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CONTROLLABILITY STUDIES FOR AN ADVANCED CANDU BOILING LIGHT WATER REACTOR

by

R.M. LEPP and H.W. HINDS

Presented at the 1975 Summer Computer Simulation Conference,
San Francisco, July 21-23, 1975

Chalk River Nuclear Laboratories
Chalk River, Ontario

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Etudes de contrôlabilité pour un réacteur CANDU avancé à eau légère bouillante*

Par

R.M. Lepp et H.W. Hinds

*Rapport présenté au Congrès de la simulation sur ordinateur tenu à San Francisco du 21 au 23 juillet 1976. Tiré à part avec l'autorisation de Simulation Councils, Inc. LaJolla, California.

Résumé

On donne un aperçu des études de contrôlabilité d'ensemble effectuées dans le cadre de la conception d'un réacteur CANDU de 1200 MWe à eau légère bouillante employant comme combustible de l'U-235 ou de l'oxyde d'uranium enrichi de Pu.

On décrit le concept, les divers modèles développés pour sa simulation sur un ordinateur hybride et les perturbations employées pour mettre à l'essai la contrôlabilité du système. Les résultats montrent que ce concept aura une meilleure contrôlabilité d'ensemble que des réacteurs CANDU-BLW semblables alimentés en combustible avec de l'uranium naturel.

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Chalk River, Ontario, Canada

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ABSTRACT

Bulk controllability studies carried out as part of a conceptual design study of a 1200 MWe CANDU boiling-light-water reactor fuelled with U^{235}- or Pu-enriched uranium oxide are outlined.

The concept, the various models developed for its simulation on a hybrid computer and the perturbations used to test system controllability, are described. The results show that this concept will have better bulk controllability than similar CANDU-BLW reactors fuelled with natural uranium.

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ABSTRACT

Bulk controllability studies carried out as part of a conceptual design study of a 1200 MWe CANDU boiling-light-water reactor fuelled with U^{235}- or Pu-enriched uranium oxide are outlined.

The concept, the various models developed for its simulation on a hybrid computer and the perturbations used to test system controllability, are described. The results show that this concept will have better bulk controllability than similar CANDU-BLW* reactors fuelled with natural uranium.

INTRODUCTION

Canada currently has approximately 2500 MWe of nuclear generating capacity installed and operating as well as just under 10,000 MWe of domestic commitments at various stages of completion. All but one of the stations are of the CANDU-PHW** type which uses heavy water as the primary coolant to carry heat from the reactor core to heat exchangers where steam is generated to drive the turbine. The station which is different is the Gentilly-1 CANDU-BLW Station, located at Gentilly, Quebec, which uses boiling light-water as coolant.

Both concepts use 50 cm long bundles of natural uranium, sheathed in Zircaloy and installed in pressure tubes that traverse the reactor core. Each pressure tube is surrounded by a calandria tube which insulates the hot pressure tube from the surrounding cool heavy-water moderator. Both reactor types are refuelled on-power under remote control. The main difference between the two systems is that the BLW plant uses a direct cycle of steam from the core to drive the turbine whereas the PHW system has heat exchangers to produce steam for the turbine.

The simulation studies carried out in support of the design, commissioning and operation of the CANDU-PHW nuclear generating stations have been reported at earlier conferences [1, 2, 3, 4]. The simulation work carried out in Canada in support of the Gentilly-1, 250-MWe prototype BLW reactor has also been presented previously [5]. What has not yet been presented is the simulation work carried out on advanced concepts which are a natural progression of the CANDU system.

One such concept is a BLW reactor, larger in size than Gentilly-1 and fuelled with U^{235}- or Pu-enriched uranium oxide. To assess the potential of this concept, Atomic Energy of Canada Ltd. launched a study early in 1973. One ground rule for this study was that the reactor had to be inherently easier to control than Gentilly-1. Thus the objective of the bulk control studies was to determine how well the system maintained control during

- on-power refuelling
- disturbances originating in the heat-transport system
- changes in electrical demand, i.e. load-following

for both start-of-life and equilibrium operating conditions, with either U^{235}- or Pu-enriched uranium-oxide fuel.

