MNRSIM: An Interactive Visual Model which Links Thermal Hydraulics, Neutron Production and other Phenomena

D. Gilbert, Wm. J. Garland, T. Ha

McMaster University, Hamilton Ontario, CANADA

6th International Conference on Simulation Methods in Nuclear Engineering

Oct 12-15, 2004

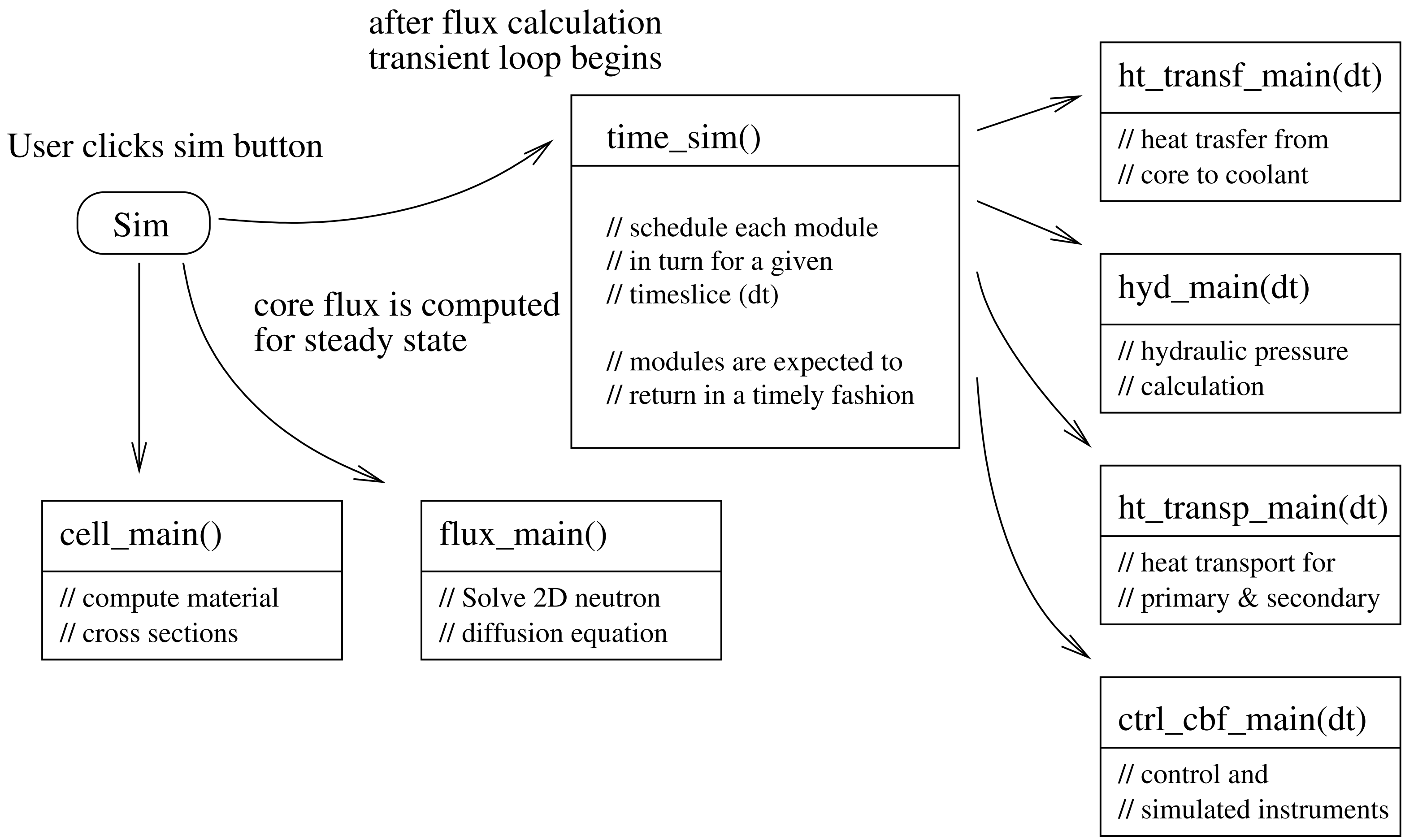
MNRSIM

The goal for the McMaster Nuclear Reactor Simulator (MNRSIM) is a first order visual approximation of the major elements of the reactor including

