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WORKING MATERIAL

NUCLEAR POWER PLANT DIAGNOSTICS - SAFETY ASPECTS AND LICENSING

REPORT OF A TECHNICAL COMMITTEE MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN
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FOREWORD

It has been widely recognized that plant safety, reliability and performance can be enhanced through the appropriate monitoring of the electrical and mechanical components and systems. Monitoring and diagnostic systems have been applied to reactor vessel internals, pumps, safety and relief valves as well as to systems and components on the secondary side. The techniques applied include noise analysis, leakage, vibration and fatigue monitoring, core diagnostics, etc.

Monitoring and diagnostic systems can be considered as a type of operator support systems which provide more processed, integrated information than it is available from conventional instrumentation. They guide and advise operators and enable them to make better strategic decisions during both normal and abnormal operation. They also provide a capability for supervisory management during emergency conditions. More recently, expert system methods have also been introduced in order to improve the performance of such systems.

This meeting was organized in the framework of the International Task Force on Nuclear Power Plant Diagnostics, which has been jointly established in 1995 by the Division of Nuclear Installation Safety and the Division of Nuclear Power and Fuel Cycle of the IAEA. The terms of reference and scope of the task force cover both technological developments and safety/licensing aspects of diagnostics. In 1995, a Technical Committee Meeting on "Advances in Safety Related Diagnostics and Early Failure Detection Systems" has been organized in Vienna with emphasis on safety and licensing issues. In 1996, a Specialists Meeting on "Monitoring and Diagnosis Systems to Improve Nuclear Power Plant Reliability and Safety" was held in the United Kingdom, which placed emphasis on technical development of diagnostic systems.

INTRODUCTION

The aim of the Technical Committee Meeting (TCM) was to review developed systems and methods in diagnostics in the scope of their impacts and importance to the safety of Nuclear Power Plants (NPP). Papers presented on TCM came from different sources, from developers, from manufacturers, from licensing authorities and from NPP personal. They reflect up to date status in the given subject.

Participants of TCM formulated three working groups to elaborate different questions which were raised during the discussions. Their results are reflected in the three chapter titles of the given material. Annex 1 to this document contains presentations made at the Technical Committee Meeting.

1. SYSTEMATIC APPROACH FOR ADVANCED MONITORING: SUPERVISORY MONITORING

Monitoring of a power plant is one of the essential tasks during operation and the computer-based implementations are nowadays quite mature. However, presently these are still not satisfactory enough to meet the high standards of the licensing requirements and they are mostly not truly integrated into the plant's design-based monitoring system. This is basically due to the robustness problem as the majority of the methods are not robust enough for the monitoring of the safety parameter set in a plant or for intelligent supervision. Therefore an advanced, intelligent monitoring approach is necessary in a plant to supervise the monitoring tasks to be executed.

Such a monitoring is coined here as supervisory monitoring system (SMS) which determines the objectives to be obtained and finds the means to support and fulfil them. SMS deals with the changing plant status and the co-ordination of the information flow among the monitoring subunits. By means of these the robustness and consistency in monitoring is achieved.

The supervisory monitoring is performed through the accurate system model in multilevel form and it addresses higher level monitoring aspects. Modelling can be constituted by several components like static modelling, dynamic modelling and computational modelling.

Operating in real-time, the tasks of a SMS can be divided into two major categories as:

⇒ Fault detection and diagnosis which includes optimal state estimation

⇒ Model management which includes simulation and learning.

Fault detection and diagnosis performs the detection of incipient failures and causes of the failures. It should also report the failures to the operator. This pass of information should be done intelligently so as to help the operator to focus his/her on the current part of interest. Fault detection can be carried out in several ways, namely by processing received alarms, by model referenced process verification and by data and trend analyses. In each case the diagnosis has to be done as fast as possible to avoid the obscurity of the real cause of the fault. To achieve these, the sensory signals should be validated where analytical as well as physical signal redundancy should be provided.

Model management maintains and exploits a process model reflecting the current state of the process. Measurement values, trends, failures and structure changes are recorded in a data base so that topological and behaviourally correct process model is always available for the other tasks of the SMS such as prognosis. Hence, the model management acts as a data base as well as it identifies current and future states and trends of the plant and evaluates the model. This is achieved by model and parameter updating and learning.

In the supervisory monitoring system, knowledge-based expert systems play the essential role. For knowledge representation the computational effectiveness is required. Solutions to the problems in knowledge representation and inferences should satisfy the real-time constraints. Also, it is necessary to identify and formalise inference structures appropriate for dealing with incompleteness and uncertainty. An artificial intelligent (AI) system must be able of reasoning with incomplete and uncertain information.

The expert system outcome should be addressed to experts in the plant and a special effort has to be made so that it has to exclusively addresses to the plant operator as well.

To process the massive amount of low-level data from the plant two different approaches can be used. These are parallelism and hierarchy. For parallel data handling neural network for solving pattern recognition and minimisation problems is of particular interest. In the hierarchical approach the data structured as efficiently as possible in order to concentrate processing where it is needed. The two approaches are not mutually exclusive as the approaches are for the same problem.

For data manipulation one can distinguish between two issues which are trend analysis and uncertainty processing. Trends of process measurement are important source of information and advanced information processing methods should be used for exhaustive information extraction from the sensory data. Uncertainty can be handled by means of two different paradigms. These are analytical paradigms and rule-based paradigms. The analytical paradigm concerns the probability density of the certain quantity and the associated confidence levels. The rule-based paradigm concerns the fuzzy sets and the associated logic.

The verification and validation (V&V) of the supervisory monitoring system software should be done in a modular-wise and in two different stages. Verification is performed verifying the coherence and the functionality of the software with respect to the specification requirements and the design. The modular software systems should be verified altogether for the global functionality of the software.

The validation of the software should be done by another independent group of the knowledgeable people repeating the verification process in a challenging way so that one can gain the confidence for the functionality and the optimality of the approach. Verification and validation is schematically shown in the Figure 1.

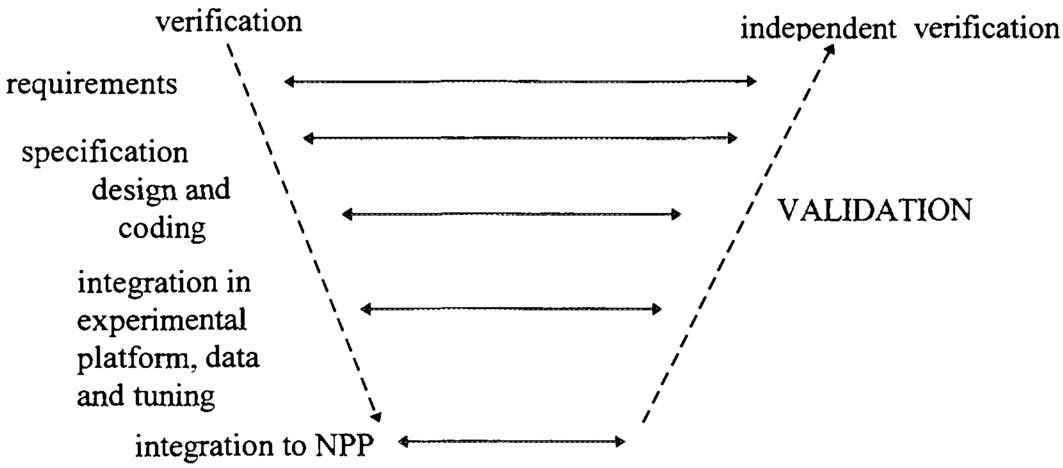

Fig.1. Schematic description of the verification and validation of the supervisory monitoring system software

The tuning of the SMS software has to be done by the people from the plant and by the people responsible for software.

Since the supervisory system outcome is exclusively addressed to the operator, taking an operator perspective is imperative while designing the human-machine interface for the supervisory monitoring system. It is clear that the complexity of the supervisory monitoring system should be hidden behind a simple, easy to understand interface. This implies that data should be presented conceptually, that is symbolically and linguistically, where it is applicable, exploiting the vast pattern recognition capabilities of human.

2. ISSUES ON MONITORING AND DIAGNOSTICS BASED PREDICTIVE MAINTENANCE FOR ENCHANCED NUCLEAR POWER PLANT SAFETY

Predictive maintenance is important for nuclear power plant (NPP) operation due to trend and early failure detection (EFD) and diagnostic analysis [1,2] for eliminating the faults in the incipient failure conditions. Predictive maintenance can be defined as follows: it is the corrective activity in case of a degrading trend of a component as well as monitoring of the operational status of that component subject to maintenance. This activity is directly related to the component maintenance procedures and also to the safety issues, in the case of monitoring of a critical, i.e. safety related component.

Critical safety related monitoring activities involve:

- core monitoring
- vibration monitoring of components and machinery
- loose parts monitoring
- leakage monitoring
- monitoring of motor operated valves.

Such a predictive maintenance directed monitoring system is shown in Fig.2., where supervisory monitoring is an intelligent monitoring system providing robust and consistent information.

It is essential to select components or systems to be monitored with certain priorities. The selection of components to be monitored comes from the need to improve safety and reliability, to increase the availability and to improve the maintenance. Analysing plant vulnerabilities with different methodologies, information to decide which type of monitoring system to be implemented should be available. The following is a list of sources that can be used to select components with major impacts on safety:

- Operational experience
- Deterministic safety assessment
- Probabilistic Safety Assessment (PSA)
- Surveillance and In-Service Inspection (ISI) program
- Plant Life Extension (PLEX)
- ALARA principles

Diagnostic activities are the integral part of the predictive maintenance where the failure type and its location with the cause are identified.

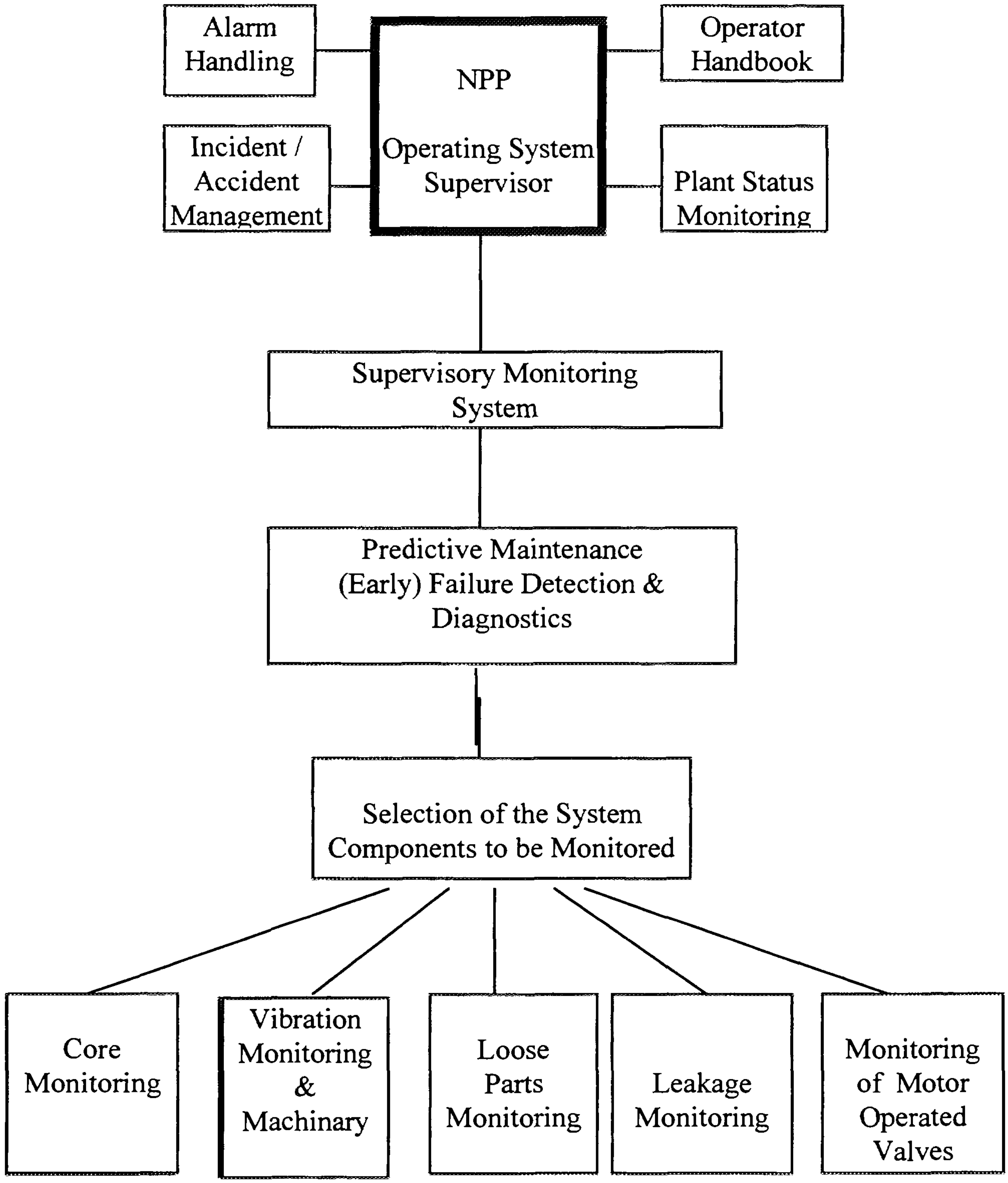

Fig.2. Monitoring scheme for predictive maintenance of safety related power plant components

The stream of monitoring and diagnostic activities for predictive maintenance are schematically shown in Fig.3. In this figure process monitoring is used for trend analysis and in the case of a trend formation, a root-cause relationship is established to take necessary steps for the elimination of the trend. For diagnostic modelling of the power plant, this is an essential tool since it is not practical to examine the performance of a power plant by other means.

