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Studsvik Report

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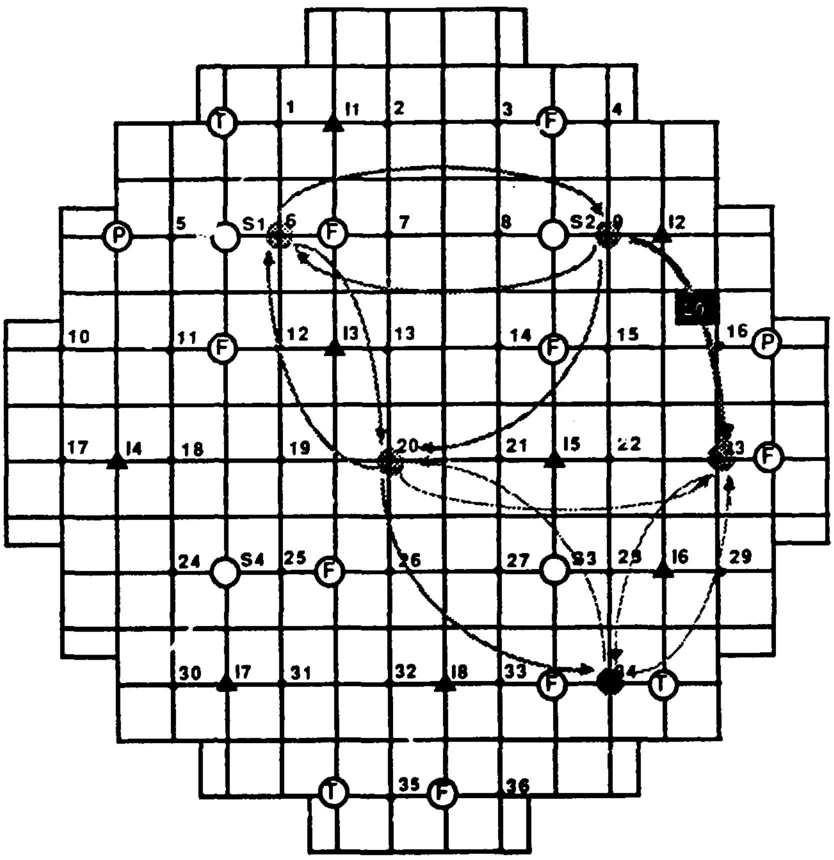
INVESTIGATION OF BWR STABILITY IN FORSMARK 2 BASED ON MULTIVARIABLE NOISE ANALYSIS

R Oguma, F Reisch, B-G Bergdahl, J Lorenzen, F Åkerhielm, S Kellner



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INVESTIGATION OF BWR STABILITY IN FORSMARK 2

Based on Multivariable Noise Analysis

(Related to Studsvik Arbetsrapport NI-88/2 on BWR stability study in F1)

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Sammanfattning

En serie brusmätningar har genomförts vid Forsmark-2 under effektuppgången efter revisionen sommar 1987. Målsättningen med mätningen var analysen av BWR-stabiliteten. Med hjälp av processidentifiering och signal-path-analysis har en resonant effektsvängning vid 0.5 Hz studerats, som förstärks vid hög effekt och lågt HC-flöde samt påverkas av en yttre störning i form av en trycksvängning.

Med analysen har källan till svängningarna bestämts samt stabilitets-villkoret for F-2 angetts med måttet Decay Ratio (DR). Dessutom har andra säkerhetsrelaterade problem till BWR-stabiliteten beskrivits.

Resultatet indikerar att effektsvängningen beror på en dynamisk koppling mellan neutron kinematiken och termo-hydrauliken via voidreaktivitetsåterkoppling. Värdet för Decay Ratio vid 63% effekt och 4100 Kg/s summa kylflöde blev så högt som DR ~0.7, (där DR=1 innnebär instabilitet). Analysen av tryckstörningen vid 0.33 Hz gav vid handen att om en sådan svängning sammanfaller med effektsvängningen vid 0.5 Hz (härdresonansen) kan detta utlösa snabbstopp.

Den föreliggande analysen belyser vikten av on-line monitering av BWR-stabilitetsmarginalen samt diagnos av reglersystem.

Abstract

A series of noise measurements have been conducted at the Forsmark-2 reactor during its start-up operation after the revision in 1987. The main purpose was to investigate the BWR stability problem based on noise analysis, i. e. the problem of resonant power oscillation with frequency of about 0.5 Hz, which tends to arise at high power and low core flow condition.

The noise analysis was performed to estimate the noise source which gives rise to the power oscillation, to evaluate the stability condition of the Forsmark-2 reactor in terms of the decay ratio (DR), as well as to investigate a safety related problem in connection with the BWR stability.

The results indicate that the power oscillation is due to dynamic coupling between the neutron kinetics and thermal-hydraulics via void reactivity feedback. The DR reached as high as ~0.7 at 63% of the rated power and 4100 Kg/s of the total core flow. An investigation was made for the noise recording which represents a strong pressure oscillation with a peak frequency at 0.33 Hz. The result suggests that such pressure oscillation, if the peak frequency coincided with that of the resonant power oscillation, might become a cause of scram.

The present noise analysis indicates the importance of a BWR on-line surveillance system with functions like stability condition monitoring and control system diagnosis.

CONTENTS

I. Introduction

A series of noise measurements has been carried out in two boiling water reactors(BWRs), Forsmark-1 (F-1) and -2 (F-2) in order to investigate the BWR stability problem, i.e. a resonant power oscillation seen at frequencies between 0.1 and 1 Hz.

The results of noise analysis for F-1 have already been presented in the previous report[1], where the objective and scope of the present work, experimentals and highlights from the results obtained are described together with proposals for continuation of the work to get further insight into the BWR stability problem.

The present report is the second volume on the BWR stability analysis, which is primarily concerned with F-2. Forsmark-2 is a same type of reactor as F-1, i.e. ASEA-ATOM BWR with internal recirculation pumps. One feature which distinguishes from F-1 is the core configuration with fuel loading. Namely approximately 50% of the core is occupied by Svea rods and the rest is by 8x8 rods.

The noise experiments were carried out in F-2 during reactor start-up operation after the revision in 1987. The experiments consisted of two kinds;

- recording noise signals, i.e the signal's natural fluctuation around the mean value, in an operational region with less stability margin and
- recording signals while applying small artificial random perturbations to a pressure control system.

As already discussed in the previous report and other open publications[2,3] there are in principle three types of instabilities considered possible in operating BWRs;

- plant instability, which is related to the whole plant dynamics including various control systems in the reactor plant,
- channel instability or local instability, which is due to feedback between flow rate and steam void in the heated channel,
- reactivity instability or core instability (hereafter we call it core instability), which is concerned with chain loop of dynamic interaction consisting of neutron kinetics, fuel heat transfer characteristics, coolant thermal-hydraulics and negative reactivity feedback of the void.

These three instabilities may be related with each other.

The main purpose of the analysis is, therefore, addressed on the following points;

a) to get insight into the mechanism of reactor instability which may be associated with F-2 among those mentioned in the above,
b) to evaluate the stability condition for F-2 in an operational region with less stability margin,
c) to investigate safety related problems which may occur in conjunction with the current reactor stability problem.

In Sec. II, we present results of the analysis for a), in which the possible source of the resonant power oscillation for F-2 is investigated using a multivariable noise analysis method. In Sec. III, we present the pressure to neutron transfer functions obtained via process identification for measurements with artificial perturbations and with natural noise only, respectively. The result is used to study the BWR instability mechanism. In Sec.IV, we evaluate the stability margin for F-2. In Sec. V, we discuss the problem c) which may arise from interaction between plant and core dynamics. The concluding remarks is givan in Sec VI.

