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Institut für Neutronenphysik und Reaktortechnik
Projekt Schneller Brüter

Two On-Line Methods for Routine Testing of Neutron and Temperatur Instrumentation of Power Reactors

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Abstract

Two on-line methods for in situ testing of neutron and temperature instrumentation of power reactors have been developed. They provide a means for monitoring the sensitivity and response time of neutron and temperature instrumentation including neutron detectors and thermocouples, respectively. These parameters characterize the over-all performance of a signal channel. Performance information of signal channels is of particular interest in safety systems where deviations from the normal and safe conditions of reactor operation have to be detected reliably and as fast as possible.

The testing procedures proposed and described in this report use inherent fluctuations or modulations of the physical quantities being measured as dynamic test input to the whole signal channel. They can be applied therefore during normal reactor operation at power. No additional testing equipment is needed. Signal channel performance information is obtained from the fluctuations of the available signals only using simplified noise analysis techniques.

Neutron instrumentation testing is based on the prompt jump in reactor power subsequent to single reactivity steps produced by the control system during normal operation to keep the power at the prescribed level. It is shown that the signal response to single steps can be clearly identified in most practical situations. Missing a prompt jump in the signal would be an indication of a significant failure. Small changes in the transmission characteristics can be detected by measuring the averaged step response.

For testing of outlet temperature instrumentation a different procedure is necessary. In the proposed method the relationship between temperature and power noise is used. It was found that the ratio of the maximum value of the cross correlation function between neutron and temperature noise signals normalized to the rms value of the neutron noise is a suitable quantity for monitoring the performance of temperature instrumentation in a reactor. Monitoring with small and large averaging time constants simultaneously enables quick indication of suddenly occuring significant failure and detection of small changes of the response characteristics, respectively.

Zwei Methoden zur betrieblichen Funktionskontrolle von Neutronenfluß- und Temperaturmeßkanälen in Leistungsreaktoren.

Zusammenfassung

Es wurden zwei on-line Methoden zur laufenden Funktionskontrolle von Neutronenfluß- und Temperaturmeßkanälen in Leistungsreaktoren entwickelt. Damit können Empfindlichkeit und Ansprechzeit der Meßkanäle einschließlich der Neutronendetektoren bzw. der Thermoelemente im Reaktor überwacht werden. Die Kenntnis dieser Parameter ist besonders wichtig für die Instrumentierung innerhalb des Sicherheitssystems, mit der Abweichungen vom normalen, sicheren Betriebszustand möglichst schnell und zuverlässig nachgewiesen werden müssen.

Die in diesem Bericht beschriebenen Prüfmethoden nutzen die betrieblichen Schwankungen der Meßgrößen als Prüfsignale für die gesamte Meßstrecke. Sie können während des normalen Leistungsbetriebes ohne zusätzliche Testeinrichtungen eingesetzt werden. Die gewünschte Information erhält man allein aus den Rauschanteilen der vorhandenen Signale nach vereinfachten Methoden der Rauschanalyse.

Die Prüfung der Neutronenkanäle beruht auf der Messung des prompten Sprunges in der Reaktorleistung während der Bewegung der Regelstäbe, durch die die Reaktorleistung bei Normalbetrieb konstant gehalten wird.

Zur Prüfung der Temperaturkanäle mußte ein anderes Verfahren entwickelt werden. Es beruht auf dem ursächlichen Zusammenhang zwischen Leistungs- und Austrittstemperaturschwankungen bei niedrigen Frequenzen. Als Prüfgröße eignet sich das durch den Effektivwert des Neutronenrauschsignals dividierte Maximum der Kreuzkorrelationsfunktion zwischen Neutronen- und Temperatursignal.

Contents

1. Introduction

Safe and reliable operation of a power reactor requires that partial failures in its safety installations do not reduce the availability of the safety system as a whole. This is accomplished by providing redundant installations and by adequate testing of their performance during reactor operation. For the majority of components of a safety system one can prove from time to time or continuously that they would work as scheduled by applying electronic testing signals and checking whether there is a proper response or not.

Presently two important exceptions from such active testing exist. One is the shut-off system, the other concerns the signal transducers of the safety instrumentation. Of course testing the performance of the shut-off system cannot include an actual shut-down of the reactor. It can only show that the safety rod drive mechanisms will be activated by a true scram signal. Actual movement of the safety rods has to be guaranteed by an absolute reliable mechanical design.

For the transducers of the safety instrumentation the situation is different. They can be tested directly by modulating the physical quantity which is to be monitored. Fortunately, in a power reactor exist inherent fluctuations (power noise) which can be utilized as test input to some of the safety instruments. Active testing of complete signal channels becomes feasible then without any interference with normal reactor operation. This paper describes two methods for active testing of neutron and temperature instrumentation of power reactors using noise analysis techniques.

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2. Theoretical considerations

2.1 Testing procedure for neutron instrumentation

The current testing of neutron instrumentation is restricted to the electronic networks. The neutron detector itself is not included. Malfunction of a neutron detector may be discovered from an intercomparison of signals from different neutron detectors. However, this method is not very sensitive because of flux tilting and burnup effects which produce significant changes in individual neutron signals even when there is no deviation from the normal behaviour of the detector.

Conversely, one could think of a malfunction which preserves the output level of a signal when the neutron flux and reactor power are changing. Such a failure would go undetected when the reactor is operated at a constant power level. Whether a neutron detector would respond properly to changes of reactor power or not can be tested by modulating the reactor power in such a way that the response of a neutron detector can be predicted by a theoretical model. This is possible for the normal actions of the control system to keep the reactor power at a prescribed level.

