ANALYSIS OF A LOAD REJECTION EVENT AT THE NUCLEAR POWER PLANT GOESGEN
WITH COUPLED NEUTRONIC/THERMAL-HYDRAULIC CODE SYSTEMS

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ABSTRACT

The use of coupled 3-D neutron kinetics/thermal-hydraulic system codes is becoming increasingly important as a means to accurately simulate plant transients and to analyze complex accident sequences which are characterized by a significant change in the power distribution of the reactor core and by interaction between the core and the plant. In the development of these large computer codes the verification and validation procedure is vital to establish how well the codes perform. Part of the code verification includes the comparison of results with those of other codes. This can most easily be done through the calculation of benchmark problems, but the accuracy of the results can only be determined through comparisons with experimental data. Since only a few adequate experimental data exist to validate coupled codes an inadvertent event at the NPP Goesgen - a load rejection to auxiliary power supply - was used for the validation of RELAP5/PANBOX. The comparison of calculated and measured data shows the ability of RELAP5/PANBOX to analyze this type of transient and validates in-core and ex-core detector models and certain aspects of the coupled core and thermal-hydraulic models. Similar investigations are being performed with TRAC-PF1/NEM. Code-to-code comparisons with RELAP5/PANBOX will help to estimate and further reduce modeling uncertainties.

1. INTRODUCTION

The accurate computer simulation of the core surveillance and protection systems is important for safety and design calculations. The determination of the accuracy of these computer models is accomplished through validating the models by benchmarking the results of the models with experimental data. The ex-core and in-core detector simulation models are developed for the coupled thermal-hydraulic/neutronic code systems RELAP5/PANBOX[1] and TRAC-PF1/NEM[2]. The development and validation of these models is part of a cooperation between Siemens KWU and The Pennsylvania State University. This effort is also important in the validation of the entire coupled code systems. This will help validate the nodal model, coupling, and thermal-hydraulics model for both of these code systems.

2. SIEMENS PWR CORE SURVEILLANCE CONCEPT

The PWR core surveillance concept developed by SIEMENS/KWU is consistently based on in-core instrumentation. A schematic representation is given in Fig. 1. This concept combines two complementary systems with different tasks - the Aeroball system as movable flux mapping system, and the Power Density Detector (PDD) system as monitoring system employing fixed in-core detectors. The calibration process functionally links the two systems; the monitoring signals are calibrated at regular intervals under reference conditions using the results of the Aeroball system.

2.1 THE AEROBALL SYSTEM

The Aeroball system is a flux mapping system based on movable activation probes. The activation probes consist of 1.7 mm diameter steel balls forming stacks guided in tubes (outside diameter 3 mm). The balls are made of steel alloyed with 1.5 % vanadium, which acts as neutron sensitive material. The ball stacks normally reside outside the core. To take a measurement, all the stacks are transported into the core at the same time where each extends over the entire active core height. The balls are transported pneumatically by applying carrier gas pressure. Once in the core, they are activated by the neutron flux for three minutes. This short irradiation time makes flux mapping like a snapshot and enables accurate measurement even in semi-transient conditions (xenon-redistribution). During irradiation, a process computer registers all data needed for subsequent evaluation. On completion of irradiation, the carrier gas pressure is applied from the other direction, driving the ball stacks out of the core and into a detector array that is located outside the biological shield. The gamma radiation emitted by the balls is then read by detector arrays arranged in a measuring table. The recorded counts are the primary data that are further evaluated by a computer to yield the 3D-power density distribution and other parameters representative of core conditions.

