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Safety and Control Studies of a Boiling Water Reactor Plant, Using a Real Time Simulator

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Safety and control studies of a boiling water reactor plant, using a real time simulator

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Summary

The mathematical model presented in this paper simulates the dynamics of the main components of a BWR power plant in real time. The model of the reactor core is one dimensional. The other components are described by lumped parameter models. The model is programmed for a hybrid computer, and is currently used for safety and control studies of a power plant under construction. Some results of these studies will be presented in the paper.

Sicherheits- und Regelungsstudien über eine Siedewasserreaktor-Anlage unter Anwendung eines Real-Zeit-Simulators

Zusammenfassung

Dieser Artikel beschreibt ein mathematisches Modell, das die Dynamik der Hauptkomponenten eines BWR Kraftwerkes in Real-Zeit simuliert. Das Modell des Reaktor-Cores ist eindimensional. Die anderen Komponenten werden mit einem Punkt Modell beschrieben. Das Modell ist für einen Hybridrechner programmiert und wird häufig bei Sicherheits- und Regelungsstudien für im Bau begriffene Kraftwerke benutzt. Resultate von diesen Studien werden in diesem Artikel präsentiert.

Etudes de la sécurité et du contrôle d'une centrale nucléaire à eau bouillante, en utilisant un simulateur de temps réel

Résumé

La maquette mathématique presentée dans ce document simule la dynamique des principaux organes d'une centrale d'énergie nucléaire à eau bouillante dans le temps réel. La maquette du noyau du réacteur est à une dimension. Les autres organes sont décrits par des maquettes parametriques groupées. La maquette est programmée pour un ordinateur hybride. Elle est généralement utilisée pour des études de la sécurité et du contrôle d'une centrale d'énergie en construction. Certains resultats de ces études seront présentés dans ce document.

Introduction

For the control- and safety studies of a nuclear power plant a mathematical model is highly desirable, which describes the dynamics of the plant in as detailed a way as possible. On the other hand the model should not be too complicated, so that it becomes difficult or expensive to use. It is also a great advantage if the model can be run in real time or faster. Such a model can, for instance, be used by the utilities already before the station comes into use

- to study the cooperation of different control systems within the plant and the philosophy of the operating instructions under varying operating conditions
- to forecast the consequences of component malfunctioning
- to simulate accidents.

Possible difficulties during the commissioning of the power station, or in connection with a change of its operational mode, can be foreseen and improvement of the control systems and/or operation instructions can be suggested.

Another important application of the simulator is for design and testing of control systems particularly with regard to advanced automation of the power plant with the use of a computer. In this case a model, which can be run in real time, is of great advantage.

The model of the plant presented in this paper simulates the dynamics, in real time, of a BWR reactor and the main components of a reactor power plant.

As the power control of a BWR to a great extent is controlled by the main circulation pumps, a good description of the influence of the coolant on the reactor dynamics is necessary, so that void, coolant temperature and fuel temperature feedbacks to the nuclear power can be properly taken into account. For this reason the model describes these variables in one (axial) dimension. The other components are described by lumped parameter models. The model is programmed for a hybrid computer.

Mathematical Model

Figure 1 shows the main components which are simulated in the model. The neutron diffusion equation is represented in one energy group and one dimension (axial). The reactor core is divided into 8 or 16 vertical sections. The time dependent equations for 3 delayed neutron groups, fuel and core hydraulics are written in one dimension as well. The program explicitly covers the axial direction of the core. Therefore, the influence of the radial direction must be taken into account by properly averaging the cross section data and other nuclear data, such that an average channel is represented. The nuclear parameters (cross sections etc.) are functions of fuel temperature, coolant temperature, void fraction and the position of the control rods. These parameters are described as polynomials. The coefficients of these polynomials are input data.

In the program there are two possibilities to get a critical reactor with a prescribed power level. Either the one-dimensional rod pattern can be chosen as input data or the neutron flux distribution. In the latter case the rod distribution is computed.

During the dynamic calculation a group of equally inserted control rods can be moved continuously. Furthermore, the control rods can be inserted rapidly into the core simulating a scram action.

