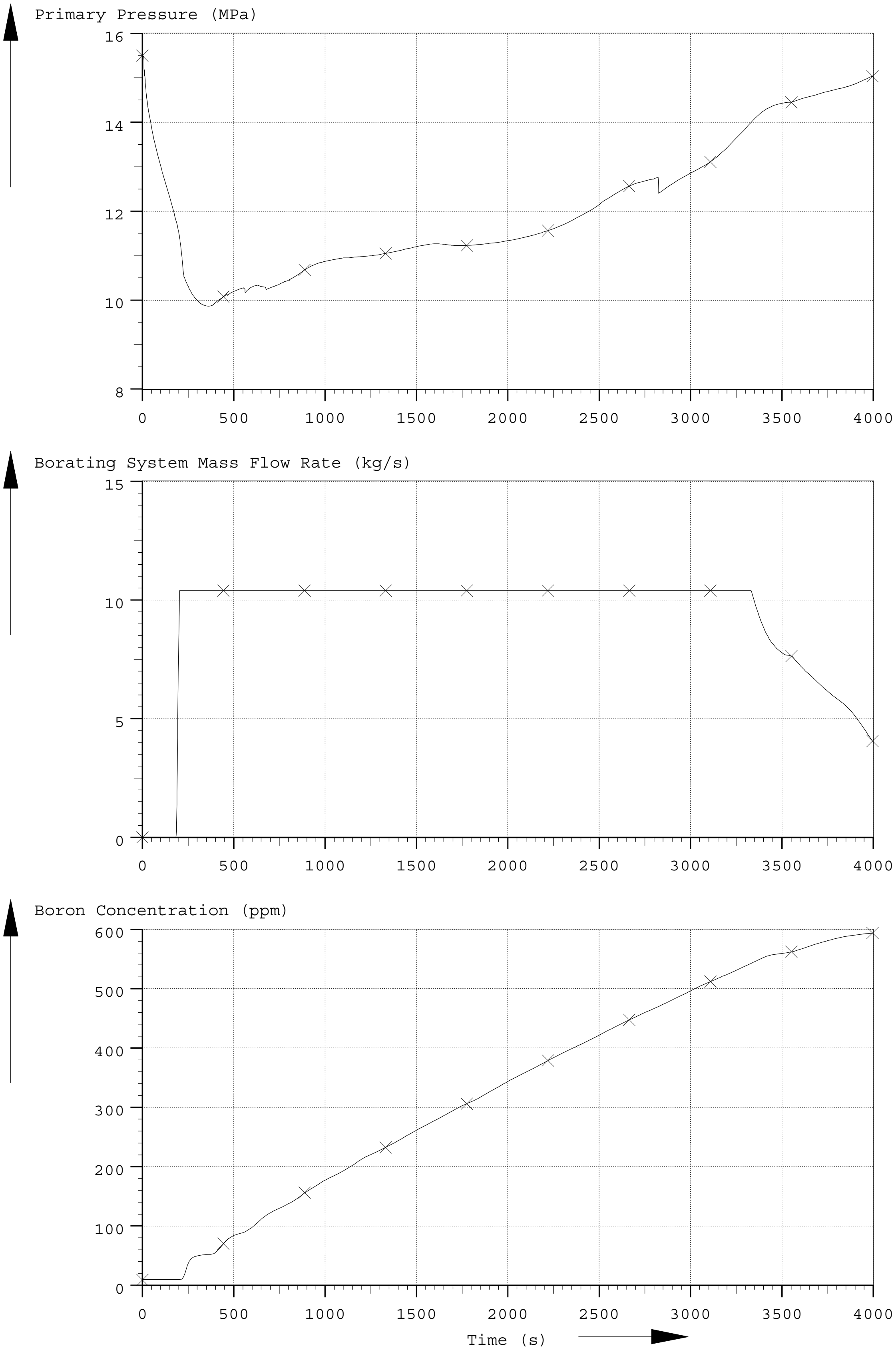

Figure 5. R/P/C Results for a Multiple Steam Line Leak

CONCLUSIONS

The CASCADE-3D system is a consistent and comprehensive design tool for the analysis of plant behavior under both normal and off-nominal operating conditions. This work described the system architecture and presented capabilities and applications of the core design procedure SAV and of RELAP5/PANBOX - the transient part of CASCADE-3D. Most of the CASCADE-3D components for in-core fuel management and accident analysis are also available in a hexagonal geometry option and can be applied to the analysis of VVER-type reactors[11]. Ongoing developments of RELAP5/PANBOX include, on the one hand, the continuing validation process using experimental data and international benchmarks and, on the other hand, the implementation of the Adjoint Method of Sensitivity Analysis[12].

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