As the monitoring implies real-time operation dynamic modelling becomes imperative.

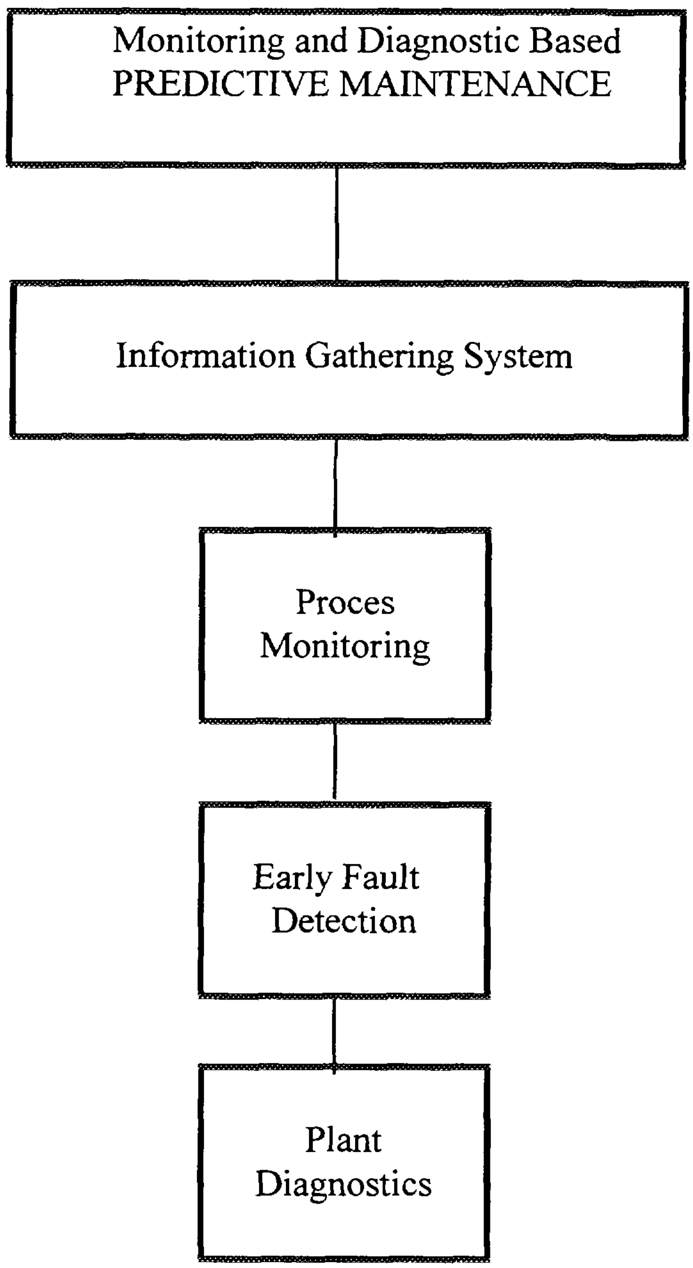

Fig.3. Monitoring and diagnostics process fro predictive maintenance

The EFD systems enhance the safety and the reliability of the NPP and it should focus on performance problems coming from NPPs and from Nuclear Regulatory Bodies (NRB). Further, on-line monitoring activities can be extended with

- sensor surveillance (response time monitoring, etc.)
- fatigue monitoring of the components
- rotating machinery diagnostics (bearing degradation, shaft crack detection etc.)
- critical safety parameter (CSP) monitoring (for example, subcooling margin determination)
- chemical monitoring

3. SPECIFIC SYSTEMS AND METHODS BASED ON ACOUSTIC SIGNAL MONITORING FOR NPP

Scope

General

This chapter analyses specific systems, methods and applications based on acoustic signal monitoring for NPP. Within the scope of the chapter are:

- methods using microphones, accelerometers or ultrasonic transducers
- analysis that belong to acoustics
- methods especially developed or applied in NPP's

Vibration measurements and analysis as well as methods for non-destructive material testing are out of the scope of this chapter.

Systems and their tasks

Loose parts monitoring

Loose or loosened parts in the reactor cooling system of nuclear installation may cause severe damage of the primary circuit components and reactor pressure vessel internals.

The tasks of loose parts monitoring systems are:

- detection of loose parts,
- localisation of the detected loose parts and
- estimation of the mass.

Leak detection

The tasks of leak detection systems are:

- automatic detection and localisation of leakage's from cracks through the wall of large pipes in the reactor cooling system (main coolant line, surge line, lines of the emergency core cooling system, etc.),
- automatic detection and localisation of leakage's through flanges and penetrations at the NPP main components (reactor vessel top, steam generator, main coolant pump, pressurizer, etc.),
- automatic detection and localisation of inner leakage's through closed valves (i.e. pressurizer safety valve).

In addition, the leakage rate has to be estimated.

The LBB (leak before break) requirements concerning leak detection systems have to be fulfilled.

Acoustic emission

The task of the acoustic emission methods are the detection of cracks and defect developing processes in metals during hydraulic tests or operation.

Valve diagnostics

The task of acoustic methods at valves is the estimation of the mechanical condition of the valves.

Boiling detection

The task is the detection of boiling occurrence in nuclear reactor core or in fuel channels of channel type reactors. Additional the steam-water-ratio has to be estimated.

Liquid level estimation in vessels

The task is the determination of the liquid level in vessels of nuclear power plants (e.g. storage of liquid waste, boiling water reactor vessels, etc.)

Statement about the achievements

System / Application	Method	Sensor (type, number, freq. range, environm. condition)	Sensitivity	Accuracy of localisation	Reactor types for application	Status of development	Other and additional methods for the same task
Loose parts monitoring	Detection of bursts in structure-borne noise resulting from loose parts impacting with walls or internals of the reactor coolant system. Comparison of the signal level with absolute and/or relative thresholds.	accelerometers mounted on the outer surface of RCS 10...24 sensors 1 to 10 (-20) kHz up to 350°C	loose parts with mass between 0,1 kg and 20 kg	1 m	PWR VVER	full developed and standard applied system	none
Leak detection (metal born acoustic signal)	Detection of the noise produced by water or steam escaping through a crack.	ultrasonic pickup's mounted on the outer surface of RCS 10 ... 50 sensors 100 to 400 kHz up to 350°C humidity up to 100%	better than 230 kg/h (achievement about 10 kg/h)	1 ... 2 m (achievement is about 1 cm)	VVER RBMK (other types possible)	full developed and standard applied system (for RBMK: experimental system)	humidity measurement, water level measurement, detection of radioactive isotopes
Leak detection (air born acoustic signal)	Analysis of the air born noise produced by water or steam escaping through a crack.	high temperature microphones installed in the compartments of the to be monitored NPP components about 40 sensors (In case of beam microphones: about 10 sensors) 8 to 30 kHz up 300°C humidity up to 100%	better than 230 kg/h (achievement about 25 kg/h)	about 1 m	RBMK (other types possible)	experimental system	humidity measurement, water level measurement, detection of radioactive isotopes

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System / Application	Method	Sensor (type, number, freq. range, environm. condition)	Sensitivity	Accuracy of localisation	Reactor types for application	Status of development	Other and additional methods for the same task
Acoustic emission	Analysis of metal born burst acoustic signals of cracking processes inside the material	ultrasonic sensors mounted on the outer surface of the to be monitored NPP component 500 kHz ... 1 MHz		about 10 cm		laboratory equipment	non-destructive flaw detection methods (ultrasonic, eddy current, etc.)
Valve diagnostics	Analysis of metal born or air born acoustic noise generated by the valve equipment during operation	piezosensors 1 per valve 20 ... 100 kHz beam microphones 1 per room 5 ... 50 kHz	determination of the real valve condition determination of the real valve condition		RBMK VVER RBMK VVER	experimental system under development	systems using other techniques: vibration, current, torque, magnetic and pressure signal
Boiling detection	Analysis of water born noise generated by boiling process	piezosensors	?	?		laboratory equipment	temperature control
Liquid level estimation in vessels	Analysis of ultrasonic signal reflection from the liquid surface (active method)	piezosensors ultrasonic signal generator 1 per vessel	about 1 cm	about 1 cm	RBMK (other types possible)	full developed and standard applied system	standard I&C systems

References / Examples

Loose parts monitoring

Standard application in pressurised water reactors in most of the countries. Typical systems installed in the PWR's in Belgium, Czech Republic, France, Germany, Slovakia, Spain, Switzerland, USA, etc.

Leak detection

Standard application for leakage monitoring at VVER reactors in Bulgaria, Czech Republic, Hungary, Russia, Slovakia.

At RBMK reactors the Leningrad NPP (Russia) is equipped with an experimental system.

Acoustic emission

Because of difficulties to confirm the achieved results the method is nowadays seldom used.

Valve diagnostics

Systems installed at Kalinin NPP (Russia) in an experimental stage, at Balakovo NPP a laboratory equipment is installed.

Liquid level estimation in vessels

Systems used for liquid waste level control of storage vessels at Chernobyl NPP (Ukraine).

Recommendation

Loose parts monitoring

International standards (KTA, DIN, IEC, ASME) requires or recommend loose parts monitoring of NPP's. The occurrence of loose parts is a common problem in pressurised water reactors and not specific for certain PWR design.

The implementation of loose parts monitoring systems is highly recommended for PWR for use during NPP operation and even during start up of the reactor.

A guide compiled from the available codes would be useful for the implementation and operation of loose parts monitoring systems.

Leak detection

The development of acoustic leak detection systems is in a stage that a wide use in different reactor types has been done. Special codes or standards are not existing for this method up to now. The general standard for leak detection systems is IEC 1250.

One essential application is to fulfil the requirements for leak detection systems for the application of the LBB criterion (leak before break).

It is recommended to compile guidelines for acoustic leak detection systems to get standard requirements for manufacturing, mounting and operation of this type of leak detection systems.

Valve diagnostics

It is recommended to estimate the state of the art of the existing different valve diagnostic systems e.g. by organising international benchmark tests for valve diagnostic systems.

Boiling detection

Because the method is still under development it is recommended to improve the reliability of that method. The achievements of this method in different countries should be observed, compiled and kept updated.

Liquid level estimation in vessels

It is recommended to test other application of this method for liquid level measurements in other reactor installations. The achievements of this method should be observed and kept updated.

REFERENCES

[1] Advances in Safety Related Diagnostics and Early Failure Detection Systems, IAEA-J4-TC698, Vienna, November 20-24, 1995

[2] Monitoring and Diagnosis Systems to Improve NPP Reliability and Safety, Proc. of the Specialists' Meeting jointly organised by IAEA and Nuclear Electric Ltd, Gloucester, UK, May 14-17, 1996

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ANNEX I:
PAPER PRESENTATIONS

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EXPERIENCE WITH DIAGNOSTICS AT ANGRA I

F.R. FEINER

1. INTRODUCTION

Angra I NPP was ordered in the early seventies as a turn key contract from Westinghouse. It is a PWR, two loops, 657 MWe capacity project, and was the first Brazilian initiative on the nuclear electric field.

As a large tropical country, we benefit from a number of rivers and proper geography, such that our electric matrix rely on hydroplants as much as 95%, or more. Typically we have a hydro-electric culture in our electricity industry.

Without a nuclear culture, and being a hydro system, we had to rely on our contractors on every aspect of the project. We ordered the plant, and receive its hardware and software as they were twenty years ago, before TMI and Chernobyl. The system was typically an operation environment. Focus on ops was the major aspect, with some attention to Reactor Engineer, Health Physics and Chemistry. Training was toward ops needs, only.

In such an environment, the Maintenance needs were secondary issues, with a lack of resources, either material or human. Items such as work control system, training, types of maintenance (corrective or preventive), equipment history, procedures, maintenance operational experience (inhouse/outside), maintenance equipment and so on, were not among those considered as plant priorities. As a result, the maintenance performance was extremely low, and the overall plant performance was a great disappointment.