- Flux calculations
- Reactor control
- Thermal hydraulic calculations
- Eventually fuel management

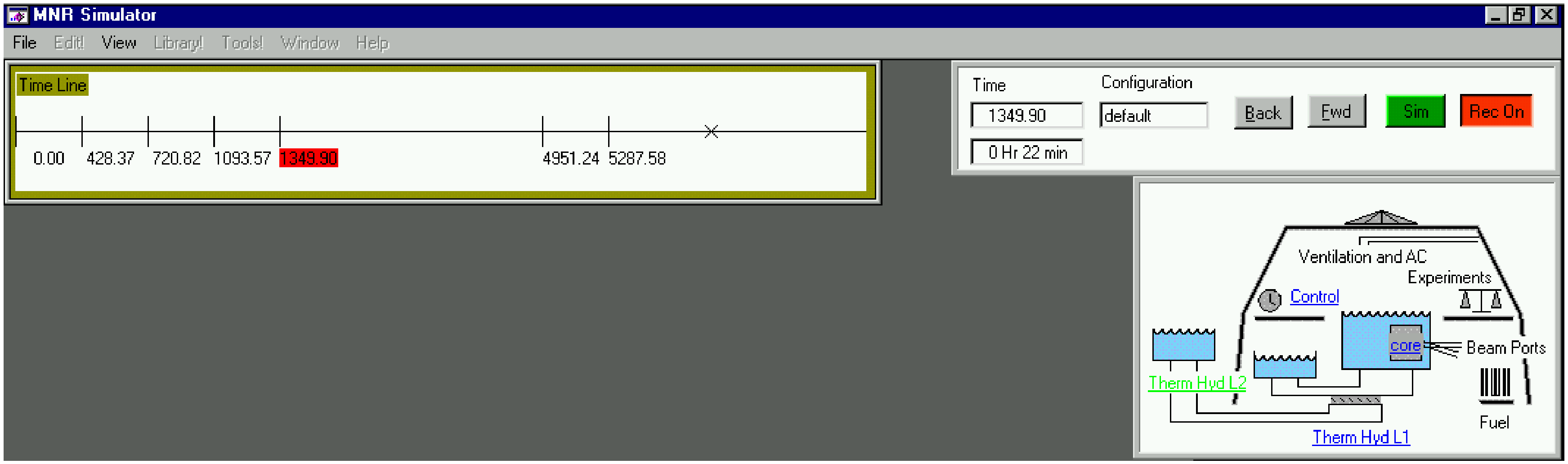
The current version of MNRSIM simulates two major physical processes, the reactor core, and the reactor cooling system using six basic nuclear physics modules: a cross section collapsing module, a two dimensional core flux module, a hydraulic pressure module, a cooling system heat transfer module, a core heat transfer module, and a simple control and instrumentation module.

Program Design



MNRSIM is written in C and compiles under LabWindows. 12,000 lines of code define 20 modules, 7 for physics, 9 for user interface, 4 are general purpose.

User Interface Design



The physical systems that are modeled by MNRSIM are represented diagrammatically by a schematic figure.

- Time-line in upper left stores a sequence of model snapshots
- VCR buttons in upper right manipulate the time-line
- Links on Schematic are used to change reactor views