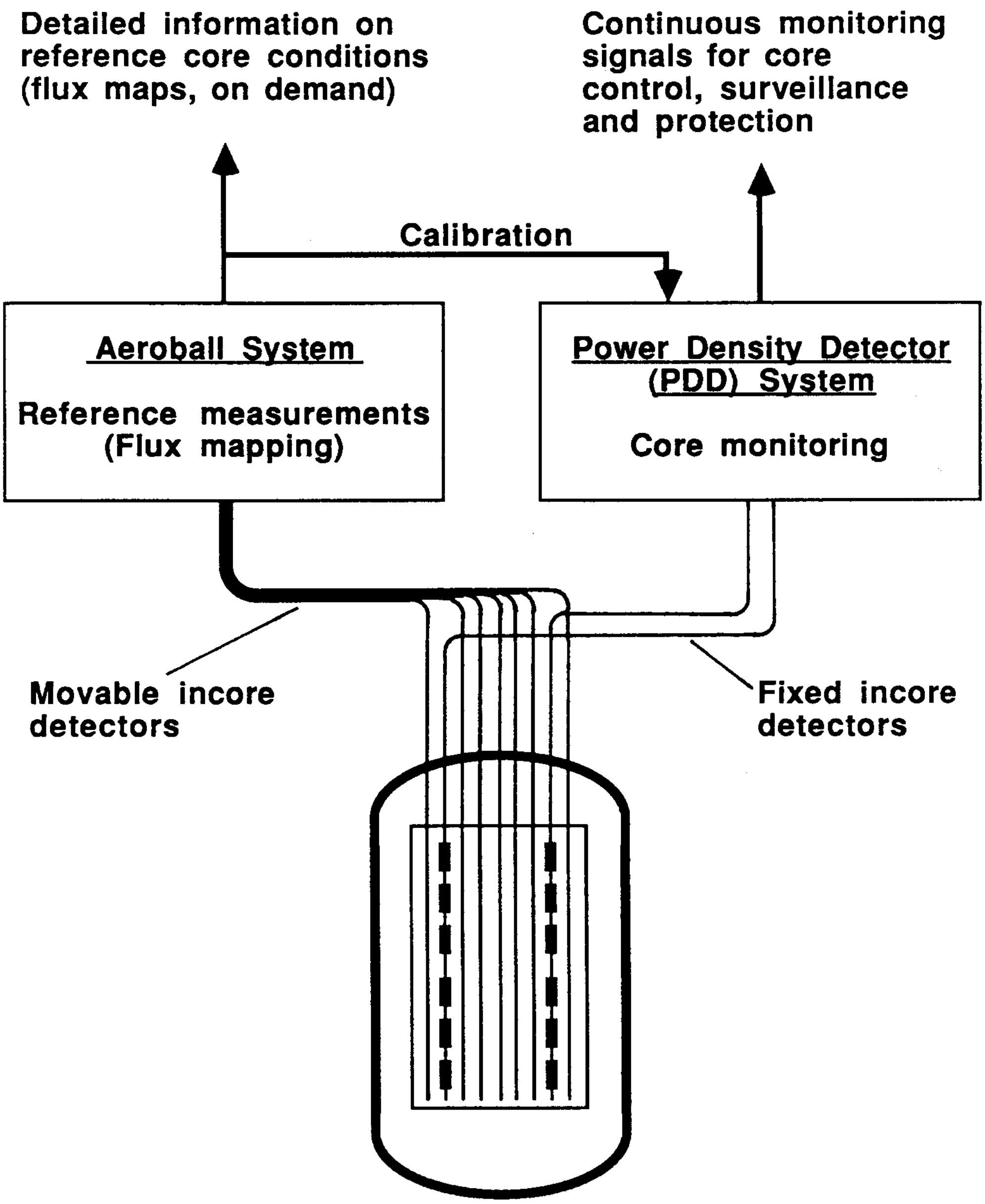

Figure 1: Combined In-core Aeroball and PDD Systems

2.2 THE POWER DENSITY DETECTOR SYSTEM

The Power Density Detector system is a monitoring system which employs self-powered neutron detectors with prompt response, i.e. fixed in-core detectors called Power Density Detectors (PDDs). These signals are used for core control and core protection purposes; they are processed together with selected other process variables to yield continuous monitoring signals representative of core conditions. The in situ calibration capability of the PDD system as well as the proximity of the PDDs to the hot spots or hot channels in the core provides a higher accuracy than achievable by ex-core instrumentation. As a consequence, the allowance for monitoring system imperfections is smaller for the PDD system than for ex-core systems. This provides additional margins for core design and flexibility of operation. The number and distribution of PDDs over the core is determined by considering the two main requirements related to the application of monitoring signals: the Core Surveillance and the Core Protection.

The PDD distribution within the core must support the capability to detect and assess local power increases caused by flux and power redistributions that occur under non-steady-state conditions. The PDD distribution must also make allowance for proper signal redundancy. Examples of power shape perturbation modes relative to an unperturbed reference condition and the arrangement of PDDs is given in Fig. 2. Signal redundancy is attained by increasing the number of in-core detectors above the theoretically required minimum in order to achieve a high degree of reliability as required for safety grade instrumentation. This is a “functional” redundancy in the sense that power density increases are “seen” by several detectors. If one or more of these detectors fail, the remaining detectors are still able to register the increase. In this way, failure of one or even more in-core detectors does not degrade monitoring system performance and does not impair significantly the accuracy of the derived monitoring signals used for core surveillance and protection.

The calibration of the PDDs has to be matched to the reference conditions at regular intervals. Reference values for this calibration are provided by the Aeroball system. The calibration - usually performed every 14 EFPD in unperturbed reference conditions - transfers the accuracy of the Aeroball system to the PDD system.

2.3 THE EX-CORE DETECTOR SYSTEM

The ex-core detector system is used for surveillance and limitation of the integral reactor power and provides information about global axial or azimuthal power shape perturbations. The system relies on neutron flux signals of the out-of-core (ex-core) instrumentation and on loop temperature rises. The arrangement of the neutron flux ex-core detectors is provided in Fig. 3.

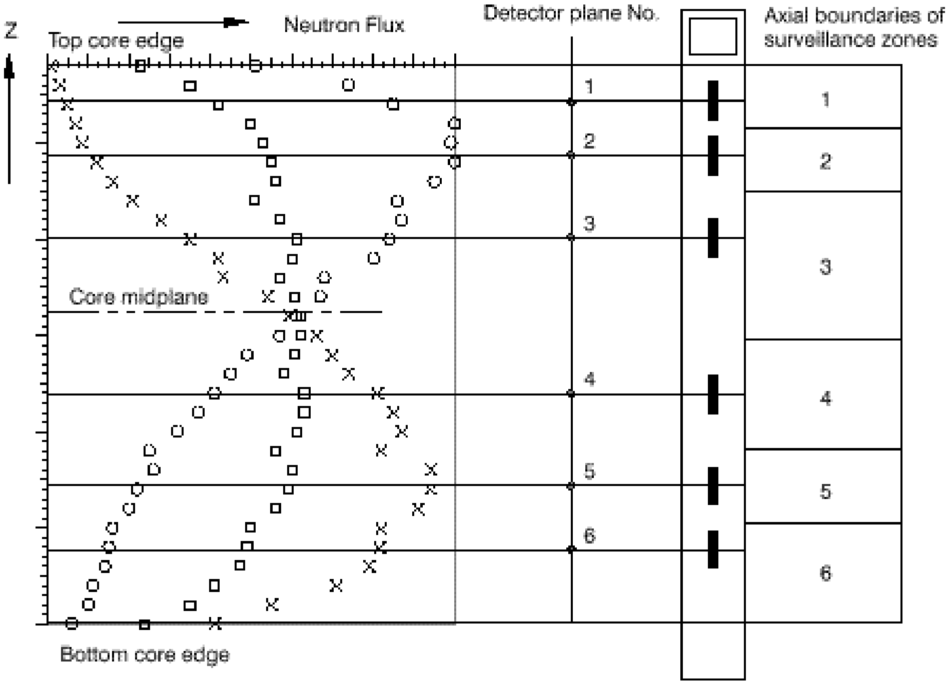

Figure 2: Axial Power Shape Perturbations (Aeroball System Measurement) and Axial Arrangement of PDDs within Fuel Assembly

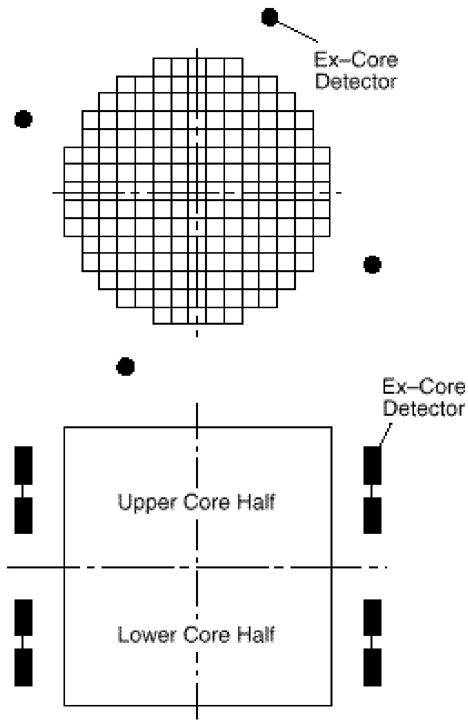

Figure 3: Arrangement of Ex-Core Detectors

3. COUPLED CODE SYSTEMS

3.1 RELAP5/PANBOX

Siemens KWU uses the coupled code system RELAP5/PANBOX (see reference 1) as their basis for three-dimensional thermal-hydraulics/neutron kinetics analysis. RELAP5 is a full system thermal-hydraulics code based on the non-equilibrium, non-homogeneous, six-equation model of two-phase flow. PANBOX is a nodal space-time kinetics core model. It is capable of solving three-dimensional transients using both polynomial and (semi-) analytical nodal expansion methods. PANBOX is coupled to RELAP5 via the general Siemens interface package EUMOD. Other external code modules (e.g. I&C) may also be linked through EUMOD to RELAP5/PANBOX. Core thermal-hydraulics can be calculated using RELAP5 or COBRA, the core-wide and sub-channel thermal-hydraulic module of PANBOX. Thus RELAP5/PANBOX has the capabilities of RELAP5 with the added ability to calculate space-time kinetics with PANBOX and thermal margins with COBRA used with an embedded sub-channel grid. This allows a large range of coupled thermal-hydraulic/neutronics phenomena to be simulated. The coupling of PANBOX and RELAP5 via the EUMOD interface package establishes a code system that is capable of accurately predicting complex transients with coupled core/plant interactions including I&C actions of pressurized light water reactors.

