

Chapter 5

SIMULATION MODES: EXAMPLES AND DISCUSSIONS

The flow chart of the DNRR simulation code is shown in Fig. 5.1. The code listing is given in Appendix C.

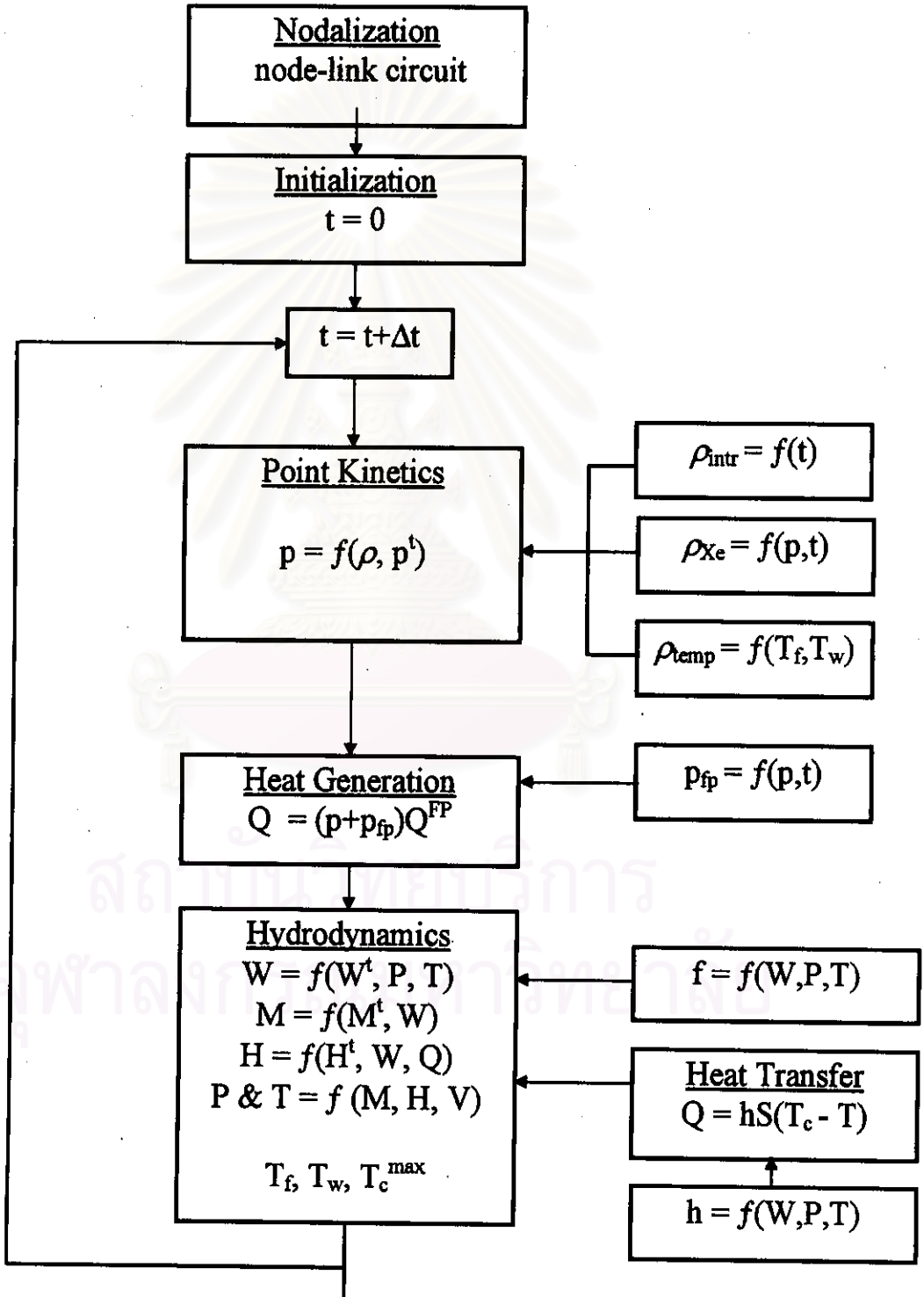


Fig. 5.1. Flow chart of the DNRR simulation code

The code begins with the formation of nodes and links for the reactor hydraulic network as described in Chapter 3. Here, every node and link is assigned with geometry parameters such as node centre, volume, heat transfer area as well as link flow area, length, hydraulic diameter, resistance, etc., most of which do not change during run-time.

Next, the code initializes all time-dependent parameters. Traditional initialization is made at some given steady-state condition. It must be noted that, except for the first start-up of the reactor, there is only a steady-state approximation for thermal-hydraulics, when heat generation is approximately balanced with the heat transported by the coolant flow. As temperature of the vast amount of pool water and the inventory of fission products, affecting decay heat fraction, are changing slowly with time, it will take several hours to reach so called real steady-state conditions if power, pump flow and inlet coolant temperature are maintained constant. However, the code can restore parameters from a file at a time when these parameters were saved during a transient. Thus, it allows the simulation to continue the transient from any time without starting from the beginning if all current neutronic and thermal-hydraulic parameters have been saved for that time.

Two simulation modes were investigated, the first was a power change and the other one a reactivity change. For the power change mode, the reactor power is given as a function of time, usually as an exponential with a constant period. The reactor is always critical in this mode. That is a very practical case, when power is usually changing with sufficiently large period (more than 30 sec for the DNRR). In this case, the reactivity change is too small and it is not worth solving the point kinetics.

In the reactivity change mode, an external reactivity is inserted as a function of time and reactivity feedback is considered. For example, a large amount of positive or negative reactivity is inserted by control rods' movement in the reactor core during its start-up and shut-down, or when an irradiation sample is fired in and out by the pneumatic system. Actually, the reactor automatic control system and safety protection system always respond to any change in reactivity and will trip the reactor when the period is less than 30 sec. and/or the power is more than 110%. As the simulation of the reactor control system is beyond the scope of this work, we will investigate reactivity transients in cases when the reactor is on manual control regime (no automatic regulation) and the reactor protection system is switched off or fails. However, a simple shut-down of the reactor by scram power signal can be simulated.

In either of these two modes, pump flow and coolant temperature at the inlet into the reactor pool are given as input with time. Thus, pump flow can range from zero to full capacity of the pump. Inlet coolant temperature can be constant or changed during a transient. Two cases attracting more attention are when the secondary cooling loop is not working and when the reactor is in self-cooling mode. In both cases, the inlet coolant temperature is taken as the upper pool water temperature.

In this simulation, reactor power is taken as equal to neutron power. The reactor thermal power then consists of fission heat, proportional to neutron power, and the decay heat of fission products. The latter is tracked dynamically. Thus, the total heat deposited in the reactor fuel consists of approximately 95% of fission heat plus all decay heat. The other 5% of fission heat takes place in the coolant/moderator, i.e. the water.