This paper describes the work associated with the first two of the above items, i.e. the load-following study is not discussed here.

THE CONCEPT

The concept examined is a 1200 MWe design which operates on either a Pu- or a U^{235}-enriched fuel cycle, designated CANDU-BLW(PB)-1200. A simplified station schematic is shown in Figure 1. The reactor core is vertically oriented with 872 pressure tubes containing the fuel bundles [6].

During operation, ordinary water enters the bottom of the pressure tubes, cools the fuel as it flows upward and leaves the top as a two-phase mixture. From there, the two-phase mixture travels to two steam drums where the steam is separated from the water by cyclone separators. The saturated steam drives a turbine-generator set which delivers power to the electrical grid. The water removed by the separators collects at the bottom of the steam drums from where it is circulated again through the reactor core. The steam removed from the primary circuit is replaced by feed-water returning from the turbine condenser.

The feedwater enters the primary circuit at the reactor inlet header. This is in contrast to Gentilly-1 which has the feedwater entering the steam drum. The advantages of the former are, simpler emergency-core-cooling system and smaller primary coolant pumps. The disadvantages are, more difficult drum-level control and a greater sensitivity to feedwater-train disturbances. Feedwater flow is regulated to maintain a constant water level in the steam drum.

The mass flow of steam to the turbine is regulated by a turbine throttle-valve. In a turbine-following-reactor nuclear generating station the position of this valve is regulated to maintain a constant pressure in the steam drum. In parallel with this valve is a bypass valve, which regulates the flow of steam to the turbine condenser. During normal operation this valve would be completely closed. However, if the reactor were producing more steam than the turbine could handle the bypass valve would open. Similarly, it would open in the event of a turbine trip.

After passing through the turbine the steam is condensed in the turbine condenser. This low-pressure condensate is pressurized and then heated by feedwater heaters on its way to the core inlet header.

THE SYSTEM MODEL

A detailed small-signal simulation of the concept was assembled on the Chalk River Hybrid Simulator for both frequency- and transient-response analysis. A block diagram of the regulating-system model is shown in Figure 2.

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*CANada Deuterium Uranium, Boiling Light Water cooled
**CANada Deuterium Uranium, Pressurized Heavy Water cooled

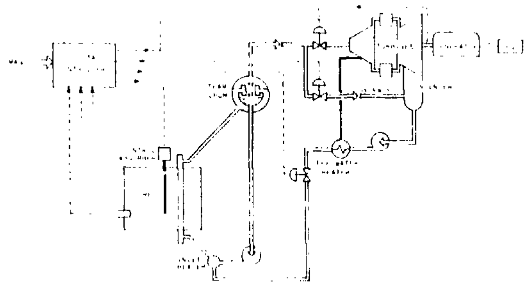

FIGURE 1 - CANDU-BLW(PB)-1200 SCHEMATIC

In this model the reactor core is represented by a point-kinetics model with 6 delayed-neutron groups, a 3-shell fuel model and a single coolant channel. As shown in Figure 2, the internal reactivity feedback varies in response to changes in fuel temperature and coolant density. The latter is simulated in the heat-transport-circuit model as a function of four variables, i.e. changes in

- power-to-coolant
- coolant pressure
- coolant inlet enthalpy, and
- coolant mass flux.

A block diagram of the heat-transport circuit, divided for simulation purposes, is shown in Figure 3 and described in Appendix A. Also shown are the steam-drum-level controller and the pressure controller. The entry points for the various expected heat-transport-circuit disturbances are also indicated.

The plant model was almost entirely simulated on the analog portion of our hybrid facility, 3 EAI TR-48 analog computers. Only the transport delays were simulated in the digital computer, a DEC PDP-8. The controller model, indicated in Figure 2, can be simulated on either the analog or the digital portion of our facility and we have done both. In existing CANDU nuclear generating stations the major control functions are carried out by digital computers. Therefore, our hybrid simulations are more realistic if those station control functions which are carried out digitally are also simulated digitally. This is not always possible with our present hybrid facility. Therefore, we have taken care to ensure that the effects of 'signal conditioning' and 'sampling rates' were considered when the analog computer was used to simulate a digital-control function.