The prompt and the delayed power are both assumed to be generated partly in the fuel, partly in the cooling channel and partly in the control rod cooling channel. The fuel is represented by three radial zones, separately calculated for each axial segment. The hydraulics of the cooling circuits are described by the equations for the conservation of mass, energy and momentum. Thermal equilibrium between steam and water phases is assumed throughout. Empirical correlation functions for slip and friction are used. Within the core the axial variations are represented giving feedback to the neutronics while the variables in the riser, top reflector and downcomer are lumped to a greater extent.

The external circuits, i.e. the steam line system, the turbine system, the feedwater system, the main circulator circuits and the various regulators are simulated essentially on the analog part of the computer system. The turbine system with a high pressure part, a reheater and a low pressure part is represented by a linear model. Further, the model comprises a non linear representation of the steam governor valves to the turbine, the dump valves and two groups of safety valves. The feed-water flow appears with its adequate magnitudes and temperatures at the outlet of the condenser and the inlet of the reactor vessel. The feedwater flow is controlled by the magnitude of the steam flow and the water level in the reactor vessel.

The main circulation pumps are simulated by one pump characteristic representing one or more pumps in parallel. As an option, pump-stop can be simulated.

Control and safety systems are included in the model and can easily be modified to fit different needs. By the studies described in this paper the three main control systems of the plant, namely the systems controlling the turbine generator power, the reactor vessel pressure and the water level in the reactor vessel are simulated on the analog computer.

The power is controlled by changing the void fraction in the reactor core. The void is changed by the coolant flow through the reactor core. The void reactivity coefficient is negative. Thus an increase of the coolant flow will cause an increase of the reactor power and vice versa.

The power control system (See fig. 2A) consists of a master controller, which handles the power error and a slave controller, which handles the pump speed error and acts on the main circulation pumps through a hydraulic-coupling. The hydraulic coupling controls a frequency converter which feeds the pump motors.

The pressure control (see fig. 2B) is performed by processing the pressure error in the controller to provide opening signals to the turbine valve.

The feed water control is done by a level controller and a flow controller in combination. The level controller treats the error of the water level in the reactor vessel. The flow controller treats the difference between the steam flow in the reactor outlet and the feed water flow in the reactor inlet. The sum of the output signals of the level controller and the flow controller feeds the slave controller which controls the hydraulic coupling of the frequency converter which feeds the feed water pump motors.

The model is programmed for the hybrid computer in Studsvik, consisting of a small digital computer (PDP-9) linked to a medium-sized analog computer (AD4 backed up by PACE 231R and AD2). The hardware configuration including the peripherals used is shown on Figure 3. The equations for the core are mainly solved digitally. For the solution of the equations describing the hydraulics of the cooling channel a hybrid method (continuous space-discrete time) is used. The remaining equations are solved on the PACE 231R and the AD2 using analog methods.

Most of the data required by the program are entered via the paper tape reader. The operator should, however, from the teletype enter some initial data, such as power level, main circulation flow and the initial axial distribution of either the control rods or the neutron flux and the position and relative number of rods which can be moved continuously.

The calculation of coefficients for the analog computers is performed by a separate digital program.

The initial axial distribution of power, control rods, temperatures and void within the core is computed digitally and is printed out on the teletype.

The operator starts and stops computation by means of teletype commands. In the same way it is possible to select one of the space dependent variables to be displayed on a CRT-display.

Perturbations are normally introduced at the analog computer. This provides a maximum of flexibility. During the calculation of transients a number of variables can be recorded on 8-channel recorders. The most essential of these variables are power level, pressure, temperatures in fuel and coolant, coolant flow, steam flow, feedwater flow and control variables. The space dependent variables can be displayed on CRT´s. Among those are the axial distribution of the power, fuel temperature, coolant temperature and void. When the dynamic run is completed, the operator can request a printout of initial and maximum/minimum values of a few important variables.

Application of the model

The program is primarily intended for studying transients in the second and minute time scale. The time scale is in these cases normally chosen such that the solution time is equal to real time. Very rapid transients may to some extent be studied by a change of time scale.

Typical transients which can be studied with the model are those caused by perturbations such as:

- movement of control rods
- changes in power load demand
- changes in the steam flow caused by changes of the steam valve setting or pipe rupture
- changes of pump speed in the main circulation loop
- changes in the feedwater flow
- changes in the coolant or feedwater temperature
- turbine trip and dumping
- failure of control systems such as power control system, pressure control system, feedwater control system.