Most of the code running time is spent in solving hydrodynamic equations and calculating water properties. The hydrodynamics also dictates the size of time step for the entire code. Actually, the larger the number of nodes, the better the accuracy of the simulation. However, the time step size is inversely proportional to the total number of nodes. In the following examples, the reactor circuit is divided into 10 core nodes plus 3 other nodes of the chimney, upper pool and lower pool. Time step for normal operations is typically about $\Delta t \approx 10^{-4}$ sec. For reactivity transients at very high or very low power levels, it can be reduced to as small as $\Delta t = 10^{-6}$ sec.

Heat transfer always satisfies the conditions for applying the lumped form, i.e. $Bi < 0.1$, as described in Chapter 4, section 4.4. Correlations for heat transfer coefficient and, similarly, friction coefficient calculations are validated beforehand by comparison with experimental data.

In order to analyze the reactor safety, hot channel calculations will give the maximum fuel surface temperature as well as the margins to ONB and DNB.

5.1. Power Mode

5.1.1. Power changes with a constant period

Fig. 5.2 shows the most important thermal-hydraulic parameters of the reactor during transients changing from $p_0 = 50\%$ at steady-state conditions to $p = 100\%$ by step and by exponential with a period $\tau \neq 0$ as

$$p(t) = p(0)\exp(t/\tau).$$

Maximum fuel surface temperature (T_{cmax}) reacts immediately to power change and reaches a peak in less than 20 sec. after the power has stopped changing. Then it stabilizes at a temperature that is about 2°C lower than the peak. Coolant temperature (T_2) at the outlet from the core behaves similarly to T_{cmax} but the change is much smaller. It can be seen that the normal operating conditions could be satisfied. Core flow (W_{core}) lags behind reactor power. It takes 1.5-2 min. for W_{core} to be developed to the new stabilizing value after power has reached the new level. Also, with larger power changing period, all parameters keep following the power more closely, and the temperature peak can hardly be seen when τ is greater than 60 sec.

Table 5.1. Maximum temperature for power changes from 50 to 100%

Max. temperature	Step	$\tau = 30$ s	$\tau = 60$ s	Stabilized
Fuel surface	100.6 (21s)	100.4 (15s)	100.1 (13s)	98.2
Coolant at core outlet	49.1	48.9	48.4	45.5

Note: the value in parentheses is the time to peak after the power has stabilized.

The transients shown in Fig. 5.3 are from different power levels p_0 to $p = 100\%$ with the same period $\tau = 30$ s. Fuel surface temperature peak is higher with lower initial power level. However, as expected with $\tau > 30$ sec, any power maneuver of the reactor should not result in exceeding any safety limits.

Table 5.2. Maximum temperature for power changes with a period of 30 sec.

Max. temperature	50 to 100%	20 to 100%	0.5 to 100%	Stabilized
Fuel surface	100.4 (15s)	102.6 (12s)	105.3 (10s)	98.2