Reactor Core Physics

Core physics is modeled in two dimensions

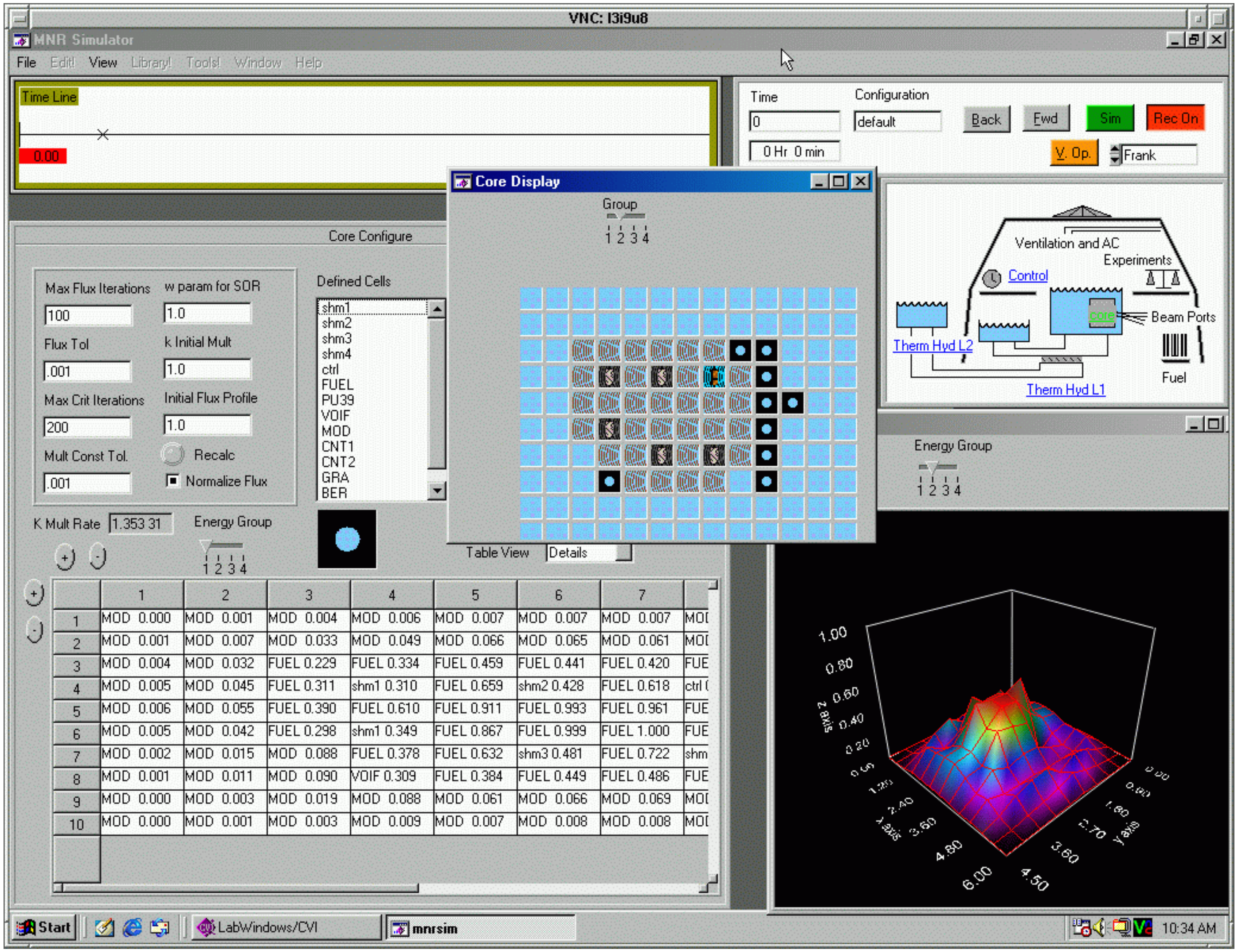
- One, and three dimensional extensions to model are planned

Flux module computes a steady-state shape for the core flux based on the neutron diffusion equation, the following finite differenced approximation is used:

$$\left[\Sigma_{R_i}^{g} + \sum_{j}^{J} \frac{D_{ij}^{g}}{\triangle_{ij}^{2}}\right] \phi_{ig} - \sum_{j}^{J} \frac{D_{ij}^{g}}{\Delta_{ij}^{2}} \phi_{jg} - \sum_{g'=1}^{g-1} \Sigma_{s_i}^{g' \to g} \phi_{ig'} = \frac{\chi^{g}}{k} \sum_{g'=1}^{G} \nu_{g'} \Sigma_{f_i}^{g'} \phi_{ig'}$$

- 4 Energy group approximations are used
- Successive relaxation solver is implemented in C
- Materials cross sections derived from WIMS library

Core Assembly Interface



A wide variety of cores can be easily assembled and saved in archives.

