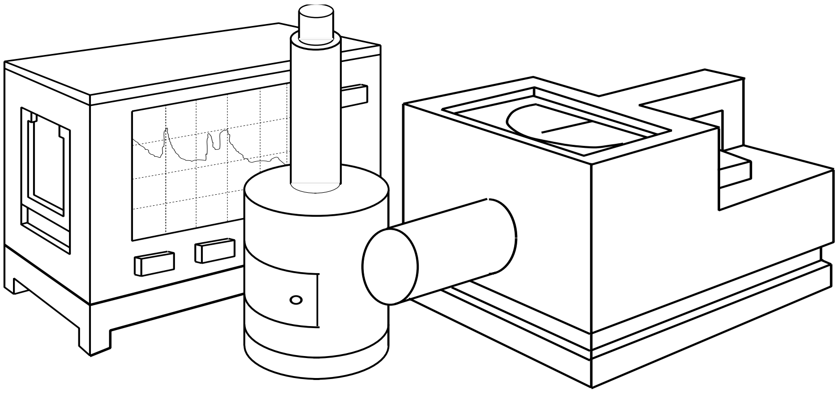
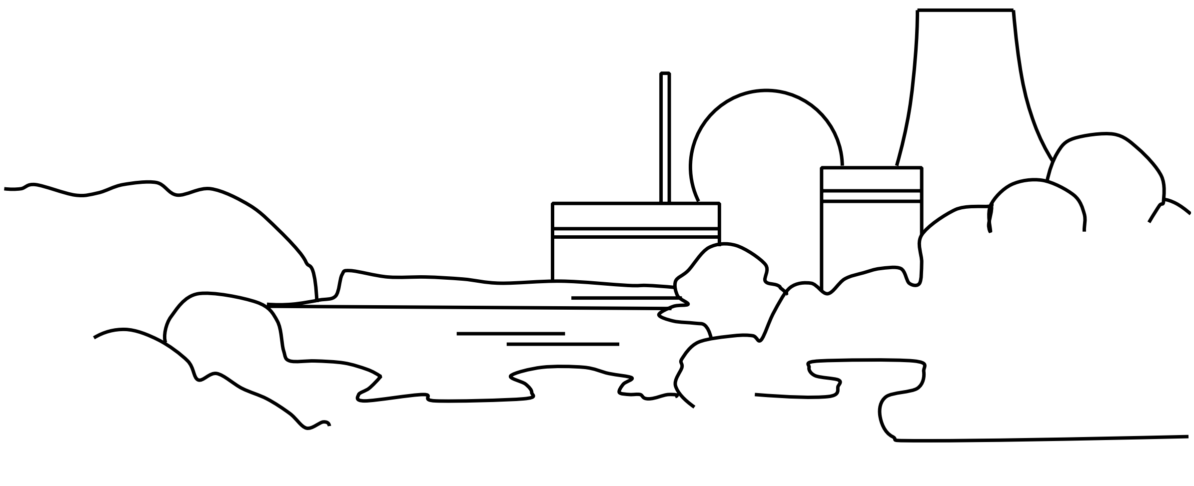
TECHNICAL UNIVERSITY DRESDEN
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Reactor Start-up Procedure"



Instruction for
Experiment “Reactor Start-up”

Content:

Figures:

(issued: March 2015)

1. Motivation

With help of training reactor AKR-2 of the Technical University Dresden it can be studied:
- the principle design of a thermal nuclear reactor,
- the function of main components required for a controlled nuclear fission chain reaction,
- basic design of a digital reactor instrumentation and control (I&C) system.

In order to understand the typical control behaviour, the results of the reactor theory which can be derived from the kinetic reactor equations should be known. By means of reactor power control display and of the reactor period readings the reactor status should be certainly recognised.

2. Tasks

1. The safety check of the reactor has to be carried out and recorded in the operation logbook in accordance with the pre-defined check list.
2. The reactor has to be put into operation by a normal restart and should become critical at that thermal power given in the service instruction (BA) or by the supervisor.
3. For qualitative investigation of the behaviour of a zero-power reactor, power changes have to be carried out according to instructions by the supervisor.
4. The gamma-ray dose rate has to be measured in dependence on thermal reactor power at selected test points in the reactor hall.
5. Critical control rod positions have to be determined and discussed in dependence on the thermal power of the reactor.

3. Types of Reactor Start-up

Starting-up a nuclear reactor means to generate a controlled nuclear fission chain reaction. This process needs to be controlled clearly and handled safely at all times.

The start-up of a nuclear power plant reactor, i.e. the restart from shut-down condition to high thermal power (> 1000 megawatts) is an extraordinary complex process because of the necessary heat removal. The basics of reactor physics can be studied more easily at so-called **Zero-Power Reactors** like the AKR-2.

Zero-power reactors are operated only at such low power (watts to kilowatts) that power effects like
- fuel burn-up,
- poisoning or
- temperature effects

can be virtually neglected during the time of the training. Hence, every start of such a zero-power reactor corresponds, regarding its nuclear part, to a cold start-up of a power reactor.

Any start-up of zero-power reactors can be classified into three categories:
- normal restart,
- comprehensive start-up experiment,
- critical experiment.

The **normal restart** is the start-up in routine reactor operation. Restarts are possible, if nothing has changed since last reactor operation with effect on the reactivity properties of the facility, neither in the reactor itself nor in the internal experimental arrangements.The reactivity characteristics and the control rod positions in the critical condition of the reactor are known.

A **comprehensive start-up** experiment is necessary for a safe reactor start-up after minor material or geometry changes (e.g. after the installation of new internal experimental arrangements) which do not require a critical experiment. The behaviour and the reactivity properties of the reactor are known, i.e. the fuel loading, the reflector properties, and the reactivity characteristics of the control rods.

A **critical experiment** has to be carried out, if the physical parameters (e.g. critical mass, reactivity characteristics of the control rods, excess reactivity etc.) are known from calculations only. This applies, in any case, to the first start-up of a newly build reactor and to any further start-up after variations of the assembly which let expect a considerable change in the reactivity behaviour.

4. Theoretical Background

At zero-power reactors, the neutron flux density Φ (unit: neutrons/(cm² · s)) is the quantity to be controlled. It is proportional to the neutron density (unit: neutrons/cm³) as well as to the reactor power (unit: watt) and to the total number of the neutrons N in the reactor.

