EXPERIENCE IN MODELLING THE ZAPOROZH'YE NUCLEAR POWER PLANT USING RELAP5

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INTRODUCTION

Unit 5 of the Zaporozh'ye Nuclear Power Plant (ZNPP5), equipped with a VVER-1000/320 4-loop reactor, has been modelled in detail using the RELAP5/MOD3.2 thermal-hydraulic system code (Ref. 1). The 4-loop model affords a fidelity with ZNPP5 in terms of the system geometry such as the point of emergency core cooling (ECC) injection, for example. Both the reactor vessel and steam generators were nodalized in a quasi 3-dimensional (3-D) fashion thus allowing to capture asymmetric effects in the main reactor system components and realistic heat transfer distribution in the steam generators.

Besides its use for accident analysis, the present model is intended to closely simulate operational events such as pre- and post calculations of anticipated transients and tests. At present, the model is used to support justification of the new - symptom-oriented - set of emergency operating instructions.

ZNPP5 makes use of both digital and analog controls. They have been modelled in the RELAP5 model allowing to analyse in detail workings of various plant equipment. The present model was validated using three ZNPP5 transient events.

(QUASI) 3-D MODELLING APPROACH

REACTOR

The reactor model, shown in Figure 1, consists of several parts based on several flow paths through the reactor vessel:

- 28 in four segments of the annular downcomer;
- 20 in four segments of the lower plenum, including the lower core support region with the perforated support tubes;
- 80 in four core segments, including 40 in 4 "hot" fuel assemblies and 40 in 4 "average" core sectors;
- 3 in the core bypass, including 1 in the flow paths through the baffle cooling channels and the gap between the baffle and the barrel, and 2 in the flow paths through the assembly central and guide tubes;
- 38 in the upper plenum, including the upper core unheated length and the upper head volumes;
- 6 in the upper plenum bypass, including the flow paths through the protective tubes of the control rods, temperature and neutron flux measurement channels; the bypass between the inlet and outlet nozzle regions is modelled by junctions.

When modelling the reactor vessel 3-D capabilities of the RELAP5 code were widely used in such regions as the downcomer, lower plenum, core and upper plenum. The vessel model is based on four communicating segments consistent with the locations of the inlet and outlet loop legs. Such nodalization allows to model adjacent and opposite loops and to analyse the asymmetric loop behaviour in transients with partial main circulation pumps work, secondary side break, ECC injection, etc. The nodalization of the upper part of downcomer allows taking into account ECC flow bypass through a break in the cold leg.

The core is modelled by eight parallel channels representing four equal sectors. Each sector has 1 ''hot'' fuel assembly and rest of "average" assemblies. In the "hot" assembly two types of heat structures are present for the hottest fuel pin and average fuel pin. The model takes into account flow mixing between the channels using cross-flow junctions.

Using "artificial" boundary conditions, the reactor vessel model was tested with various perturbations of flow and pressure. It showed acceptable stability.