II. Evaluation of Noise Source for The Resonant Power Oscillation

II-1. Power Spectral Density

Figure 2.1 shows a typical example of power spectral densities (PSDs) of the average power range monitor (APRM) for F-1 (a) and F-2 (b), respectively. The signal measurement conditions are summarized in Table 2.1. From these, it is seen that the noise measurement condition was nearly the same between the two measurements.

reactor plant	Forsmark-1	Forsmark-2
reactor power(%)	63.7	63.1
total core flow rate (Kg/s)	4774	4104
signal's stand. deviation (%)	0.844	0.874

Table 2.1 Noise measurement conditions of APRM signals for F-1 and F-2, respectively, which are used for the calculation of PSDs shown in Fig. 2.1.

In both PSDs, there is a sharp resonance in the frequency at 0.5 Hz in F-1 and at 0.45 Hz in F-2. The magnitude of signal variation is nearly the same. This means that the both reactors have a similar power oscillation with a period of about 2 sec. At the same time, the PSD in F-2 has another weak peak at about 0.1 Hz, representing more complicated PSD pattern than that in F-1.
In this section we investigate the main source of these PSD peaks.

II-2. Multivariable Noise Analysis

Figure 2.2 shows an example of measurement signals for APRM, reactor pressure (RPRS), total core flow (TCFL), steam flow in loop-1 (STFL1) and steam flow in loop-2 (STFL2), respectively, for the noise measurement in F-2 specified in Table 2.1. The corresponding PSDs are shown in Fig. 2.3. It is seen that the RPRS has two weak peaks at about 0.3 and 0.8 Hz, and the TCFL also a resonant peak at about 0.5 Hz. A sharp resonance is seen in STFL1 at 0.8 Hz and also vague two peaks in STFL2 at about 0.35 Hz and 0.8 Hz.

Our interest here is to identify the process which induces the resonant power oscillation. In order to achieve this, we apply a spectrum decomposition technique called noise contribution analysis[4]. Using a multivariable noise analysis technique, it is possible to decompose the PSD of the variable under evaluation into contributions of other variables, thus allowing the evaluation of how much in percentage the signal variation is influenced by other variables. The result is shown in Fig. 2.4; Figure 2.4-(a) is interpreted such that the power influence from plant variables to neutron flux is at most about 20%, indicating that their influence to the PSD peaks of APRM is small. On the other hand, relatively strong influence is seen from APRM to RPRS (Fig. 2.4-(b)) and also to STFL2 (Fig. 2.4-(e)) at about 0.1 Hz and 0.45 Hz. The TCFL is not influenced by the other variables (Fig. 2.4-(c)).

From the present analysis we can deduce the followings;
- Typical PSD peaks in the APRM are independent of plant variables and hence induced inside the core.
- Signal variations of the APRM are propagated to RPRS and STFL2.
- APRM and TCFL, although they have a PSD peak in the same frequency region, are slightly correlated with each other.

The result obtained here is qualitatively in good agreement with that for F-1 [1].

Further analysis was made for local power range monitor (LPRM) and local core flow (LCFL) signals to evaluate local stability condition, which resulted in the similar conclusion for F-1. Namely, the local core flow is less influencial to the power oscillation. In addition, the coherence analysis for local core flow exhibited that the correlation among LCFL signals of different channels is weak in frequencies of interest.

II-3. Discussions

In the light of the aforementioned three types of instabilities, it is most likely that the resonant power oscillation in F-2 is related to the core instability which arises from the neutron and thermal-hydraulics coupling with strong void reactivity feedback.

Concerning the neutron and thermal-hydraulics coupling, at the same time, the present analysis points out that there is little coupling between the neutron and total core flow. However, this does not mean that the two phase flow characteristic along the core does not play any significant role. On the contrary, it should be essential for the BWR stability problem. The fact is that the core flow measurement in ASEA-ATOM reactors is based on the pressure drop measurement at the inlet of selected channels. Accordingly, the present result can be interpreted such that there is no strong coupling between the neutron and core inlet flow fluctuations. On the other hand, GE reactors are equipped with an instrumentation to measure the pressure drop over the core height. Noise analysis of this signal demonstrated a strong influence to the neutron flux at the frequency of resonant power oscillation[5], indicating that the void generation and propagation process inside the core and its interaction with neutron flux is essential for the BWR stability problem.

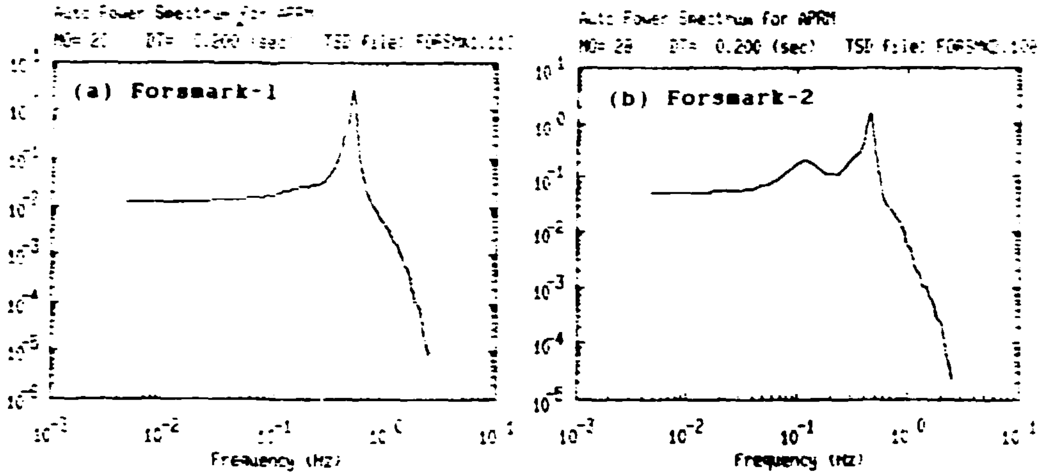


Fig. 2.1 Power spectral densities of APRM signal for Forsmark-1 (a) and for Forsmark-2 (b), respectively, measured at about same operational condition.

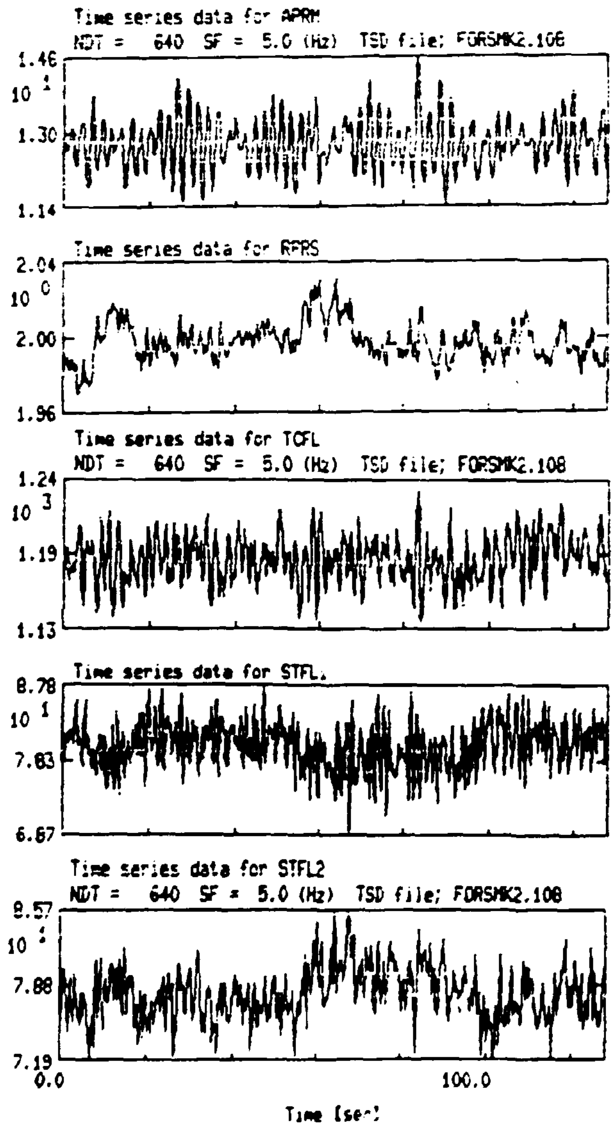


Fig. 2.2 An example of recorded signals which are used for the stability analysis of Forsma k-2.

