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Object-Oriented Modeling, Simulation and Control of the IRIS Nuclear Power Plant with Modelica

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Abstract

The paper presents the application of the Modelica language to the modeling, simulation, and control of the new IRIS nuclear power plant, under development by an international consortium. The plant model, developed by using components from the ThermoPower library, as well as custom-built nuclear components, is described, as well as the digital control system model, which will eventually become very realistic. Special emphasis is put on the use of inheritance and replaceable objects for the management of a family of model variants over the project life-time. Selected simulation results are included.

1 Introduction

The IRIS project [3] involves 21 organizations from 10 countries and refers to the design of an innovative, light water reactor with a modular, integral primary system configuration. The reactor pressure vessel houses the nuclear fuel, control rods and control rods drive mechanisms, but also all the major reactor coolant system components, including the coolant pumps, the steam generators and the pressurizer (Fig.1).

IRIS is basically a PWR (Pressurised Water Reactor): in the primary loop, liquid water is heated by the nuclear fuel rods in the core, and is then sent by the pumps to the primary side of heat exchanger; the secondary loop actually generates steam which is sent to turbines to produce power.

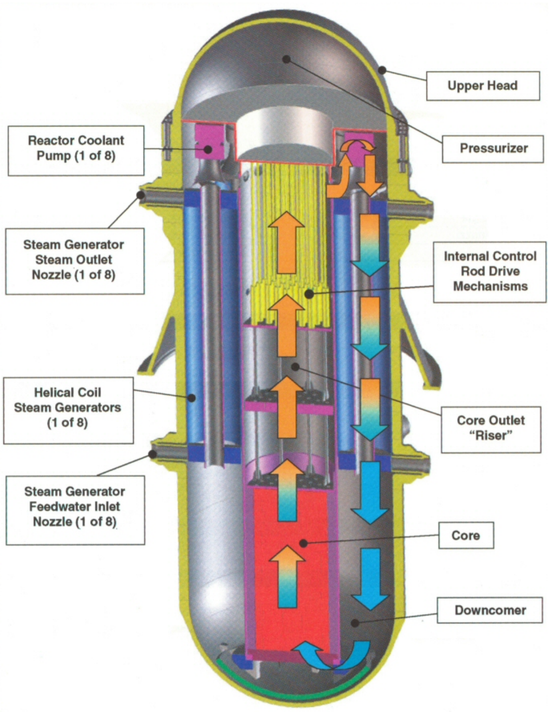

Figure 1: The IRIS Reactor

Compared to conventional PWR plants, however, IRIS has a set of distinctive features, which directly affect the control system design:

- the integral configuration requires a large water inventory in the primary loop, whose residence time is much greater than usual;
- a helicoidal once-through steam generator is employed on the secondary side, which has a very short residence time, compared to the more widespread U-tube recirculating steam generators;
- sprayers are not available to reduce the pressure in the primary loop during fast transients.

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The control strategy must take these facts into account, and a dynamic simulation tool is essential to ensure that the control objectives can be achieved.

Highly detailed dynamic simulators have been developed for the IRIS reactor [6]. Such simulators, based on the complex computational fluid-dynamics code named *RELAP* [10], are perfectly suited for accident analysis and safety-oriented evaluations of the reactor design features. On the other side, due to the amount of the details involved, they cannot be proficiently used for control-oriented dynamic simulation.

Within this framework, the use of the Modelica language offers a viable solution, allowing the development of dynamic simulators that are detailed enough for control-oriented analysis and yet with limited computational requirements.

To provide the required capabilities for the analysis, specific models for nuclear reactor components have been developed, to be applied for the dynamic simulation of the IRIS integral reactor, albeit keeping general validity for PWR plants. In addition to that, specific digital control blocks have been developed, so that a complete model of the plant and of its digital control system is available.

The paper is organized as follows. An overview of the plant model is presented in Section 2, while in Section 3 the models specifically developed for nuclear components are analyzed in detail. Section 4 contains an overview of the plant digital control system and, in Section 5, the problem of managing a library of plant models with different detail levels is tackled. Section 6 presents some closed-loop simulation results. Finally, Section 7 draws some conclusions and outlines possible future developments.

2 Plant Model

The model of the IRIS plant basically describes the primary circulation loop, i.e. the reactor coolant loop, and the secondary loop, i.e. the once-through evaporators, along with the feedwater and turbine systems. Most of the required models have no specific *nuclear* features, and were thus borrowed from the general-purpose ThermoPower library, designed for the modelling of generic thermo-hydraulic power plants; the library is an open-source project, described with more detail in [4]. The only notable exceptions are the reactor core and the pressurizer, which are described in the next section.

