Chapter 4

TRRSIM modeling

TRRSIM stands for Thai Research Reactor 1/M1 simulation model. This model is done in one dimension and one group of neutron energy. In neutonic model, it is prepared along the length of fuel pin which in the radial plane is smeared for all materials consisting in. The radial axis is used in thermal hydraulic modeling which only an averaged fuel pin is used.

Thermal hydraulic model is done in one dimension in cylindrical coordinate. Heat is by-product of fission reaction generated in fuel-moderator. This amount of heat need to be removed to coolant by conduction inside fuel pin and convection on fuel-moderator clad and coolant interface. The thermal hydraulic model is using neutron flux getting from neutronic model on calculation of generated heat density.

Due to a certain amount of temperature coefficient, the neutronic model get a feedback from changing in fuel-moderator temperature. The fission cross section, total absorption and diffusion coefficient would be affected.

4.1 TRRSIM assumption

This model is prepared in simplified form with many assumptions as followings.

1. Neutron energy used in model is only one group.
2. The neutronic model is prepared in one-dimension along the fuel pin.
3. The elements in regions are homogenized together forming homogenized mixture.
4. The neutron flux distribution in-core radial axis is not concerned.
5. Neutron cross section used for any materials in core is based on cross section at thermal energy except fission cross section which change with fuel-moderator temperature.
6. The negative temperature coefficient is fixed at a constant value.
7. Temperature used on fission cross section calculation base on individual temperature of the fuel-moderator in each region.
8. Heat transfer coefficient of clad and coolant interface is assumed to be constant over the whole operating range of operating power.
9. Heat conduction equation is done in one dimension, radius in cylindrical coordinate. The axial conduction along the fuel-pin is ignored.
10. Coolant flow in channel is assumed to be force flow and constant over all conditions.

4.2 Neutronic

The homogenized core is devided in to 29 regions; 15 in active region, 6 regions in graphite reflector and 8 regions in water reflector. In the active region the space between each node is 2.54 cm, 2.9 cm in graphite reflector and 25 cm in water reflector.

As we know that the finer the elements, the better accuracy of the calculation result. The limitation is that equations are in matrix form and will highly increased calculation load if higher series of equations are solving.

In active region 1" interval is enough and fine for a certain level. Neutron flux vary quite a lot in active regions. These regions contain fissionable material, neutron absorber and structural materials. So changing in absorber rod position will effect neutron flux.

The larger interval in graphite reflector and water reflector are not much effect in flux variation. They do contain less fuel and only one neutron energy considered.

4.3 Thermal Hydraulic

The thermal hydraulic model will be calculated on only one average fuel pin. It is done in 1 dimension in radial of the fuel pin included in the coolant Cylindrical coordinate is applied and used. There are totally 47 regions in cross sectional area; 10@0.03175 cm in central Zirconium rod, 25@0.0599 cm in fuel-moderator mixture, 1@0.0022 cm in air gap, 10@0.0051 cm in clad and 1 in coolant.

Coolant flow is actually in natural convection flow. But due to the difficulties of flow calculation on various power level, heat flux distribution and operating conditions, forced flow is assumed and fixed as constant over all conditions. The flow velocity gets from standard TRIGA Mark III, safety analysis report at power level 2000 kW is 18.2 cm/sec. This flow was used for convenience in this calculation and is at a maximum power level in steady state condition.

Cooling system have 2 loops; primary and secondary. The primary cooling system is the water in pool where reactor core sunk in . The pool contains demineralized water with capacity around 153,000 liters. This system is directly contact with reactor core and irradiated by neutron flux. Some contaminated elements in water will become radioactive. Diffuser pump will slow down flow of radioactive elements to pool surface by spiralize flow.

Secondary cooling loop will cool down the primary loop by non-contact heat exchanger. Cooling tower dissipate heat to environment. The ambient temperature and humidity might effect the return flow temperature. Constant secondary loop temperature is assumed to ease the calculation.

Heat transfer coefficient is assumed to be constant over the whole operating power and conditions. One important factor to define heat transfer coefficient is the coolant velocity which is assumed to be fixed as mentioned before.

Heat conduction along the fuel pin length is assumed to be negligible. All heat generated will only be transfer in radial axis.