Normally, the shut-down reactor contains almost no free neutrons. Its state regarding neutron reproduction is characterised by the multiplication factor **k < 1**. This **subcritical condition** is maintained by material and/or geometric conditions in the core, e.g. due to addition of neutron absorbers, removal of moderator or reflector material, separation of the core into subcritical masses, etc. By gradually eliminating these conditions, the reactor state can be converted to the **critical condition (k = 1)** or to a **supercritical condition (k > 1)**.

In this process it is remarkable, that the neutron flux density rises by several orders of magnitude, i.e. from a few neutrons up to many millions of neutrons per second and cm² depending on the eventual power. Therefore, certain start-up rules have to be kept to avoid start-up incidents.

These rules avoid
- mistakes of the reactor operators resulting from misinterpretation of the reactor state,
- too high reactivity change rates,
- a too high excess reactivity ($\rho_{excess} > \beta$).

Since criticality can be achieved also at very low power levels, there is the danger of exceeding the critical point already at very low neutron flux densities and without the reactor operator being aware of it. Because of the statistical fluctuations at low neutron flux densities, measurements require detectors with a sluggish time response. The time delay between the occurrence and the display of flux density changes may cause the operator to assess the reactor state incorrectly and possibly to increase the multiplication factor to inadmissible supercritical ranges. The resulting very fast increase of the neutron flux density might be recognised only when intervention is already impossible. Then, harm to the operational staff and damage to the facility could be a consequence.

Thus, following prerequisites ensure a safe reactor start-up:
- The reactor must be started from an adequately high value of the neutron flux density. The initial neutron flux density has to be raised such that the neutron detectors deliver proper values. Usually, at zero-power reactors, an artificial (external) neutron source is used for controlling the subcritical condition. The subcritical multiplication of the source neutrons provide the reqiured values of the flux density for the neutron detectors of the reactor instrumentation. At high power reactors, also neutrons from spontaneous fission and (γ,n)-reactions may be sufficient for this purpose.
- Limitation of the positive reactivity change rate:
  Only those positive values of the reactivity change rate are allowed that
  - the reactor can be controlled safely manually during normal operation and
  - the I&C-system of the reactor is able to shut-down the reactor reliably if required (international recommended limit: $(d\rho/dt)_{max} = 10^{-4}\ s^{-1}$).
- Limitation of the excess reactivity:
  The excess reactivity is limited to values of $\rho_{excess} < ß$ (at AKR-2: $\rho_{excess} \approx 0.3\ \%$).

4.1. Prompt and Delayed Neutrons

Fission of nuclei of the reactor fuel (e.g. U-235) releases fast neutrons. These neutrons are released either immediately after the nuclear fission as the so-called **prompt neutrons** or originate from a special case of radioactive decay as the so-called **delayed neutrons**.

The average time between the birth of a fast neutron and the next nuclear fission after slowing down and diffusion with the formation of a new generation of fission neutrons is called **neutron life time l**.

The neutron life time consists of the moderation time (from fast to thermal energy, about 10^{-5} s), the diffusion time (entering of the neutron into the fissionable nucleus, about 10^{-4} s) and the reaction time (nuclear fission, about 10^{-15} s).

Consequently, the total time is determined by the longest part, which is the diffusion time.

All neutrons that arise within one neutron life time are considered as one neutron generation. These neutrons vanish by leakage out of the reactor ($N_{leakage}$) and by absorption (N_{abs}) within the reactor core. Partly, absorption causes new fissions (provided that the absorption is in the fuel).

At the time t + l, a new neutron generation has been produced (N_{gen}). Therefore, the multiplication factor k can be written in the following form:

$$k = \frac{N_{gen}}{N_{abs} + N_{leakage}} = \frac{number\ of\ neutrons\ at\ time\ t + l}{number\ of\ neutrons\ at\ time\ t} \qquad \textbf{(1)}$$

Consequently, the number of neutrons per time unit being in the reactor is the balance of those neutrons being produced ($k \cdot N / l$) and those disappearing ($- N / l$). Hence, the change of the number of neutrons is given by:

$$\frac{dN}{dt} = k \frac{N}{l} - \frac{N}{l} = N \frac{k - 1}{l} \tag{2}$$

or

$$\frac{dN}{N} = \frac{k - 1}{l} dt \tag{3}$$

which is solved by

$$N(t) = N_0 \cdot e^{\frac{k - 1}{l} \cdot t} = N_0 \cdot e^{\frac{t}{T}} \qquad with\ T = \frac{l}{k - 1} \tag{4}$$

where T is called **reactor period**. The reactor period is that time interval during which the neutron number N (or in same way Φ, n or P) increases by a factor of e (≈ 2.71). Because it is more convenient to determine a change for a factor of 2 (instead of e), it is often common practice to use the **doubling time** T_2 instead of reactor period. Reactor period and doubling time are connected with each other by the simple relation:

$$T_2 = ln2 \cdot T \tag{5}$$

Thus, according to equation (4), the time behaviour of a reactor follows always an exponential function.

For the assumption that only prompt neutrons would exist in a nuclear reactor (with typical life times $l \approx 10^{-4}$ s), a change of the multiplication factor from 1 to 1.001 would cause an enormous increase of the neutron number and consequently of the reactor power within only one second by a factor of

$$\frac{N(t + 1s)}{N(t)} = e^{\frac{1.001 - 1}{0.0001\ s} \cdot 1s} \approx 22000 \tag{6}$$

Such dynamics would be virtually uncontrollable. A nuclear reactor with a <u>controlled</u> chain reaction would not be feasible.

The control of the chain reaction is possible only due to the existence of the delayed neutrons, whose properties are summarised in Tab. 1. Delayed neutrons arise in 6 groups (from 6 groups of radioactive precursor nuclei as a very special case of nuclear decay) with half-lives $T_{1/2}$ between 0.23 s and 56 s. From fission of U-235 only a fraction of ß = 0.641 % is released as delayed neutrons. The remaining 99.359 % are prompt neutrons.

Group i	pre-cursor	average kinetic energy / keV	$T_{1/2}$ / s	$\lambda_i = \ln 2 / T_{1/2}$ / s^{-1}	fraction $ß_i$ compared to all fission neutrons / %	relative fraction $a_i = ß_i / ß$	neutrons per 10^3 fissions (absolute value)
1	Br-87	250	55.72	0.0124	0.021	0.033	0.52
2	J-137	560	22.72	0.0305	0.140	0.219	3.46
3	Br-89	430	6.22	0.111	0.126	0.196	3.10
4	?	620	2.30	0.301	0.253	0.395	6.24
5	?	420	0.61	1.14	0.074	0.115	1.82
6	?	-	0.23	3.01	0.027	0.042	0.66
total					0.641	1.000	15.80

Tab. 1, Properties of delayed neutrons caused by fission of U-235
/Reactor Physics Constants, ANL-5800/

The fraction of delayed neutrons compared to all fission neutrons can also be derived using the last column of Tab. 1. Considering an average of 2.47 released neutrons per fission of a U-235 nucleus, it results ß = 15.80/1000/2.47 = 0.0064 = 0.64 %.