TMI accident, and the related results of event evaluation, brought additional impacts to our plant, since a large number of modification, either on the systems and procedures, were required. Again these changes did not addressed maintenance and its role as an important contributor for plant safety, availability and cost effectiveness. TMI improvements focused mainly on ops and safety back up systems to avoid, isolate or mitigate an event. These improvements, although very beneficial for the industry, raised the overall cost of the plant. With a single unit, inside a hydro environment, with low capacity factors, the organization struggles to improve plant performance addressing other issues, but not maintenance.

By the end of the eighties the low plant performance enhanced public and corporation disappointment with the nuclear industry in Brazil, since we were at the lower quarter among nuclear plants worldwide.

As the first quarter of the ninites has passed, the plant had accomplished most of TMI items and started to turn its attention to maintenance. Since the first core load Angra I has performed as a very safe plant. But just now, as we focus on maintenance and its role, the plant is raising its availability indicators and the early failure detection techniques emerges as one of the key factors in this recovery.

2. PREDICTIVE MAINTENANCE PROGRAM at Angra I NPP

The purpose of the Predictive Maintenance Program at Angra I NPP is to enhance plant safety and reliability though early detection and diagnosis of equipment degradation prior to equipment failure.

The accomplishment of this role rely on an organization inside the Maintenance Division. This group will perform this function through use of installed and portable diagnostic tools which will monitor selected equipment parameters to detect degradation and monitor equipment condition. It will identify which equipment is to be trended, select parameters to be monitored, establish base lines, set Alert and Action Points for each parameter, and coordinate predictive maintenance activities with other plant organizations. In addition, this group will establish and maintain the data base needed to collect and manipulate the data collected under this program and will issue periodic reports for management.

The Predictive Maintenance Program does not override the Surveillance Program, required by the Brazilian Regulatory Body in accordance with ASME Section XI requirements, but it provides detailed component evaluation and condition assessment, which adds value to the Surveillance Program.

The Maintenance Division Manager has the overall responsibility for the program. The Predictive Maintenance Group Supervisor is responsible for the development of a PREDICTIVE MAINTENANCE MASTER LIST, for its periodic review, to ensure that results of the Predictive Maintenance data collection activities are properly stored for trending purposes and future use, to ensure that data is reviewed, trended and analyzed to detect degradation of equipment condition, and for changes recommendations to the Predictive or Corrective Maintenance activities, based on the Predictive maintenance results. In addition he is responsible for providing timely notification to the Work Control System Coordinator of equipment whose condition is deemed to be deteriorating and in need of further diagnostic activities or corrective maintenance.

The Predictive Maintenance Master List plays a key role in the program. It shall be periodically reviewed (annual basis), must identify the frequency at which each component is monitored, shall contain equipment/components that are not monitored on a routine basis, and must be revised as dictated by experience, cost effectiveness and maintenance history.

The selection of equipment/component will be based on safe and reliable plant operation, equipment experiencing repeated Corrective Maintenance, ALARA requirements and equipment that justify a "run to failure" approach.
Nuclear safety-related equipment is top priority, followed by load threatening equipment without and with spare. Support equipment and high-maintenance items are the following priorities.

In particular, as they are an important source of problems for safety and reliability of the plant, as identified by the industry, Check Valves and Motor Operated Valves shall receive special focus inside the Predictive Maintenance Program.

The Predictive Maintenance Group will use a variety of techniques in implementing the program, such as:

- vibration monitoring,
- lubricant analysis (viscosity, moisture, and other contaminants, ferrography, grease analysis specially for MOV),
- infrared thermography,
- motor operated valve diagnostic information,
- acoustic monitoring,
- bearing temperatures monitoring,
- in-leakage detection,
- insulation resistance monitoring,
- eddy current testing,
- temperature differential monitoring of heat exchangers,
- polarization index.

Special consideration shall be made in selecting and training the personnel responsible for obtaining and analyzing predictive maintenance data. A training program shall be implemented to support needed competency and to upgrade knowledge as technology enhances.

Equipment not in operation shall not be placed in operation for the sole purpose of predictive maintenance monitoring. Such monitoring shall be rescheduled to a time consistent with normal plant and equipment operations. In particular, standby safety-related equipment shall not be made unavailable solely to perform predictive maintenance.

Latest acquired data shall be compared with previous data to detect any degradation. If degradation is observed that indicates the integrity of equipment may be endangered, the Predictive Maintenance Group shall promptly notify the Work Control Center Coordinator as well as Ops Division, and may take the following actions: recollect the data to verify its validity, increase the frequency at which data is collected, perform any additional types of diagnostic testing to determine the extent and cause of the degradation, or schedule corrective maintenance.

Failure of equipment included in the Predictive Maintenance Program but not predicted must have detailed root cause investigation to determine why the program did not detected degradation before the failure occurred.

3. ANGRA I PREDICTIVE MAINTENANCE MASTER LIST

Angra I Predictive Maintenance Master List addresses the following equipments:

- Compressors
- Emergency Diesel's
- Rod Driver MG's
- Main Generator
- Main Exciter
- Turbines
- Transformers
- Breakers
- Electrical buses
- Batteries
- Switchgears
- Safety screen washes
- Fans
- Motors
- MOV's
- Pumps
- Oil reservoirs
- Traps
- Valves

Frequencies and types of techniques are included in the Predictive Maintenance Master List.

4. MAINTENANCE EVOLUTION AT ANGRA I NPP

The Angra I maintenance program has incorporated several different philosophies since plant startup. Initially, a Preventive Program was developed based on vendor manuals, which resulted in a very heavy list of PM tasks. That program didn't assess plant mode of operation, and was basically a list of tasks to be performed from time to time, no matter what the plant status was. Since plant training program has not assessed maintenance needs, this heavy program let to very frequent interventions on the machinery, with a high rate of malfunctions due to human performance weakness.

A different approach was introduced as maintenance personnel realized they were spending a high amount of resources with low output, and after some mistakes, which damaged major equipments. Basically we changed the plant preventive maintenance using sound engineering judgment. As a result we felt the maintenance resources were optimized. Indeed, some failures with high impact on availability were still occurring.

At this time we realized that Preventive Maintenance based on component replacement between fixed intervals was not adding value to our program due to the aleatory characteristic of the rotating machinery failure. Moreover, the replacement often resulted in equipment malfunction due to human failure.

As a result of experience exchange with foreign organizations, such as INPO, we enhanced our PM program with operational experience on maintenance area. With this evolutions we benefit from industry development in selected components and systems, like MOV's and check valves. We also learned from the benefits of new technology, what let us to introduce the Predictive Maintenance concept to our plant.

At Angra I NPP we believe the equipment in general is much more reliable than we thought before, and whenever a fail process initiates the equipment asks for help. The whole issue is to be vigilant, understand its language, and deliver the proper care at the right time. Again the COMMUNICATION PROCESS shows its importance in nuclear industry, just like it was established in INSAG-4 IAEA DOCUMENT (a questioning attitude + a rigorous and prudent approach + COMMUNICATION).

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ANGRA I NUCLEAR POWER PLANT

MOVs EVALUATION PROGRAM

1. ANGRA I MOVs SUMMARY

MOST OF THE SAFETY RELATED VALVES REQUIRED FOR SAFETY SHUTDOWN AT ANGRA I ARE MOTOR OPERATED VALVES.

TOTAL OF SAFETY RELATED MOVs:
108 MOVs

SAFETY FUNCTIONS OF THESE VALVES:

- COMPONENTS ISOLATION
- SYSTEMS ISOLATION
- CONTAINMENT ISOLATION
- PRESERVATION OF RCS PRESSURE BOUNDARY.

ALL THE ANGRA I SAFETY RELATED MOVs ACTUATORS WERE MANUFACTURED BY LIMITORQUE AND INCLUDE THE FOLLOWING MODELS:

SB-0:	04 ACTUATORS
SB-00:	41 ACTUATORS
SB-1:	11 ACTUATORS
SMB-00:	29 ACTUATORS
SMB-000:	21 ACTUATORS
SB3:	02 ACTUATORS

SAFETY RELATED MOTOR OPERATED VALVES TYPES AND MANUFACTURERS:

GATE WESTINGHOUSE:	**37 VALV.**
GLOBE VELAN:	**07 VALV.**
GATE VELAN:	**01 VALV.**
GLOBE COPES VULCAN:	**02 VALV.**
GATE COPES VULCAN:	**31 VALV.**
BUTTERFLY COPES VULCAN:	**01 VALV.**
BUTTERFLY FISHER:	**18 VALV.**
GATE ITT GRINNEL:	**02 VALV.**
BUTTERFLY JAMESBURY:	**06 VALV.**
BUTTERFLY POSI:	**02 VALV.**
GLOBE GIMPEL:	**01 VALV.**

2. INITIAL PROGRAM

IN ORDER TO COMPLY WITH THE REQUIREMENTS SETFORTH IN THE US NUCLEAR REGULATORY COMMISSION GENERIC LETTER 89-10, FURNAS HAS DEVELOPED, BY ITS OWN INITIATIVE, A PROGRAM FOR EVALUATION AND TEST OF ALL THE SAFETY RELATED MOVs INSTALLED AT ANGRA I.

INITIALLY FURNAS CONTRACTED ITI MOVATS TO PERFORM THE EVALUATION AND TEST PROGRAM IN 20 MOVs CONSIDERED PRIORITARY.

THE ENGINEERING EVALUATION OF THIS FIRST PART OF THE PROGRAM STARTED IN JANUARY 1996 AND THE TESTS WERE PERFORMED DURING PLANT P-6 OUTAGE, IN APRIL AND MAY 1996.

THE INITIAL PROGRAM INCLUDED A COMPLETE INSPECTION AND MAINTENANCE OF ALL THE MOVs WHICH WOULD BE EVALUATED AND TESTED. THESE ACTIVITIES WERE PERFORMED CONSIDERING ALL THE RECOMMENDATIONS AND PROCEDURES OF THE MANUFACTURERS.

AS A RESULT OF THIS INITIAL EVALUATION, SEVERAL PROBLEMS WERE IDENTIFIED AS DESCRIBED BELLOW:

- 40% OF THE ACTUATORS HAD DEGRADED GREASE.

- 30% OF THE ACTUATORS HAD NON QUALIFIED TORQUE AND LIMIT SWITCHES.

- 10% OF THE ACTUATORS HAD MOTOR PINION INSTALLED IN AN INVERTED POSITION.

- 10% OF THE ACTUATORS WERE FOUND WITH THEIR MOTOR PINION KEY ALMOST SPLITTED.

- 60% OF THE ACTUATORS HAD RELAXED SPRING PACKS.

- 10% OF THE ACTUATORS SHOWN THE WORM SHAFT BEARING DAMAGED.

- 60% OF THE MOVs HAD THEIR TORQUE SWITCHES SET POINTS READJUSTED.

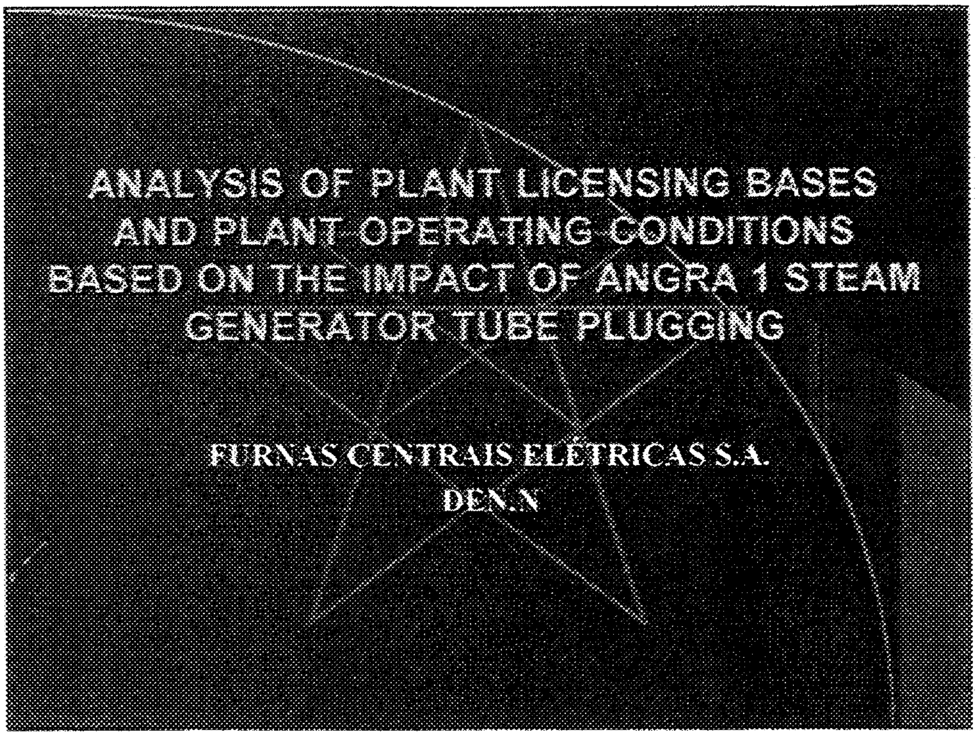
- THE AUXILIARY FEEDWATER PUMPS ISOLATION VALVES PV-1527 AND PV-1528 WERE FOUND QUITE DAMAGED.
THE ENGINEERING EVALUATION CONCLUDED THAT BOTH THE VALVES AND ACTUATORS WERE UNDERSIZED.