Core Heat Transfer

- A one dimensional heat distribution is computed within the plate

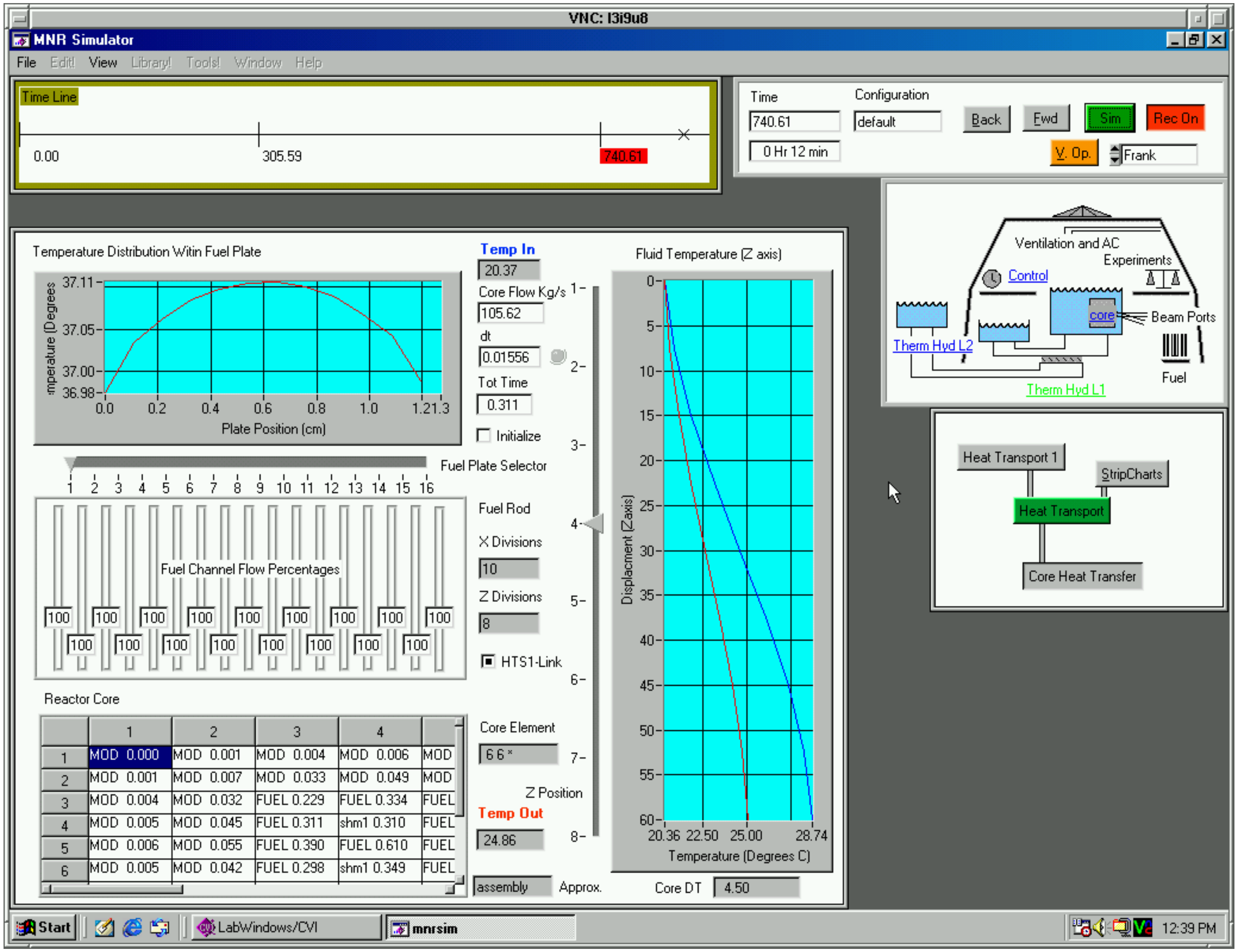
$$\rho C_p \frac{\partial T}{\partial t} = q''' + \frac{\partial}{\partial x} k \frac{\partial T}{\partial x}$$

Since the fuel plate is so thin a very small time step was required to maintain stability in the computation. A Crank Nicholson formulation was used.