The dynamic calculations can be started from steady state power levels ranging from a few percent of the nominal power to at least twice the nominal power. The initial system pressure may range from a few bar to well above the normal operating value. A suitable choice of scale factors will permit computation of both small and rather large transients.

The model can also serve as a powerful and flexible tool for the control system designer. The possibility for simulation in real time gives an instructive representation of the behaviour of a nuclear power plant. This makes the model attractive also for education and training.

Figure 4 shows the result of a fast ramp change (25 % in 10 secs) of the set point of the turbo-generator power in the model of the nuclear power plant with a preliminary optimized control system.

The recordings show the responses, of 16 variables at the same time.

The recorded signals are mainly control signals to the power control system, to the pressure control system and the feedwater control system.
The figure gives a good impression of the behaviour of the station.

The first three channels (fig. 4) show the changes of the neutron flux, the steam flow through the steam separators in the reactor vessel and the turbine generator power.
The five following variables (fig. 4) (channels No. 4 - 8) are those indicated in the block diagram of the power control system (see fig. 2A).

The changes of the reactor pressure as well as the turbine valve lift and the steam flow through the valve by the same disturbance of the set point of the generator power are shown in channels 9 - 11 and marked in fig. 2B.

The channels 12 - 16 show the transients of the variables in the block diagram of the feedwater control system (fig. 2C).

Figure 5 illustrates some system transients following reactor scram and safety valve opening initiated by a fast turbine stop when dumping to the condensor is not allowed. The delay time of the reactor shut down is conservative.

Figure 6 which shows photographs, taken at discrete time intervals, of the oscilloscope screen, illustrates the vertical distribution of the power and void in the reactor core during a control rod withdrawal (z) from top to bottom in 10 secs. The time written below the diagrams is the time after the start of the rod withdrawal. A comparison between the two pictures at the time t = 8 secs, without and with pressure control, shows the influence of the pressure control.