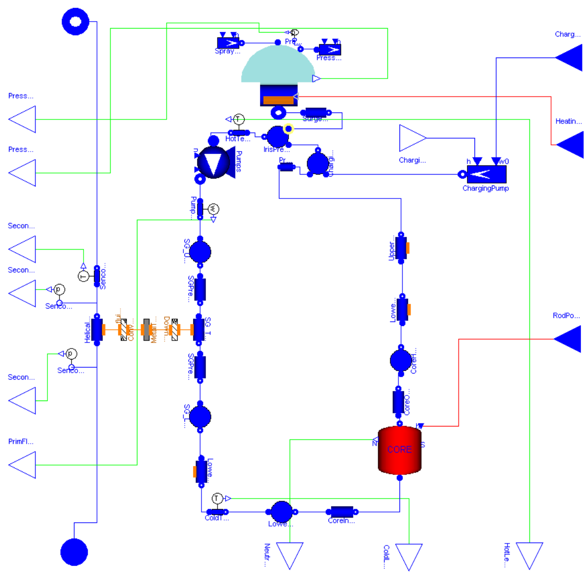

Figure 2: Plant flow diagram

2.1 Primary loop

The primary loop (see Fig. 2) starts with the pressurizer (top of the diagram); the pressurizer is connected by a pressure-loss component to the upper mixing volume, taking into account the mass and energy balances. Starting from the top of the diagram, counterclockwise, the centrifugal pump model can be found, followed by another plenum model. The primary side of the heat exchanger between the primary and secondary loop is then encountered, modelled by three cascaded, finite-volume pipe models; the middle one describes the section where the coolant is actually in contact with the secondary side tube bundle. Proceeding onwards, other two plenum models followed by a pressure loss can be found, leading to the inlet of the core model (see next Section). This in turn is followed by another pressure loss, another plenum, and the two riser sections, modelled by two pipes having different diameter. The loop is closed by a simple model of the chemical and volume control system (CVCS), basically a mixer and an ideal flow source. The fluid in the whole loop is one-phase water, with the exception of the steam filling the upper pressurizer dome.

The heat transfer between the primary and secondary loop is modelled by two heat transfer modules and by the thermal model of the tube metal mass. The primary-side heat transfer coefficient is held constant to its nominal value, since the Reynolds and Prandtl numbers does not vary substantially; on the secondary side, the heat transfer coefficients can be computed according to different laws, e.g. Chen's correlation, or much simpler, empirically tuned curves.

2.2 Secondary loop

The secondary loop is composed of the feedwater system, the helical coil once-through steam generator and the turbine system. The once-through generator is represented by a finite-volume, 10-node model of the two-phase fluid flow, assuming homogeneous flow, i.e. the same velocity for the liquid and vapor phases.
Currently, the feedwater system is represented by an ideal flow source, whose flow rate is determined by the control system, and whose enthalpy is a function of the plant load level, determined from balance-of-plant calculations. The turbine system includes a simplified, linear model of the high- and low-pressure turbines, plus simplified models of the connection to the grid, including an idealized synchronous generator, local loads, and a grid model. The latter ones are included to provide suitable boundary conditions for the (much slower) plant dynamics; therefore, they only model active power flows, neglect the electro-mechanical dynamics, and assume perfect synchronism between the generator and/or the grid.
In the near future, it is planned to replace the feedwater and turbine system models with more realistic counterparts, including steam bleedings and condensate train, to better represent the actual steam generator dynamics under large load variations. On the other hand, the finite-volume fluid evaporator model could be replaced by a simpler version, with moving boundaries between the liquid, 2-phase, and vapor sections, and an averaged description of each section.

3 Nuclear Components

The *Modelica* models for "nuclear" components have been developed to provide solutions which are suitable both for "general" use and specifically for the IRIS nuclear plant modelling. The main components are the *core*, (with separate models for the point kinetic neutronic generation, the fuel thermal dynamics and the moderator, as depicted in Fig. 3) and the *pressurizer*; the main modelling principles are summarized here, for more details see [1, 2].