The value ß can vary between 0.5 % and 0.7 % depending on the fuel enrichment, on the moderator temperature, and in particular on fuel burn-up. At a given nuclear reactor, the fraction of delayed neutrons is called $ß_{eff}$.

For reasons of simplicity, the 6 groups of delayed neutrons can be approximately condensed to one single group with averaged parameters. Thus, using the values of Tab. 1, the average life time $l_{delayed}$ of the delayed neutrons, can be written as

$$l_{delayed} = \frac{1}{\lambda} = \frac{1}{\beta} \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} = 13.00 \ s \qquad \textbf{(7)}$$

This corresponds to an average decay constant $\lambda = 0.0769 \ s^{-1}$.

For prompt and delayed neutrons together, an effective life time can be calculated by

$$\bar{l} = l_{delayed} \cdot 0.00641 + l_{prompt} \cdot 0.99359 = 0.083 s \qquad \textbf{(8)}$$

If the power increase within one second is now calculated again according to (6), taking into account a change of the multiplication factor from 1 to 1.001, it results

$$\frac{N(t + 1s)}{N(t)} = e^{\frac{1.001 - 1}{0.083\ s} \cdot 1s} \approx 1.01 \qquad \textbf{(9)}$$

i.e. an increase of reactor power by only 1 %, which can be controlled without any difficulties.

4.2. Subcritical reactor

It can be assumed that the reactor does not contain any free neutrons at the beginning of start-up ($n_{(t=0)} = 0$). At the time t = 0, a neutron source shall be inserted into the reactor. After t = l (l = life time of a neutron generation), the source has produced an average neutron source density $n_{Source} = S \cdot l$. After another neutron generation life time l, this value has increased by the factor k according to equation (1) while, at same time, the source has released additional $n_{Source} = S \cdot l$ neutrons. Hence, one has

$n(t=l) \quad = n_1 = S \cdot l$

$n(t=2 \cdot l) = n_2 = S \cdot l + S \cdot l \cdot k$

After another life time, n_2 has got multiplied by k again while at same time, the source has released additionally $n_{Source} = S \cdot l$ neutrons, resulting in

$n(t=3 \cdot l) = n_3 = S \cdot l + (S \cdot l + S \cdot l \cdot k) \cdot k$

or in general

$n(t) = S \cdot l \cdot (1 + k + k^2 + k^3 + ...)$

This is a geometric series that has the limit (in given case of a subcritical reactor with k < 1)

$$n(t = \infty) = S \cdot l \ \frac{1}{1 - k} \qquad \textbf{(10)}$$

The factor

$$M = \frac{1}{1 - k} \qquad \textbf{(11)}$$

is called the **subcritical multiplication factor**.

Equation (10) has the following consequences:
- The subcritical reactor acts as a neutron amplifier (for the neutrons released by the source).
- For the critical reactor (k = 1), after inserting a neutron source, the neutron density would increase linearly up to infinity. That is because after every single fission, exactly one neutron

is left for a new fission, while the neutron source constantly adds more neutrons. Therefore, all neutrons from the source sum-up in linear dependence.
However, this source effect can be noticed only at very low reactor powers since already at a power level of 1 W the neutrons being added by the source contribute only a small fraction of the total neutron contents in the facility.

For a subcritical reactor with a neutron source inserted, the **time dependence of the neutron density** is described by:

$$n(t) = \frac{S \cdot l}{1 - k} \cdot \left(1 - e^{-\frac{(1-k)}{l} t} \right) \tag{12}$$

Because of the negative exponent, the neutron density in the reactor asymptotically approaches the value:

$$n_{\infty} = \frac{S \cdot l}{1 - k} = \frac{S \cdot l^*}{-\rho} \tag{13}$$

with $l^* = l / k$ und $\rho = (k - 1) / k$. A comparison with equation (11) shows that the neutrons of the source have been amplified by the subcritical multiplication factor. For values of k approaching 1, the subcritical amplification gets larger and larger. On the other hand, according to equation (12), the time increases more and more until the approximation to the asymptotic limit value of the neutron density (see also Fig. 1)

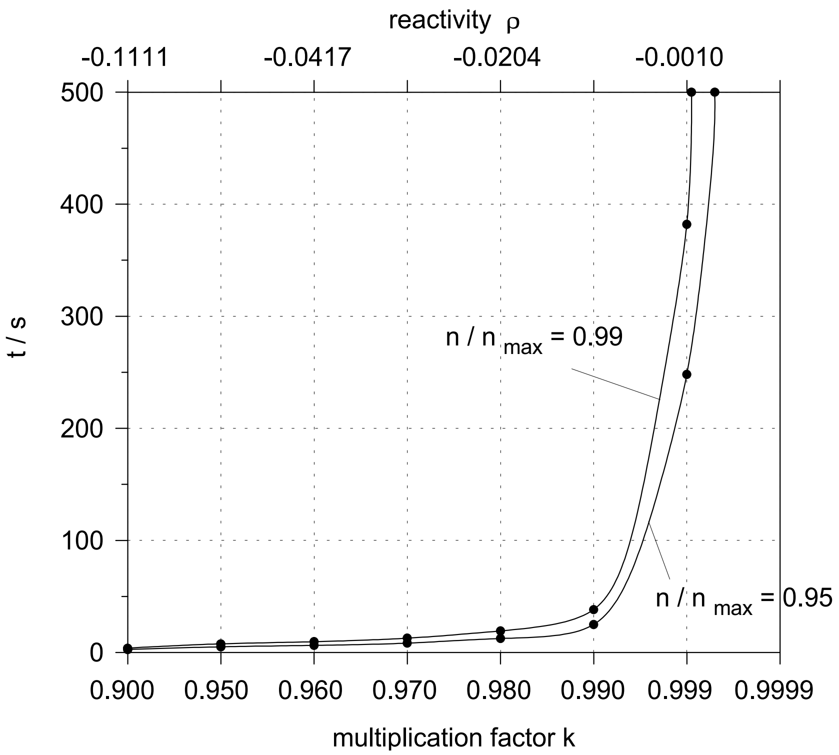


Fig. 1, Times required for 95 % and 99 % approximation to the maximum asymptotic neutron density n_{max} in dependence on the multiplication factor k and the reactivity ρ