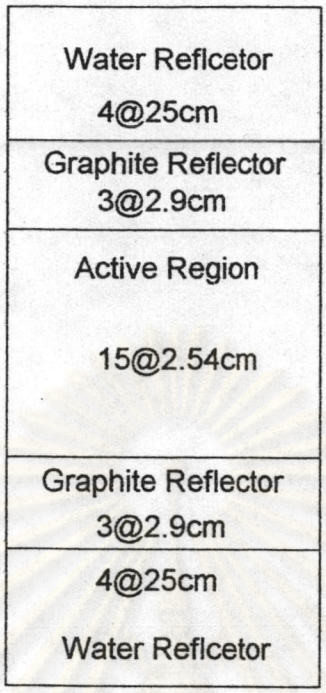


Figure 4.1 Regions in fuel pin in neutronic model

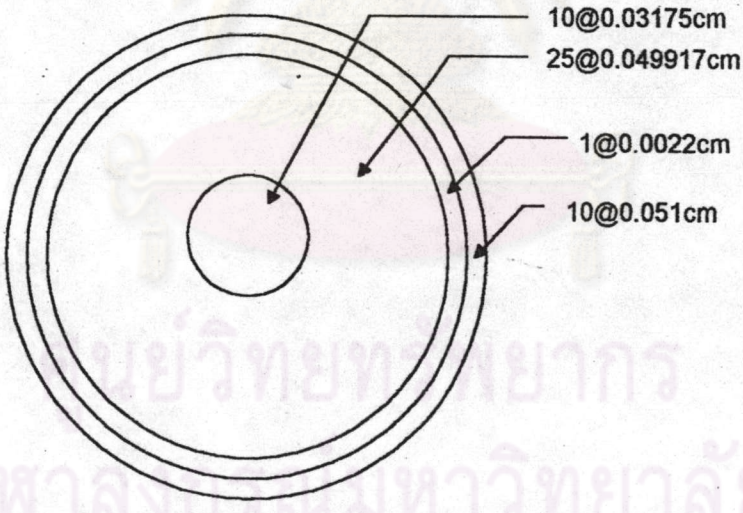
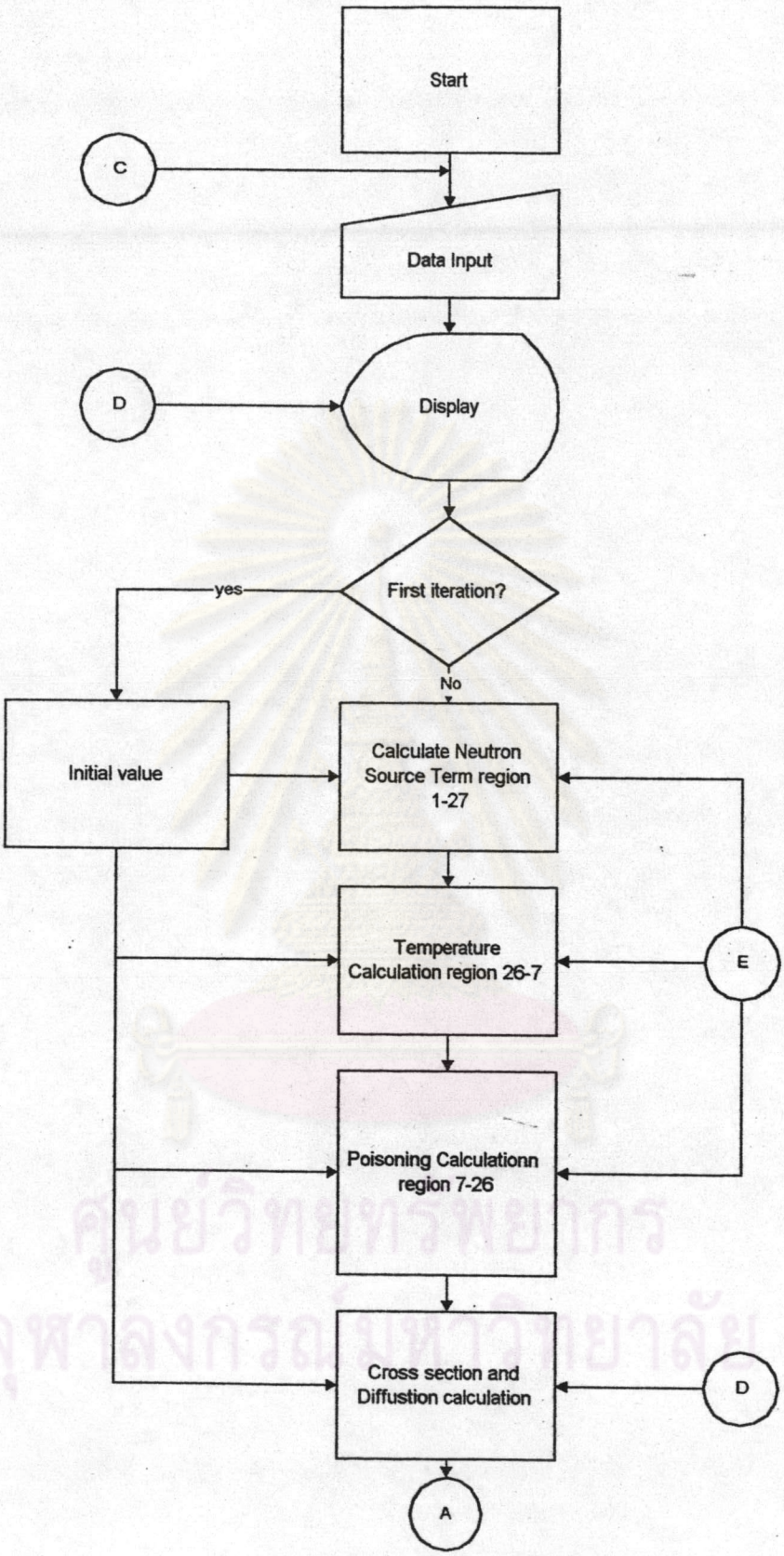


Figure 4.2 Regions in fuel pin in thermal-hydraulic model



Cassim Block programming Page 1

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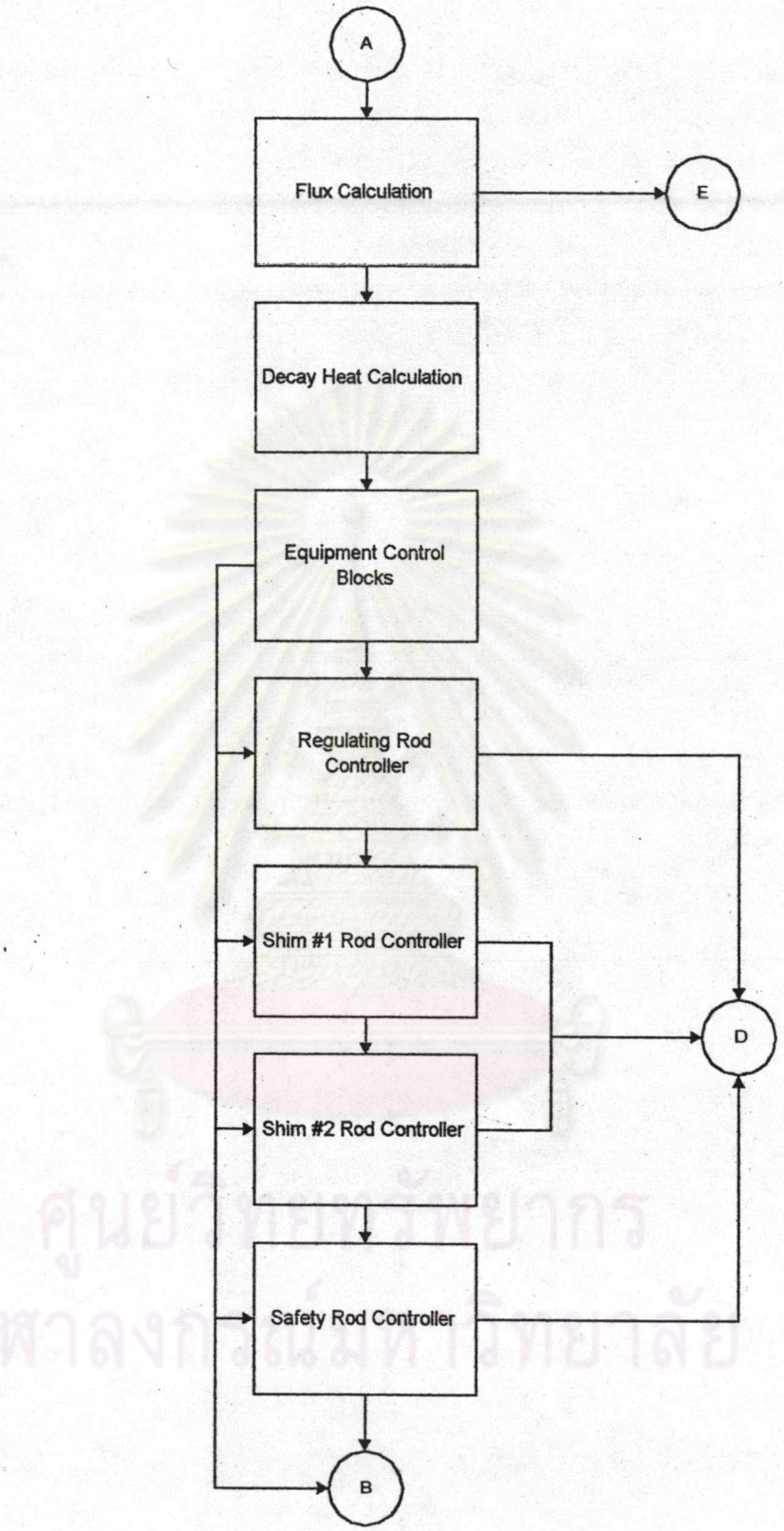