3.1 Point Kinetics Neutronic

The point kinetic neutronic model describes the dynamics of the neutron generation processes in the core. The model is based on standard point kinetic

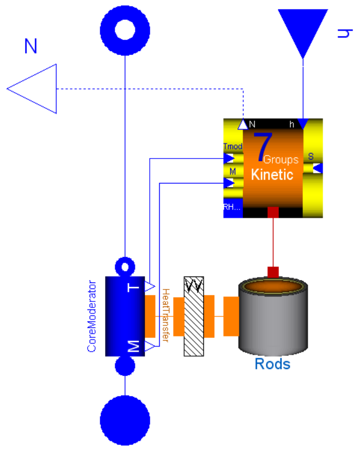

Figure 3: The Core Model Internal Structure

dynamic balance equations, describing the evolution of the neutronic population and of the precursor concentration. Reactivity feedback from coolant density, fuel Doppler effect, and rod insertion are accounted for. The dynamic terms can be switched off, to obtain a simplified static model, neglecting the fast dynamics. The neutronic power generated into the fuel is proportional to the neutronic population n, which responds to the point reactor kinetics balance equations :

$$\begin{aligned} \frac{dn}{dt} &= \frac{\rho-\beta}{\Lambda} n + \sum_{i=1}^{6} \lambda_i c_i \\ \frac{dc_i}{dt} &= \frac{\beta}{\Lambda} n - \lambda_i c_i \quad i = 1, \cdots, 6 \quad , \end{aligned} \tag{1}$$

where c is the precursor concentration leading to a delayed neutron source, ρ is the total reactivity of the core, β is the fraction of delayed neutrons, λ is the decay constant of the precursors and Λ is the characteristic period of the reactor or mean neutron generation time.
Reactivity feedbacks are taken into account as well, by considering linear or non linear feedback coefficients, always negative, for the coolant density effect (α_c), the fuel Doppler effect (α_f), the effect of the boron concentration (α_B) into the primary fluid as a neutronic poison and the level of insertion of the control rod banks into the core (α_{CR}). These relations are

$$\begin{aligned} \rho &= \rho_{CR} + \rho_f + \rho_c + \rho_B \, , \\ \rho_f &= \alpha_f \left(T_{eff} - T_{eff_0} \right) \, , \\ \rho_c &= \alpha_c \left(\frac{1}{v_c} - \frac{1}{v_{c0}} \right) \, , \\ \rho_B &= \alpha_B \left(C - C_0 \right) \, , \end{aligned} \tag{2}$$

where T_{eff} and T_{eff_0} are the instantaneous and reference effective fuel temperature, respectively, obtained from the fuel model, v_c and v_{c0} are the instantaneous and reference specific volumes of the coolant, C and C_0 are the instantaneous and reference boric acid concentration in the coolant. The boric acid concentration in the coolant depends mainly on the control rods insertion.

The reference values are those corresponding to the nominal, full power operation of the reactor.

3.2 Fuel model

The fuel model describes the dynamics of the thermal power generated within the core by the nuclear chain reactions. The neutronic generation model and the fuel model are linked by a connection between two *Modelica* standard `HeatPort`, where the connectors variable are the total power generated and the fuel temperature. A *ThermoPower* `DHT` distributed heat transfer connector is used as well, as an interface with the moderator, modelled by a 1-D flow model.

The model is based on the application of the time dependent Fourier equation (in monodimensional cylindrical geometry) to the three fuel zones: pellet, gap and cladding (Fig. 4).

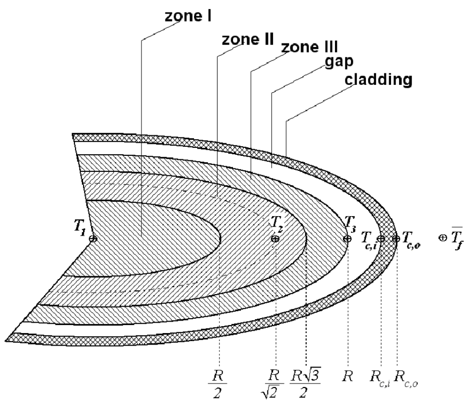

Figure 4: Fuel pellet radial scheme for heat transfer modelling

The main assumption of the model is to consider only the radial heat transfer, thus disregarding both the axial and the circumferential diffusions. Fourier's equation is discretized radially in five zones, and longitudinally in a user-decidable number of segments (N). For the pellet, gap and cladding the corresponding balance equations read:

$$\begin{aligned}
&\rho_p c_{p,p}\frac{\partial T_p}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_p \frac{\partial T_p}{\partial r}\right) + q''' \ , \\
&\frac{\partial}{\partial r}\left(k_g \frac{\partial T_g}{\partial r}\right) = 0 \ , \\
&\rho_c c_{p,c}\frac{\partial T_c}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_c \frac{\partial T_c}{\partial r}\right) \ .
\end{aligned} \tag{3}$$