- Heat removal by axial coolant flow is computed in one dimension

$$A\rho C_p \frac{\partial T}{\partial t} = -w \frac{\partial h}{\partial z} + q'(z)$$

Core Heat Transfer Interface



Flux shape is used to compute transient temperatures in fuel rods.

The Hardy-Cross Method

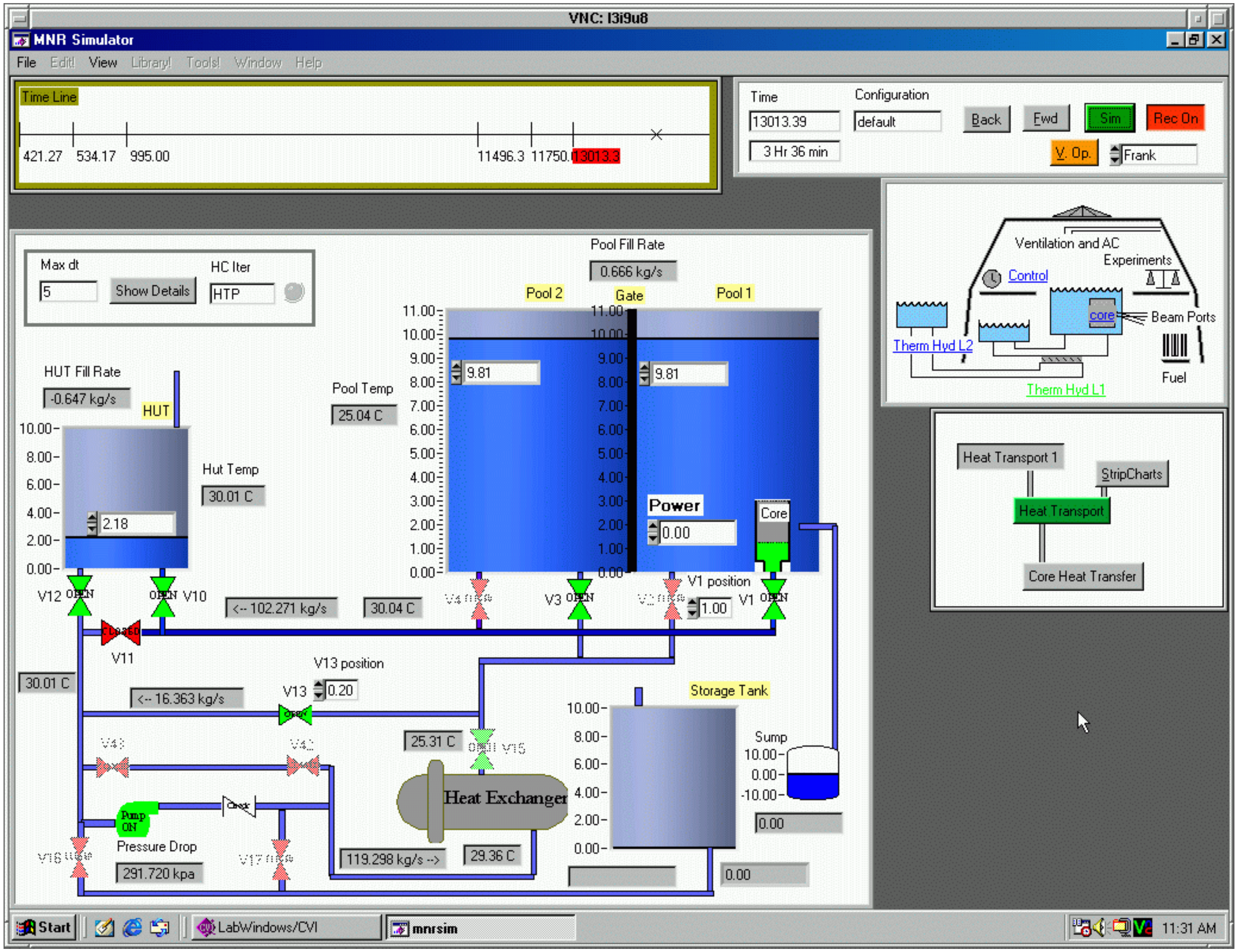
Flow in a pipe $LA\rho\frac{\partial v}{\partial t} = A\left[\Delta P - \left(\frac{fL}{D} + k\right)\frac{\rho v^2}{2g_c} - \rho g \bigtriangleup z\right]$