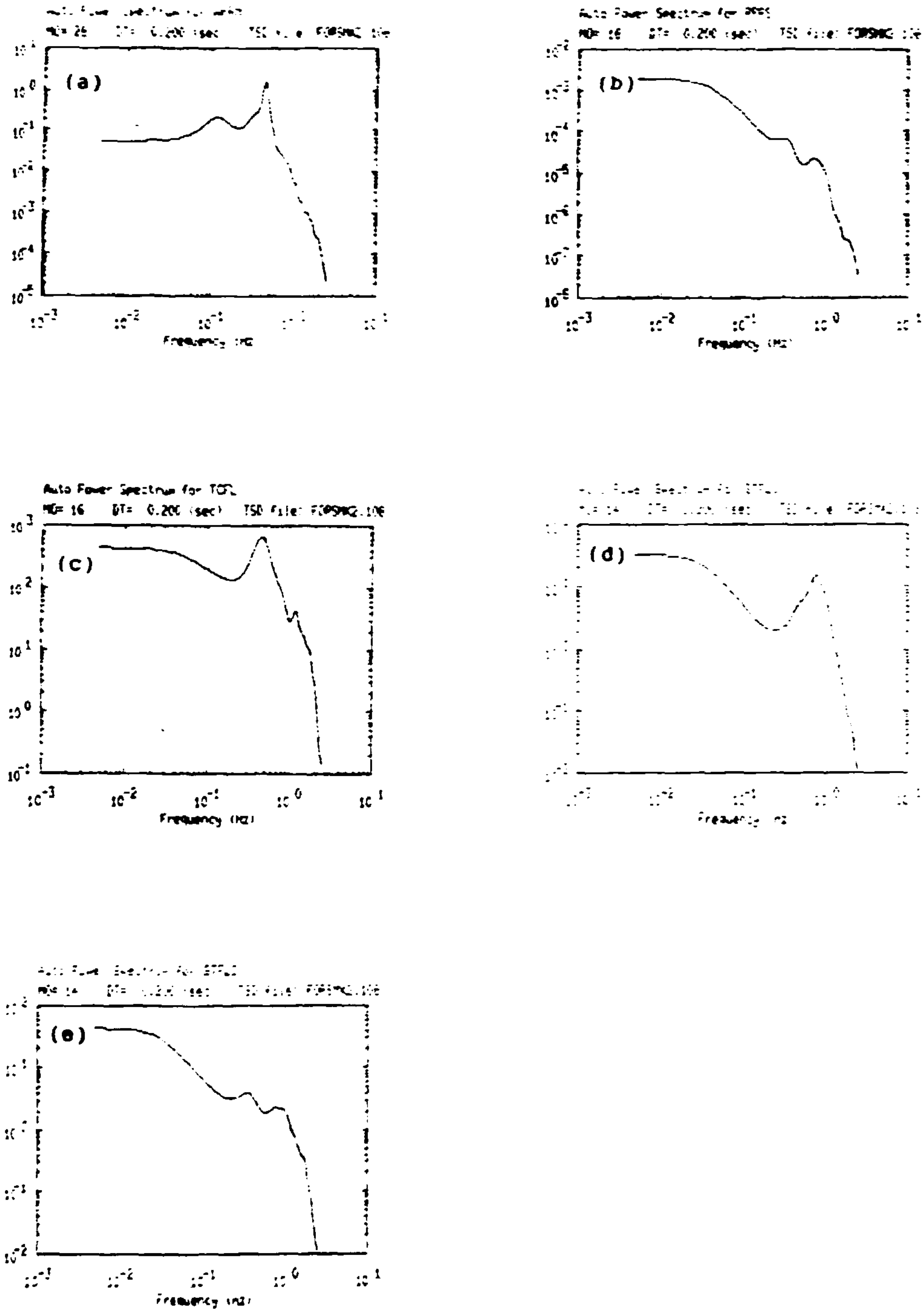


Fig. 2.3 Power spectral densities for noise signals of APRM (a), reactor pressure (RPRS) (b), total core flow (TCFL) (c), steam flow in loop-1 (STFL1) (d) and steam flow in loop-2 (STFL2) (e), respectively, measured under the condition described in Table 2.1.