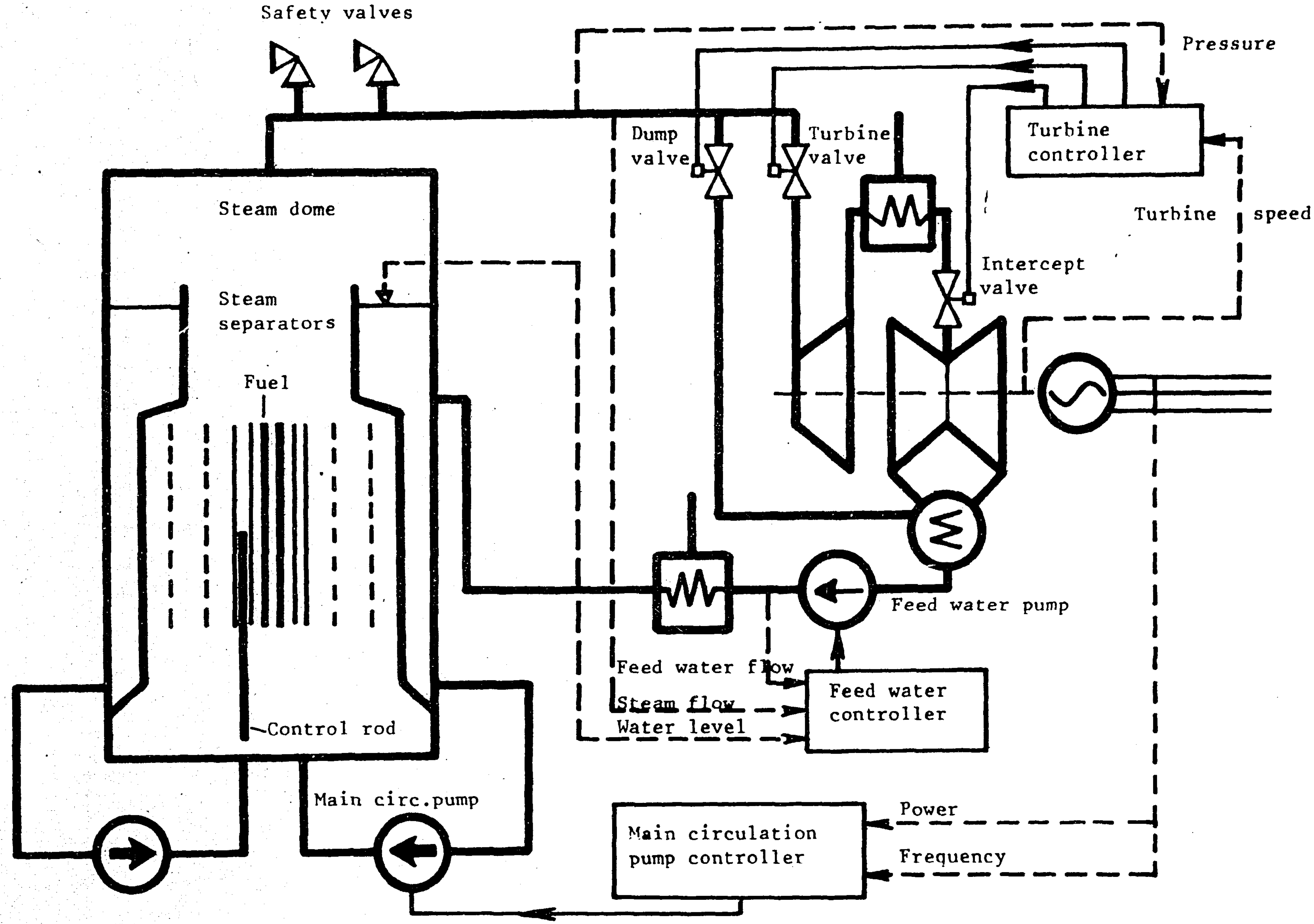

Figure 1. Principal Circuits of the Nuclear Power Plant

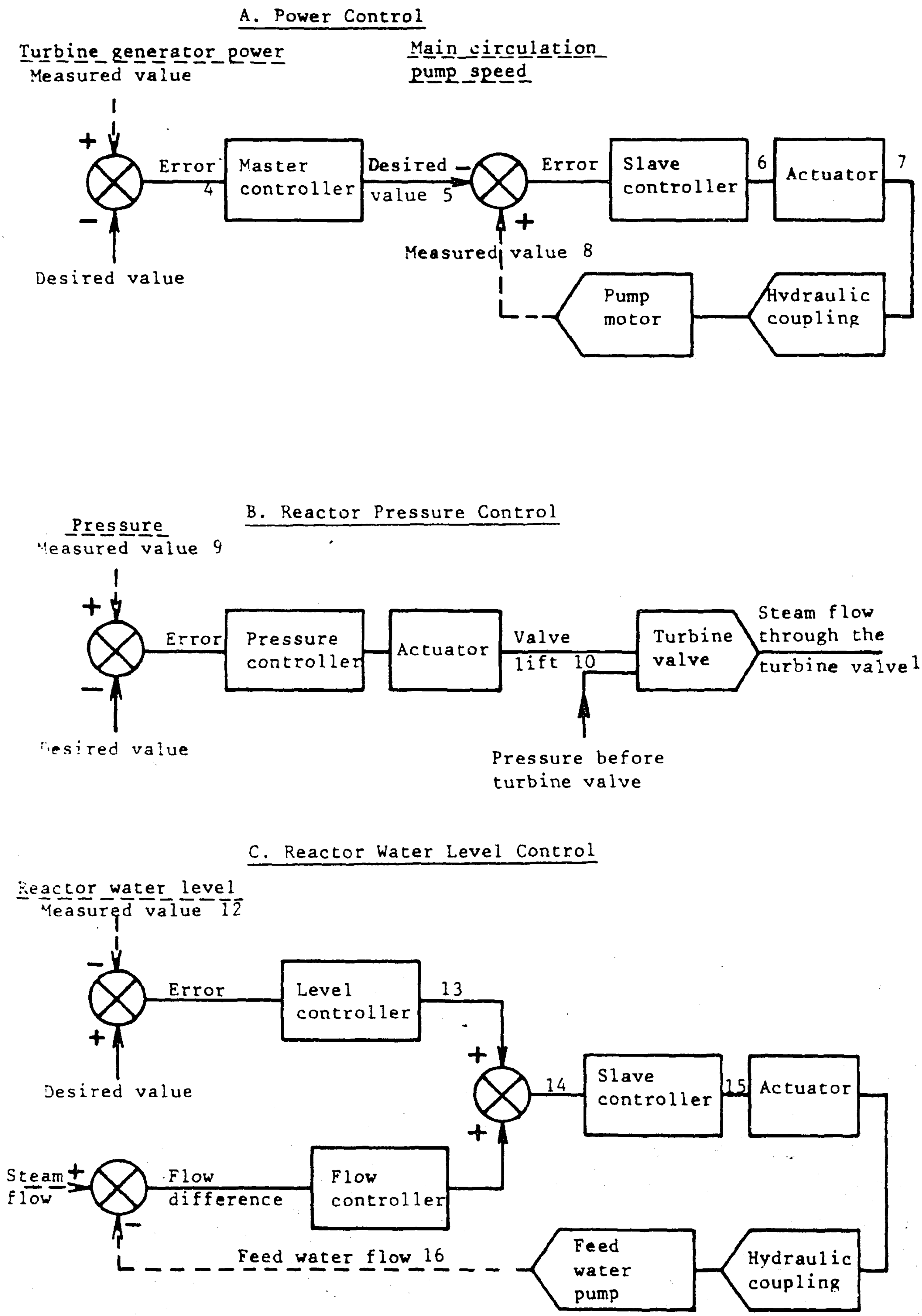