Usually this is accomplished by producing small reactivity steps with a control rod each time the prescribed upper or lower limit of reactor power is reached. In general the amplitude of a reactivity step is less than 1 ¢. The rise time of the step is in the order of a few tenths of a second. The resulting change δP in reactor power can be calculated from the kinetics equations of the point reactor model. Feedback effects can be neglected because of their large time constants.

For small sinusodial reactivity perturbations the kinetics equations can be linearised. Using conventional symbols one obtaines the reactivity transfer function /1/ in the frequency domain

$$H(i\omega) = \frac{\delta P(i\omega)/P}{\delta\rho(i\omega)/\beta} = \frac{\alpha}{i\omega(1+\alpha\sum\frac{a_I}{\lambda_I+i\omega})} \tag{1}$$

Neglecting delayed neutrons yields

$$\frac{\delta P(i\omega)}{P} = \frac{\alpha}{\alpha+i\omega} \cdot \frac{\delta\rho(i\omega)}{\beta} \tag{2}$$

It can be seen very easily that the prompt jump of the reactor power due to a reactivity step devided by the stationary mean power level has the same shape and amplitude as the reactivity step measured in $ units if the rise time of the step is larger than the prompt reactor period $R = \frac{1}{\alpha}$ by a factor of 2 π at least. This is true because the spectral composition of the power jump and the reactivity step are approximately the same then. Feedback effects and delayed neutrons can be neglected only if their time constants are larger than the rise time of the reactivity step.

The linear spectrum of a unity step function with the finite rise time τ is given by /2/:

$$\delta\rho(i\omega) = \tau \cdot \frac{(\cos(\omega\tau) - 1) - i\sin\omega\tau}{(\omega\tau)^2} \tag{3}$$

The shape of this spectrum is shown in Figure 1 together with a low-pass filter curve using $\tau = 0.1$ for the rise time and $R = \frac{1}{\alpha} = \frac{0,1}{2\pi}$ for the time constant of the reactor which in the prompt neutron approximation represents a first-order low-pass filter. It is seen that frequencies larger than $f_o = \frac{1}{\tau} = 10$ Hz do not contribute significantly to the reactivity step function.

The same statement is valid for the resulting prompt power jump and the corresponding signal of the neutron detector if the prompt reactor period R and the time constant T of the whole signal channel are sufficiently small compared to the step rise time. It has been shown /3,4/ for thermal and fast reactors that the neutron flux variations normalized to the mean flux do not depend on space variables. Therefore the relative change of a neutron signal is independent of the position of the neutron detector.

Assuming for the signal channel including the neutron detector a first-order low-pass characteristics

$$G(i\omega) = \frac{1}{1+i\omega T} \qquad (4)$$

the response in the neutron signal to a reactivity step can be obtained by multiplying the spectrum (3) by the transfer function (4) and applying straight-forward Laplace transform techniques to the result. One obtaines

$$\delta S(t) = \frac{1}{\tau} \cdot \begin{cases} t-(T+R)+\frac{R^2}{R-T}\, e^{-\frac{t}{R}} - \frac{T^2}{R-T}\, e^{-\frac{t}{T}} & \text{for } t \leq \tau \\ \tau+\frac{R^2}{R-T}\left(e^{-\frac{t}{R}}-e^{-\frac{t-\tau}{R}}\right)-\frac{T^2}{R-T}\left(e^{-\frac{t}{T}}-e^{-\frac{t-\tau}{T}}\right) & \text{for } t \geq \tau \end{cases} \qquad (5)$$

In Fig. 2 this function is plotted using the parameters of the KNK reactor at Karlsruhe $R = 3.5 \cdot 10^{-3}$ sec and $\tau = 0.1$ sec. The time constant of the signal channel was varied between $4 \cdot 10^{-3} \leq T \leq 0{,}4$ sec.

The dotted line indicates the signal step for $T = 1 \cdot 10^{-4}$ sec reproducing the reactivity step function to a good approximation. From eq. (5) and Fig. 2 it follows that an increase of the time constant of the signal channel including the detector results in a much smaller signal amplitude at the end of a reactivity step if the time constant T multiplied by 2 π exceeds the rise time τ.

In Table 1 the fractions of the signal step at two times t after the beginning of a reactivity step are listed for different values of the time constant T. Obviously, the difference of signal amplitudes prior and shortly after a control step is a quantity which is very sensitive to perturbations of the performance of a neutron detector and the connected electronic network. Monitoring this quantity would enable one to detect deviations from the prompt and linear response of the complete signal channel.

Measuring the absolute value of the signal channel response to a single reactivity step would immediately reveal failures which reduce the detector sensitivity significantly between two succeeding steps of the control system as, for instance, loss of filling gas, break down of the high tension supply and decrease in amplifier gain as well as overload conditions in the channel. Small changes of the performance of the whole channel could be detected by averaging the step response or low pass filtering the signal. This would reduce the effect of high frequency noise in the signal due to stochastic power noise which is always present in a power reactor under normal operating conditions. The low frequency noise for $f < 1/\tau$ is eliminated automatically from the quantity to be monitored because it is defined as a "high frequency" signal component by differentiating within a short time interval. The number of steps which have to be averaged depends on the frequency distribution of background power noise.