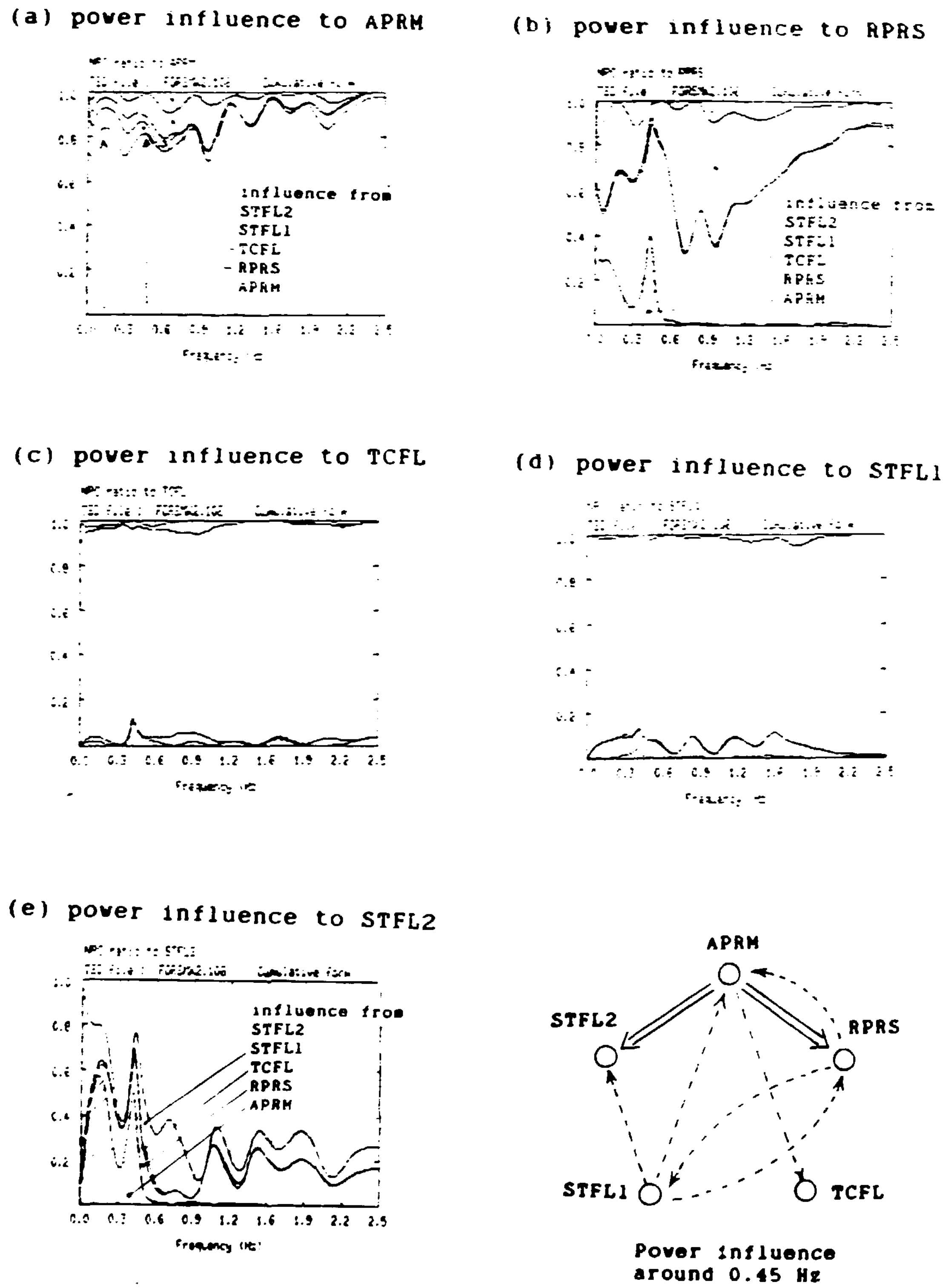


Fig. 2.4 Result of the noise power contribution analysis for APRM, RPRS, TCFL STFL1 and STFL2, indicating little influence from plant variable to neutron flux (a), relatively strong influence from APRM to RPRS and to STFL2 around 0.1 Hz and 0.45 Hz (b and e), and weak correlation between APRM and TCFL (a and c).

III. Estimation of Pressure to Neutron Transfer Function

III-1. Purpose

In one of the noise experiments, we recorded signals while applying random perturbations to the pressure signal. The reactor power and the core flow conditions during the experiment were same as those in Table 2.1. The perturbation was given to the set-point of pressure control system so that the pressure varies randomely with the magnitude of 0.6 bar in peak-to-peak. Figure 3.1 shows an example of signal recording for APRM and RPRS during the perturbation experiment, and Fig. 3.2 the PSD of these signals. The standard deviation of APRM was 2.8% around the mean value and thus the signal variation was about 3.2 times larger than the measurement without perturbation.

The purpose of the study was two fold;

1. Accurate determination of the reactivity to neutron transfer function.
   By applying the perturbation from the pressure, we can give disturbances to the void generation process. Therefore, the transfer function thus obtained can be interpreted as the reactivity to neutron transfer function, leading to the key information on the core stability problem as mentioned in Sec. I.
2. Accurate estimation of the reactor stability margin.
   The DR which is determined from this transfer function gives us a reliable information on the reactor stability condition and can be used to determine the stability margin for F-2.

III-2. Transfer Function and Impulse Response from Pressure to Neutron

The pressure to neutron transfer function was identified using a model identification technique. In this procedure, we assume the following difference equation as the dynamics model from the pressure (P) to neutron (N);

$$N(t) = \sum_{m=1}^{M} A(m)N(t-m) + \sum_{m=1}^{M} B(m)P(t-m) + E(t) \qquad (3\text{-}1)$$

where $\{A(m), B(m); m=1, .., M\}$ are model parameters which are estimated by means of the least-squares fitting to the experiment data. Parameter M denotes the model order and E(t) the noise term.

Once the model is obtained the transfer function or frequency response function is calculated by simply Fourier transforming the identified model. The impulse response can be also calculated by applying an impulse to P(t) in Eq. (3-1).

Figure 3.3 shows thus obtained transfer function. A very sharp peak is seen in the gain curve at 0.45 Hz. The phase curve exhibits a sudden shift by more than 90 degree there. This result shows qualitatively good agreement with similar measurements with Asea-Atom reactors[6] and GE reactors[7-9]. Therefore it is understood that there is a potential mechanism in the reactivity to neutron transfer function which gives rize to the resonant power oscillation under the condition that random noise in the void generation and propagation process persistently excites this process in the core.

The impulse response calculated from the identified model is shown in Fig. 3.4. The DR estimated from the response curve is 0.73 ± 0.03.

Similar analysis was performed using the noise data which we studied in Sec. II. Figures 3.5 and 3.6 show the corresponding transfer function and impulse response, respectively. The pattern of the gain and phase curves is fairly similar to those in Fig. 3.3 but represents considerable difference. The gain is much smaller around 0.45 Hz. The DR calculation resulted in 0.63 ± 0.09, leading to smaller value, i.e. less conservative value than that from the perturbation experiment.
This is because there is not sufficient signal excitation over wide frequencies so as to permit the identification of the dynamics from the pressure to neutron. This guess is supported by the analysis made in Sec. II, where the signal power propagation from the pressure to neutron was shown to be less than that to the opposite direction.

Here it should be kept in mind that as discussed in the previous report[1], the reactor dynamics in connection with the BWR stability problem is nonlinear. Therefore, in principle, the present result holds true for a reactor operational condition only in the neighborhood of the current experiment condition.

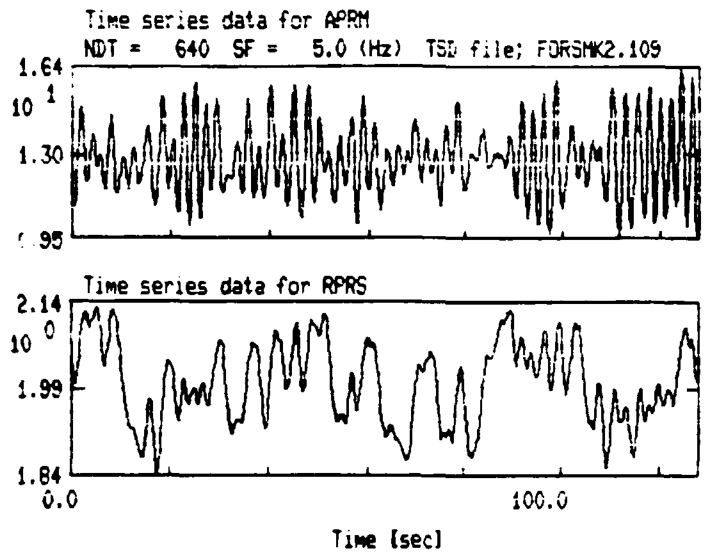


Fig. 3.1 Time behaviour of the noise signals which were recorded during the experiment with artificial random perturbations.

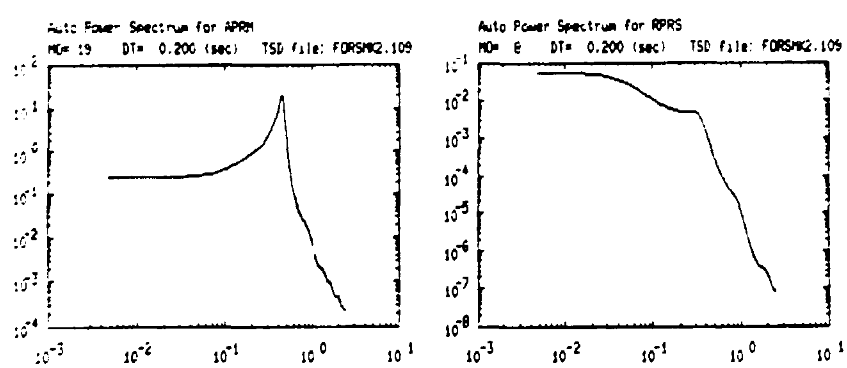


Fig. 3.2 Power spectral densities for APRM and RPRS signals recorded during the experiment with artificial random perturbations.