where ρ is the density, c_p is the specific heat, T is the temperature, k is the thermal conductivity, q''' is the volumetric source term, r is the radial dimension and t the time, while the subscripts p, g and c stand for the pellet, the gap, and the cladding, respectively.
The heat transfer model is represented in Fig. 4, with the pellet discretized into three zones of equal volume. Eqs.(3), together with the conditions of heat flux vanishing at the pellet center and the continuity of the temperatures and heat fluxes at the three boundaries pellet-gap-cladding-coolant allow the determination of $T_p(r,t)$, $T_g(r,t)$ and $T_c(r,t)$.
In addition to the above equations, five correlations synthesizing the dependance of $c_{p,p}$, k_p, $c_{p,c}$ and k_c as a function of the fuel temperature and of k_g as a function of both the reactor power and the burn-up have been adopted.
The condition at the cladding-coolant interface is determined by the distributed heat transfer connector variables.
Finally, the effective fuel temperature, used to evaluate the Doppler feedback contribution on neutronics, is defined as follows:

$$T_{eff} = \frac{4}{9}T|_{r=0} + \frac{5}{9}T|_{r=R} \ . \tag{4}$$

3.3 Moderator

The core moderator is modelled by the ThermoPower library `Water.Flow1D`, with a small extension to make the fluid density available to the point kinetics model. The coolant model has the same number of volumes as the fuel. The convective heat transfer between the two components is calculated at each node by

$$\begin{aligned}
&\phi_{mod} = -\phi_c \ , \\
&\phi_{mod} = \gamma(T_c - T_{mod}) \ ,
\end{aligned} \tag{5}$$

where ϕ_{mod} and ϕ_c are, respectively, the moderator and the fuel cladding heat flux, γ is the heat transfer coefficient, and T_{mod} and T_c are the moderator and fuel cladding temperatures. Detailed RELAP simulations have shown that the heat transfer coefficient is approximately constant for all the operating conditions the control system is concerned with.

3.4 Pressurizer Model

The pressurizer model is based on a lumped parameter approach, which is appropriate to the IRIS case. Both water properties in the liquid volume and in the steam volumes are assumed as homogeneous, at equal pressure but not at thermodynamic equilibrium.

The model is based on two groups of dynamic mass and energy balance equations, the first for the liquid phase and the second for the vapor phase inside the tank. Mass and energy transfer between the two phases is provided by bulk condensation and surface condensation of the vapor phase, and by bulk boiling of the liquid phase. Additional energy transfer can take place at the surface if the steam is superheated.

External interfaces are provided for connections to the hydraulic loop by a bottom flange and to a safety circuit by a *safety* flange; also available are a heating power command input and a level signal probe output.

The heating power input is processed by a limiter and a low pass filter block to simulate the delay in heating effect and the limited heaters power. The resulting effective heating power signal drives the production of saturated steam by the heaters at a rate corresponding to the difference between the enthalpy of the liquid holdup and the enthalpy of saturated steam. For simplicity, the corresponding steam flow enters directly the steam holdup, without causing heating of the liquid holdup.

The bottom flange's flow enters directly the liquid volume; its pressure is increased depending on the liquid holdup's level.

The metal wall dynamics is taken into account, assuming uniform temperature. Heat transfer takes place between the metal wall and the two phases and between the wall and the external ambient at fixed temperature.

4 Control

The control design of the IRIS nuclear power plant is a complex task, with objectives that, depending on the plant operating conditions, vary from the management of start-up sequences to the recover from turbine or reactor trips and to the grid power/frequency regulation at full nuclear power.

Classic design concepts, for early nuclear units, relied on separate control systems for each control loop, and limited signal interaction between the loops [9]. This simplified the design of each loop, particularly with analog control systems where each interconnection added hardware expense. On the other side, the current trend is for more integrated systems that can take advantage of coordinating the different control loops [7]. This allows for more effective plant control, but complicates the control system failure analysis. A viable solution for IRIS is the choice of a hierarchical control system, as depicted in Fig. 5.

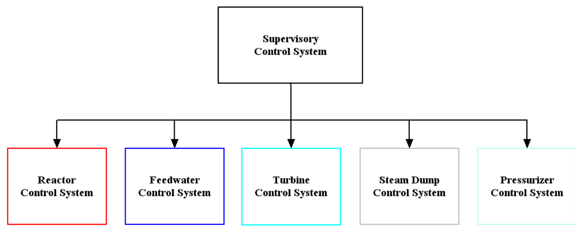

Figure 5: Control system architecture

At the top level is located a supervisory control system with the following functions:

- Establish the plant electric power reference signal. Such reference signal will be used to derive reference and/or feedforward signals for the other major control loops.
- Monitor plant conditions and determine/coordinate the appropriate operating modes for the major control systems.

The control sub-systems have different settings and a varying structure (i.e., different inputs and different controller structure) depending on the specific operating mode of the plant.