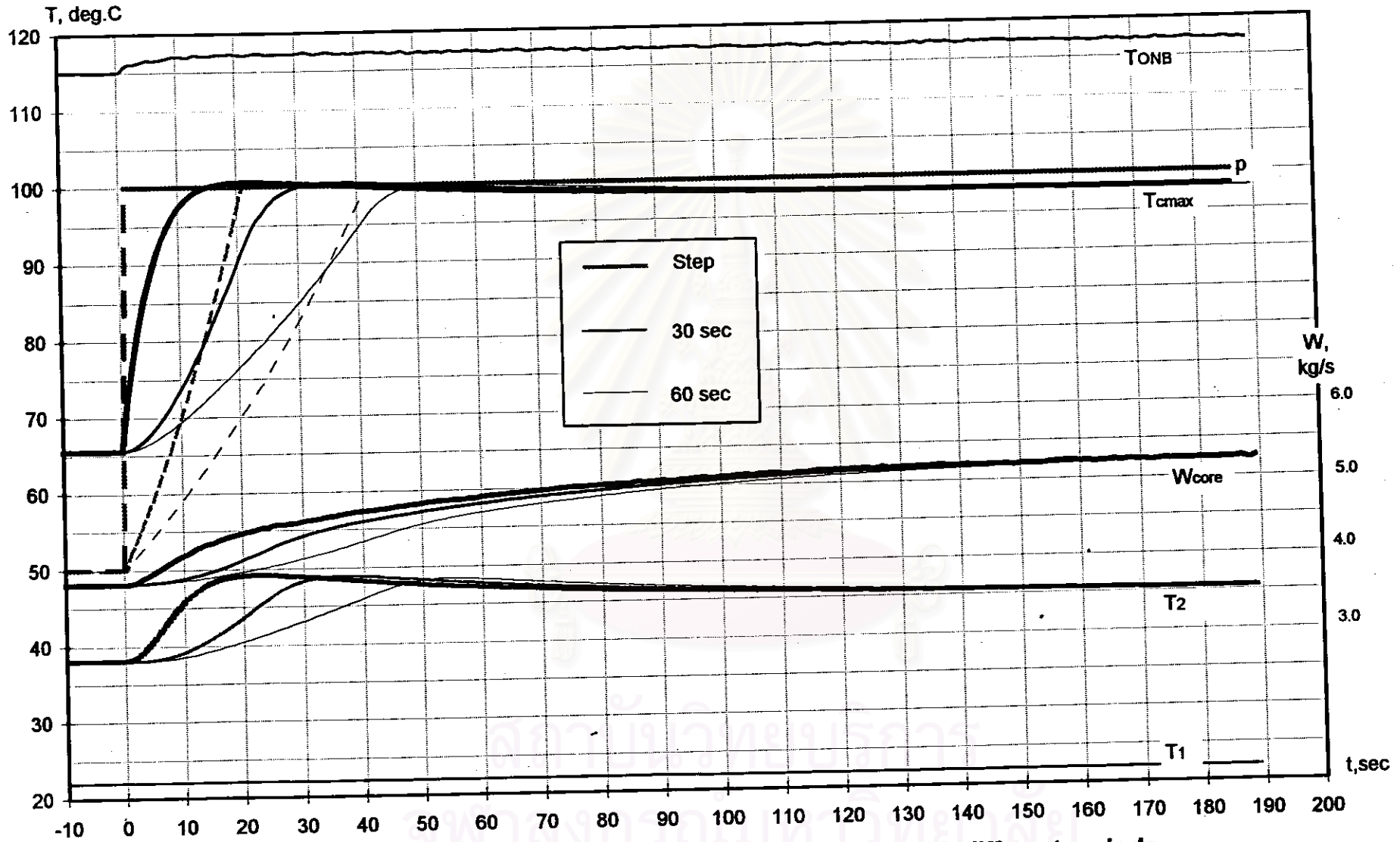


Fig. 5.2. Transients of power changing from 50% to 100% with different periods

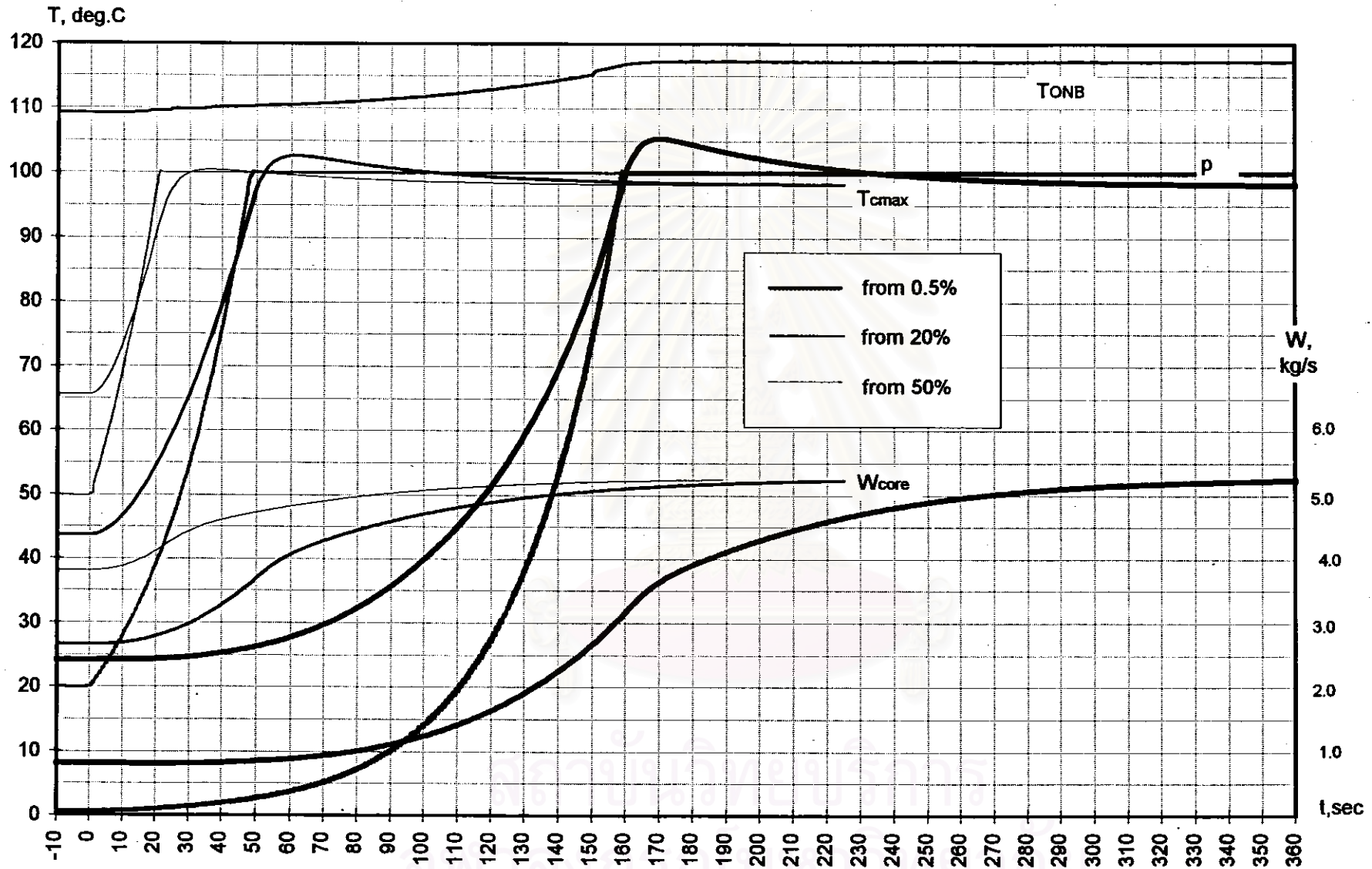


Fig. 5.3. Transients of power changing from different levels to 100% with period of 30 sec.

5.1.2. Pump and Inlet Coolant Temperature Effects

In Fig. 5.4, there are transients due to the changes of inlet coolant temperature (T_1) caused by changing pump flow (G) at constant power ($p = \text{const}$). It can be seen that the case of pump not working ($G = 0$, i.e. self-cooling) is not worse than the case of pump working ($G = 50 \text{ m}^3/\text{h}$) but without secondary cooling. That is, changing G does not affect in-core thermal-hydraulic parameters directly, but through slowly changing T_1 . Without heat transfer to the outside atmosphere, water temperature (T_{in}) at the inlet to the lower pool should be the same as water temperature in the upper pool. When T_1 is increased, W_{core} and T_2 as well as T_{cmax} increase.

Table 5.3. Effect of inlet coolant temperature at power level of 100%

$T_1, ^\circ\text{C}$	22	24	26	28	30
$T_2, ^\circ\text{C}$	45	47.2	48.9	50.5	52.1
$T_{\text{cmax}}, ^\circ\text{C}$	96.2	99.5	101.5	103.2	105
$W_{\text{core}}, \text{kg/s}$	6.19	6.22	6.33	6.45	6.56

Obviously, after some time, T_{cmax} will approach the safety limit points and normal operating conditions eventually are no longer satisfied. The allowable operation time of the reactor without external cooling will depend on the power level as well as the initial pool water temperature. In this example, from the start-up, the reactor can be operated at full power for about 7 min. without secondary cooling. As noted before, the reactor is controlled by the neutron power (p , %FP), then, due to increase of decay heat fraction (p_{fp}) of building-up fission products, T_{cmax} is increasing with time until equilibrium of the fission products is reached at constant reactor power level. This has to be taken into account because after several hours of operation, a significant fraction of up to $p_{\text{fp}} = 6\%$ of full power and warm-up cooling water altogether may cause T_{cmax} to overtake the maximum value for the aluminium cladding (105°C) and even reach the ONB.

Table 5.4. Effect of building-up fission products' decay heat

$p = 100\%$ $T_{\text{in}} = 22^\circ\text{C}$ $G = 50 \text{ m}^3/\text{h}$	Initially ($t \approx 60 \text{ sec}$)	Equilibrium ($t > 5 \text{ hrs.}$)
Decay heat fraction, p_{fp} , %	2	6
Max. fuel surface temperature, T_{cmax} , $^\circ\text{C}$	98	100.8