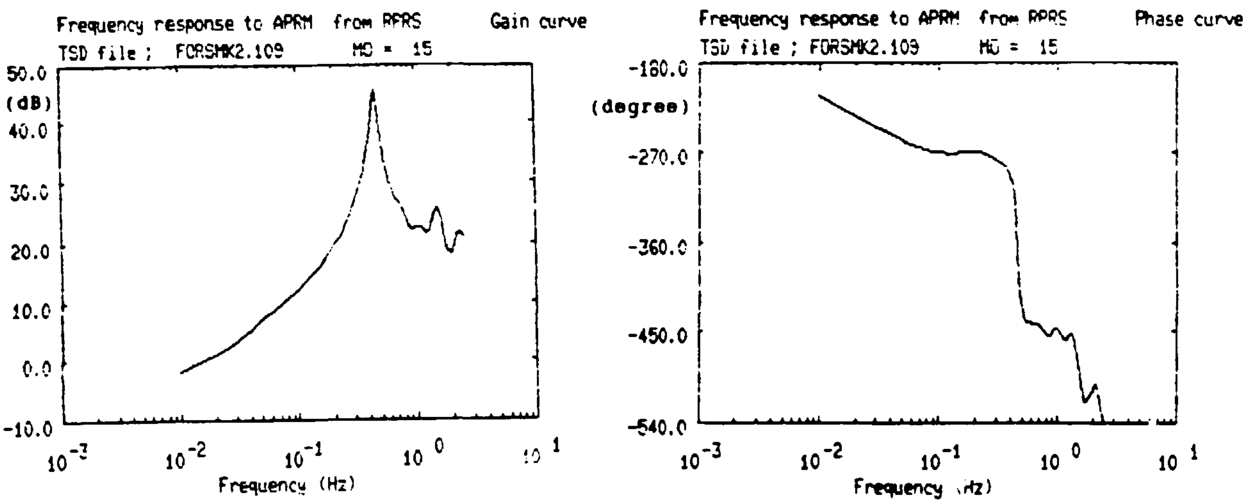


Fig. 3.3 Transfer function from pressure to neutron determined by the experiment with artificial random perturbations, showing gain (left) and phase (right) curves.

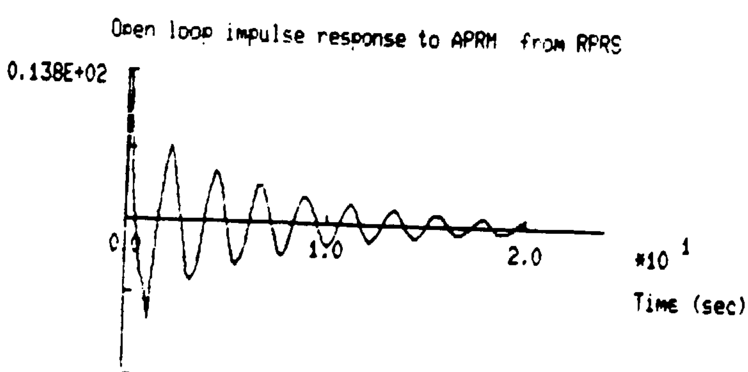


Fig. 3.4 Impulse response calculated in terms of the transfer function which was obtained by the experiment with artificial random perturbations.

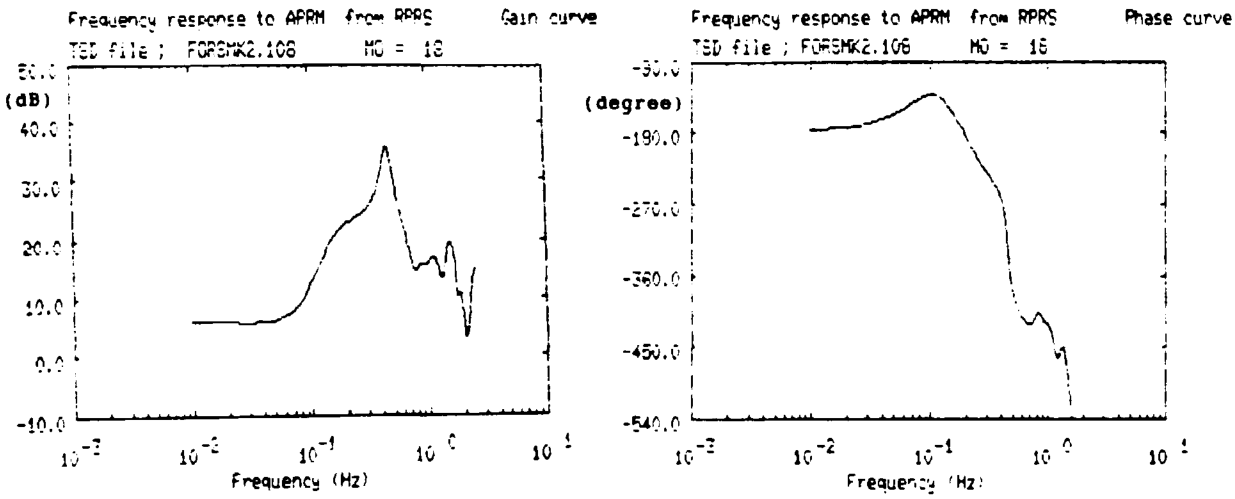


Fig. 3.5 Transfer function from pressure to neutron determined by the experiment with natural noise only, showing gain (left) and phase (right) curves.

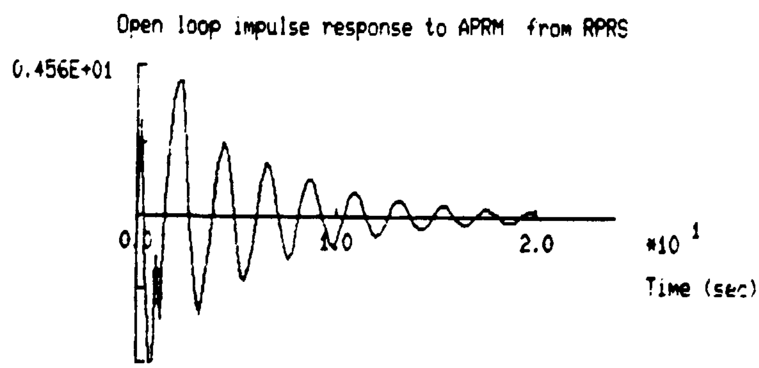


Fig. 3.6 Impulse response calculated in terms of the transfer function which was obtained by the experiment with natural noise only.

IV. Evaluation of The Reactor Stability by The Decay Ratio

The decay ratio (DR) is the most commonly used criterion to determine the BWR stability condition [10-12,3]. The DR is defined as the ratio of consecutive peaks in the impulse response, i.e.

$$DR = \frac{pk(k+2)}{pk(k)}, \qquad (4\text{-}1)$$

for k=1, 2, ...

There are two ways of obtaining the impulse response; one from the pressure to neutron transfer function, e.g. Eq. 3-1, and another from a univariate model identification for the APRM signal. In the former case, the impulse response is calculated by applying an impulse to Eq. 3-1. While in the latter case, the time behaviour of APRM signal is modeled using the so-called auto-regressive (AR) model and then the impulse response of the APRM is calculated by applying an impulse to the identified AR model.

Figure 4.1-(a) and -(b) show the impulse responses from pressure to neutron and that of APRM obtained by the AR method, respectivey, both for the same experimental data with random perturbations. The mean value of DRs is also shown in each figure. The DR by the AR method exhibits relatively close value to that by the pressure to neutron transfer function. However, it represents a slightly smaller and hence less conservative value.

Figure 4.1-(c) shows the impulse response and the DR obtained by the AR method for the natural noise only. The scatter of the DRs is much larger than the case of perturbation experiment but the mean value is quite similar to each other.

The DR evaluation for F-1 (using the APRM signal examined in Sec. II) is shown in Fig. 4.1-(d) for comparion. The scatter of the DR is less than the case for F-2. The mean value is much larger, thus indicating that F-1 is somewhat less stable than F-2. The larger scatter in the DR calculation for F-2 is probably due to more complicated spectral pattern in the APRM signal.

Summarizing the results obtained here, we can deduce the followings;

- The DR evaluation based on the AR method is an effective method to determine the reactor stability condition. However, it yields a slightly smaller value than by the pressure to neutron transfer function, and thus leading to a bit less conservative value,
- The DR estimates have larger scatter for F-2 than for F-1 because of more complicated spectral pattern in the APRM signal.