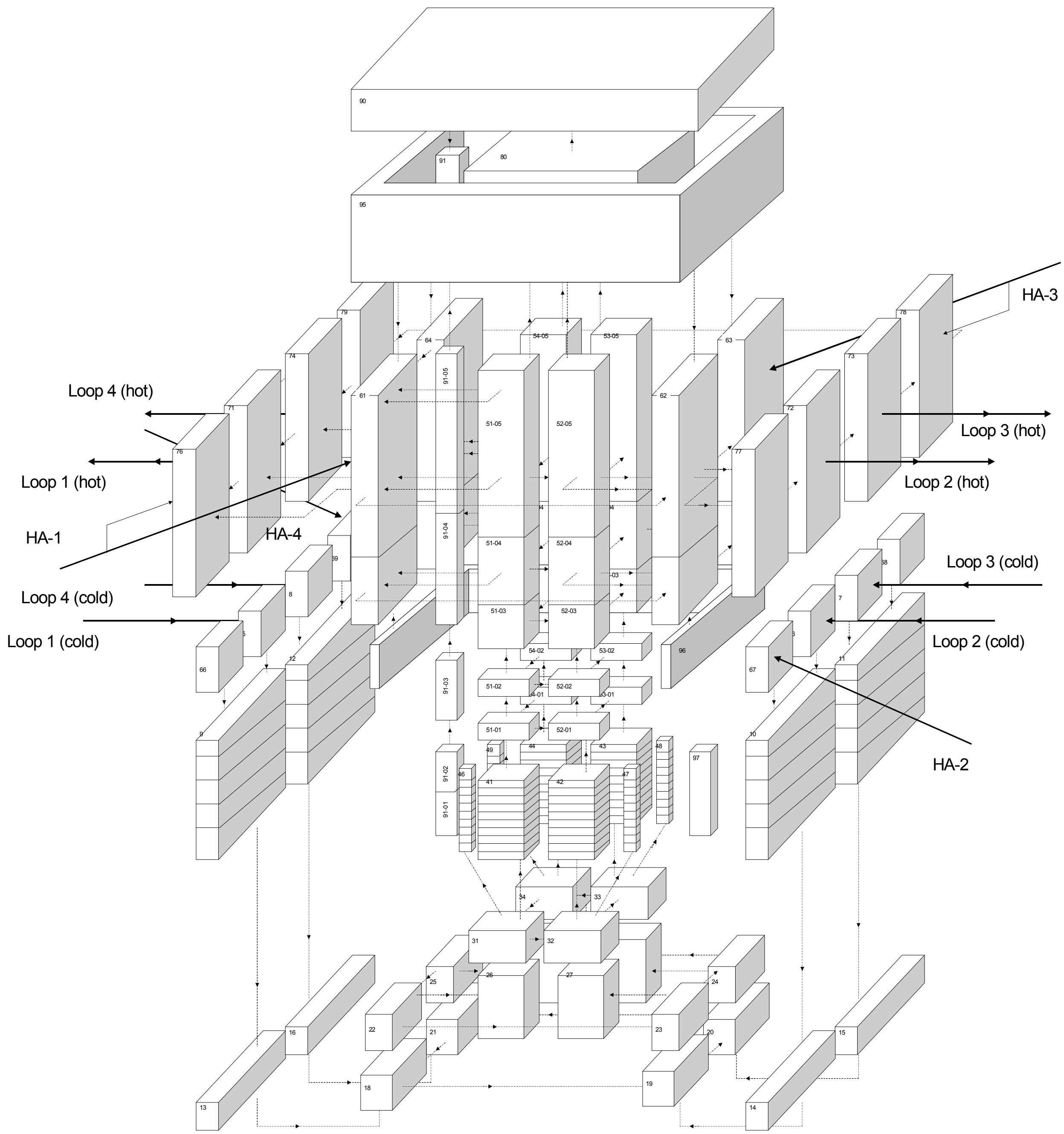

Figure 1 Reactor nodalization

STEAM GENERATOR

The horizontal steam generator (HSG) PGV-1000M is the most "uncomfortable" component of the plant for modelling by the thermal-hydraulic codes like RELAP5. The principal difficulty consists in modelling of the real physical phenomena of the HSG secondary side (Ref. 2). To better analyse the thermal-hydraulic behaviour of the HSG and plant systems response to changes of the HSG parameters the 3-D capability of RELAP5 was applied to secondary side modelling. The HSG model, shown in Figures 2 and 3, consists of the primary and secondary models.

The HSG primary side includes 78 volumes:

- 60 in five layers of the tube bundle. Each layer consists of two 6-cell pipes;
- 9 in the “hot” collector and 9 in the “cold” collector.

The HSG secondary side includes 59 volumes:

- 20 in four segments around the tube bundle;
- 20 in four segments of the bypass region between the HSG vessel and the tube bundle;
- 5 in the bypass region in the centre of the tube bundle;
- 8 in the bypass region between the HSG vessel and the submerged perforated sheet;
- 1 in the region between the submerged perforated sheet and the tube bundle;
- 5 in the steam dome of the HSG.

The HSG secondary side quasi 3-D approach was used in all HSG regions with the exception of the steam dome region. The tube bundle region is divided into 4 communicating segments. Each segment is subdivided in vertical direction into 5 layers. Such nodalization allows more realistic modelling of the spatial heat and void fraction distribution on the secondary side. The bypass regions are connected with the tube bundle regions by cross-flow junctions to facilitate natural circulation. The important feature of this approach is the capability to determine and analyse the steam generator level in different parts of the HSG secondary side. It allows to model the setpoints which depend on the water level in the “hot” and “cold” collector regions and in the bypass region. These setpoints determine trips of the main coolant pumps, feedwater pumps, turbine stop valve closure, scram, etc. An accurate determination of the HSG level is also important for the accurate operation of the feedwater controllers. The steam dome region is modelled from point of view the level measuring devices connecting and the location of the separator.