4.3. Supercritical reactor

With ß being the fraction of delayed neutrons arising from nuclear fission, the number of prompt neutrons being produced in the reactor within one neutron life time is

$$n_{prompt} = k \cdot n - \beta \cdot k \cdot n = n \cdot k\ (1 - \beta) \qquad \textbf{(14)}$$

For the change of the prompt neutron density within one neutron life time one has:

$$\frac{dn_{prompt}}{dt} = \frac{n \cdot k\ (1 - \beta) - n}{l} = \frac{n}{l}\ [k \cdot (1 - \beta) - 1] \qquad \textbf{(15)}$$

At the same time, delayed neutrons $n_{delayed}$ result from radioactive decay of certain radionuclides (delayed neutron precursors) which have been produced in previous fissions. The number of delayed neutrons can be calculated by means of the law of radioactive decay (considering all 6 groups of delayed neutrons):

$$\frac{dn_{delayed}}{dt} = \sum_{i=1}^{6} \lambda_i \cdot C_i \qquad \textbf{(16)}$$

Consequently, the total change of the neutron density within one neutron life time is:

$$\frac{dn}{dt} = \frac{dn_{prompt}}{dt} + \frac{dn_{delayed}}{dt} + S = \frac{n}{l}\ [k \cdot (1 - \beta) - 1] + \sum_{i=1}^{6} \lambda_i \cdot C_i + S \qquad \textbf{(17)}$$

Additional application of $l^* = l / k$ und $\varrho = (k - 1)/k$ to equation (17) gives the so-called **reactor kinetic equations**:

$$\begin{aligned} \frac{dn}{dt} &= \frac{\rho - \beta}{l^*} \cdot n + \sum_{i=1}^{6} \lambda_i \cdot C_i + S \\ \frac{dC_i}{dt} &= \frac{\beta_i}{l^*} \cdot n - \lambda_i \cdot C_i \qquad (i = 1, \ldots, 6) \end{aligned} \qquad \textbf{(18)}$$

with $n \cdot \beta_i / l^*$ being the number of generated delayed neutron precursors and $\lambda_i \cdot C_i$ being the number of precursors decaying under emission of one neutron. Equations (18) are a system of 7 coupled differential equations. A quite simple approximated solution is achievable by summarizing the 6 groups of delayed neutrons in only one single group with the following averaged values (see also chapter 4.1.):

$$\beta = \sum_{i=1}^{6} \beta_i \qquad\qquad \frac{1}{\lambda} = \frac{1}{\beta} \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} \tag{19}$$

Thus, the system of differential equations will reduce to only 2 coupled differential equations:

$$\begin{aligned} \frac{dn}{dt} &= \frac{\rho - \beta}{l^*} \cdot n + \lambda \cdot C + S \\ \frac{dC}{dt} &= \frac{\beta}{l^*} \cdot n - \lambda \cdot C \end{aligned} \tag{20}$$

For a reactivity jump ($\rho = 0$ für $t < 0$ und $\rho = const$ für $t \geq 0$) and if additionally the source neutrons S are neglected, equation (20) results in

$$n(t) = n_0 \left[\frac{\beta}{\beta - \rho} \, e^{\frac{\lambda \cdot \rho}{\beta - \rho} \cdot t} - \frac{\rho}{\beta - \rho} \, e^{-\frac{\beta - \rho}{l^*} \cdot t} \right] \tag{21}$$

By means of equation (21) two important reactor conditions can be discussed:

4.3.1. Delayed supercritical reactor ($\rho < \beta$)

In this case, the exponent of the second term in equation (21) is negative and has, because of $l^* \approx 10^{-4}$ s , a large absolute value. Consequently, this term vanishes within a few seconds and only the first term in equation (21) remains after short time.

$$n(t) = n_0 \frac{\beta}{\beta - \rho} \, e^{\frac{\lambda \cdot \rho}{\beta - \rho} \cdot t} \tag{22}$$

Therefore, it can be seen that a sudden increase of the reactivity results in a sharp rise of the neutron density, which is called **prompt jump** as it is caused by prompt neutrons. The height of this jump is

$$\frac{\Delta n_{prompt}}{n_0} = \frac{\beta}{\beta - \rho} - 1 = \frac{\rho}{\beta - \rho} \tag{23}$$

Considering prompt neutrons only the reactor is subcritical (since $\rho < ß$). The neutron density increases only due to subsequent addition of delayed neutrons, and that according to the stable reactor period T_s (see Fig. 2). The value of the **stable reactor period** can be derived from equation (22) to

$$T_s = \frac{ß - \rho}{\lambda \cdot \rho} \qquad \textbf{(24)}$$

This description of the reactor period, which was derived for the approximation of only one averaged group of delayed neutrons, is sufficiently precise for small values of the reactivity ($\rho < 0.001$).