Define $K = \left(\frac{fL}{D} + k\right)\frac{1}{2\rho A^2 g_c}$ also Define $P^* = P + \rho g \bigtriangleup z$

$$0 = P_1^* - P_2^* - Kw^2$$

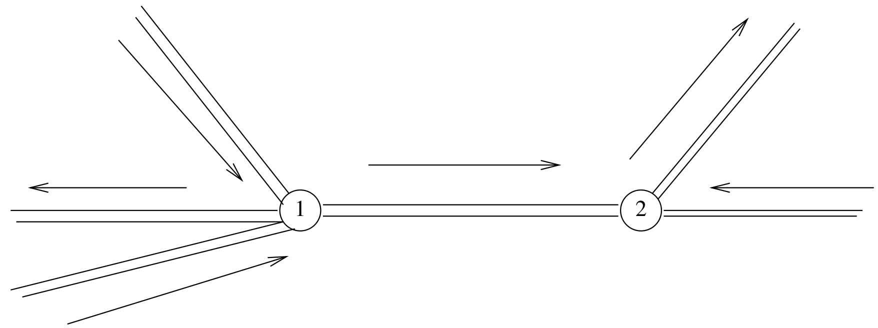
$$P_n^* = \frac{1}{\displaystyle\sum_{j=1}^{\#links} \frac{1}{\sqrt{\left|P_j^* - P_n^*\right| K_j + \varepsilon}}} \times \sum_{j=1}^{\#links} \frac{P_j^*}{\sqrt{\left|P_j^* - P_n^*\right| K_n + \varepsilon}}$$

Primary Heat Transport User Interface



Interface combines flow, temperature distributions, and valve controls.