Using “artificial” boundary conditions, the HSG model was tested with various perturbations of flow and pressure. It showed acceptable stability.

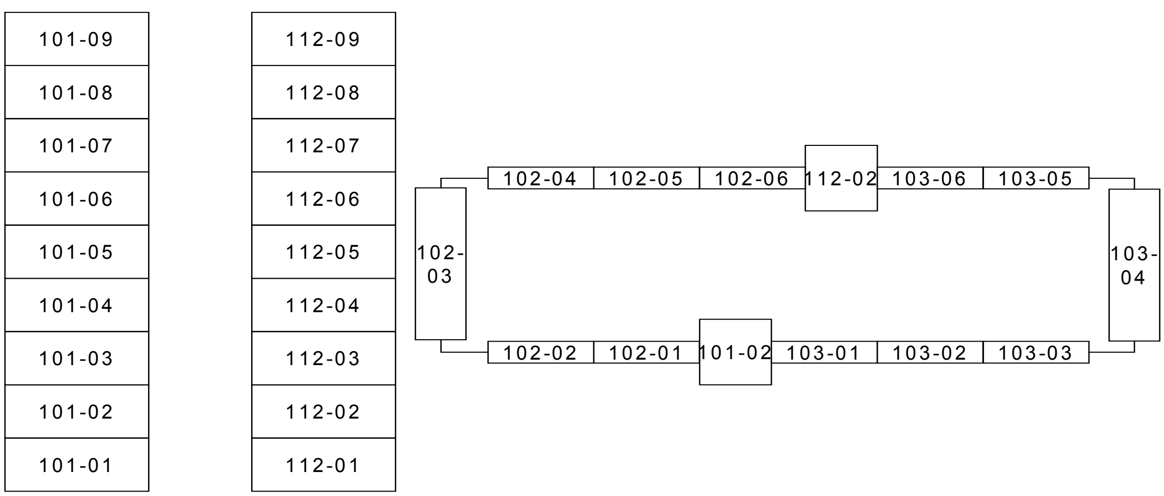

Figure 2 Steam generator primary side nodalization (collectors and tube bundle layer)