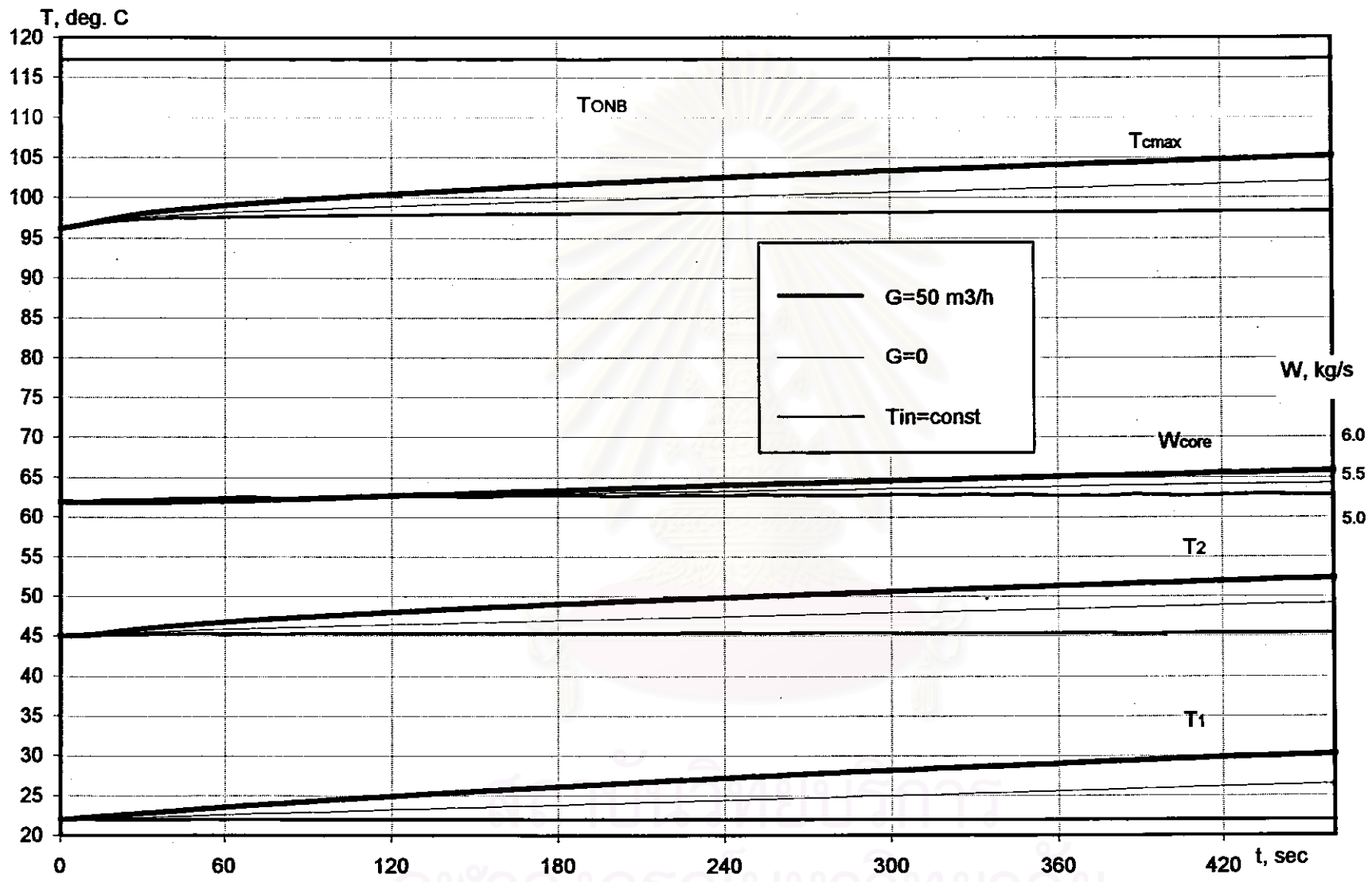


Fig. 5.4. Pump Flow and Inlet Coolant Temperature Effects

5.2. Reactivity Mode

In the reactivity changing mode, the reactor is initially in either subcritical ($\rho_0 < 0$) or critical state ($\rho_0 = 0$). A positive or negative reactivity amount is then inserted by step or by ramp. The neutron power (p) is obtained from the point kinetics. Reactivity feedback due to xenon poisoning and due to changes in core temperature is calculated dynamically. Unfortunately, because the void formation and its reactivity feedback are not studied in this thesis, whenever T_{cmax} reaches the ONB, further changes of power are no longer correct. However, if the void feedback is negative as assumed, the code will overestimate the thermal-hydraulic parameters, which will be indicated to be at a higher power than the actual. In actuality, the power should be lower than that given by the point kinetics and, consequently, calculated value for T_{cmax} will be conservative from a safety viewpoint. As the DNB is reached, T_{cmax} will increase drastically and may lead to deformation or melting of the cladding.

5.2.1. Positive Reactivity Insertion into the Critical Reactor

Fig. 5.5 shows the power (p) and T_{cmax} changing due to a step positive reactivity insertion. The reactor is initially critical at steady-state conditions of 100% power level. With $\rho < \beta$, in very short time of milliseconds the power rises drastically to the value close to the “prompt jump” approximation as

$$p \approx p_0 \frac{\beta}{\beta - \rho}$$

due to rapid increase of prompt neutrons. Then, when delayed neutrons gradually take control, the power rises exponentially with a nearly stable constant τ (reactor period). With $\rho \geq \beta$ (1\$), the reactor is critical on the prompt neutrons alone, i.e. prompt critical, and power would jump extremely high. The power dynamics obtained by the code is in very good agreement with the theory. Moreover, because multiple delayed neutron groups and reactivity feedback mechanisms are implemented, the results are better than the theoretical approximation.

Table 5.5. Step reactivity insertion at initial 100% power level

ρ , \$	1	0.5	0.3	0.1
p ‘prompt jump’, %	> 3000	200	140	110
Time to ONB, sec	0.14	1.60	5.08	long
Time to DNB, sec	0.14	2.09	6.63	never