All the operating modes to drive the plant during the non-emergency maneuvers have been designed [8]; nevertheless, only the "full-power" control mode (nuclear flux from 20% to 100%) has been fully implemented, simulated and tested yet, so, from here on, the description will cover only such operating mode.

4.1 Supervisory Control

The supervisory control system uses the normalized desired power as an input signal to derive the reference and feedforward signals for the lower-level control systems. On the base of the desired power the temperature and nuclear flux reference for the reactor control are derived, along with a pressure reference for the turbine and steam dump control systems and with the flow rate reference for the feedwater control. The signals to be fed to the lower systems are derived from the desired power reference with linear filtering and through look-up tables based on steady-state plant balances.

4.2 Reactor Control

The aim of the reactor control is to control the coolant temperature, and thus the reactor nuclear power, by driving the control rods stepping system. As a matter of fact, the reference is a temperature signal coming from the upper level, while the measurements include the core coolant average temperature (obtained as the mean between the temperature at the core inlet and the one at the core outlet) and the nuclear neutron flux (obtained through special sensors enclosed within the core shielding).

The temperature error, with suitable dynamic compensation, is used to generate an error signal for determining the speed request for the control rods, along with a power mismatch signal which is used to improve the stability and the velocity of the reactor control system response. The power mismatch signal (i.e. the difference between the reference and the measured neutronic flux) is fed into a rate compensation filter, to eliminate steady-state influence, and then into a non-linear, power-dependant gain, to improve low-power response while avoiding high frequency excitation of the rod stepping system.

The combined error signal enters a rod speed program that features a small dead band to avoid high frequency rod stepping. The speed request thus generated is then serviced by a servo control system embedded within the control rods drive mechanism. This servo is currently described by a high-level behavioral model, which could be eventually replaced by a physical-based model.

4.3 Turbine Admission Valve Control

The turbine system for the IRIS power plant has not been completely designed yet and it is reasonable to assume that the turbine supplier will provide most of the requirements for the turbine control system; however, the design must be compatible with the overall IRIS plant control strategy. The most important constraint is that the IRIS turbine control will have the responsibility for controlling steam pressure by acting on the turbine admission valve (TAV).

The control is based on a PID, its input being the reference pressure signal coming from the supervisory control system and the actual steam pressure measured at the turbine inlet, with suitably low-pass filtering action. The PID output is then summed to the amplified frequency mismatch (i.e., the difference between the actual generator frequency and the desired frequency), with the gain depending on the grid droop setting. The resulting signal is fed to the TAV drive system after being filtered by a non-linear algebraic function, which is an approximate inverse of the TAV characteristic.

4.4 Steam Dump Control

The steam dump control system must control steam pressure when the turbine admission valve control is not doing so, and must provide a backup in all other cases. Experience shows that a simple PID control performs well, particularly if the system uses hydraulic steam dump valves, as it will be in the IRIS case.

The controlled variable is the steam dump valve opening, while the controller inputs are the pressure reference (from the upper level) and the turbine inlet steam pressure (low-pass filtered). Additional steam-dump action is available in case of need: the power reference, filtered through a rate compensator and a suitable gain, is added to the steam dump valve control signal, to provide a faster response in case of sudden changes in the requested power (e.g., when a reactor or a turbine trip occur and the supervisory control system instantaneously lowers the power reference).

4.5 Feedwater Control

The feedwater control system directly controls the feedwater flow in the secondary side by acting on a valve located at the feedwater pumps. The structure is based on two PID controllers in cascade configuration. The inner loop acts to control feedwater flow to the reference value obtained from the supervisory control. In the ideal case with perfect settings in the supervisory controller, this would result in the plant operating at the desired power, at least in steady state. Of course, such an open loop control on power would be sensitive to parameter variations, so the outer loop provides a trim signal to adjust feedwater flow to achieve the desired power by the action of a PID controller with the reference and the actual power as inputs. The feedwater valve control signal is then filtered by a non-linear algebraic function, which is an approximate inverse of the valve characteristic.

4.6 Digital PI controller

Models for digital PI and PID controllers, in ISA form, have been implemented. Here, for the sake of brevity, only the PI development is briefly showed: the PID model development is quite similar.
The model is based on the standard industrial ISA formulation, with the output calculation formula obtained with a Tustin discretization:

$$
\begin{aligned}
CS(s) = & K_p\left((bSP(s)-PV(s))+\frac{1}{T_I\,s}(SP(s)-PV(s))\right)\\
& \Downarrow\left(\text{Tustin: } s=\frac{2}{T_s}\frac{z-1}{z+1}\right)\\
CS(z) = & SP(z)\frac{a_0+a_1\,z^{-1}}{1-z^{-1}}+PV(z)\frac{b_0+b_1\,z^{-1}}{1-z^{-1}}
\end{aligned}
\tag{6}
$$

with

$$
a_0=\frac{2\,Kp\,b\,T_I+Kp\,T_s}{2\,T_I}\quad,\quad a_1=\frac{-2\,Kp\,b\,T_I+Kp\,T_s}{2\,T_I}\quad,
$$
$$
b_0=\frac{-2\,Kp\,b\,T_I-Kp\,T_s}{2\,T_I}\quad,\quad b_1=\frac{2\,Kp\,b\,T_I-Kp\,T_s}{2\,T_I}\quad.
$$