3. PROGRAM FOR THE P-8 AND P-9 OUTAGES

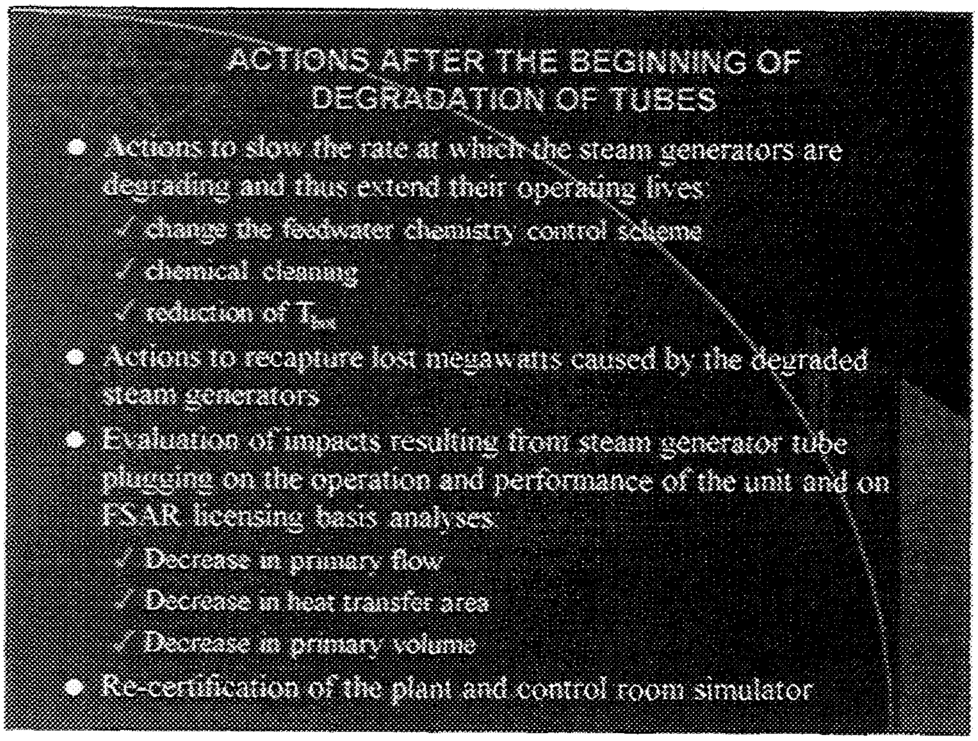
FOR THE NEXT OUTAGE (P-8) , FURNAS WILL CONTRACT SERVICES TO EXTEND THE NRC GENERIC LETTER 89-10 PROGRAM TO 50 MOVs.

THE TESTS PERFORMED AT P-7 OUTAGE AND THE TESTS THAT WILL BE PERFORMED IN THE P-8 OUTAGE, DON'T CONSIDER DYNAMIC CONDITIONS.

DYNAMIC TESTS ARE SCHEDULED FOR THE P-9 OUTAGE, WHEN FURNAS INTEND TO HAVE COMPLETED THE ENGINEERING EVALUATION FOR ALL THE VALVES, AND THE GROUPING STUDY.



This project under development in Furnas Centrais Elétricas S.A. analyse impacts on licensing basis and operating conditions of the increase in Angra 1 steam generator tube plugging levels.



• Experience has shown that once steam generator tubes begin to degrade, there is no way to stop the process. Furthermore, if no action is taken, the rate of degradation will accelerate. Therefore most plant owners have undertaken some positive actions to slow the rate.

• As the tubes degrade, the amount of heat transferred to the secondary system decreases. Initially, the plant operators can compensate by opening the turbine control valves wider. Eventually these valves will be fully open and any additional loss in steam pressure will result in the plant no longer being able to generate 100% electrical power. These lost megawatts can be recaptured by making modifications to the secondary side to improve the efficiency of feedwater/steam cycle.

• The evaluation of the thermal-hydraulic impacts on plant operation, performance and licensing basis fall mainly into three areas:

• Decrease in primary flow

Primary circuit flow will decrease following any modification that increases the loop hydraulic losses.

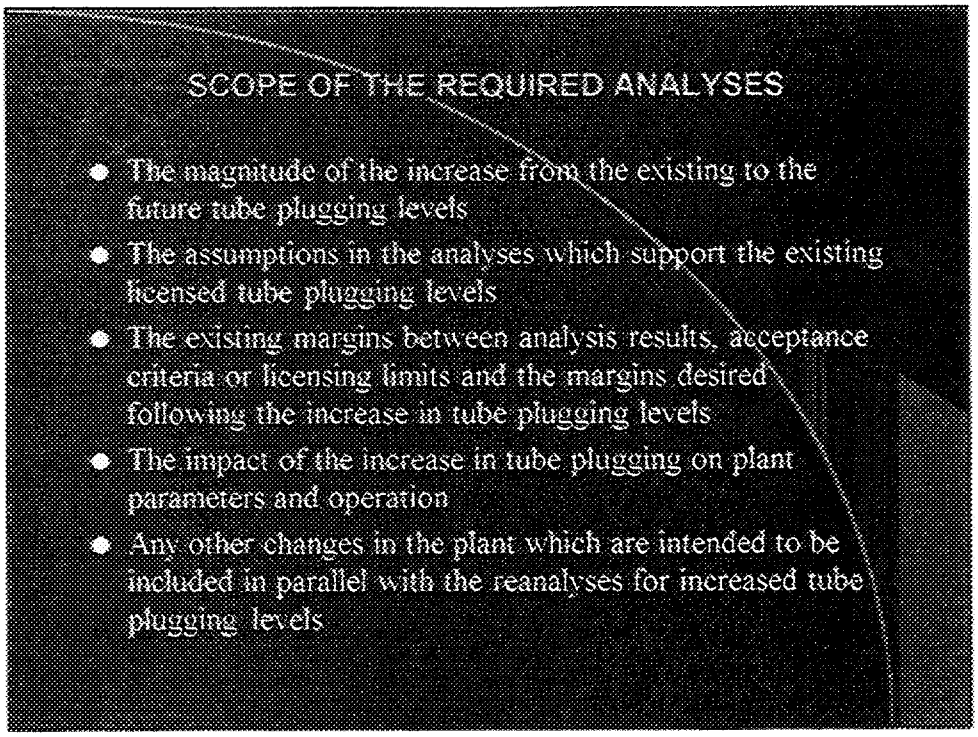
• Decrease in heat transfer area

As steam generator tubes are removed from service by plugging, there is less active heat transfer area.

• Decrease in Primary Volume

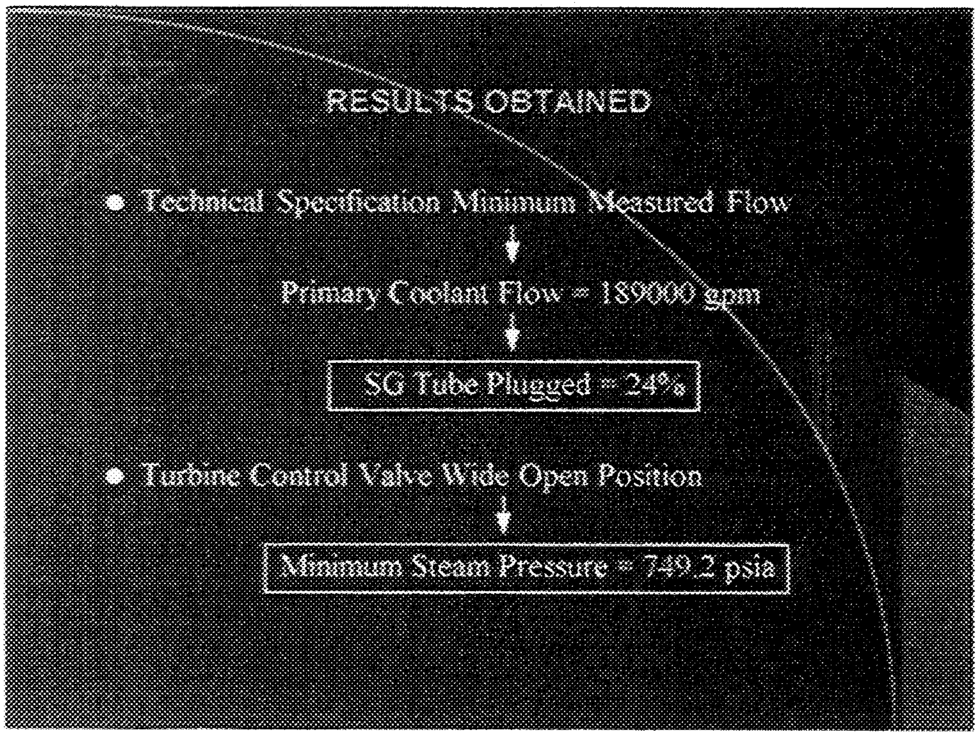
Although not as significant as the two above impacts resulting from tube plugging, there is some potencial for a smaller primary volume to contribute to a more adverse transient response.

• As the plant is modified and as new analytical models are developed, the plant simulator will need to be checked to verify that it properly represents plant performance.



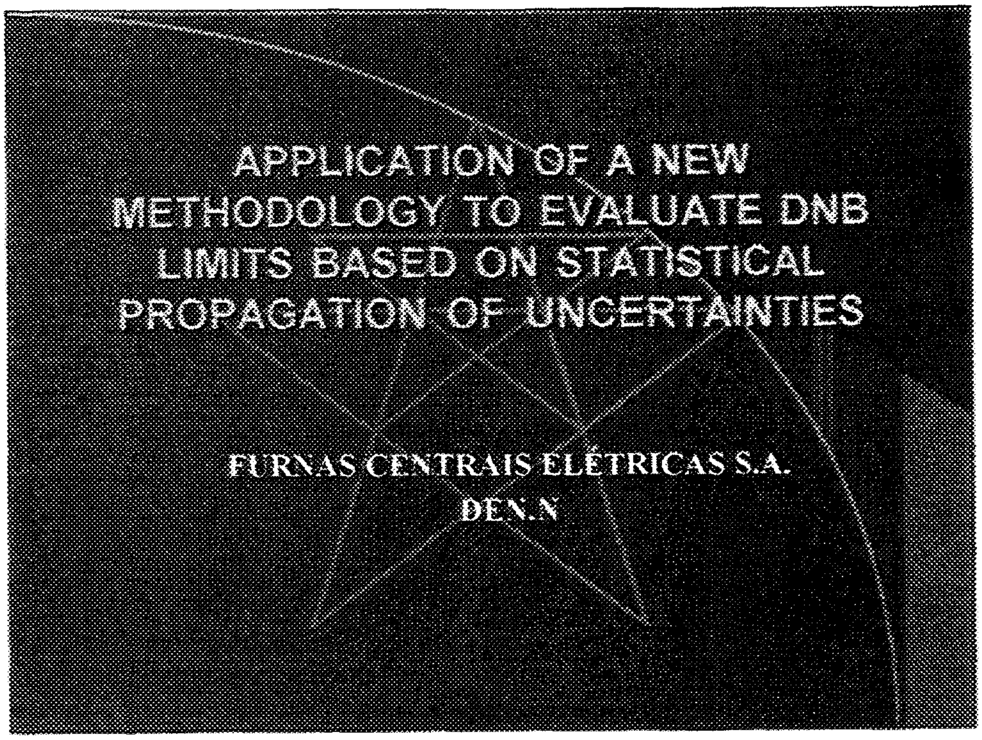
The scope of reanalysis required to justify an increase in the licensed tube plugging level depends on the following considerations:

Concerning the potencial impacts of tube plugging discussed before, only the decrease in primary flow has a Technical Specification impact.