Cassim Block Programming page 2

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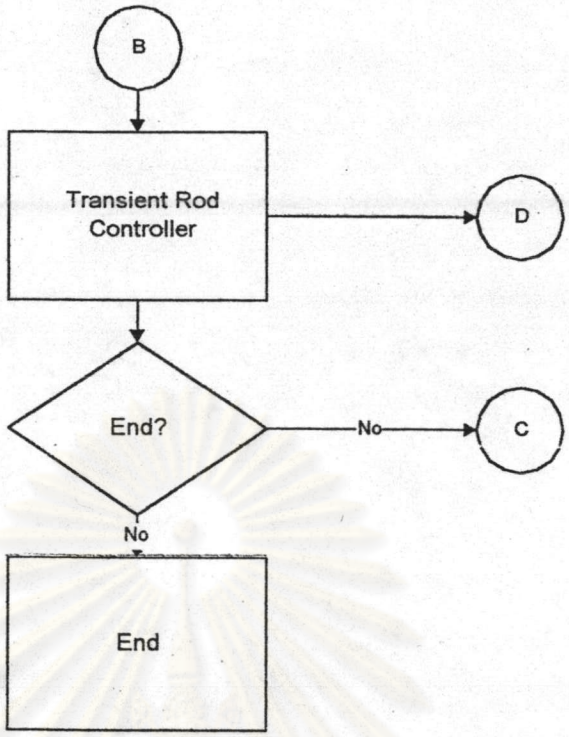

Figure 4.3 Block diagram of neutronic & thermal hydraulic calculation model

4.4 Merging neutronic and thermal hydraulic model

The neutronic model get feedback from thermal hydraulic model with a factor named as " temperature coefficient". As mentioned in the previous chapter, this reactor has this value in minus sign. Whenever fuel temperature increase, the core reactivity decrease.

If starting with a certain initial condition, we would know the atoms density in each region. Then the fission cross section, absorption cross section and diffusion coefficient are calculated to be used in the diffusion equation. After that neutron flux is solved throughout the matrix form of known parameters in equations. The by-product of fission reaction is energy in heat form which cause a temperature change in fuel pin. Changing in fuel-pin temperature cause changing in fission cross section which shall effect the fission rate in the next iteration.

This calculation is repeated till getting stop signal. Block diagram of the calculation model is shown on figure 4.3.

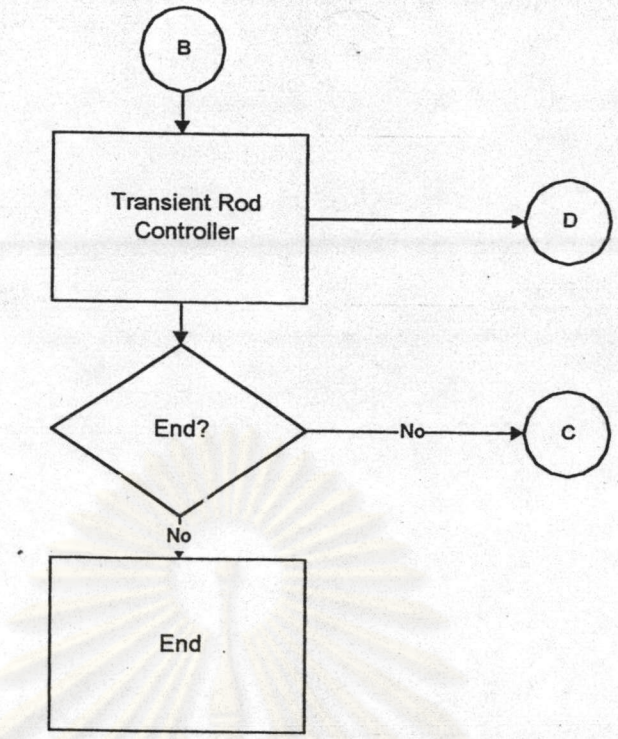

Figure 4.4 Block diagram of neutronic & thermal hydraulic calculation model

In the model development section, Cassbase® is the data file stored calculation sequence of model. TRRSIM 's blocks are classified into 6 groups which have different prefix. We start with AAA which is used as calculation, output and input blocks. The input and output interface blocks were placed at the beginning.

Some inputs are pulse signal which stay for only one iteration and will change state to the default state on the next iteration. The self reset switch block is used to deal with this pulse. The output interface blocks are just blocks gather calculation results from any other blocks in model. They will ease you while programming the interface module in the front end preparation.

Then neutron source term in regions are calculated with the initial conditions on the first iteration. Temperature calculation blocks are placed next to the neutron source term and again some parameters are getting from initial condition at the first iteration.

Fission products calculation blocks just follow the temperature calculation blocks. Then atoms density, absorption cross section and fission cross section in regions are considered taking parameters from the above calculation and rods position.