3.2 TRAC-PF1/NEM

PSU uses the TRAC-PF1/NEM coupled code system (see reference 2) as the basis of their safety analysis of pressurized light water reactors. NEM is capable of calculating three-dimensional neutronics transients in three different geometries: Cartesian, cylindrical, and hexagonal. It uses the transverse integration approach for de-coupling the three-dimensional spatial dependence based on a semi-analytical one-dimensional flux expansion and improved quadratic transverse leakage approximation. It also has a one-dimensional decay heat model. TRAC-PF1 is a full system thermal-hydraulic code. It has a three-dimensional vessel model and is also based on the non-equilibrium, non-homogeneous, six-equation model of two-phase flow. NEM is fully integrated into TRAC-PF1 and is capable of modeling a large range of coupled thermal-hydraulics and neutronics conditions. The core thermal-hydraulics can be coupled to the neutronics using a very coarse or a fine thermal-hydraulic nodalization in the three-dimensional vessel component.

3.3 EX-CORE DETECTOR MODEL

The ex-core detector simulation model is developed to accurately simulate the ex-core detector response. Four detector sections are located around the bottom half and four around the top half of the core. The detectors are positioned so that each is assigned to one eighth of the core. The detector responses are calculated by summing up the outgoing currents from the core over the area of each detector view times a correction factor. The correction factor includes the macroscopic removal cross-sections for steel and water and the angle between the outgoing current and the detector position. The calculated detector signals are normalized to the initial steady-state value for easy comparison with experimental data.

3.4 IN-CORE DETECTOR MODEL

The in-core n-β simulation model is developed to simulate the in-core Power Density Detector (PDD) system in nuclear power plants. The PDD system described above is an integral part of the core monitoring and control system of Siemens KWU designed pressurized water reactors. It is essential that the n-β detectors are modeled accurately so that transient analysis can be accurately predicted. The in-core detectors are placed in control rod guide tubes of fuel assemblies not occupied by control assemblies. Usually six detectors are axially arranged in one lance and maximum eight lances are intalled in a core of a Siemens KWU PWR. The activation of the n-β detectors is calculated by integrating the flux shape over the detector length. A second or fourth order integration scheme can be used to integrate the axial activation rate over the detector length.

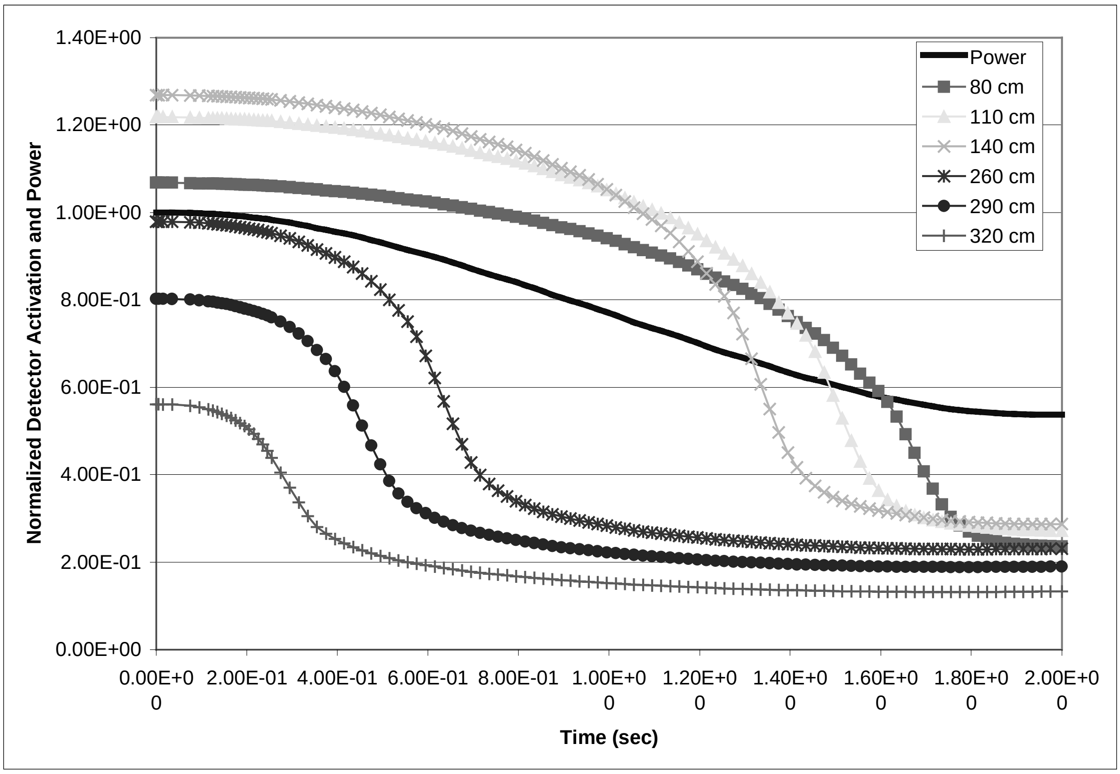

Figure 4: Plot of Core Average Power and Detector Activations for Fuel Assembly Where Control Assembly is Inserted

These models have been implemented into the RELAP5/PANBOX code system and the Penn State version of TRAC-PF1/NEM and successfully verified (Fig. 4).

4. LOAD REJECTION EVENT AT NPP GOESGEN

The analysis of the Goesgen Load Rejection Transient is performed by the Siemens KWU code system RELAP5/PANBOX. The results are compared with measured signals of the plant. This validates the in-core and ex-core models and helps to validate the analysis capabilities of the coupled code system. Similar investigation is being performed with TRAC-PF1/NEM to estimate the modeling uncertainties by code-to-code comparisons.

The load rejection to auxiliary power supply on 8 April 1998, of the NPP Goesgen (KKGg) was caused by an inadvertent opening of the unit switch. As a consequence the energy transmission to the grid was cut and the plant was reduced to isolated operation. The measures for load rejection were automatically initiated and worked as designed. The load rejection caused a short-term frequency rise of the turbo-generator unit and hence also of the main coolant pumps. The associated increase of core coolant flow led, via the negative coolant temperature feedback, to an also short-term overshoot of reactor power. Intervention of the frequency control, according to design, occurred so effectively that no reactor trip was triggered by this power increase. Only some control assembly pairs were dropped for power limitation, which was efficient enough to keep the reactor markedly below the trip threshold also in the following phases of the transient.