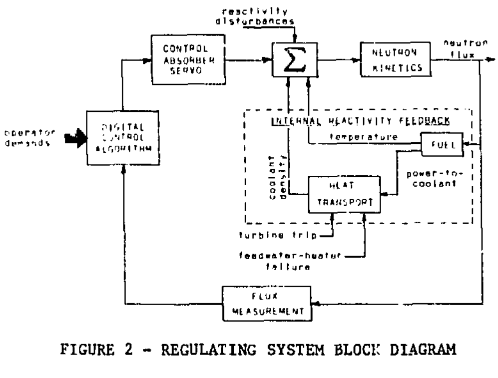

FIGURE 2 - REGULATING SYSTEM BLOCK DIAGRAM

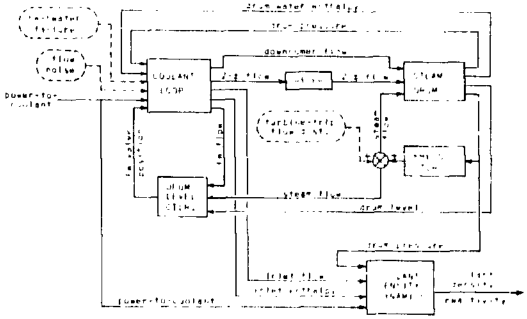

FIGURE 3 - HEAT-TRANSPORT CIRCUIT BLOCK DIAGRAM

A simpler model than that described above was programmed into our BODE/NYQUIST PLOTTER code [7] to obtain the open-loop transfer functions for the system. In this model the reactor was simulated by point kinetics with two delayed-neutron groups, the fuel by a first-order lag, and the reactivity feedback effects due to fuel temperature and coolant density by a power coefficient [8]. The neutron-flux controller, the signal conditioning and the sampling rate were also included.

This simpler model was used to arrive at neutron-flux-controller parameters which gave good system-flux control. The power coefficient used was derived from the best estimates of reactivity effects determined by large digital reactor-physics codes. Subsequently, the model was used to determine system open-loop frequency response for a range of feedback power coefficients. In this way, the small-signal stability of the system was determined as a function of power coefficient. Because of the uncertainty in the reactor-physics calculation of reactivity effects, it is important that the system open-loop stability characteristics (phase and gain margins) be relatively insensitive to power coefficient.

SYSTEM CONTROLLERS

Unlike Gentilly-1, the BLW(PB) concept does not have primary coolant-flow control. Instead, the flow of primary coolant through the core varies depending on the pressure drops around the circuit.

The main control loops examined in this simulation were

- neutron-flux control
- steam-drum-pressure control, and
- steam-drum water-level control.

The sampling period assumed for neutron-flux control was 0.064 seconds, which is the sampling period currently used in the Gentilly-1 plant.

The control loops were treated separately even though there are strong interactions between them because of the direct-cycle nature of the plant. Also, the individual controllers were programmed as if the station were a turbine-following-reactor unit. This means that the station can be operated either at a steady power or it can be used for daily load-cycling. The controllers used would not enable the plant to carry out the other forms of load-following, namely fringe control or spinning-reserve duty.

The objectives of our work did not include

- developing a co-ordinated control system using multivariable or optimal control theory

- studying all aspects of load-following, including its effects on plant dynamics.

These are natural progressions of the work done to date [9].

The controller parameters were chosen to give satisfactory control in response to perturbations at high power (50-100%). It is assumed that, if the concept can be controlled well at these powers, control at low power will not be a problem. In other words, any control peculiarities at low power should be readily overcome. Based on the experience at Gentilly-1 [5], it is likely that the drum-level controller, satisfactory at high power, will not be satisfactory at low power, because of a 'non-minimum phase' effect. This problem was overcome at Gentilly-1 by adding a feedforward loop to the controller, using rates-of-change of reactor power and primary coolant flow.