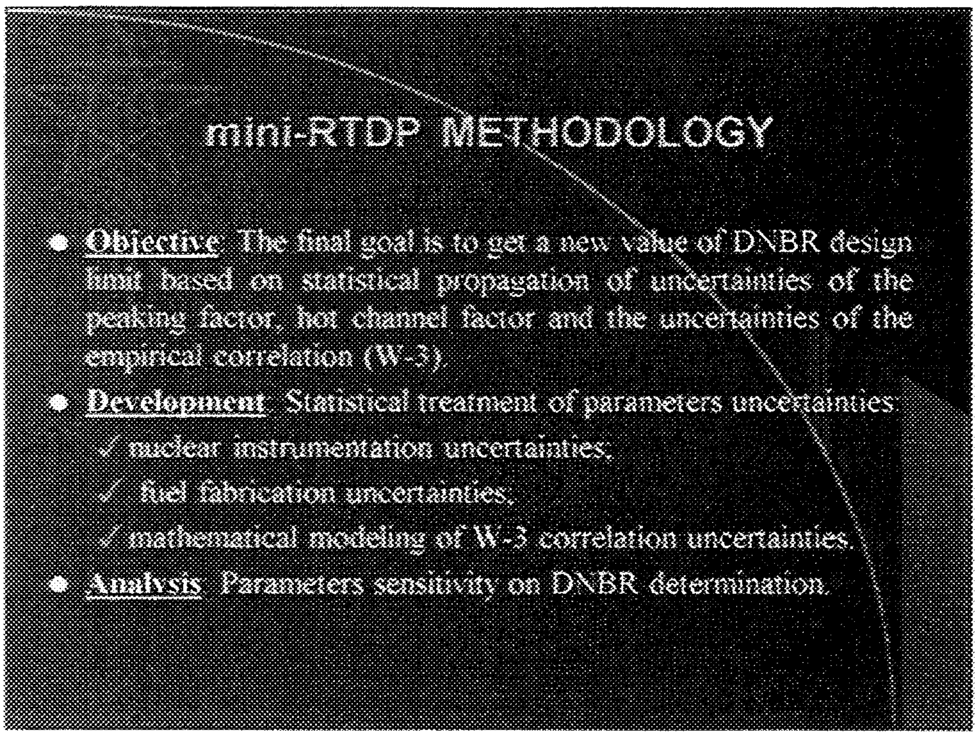


• The first step was to determine up to what level S.G. tube plugging in Angra 1 would not constitute an unreviewed safety question and that no Technical Specification changes would be required.

• Then, the minimum steam pressure required to generate full power and associated with the turbine control valves wide open position was determined.



• This project developed in Furnas apply a new methodology named mini-Revised Thermal Design Procedure (mini-RTDP). This methodology is based on statistical propagation of uncertainties. This new methodology allow a gain in safety margin avoiding licensing problems for operation of the reactor on its maximum power. This new methodology reduces the level of conservatism in parameters used in the DNBR calculation, wich are in their most unfavorable values with the standard methodology, by using their best estimate values.



•Hot channel factor ($F^{E}_{\Delta H,1}$) = Engineering enthalpy rise hot channel subfactor.

This subfactor accounts for variations in those fabrication variables which affect the heat generation rate along the flow channel. These variables are pellet diameter, density and U-235 enrichment.

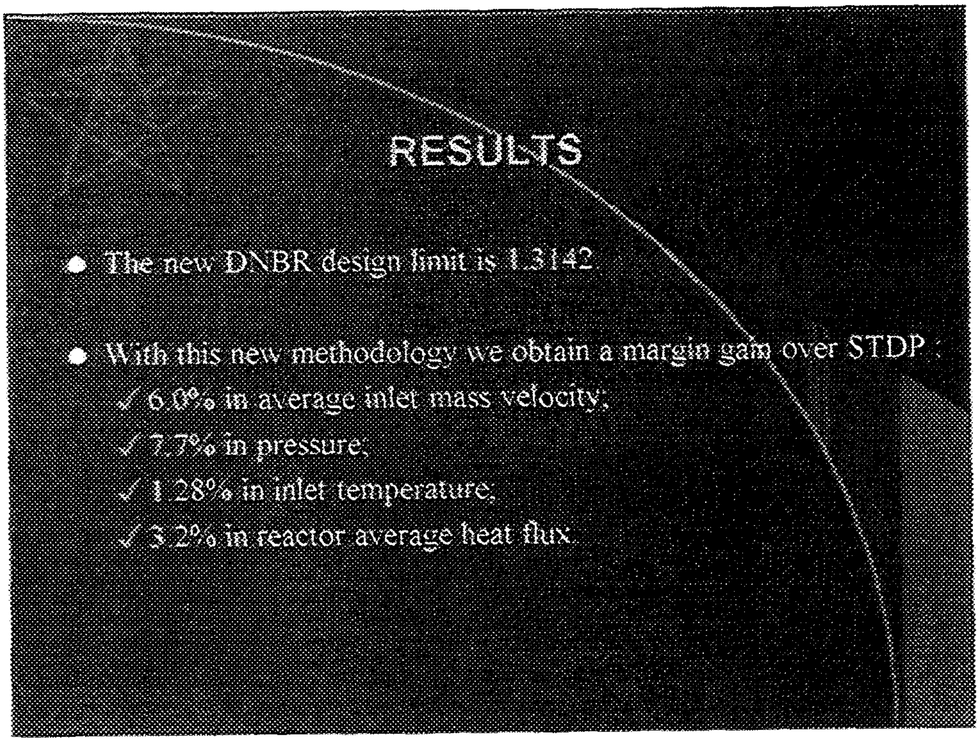
•Peaking factor ($F^{N}_{\Delta H}$) = Nuclear enthalpy rise hot channel factor.

The ratio of the linear power along the rod with the highest integrated power to the average rod power (including uncertainties).

Related with nuclear instrumentation errors.

•We must analyse the influence of these two parameters above in COBRA IIIC/MIT code. This is made by a sensitivity analysis of the code output due to isolated variations of each parameter.

•mini-RTDP Methodology = Parameters associated with plant instrumentation (temperature, pressure, power, flow) are at their worst values. Peaking factor ($F^{N}_{\Delta H}$) and hot channel factor ($F^{E}_{\Delta H,1}$) are at their nominal values.



•The new Departure from Nucleate Boiling Ratio (DNBR) design limit is based on statiscally combining the peaking factor and hot channel factor uncertainties with the DNB correlation uncertainties.

•The new value is 1.3142 that is greater than 1.3000 used in standard methodology (STDP).

•The standard method (STDP) currently in use is extremely conservative, and may result in penalties to the reactor power due to an increasing plugging level of steam generators tubes.

•With the new methodology we obtain flexibility in nuclear, thermal and hydraulic design. If average inlet mass flow is reduced in 6.0% DNB will not occur with 95 percent probability at a 95 percent confidence level.

3P

XA9744870

HUMOS MONITORING SYSTEM OF LEAKS IN TO THE CONTAINMENT ATMOSPHERE

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1. INTRODUCTION

HUmidity MOnitoring System (HUMOS) has been developed and designed to detect the presence of leak in selected primary circuit high energy pipelines and components that are evaluated from the point of view of Leak Before Break (LBB) requirements, Ref [1].

Ref [2] requires to apply technical tools for detection and identification of coolant leaks from primary circuit and components of PWRs reactors. Safety significance of leaks depends on:

- leak source (location)
- leak rate, and
- leak duration.

Therefore to detect and monitor coolant leaks in to the containment atmosphere during reactor operation is one of important safety measures.

As potential leak sources flange connection in the upper head region of VVER reactors can be considered, Ref [3].

HUMOS does not rely on the release of radioactivity to detect leaks but rather the relies on detection of moisture in the air resulting from a primary boundary leak. Because HUMOS relies on moisture and temperature detection, leaks can detected without requiring the reactor to be critical. Therefore leaks can be detected during integrity pressure tests and any other mode of operation provided the reactor ventilation system is operating and primary circuit and components are pressurized.

2. FUNCTIONAL REQUIREMENTS

HUMOS shall be capable of:

- detecting leak rates from high energy pipelines and components of 4 liter per minute within one hour,
- detecting required changes in air humidity,
- detecting required changes in air temperature,
- leak rate estimation.

HUMOS also meets following functional requirements:

- on-line (automatic) operation,
- off-line operation (dialog with operator)
- to provide alarms and diagnostic messages,
- electromagnetic compatibility (to meet requirements of IEC801-2, IEC801-3, IEC801-4 and IEEE C62.41),
- environmental requirements and radiation resistance
- self testing during operation
- quality assurance.

3. HUMOS DESIGN

3.1. Hardware Design

A schematic diagram of the HUMOS hardware is in Fig.1. Hardware consists of:

- integrated humidity and temperature sensors
- cables (sensors - DAQ units)
- DAQ units
- communication and power cables (DAQ units - containment penetrations - cabinet)
- cabinet with an industrial PC and accessories (power supply units, power distribution etc.)

Cabinet is located in the diagnostic room.

3.2. Software

The HUMOS version~~ for NPP Temelín runs under op~~erating system MS DOS 6.0 or higher. HUMOS soft~~ware design meets requirements l~~isted be~~llow, namely~~:

3.2.1. User Environment is:

- ~~Using the~~ mouse
- Graphics environ~~ment that fully supports keyboar~~d and/or mouse control
- Menu choice
- Windows (program user communication takes place in user windows)
- Text windows
- Graphics windows
- Information windows
- Messages windows

3.2.2. Screen Layout

The screen is divided into five region from top to bottom:

- Application name,
- information area,
- main working area,
- messages area, and
- main menu.

3.2.3. Menu (Mode)

Alone quick change mode:

- Entering passwords
- Mode change - on-line, off-line, stand-alone.

3.2.4. Menu System

In this window user can set some system variables, such as a working mode or passwords, perform system tests, obtain information about program and system or quit to operating system.

- Entering passwords.
- Window system (Mode, Computer Test, Measuring System Test, Communication Test, General Test, System information, Program information, External Tests, Quit).

3.2.5. Menu Diagnostic

Shows the current system diagnostic, list of all active alarms in given moment.

3.2.6. Menu Display

User can select a format for data display. Data are displayed direct in the main working area or a separate window is opened.

- Graphics display shows the time history of selected output.
- Digital display can be used to display current values and status of all sensors.
- Sensor layout can be displayed.

3.2.7. Messages

In this window (screen area) all alarms, warnings and other messages are displayed.

3.2.8. Menu Area-Graph

In this menu all functions to control graph area are listed:

- Move.
- Zoom.
- Offset (allows to change relative position of displayed time history).
- Show (this function allows to identify any from displayed lines include display of values in given point and change of lines setting).
- Normal.
- Back (previous graph is redrawn).
- Load Graph (allow to load predefined or user stored graphs).
- Save Graph (the graph will be saved under the name).
- Set Axis X.
- Set Axis Y.
- Select Showed Sensor Output.
- Lines Identification.
- Graph Set.

3.2.9. Menu Area - Digital

Following menu can be displayed:

- Update (list of digital displayed points according to newest value).
- In Color (lines with sensor in alarm state are displayed).

3.2.10. Menu Print

If a printer is connected, the~~ output goes to this prin~~ter.

3.2.11. Menu Help

Corresponding help to action or current window can be displayed.

3.2.12. Error Messages

If system error occurs, this error is reported in an error message window. (Examples: PC/TCP resident module is not loaded, not enough memory, cannot write temporary file etc.).

3.2.13. Alarms and Diagnostic Messages

System alarms show that a phenomenon pointing to a possible hardware or software error was detected and/or leaks are detected.

3.2.14. Sensor List

3.2.15. Leak Location and Leak Estimates

4. HUMOS TESTS

4.1. Test Design and Test Apparatus

Test were performed to demonstrate that small amounts of humidity released by a leak into air stream in containment can be detected. The tests attempted to simulate the specific mass flow rates of both the air and potential leaks during normal reactor ventilation operation. In order to accomplish this test apparatus of duct, fan and injection devices were designed and built in Energovýzkum.

A schematic diagram of the test apparatus is in Fig.2. The test apparatus housed the following components:

- fan powered by an asynchronous electric motor (fan volumetric rate up to 30 m^3 per second),
- inlet air filter,
- exhaust pipeline section of the diameter of 1000 mm,
- water/steam into air injection unit,
- humidity, flow rates, temperature and pressure sensors,
- data acquisition and evaluation system.

4.2. Water / Steam into Air Injections (leak simulations)

Series of water and steam injections into air were performed at the air volumetric flow rates of 2 to 10 m^3/s and injected mass flow rate of 0,04 to 130 g/s (0,24 to 10 l/min) and sensors responses were measured and evaluated.