The event was recorded by the fast signal recording system. Two times per second 250 signals (including selected PDD traces) were recorded. In this way all essential data of transients are stored for further evaluation, e.g. processing by computer codes.

The load rejection event is especially suited for the validation of complex coupled neutronic/thermal-hydraulic codes with respect to their ability to calculate local safety-related parameters. Various plant I&C actions (core surveillance and protection system) influence the power history. In addition the asymmetric drop of selected control assemblies results in the challenging problem for recalculation of time-dependent asymmetric neutron flux deformations.

The schematic cross sections of the reactor core in Fig. 5 help to identify the radial positions of the control assemblies, the Aeroball system and the in-core n-β detectors.

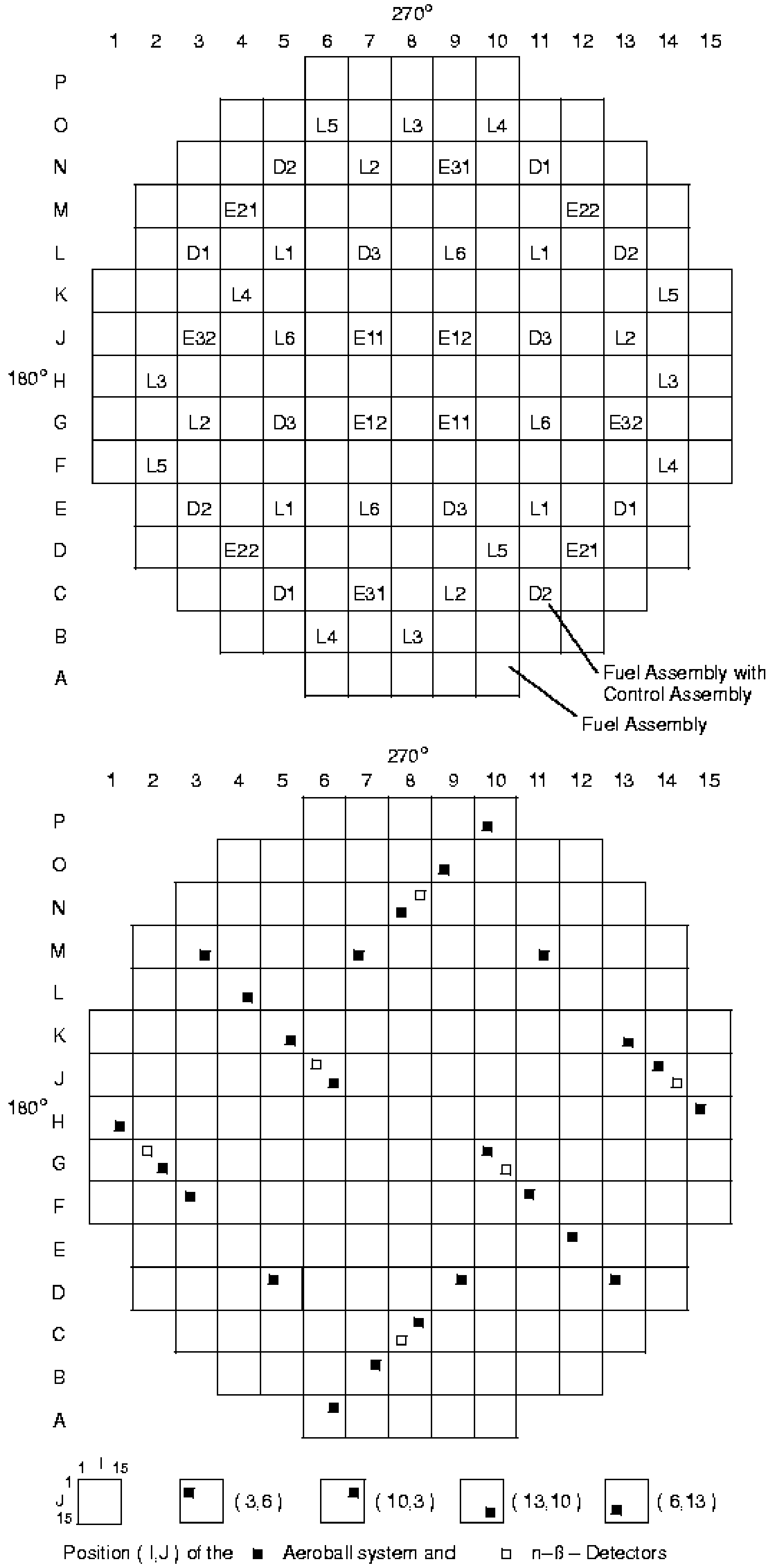

Figure 5: NPP Goesgen, Positions of Control Assemblies, Aeroball System and n-β Detectors

4.1 CORE SPECIFIC SEQUENCE OF EVENTS

Clock Time	Event
11.05.03	Opening of the 400 kV circuit breaker and reduction of the generator power to auxiliary power supply of approx. 50 MW.
11.05.03	Over-speed of the turbine following the load rejection. As a consequence, the coolant flow increases and the core inlet temperature decreases. The reactor power increases (effected by the moderator feedback) to nearly103 % (2nd maximum of measured reactor power). (Burn up state is 20 days before EOC).
11.05.03	0.3s after the load rejection the signal "synchronous control assembly dropping" appears. This signal is formed as follows: Reactor power minus generator power > 30% AND Reactor power > 42% AND Generator power < 15% From the signal of the "synchronous control assembly dropping" the following control assemblies are selected for dropping (see Fig. 5). E32 (J03/G13), E31 (C07/N09), E12 (G07/J09), E11 (J07/G09), E22 (D04/M12) As a result of the drop of the above 5 pairs of control assemblies reactor power decreases to about 38 %.(2nd maximum of measured reactor power). As can be seen from Fig. 5 the dropping is asymmetrical within the different core quadrants; thus the corresponding ex-core detector signals are different. The formed power signals for control, limitation and protection are therefore different. For actions of the control-, and limitation - systems the 2nd maximum of the measured signals is used.
11.05.07	After the power reduction and the following decrease of the coolant temperature the power increases fast and a further control assembly dropping is released. The release signal is triggered if the set point “reactor power > 47 %” is exceeded by the 2nd maximum of the measured signals. After the first "synchronous control assembly dropping" the position of the selected control assemblies for dropping due to “power higher than the relevant set points” has changed. Thus the dropped pair of control assemblies is E21 (M04/D12).
11.05.07	Activation of the “back-position control”. The dropped control assemblies (E11 to E32) are withdrawn by the mean coolant

temperature control; countercurrent to this the control assemblies of the remaining D-bank (D1 to D4) are inserted.