Neutron-Flux Controller

Neutron flux is measured by ion chambers installed at the edge of the core. The error between the desired and actual neutron flux is used in a proportional-plus-rate control algorithm, the output of which is a signal that moves the solid control absorbers at a velocity proportional to the signal strength. Therefore we have

$$\frac{\delta k_c}{\Delta\phi} = \frac{K}{s}\left[K_1 + \frac{K_2 s}{(1 + \tau_1 s)(1 + \tau_2 s)}\right] \qquad (1)$$

where δk_c = control reactivity (mk)

$\Delta\phi$ = neutron flux error (% of actual flux)

K_1 = controller proportional gain (cm/%.s)

K_2 = controller rate gain (cm.s/%.s)

K = absorber worth (mk/cm)

Drum-Pressure Controller

It was assumed in the simulation that the station would be of the turbine-following-reactor type. This means that reactor power does not change in response to changes in demand on the grid. Instead, the reactor power follows the operator demand and the output of steam to the turbine is regulated to maintain the steam-drum pressure at its setpoint. A proportional-plus-reset controller, operating on the drum-pressure error signal, is used to accomplish this. Consequently,

$$\frac{\Delta W_s}{\Delta P} = K_3 + \frac{K_4}{s} \qquad (2)$$

where ΔW_s = change in steam flow (kg/s)

ΔP = pressure error (MPa)

K_3 = controller proportional gain (kg/MPa.s)

K_4 = controller reset gain (kg/MPa.s^2)

Drum-Level Controller

The level of water in the steam drum is controlled by changing the flow of feedwater into the core inlet header. The inputs used by the controller are the
- difference between steam flow and feedwater flow, and
- water-level error in the drum.

The controller transfer function is

$$\Delta X = K_5 \,(W_s - W_{FW}) + \left(K_6 + \frac{K_7}{s}\right)\Delta L_D \qquad (3)$$

where ΔX = change in feedwater-valve lift (% of full stroke)

W_s = steam flow (kg/s)

W_{FW} = feedwater flow (kg/s)

K_5 = flow-error proportional gain (%.s/kg)

ΔL_D = drum-level error (cm)

K_6 = drum-level proportional gain (%/cm)

K_7 = drum-level reset gain (%/s.cm)

SYSTEM TESTING

The detailed transient-response model described above was used to assess the control system's ability to regulate
- neutron flux
- steam-drum water level, and
- steam-drum pressure

in response to
- step reactivity perturbations
- turbine-trip pressure transients, and
- feedwater-heater failures.

These were considered to be representative of the worst disturbances one could expect in the actual plant.

Reactivity Disturbance

The first tests carried out in the simulation were step changes in reactivity. These, although perhaps not as realistic as ramp reactivity changes, for example due to on-power refuelling, are useful for controllability comparisons between different
- reactor concepts
- core power levels, and
- fuel burnups

The maximum reactivity step the system can tolerate is, for our purposes, defined as that which causes certain system parameters to approach their reactor-trip settings, i.e.
- a neutron flux increase of 10% above the equilibrium level, or
- sustained maximum control-absorber velocity in one direction for five seconds,

whichever comes first.

Turbine-Trip Transient

The second series of tests was aimed at simulating the consequences of a turbine-trip pressure transient. The sequence of events following a turbine trip are best explained with the aid of Figure 4.

A turbine trip generates a signal that rapidly closes the stop valve to cut off the flow of steam to the turbine. This in turn causes a rapid pressure rise in the steam drum. The pressure controller responds to the pressure-error signal by opening the valve to bypass steam to the condenser. The coolant density, and hence the core reactivity, is affected by the pressure transient such that control action is required to bring the neutron flux back to its setpoint value. The control system must be able to do this without the reactor exceeding the trip settings given above. In Gentilly-1, a '10% capture rule' has been programmed