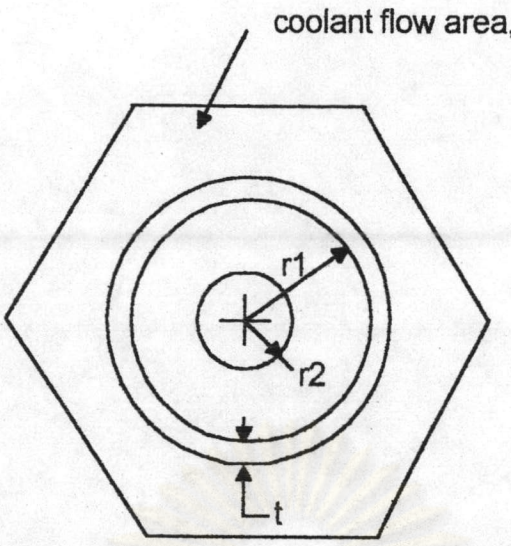

Figure 4.5 Enlarged one unit fuel-pin cell: the inner circle represent Zirconium rod, second ring represent fuel-moderator and the last ring is clad.

zirconium rod density	ρ_2	gram/cc.
hydrogen content in moderator (zirconium hydride)	X	%

If we assume that 1 cm. length of fuel pin is being considered

Total unit cell volume in 1 cm. length (V_{tot})	$\pi \times (r_1+t)^2 + Acf$	cc.
fuel-moderator volume	$\pi \times ((r_1)^2-(r_2)^2)$	cc.
U-235 concentration	$W1 \times W2 \times \rho_1 \times \pi \times ((r_1)^2-(r_2)^2)/V_{tot}$	gram/cc.
U-235 atom density	$W1 \times W2 \times \rho_1 \times \pi \times ((r_1)^2-(r_2)^2)/V_{tot} \times N_A/235$	atoms/cc

In the same way, U-238, zirconium and hydrogen atom in fuel-moderator, central zirconium rod and coolant can be written as.

U-238 atom density	$W1 \times (1-W2) \times \rho_1 \times \pi \times ((r_1)^2-(r_2)^2)/V_{tot} \times N_A/238$	atom/cc.
zirconium in moderator atom density	$(1-W1) \times (1-X) \times \rho_1 \times \pi \times ((r_1)^2-(r_2)^2)/V_{tot} \times N_A/91.22$	atom/cc.
hydrogen in moderator atom density	$(1-W1) \times X \times \rho_1 \times \pi \times ((r_1)^2-(r_2)^2)/V_{tot} \times N_A$	atom/cc.
zirconium in central atom density	$\pi \times (r2)^2 /V_{tot} \times \rho_2 \times N_A/91.22$	atom/cc.
coolant(water) in flow channel	$Acf/V_{tot} \times N_A/18$	atom/cc.

Stainless steel cladding is alloy metal consisting of 3 main ferrous elements; Cr 18%, Ni 8% and the rest is Fe. It's physical properties is 7.817 gram/cc. in density, 16.3 W/m C in thermal conductivity and specific heat 0.46 kJ/kg C.

Fe atom density in cell	$\pi \times ((r1+t)^2-(r1)^2) \times 7.817 \times 0.74 \times N_A/55.85/V_{tot}$	atom/cc.
Cr atom density	$\pi \times ((r1+t)^2-(r1)^2) \times 7.817 \times 0.18 \times N_A/52/V_{tot}$	atom/cc.
Ni atom density	$\pi \times ((r1+t)^2-(r1)^2) \times 7.817 \times 0.08 \times N_A/58.71/V_{tot}$	atom/cc.

The above atom density calculation is an isolating unit cell without any disturbance from control rod. The control rods occupy the same cell area as fuel pin does (Acf) but differ in pin size and chemical compositions. They all contain Boron Carbide which is neutron absorber. Changing in rods position will affect absorber concentration in region and as a result will affect neutron flux and neutron flux distribution in reactor core.

If we assume the total cell number in TRR 1/M1 is 100 units, position changing in one control rod will be averaged all over the cells. There are four control rods having fuel-moderator follower and one with air void follower. Consider fuel-follower in figure 4.6 and figure 4.7.

Control rod cell consists of water, clad, fuel moderator, zirconium rod and absorber. In the same way as done in fuel moderator cell, the atoms density are defined.

coolant flow channel area Ac,cr cm^2

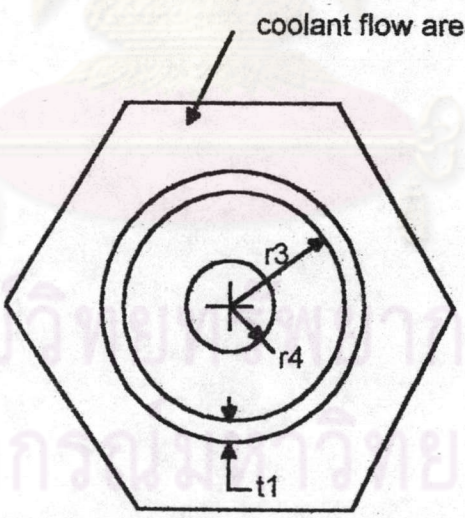

Figure 4.6 Fuel-follower control rod lower region

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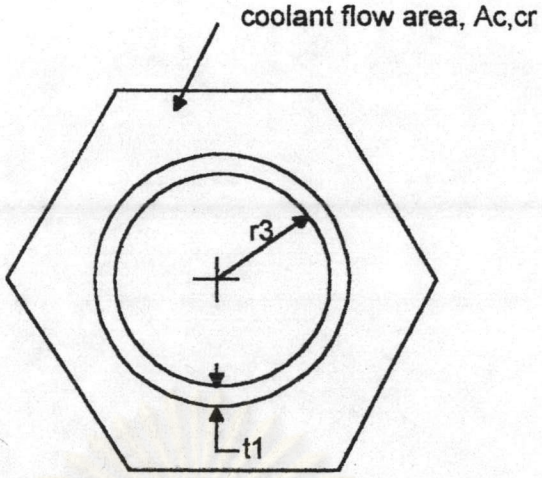

Figure 4.7 Fuel-follower control rod upper region

Assume that the 1 cm in axial length is considered with partial insertion of absorber in Y fraction.