4.3. Sensor Response to Leaks

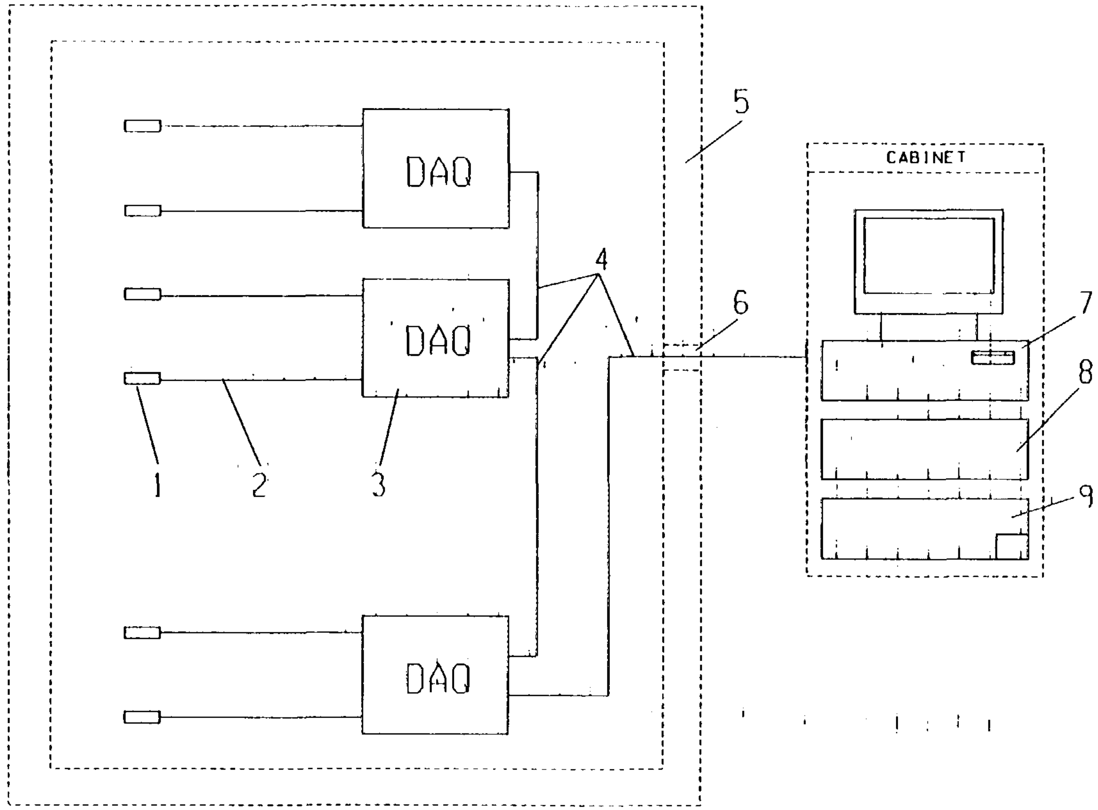
An example of sensors response on leak rates at a constant air flow rate is in Fig.3. Obtained test results correspond to theoretical predictions.

5. REFERENCES

[1] Procedure for LBB qualification, Requirements on Safety Reports Content and Scope, ČSKAE 1/1991

[2] U.S. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems

[3] Requirements Nr. 352-TZ-281, OKB Gidropress, 1987



1 -	INTEGRATED SENSOR	6 -	CONTAINMENT PENETRATION
2 -	CABLE	7 -	IBM PC INDUSTRIAL COMPUTER
3 -	DAQ UNIT	8 -	POWER SUPPLY
4 -	COMMUNICATION AND POWER CABLE	9 -	POWER DISTRIBUTION UNIT
5 -	CONTAINMENT		

Fig.1: HUMOS hardware configuration

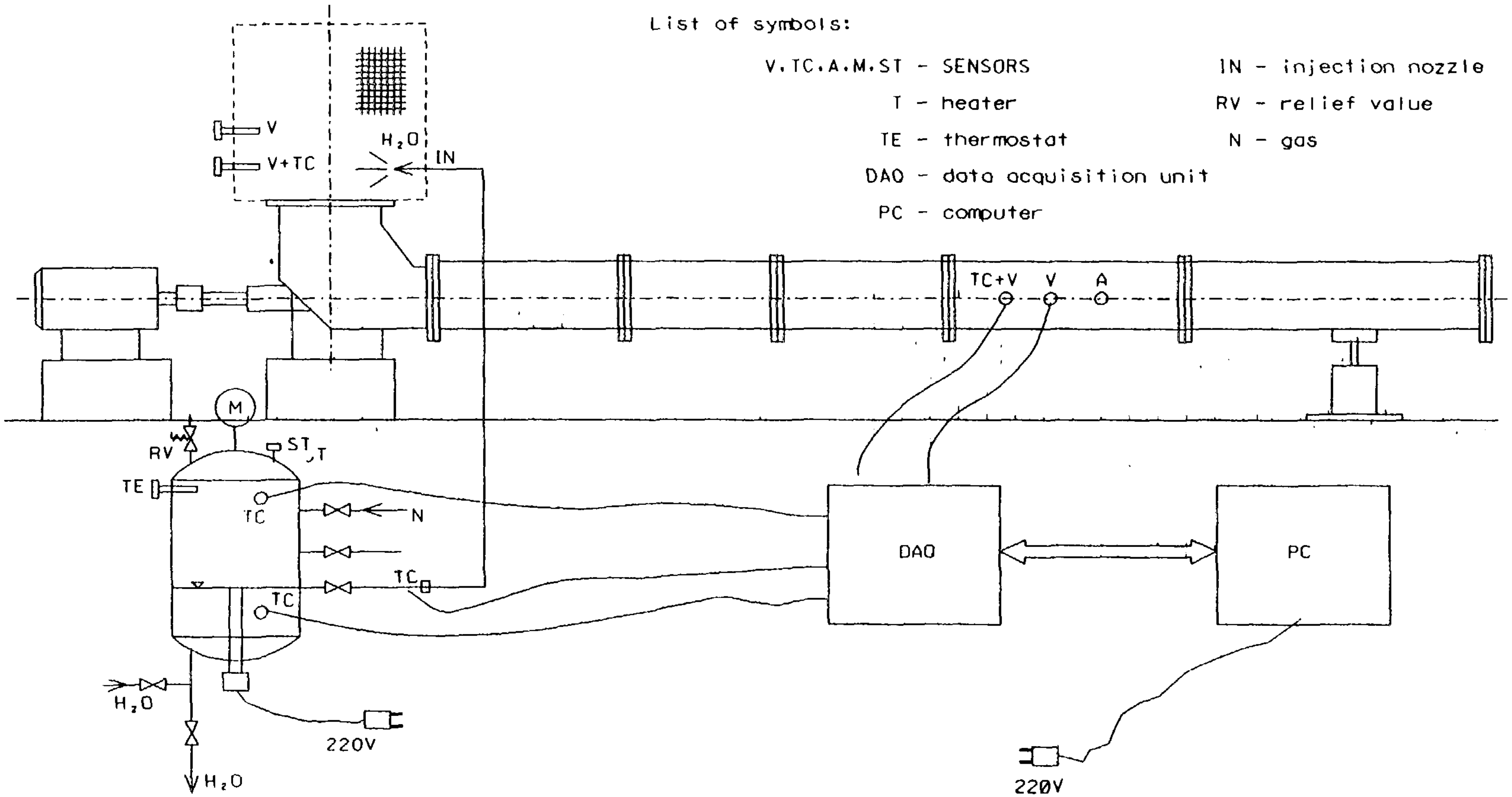


Fig.2: Schematic diagram of the HUMOS test apparatus

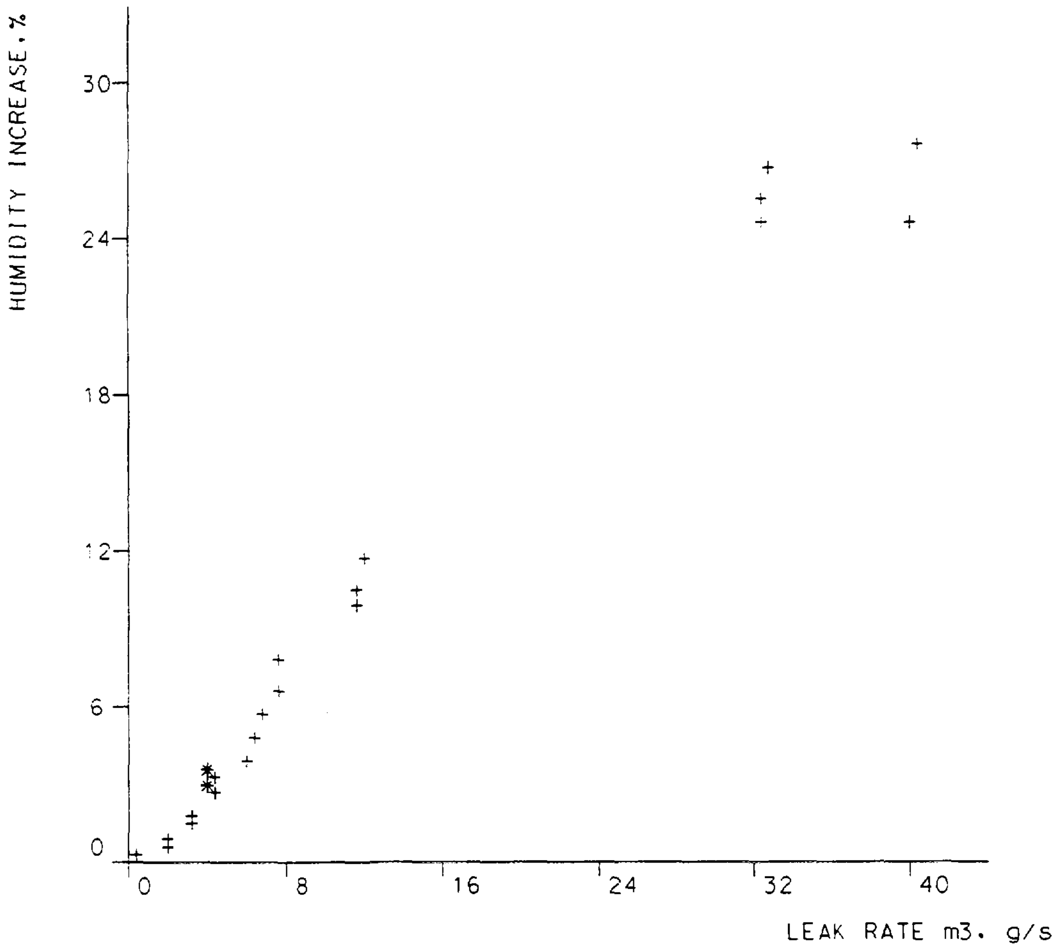


Fig.3: Humidity sensor responses on injected water (+) and stem (*) rates m_3 at a constant air flow rate

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EARLY DETECTION OF POWER SYSTEM DITURBANCES AS A CONDITION FOR SAFE OPERATION OF THE DUKOVANY NPP IN THE "ISLAND OPERATION MODE"

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Abstract

System failures are statistically documented facts in the operation of large power system. There are many failure modes (e.g. sudden short cuts, inadequate interventions, and malfunction of protective devices), the majority of the root causes, however, originates in a transmission grid itself. The probability of grid disturbances is generally low but the impact of a single event could have resulted in heavy economical losses. (The severe disturbance of the Californian power system in August 96, leading to tripping of a number of thermal and nuclear unit, may serve as a recent example of the event of interest.)

The power plants have to be of satisfactory design level to overcome all events accompanying the disturbances. They also must cope with non-nominal parameter values. This operation is typical after any heavy grid disturbance causing the power system to separate into several less-stable islands.

A change in the frequency values serves as a problem indication. If the frequency difference exceeds the limit value of 0.2 Hz, the system regulation reserve would probably be exceeded and the system should be separated into islands. A load should be decreased by frequency relays in accordance with predefined set of frequency levels. If the intervention of these relays is insufficient, the frequency is dropping further and eventually activates the local frequency relay in generators, which results in cutting off the units from the power system. The "Frequency Plan" worked out for the Czech Power System specifies a set of preventive measures along with the set of the frequency ranges.

This problem was solved in details for the Dukovany nuclear power plant. The design changes under preparation support a reliable as well as safe operation in the island-operation mode until a full restoration of the whole power system. The design modifications are based on the following three main innovations

- *An incorporation of the FREA 16 frequency relay into the protection circuits*
- *Large modifications in turbine control loops*
- *Installation of software routine for the operator to support the island operation mode*

The capability of the island operation has been demanded by relevant regulations both for the operating units and for that under construction. The capability can be tested under the transition to houseloads, as well as through the direct simulation of abnormal grid conditions.

1. Power system disturbance

Nuclear power plants are a part of the Power System (PS). Individual elements (simultaneously operated power plants, transformer stations, transmission lines, electrical loads) influence each other and a failure of any of them affects the other ones. The most serious disturbance is the splitting of a large synchronous system into several less-stable islands.

The causes of these failures vary (heavy short-circuits, inadequate interventions, and malfunction of protective devices), and, mostly originate in the transmission network itself. The failure propagates very quickly and worsens operating conditions of all power system elements, particularly the power plants.

The major disturbance generally arises from multiple unfavorable incidental events. Each disturbance is, therefore, analyzed in detail and corrective actions are taken. Despite of this the power system disturbaces occur relatively very frequently.