Figure 2. Simplified Block Diagrams of the Control System

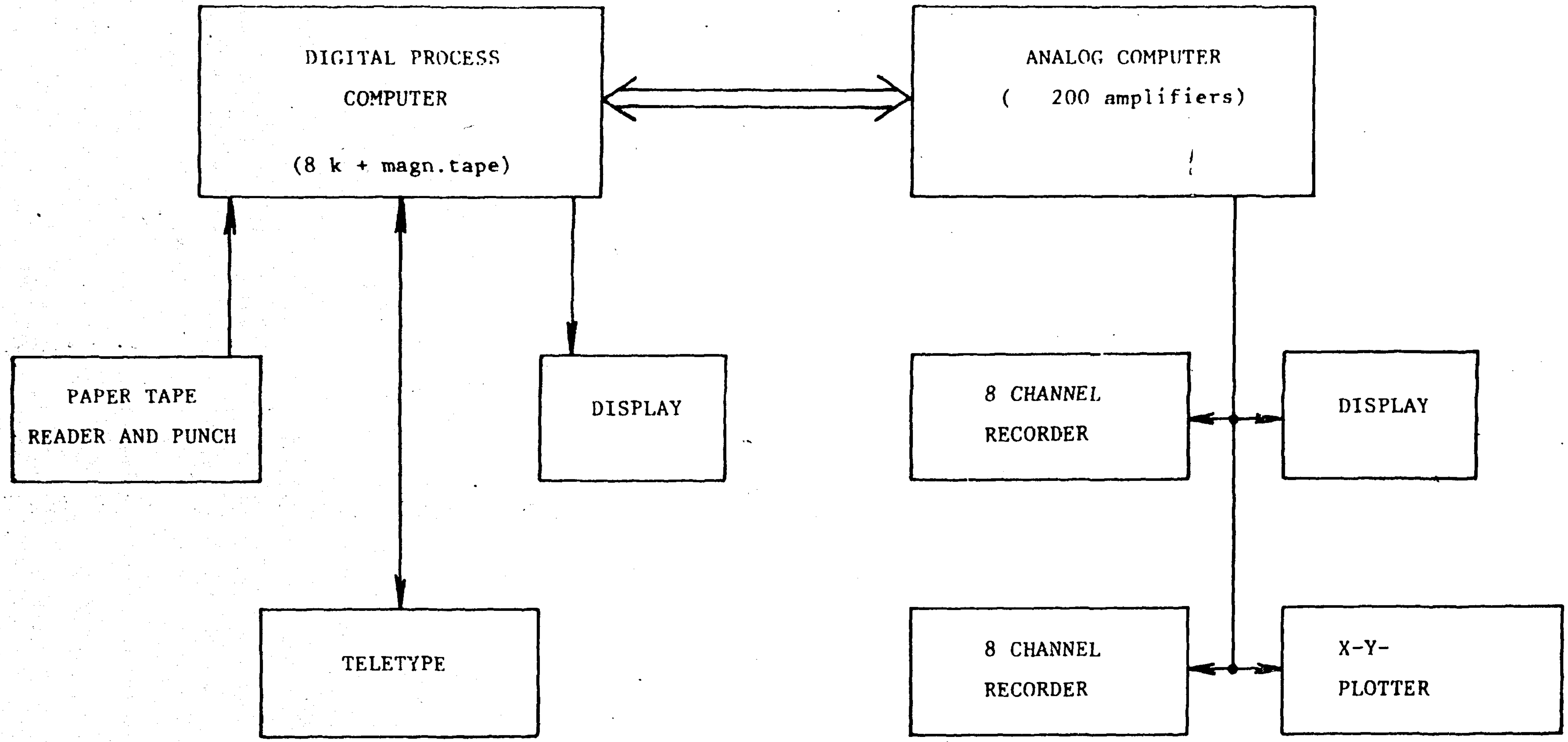

Figure 3. Hardware configuration for the simulator