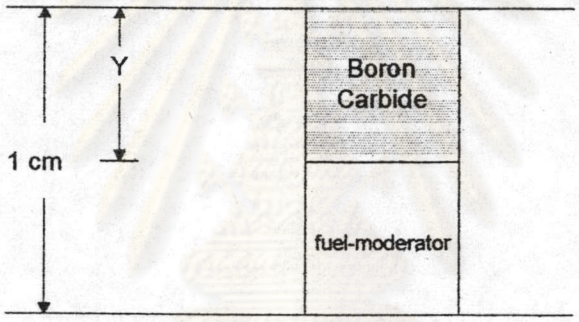

Figure 4.8 Partial Insertion of control rod in region

In the lower section (fuel-moderator)

central zirconium rod radius	r4	cm
fuel-moderator rod radius	r3	cm
clad thickness	t1	cm
Uranium content by weight in fuel-moderator	W_1	%
Uranium-235 enrichment	W_2	%
fuel-moderator density	ρ_1	gram/cc.
zirconium rod density	ρ_2	gram/cc.
hydrogen content in moderator (zirconium hydride)	X	%

in the upper region

absorber outer radius	r3	cm
absorber rod density	ρ_3	gm/cc
Total unit cell volume in 1 cm. length (V_{tot})	$\pi \times (r3+tl)^2 + Ac,cr$	cc.

In fact the total cell volume of control rod and fuel-moderator is exactly the same but differ in the coolant flow channel area.

fuel-moderator volume	$\pi \times ((r3)^2-(r4)^2) \times (1-Y)$	cc.
U-235 atom density	$W1 \times W2 \times \rho_1 \times \pi \times ((r3)^2-(r4)^2) \times (1-Y)/V_{tot} \times N_A/235$	atoms/cc
U-238 atom density	$W1 \times (1-W2) \times \rho_1 \times \pi \times ((r3)^2-(r4)^2) \times (1-Y)/V_{tot} \times N_A/238$	atom/cc.
zirconium in moderator	$(1-W1) \times (1-X) \times \rho_1 \times \pi \times ((r3)^2-(r4)^2) \times (1-Y)/V_{tot} \times N_A/91.22$	atom/cc.
hydrogen in moderator atom density	$(1-W1) \times X \times \rho_1 \times \pi \times ((r3)^2-(r4)^2) \times (1-Y)/V_{tot} \times N_A$	atom/cc.
zirconium in central atom density	$\pi \times (r4)^2 \times (1-Y) \times \rho_2/V_{tot} \times N_A/91.22$	atom/cc.
Boron Carbide volume	$\pi \times (r3)^2 \times Y$	cc.

Boron Carbide consists of 4 atoms of Boron and one of Carbon with molecular weight 52.255.

Boron atom density	$\pi \times (r3)^2 \times Y \times \rho_3 \times 4 \times N_A/52.255/ V_{tot}$	atom/cc.
Carbon atom density	$\pi \times (r3)^2 \times Y \times \rho_3 \times N_A/52.255/ V_{tot}$	atom/cc.
coolant(water) in flow channel	$Ac,cr/V_{tot} \times N_A/18$	atom/cc.

Control rods' clad are made from stainless steel 304 which is as the fuel-moderator pin.

Fe atom density in cell	$\pi \times ((r3+t)^2-(r3)^2) \times 7.817 \times 0.74 \times N_A/55.85/V_{tot}$	atom/cc.
Cr atom density	$\pi \times ((r3+t)^2-(r3)^2) \times 7.817 \times 0.18 \times N_A/52/V_{tot}$	atom/cc.
Ni atom density	$\pi \times ((r3+t)^2-(r3)^2) \times 7.817 \times 0.08 \times N_A/58.71/V_{tot}$	atom/cc.

Once the elements concentration of cell are calculated, we have to divide the above value by 100. This is because only one control rod is used in the above equations. All the fuel-follower control rod use the same equations with the same physical properties and chemical properties.

The last control rod is transient rod. It consists of air void in the upper part and absorber in the lower part. The main purpose of this rod is for pulsing. After pulse we will get an extremely high neutron flux. Air void decrease the rod' weight and make the faster step increase on lifting up rod. Elements

concentration can be written in the same way as we did in fuel-follower control rod except there are not any Zirconium, Uranium and Hydrogen.

Elements concentration are used to calculate macroscopic cross section just multiply them with their own neutron microscopic cross section.

4.6 Diffusion Coefficient (D)

The diffusion coefficient is used in the diffusion equation and defined as:

$$D = \frac{1}{3\left(\Sigma_a + \left[1-\overline{\mu}_0\right]\Sigma_s\right)} \qquad (56)$$

where Σ_a = neutron absorption cross section

Σ_s = neutron scattering cross section

$\overline{\mu}_0$ = average scattering angle cosine

In the reactor consisting of various elements, diffusion coefficient could be calculated by replace the absorption cross section by the total absorption cross section of all elements. The second term, $\left[1-\overline{\mu}_0\right]\Sigma_s$, replace with the summation of this value on all elements. The value of $\overline{\mu}_0$ in the second term could be written in equation as $\frac{2}{3A}$ which A is the atomic weight of elements. For example, in the active region of this model:

In case of without any disturbance from control rod insertion:

$$\Sigma_a^{tot} = \Sigma_{a,U28} + \Sigma_{a,U25} + \Sigma_{a,Zr} + \Sigma_{a,H} + \Sigma_{a,Fe} + \Sigma_{a,Ni} + \Sigma_{a,Cr} + \Sigma_{a,H_2O} + \Sigma_{a,Sm-149} + \Sigma_{a,Xe-135}$$

$$\begin{aligned}\left[\left(1-\overline{\mu}_0\right)\Sigma_S\right]_{tot} &= \left(1-\overline{\mu}_{0,U28}\right)\Sigma_{S,U28} + \left(1-\overline{\mu}_{0,U25}\right)\Sigma_{S,U25} + \left(1-\overline{\mu}_{0,Zr}\right)\Sigma_{S,Zr} + \left(1-\overline{\mu}_{0,H}\right)\Sigma_{S,H} \\ &+ \left(1-\overline{\mu}_{0,Fe}\right)\Sigma_{S,Fe} + \left(1-\overline{\mu}_{0,Ni}\right)\Sigma_{S,Ni} + \left(1-\overline{\mu}_{0,Cr}\right)\Sigma_{S,Cr} + \left(1-\overline{\mu}_{0,H_2O}\right)\Sigma_{S,H_2O} \\ &+ \left(1-\overline{\mu}_{0,Sm-149}\right)\Sigma_{S,Sm-149} + \left(1-\overline{\mu}_{0,Xe-135}\right)\Sigma_{S,Xe-135}\end{aligned}$$

If we do calculate diffusion in other situations and regions, the above procedure is kept but replace by elements contained in that place.