The complete controller model includes also advanced features like *manual* and *tracking* working mode, output saturation, and anti wind-up mechanism.
The resulting block has two boolean inputs (automatic and tracking switch signals), four discrete real inputs (set-point, process value, manual and tracking signals) and a discrete real output (control signal).
The *Modelica* implementation exploits the language features for digital blocks, using discrete variables and with the instructions enclosed within a sampling loop:

```
when {initial(),sampleTrigger} then
...
[PI computations]
...
end when;
```

The anti wind-up mechanism is implemented via an auxiliary variable:

```
CSwind=pre(CS)+a0*SP+a1*pre(SP)+b0*PV+
       b1*pre(PV);
```

where `Cswind` is the auxiliary variable, `CS` the control variable, `SP` the set-point and `PV` the process value.
The actual control value is chosen depending on the controller logic state (automatic, manual or tracking) and on the saturation values, e.g. :

```
if AUTO then
   if CSwind >= CSmax then
       CS = CSmax;
       CSport.signal[1] = CSmax;
   elseif CSwind <= CSmin then
       CS = CSmin;
       CSport.signal[1] = CSmin;
   else
       CS = CSwind;
       CSport.signal[1] = CS;
   end;
else
   ...
```

where the parameters `CSmax` and `CSmin`, are the upper and lower saturation limits for the control action.
With this implementation structure, the controller integral state is automatically updated at every execution cycle so to be coherent with the last output sample.

5 Model Management through the project life-cycle

Object-oriented features such as inheritance and replaceable components are often described as key factors in the development of reusable model libraries. In fact, they can also be extremely useful for the proper management of families of application models throughout an engineering project's lifetime, as it will be explained in this section with reference to the IRIS project.

5.1 Requirements

During the IRIS project lifetime, a considerable number of model variants will have to be built and analyzed; some of them will become obsolete and will have to be discarded, while others should be kept consistently up-to-date. The motivations of the model variants are now briefly discussed.
Depending on the specific simulation to be performed, different accuracy vs. computational load trade-offs are required. Reference simulations should be performed with the maximum level of accuracy and detail, and cross-checked with the results of the reference simulations performed with the certified RELAP code. When performing simulations around a certain operating point, some approximations could then be introduced, which are only valid for that operating region; it should be possible to easily check simplified versions against their more accurate counterparts.
Some of the plant parameters (e.g. the pump characteristics, or some plenum volumes) are not yet definitive, and could change in the future; when one of such parameters is changed, it is essential that all the current model variants are updated consistently.
Once the initial phase of the control system design has been carried out, a systematic simulation campaign

must be performed to check that the operational constraints (i.e., the activation thresholds of the protection system) are never violated in all the predicted operating conditions and transients; thousands of different simulation runs can be required. To carry out this task, the simplest and fastest possible variant of the plant model should be used.

It should be also kept in mind that the plant models will be developed, used, and maintained by different people over a wide time span (several years) and at widely spaced sites (US, Europe). For instance, the model presented in this paper will be presumably frozen for some months, and then possibly resumed when the project will enter the commercial phase. It is therefore essential to avoid building a plethora of distinct models, differing only by some details, which would be extremely difficult to maintain and document consistently.