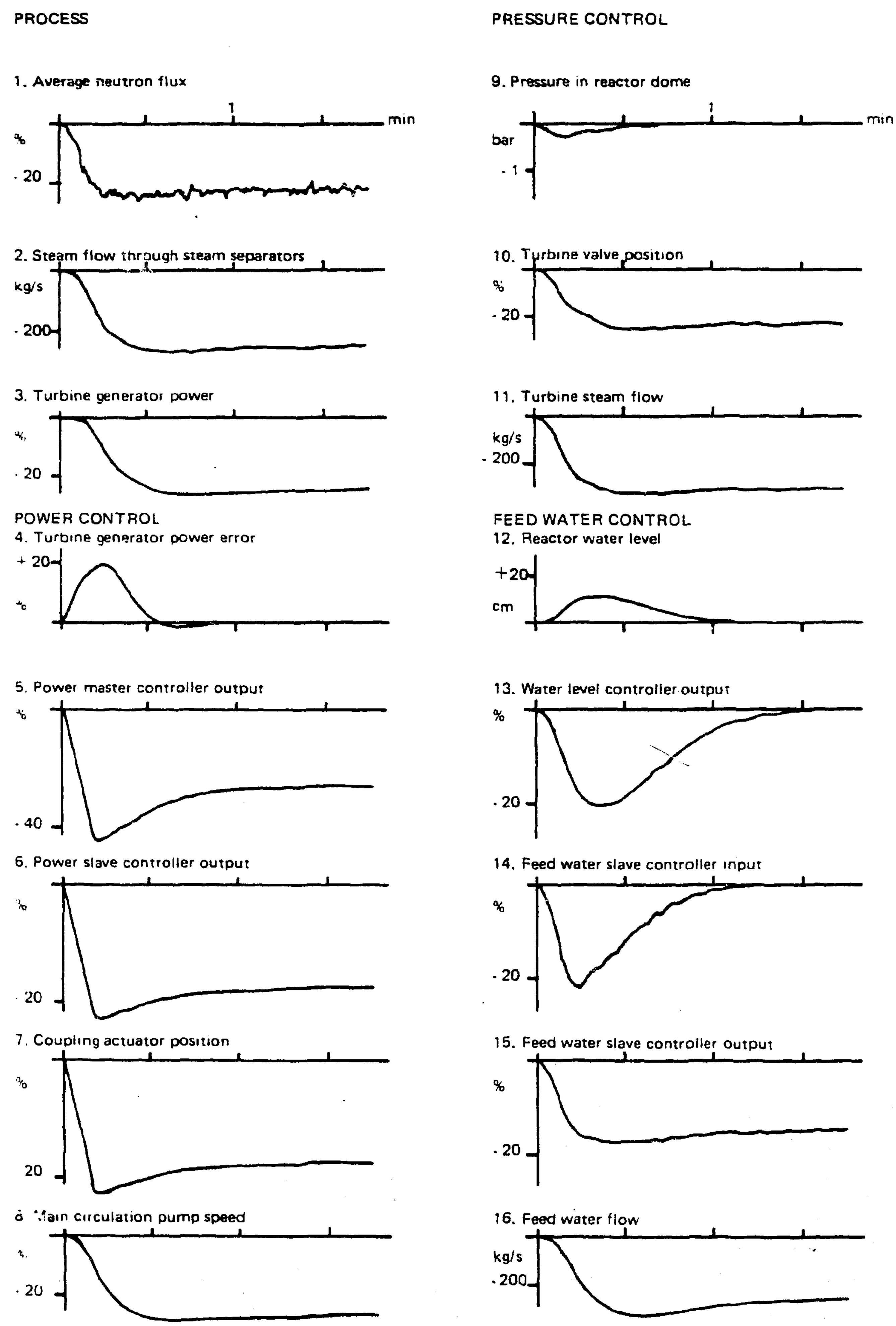

Figure 4. Response of control system variables to a linear change of the power demand from 100 to 75 % in 10 seconds