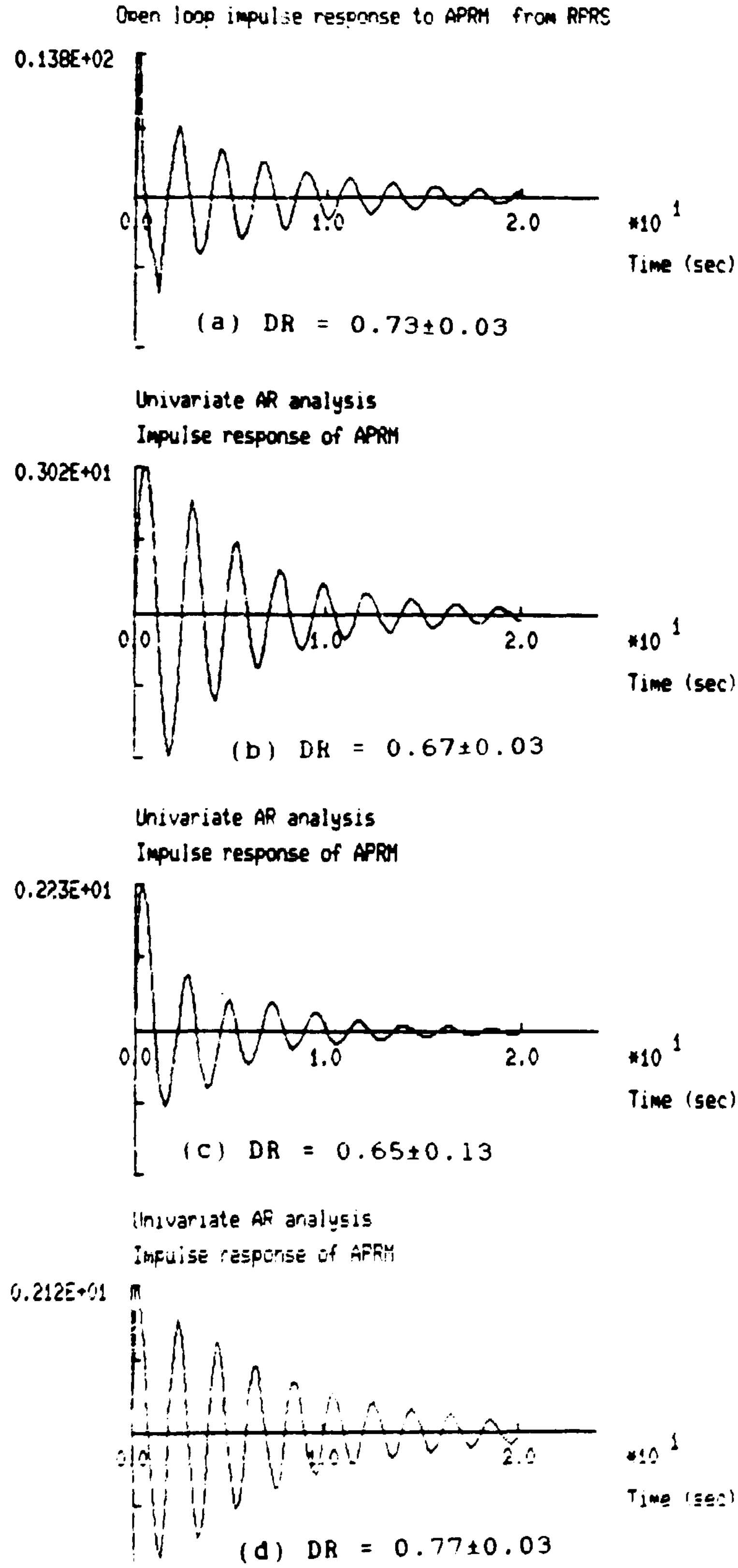


Fig. 4.1 Decay ratio calculation based on
(a) the pressure-to-neutron transfer function,
(b) APRM signal for the artificial perturbation experiment,
(c) APRM signal with natural noise for Forsmark-2, and
(d) APRM signal with natural noise for Forsmark-1.

V. On Interaction between Core and Plant Dynamics

Through investigation in the previous sections it bacame clear that F-2, like other BWRs, also has such a characteristic that causes a resonant power oscillation originated from neutron and thermal-hydraulics couplings with strong void reactivity feedback effect. This is therefore an inherent in-core phenomenon. However, this never means that the plant disturbances are of less significance. On the contrary, any kind of plant disturbance which may lead to reactivity variation causes more significant influence on the reactor operation.

In the course of noise measurements in F-2, a sudden change occured in the process behaviour when the reactor power reached 70%. Figure 5.1 shows the time behaviour of the APRM and RPRS signals before and after the onset of the oscillation together with their PSDs. Another very sharp and more pronounced peak appeared on the PSD of APRM signal at 0.33 Hz. The resonance is more remarkable on the pressure PSD.

An investigation was made using a method called signal transmission path (STP) analysis[13,14] for this set of data. To put it simply, the STP analysis is a multivariable noise analysis method for investigating mechanism of signal power propagation among variables under evaluation. It is pursued in such a way that we evaluate the rate of signal power influence from one variable (V1) to another (V2) and then compare it with the coherence function between the two. The value of coherence function is the degree of correlation between the two variables at that frequency. The theory tells that

- if there is one directional influence only from V1 to V2, then the magnitude of signal power influence from V1 to V2 agrees with the coherence at that frequency and the signal power influence from V2 to V1 becomes zero, (The similar relation holds true for the opposite direction of signal power transmission),
- if there is no correlation between the two variables, the signal power influence to both directions (from V1 to V2 and also from V2 to V1) becomes zero, and so becomes the coherence, too,
- if there is a feedback effect, the signal power influence exists to both directions and in addition their magnitude does not agree with the coherence.

Accordingly, the STP analysis yields the information about;

- which variable is the cause and which one is the result, i.e. cause-consequence relation,
- whether or not there is a feedback effect between the two variables, i.e. detection of feedback.

The method has been proved to be useful for identifying disturbance sources of various random signals.[15,16].

As shown in Fig. 5.2, the STP analysis for this measurement clearly indicates that the two PSD peaks in APRM are originated from different sources; 0.33 Hz peak from the pressure and 0.45 Hz peak from the APRM itself.

Because of this strong cyclic perturbation from the pressure to neutron, the neutron variation became about 1.27% in the standard deviation against the rated power, which is about 46% larger than before. Accordingly a concern arises about what would have happend if the pressure oscillation frequency had coincided with the core resonant frequency. We can make a rough estimation on this consequence with the aid of the pressure to neutron transfer function obtained in the previous section.

Comparing the gain at 0.33 Hz and that at 0.45 Hz in Fig. 3.3, the latter is about 6 times bigger than the former. This means that the neutron variation would have become 7.5% in the standard deviation (STD). According to the statistics theory, for a random signal with this STD there is a possibility of more than ±15% of power variation (2 times of the STD) with a probability of 5%. It can be well imagined that such a magnitude of power variation might cause a reactor scram.

It is worthwhile to note that in the light of the earlier publication[6] the above mentioned argument is not quite unrealistic. Namely the peak frequency of resonant power oscillation can shift depending on the reactor operational conditions and therefore may coincide with the pressure oscillation resonance.