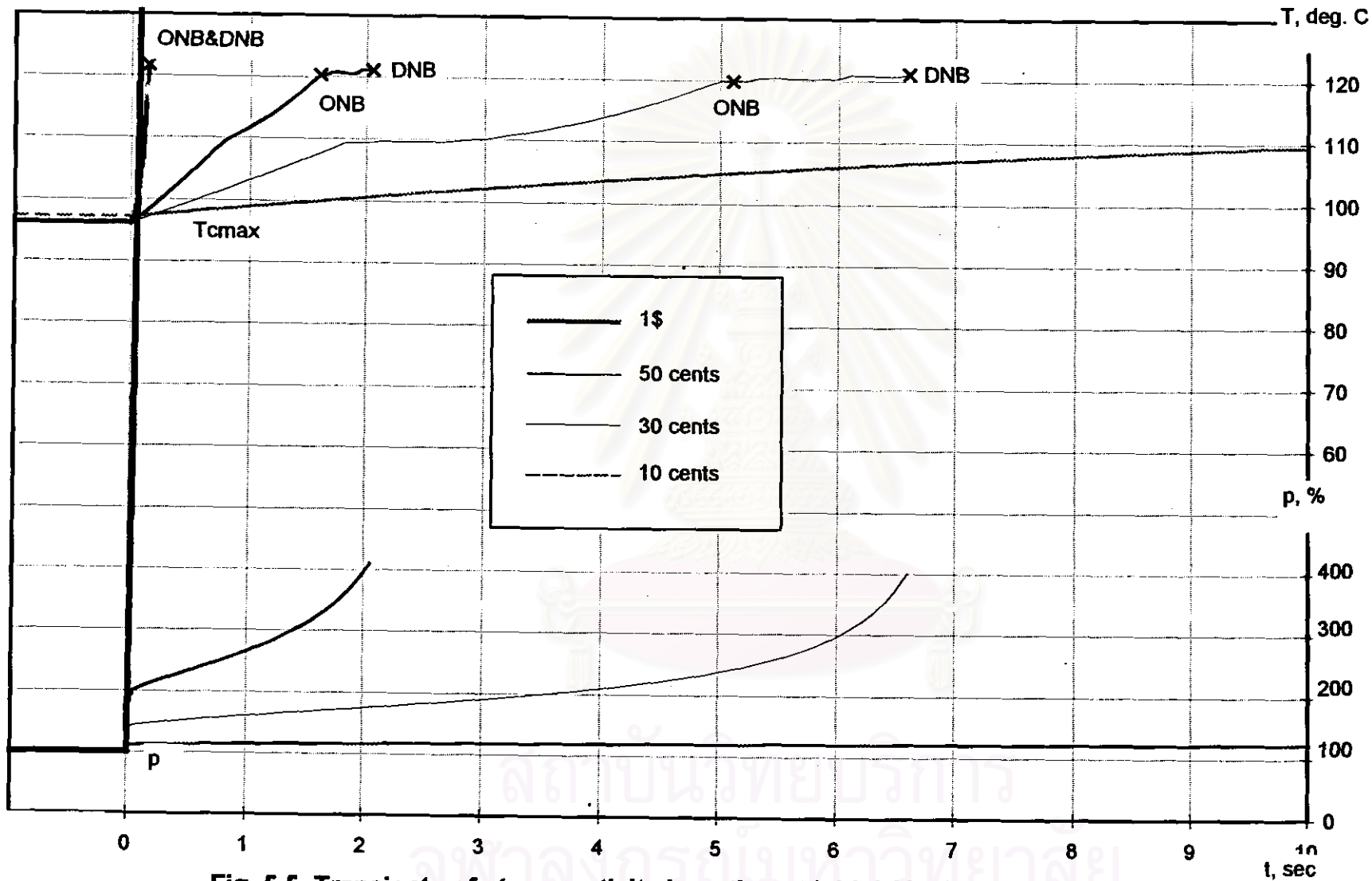


Fig. 5.5. Transients of step reactivity insertion at initial 100% power level

In the cases studied, sooner or later ONB always occurs. If the inserted reactivity is small enough, the ONB could take longer and the DNB would not come. That is because the heat flux is still less than the critical heat flux ($q_c < q_{CHF}$). As the change of fuel temperature is significant, the fuel temperature feedback will dominate the reactivity feedback effect. However, the fuel temperature coefficient is decreasing in absolute value with increasing temperature. Therefore, following the prompt jump of power, the power rate slows down due to the negative fuel temperature feedback. Then, this negative feedback is reduced due to fuel temperature increase, so the power rate will rise more rapidly. If the power exceeds than 400%, the DNB occurs very quickly just after the ONB.

Fig. 5.6 shows transients due to a ramp insertion of 1\$-reactivity at an initial 100% power level. With a ramp insertion of reactivity, there is no prompt jump of the power. But for the thermal-hydraulics, the pictures do not differ much from the case of the step insertion, except the time to ONB takes a little longer.

In Fig. 5.7, the same reactivity insertion transients as in Fig. 5.4 but for a low initial power level of $p_0 = 0.5\%$ (practical "zero" power state) are shown. The fuel reactivity feedback effect can be seen more clearly. As the initial fuel temperature at low power is low, the absolute value of the fuel reactivity coefficient is relatively large, so that its feedback could even make the power drop if fuel temperature increases sufficiently quickly. But, then, when fuel temperature gets higher, its effect is reduced and power begins to rise again. If the DNB does not come, as water temperature increases slowly, the moderator reactivity feedback begins to take effect to slow down the power increase.

It must be noted that, for the DNRR, the reactivity accidents are extremely dangerous as the reactor has very small negative temperature feedback. Any step reactivity insertion must be avoided, otherwise the reactor will not withstand the resultant transient.