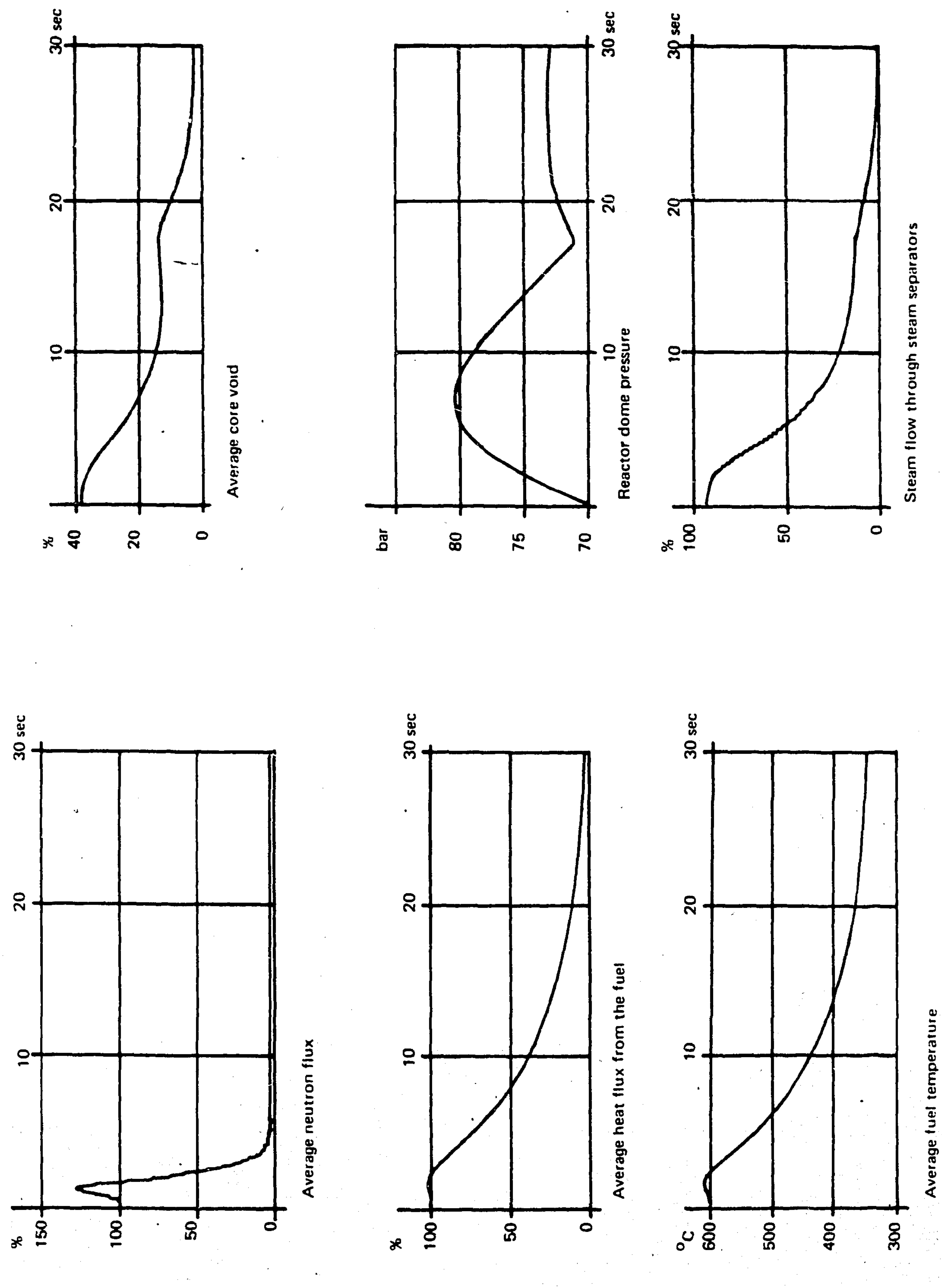

Figure 5. Shut down transients following a turbine stop initiating a reactor scram and reactor safety valve opening