In general the power spectral density decreases rapidly with frequency. In the frequency range around $f = 1/\tau$ which is of interest here the amplitude of power fluctuations is smaller than the magnitude of the prompt power jump caused by a control step. In most practical cases it should be possible therefore to detect a single step response reliably if the signal channel is working properly. Missing a step in the signal indicates a malfunction of the channel.

The averaged absolute value of the step response depends on the ratio of local neutron flux at the detector position and reactor power. This ratio is changing with time due to burnup, reloading and other operating procedures during normal operation. The resulting influence on the step response can be roughly predicted by reactor calculations as a function of time. The magnitude and time scale of these effects on the step response represent a limitation to the sensitivity of the described method for detecting malfunctions of neutron instrumentation in a power reactor.

Monitoring the relative step response avoids this limitation. However, in this case a failure which changes the signal without disturbing its frequency composition significantly (decrease in detector sensitivity or amplifier gain) can not be detected. An optimum detection scheme would include monitoring both the absolute and relative step response as well as a comparison of the results from different signal channels to each other.

A process computer could perform all the necessary data processing very easily. Each time the control system is activated all the neutron signals are sampled at the beginning and shortly after completion of the reactivity step. Absolute and relative changes of the signals are calculated and compared with predetermined thresholds and to each other. Single steps and their average are monitored simultaneously and in the same way. If a deviation of normal behaviour is observed, a specific message is send to operator. The minimum value for the results is adapted in adequate time intervals to the current power distribution taking into account burnup and reloading of fuel elements.

If the reactivity steps produced by the control system have variable amplitudes the single step responses of the neutron signal have to be normalized to the magnitude of the reactivity step first before the described testing procedure is applied. The magnitude of the reactivity step can be obtained from the control rod position prior and after the step. To eliminate burnup and flux profile variations the control rod has to be recalibrated in suitable time intervals.

2.2 Testing procedure for temperature instrumentation

Testing of temperature instrumentation is of particular interest for sodium cooled fast reactors. Due to the high power density in these reactors flow obstructions in a fuel element have to be considered as initiating events for dangerous accidents which have to be detected at an early stage by monitoring the coolant temperature at all fuel element outlets. This temperature instrumentation will be included in the safety system of LMFBR's. Therefore it should be proved continuously during reactor operation that the temperature signals would reveal abnormal outlet temperatures reliably and as fast as possible.

The testing procedure for the neutron instrumentation described in the foregoing paragraph is not applicable to the temperature instrumentation due to the large time constants of the heat transfer and the large amplitude of outlet temperature fluctuations at low frequencies caused by the normal (stochastic) power noise.

A simple relationship between power and temperature fluctuations can be **derived** from a lumped parameter model of the heat transfer in a fuel element. In a two region model /5,6/ consisting of fuel and coolant only one obtains the power- to temperature transfer function

$$\frac{\delta T_p(i\omega)}{\delta P(i\omega)} = F(i\omega) = \frac{A}{(1+i\omega\tau_1)(1+i\omega\tau_2)-B} \qquad (6)$$

where A, B are constants describing the heat transfer and τ_1, τ_2 denote the time constants for heating of fuel and coolant. $T_p(i\omega)$ represents the temperature fluctuation which is caused by the power fluctuation $\delta P(i\omega)$ only. Because the time constants of the heat transfer are in the order of a second there is no prompt jump in outlet temperature signals as in the neutron signal when the control rod moves. The outlet temperature can only follow the rather slow changes in reactor power subsequent to the prompt jump. This part of the step response of reactor power is determined by delayed neutrons and reactivity feedback which cannot be calculated precisely. Furthermore, the slow part of the step response is superimposed by the large low frequency component of background power noise. Therefore the response of a fuel element outlet temperature signal to a normal reactivity step by the control rod can be neither predicted nor measured with sufficient accuracy.

A different method for monitoring the performance of temperature instrumentation is proposed therefore which utilizes the large stochastic fluctuations of reactor power at low frequencies as an indirect test input for the temperature instrumentation. These fluctuations can be measured with a neutron detector. The resulting ac component $\delta S_p(i\omega)$ of the temperature signal can be calculated in the frequency domain using the transfer functions (6) and (4) when for the temperature instrumentation including the thermocouples a first-order low pass characteristics is assumed again as for the neutron instrumentation. We obtain

$$\delta S_p(i\omega) = K(i\omega)\delta P(i\omega) = \frac{A.C}{[(1+i\omega\tau_1)(1+i\omega\tau_2)-B](1+i\omega T)}\delta P(i\omega) \qquad (7)$$

where C is a calibration factor relating the temperature with the temperature signal.

Operational testing the performance of temperature instrumentation of a power reactor can be based directly on this equation if a frequency interval $\omega_1 \leq \omega \leq \omega_2$ exists wherein power noise is the only source of temperature noise. Either gain or phase relations at suitably selected frequencies can be checked for testing purposes. Monitoring the complete frequency spectra explicitely would be a too complicated procedure for real-time applications at power reactors because frequency spectra of many temperature and neutron noise signals would have to be calculated continuously.