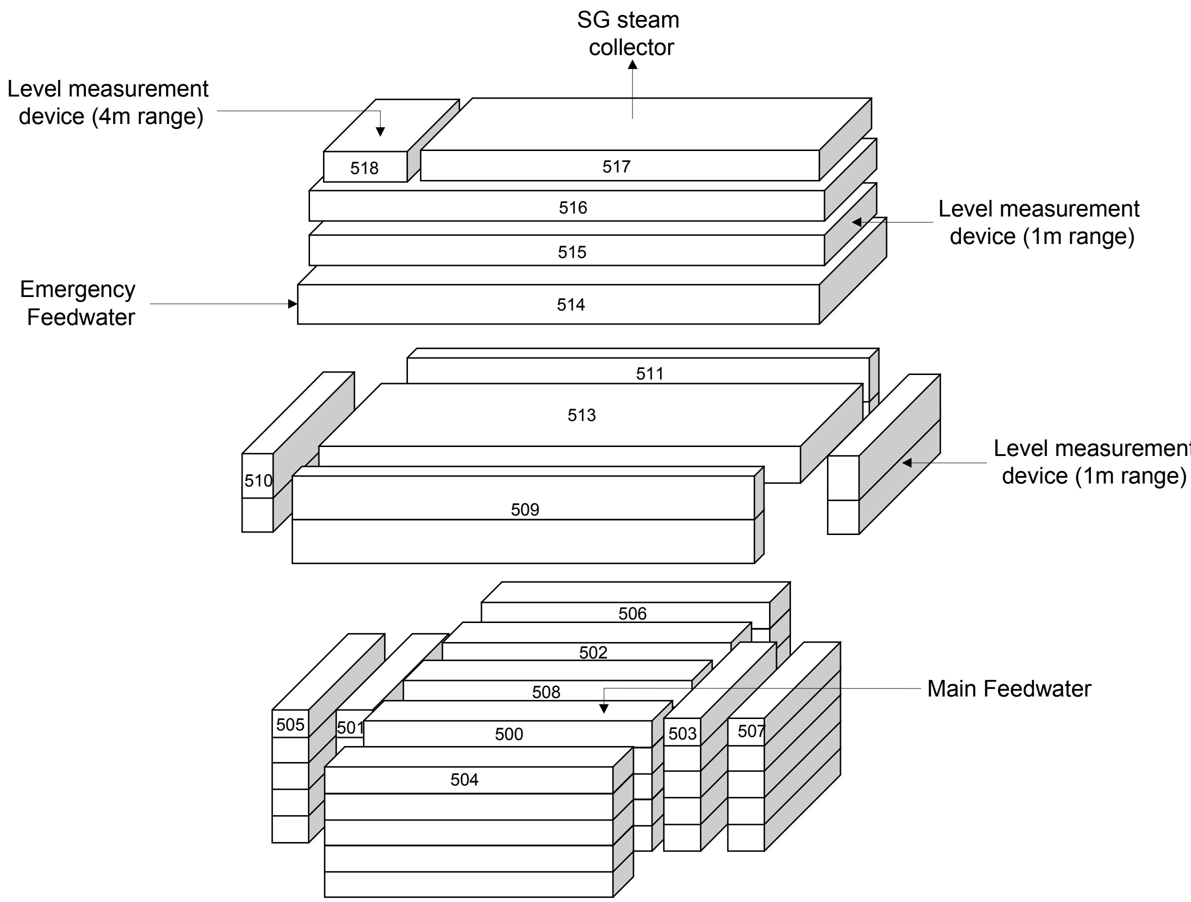

Figure 3 Steam generator secondary side nodalization

CONTROLLERS MODELLING

The following control models, representing ZNPP5 controllers, were developed and implemented:

- reactor power control and protection system model;
- model of make-up and let-down system with pressurizer level controller;
- feedwater controllers model;
- main feedwater pump controller model;
- model of electro-hydraulic turbine control system;
- BRU-A (steam dump to atmosphere) controller model;
- BRU-K (steam dump to turbine condenser) controller model;
- primary pressure controller model.

These control models were developed on the basis of the design algorithms of the digital and analog controllers. Modelling principles consisted of the next important stages realized in the controllers models:

- choice of operation mode;
- calculation of deviation;

- forming of calculation cycles;
- calculation of actuator movement time;
- determination of conditions for actuator movement.

The modelling principles were successfully validated against plant transients, allowing to analyse both equipment operation and all components of controller algorithm, making recommendations to plant personnel related to tuning coefficients, controller feedbacks and conditions for actuator movement.

VALIDATION RESULTS

The qualification process of the ZNPP5 RELAP5 model consisted of two main steps: steady state and transient analyses.

The steady state analysis dealt with achieving certain ZNPP5 conditions known to exist at steady-state and full power. Special comparison was made for parameters of quasi-3D modules with design calculation values.

The transient analysis was performed to assess the capability of the model to adequately simulate phenomena that are of interest, and to determine differences between the calculated and measured. Three transients occurred at ZNPP Unit 5 were used for the model validation:

- Event #1. Power reduction due to malfunction of the main feedwater pump No. 1.
- Event #2. Reactor scram due to a trip of all main circulation pumps.
- Event #3. Loss of grid (via the main breaker VNV-750 kV).

Some validation results are presented in Table 1 and in Figures 4 – 12.

Figures 4-12 show the comparison of the calculated and measured data.

Figure 4 shows the behaviour of the neutron reactor power during unloading of the unit due to disabling of the main feedwater pump (event #1). The drop in power is modelled by the reactor power control and protection system model.

Figure 5 shows the feedwater flow rate from operational main feedwater pump 2 after main turbine driven feedwater pump 1 disabling (event #1). The behaviour and value of the flow rate are determined by the main feedwater pump controller model. The model of the main feedwater pump was constructed using widened pump curves simulation, the load controller, and the re-circulation line.

Figure 6 shows the behaviour of the primary pressure after actuation of scram due to disabling of 4 main circulation pumps (event #2). The behaviour and value of the primary pressure are controlled by the primary pressure controller model.

Figures 7, 8 show the behaviour of the coolant temperature in the “hot” and “cold” legs of loop 1 after actuation of scram due to disabling of 4 main circulation pumps (event #2). The behaviour and value of the coolant temperature are determined by the heat decay value and the natural circulation flow in the primary.