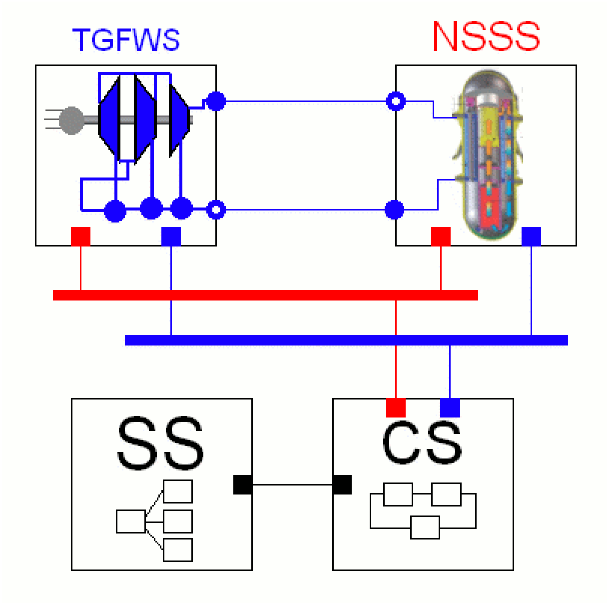

Figure 6: The Base Simulator Model

5.2 Implementation

The top-level structure of the simulator is represented in Fig. 6: the TGFWS block contains the Turbine/Generator/Feedwater system model; the NSSS block contains the Nuclear Steam Supply System model, i.e. the nuclear reactor, with the primary and secondary loops. The two are connected to each other by thermo-hydraulic connectors. The control side is represented by the CS (Control System) block, collecting all the control loops, and the SS (Supervisory system) block, which generates the set points for the CS based on the plant load request. Three bus connectors carry the sensor, actuator, and reference signals. This structure is common to all the possible variants of the model, and thus contained in the `IRISSimulatorBase` partial model. Different versions of the simulator can be instantiated by selecting the actual content of each block; for instance, one could use the simplified TGFWS model described in Section 2, or a more detailed one.

The NSSS model contains a replaceable model (`HelicalCoil`) for the secondary side of the once-through steam generator, which can be implemented by either the finite-volume or the moving boundary model, and by adding through inheritance the desired equations to compute the heat transfer coefficient.

Besides that, it is possible to vary dramatically the degree of detail and the computational load of the model by changing the number of nodes in the core and once-through generator models, as well as by redeclaring the medium models in the primary and secondary loop components. The default medium models are the IF97-based water models taken from `Modelica.Media`, but it is possible to use much faster models, based either on table interpolation or on equation-based simplified medium models. The thermodynamic conditions of the fluid in the primary loop conditions vary in a rather narrow range (140 to 160 bar, 270 to 330 degrees Celsius), so that extremely simplified models can still be acceptable; the fluid conditions in the secondary loop vary in a broader range, from subcooled liquid to superheated steam, albeit in a narrow pressure range around 58 bar, due to the pressure control system action.

Last, but not least, if an incompressible fluid model is adopted for the primary loop, the fast pressure states caused by the small compressibility of the fluid, coupled with the small hydraulic resistances around the circulation loop, are automatically avoided, without any need to change the component models. This is essential to allow the use of the faster explicit integration algorithms (e.g. forward Euler).

The simulation suite is then organized as a small library (Fig. 7), containing the “empty” base models, and the actual models of the different parts, without any unnecessary duplicate of data. Any specific variant of the simulation model can be instantiated from this library by using suitable modifiers. For example, the variant V2 of the simulator, using a simple incompressible water model for the primary loop, 7 nodes in the core model, a finite volume model of the steam generator with 15 nodes using Chen’s correla-

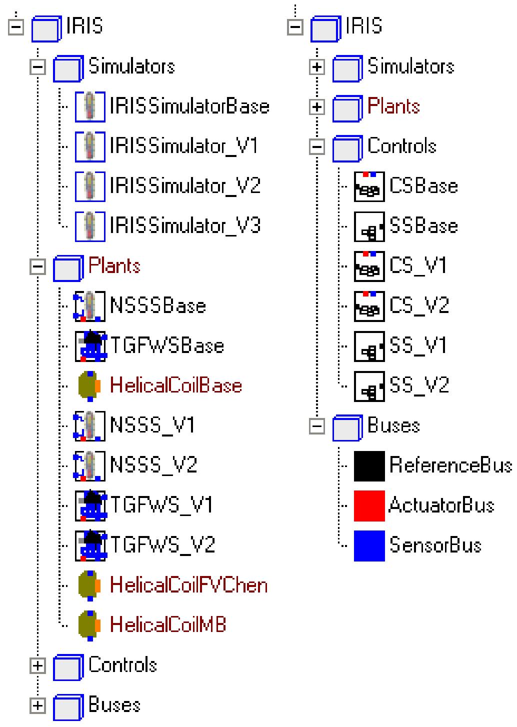

Figure 7: Iris Simulation Suite

tion for the heat transfer coefficient, the variant V1 of the TGFWS, and the variant 2 of CS and SS, is instantiated as follows:

```
model IRISSimulator_V2
 extends IRISSimulatorBase(
    redeclare Plants.NSSS_V1 NSSS(
      redeclare package PrimaryMedium =
        Media.SimpleIncompressibleWater,
      Core(N = 7),
      redeclare Plants.HelicalCoilFVChen
        HelicalCoil(N=15)),
    redeclare Controls.CS_V2 CS,
    redeclare Controls.SS_V2 SS,
    redeclare Plants.TGFWS_V1 TGFWS);
end IRISSimulator_V2;
```

`IRISSimulatorBase` is the empty base model described at the beginning of the section, and its four replaceable components `NSSS`, `TGFWS`, `CS`, `SS` are of type `NSSSBase`, `TGFWSBase`, `CSBase` and `SSBase`, which again only contain the interfaces. The NSSS model in turn contains the replaceable steam generator model `HelicalCoil`.