However, malfunctioning of a signal channel can also be detected by monitoring the ratio γ of the mean square values of the two signals which can be measured much easier. According to /7,8/ this ratio is given by the equation

$$\gamma \equiv \frac{\overline{\delta S_p^{\,2}(t)}^{L}}{\overline{\delta P^2(t)}^{L}} = \frac{\int_{\omega_1}^{\omega_2} \delta S_p(i\omega)\,\delta S_p^{*}(i\omega)\,d\omega}{\int_{\omega_1}^{\omega_2} \delta P(i\omega)\,\delta P^{*}(i\omega)\,d\omega}$$

$$= \frac{\int /K(i\omega)/^2 /\delta P(i\omega)/^2 / L \; d\omega}{\int /\delta P(i\omega)/^2 / L \; d\omega} \qquad (8)$$

where $\delta S_p(i\omega)$ and $\delta P(i\omega)$ are the frequency spectra of the finite signal records $\delta S(t)$ and $\delta P(t)$ of length L in the time domain. $-L$ and $*$ denote the time average over L and the conjugate complex of a quantity, respectively. The square modulus of the two spectra devided by L defines statistical estimates of their auto power spectral densities. The original signals have to be band-pass filtered to restrict their evaluation to the frequency range $\omega_1 < \omega < \omega_2$ where power noise is the only source of the outlet temperature noise.

The ratio γ of the mean square values is equal to the first moment of the square modulus of the transfer function $K(i\omega)$ (7) using the auto power spectral density of the power noise as a weighting function. If the parameters of the heat transfer function (6) can be assumed as time-independent numbers and the power spectral density $\frac{/\delta P/^2}{L}$ does not change significantly with time the value of γ will be constant too provided there is no change of the signal transmission characteristics. Any malfunction of the signal channel which influences either the gain or response time however, would change γ. Thus, the mean square of the temperature fluctuation due to power fluctuations devided by the mean square of these power fluctuations is a suitable quantity to be monitored for dynamic testing of temperature instrumentation in the core of a power reactor.

In general there will be other independent sources of noise in the coolant outlet temperature besides of reactor power noise representing a background in the whole frequency range of interest. This background noise can be eliminated by cross correlating temperature and neutron (power) noise signals. Consequently, the maximum value of the cross correlation function at the delay time t_m, $\overline{\delta S(t)\cdot\delta P(t-t_m)}$, divided by the mean square value $\overline{\delta P^2(t)}$ (which is equal to the maximum value of the auto correlation function of power noise) is to be monitored instead of the ratio γ from eq. (8). This ratio q depends on the transfer function $K(i\omega)$ in a slightly different way than γ:

$$q = \frac{\overline{\delta S(t)\cdot\delta P(t-t_m)}^{L}}{\overline{\delta P(t)^2}^{L}} = \frac{\int_{-\infty}^{\infty}\delta S_p(i\omega)\cdot[e^{-i\omega t_m}\delta P(i\omega)]^{*}d\omega}{\int/\delta P(i\omega)/^2 d\omega}$$

$$\cong \frac{\int/K(i\omega)/\ /\delta P(i\omega)/^2 d\omega}{\int/\delta P(i\omega)/^2 d\omega} \qquad (9)$$

with $\delta S(t) = \delta S_p(t)+\delta S_x(t)$, $\delta S_x(t)$ uncorrelated background.

The last equation is valid only if $/\delta P/^2$ decreases rapidly with increasing ω so that for $\omega \gtrsim \frac{1-B}{2(\tau_1+\tau_2)} = \omega_o$ there is no significant contribution to the integrals in eq. (9). This is true in most practical cases. Measured power spectral densities in general have an absolute maximum value at an angular frequency in the order of 10^{-2}/sec and decrease toward higher frequencies. For frequencies smaller than ω_o the phase angle of the heat transfer function (6) can be approximated by

$$\angle F(i\omega) = \text{arc tg} - \frac{(\tau_1+\tau_2)\omega}{1-B-\tau_1\tau_2\omega^2} \approx - \frac{\tau_1+\tau_2}{1-B}\omega \approx - t_m\omega \qquad (10)$$

Thus, for $T \ll \tau_1, \tau_2$ we can write

$$K(i\omega) = e^{-i\omega t_m}/K(i\omega)/ \qquad (11)$$

This means that in the time domain there is a delay t_m between the power fluctuations and the resulting temperature fluctuations and a smoothing effect on the temperature signal described by $/K(i\omega)/$.

From eq. (9) it follows that q is equal to the first moment of the absolute value of the transfer function (7) with $/\delta P/^2$ as weighting function. This can equally be used for testing of temperature instrumentation as the ratio γ of the mean square values of temperature and power noise.

For routine testing of temperature instrumentation by a process computer the signals of a neutron detector and of all the temperature measuring channels to be monitored have to be sampled at a low sampling rate which is determined by the time constants τ_1, τ_2 of the heat transfer or the bandwidth of the power fluctuations ($\omega \leq \omega_o$).

The computer then would have to calculate continuously the cross correlation functions between the temperature signals and the neutron signal at delay time t_m and the mean square value of the neutron signal (after subtracting the dc components) according to eq. (9) using digital (RC or exponential) filtering for averaging of the current signal products. The averaging time constant L is chosen by compromising between high sensitivity and quick response to malfunctions and low rates of false alarms. Finally the individual q values obtained are compared with predetermined thresholds to check whether there are deviations from normal behaviour of the signal channels.

All these operations have to be performed within the sampling time interval for real-time application. At larger periods of time the individual thresholds for the q values have to be recalculated to account for changes of the power distribution due to burnup and reloading of fuel.

3. Experimental results

To confirm the theoretical predictions about the two methods for testing of neutron and temperature instrumentation measurements have been performed at the reactor KNK I at Karlsruhe.