4.7 Heat Transfer Coefficient

From our assumption, the force flow has taken over natural flow. The coolant velocity is known at the point of 2 MW theraml at steady state condition. This value was used for all over the reactor power level. The heat transfer coefficient would be found from equation:

$$Nu = C\,Re^{0.8}\,Pr^{\frac{1}{3}} \tag{57}$$

where

$$Nu = \text{Nusselt number} = \frac{hD_e}{k} \tag{58}$$

$$Re = \text{Reynolds number} = \frac{D_e V \rho}{\mu} \tag{59}$$

$$Pr = \text{Prandtl number} = \frac{C_p \mu}{k} \tag{60}$$

The constant C depend on the lattice arrangement given by

$$C = 0.026\frac{S}{D} - 0.006 \tag{61}$$

S = distance between rod center

D = diameter of the individual rods

h = heat transfer coefficient, $W/cm^2.C$

D_e = equivalent diameter of noncircular flow channel, cm

k = thermal conductivity of coolant, W/cm.C

V = flow velocity, cm/s

ρ = coolant density, gm/cc.

μ = absolute viscosity of coolant, gm/s.cm

Cp = specific heat of coolant, Joule/gm.C

For the TRR 1/M1 model

$$D_e = \frac{4A_c}{P} \qquad (62)$$

where

Ac = coolant flow channel area, cm2 = 5.39 cm2

P = wetted perimeter of the channel, cm = $\pi(3.73) = 11.718$ cm.

$$D_e = \frac{4(5.39)}{11.718} = 1.84\ 0 \text{ cm}$$

V = 18.2 cm/s

The water properties are taken at bulk temperature. From the safety analysis report of TRIGA Mark III mentioned that at 2 MW thermal in steady state condition, the average bulk temperature is 57.2 C. The status of coolant is subcooled liquid.

k = 0.38095 Btu/hr.ft F

= 6.594×10^{-3} W/cm C

ρ = 61.662 lbm/ft3

= 0.987 gm/cc.

μ = 1.199 lbm/hr.ft

= 4.957×10^{-3} gm/s.cm.

Cp = 0.9938 Btu/lbm.F

= 4.187 Joule/gm.C

$$\text{Re} = \frac{1.840 \times 18.2 \times 0.987}{4.957 \times 10^{-3}} = 6667.9$$

$$\text{Pr} = \frac{4.187 \times 4.957 \times 10^{-3}}{6.594 \times 10^{-3}} = 3.148$$

S for TRR 1/M1 = 4.34 cm

D for TRR 1/M1 = 3.73 cm

$$C = 0.026\frac{4.34}{3.73} - 0.006 = 0.024$$

$$\text{Nu} = 0.024(6667.5)^{0.8}(3.148)^{\frac{1}{3}} = 40.31$$

$$h = \frac{Nu.k}{D_e} = \frac{40.31(6.594x10^{-3})}{1.84} = 0.144\ W/cm^2.C$$

4.8 Delayed neutron constant

The delayed neutron is part of the neutron souce and devided in to 6 group depending on its half life. The fission yield of each group are also difference group by group.

Table 4.1 delayed neutron group constant (from data sheet by OAEP)

delayed neutron group	fission yield	decay constant
1	0.0002774	.0134566
2	0.0015549	0.03173751
3	0.0013724	0.11552453
4	0.00297110	0.31082833
5	0.0009344	1.39747415
6	0.00018980	3.78233062

4.9 TRRSIM programming

This model is a combination of software; Cassbase®, Casseng®, MS Fortran and Labview®. It start with block structure modeling in Cassbase®. Fortran source code is written as sub-routine and save as executable algorithms stored in Dynamic Link Library (DLL).The stored algorithms will be used by Casseng® to calculate any equations or logical control.

The front end of simulation take calculation results from Casseng® and display on Labview® which is graphical interfaced. The calculation engine pick up command from Labview® front end to change operating condition.

These programs is running on Window® base.

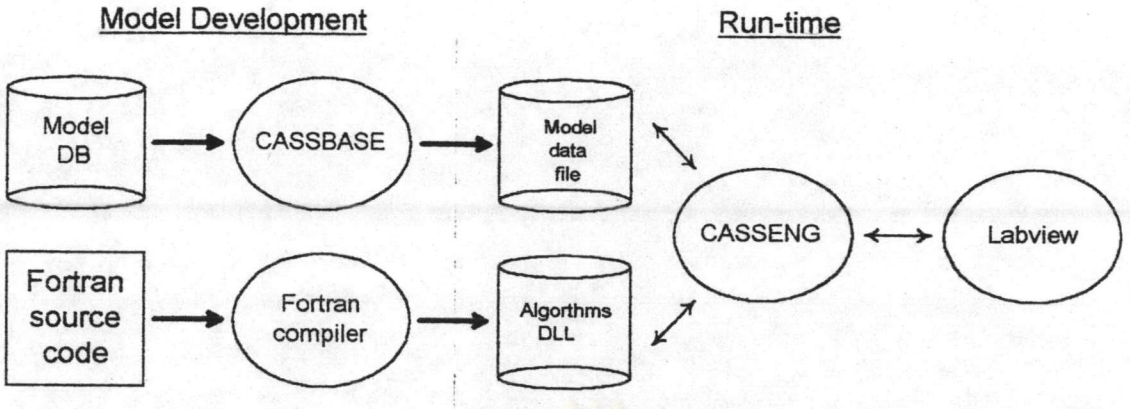

Figure 4.9 Schematic Diagram of TRRSIM

In the model development section, Cassbase® is the data file stored calculation sequence of model. TRRSIM 's blocks are classified in 6 groups which have different prefix. We start with AAA which is used as calculation, output and input blocks. The input and output interface blocks were placed at the beginning.

Some inputs are pulse signal which stay for only one iteration and will change state to the default state on the next iteration. The self reset switch block is used to deal with this pulse. The output interface blocks are just blocks gather calculation results from any other blocks in model. They will ease you while programming the interface module in the front end preparation.

Then neutron source term in regions are calculated with the initial conditions on the first iteration. Temperature calculation blocks are placed next to the neutron source term and again some parameters are getting from initial condition at the first iteration.

Fission products calculation blocks just follow the temperature calculation blocks. Then atoms density, absorption cross section and fission cross section in regions are considered taking parameters from the above calculation and rods position.