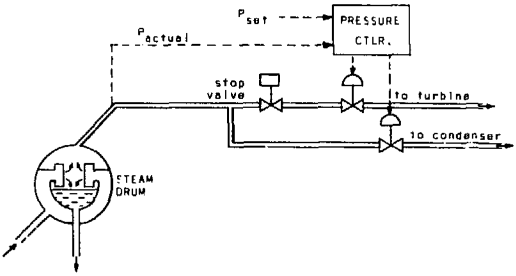

FIGURE 4 - PRESSURE CONTROLLER FOR CANDU-BLW(PB)-1200 AS A 'TURBINE-FOLLOWING-REACTOR' STATION

into the digital flux controller such that when the neutron flux decreases from 100% to 90% (or less) the flux setpoint is also reset to 90%. This has the effect of decreasing parameter overshoots (e.g. neutron flux, control-absorber position) following a turbine trip. Such a 'capture rule' was not included in the simulation presented here.

The turbine-trip was simulated, as shown in Figure 2, by a step decrease in steam flow to the turbine.

Feedwater-Heater Failure

The purpose of the third series of tests was to determine the effects of a feedwater-heater failure on system control. There are five feedwater heaters in the system, which raise the temperature of the feedwater by ∿ 155°C, using bleed steam from the turbine. A feedwater-heater failure could be the sudden unexpected closure of the valve in any of the bleed-steam lines. This would cause a rapid drop in feedwater temperature and, in turn, a rapid drop in core inlet enthalpy, resulting in a decrease in core reactivity, due to an increase in coolant density.

The feedwater-heater failure was simulated by a step decrease in feedwater enthalpy (see Figure 2).

SIMULATION TESTS AND DISCUSSION OF RESULTS

All the transient-response tests were carried out in real time. Eight output variables were recorded on strip-chart recorders during each test. Of these, only the responses of the five most important variables are shown in the figures which follow.

Response to Step Reactivity Disturbance

The system response to a positive reactivity step is shown in Figure 5. The size of the reactivity step was chosen so that the neutron-flux error would approach its 10% overpower trip setting. As seen from the response curves, the control absorbers travel at their maximum velocity in one direction for ∿ 3 seconds and then rapidly come to a stop after a few small oscillations. These oscillations can be attributed to reactivity feedback effects that return to the core after having travelled around the heat-transport circuit. Control of drum pressure and drum level are also satisfactory, as evidenced by the response curves.

It was found that the plant became more sensitive to this disturbance with increasing fuel burnup. This is as one might expect because the maximum tolerable reactivity step depends primarily on the core-kinetics gain which increases with burnup since the delayed-

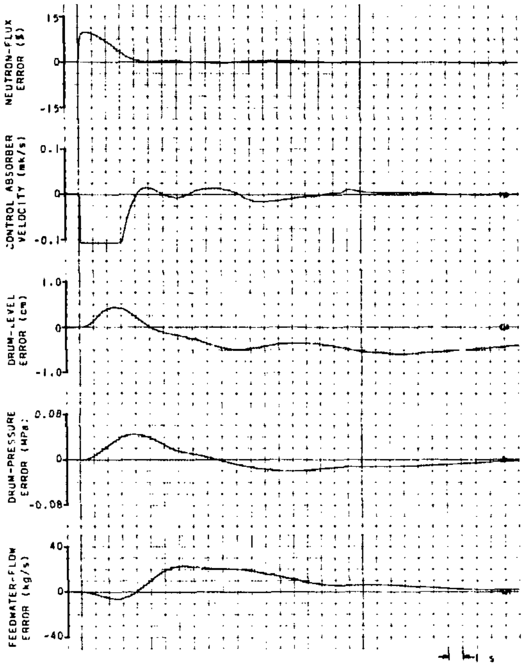

FIGURE 5 - TRANSIENT RESPONSE OF CANDU-BLW(PB)-1200 TO A 0.4 mk REACTIVITY STEP

neutron fraction decreases. The core feedback reactivity in this case is relatively unimportant because the system overpower trip level is approached within a fraction of a second of the disturbance.