KNK is a sodium cooled zirconiumhydrid moderated reactor of 58 MWth nominal power. Neutron signals were obtained from ionisation chambers placed outside of the reactor vessel. Sodium outlet temperature was monitored by thermocouples on top of the individual fuel element outlets. The reactor itself is unstable due to a positive temperature coefficient of reactivity. Therefore the control system is activated rather frequently. Each time when the outlet temperature reaches the setpoints the control rod produces reactivity steps of constant amplitude. The shape of the reactivity steps is indicated by the dashed line in Fig. 2. In a reactor with a negative power feedback the frequency of control steps might be too low for testing purposes. Then additional control steps could be produced to limit the test interval for the neutron instrumentation.

In Fig. 3 sample records of low-pass filtered neutron and temperature noise signals obtained at KNK I at full power are shown. The corner frequency of the low-pass filter was 0.4 Hz in all cases. The vertical dashed lines with arrows indicate time and direction of control steps. The prompt response to single steps is clearly seen in the neutron signal in spite of the large background noise. In the temperature signal the step response is not observed.

Fig. 4 shows the averaged step response of a neutron signal to approximately 300 steps of the control rod (during 5 hours of reactor operation). The prompt jump of the signal equals 200 mV $\pm$ 4 %. Using the same gain factor for the mean value of the signal yields a dc component of 180 V. For the reactivity worth of a control step we obtain 0.11 ¢ therefore. The signal response of each individual control step could be measured reliably with a standard deviation of $\pm$ 61 %.

The testing procedure for temperature instrumentation was applied to the signals from thermocouples at the outlet of two fuel elements at different radial core positions. A minimum averaging time of 2 min was found for obtaining always positive values for normalised cross correlation between temperature and neutron signals as defined in eq. (9). The standard deviation of the q values was $\pm$ 50 % and $\pm$ 57 % for the two signal channels, respectively. In Fig. 5 two short-time cross correlation function estimates as obtained from neutron and temperature signals after 1 min of measurement time are plotted. For comparison, the cross correlation function from 5 hrs long records of the same signals is also shown in the figure. The standard error of the maximum value at $\tau \approx 6$ sec is reduced to 4 % in this case. For testing of temperature instrumentation only the cross correlation for a fixed delay time between the two signals $\tau \approx t_m$ according to eq. (10) has to be measured.

4. Conclusion

Results from theoretical considerations and preliminary measurements have shown that the reactivity modulation by the control system and the inherent stochastic fluctuations of reactor power can be used for real-time routine testing of neutron and temperature instrumentation of power reactors. The testing procedures use noise analysis techniques and can be applied at normal operating conditions without disturbing the reactor operation.

Acknowledgement

The author is much indepted to Mr. P. Hoppé and Dr. F. Mitzel for providing tape records of KNK signals and to Mr. H. Massier for his support in the digital data processing.

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Table 1 Signal fractions for different response time constants T

$t/_\tau$	$2\pi\ T/_\tau$						
	0,25	0,5	1,0	π	2π	4π	8π
1	0,94	0,90	0,82	0,55	0,36	0,21	0,11
1,5	1,00	1,00	0,99	0,82	0,60	0,37	0,21