The flux calculation is using parameters from neutron source blocks and atom density block to calculate neutron flux in the next iteration. These flux will be used in the neutron source blocks, temperature calculation blocks and fission product blocks to recalculate flux in the next iteration.

Heat getting from fission is partially from decay of radioactive fission products. It would share up to 8% of the total heat generated. The total heat emitted is the sum of decay heat with instantaneous heat. These blocks are placed after the flux calculation block. Equipment interface blocks, logical control blocks and some calculation blocks are computed in the last section of AAA prefix group.

The second group is CAA prefix. These blocks represent controls of equipment such as pumps, power supply, trip signal, up-down control rods. They are mostly flip flop switches and convert pulse from

self reset switch to constant signal. The constant signal would change its state if it get the state change pulse.

Regulating rod group represent by RE prefix is next to the CAA prefix group. As mentioned by name, they are dealing with regulating rod. Most of all are logical control of which condition regulating rod should move in which direction. Various rod speed are selected from conditions to conditions.

The S1, S2, SA and TR groups stand for shim #1, shim #2, safety and transient rod groups. Their functions are exactly the same as RE prefix group but differ in rods.

Till end of the calculation model, It will repeat calculation of the first group with the parameters getting from the previous condition, not from initial conditions.

4.10 TRRSIM front end and calculation engine interface

The front end in Labview® will regcognize blocks and sequences of data to communicate with in Casseng®. We need to identify in the data aquistion fuction where they are able to get the right calculation value. To pick up command from the front end, just do the same as reading data. Exact blocks and sequences must be identified to transfer data.

The programming section of front end open up the connection with calculation engine by function written by Cassiopiea named as Dynamic Data Exchange (DDE) Open function. This connection is 2 way communication. Each specific type of data are catagorized by name. Then bundles of data are unbundled and distributed to the display function.

While retriving data from the calculation engine after connection open up, we must identify block number, sequence, starting block and end block to pick up data.

Just to ease us to interface with Cassbase®, block programming, the non-movable block was written to cope with this problem. It collects data from blocks in the model without processing. The output data are the same as input. When mentioned exact position to pick up data, It is quite easier to specify the succesive data than mention many blocks and data sequences.

All commands are shown in graphical pictures and link or transfer data via a connection named as wire. The wire represent the type of data being transfered or connected such as green wire stand for logical data, orange wire represent real data, blue line is integer data.

The front end displays are selectable. We may show them on chart, digital display, circular meter, bar, knob with the same real or interger data.

For Example, we consider block number 10 which is the display block. The output sequence from 1 to 11 are collected from many parts of the model. They are distributed in many kind of display such as neutron flux in sequence #1 is in digital display, fuel temperature in sequence #7 is in regtangular meter, bulk water temperature is in chart.

As shown in Figure 4.12, data in block #10 from sequence #1 to #11 are reading in the front end. Block #10 in Cassbase® is non-movable block. Input data would pass directly to output.