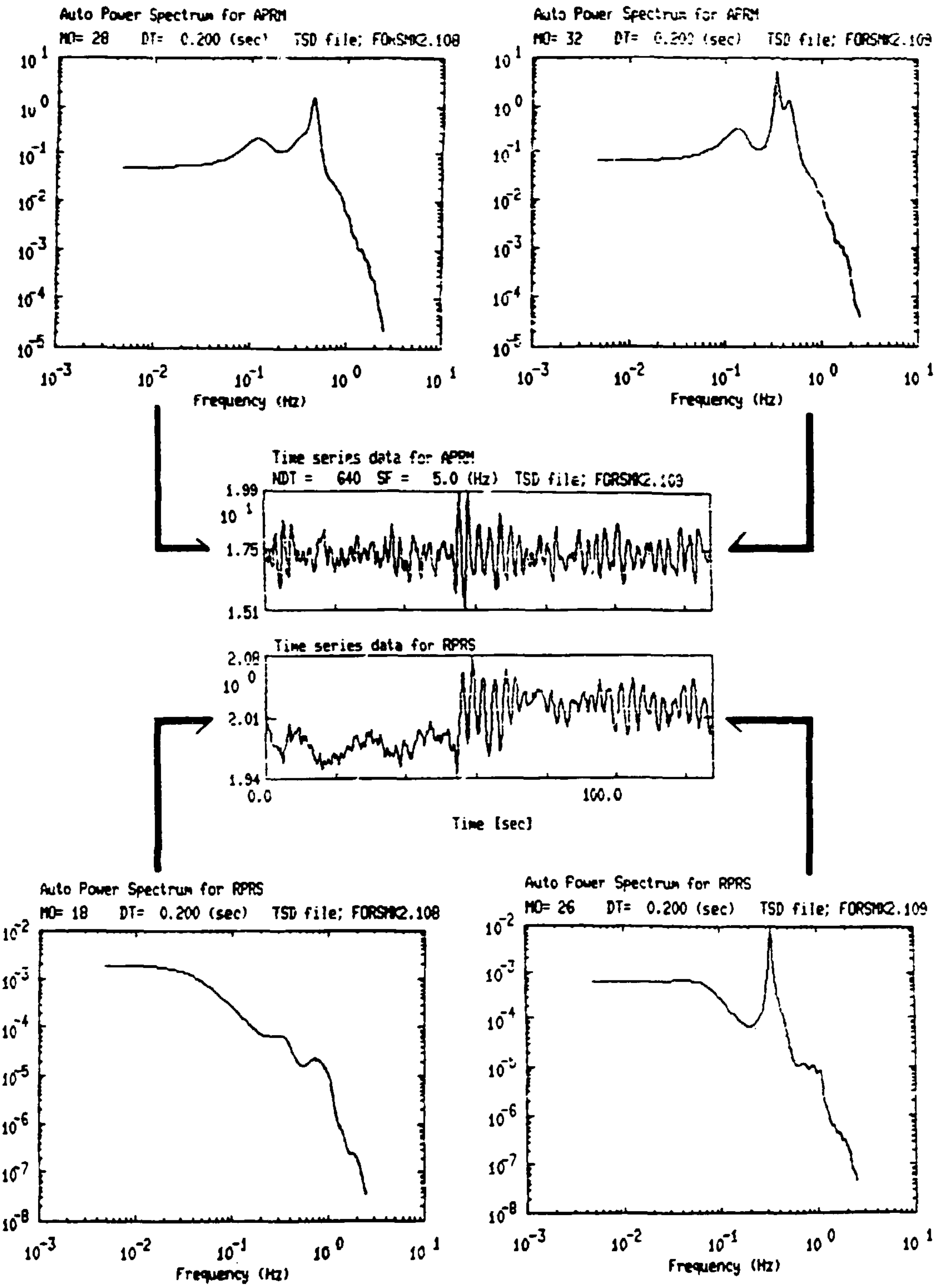
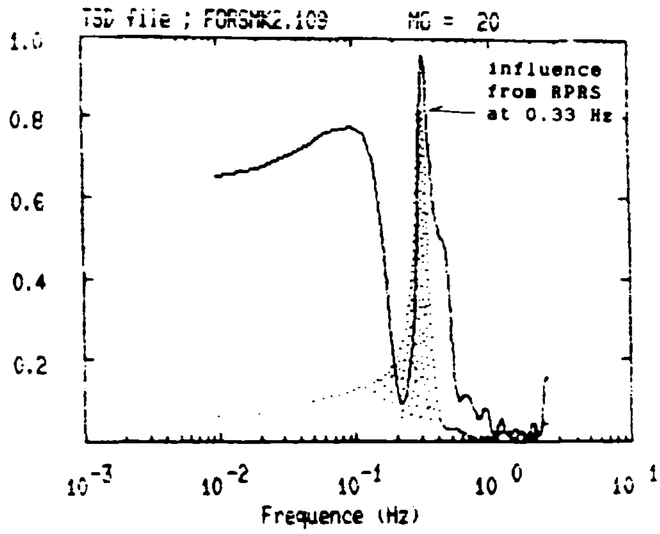
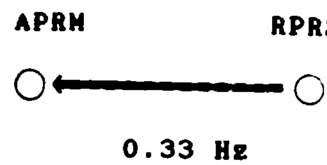


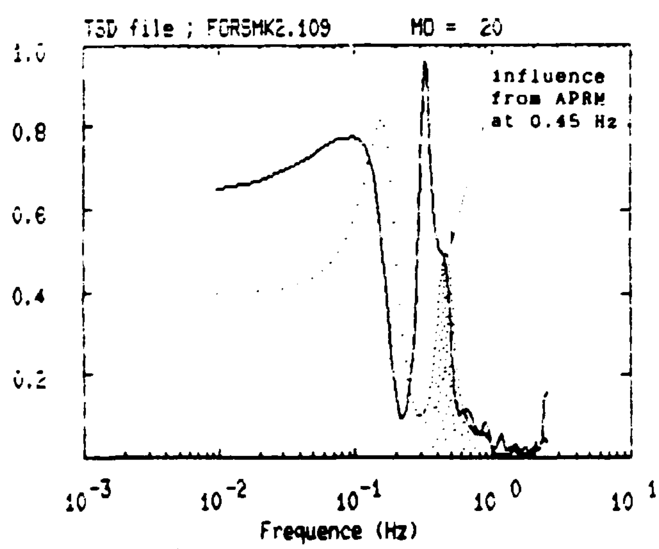
Fig. 5.1 Time series data of APRM and RPRS and their power spectra before and after the onset of signal condition change, showing an additional very strong oscillation in both APRM and RPRS at 0.33 Hz.

Power influence from RPRS to APRM





Power influence from APRM to RPRS



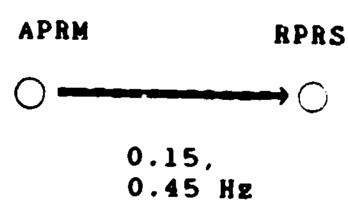


Fig. 5.2 Signal transmission path (STP) analysis for the APRM and RPRS signals, demonstrating strong influence from the pressure to neutron at 0.33 Hz and that from the neutron to pressure at 0.45 Hz. The real line in the figure represents the coherence between the two signals and the dotted line the degree of influence from one variable to another.

VI. Concluding Remarks

The F-2 reactor stability problem has been investigated based on noise analysis for signals collected during start-up operation after the regular outage in 1987.

1. It was revealed that the reactor has a strong resonant power oscillation with a frequency of 0.45 Hz at about 63% of the rated power and 4100 Kg/s of the core flow. The DR reached as high as about 0.7.

2. The transfer function measurement using random perturbations indicated that there is a strong resonant characteristic in the reactivity to neutron transfer function at 0.45 Hz. The STP analysis demonstrated that this resonant power oscillation is not induced by the reactor pressure nor core inlet flow variations but rather by the inherent mechanism inside the core. Therefore, it can be concluded that the primary mechanism of the resonant power oscillation in F-2 is the chain loop of dynamic interaction composed of the neutron kinetics, fuel heat transfer characteristics, coolant thermal-hydraulics, and void reactivity effect.