In this way, it is straightforward to maintain a consistent state for a potentially large family of simulator variants, as well as documenting all of them efficiently.

6 Simulation

The results of a closed-loop simulation, obtained with the tool Dymola ([5]), are now presented. The reference transient is a filtered step variation of the electrical load reference, from 90% to 100% and then back to 90%. Although such a rude transient will never be performed on the actual plant, it is usually employed to assess the overall dynamic response of the control system, in terms of speed of response, damping, overshoot, and so on. The normalized transients of the neutron flux (representative of the generated nuclear power) and of the generated electrical power are shown in Fig. 8, along with the reference power signal. The responses are well-damped and with limited overshoot. The neutron flux transient takes into account the effect of the step-by-step actuation mechanism, as well as of the dead-band included to avoid persistent chattering around a specific operating point. The corresponding normalized transients of some control variables (i.e. TAV opening, feedwater flow rate, and rod insertion) are shown in Fig. 9.

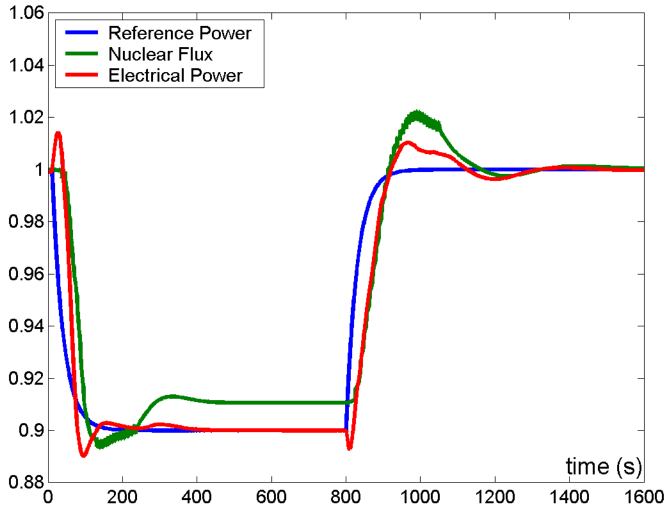

Figure 8: Normalized response to a step load variation: measured variables

7 Conclusions and Future Work

In this paper, the application of Modelica to the study of the control system of the new IRIS nuclear power plant has been presented; this is also the first industrial-scale application of the ThermoPower Modelica library.

The well-behaved nature of the closed-loop transients

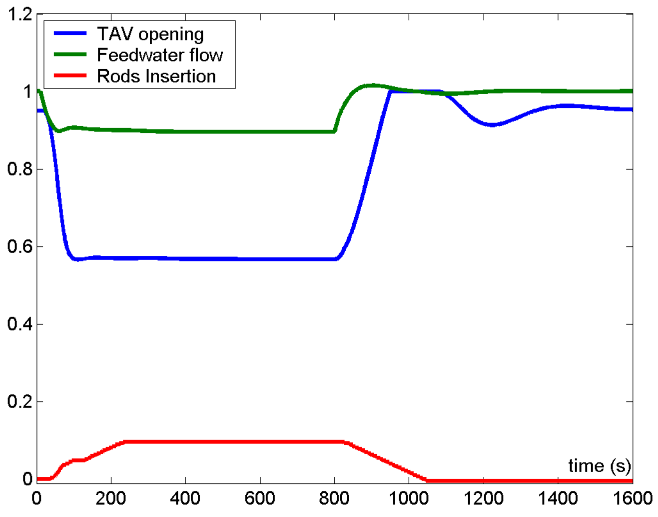

Figure 9: Normalized response to a step load variation: control variables

has confirmed that the new reactor concept will not pose exceedingly difficult problems to the control engineers, compared with already existing PWR plants. On the other hand, the availability of a detailed dynamic model will allow the study of more advanced control concepts, to cope with situations such as. e.g., load/frequency control in small grids, or improved management of blackout transients.
The object-oriented features of the Modelica language (replaceable classes in particular) have been fully exploited to allow the efficient management of all the variants of the plant simulator, which will be needed throughout the project's life-time. The structure of the simulation suite will allow an easier re-use and extension of the models developed so far, when the project will eventually enter the detailed engineering phase, prior to the construction of the first plant.

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