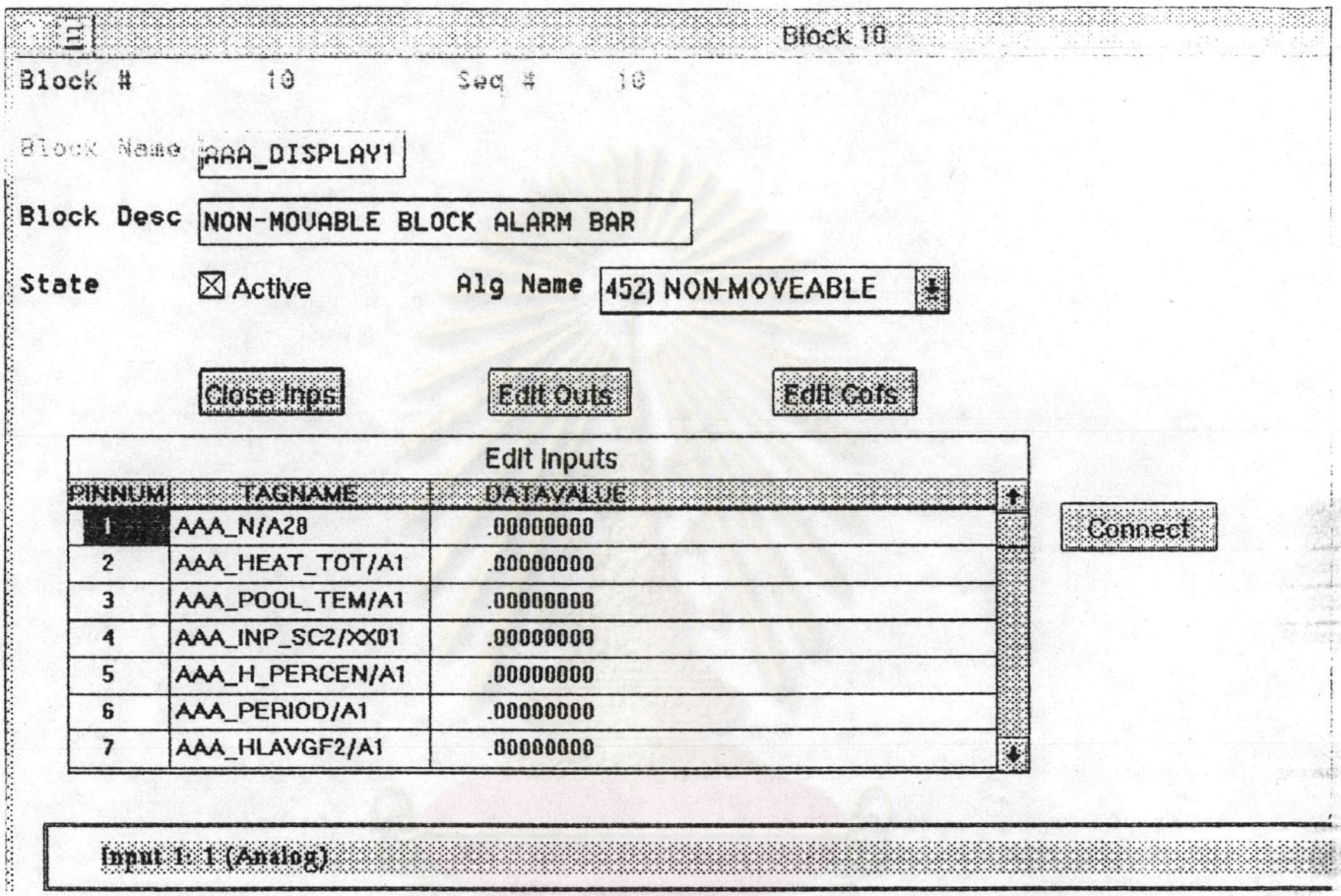

Figure 4.10 Cassbase® block #10 show partial of input data

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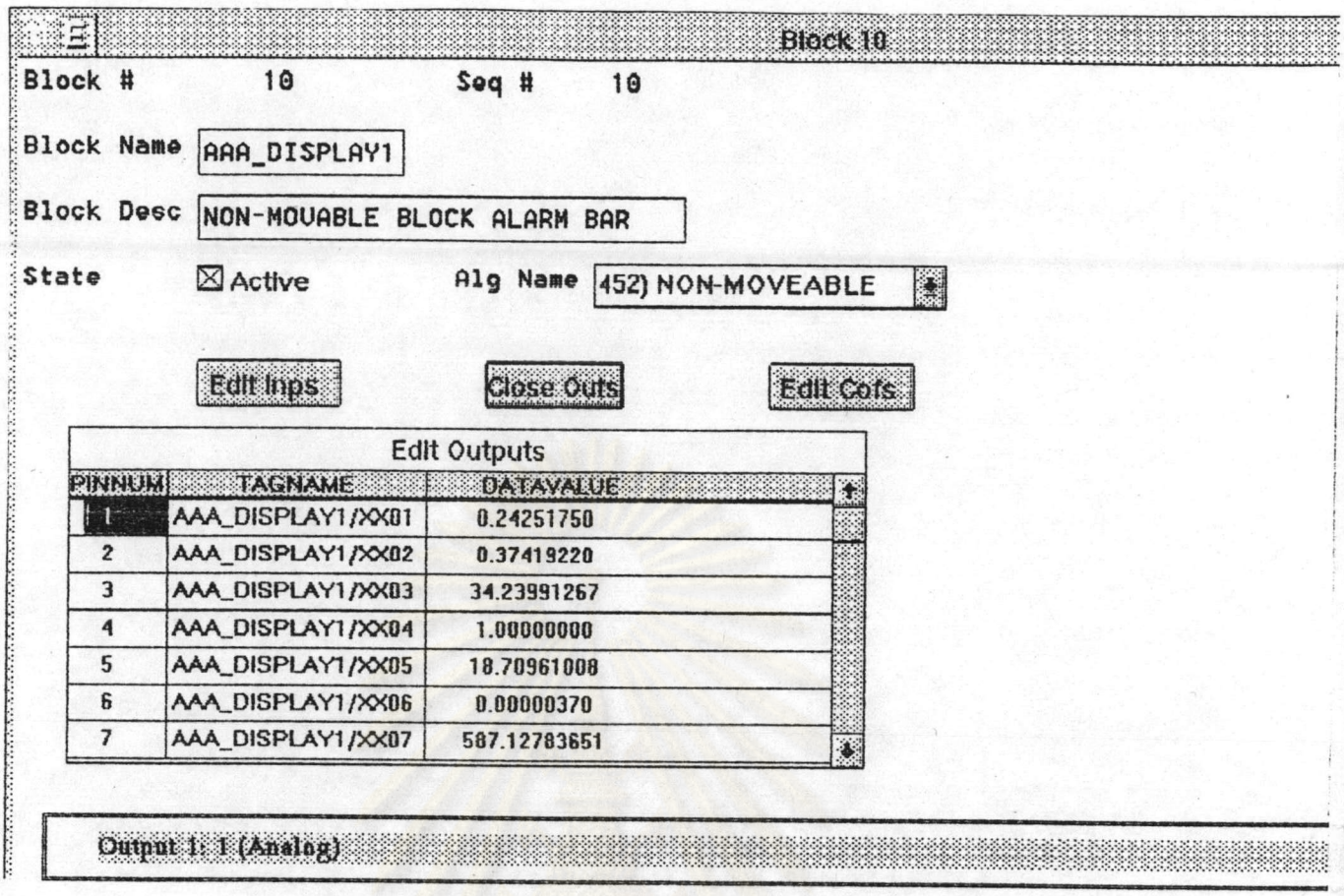

Figure 4.11 TRRSIM block #10 show partial of output data

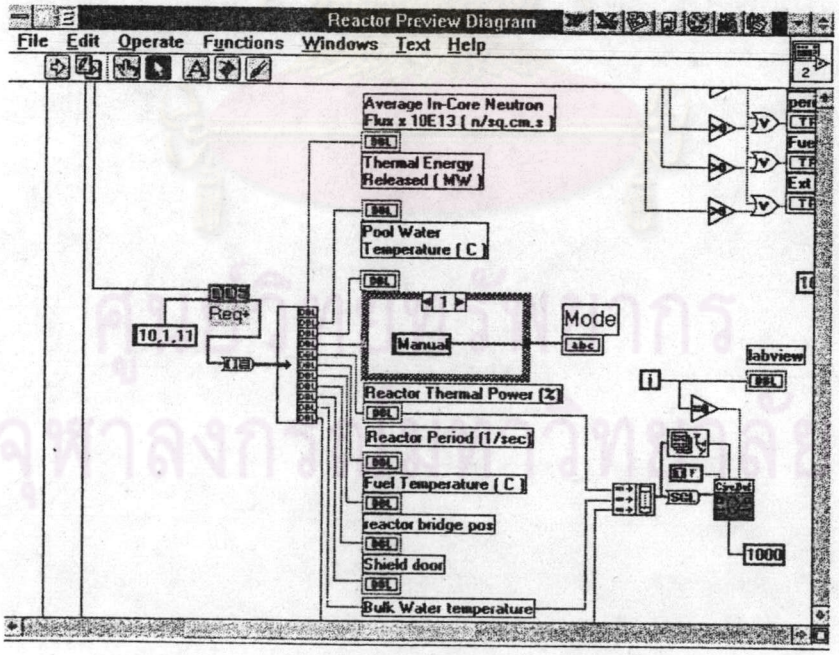

Figure 4.12 programming section in Labview front end

If we need to change the operating conditons from front end, The DDE write fuction was used. There are 2 type of data to write: analog and digital. As we did in reading data, exact postion to write must be identified. We could either write in input block or output block which up to the objective and block algorithm.

When ending the DDE connection, DDE close fuction was placed at the end to close connection.

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