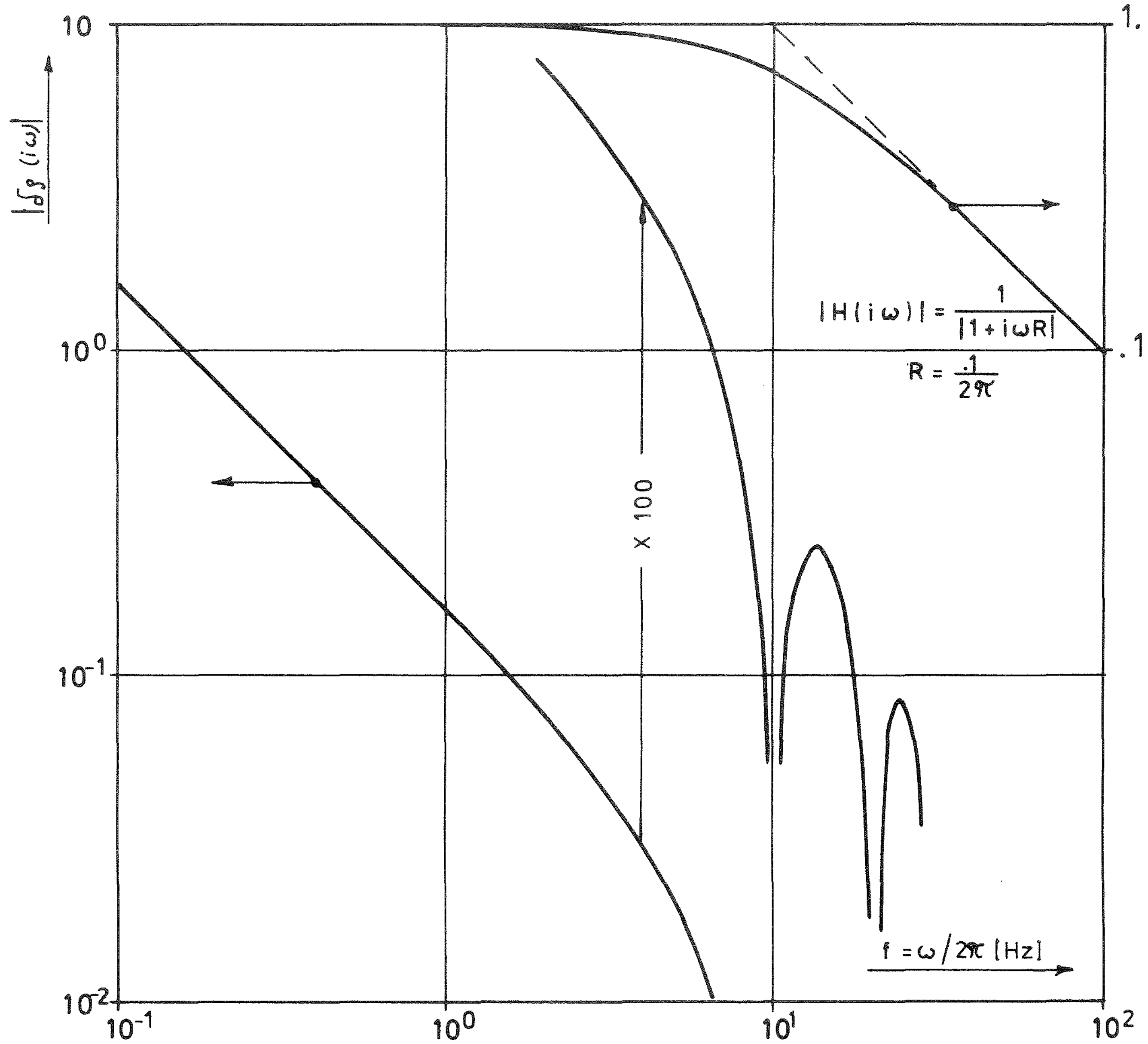

Fig. 1 Spectral composition of control step

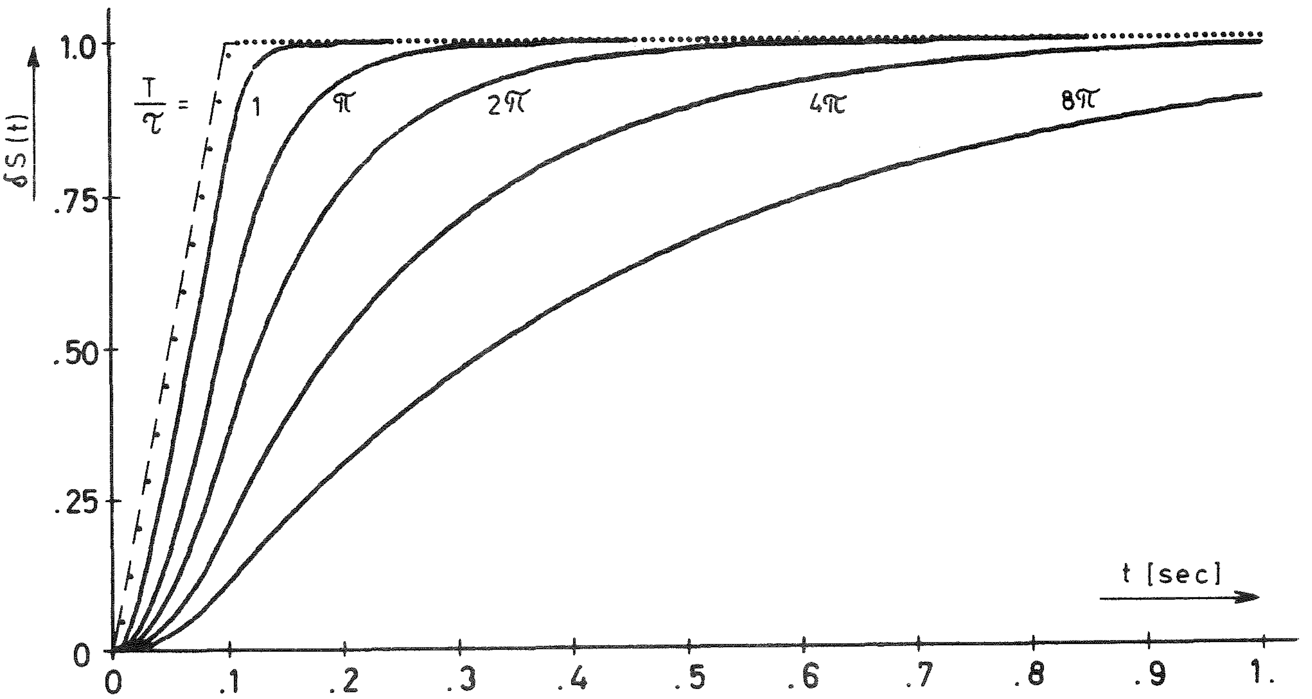

Fig. 2 Signal responses to control step (Eq. 5)