Figure 9 shows the stem position of the BRU-K valve after cutting off the unit from a grid (event #3). The valve operation is controlled by the BRU-K controller model.

Figure 10 shows the pressure in the steam header after cutting off the unit from a grid (event #3). The behaviour and value of the steam header pressure are controlled by the BRU-K controller model.

Figure 11 shows the level in the “cold” collector region of steam generator 1 after cutting off the unit from a grid (event #3). The behaviour and value of the level depend on the feedwater flow into the steam generator.

Figure 12 shows the level in the bypass region of steam generator 1 after cutting off the unit from a grid (event #3). The value of this level determines the operation of the feedwater contollers.

Table 1 contains the test results of loop heat-up distribution with 1 MCP turned off. Because the VVER-1000 loops are not located evenly in the circumferential direction, the distribution of the reverse flow in the reactor will not be symmetrical between the loops. The tests were performed using reactor vessel model only with loops boundary conditions. The measured values of temperatures and flows for the loop with tripped MCP were used as input data, to trace other 3 loops outlet temperatures from upper plenum. The table shows the results and basic information about measurement and calculation data for 3 chosen regimes.

Table 1

No of the test	Date& description	Power, MW		Pressure (gauge), kgf/cm^2		Loop heat-up, ^{0}C		Temperature measurement uncertainty, ^{0}C
		measured	calculated	measured	calculated	measured	calculated	
1	Apr 14, 1995, MCP-1 trip	1776	1776*	157.5	160.6*	-1.9 22.2 24.0 16.1	-1.9* 23.5 22.9 17.9	2
2	June 3, 1994, MCP-4 trip	2025	2025*	160.3	160.6*	18.7 24.0 24.0 -8.9	17.8 25.4 25.6 -8.9*	2
3	June 7, 1994, MCP-3 trip	2029	2029*	159.8	160.6*	24.1 15.9 -8.6 23.7	25.6 17.2 -8.6* 26.0	2