11.05.33 A further increase of the reactor power > 42 % causes control assembly dropping again. With a time interval of 2 seconds for each pair the partly withdrawn control assemblies are dropped again with the sequence “E32, E31, E11, E22, E21, E12” until the power (2nd maximum) is reduced to lower than 42 %. This control assembly drop is not very effective because the control assemblies were only some "cm" withdrawn by the control measures.

Subsequently the mean coolant temperature control initiates withdrawal of the L-bank to maintain the mean coolant temperature at the target value corresponding to a reactor power of about 35 %. In parallel the dropped control assemblies E11 to E32 are withdrawn and for compensation the control assemblies of the D-bank (D1 to D4) are inserted.

With respect to core specific events only the first 60 s are of major interest.
About 45 min after the load rejection the operator started to increase reactor power to full load.

4.2 COMPARISON OF CALCULATED AND MEASURED DATA

The comparisons of the measured data with the results obtained with the Siemens code system RELAP5/PANBOX are presented in the following Figs. 6 - 10. The recalculation by the PSU code system TRAC-PF1/NEM is planned as the next step. Only the most important of the 250 available measured signals are selected and compared with the calculated values during the first 60 s. The scale range of the parameters in the figures is chosen similarly to the measuring range.

In general the accuracy of the calculation is very good. The calculated plant as well as core related signals show the correct behavior. The determination of actions initiated by the I&C is of special importance. In particular the initialization of the drop of selected control assemblies and control assembly movements (see Fig. 8) depend very sensitively on just exceeding or just not exceeding specific set points. The subsequent sequence of events and power history is significantly influenced by such decisions. To get comparable power distribution signals (in-core and ex-core signals) the I&C actions in the calculation were adapted to the measurement, where such small differences between measured and calculated signals would have triggered different I&C actions.

Fig. 6 shows the short-term corrected thermal reactor power signals. The signals are formed using the temperature rise in conjunction with short-term corrective contributions gained from the ex-core neutron flux signals per quadrant. Therefore four signals – one per quadrant – are available. The calculation is in excellent agreement with the measurement.

The following discussion will show in detail that the resulting calculated in-core (see Fig. 7) and ex-core signals (see Figs. 9) as well as the signal of axial power shape (axial imbalance) of the ex-core detector signals (upper minus lower core half, see Fig. 8) demonstrate that also the time dependent redistribution of axial and radial power is well determined. The fast signal recording system records only one lance of the n-β in-core detector system, which is the radial core position J6. This lance was calibrated to keep the hottest fuel rod of the core under surveillance. Four of the six detectors of this lance operated correctly. The two bottommost detectors were out of operation.

The calculation reproduces very well the basic behavior of the in-core detector signal. Nevertheless remarkable deviations appear. The maximum deviation of the local power density from the registered in-core detector signals during the first 35 s of the calculated time period amounts to about 55 W/cm (24 % nominal average linear heat generation rate). From 35 s on the underestimation of the axial power offset increases by a few percents (see Figs. 7 and 8). Accordingly during the remaining time until 60 s the maximum deviation of the local power density reaches 80 W/cm (35 %). These deviations result from different effects. Only part of them result from the limited accuracy of the transient RELAP5/PANBOX calculation.

The comparison of the measured signals of the four in-core detectors with the calculation (see Fig. 7) shows that the calculation underestimates at the very beginning of the transient the power level in the area surrounding the detector lance by approx. 10 %. This deviation results from the cycle depletion calculation on which the RELAP5/PANBOX calculation is based and influences the whole transient calculation. In addition the cycle depletion calculation was performed with the assumption of completely withdrawn control assemblies, whereas the plant was operated with L-bank at 5 cm inserted. To simulate the control assembly movements during the transient, also the initial control assembly position was adapted to the measurement which means an instantaneous perturbation of the axial neutron flux distribution. The effect is a stronger bottom peaked power distribution compared to the measurement, which can also be seen in Fig. 7. Naturally this deviation remains during the whole transient. The consideration of these initial deviations allows judging the transient calculation accuracy separately.

An additional effect which has to be considered when judging the in-core detector signals is the noise signal of Co 60. At full load the calibration compensates the Co 60 noise. But at part load the relative contribution of the noise to the detector signal can increase significantly. The effect depends on the irradiation duration of the detector (amount of converted Co 59 to Co 60 by absorption of neutrons). The detectors of the recorded signal in Position J6 had been in the core since the first cycle, i.e. for 19 cycles. So the signals at part load can be increased significantly, but the effect is not amenable to quantification. In the sense of core surveillance this effect is conservative. But for comparison with the calculation it has to be considered and at part load the real deviation between measurement and calculation can be assumed to be some percents smaller.

The maximum deviations appear at the uppermost in-core detector position. This detector is positioned near the axial reflector where the neutron flux has a strong buckling. Naturally to calculate the neutron flux at this position is most challenging.

The in-core detector lance at position J6 provides information on the accuracy of the calculation at the inner region of the core. The ex-core detectors allow judging the accuracy at the core edge (see Figs 8 and 9). The registered signal of axial power shape (axial imbalance) of the ex-core detector signals (see Fig. 8) is formed as difference between the upper and lower core halves signals. It demonstrates that also at the core edge the axial power shape redistribution, which in particular is induced by control assembly drops (see also Figs. 8 and 5), is very well calculated. Due to asymmetric control assembly drops at 0.3 s after opening of the 400 kV circuit breaker (approx. at 5 s of the calculated time period) the ex-core detector signals are asymmetric, too. The calculation reproduces these asymmetric signals very well, as can bee seen from Fig. 9.

So not only the transient behavior of integral but also of local signals is calculated accurately. In particular the capability to calculate the power redistribution shows that the coupled modeling of both, of the neutronic and thermal-hydraulic behavior, is well described in the 3-D core model of PANBOX.

Also several representative signals regarding the system behavior are considered (see Figs. 9 and 10). The power reduction results in a decrease of coolant temperature and this decreases the pressurizer water level and the coolant pressure (see Fig. 10). Later on mainly due to the measures of the mean coolant temperature control system the coolant temperature increases and consequently also the pressurizer water level and coolant pressure. The transient behavior predicted by calculation is in good agreement with the measured data.

Fig. 9 shows the transient behavior of steam generator water level and of the main steam pressure at the steam generator outlet for one of the three plant loops. Due to the collapse of steam bubbles during the reduction of power the steam generator water level decreases. In case of a load rejection to auxiliary power supply the I&C reduces the generator power very fast to approx. 5 %. Accordingly the steam release to turbine is abruptly reduced by the turbine inlet valves, which results in a strong imbalance of produced reactor power and the power carried off by the turbine. Thus the main steam pressure increases rapidly. The main steam bypass station releases the excess energy. Simultaneously with load reduction to auxiliary power supply the set point for main steam pressure is increased with a gradient to 80 bar. The main steam pressure is controlled to set point by the control system. The calculated results are in good agreement with the measured values.