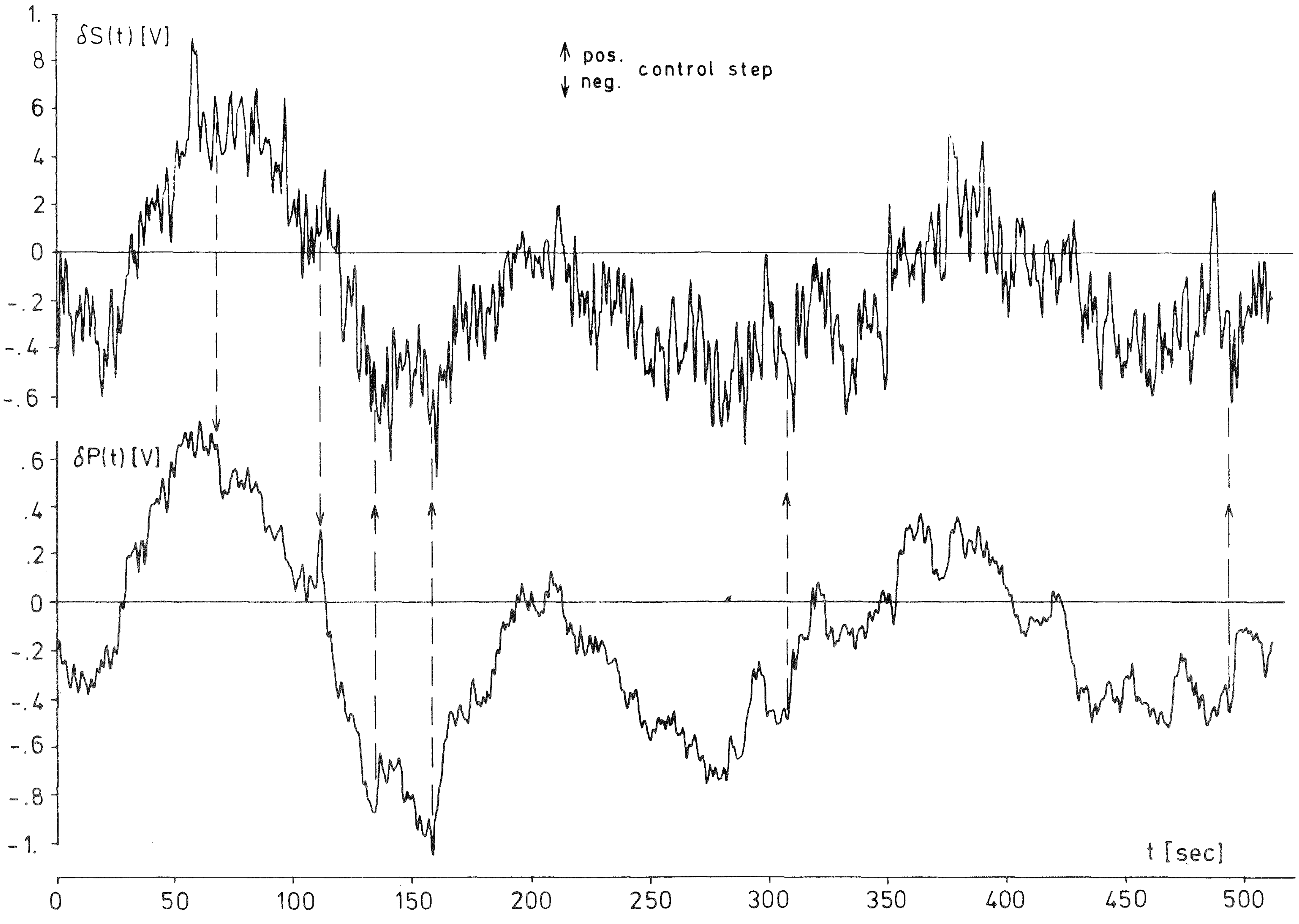

Fig. 3 Neutron (δP) and temperature (δS) noise signals from KNK I

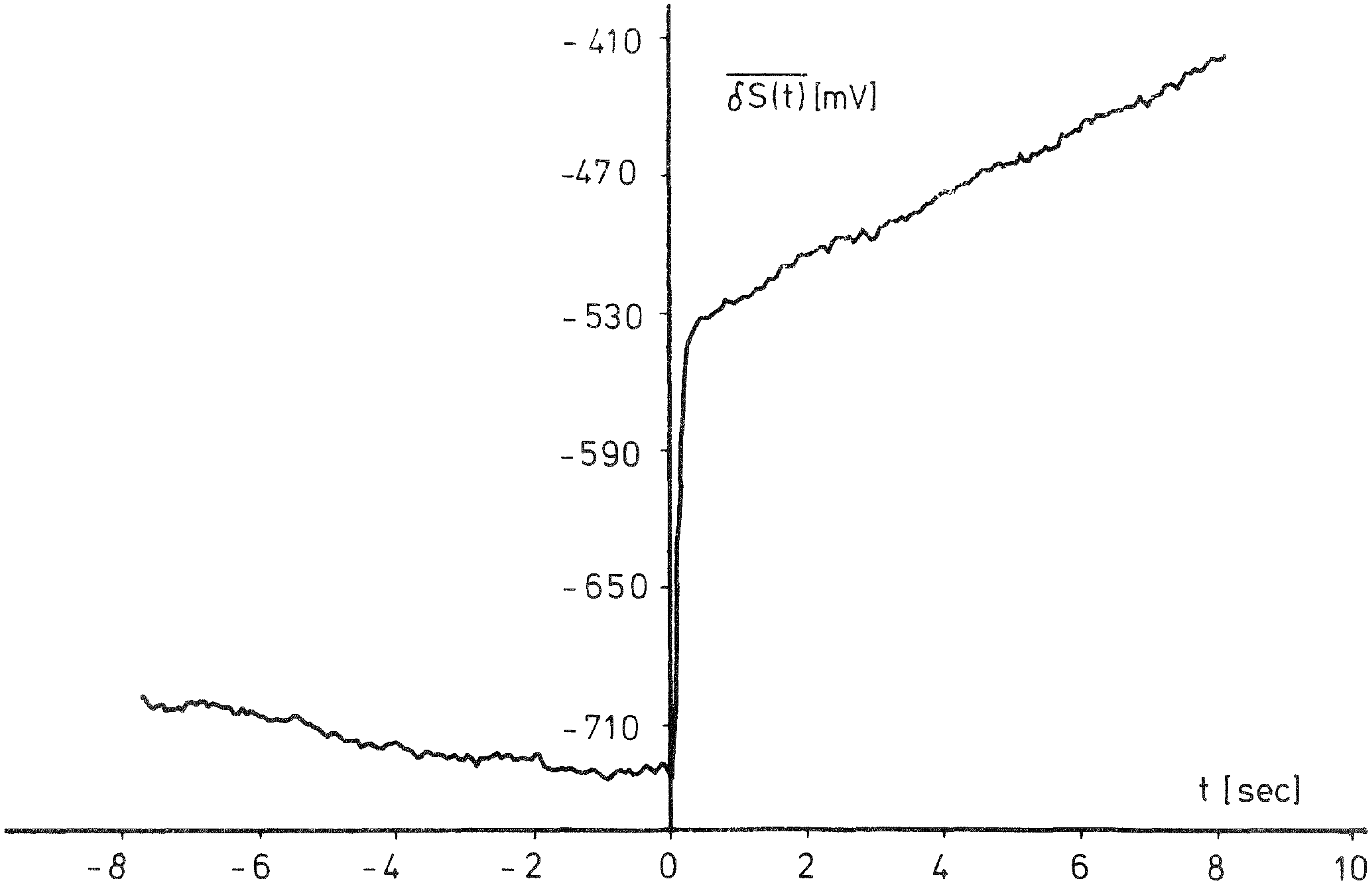