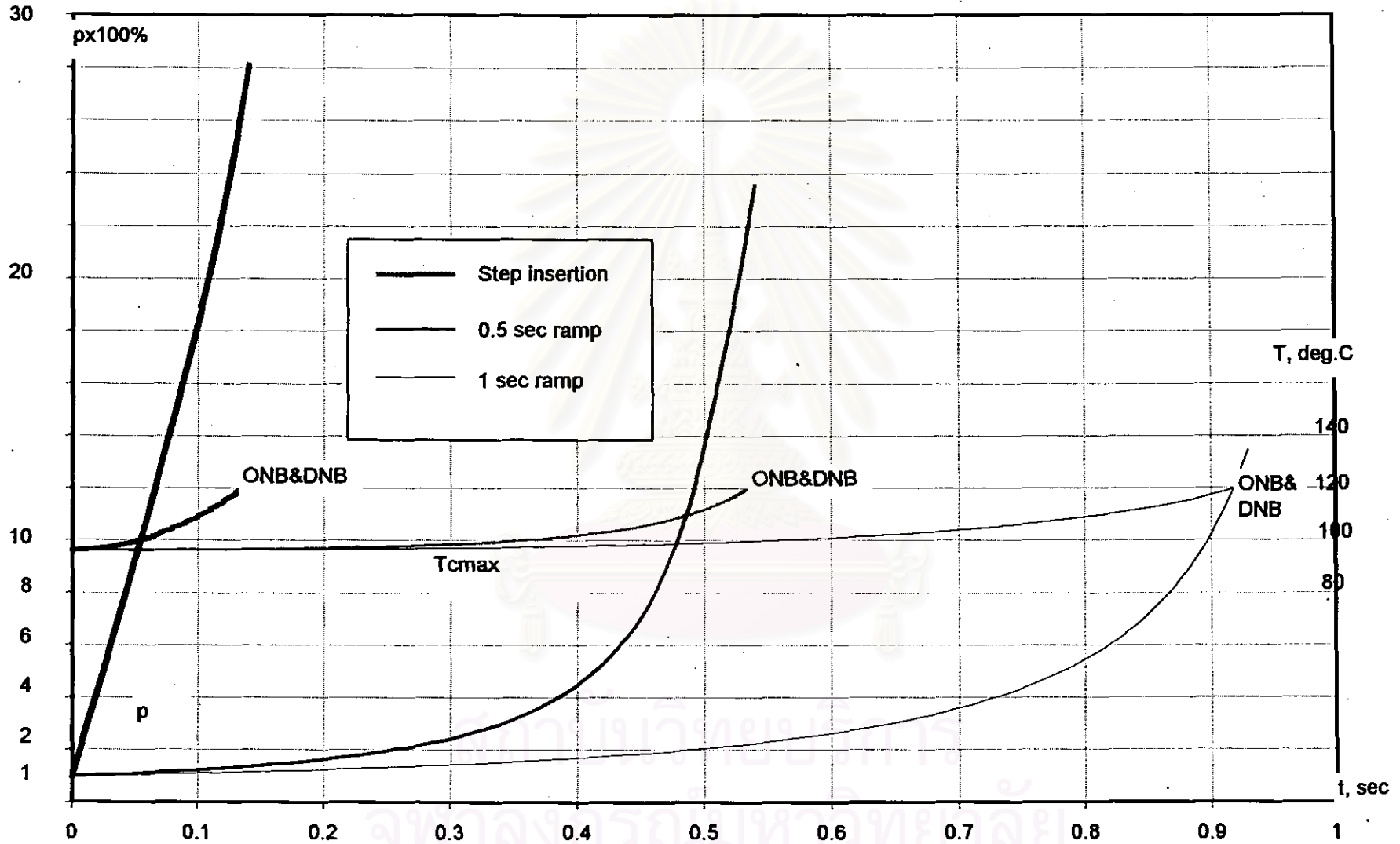


Fig. 5.6. Transients of 1\$ reactivity insertion at initial 100% power level

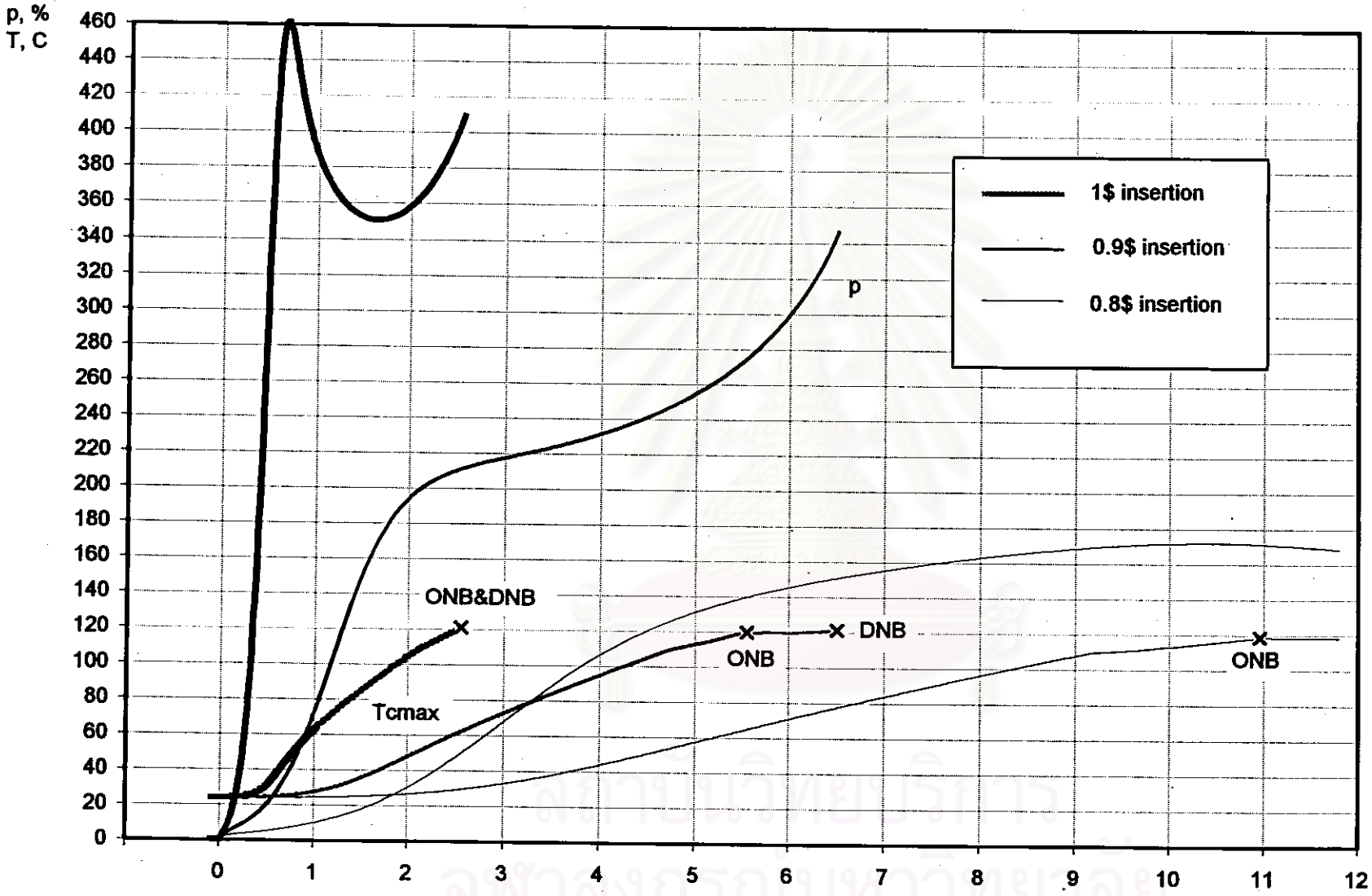


Fig. 5.7. Transients of step reactivity insertion at initial 0.5% power level

5.2.2. Positive Reactivity Insertion into the Subcritical Reactor

Fig. 5.8. shows transients resulting from a step reactivity addition to the initially subcritical reactor. The reactor is in a deeply subcritical ($\rho_0 < -1\%$) shutdown state at a power of $p_0 = 10^{-5}\%$, and is sustained by some neutron source (s_0) present in the core. If the amount of reactivity increase is not enough to make the reactor critical, i.e. $\rho_0 + \rho_{add} < 0$, power will stabilize at some higher level than previously, with both the time and the power level reached being dependent on how close to critical the reactor was at the time of the reactivity insertion. Theoretically, since we can neglect the reactivity feedback due to insignificant changes in fuel and coolant temperatures at so low a power, the stabilizing power is

$$p = \frac{s_0}{-\rho}$$

where $\rho < 0$ is the net reactivity. If ρ is closer to zero, the time for power to stabilize will be longer. If the inserted reactivity is large enough to make the reactor critical, power will increase continuously. However, due to fuel temperature feedback, the power rate of increase will slow down, and the power level will stabilize if the feedback effects can compensate the inserted reactivity.

5.2.3. The Reactor Scram Shutdown

To see the decay heat contribution (p_{fp}) to the reactor thermal power, a scram shutdown of the reactor is shown in Fig. 5.9. The reactor has been at full power for such a long time, that the inventory of fission products is at equilibrium ($p_{fp} = 6\%$). The reactor is shut down by free drop of all control rods into the core within 1 sec., inserting the total reactivity amount of approximately -10% . The neutron power (p) rapidly decreases to about 10% within 1 sec. due to rapid decrease of prompt neutrons. Then it gradually continues to decrease as delayed neutrons decay more slowly. The decay heat at the shutdown moment is $p_{fp} = 6\%$ of neutron power, but its decreasing rate is much lower than that of the neutron power. After 3 sec. from the moment of shutdown, both are comparable ($p \cong p_{fp}$). Further, the fission heat is insignificant compared with the decay heat ($p \ll p_{fp}$). For the DNRR, the residual heat does not cause any threat to its safety, as the natural convection is adequate to cool the reactor.