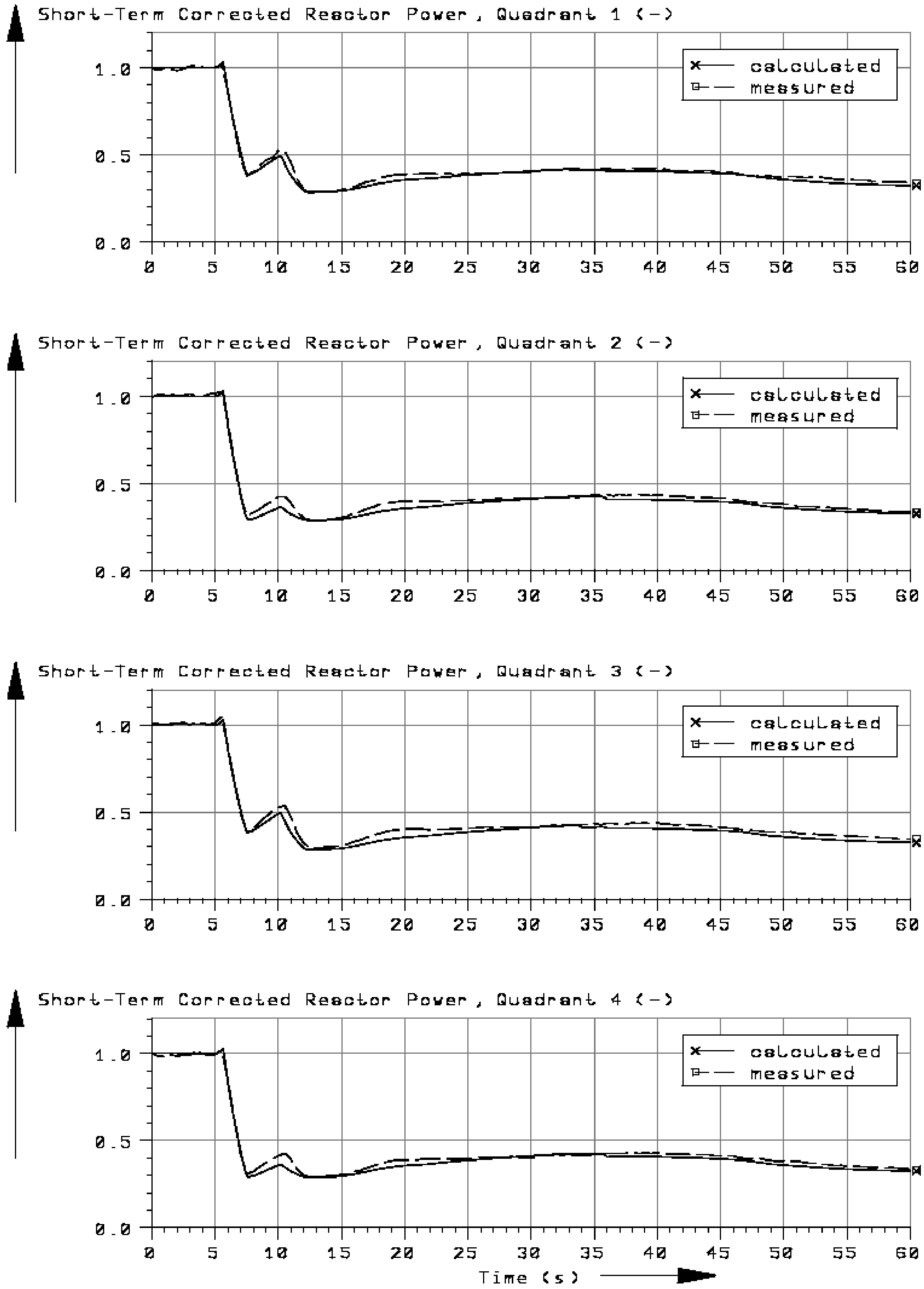

Figure 6: NPP Goesgen, Load Rejection, Short-Term Corrected Reactor Power

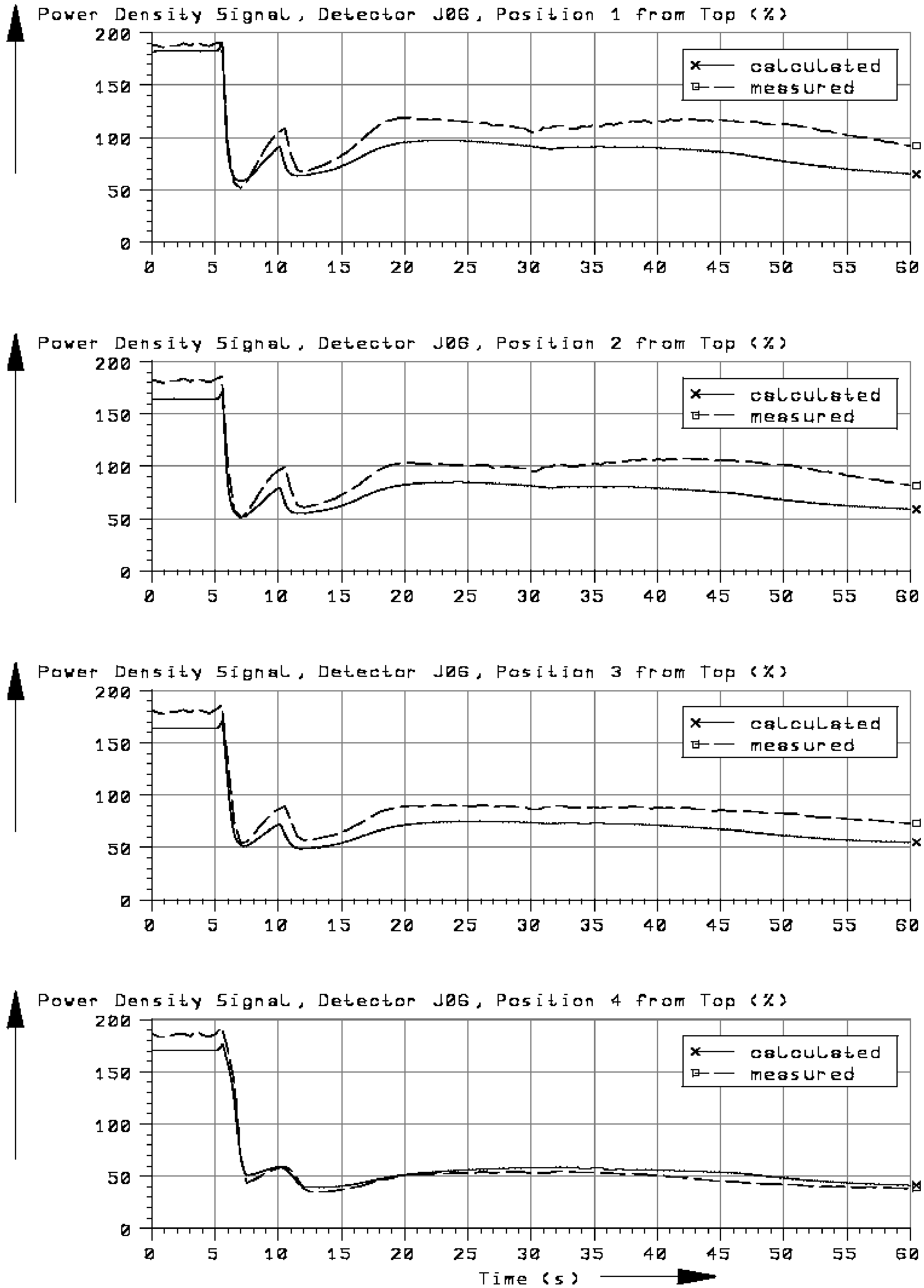

Figure 7: NPP Goesgen, Load