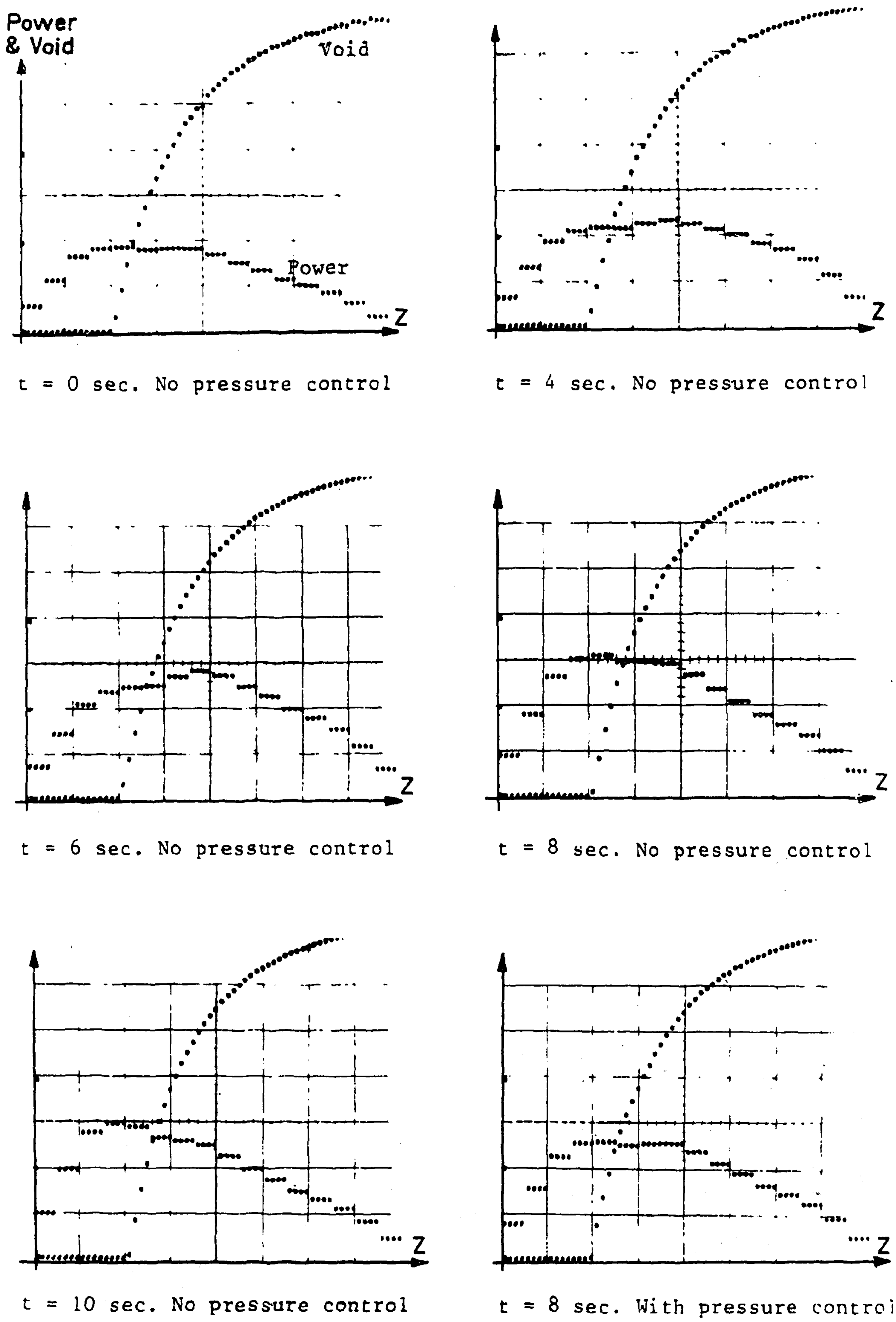

Figure 6. Vertical power and void distribution in the reactor core at discrete time intervals during rod withdrawal.