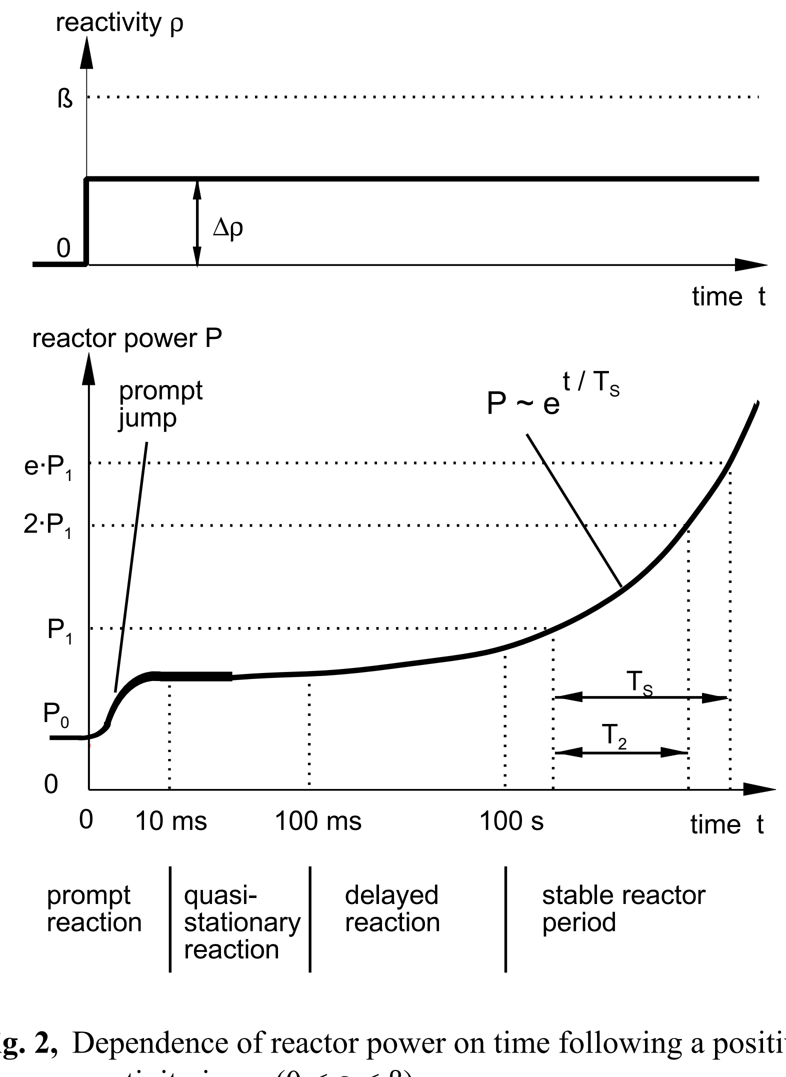


Fig. 2, Dependence of reactor power on time following a positive reactivity jump ($0 < \rho < ß$)

4.3.2. Prompt Supercritical Reactor ($\rho > ß$)

In this case, both, the second term in equation (21) becomes positive as well as its exponent. Consequently, the neutron density rises very fast (because of $l^* \approx 10^{-4}$ s). The reactor is super-critical only by the prompt neutrons. The reactor period would be in the order of milliseconds. Thus, the reactor power would increase so fast that the control rods could not be used reasonably for reactor control. This case is the accident of an uncontrollable power excursion and must never occur. Therefore, it is of high importance to keep $\rho < ß$ at any time and under any circumstances.

Table 2 summarises possible reactor conditions, resulting reactor power behaviour and the corresponding multiplication factors k and reactivity values ρ.

Because of the exceptional safety relevant importance of the transition from the controlled reactor to the uncontrollable reactor (i.e. from the delayed supercritical condition to the prompt super-critical condition), a particular artificial unit ρ' has been introduced for describing the reactivity. By relating the reactivity ρ to the fraction of delayed neutrons ß the reactivity $\rho' = \rho / ß$ is defined with the advantage that in case of safe delayed supercritical reactor ρ' remains smaller than 1. For distinguishing the two definitions of reactivity, the quantity ρ' has been added by the arbitrary unit $ (Dollar, 1 $ = 100 Cents). Thus, the transition from the delayed supercritical condition to the prompt supercritical condition occurs at the impressive value of $\rho' = 1$ $.

Reactor Condition	**Reactor Power Behaviour**	**Multiplication Factor k**	**Reactivity**	
			$\rho = (k - 1) / k$	$\rho' = \rho / ß$ [$]
subcritical	$\searrow$	< 1	< 0	< 0
critical	$\rightarrow$	= 1	= 0	= 0
(delayed) supercritical	$\nearrow$	$1 < k < 1+ß$	$0 < \rho < ß$	$0 < \rho' < 1$ $
prompt supercritical	$\uparrow\uparrow$	$k \geq 1+ß$	$\rho \geq ß$	$\rho \geq 1$ $

Tab. 2, Summary of possible reactor conditions with corresponding reactor parameters

4.4. Influence of Reactivity Change Rate

For the subcritical reactor, differentiation of equation (13) gives the neutron density change rate as a function of the reactivity change rate:

$$\frac{dn}{dt} = \frac{S \cdot l^*}{\rho^2} \cdot \frac{d\rho}{dt} \tag{25}$$

With the general definition of the reactor period

$$T = \frac{n}{(dn/dt)} = \frac{P}{(dP/dt)} \tag{26}$$

and using equation (25), the reactor period for the subcritical reactor results in:

$$T = \frac{-\rho}{(d\rho/dt)} \qquad (\rho < 0) \tag{27}$$

Considering an infinitely slow approach to criticality, i.e. $\lim(d\rho/dt) \rightarrow 0$, the power of the subcritical reactor is given by the subcritical amplification value according to equation (13), at all times. Criticality would be achieved only after infinitely long time (Fig. 3). However, in the meantime, the reactor power had been grown to infinity.

On the other hand, for limited positive values of $d\rho/dt$, the reactor becomes critical at limited values of the reactor power whereby a higher reactivity change rate corresponds to a lower reactor power when achieving criticality. In the deep subcritical reactor condition, the reactivity change rate has virtually no effect on the change rate of the neutron flux density.

For adding positive reactivity with a constant rate, i.e. $d\rho/dt = \text{const}$, the reactor period decreases with linear correspondence to the remaining amount of negative reactivity in the reactor.

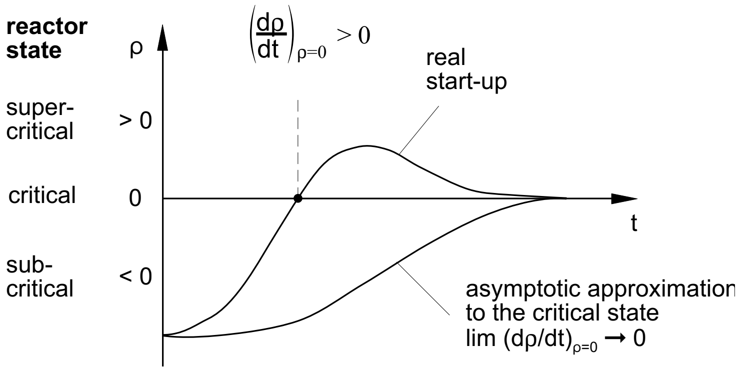


Fig. 3, Behaviour of the reactivity for an infinitely slow reactor start-up and for a real reactor start-up

The more the reactor approaches to its critical condition, the more the neutron density (and reactor power and reactor period as well) depend on the reactivity change rate.

The reactor start-up would be very time consuming, if the operator would wait for the approximation to the (subcritical) asymptotic limit of the reactor power n_∞ after each increase of reactivity. Instead of this, it is the common procedure of a normal reactor restart to rise the criticality first onto the supercritical reactor condition and then adjust criticality afterwards at the desired power level (compare to Fig. 3).

When approaching the desired reactor power, the (negative) reactivity change rate is continuously lowered to values that are appropriate for the respective differences of the actual reactor power and the desired reactor power. The reactor is critical (k = 1), if the stable reactor period is infinite ($T_S = \infty$).