Fig. 4 Averaged reactivity step response of neutron signal

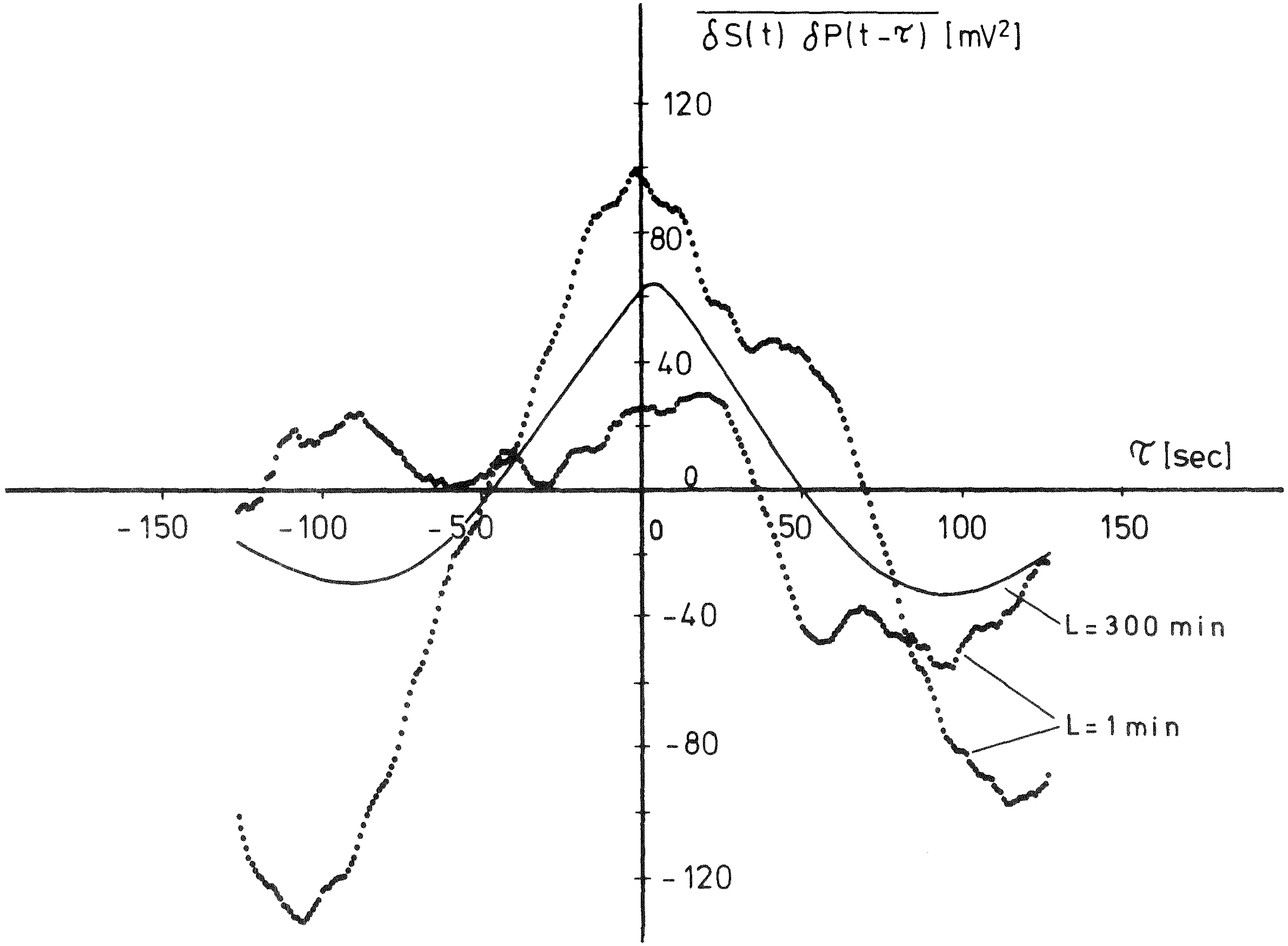

Fig. 5 Cross correlation functions calculated from signal records of diffent lengths L