Note: * – value taken as a boundary condition

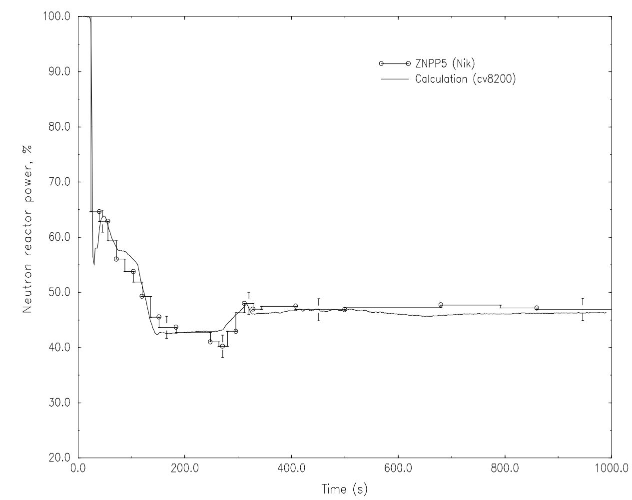

Figure 4 Neutron reactor power

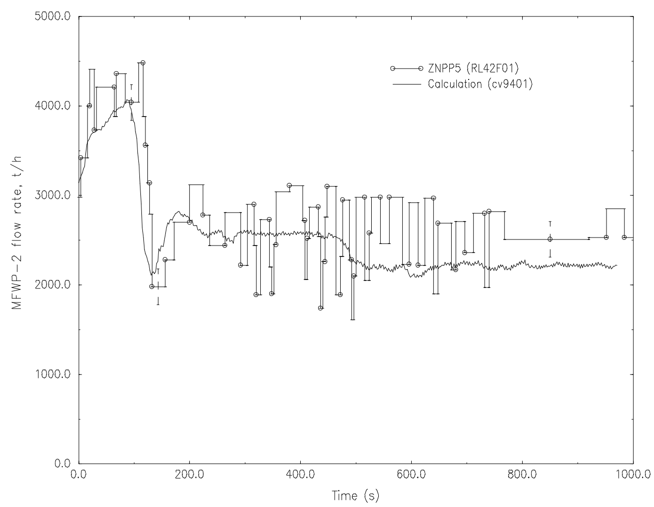

Figure 5 Main feedwater pump flow rate

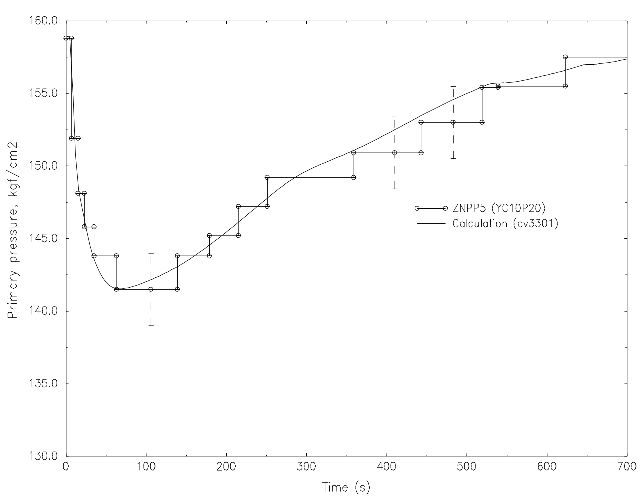

Figure 6 Primary pressure

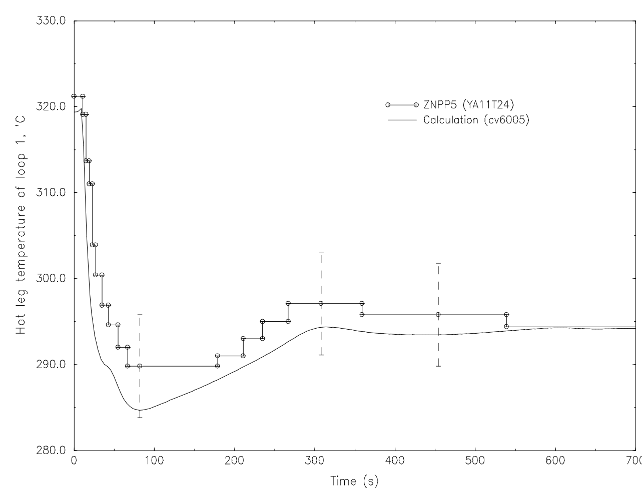

Figure 7 “Hot” leg temperature of loop 1

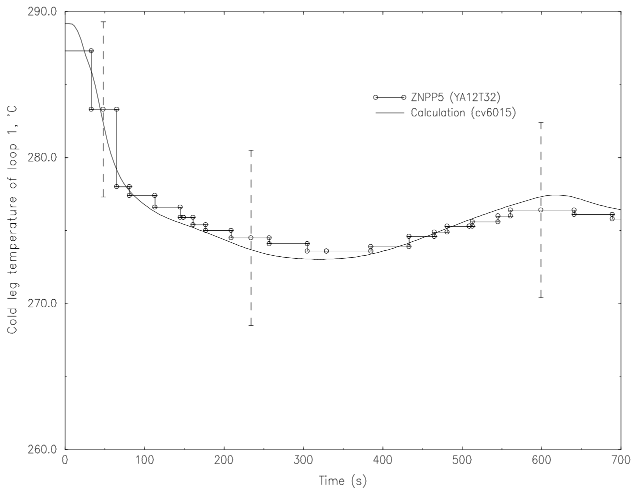

Figure 8 “Cold” leg temperature of loop 1

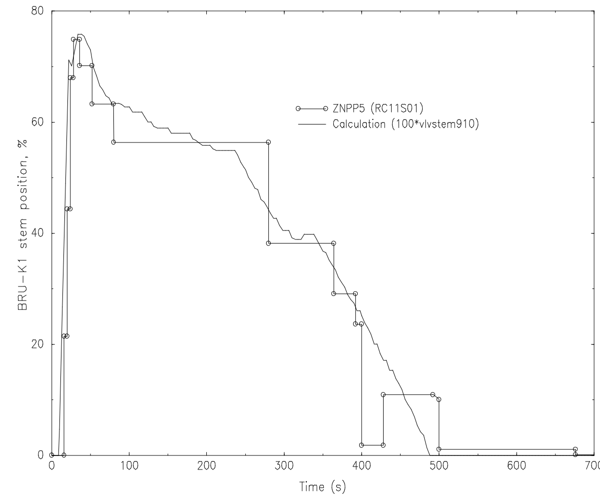

Figure 9 BRU-K valve stem position

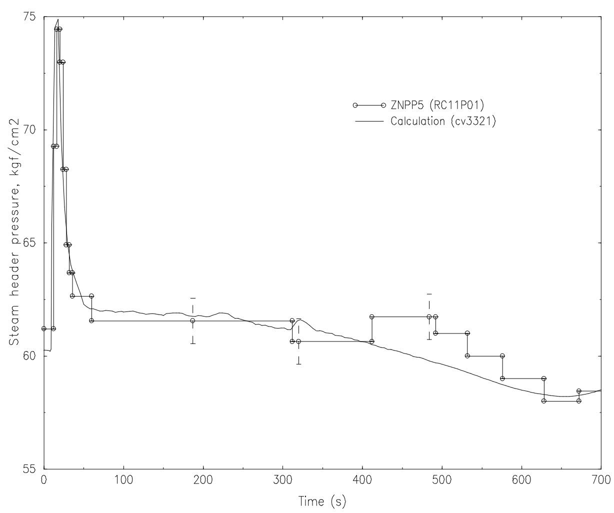

Figure 10 Steam header pressure

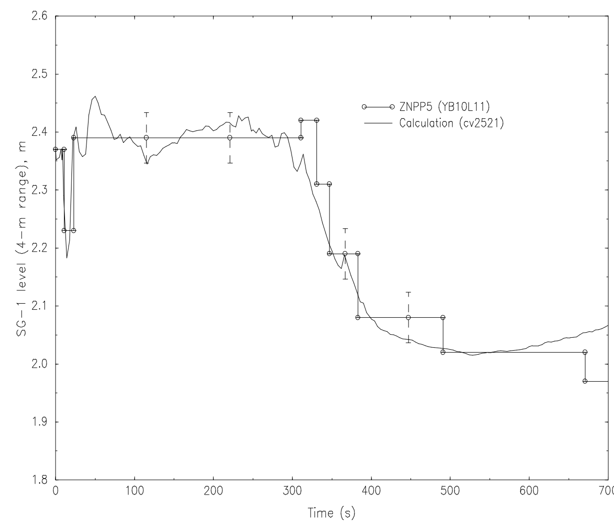

Figure 11 SG-1 level (4-m range)

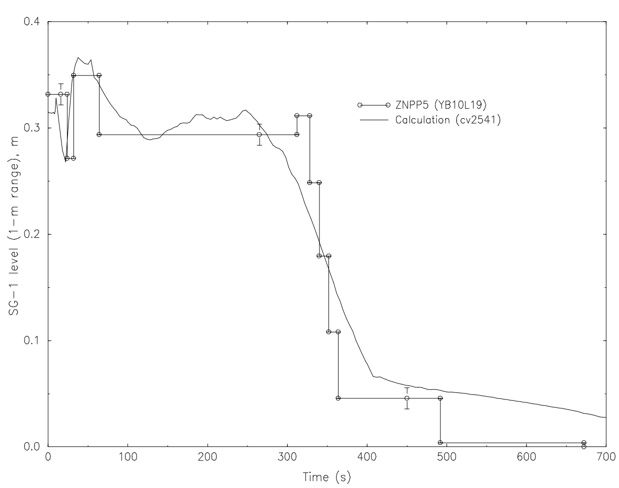

Figure 12 SG-1 level (1-m range)

CONCLUSIONS

In general, the introduced 3-D nodalization and inclusion of all reactor loops provides for "hi-fidelity" simulation. For example, coolant mixing due to slightly different operation of loops (a real situation) and the system response to asymmetric transient events such as tripping of one or two main coolant pumps can now be reliably mimicked. It also provides a better and more realistic representation of the horizontal steam generators capturing in detail the so-called “hot” and “cold” collector regions and their associated bypass.

Likewise, the present model affords a complete fidelity with ZNPP5 in terms of the location of safety (such as emergency core cooling system) and other systems important for plant simulation. This “configuration similarity” significantly facilitates preparation of analysis performed to investigate failure of specific systems and greatly contributes to transparency of presenting the results of analysis to the operation personnel.

The RELAP5 model was validated against ZNPP5 data using steady-state full power operation and three actually occurring transient events. The results of validation calculations are in excellent agreement with the ZNPP5 design parameters and the measured response.

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