5. Procedure of the Experiment

5.1. Design and Operation of the AKR-2

The AKR-2 is a homogeneous thermal zero-power reactor. A detailed description of its design and operation is given in Ref. /2/. A basic functional layout of the AKR-2 facility is shown in Fig. 4. For safety reasons, the cylindrical core (diameter 250 mm, height 275 mm) consists of two separate sections. Each section contains still the initial fuel loading as it was assembled stepwise from the disk-shaped fuel elements having various thicknesses in the critical experiment during commissioning of the reactor in 1978.

The fuel elements consist of a homogeneous mixture of nuclear fuel (uraniumoxide, enriched to 19.8 % U-235) and the moderator (polyethylene). The critical mass of the core is about 790 g of U-235. The core is surrounded on all sides by a graphite reflector of about 30 cm thickness.

Both, the upper and the lower core sections are each hermetically enclosed in an aluminium container. A second, larger gas-tight reactor tank encloses the two core sections and parts of the reflector. The pressure inside the reactor tank is lowered by (8…18) kPa compared to the ambient atmospheric pressure. This subatmospheric pressure barrier prevents an uncontrolled leakage of radioactive fission products even in the unlikely case that all the other internal retention barriers would fail.

In the shut-down condition of the reactor, the lower core section is lowered by about 50 mm. When starting the reactor, it is moved upwards by a drive mechanism and a threaded spindle including an electromagnetic holder of the core section until close contact with the upper core section is given. Prior to this, inside the spindle, the start-up neutron source (^{241}Am-Be, source strength $2.2 \cdot 10^6$ s^{-1}) is driven up to the lower side of the core.

Three control rods made from cadmium are available for both, the control of the reactor and for safety shut-down. The positions of the neutron source, of the lower core section and of the three control rods are displayed digitally and analogously on the monitor screens at the control desk.

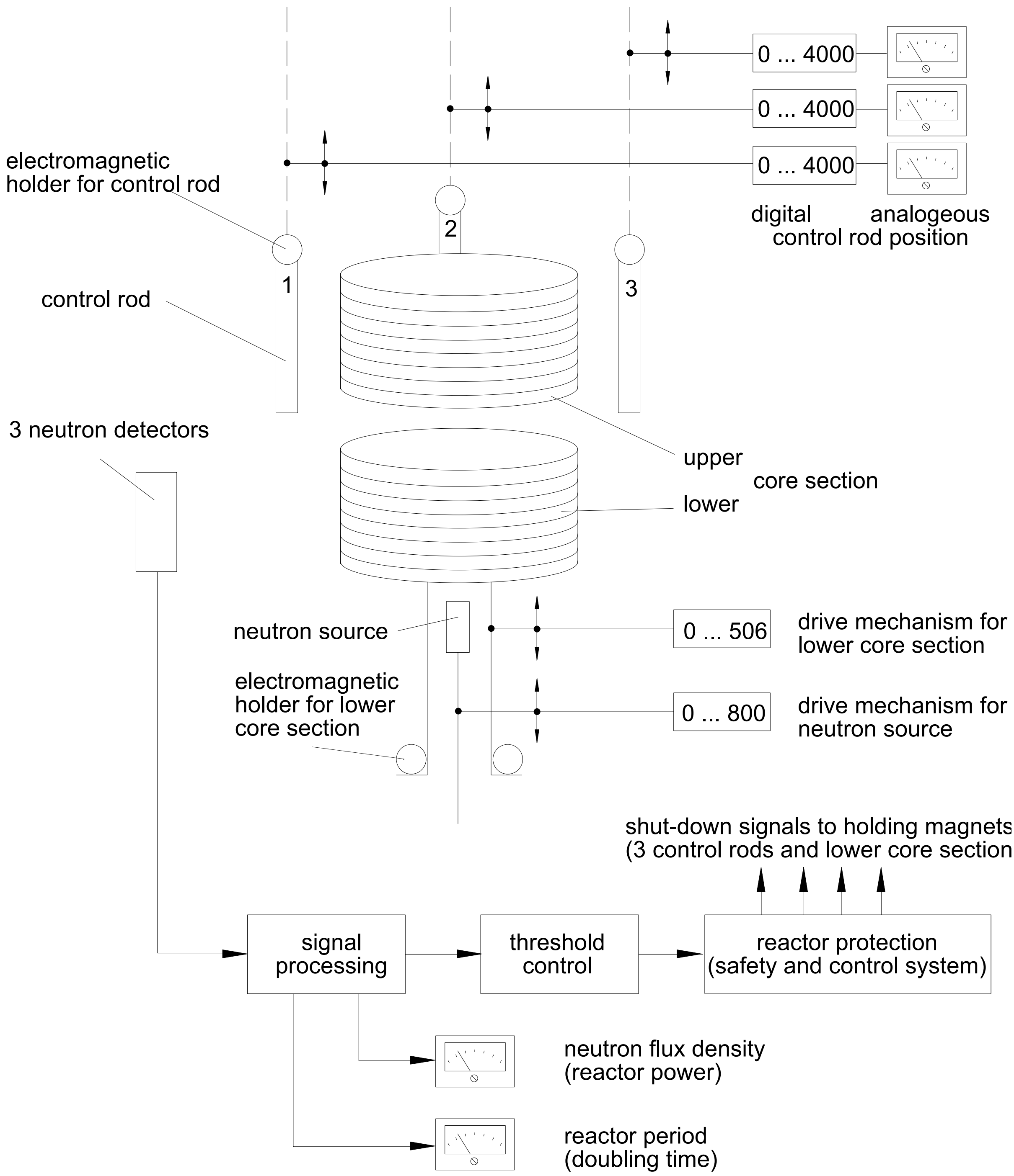


Fig. 4, Functional layout of the AKR

Three neutron detectors of different types and sensitiveness are used for measuring the neutron flux density in the reactor and in this way for supervising the operating condition of the reactor. They provide electrical signals (pulse densities, currents) that are proportional to the neutron flux density and are used, in consequence, for calculation of reactor power and reactor period.

These signals are supervised with safety threshold monitors. When exceeding the pre-defined thresholds, the safety and control system (SUS-System) gives shut-down (SCRAM) signals that release the holding magnets of the control rods and of the lower core section. These magnets lose their electricity supply and consequently the control and safety units (i.e. lower core section and all three control rods) drop down and interrupt the nuclear chain reaction, i.e. the reactor is shut-down.