Response to Turbine-Trip

Typical system response to a simulated turbine-trip pressure disturbance (i.e. step change in steam flow) is shown in Figure 6. The disturbance used was the maximum tolerable without exceeding the overpower trip level. The resulting pressure increase is then a measure of the system tolerance to this disturbance. For this reactor concept we found that, for the worst case, an overpressure of almost 0.4 MPa could be tolerated. As expected, those core loadings having the smallest coolant-density reactivity coefficient gave the best results.

It should be noted that, in current CANDU reactors, an overpressure of 0.2 MPa causes the suppression-tank valves to open. Therefore, it would probably be a requirement that the pressure-control system be good enough to ensure that a turbine-trip produce a pressure transient less than 0.2 MPa.

Response to Feedwater-Heater Failure

The transient response of the system to a sudden feedwater-heater failure is shown in Figure 7. This disturbance is simulated by perturbing the heat-transport circuit model with the appropriate step decrease in feedwater enthalpy. The parameter responses in Figure 7 show that this disturbance causes the system to approach its '5-second control-absorber saturation-velocity' trip setting.

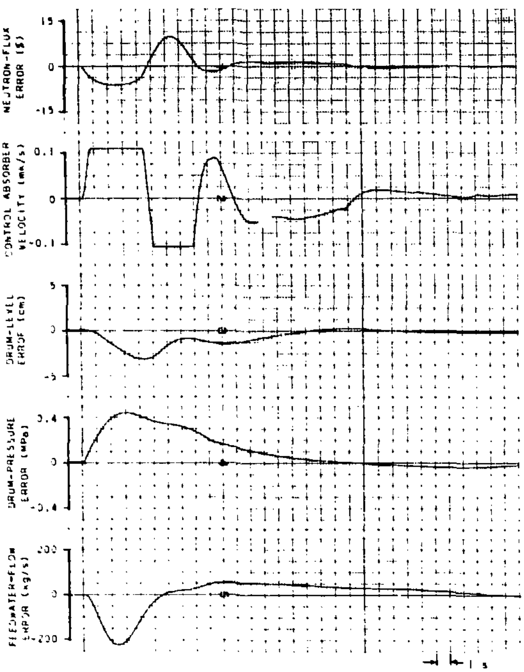

FIGURE 6 - TRANSIENT RESPONSE OF CANDU-BLW(PB)-1200 TO A TURBINE-TRIP PRESSURE DISTURBANCE

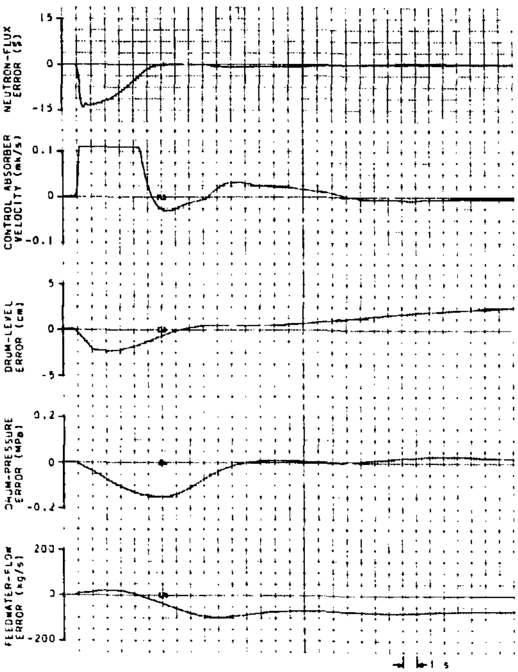

FIGURE 7 - TRANSIENT RESPONSE OF CANDU-BLW(PB)-1200 TO A FEEDWATER-HEATER FAILURE

The test results showed that the core with the highest coolant-density reactivity coefficient was initially unable to withstand this disturbance without the reactor tripping. Changes to the control system however, brought it closer to the desired objective.

In the initial analysis, the thermal time constant of the feedwater heater was neglected. When this time constant was incorporated into the simulation there was a significant decrease in the severity of the transient. Nevertheless, it is clear from the analysis that the failure of a single feedwater heater is a large disturbance to the system. This is due to the strong coupling between the feedwater train and the reactor core.