Rejection, Power Density Signals of In-Core Detectors

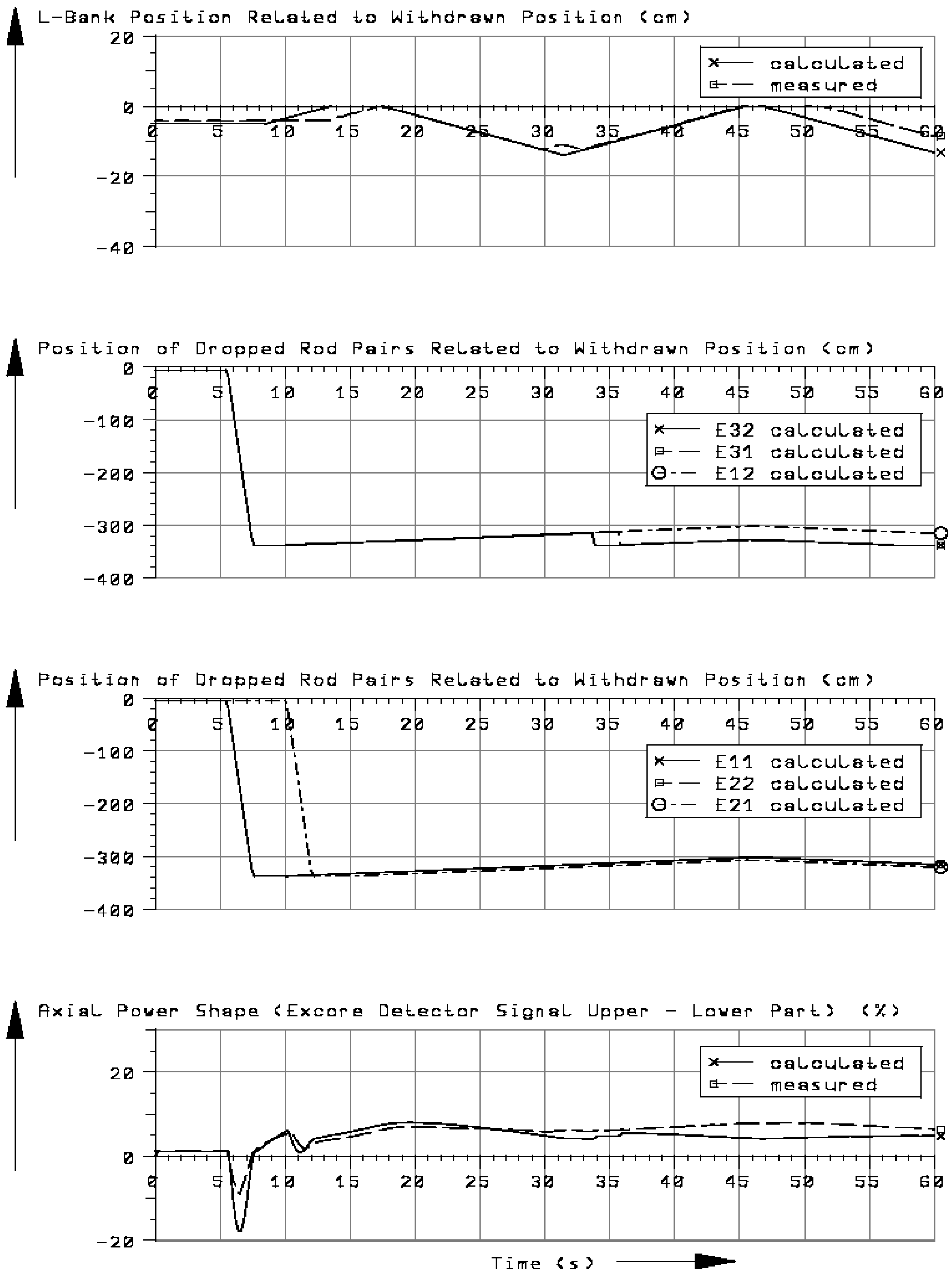

Figure 8: NPP Goesgen, Load Rejection, Control Assembly Movements, Axial Power Shape Signal