5.2. Normal Restart

It must be definitely excluded that inadmissible operating conditions and disturbances during the start-up of a reactor due to improper operation can occur. Therefore, the instrumentation is designed in such a way that necessary safety requirements are kept automatically and undue actions of the operator have no effect or lead to automatic reactor shut-down.

During the reactor start-up, the protective logic ensures the correct sequence of actions, i.e. the start-up procedure can only succeed if a certain pre-defined sequence of starting conditions is maintained and by strictly following the given necessary actions.

A flow chart of the start-up procedure is given in Fig. 5.

The processing and completion of the particular steps is indicated on the monitor screens of the control desk. The operating condition of the reactor can be concluded from the time behaviour of the reactor power and the reactor period (or doubling time). Warning and alarm signals inform the operator about inadmissible operational parameters both, visually and acoustically.

The safety and control system (SUS) supervises compliance with the given threshold conditions at any time, already during the reactor start-up. It causes automatic reactor shut-down (SCRAM) under the following conditions:
- any fault in the protective logic
- neutron flux density > 120 % in 1 of the 3 neutron measuring channels
- doubling time of the reactor power < 10 s in 1 of 2 neutron measuring wide-range channels.

Before every start-up of the reactor, all electronic and mechanical parts that are essential for a safe operation have to be checked.

The procedures of both, the functional check and the start-up procedure have to follow fixed sequences (see Ref. /2/). The results have to be documented in the operation logbook.

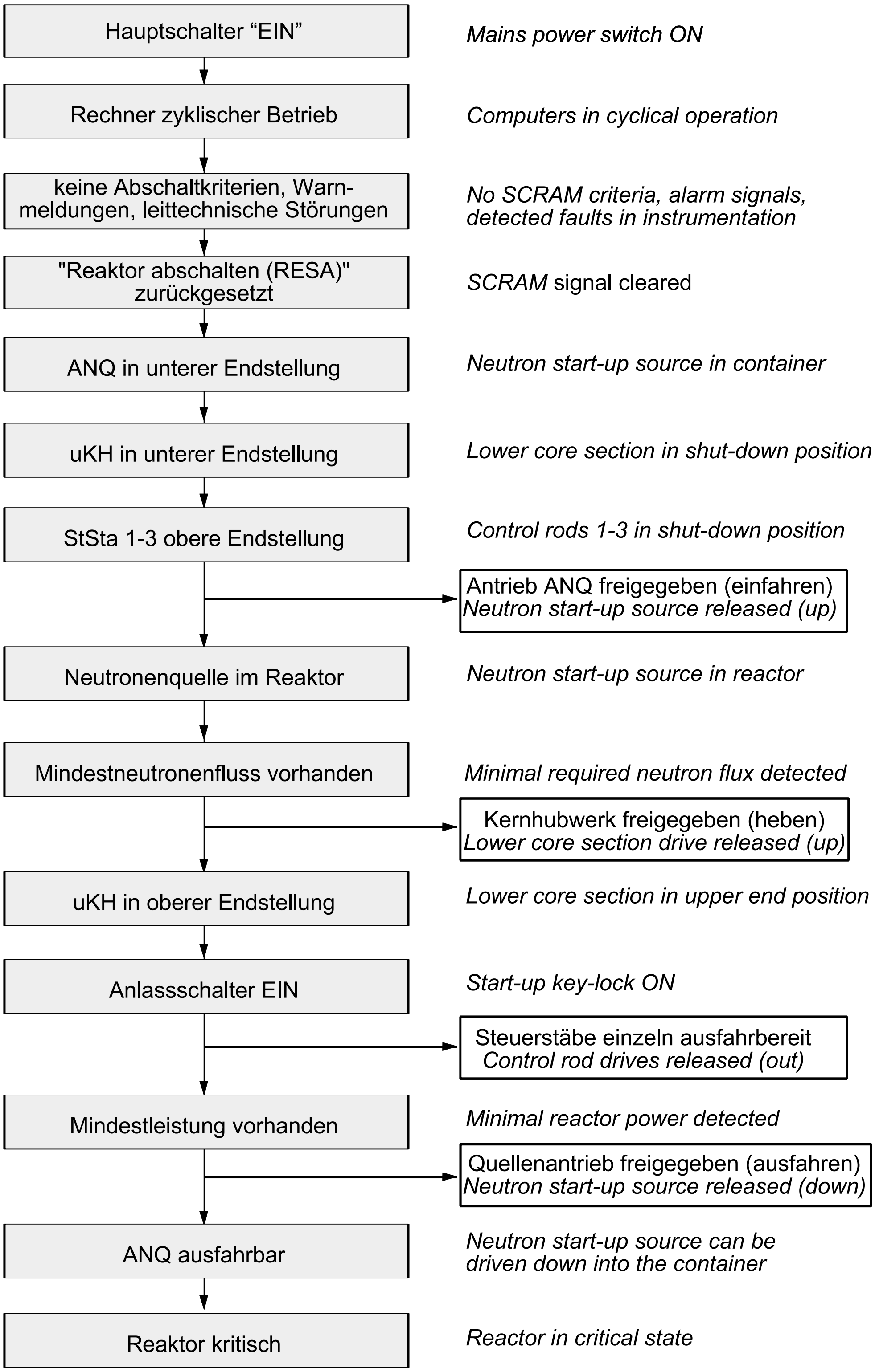


Fig. 5, Flow chart of the start-up procedure (protective logic)

Consequently, the reactor start-up has to be carried out in the following order:

1. Information about pre-defined conditions of the subsequent reactor start-up by reading the adequate service instruction (BA) and about previous conditions of reactor operation, especially about corresponding critical control rod positions, by reading the entries in the operation logbook.

2. Functional check in accordance with test instruction. The reactor is free to be started-up only at full availability of all components of the system!

3. Reactor start-up in accordance with the check list for reactor start-up and operation.

Remarks:
- The neutron source can be withdrawn at reactor power values > 0.25 W.
- If the reactor power has about 80 % of the desired power level, the (positive) reactivity has to be reduced by moving cautiously the control rod(s) in direction to the core until the desired reactor power is obtained with a reactor period of $T_S \approx \infty$.

5.3. Power Change

Power change means any intended rise or reduction in the reactor power.

Power rise:

A rise in the power is achieved by drawing out one or several of the control rods with corresponding increase of reactivity. The rods can be pulled out only stepwise and one after another.
The required change in the rod position can be pre-determined with the help of the rod reactivity characteristics. The reactor doubling time should not fall below 30 s.
If the rod reactivity characteristics are not available, the control rods have to be moved in accordance with the display for the doubling time, i.e. before obtaining the value of 30 s, the movement of the control rods has to be stopped.