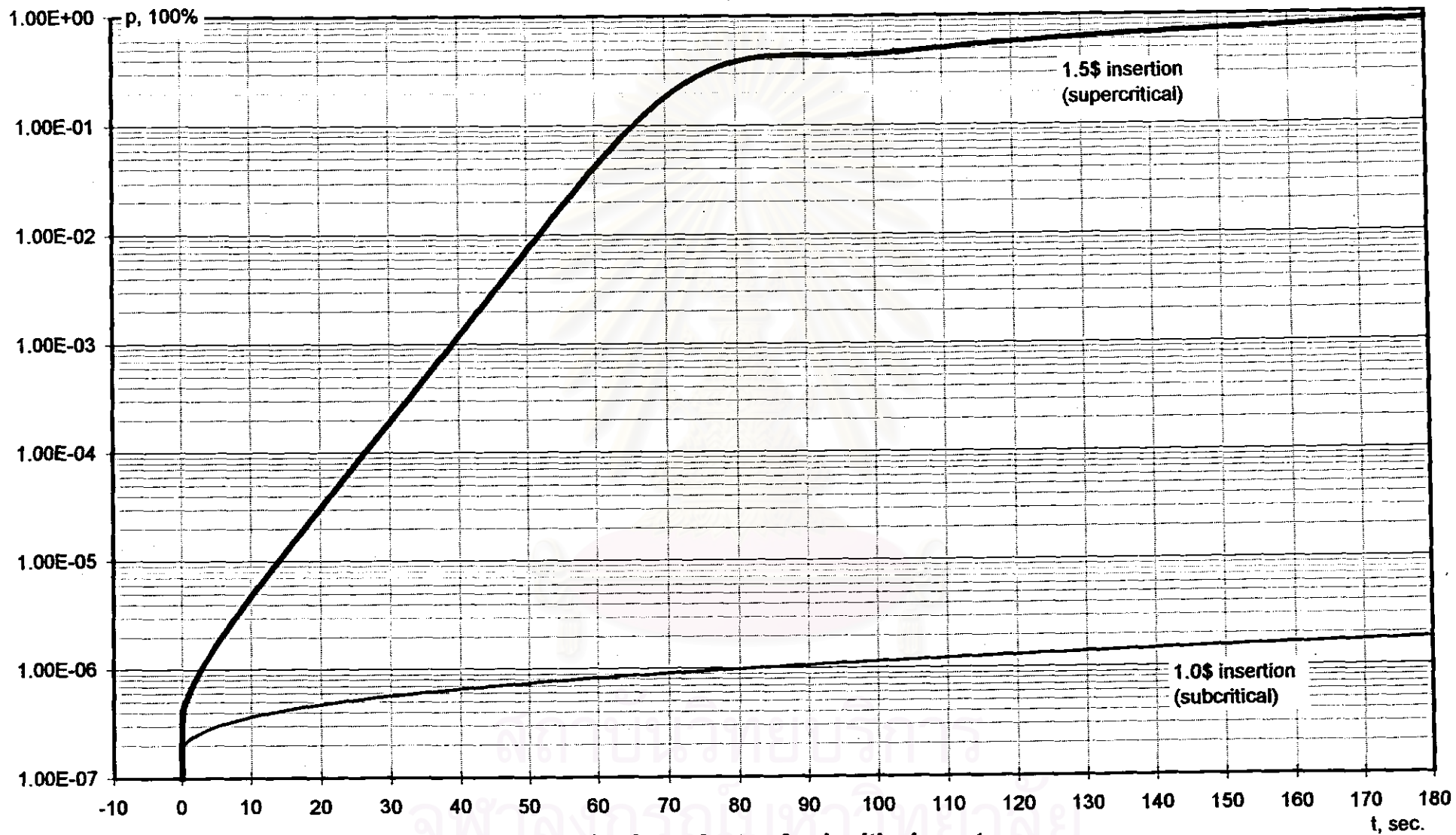


Fig 5.8. Transients of subcritical reactor

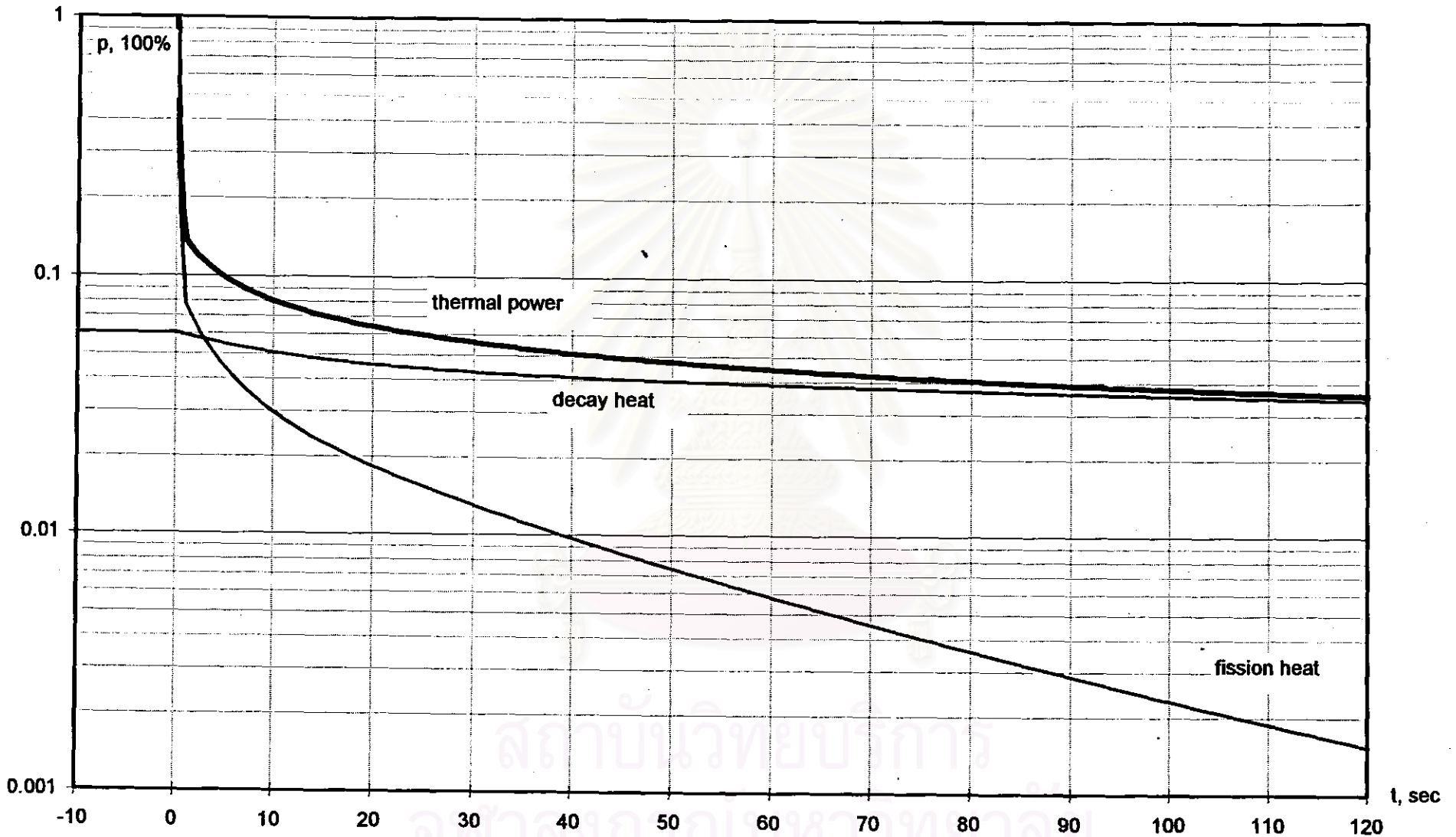


Fig. 5.9. Reactor power after shutdown

Table 5.6. Residual heat after shut-down

t (shut-down) , sec	0	3	60	120
Fission heat p, %	100	5.6	0.6	0.15
Decay heat p_{fd} , %	6	5.7	3.9	3.44
Thermal power, %	106	11.3	4.5	3.6

5.2.4. Reactivity Transients with Power Scram Setting

If the reactor protection system is able to shut the reactor down when power exceeds the scram power setting, the same reactivity insertion as in Fig. 5.5 and Fig. 5.6 will have very different consequences as shown in Fig. 5.10. The scram power is set to 110%. Whenever the reactor power reaches this setting value ($p \geq 110\%$), a scram signal is created. The protection system will take 25 msec. to respond to this signal before all control rods start to drop into the core, and they are fully inserted within 1 sec. In all cases, fuel surface temperature never reaches the ONB. What is most interesting is that the power peak is much higher for the lower initial power level. This is because by the time the scram signal is initiated in the case of low initial power, there