3. An investigation was made for the noise recording in which the pressure signal represented resonant oscillation. The result suggests that such pressure oscillation, in case that it occurred under the circumstance of high DR, has a potential of causing a forced power reduction or even a reactor scram if the resonant frequency of the neutron and that of pressure are coincided with each other. This must be seriously taken into account as a realistic scenario for a reactor scram.

The present investigation provided us with valuable information on the stability problem for F-2. However, it also disclosed a number of problems to be further studied. Some of those are considered to be very important in order to deepen our understanding of the reactor stability problem as well as to develop an effective method for the stability monitoring. Followings are the items to be investigated further by the noise analysis;

a) Investigation of two-phase flow characteristic and its interaction with neutron

The present study suggests that for deeper understanding of the BWR instability problem it is important to acquire more knowledge on the void generation and propagation processes and its interaction with neutron. In this respect, noise analysis of the space dependent neutron dynamics at near instability condition is very important. As seen in numerous earlier works[17-20], the noise analysis of in-core neutron sensor signals should provide additional information on the two-phase flow characteristic in the BWR. The followings are thought to be possible by the noise analysis;

- measurement of void transit time along the channel and its use to evaluate the axial void distribution,
- measurement of spacial correlation of the neutron,
- identification of spacially distributed perturbation sources inside the core, etc.

b) Diagnostic study of the plant control systems and their interaction with core dynamics.

As discussed in Sec. V, stability of the plant control systems is extremely important for the reactor stability especially when the reactor is near unstable. The present study suggests the significance of focused surveillance of the control systems by noise analysis during the reactor operation. There are examples of such systems in commercial power plants which have proved to be very useful to enhance the safe operation of the reactor[21,22].

In connection with the pressure control system, first of all, the cause of the resonant pressure oscillation which was observed at 70% power level must be investigated thoroughly. Having the high DR in F-2 in mind, one should, in principle, avoid to introduce such a mechanism as giving rise to control variable oscillation with resonant frequency between 0.1 and 1 Hz.

Secondly, it is desirable to have a diagnostic system which forcuses surveillance of the control systems by noise analysis during reactor operation. Early fault detection by noise analysis should be a benefitial measure to prevent abnormal reactor shut-down associated with the core stability problem.

Thirdly, it may be worthwhile to make a short survey of the past abnormal reactor shut-down events for both F-1 and F-2 in the light of the above mentioned scenario.

As to other control systems, it is desirable to make close examination about if there is a potential mechanism in the control system to induce such a resonant oscillation as in the above. It is also very useful to carry out a perturbation experiment to evaluate the sensitivity of the plant variable to neutron transfer fuction.

c) Evaluation of DR behaviour over one core cycle, and also a few core cycles in case of loading a new type of fuels with smaller diameter.

The DR may shift as a consequence of fuel burn-up, loading different type of fuel rods, especially those with smaller diameter, and so forth. Hence it is desirable to measure at least once the DR over a long period in order to follow its long term behaviour.

The significance of long term DR behaviour is rooted by the following facts;

- A pump trip at the reactor full power operation may cause the reactor to enter near instability region on the operational map. If the stability condition should change according to the fuel burn-up, the DR must be once evaluated over one core cycle in order to confirm the potantial stability condition of the reactor.

- There is a concern that the reactor stability margin decreases for smaller diameter fuel rods[23,24].

In the last, but not least, development of an on-line BWR stability monitoring system with diagnostic capability for the control systems, and its regular use would contribute to increased safety as well as economy of the reactor opration.

VII. Reference

1. B-G. Bergdahl et al., BWR-Stabilitetsundersokning i Forsmark 1, (1987).
2. T. Enomoto, et al., Recent Status and Future Perspective of BWR Stability, (in Japanese), Jour. of Japan Atomic Energy Soc. Vol. 27, No. 10, pp890-903, (1985).
3. J. March-Leuba, C. M. Smith, Development of an Automated Diagnostic System for Boiling Water Reactor Stability Measurements, Prog. Nucl. Energy, Vol. 15, pp27-35, (1985).
4. H. Akaike, On the Use of a Linear Model for the Identification of Feedback Systems, Ann. Inst. Statist. Math., Vol. 20, p425 (1968).
5. S. Kanemoto, et al., Multivariate Analyses for Neutron Noise Source Identification, Jour. Nucl. Sci. Technol. Vol. 20, No. 1, pp13-24, (1983).
6. J. H. Blomstrand, S. A. Andersson, Recording, Evaluation and Interpretation of Process Noise in Asea-Atom BWRs, Prog. Nucl. Energy, Vol. 15, pp461-471, (1985).
7. T. Mitsutake, et al., Core Stability Test Analysis Using The Autoregressive Model, Nucl. Technol., Vol. 65, pp365-373, (1984).
8. S. A. Sandos, S. F. Chen, Vermont Yankee Stability Test During Cycle 8, Trans. Am. Nucl. Soc., Vo. 45, pp754-755 (1983).
9. E. Gialdi, et al., Core Stability in Operating BWR; Operational Experience, Prog. Nucl. Energy, Vol. 15, pp447-459, (1985).
10. B.R. Upadhyaya, M. Kitamura, Monitoring BWR Stability Using Time Series Analysis of Neutron Noise, Trans. Ame. Nucl. Soc. Vol. 33, 342-343 (1979).
11. A.G. Federico, et al., Dynamic Characterization of The BWR Core/Channel for The Caorso Nuclear Power Plant with Applications to Stability Patterns Evaluation from Operating Data, Paper presented at SMORN-V, (1987).
12. S. Kanemoto, et al., Development of an On-line Reactor Stability Monitoring System in a Boiling Water Reactor, Paper presented at SMORN-V, (1987).
13. R. Oguma, A Method of Signal Transmission Path Analysis for Multivariate Random Processes, JAERI-M 84-084, (1984).
14. R. Oguma, E. Turkcan, Application of an Improved Multivariable Noise Analysis Method to Investigation of PWR Noise; Signal Transmission Path Analysis, Progress Nucl. Energ., Vol. 15, pp863-873, (1985).
15. G. Pole, et al., Comments on Practical Application of Autoregressive Signal Analysis to PWR NPP NoiseData, Progress Nucl. Energ., Vol. 15, pp897-902, (1985).
16. O. Glöckler, B.R. Upadhyaya, Results and Interpretation of Multivariate Autoregressive Analysis Applied to Loss-of Fluid-Test Reactor Process Noise Data, Paper presented at the Specialists Meeting on Reactor Noise (SMORN-V) Munchen, 12-16 Oct. (1987).
17. G. Kosaly, Noise Investigations in Boiling Water and Pressurized Water Reactors, Prog. Nucl. Energy, Vol. 5, pp145-199,(1980).

18. D. Lübesmayer, Experimental Reactor Noise - A Review on Noise-Analytic Measurements of Thermohydraulic Parameters in Operating BWRs and Their Interpretations, Prog. Nucl. Energ., Vol. 14, No. 1, pp41-93, (1984).
19. B.R. Upadhyaya, et al., Application of Noise Analysis Methods to Monitor Stability of Boiling Water Reactors, Prog. Nucl. Energ., Vol. 9, pp619-630, (1982).
20. J.A. Thie, Power Reactor Noise, American Nuclear Society, (1981)
21. T. Ando, et al., Operating Experience of a BWR Plant Diagnostic System, Prog. Nucl. Energy, Vo. 9, pp657-664, (1982).
22. T. Umeda, et al., Experience of On-Line Surveillance at Onagawa-1 BWR Plant, Paper presented at SMORN-V, (1987).
23. L. Goldstein, et al., Comparison of Advanced Fuel Designs to Current Standard Designs, 7-93, (1985).
24. D. Dayal, et al., Stability Tests at KWU Nuclear Power Plants, ppVII-48-56, (1987).