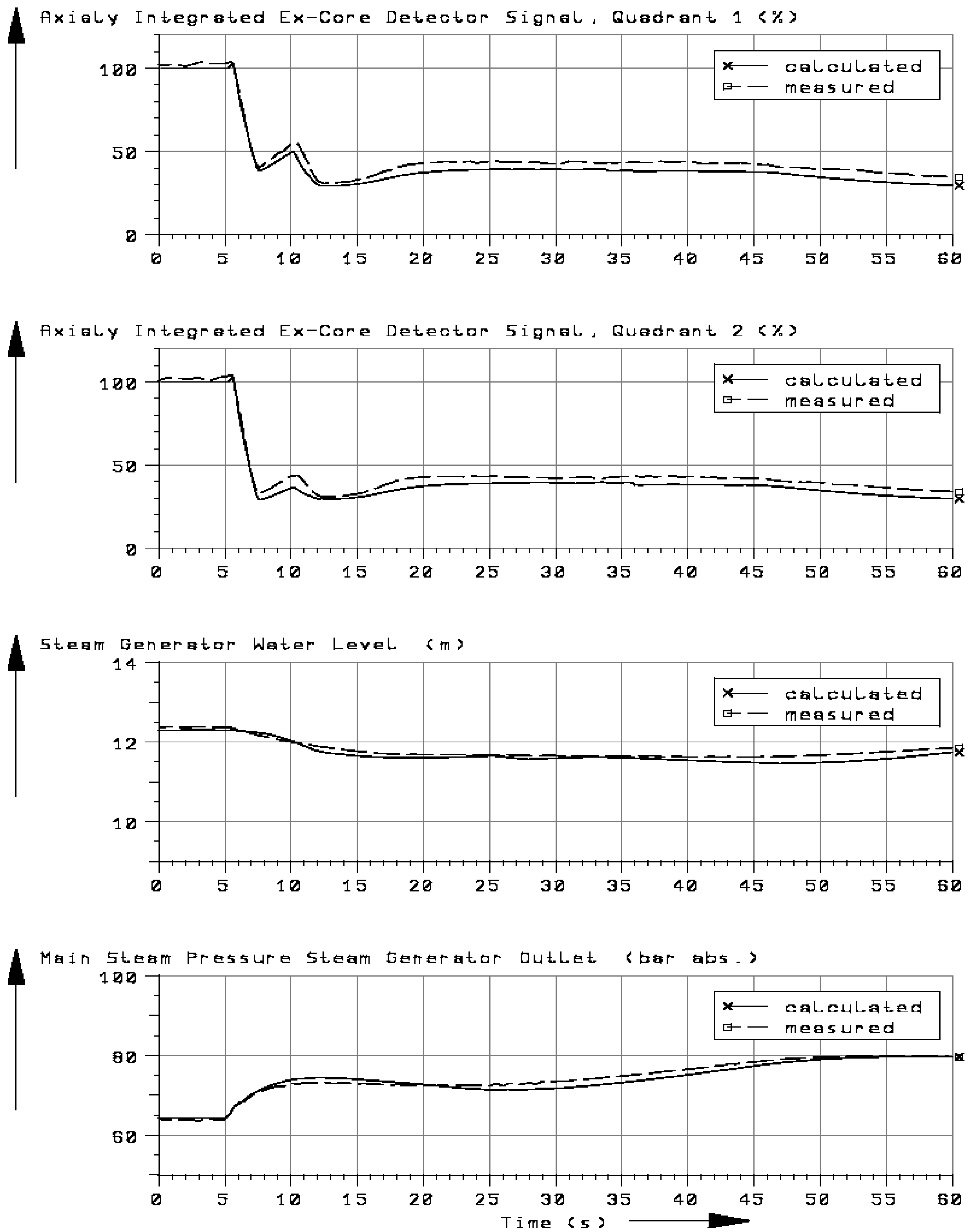

Figure 9: NPP Goesgen, Load Rejection, Ex-Core Detector Signals, Steam Generator Water Level, Main Steam Pressure

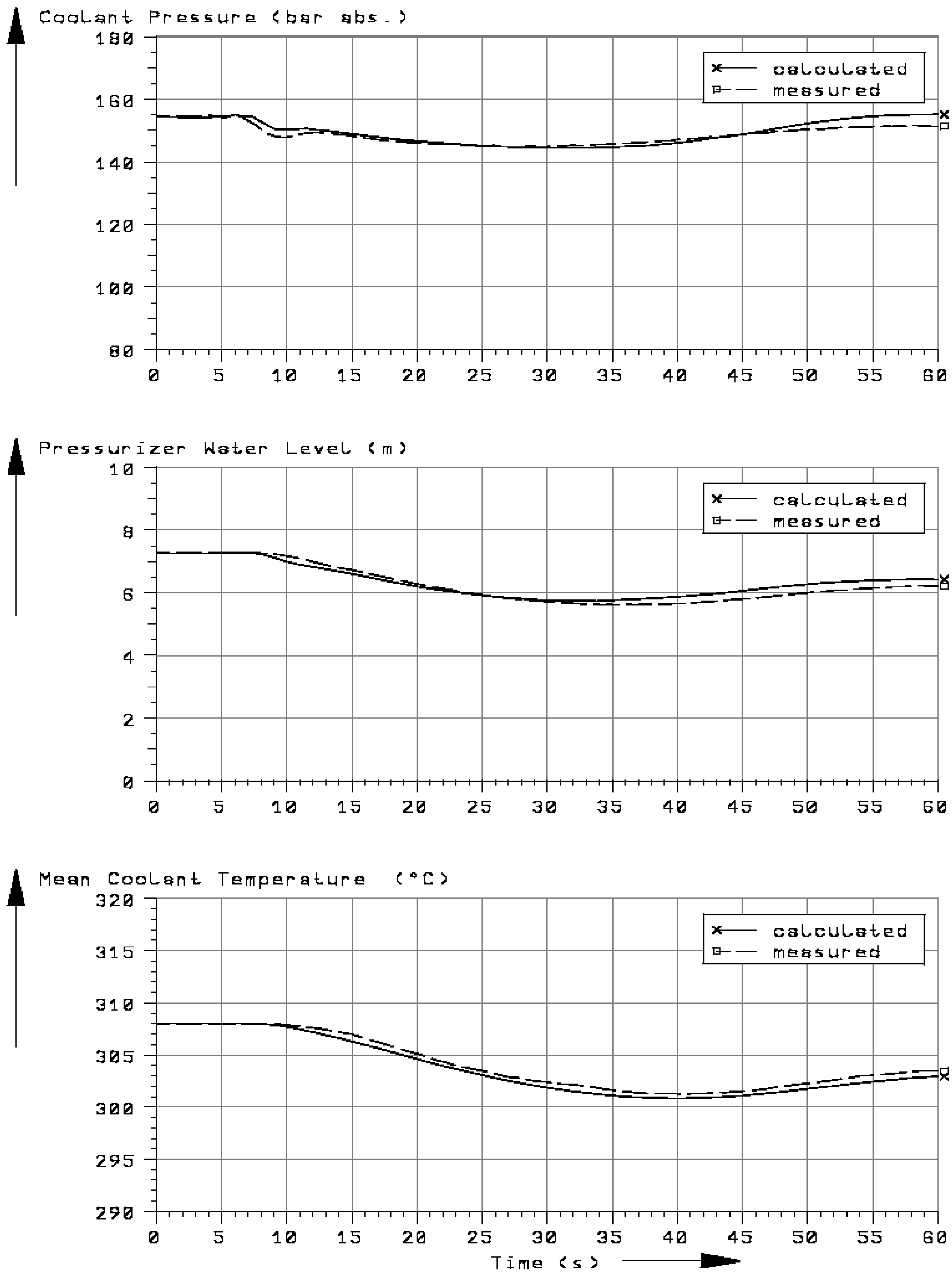

Figure 10: NPP Goesgen, Load Rejection, Coolant Pressure, Pressurizer Water Level, Mean Coolant Temperature

5. CONCLUSIONS

The transient of a load rejection to auxiliary power supply is a very challenging problem for a transient recalculation. The interaction between plant system and reactor core yields complicated I&C actions and induces asymmetrical radial and axial power redistribution. The comparison of the recalculation with a variety of measured core related and plant system signals demonstrates that RELAP5/PANBOX results are in excellent agreement with reality. It is therefore concluded that RELAP5/PANBOX can be considered validated for calculation of this kind of transients..

6. ACKNOWLEDGEMENT

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