has already been a significant amount of positive reactivity inserted that is enough to make the reactor supercritical on prompt neutrons. In the very short time that it takes for the protection system to respond and for the initial insertion of the control rods, power could jump as much as 10 times. Conversely, for high initial power, the scram signal is created just after the insertion of a relatively small amount of positive reactivity, so the power of the reactor supercritical on delay neutrons does not jump but increases slowly, until the positive reactivity is compensated by both the temperature feedback and the subsequent insertion of control rods. Therefore, the colder core is potentially less safe than the hotter core in reactivity accidents when the protection system is required to work to prevent a reactor accident.

Table 5.7. Ramp reactivity insertion with scram power setting to 110%

$\rho = 1.5\$/s$	$p_0 = 0.5\%$	$p_0 = 100\%$
Power peak, %	1662	119
Time to peak. sec	0.5	0.04
Increase in fuel temperature, °C	18	0.1

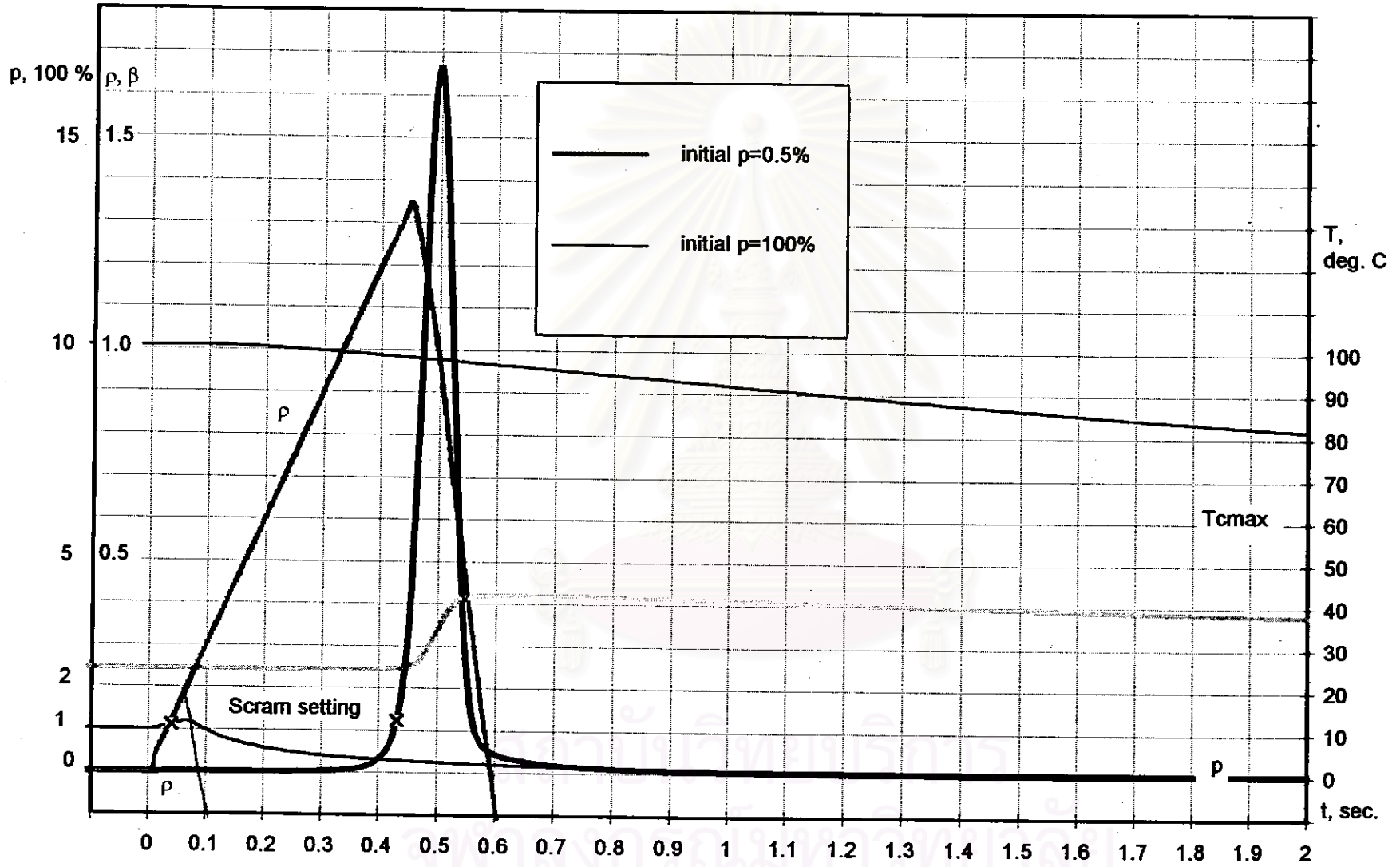


Fig. 5.10. Reactivity transients of 1.5\$/s insertion with scram power setting of 110%