CONCLUSIONS

The simulation techniques currently being used to analyze the bulk controllability of new CANDU reactor concepts have been described. The conclusions drawn for the particular concepts described here are as follows:

(1) The bulk controllability of the enriched CANDU boiling-light-water reactor will be better than the Gentilly-1 natural uranium version of this concept, even without primary coolant-flow control and with feedwater entering at the inlet header.

(2) The most severe heat-transport circuit disturbance envisaged for this concept is the sudden failure of a single feedwater heater. Three of the four reactor system concepts studied were able to withstand this disturbance without tripping. The fourth system would only survive the disturbance if the '5-second control absorber saturation velocity' trip were raised.

(3) In this study, the steam-drum pressure was controlled by manipulating the turbine-throttle valve, the principle used in turbine-following-reactor plants. Further model development is planned to evaluate the concept from the point of view of 'fringe control' and 'spinning-reserve' duty during load-following operation.

It should be noted that a complete controller parameter optimization was not carried out. This would be done, should it be decided to pursue the concept further.

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APPENDIX A

THE PLANT MODEL

The major components in the nuclear steam-supply system are included in the simulation. However, the turbine and the feedwater train are not modelled. Instead, these are treated as sources of external disturbances to the heat-transport system (i.e. turbine-trip pressure transient and feedwater-heater failure). A brief description of the various models used in the study is given below.

Reactor Kinetics

The reactor kinetics are represented by a point model with six delayed-neutron groups.

Fuel Model

The fuel is represented by an equivalent fuel element made up of three, equal-thickness, concentric cylinders, clad in a metal sheath. The sheath is treated as a pure thermal resistance. The outputs from the model are 'mean coolant temperature' and 'power-to-coolant'.

Heat-Transport Circuit

The heat-transport circuit is shown in Figure 3 and consists of models of the
- coolant loop
- riser
- steam drum
- coolant-density dynamics
- drum-pressure controller
- drum-level controller.

In the coolant-loop model, pressure-drop equations are used to derive the instantaneous flows in the down-comer and in the feedwater line. These are summed, according to the conservation-of-mass equation to give the core inlet flow. The inlet enthalpy is obtained using the conservation-of-energy equation and a digital transport delay. These quantities, along with power-to-coolant and drum pressure, are used in pressure-drop equations, to obtain the inlet-header pressure.

The steam and water flows at the fuel-channel outlet are found from the conservation equations for energy and mass. In this model it is assumed that there is no change in mass storage in the core. Consequently, any change in core inlet conditions causes an instantaneous change at the core outlet.

The riser, however, is assumed to have mass storage. The model differential equations are obtained from conservation of mass and energy and the assumption that the riser is an adiabatic duct, at constant pressure. A non-minimum phase effect occurs in the riser; an increase in steam flow into the riser causes an instantaneous temporary increase in water flow out of it. (This same non-minimum phase effect would occur in the fuel-channel model had we allowed for mass storage.) Propagation of the void-fraction change from riser inlet to outlet is achieved using a digital transport delay.

The mathematical model of the steam drum is based on the assumption of no heat transfer by conduction between the steam and water phases. Heat transfer occurs only due to spontaneous condensation and/or flashing and then at rates sufficient to prevent the formation of supersaturated steam or superheated water. Detailed derivations of the drum equations used in the simulation are given in a companion paper by Moeck and Hinds [10].

Changes in coolant density in the core affect neutron absorption and hence neutron flux. These changes in coolant density are caused by perturbations in the heat-transport circuit, i.e. changes in
- power-to-coolant
- coolant pressure
- coolant inlet enthalpy, and
- coolant mass flux.

The steady-state sensitivity of coolant density to each of these variables is calculated using a digital code. However, the transient response of density to these parameters is more difficult to calculate. We are currently using a code developed by Hancox and Nicoll [11] to calculate three of these transfer functions (effect of pressure on density is assumed instantaneous). For simulation purposes we fit a simpler transfer function to the results from their code.

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