Heat Transport



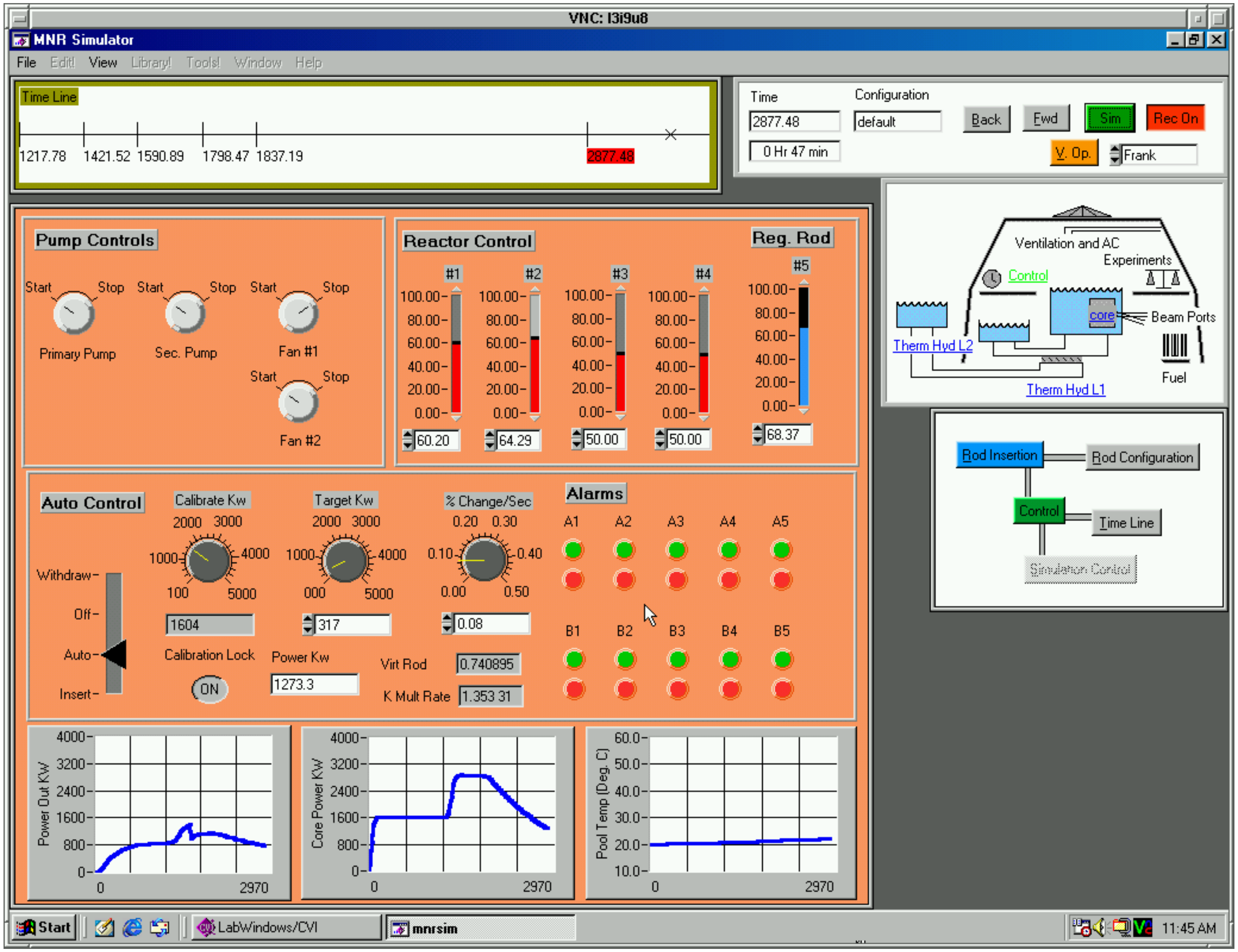
The basic heat transport equation used takes the following form:

$$C_p \rho V \frac{\Delta T_i}{\Delta t} = w_{in} C_p T_{in} - w_{out} C_p T_{out} + Q_i$$

For a general pipe like the one shown in the figure the following equation is used to compute the change in enthalpy due to flow into and out of each end.

$$\frac{\partial H}{\partial t} = C_p \sum_{i}^{\#links} w_i T_i$$

Control Panel



Calibration and Validation

Model validation has been fairly simple at this stage. The data recorder information was used to both calibrate and verify the primary and secondary thermal loops and calibrate the core power levels.

time	core.o C	core.i C	P.o HX C	2.o HX C	2.i HX C	Pool C	Air C	Pow %	Flow KG/s	Pool meters	HUT meters
14:00	27.4	27.44	22.11	n/a	22.83	28.06	-20	90	98.5	9.25	2.27
15:00	31	26.78	22.06	20	16.67	27.72	-16	91	98.5	9.25	2.27
16:00	31.5	26.28	22.00	n/a	17.00	27.11	-15	93	98.5	9.1	2.31
17:00	31.8	26.67	21.94	20.28	17.17	27.11	-15	93	98.5	9.1	2.31

Several parameters of the simulation were adjusted to calibrate the model. Among these are:

- minor loss friction coefficients for the heat exchanger
- minor loss friction coefficient of the reactor core

Conclusion

This is an open project which will hopefully continue for many years; the work which still needs to be done is extensive.

- The model needs much more careful validation
- Proposed extensions include
  - Basic fuel management
  - Failure modeling system- builds history of randomly generated scenarios
  - Detailed modeling of the MNR's control panel and alarm system
  - Detailed modeling of fluid flows in the core, and in the pool, as well as detailed air flow models within the building

Software and documentation can be downloaded at

- http://nuceng.mcmaster.ca/mnrsim