Power reduction:

A reduction in reactor power is obtained by moving the control rods into the core with corresponding decrease of reactivity to negative values. The value of the resulting negative reactor period is not safety-relevant. Therefore, all three control rods can be moved simultaneously.

5.4. Determination of the Operation Condition

The effects of changes in the positions of the neutron source, of the lower core section and of the control rods on the reactor can be expressed by means of one common physical quantity, i.e. the **reactivity** ρ (t). The reactivity is influenced by these changes in a defined way and can be seen as a global time-dependent reactor control parameter $\rho = \rho(t)$.

The AKR-2 (as most of other reactors, too) has no instruments for direct measurement of the reactivity. Therefore, the reactor operator has to determine the operating condition of the reactor from the behaviour of the reactor power and the reactor period (or doubling time).

The typical behaviour of the reactor power and of the reactor doubling time in dependence on reactivity for start-up and power changes is presented in Fig. 6.

At constant reactor power, the reactor period (or the doubling time as displayed on the control desk) is infinite. Any power increase or power reduction corresponds to a positive reactor period or a negative reactor period, respectively. After any reactivity change, the reactor takes about 1 min in order to show a stable reactor period T_S (time for getting the equilibrium between production of prompt and delayed neutrons).

At very small reactor power levels, the signal for the reactor period may fluctuate slightly because the statistical fluctuations of the neutron density have larger relative amplitudes at low powers than at high powers.

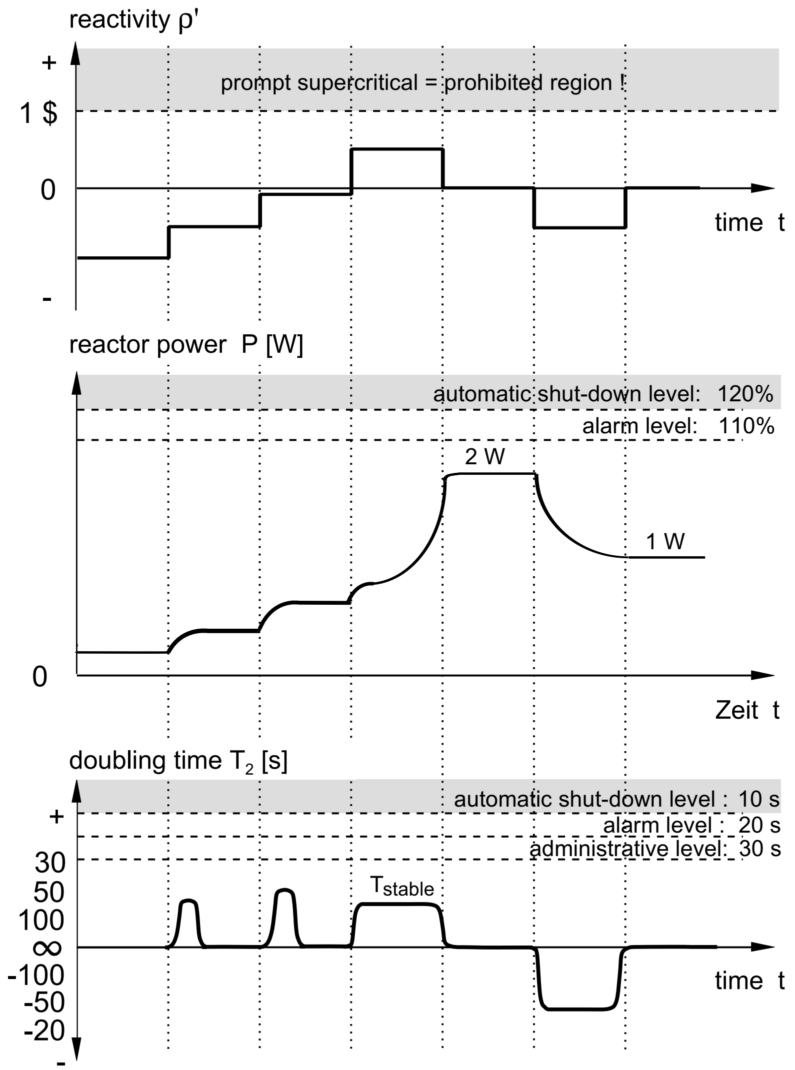


Fig. 6, Behaviour of reactor power and doubling time in dependence on reactivity ρ

6. Instructions Concerning the Protocol

The protocol should contain:
- short description of the experiment
- copy of the record in the operation logbook about the functional check
- log of the start-up procedure
- critical control-rod positions as a function of the power
  - measured values including error ranges
  - analysis and discussion of the control rod position
- measurement of gamma and neutron dose rates in dependence on the reactor power
  - cross section of the reactor hall with the points of the measurements indicated
  - graphical representation of the measured data including error ranges
  - analysis and discussion of the results

7. Index of Relevant Variables

n neutron density (proportional to the neutron flux density Φ, to the number of neutrons N and to the reactor power P)

k multiplication factor

ρ reactivity, $\rho = (k - 1) / k$

ß total fraction of delayed neutrons from fission (for U-235: ß = 0.641 %)

l neutron life time, $l^* = l / k$

λ averaged decay constant for precursors of delayed neutrons

C concentration of the delayed neutron precursors

T reactor period, i.e. time interval for reactor power change for the factor of $e \approx 2.71$

8. Questions to Answer

1. Which are the main components for construction and operation of a thermal reactor and how do they work?

2. What is a zero-power reactor?

3. Which parameter describes the reactor with regard to the criticality and the transient behaviour?

4. What is the purpose of the neutron source for operating a nuclear reactor?

5. What neutron-physical phenomenon enables the safe control of a nuclear reactor?
   Give a prove using the respective formalism!

6. Which components guarantee the safe operation of a nuclear reactor and how do they work?

7. Why should a reactor have a negative temperature coefficient of the reactivity?

8. What does a reactor operator has to take care for in a reactor start-up and what is the procedure of a start-up?

9. What does the reactor operator have to do in a safety-relevant event (exceptional event)?

References:

/1/ Glasstone, S., Edlund, M.C., The Elements of Nuclear Reactor Theory, New York 1952
/2/ Technical Description and Procedure of Operation for the Reactor Facility AKR-2, TU Dresden, Institute of Power Engineering
/3/ Ackermann, G. (ed.), Betrieb und Instandhaltung von Kernkraftwerken, Leipzig 1982