In very short period since well-known black-out in New York (1977) there had been several PS disturbances with heavy economic losses. We could mention Italy (November 1978, May 1989), North Germany (April 1979), England (August 1981), Sweden (December 1983), France (January 1987). The splitting of the power system in California in 1996 may serve as a very recent example.

On Saturday, 10th August 1996, at 15:48, a disturbance in the western system (WSCC) created four separate islands. Subsequent interruption of power supply to 7.5 million customers lasted for the period from several minutes up to nearly six hours. The disturbance began with a loss of 500 kV line in the area of Port Island, caused by carelessly performed coordination of power system control and maintenance. Parallel lines were overloaded, and transmission voltage dropped. The conditions resulted in subsequent tripping of other lines and power stations.

Besides the interruption of power supply to plenty of customers the disturbance caused the total trip of fifteen thermal and nuclear units in California. Several of these units could not start the power operation for several days.

2. Early failure detection - the condition of the NPP safe operation during PS disturbance

On occurrence of a failure in the PS, the main task is to clear the failure, protect the process equipment against damage, and restore the normal operation as quickly as possible. The power plant design must meet the following requirements:

- The power plant must not cause system disturbances
- The power plant must not worsen an already arisen disturbance
- The power plant must overcome transient effects accompanying the disturbance

- The power plant must cope with the operation on non-nominal parameters, which is characteristic for steady state following a heavy system disturbance accompanied by system splitting into islands.

Early PS failure detection is the basic presupposition to meet the above requirements. The magnitude of a change in PS frequency serves for detection of occurrence of a heavy PS disturbance. Relevant actions on the side of both power generation and consumption are initiated on reach of certain values. These limit frequencies are defined in so-called "Frequency Plan".

Table 1 Actions on frequency decrease

49.8 Hz	Automatic transition of TG control to speed control with switching off the power governor
49.0 Hz	Level 1 of frequency unloading by 12% of the load
48.7 Hz	Level 2 of frequency unloading by 12% of the load
48.4 Hz	Level 3 of frequency unloading by 12% of the load
48.1 Hz	Level 4 of frequency unloading by 14% of the load
47.9 Hz	Disconnection of Dukovany NPP from the grid and transition to the houseload conditions

Table 2 Actions on frequency increase

50.2 Hz	Transition of TG control to speed control with switching off the power governor
52.5 Hz	Disconnection of Dukovany NPP from the network and transition to the houseload conditions

3. PS disturbance detection and clearing

If frequency deviation exceeds 0.2 Hz, it indicates a splitting of a large synchronous system into several islands. It means the system regulation reserve would probably be exceeded. The power plants shall maintain the frequency within acceptable limits by changing their power outputs. If the to the disturbance propagates and the frequency deviations rise, the power plants must be disconnected from the outside network.

The effects related to the system splitting may be divided into three phases.

1. Disturbance occurrence

The first time phase means fast and big changes in all parameters of alternators. The right behavior of the unit is conditioned by as quick as possible

a) disconnection of power control of turbines (of PI character)
b) switching-over the turbine control to speed control (of P character)

2. Island operation

The development of frequency in the second time phase determines the ratio of resource power to "island" consumption.

a) In more favorable case the resource power is higher than the consumption of the area supplied. The proportional speed control of power plants ensures the balance between the sources and consumption. After attenuation of the transition effect the frequency stabilizes on the level that would be a bit higher than the one prior to the disturbance.

b) Otherwise the source power is less than the consumption. The frequency might not stabilize. Due to the proportional speed control, machines are loaded with maximum values on certain frequency, but the frequency still continues to drop. On further drop a part of the load must be cut off by unloading frequency relays. On their proper function an excess power is ensured and frequency is stabilized. On their incorrect or insufficient function the frequency continues to drop until the local frequency relays in individual plants are activated and cut off the units from the power system.

3. Island liquidation

Final phase involves controlled achievement of nominal frequency of the island. After fulfillment of synchronizing conditions, individual islands are interconnected. To ensure successful synchronization:

a) maximum level of active power peak of individual units must not be exceeded
b) network protection must not act

To ensure prevention against occurrence of power system disturbances and to eliminate their propagation, analyses have been carried out of behavior of decisive plants including analyses of protection function with heavy short-circuits in the vicinity of the plant connected with cutting off the equipment affected.

4. Assurance of safe operation of Dukovany NPP in the island mode

The requirement of Dukovany NPP resistance to heavy power system disturbances means to cope with transition to the island and to ensure reliable operation of Dukovany NPP within the island system for the period necessary to restore synchronous operation of the whole power system. Dukovany NPP design solves only two basic modes:

- basic load operation with normal power system
- houseload operation (or idle operation) on disconnection of the 400 kV output switch.

Island operation represents a new operating mode. Designs and effectiveness of protective automatics were verified on the interconnected PS model (MODES) and the model of NPP Dukovany (DYJE).

EGU Praha performed model verification of dynamic stability of the Dukovany nuclear power plant with large frequency disturbances of power system. They found out insufficient resistance of Dukovany NPP to failure states of the power system (2). Safe operation of the

Dukovany NPP in the island-operation mode **is not possible at present**, because it mainly depends on the characteristics of turbine generator unit control and cooperation with reactor power control. On big changes in frequency the hydraulic speed governor has been applied practically without limitation (static set to 5%). Pre-governor of the turbine output is working against the power system/island requirements, and goes to the manual mode after achievement of its extreme position. On frequency drop TGs increase the power irrespective of reactor capability. With long-time drop in frequency TGs achieve the nominal power with electro-hydraulic transducer opening to approx. 20%. It is unacceptable from the viewpoint of unit protection effectiveness that requires decrease in TG output.

Without special modifications the big PS failures mean higher risk for safe operation of Dukovany NPP. The actions taken can be divided into 3 groups as follows:

I) Incorporation of FREA 16 frequency relay into the protection system

The basis of the modification for island-operation mode of Dukovany NPP is installation of the frequency automatics that is based on the FREA 16 frequency relay. The automatics generates output initiating signals that are introduced into I&C and PIS.

II) I&C modifications

The modifications ensure safe transitions on occurrence of an island and safe operation of the unit in speed control

III) Installation of SW supporting the operator in the "ISLAND OPERATION" mode.

Since the "island operation" is an event with low with low frequency of occurrence the unit control room operators must have sufficient information during PS failure. By means of the SW support they will be given better overlook of the technological process during the unit operation in the island.

The modifications performed are clearly drawn in the process diagram of the Dukovany NPP unit as shown in Fig. 1.

1. Installation of frequency automatics
2. Turbine governor modifications
3. Reactor controller modifications
4. Modifications ensuring limitation of condensate flow on sudden power decrease in the island operation mode.

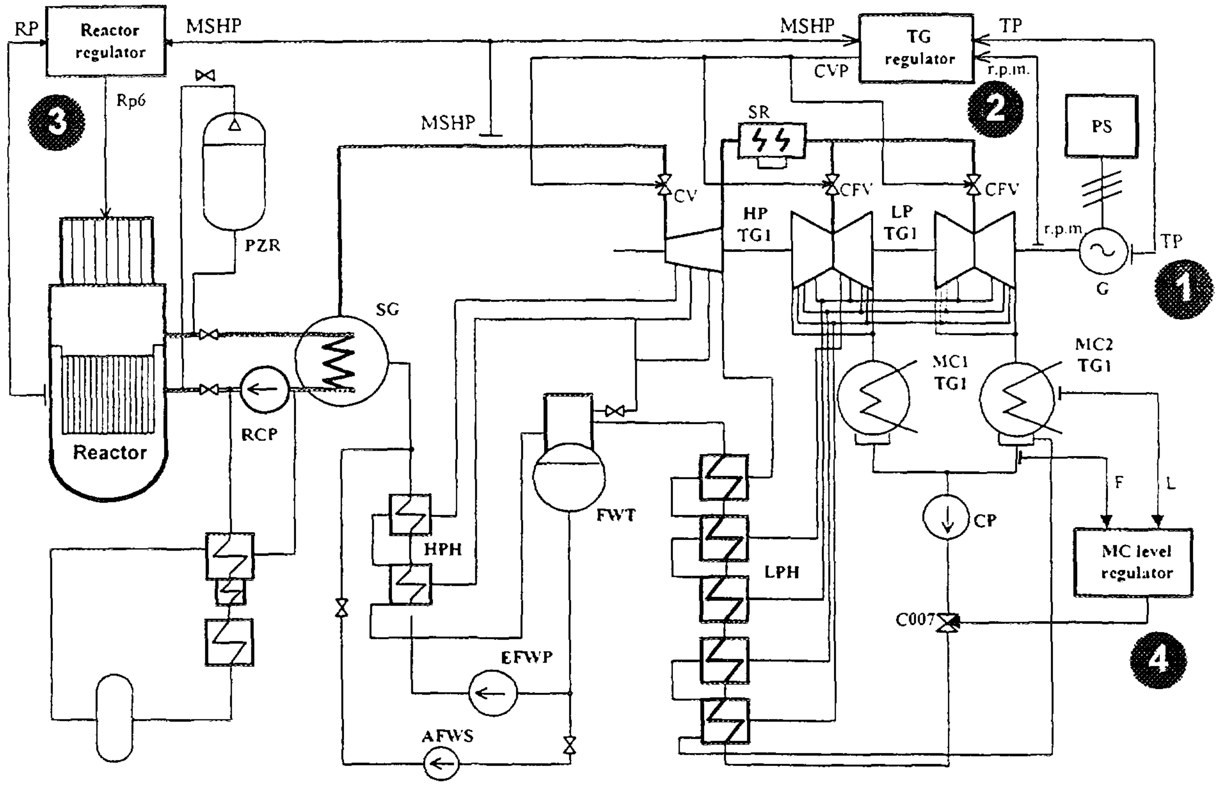


Fig.1 Principal diagram of Dukovany NPP unit with controllers and identification of modifications for island operation mode

5. Verification of safe operation of the nuclear power plant in the island operation mode

At present, when the Czech Power System has been already connected to UCPTE, the requirement of safe operation of the unit in emergency situations in the Power System becomes more important. The operating characteristic of the units is defined in the following UCPTE documents:

- Measures to prevent from occurrence of heavy failures and eliminate their extent - UCPTE document no. 13
- Measures against heavy failures in interconnected network - UCPTE document no. 16
- Thermal power station operation with lower frequency and voltage - UCPTE document no. 20

Verification of capability of the unit in the ISLAND OPERATION mode (including the method of evaluation) is stated in relevant regulations for both the operating units and those under construction. The basis is the breaking test of the unit with blocked signal of switching-off the 400 kV output switch. The test simulates real process of sudden transition of the unit from stable PS to so called minimum island, whose power corresponds to the houseload.

From the moment of its disconnection from the Power System till its re-synchronization to the grid after houseload operation, the power unit undergoes electromagnetic, electro-mechanic, thermodynamic (working medium: water-steam), thermo-mechanic, and mechanic transients.

Satisfactory behavior of the unit mainly depends on the correct function of:

- excitation system and alternator voltage controller
- steam turbine control system (speed control and protections) including by-pass stations
- reactor control system (maintaining the steam parameters within acceptable limits)
- unit electrical system
- unit operators, whose interventions influence significantly upon successful performance of the whole transient

6. Breaking test performance and evaluation

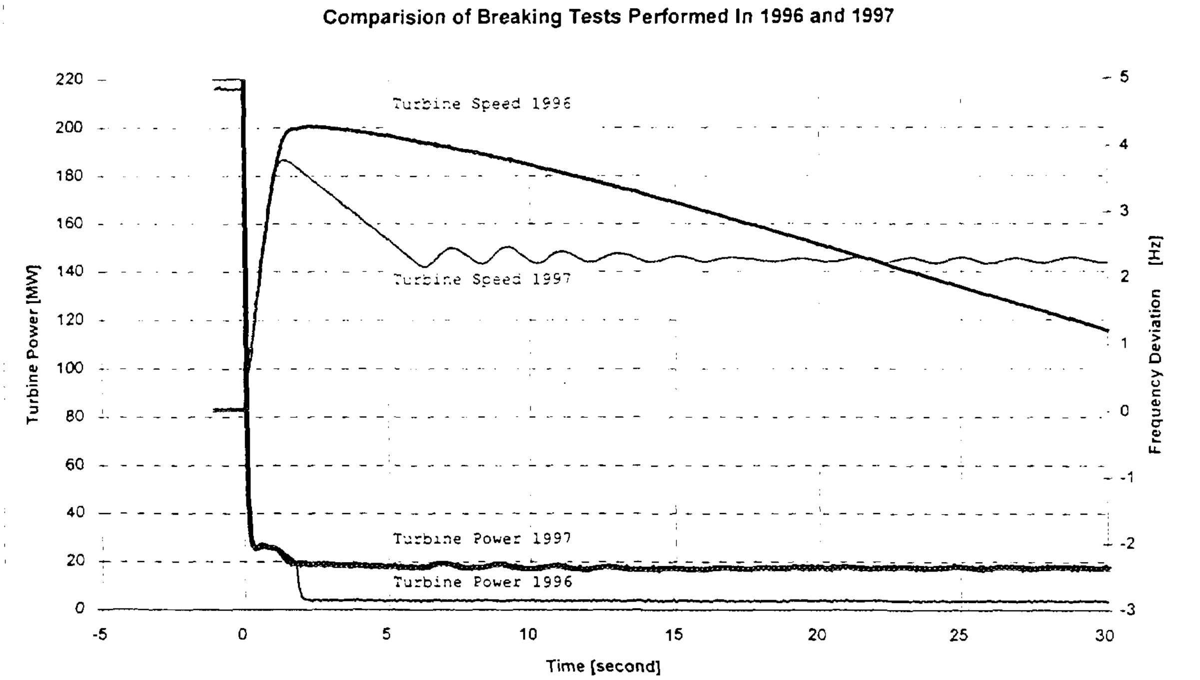
The tests have been designed and implemented in such a way that they would simulate as truly as possible the real conditions with minimal risk of damage of the unit and affected part of the power system. For the safety reason, the breaking tests differ from the real process in several aspects:

- Breaking test is implemented on the unit in good technical state, after inspection and adjustment of all decisive control and safety systems;
- Operators are prepared for and concentrated on the breaking test;
- Other qualified specialists from the power plant and manufacturers take active part in the breaking test;
- Breaking test is prepared organizationally, including collaboration with dispatching centre and relevant switching station;
- Important parameters of selected equipment are measured by special devices
- The moment of cutting-off is stated in momentary optimum operating conditions of the unit and network

The basic criterion of success is a reliable transition to the houseload with 2-hour operation in this state. In detail it means that:

- During the breaking test, the maximum transition speed of the turbo-set must not reach the limit speed, in which the turbine could be cut off by quick-acting valves. The speed characteristic during the transiet to the houseload must be damped, with fast stabilization at required value.
- Dynamic behavior of the other parameters comply with expectations and must be evaluated individually. Their incorrect characteristic should finally reflect in the resulting characteristic of the turbo-set speed.
- On the breaking test, the maximum transition voltage on the turbo-alternator terminals must not reach the limit value, for which the overvoltage protection is set up.
- Other process equipment of the unit must function in such a way that the turbo-set would get enough steam of appropriate parameters. These factors mainly influence upon the subsequent island operation, not initial phase of mechanical transition mode.

Prior to project work order the NPP Dukovany had performed the breaking test of one of the unit turbine generators (in February 1996, Unit 3). The test confirmed the conclusions of analyses, although it had not been successful. After implementation of modifications ensuring the island mode operation, the test result was positive (June 1997, Unit 3). The characteristics of TG speed and power are shown in Fig. 2 for both tests.



7. Conclusions

At present a system has been installed in all power plants in the Czech Republic to ensure safe operation on heavy power system disturbances. The capability of operation in so called island mode is verified according to the UCPTE documents.

Modifications have been implemented gradually for all units of the NPP Dukovany. In June 1997 the modifications were tested successfully for Unit 3 by means of:

- breaking test
- 2-hour houseload operation

The newly installed equipment enables to detect the PS disturbance early. At the same time the unit control room operators get sufficient information to control the transition from full load to the houseload island. The units may be operated in this mode for up to several hours. This decreases the probability of origin and mainly of propagation of disturbances in the power system in the vicinity of NPP Dukovany.

IAEA TC Meeting on „NPP Diagnostics - Safety Aspects and Licensing", Portoroz, Slovenia, 23-26 June 1997

Modern Diagnostic Systems for Loose Parts, Vibration and Leakage Monitoring

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Abstract

The modern diagnostic systems for loose parts, vibration and leakage monitoring of Siemens marked improvements in signal detection, ease of operation, and the display of information. The paper gives an overview on:

- Loose parts monitoring system KÜS '95 - a computer-based system. The knowledge and experience about loose parts detection incorporated into this system can be characterized as „intelligence".
- Vibration monitoring system SÜS '95 - a fully automated system for early detection of changes in the vibration patterns of the reactor coolant system components and reactor pressure vessel internals.
- Leak detection system FLÜS - a system that detects even small leaks in steam-carrying components and very accurately determines their location. Leaks are detected on the moisture distribution in a sample air column into which the escaping steam locally diffuses.

All systems described represent the latest state of technology. Nevertheless a considerable amount of operational experience can be reported.

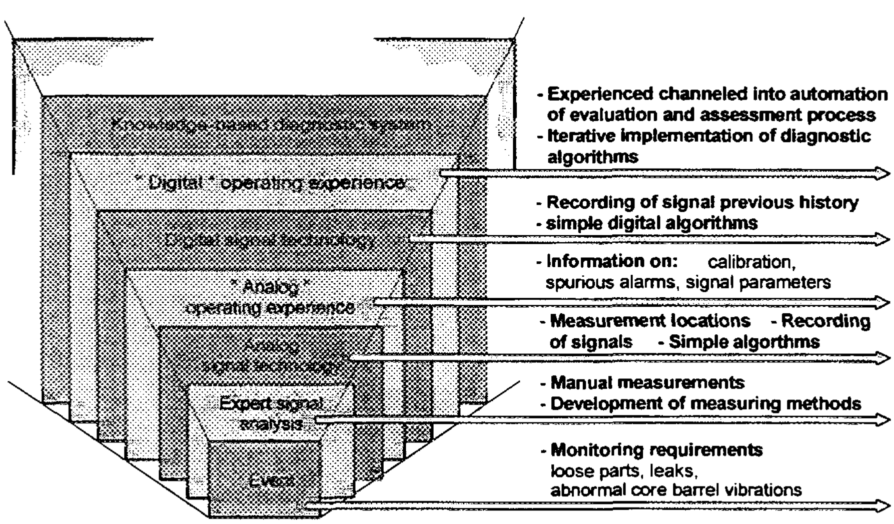

Figure 1 - System transition from monitoring to diagnostics

1 *Introduction*

Continous coolant-system monitoring has been used successfully in nuclear power plants for many years to increase plant reliability and availability, which in turn has improved plant operating

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economy. Monitoring systems provide early detection of abnormal mechanical vibrations, loose parts, leaks, incidents of material fatigue and changes in the condition of valves.

The monitoring systems used in nuclear power plants have undergone further development over the past three or four years, and are now complete diagnostic systems (Figure 1) which, in addition to analyzing measured data, perform an evaluation and visualize this in an easily understood form. This is particularly true of systems which monitor structure-borne noise, i.e., detect and localize loose parts, and those which monitor vibration.

2 *Technical Concept of Siemens Diagnostic Systems*

A new generation of monitoring and diagnostic systems has been developed by Siemens during the last years. The new generation, called Series '95, is based on PC. The new systems improve the reliability and quality of monitoring techniques and reduce the effort and staff needed for maintenance and evaluation.

The objectives behind the development of these new systems are both safety-related and economic. They include:

- Early detection of faults, and hence minimization of damage,
- Facilitation of fault clearing,
- Prevention of sequential damage,
- Reduction of inspection costs and radiation exposure.

The functions of the diagnostic systems include acquisition, processing, storage and documentation of the necessary data and also provision of the other aids required for quick and reliable analysis including the tools for automatic diagnosis.

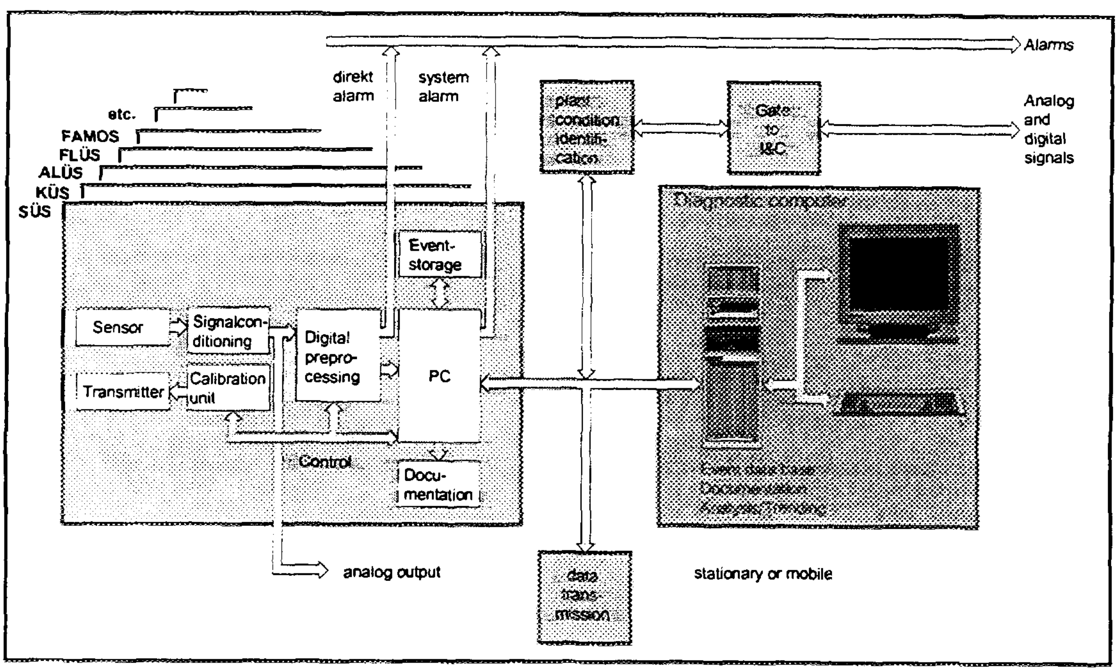

Figure 2 - Modular design of Siemens diagnostic systems

All monitoring systems (KÜS[1], SÜS[2], ALÜS[3], FLÜS[4], FAMOS[5]) follow a common technical concept in which autonomous individual modular systems perform identical functions (RMS generation,

[1]KÜS - loose parts monitoring system (Körperschallüberwachungssystem)
[2]SÜS - vibration monitoring system (Schwingungsüberwachungssystem)
[3]ALÜS - acoustic leakage monitoring system (Akustisches Lecküberwachungssystem)
[4]FLÜS - moisture leakage monitoring system (Feuchtelecküberwachungssystem)
[5]FAMOS - fatigue monitoring system

calibration, analog-digital conversion, filtering, etc.) with identical modules (Figure 2). For analyses exceeding the needs of a single system, data records may be transferred to a diagnostic computer, using remote data transmission or computer networks.

The systems features are:

- Continuous operating mode - the systems work practically without dead time. Data evaluation does not impede continued monitoring.
- All the signals sent to the systems are system captured and processed simultaneously and parallel to one another.
- Automation - the systems perform continuous monitoring without operator intervention; even complex alarm conditions can be implemented. Instrument chain adjustments, function checks, calibration, etc. are operations performed periodically and automatically.
- Comprehensive information storage - in the event of an alarm, all the necessary information, including the previous history of the event, is stored parallel and in digital form for all measuring channels.
- A detailed system documentation, supplemented by extensive evaluation and analysis functions, has been prepared for the systems and the results obtained.
- Physical calibration - a basic principle of the system design is that control is exercised over instrument chains and algorithms in order to produce physically identical variables during monitoring; i.e. for loose parts monitoring a remote impact hammer introduces periodically impact events into the nuclear power plant components.
- Informative character - a signal is transmitted to the control room, but there is no intervention in the reactor control system. Further decisions are left to the expert, whom the system supplies with the necessary tools for a rapid and reliable diagnosis.
- Inclusion of operating data - the systems can record operating data which modifies the monitoring task as a function of the operational status of the power station.
- Self-monitoring - all parts of the monitoring systems are themselves monitored by the system. Every part of the monitoring system is periodically checked for operability by the system itself, and if faults are detected the system triggers an alarm. A computer failure does not interrupt the monitoring process, but merely restricts certain functions.
- Same user interface - in line with modern PC developments, all the systems have a largely identical menu-driven user interface following Windows standard.

The various systems are based on a common technical design and are individually-working, standalone systems, equipped with the following functional components:

- Signal pickups,
- Signal conditioning unit,
- Digital pre-processing unit,
- Data acquisition computer,
- Analysis computer and peripherals.

3 The KÜS '95 - Loose Parts Diagnostic System

Two years after its introduction, the KÜS '95 loose parts diagnostic system for the detection and localization of structure-borne noise is already in operation in seven nuclear power plants. The system evaluates noises made by loose parts in an analysis and diagnostic phase. The main benefit of the KÜS '95 system is its ability to determine whether or not detected structure-borne noises coincide with previously recorded reference events. These reference events are used as the basis for subsequent diagnosis, however in addition they give the system the capability to decide if the noise in question is being made by a loose part or just a normal operational event, which is often the case. A normal event is filed as a „Known Event", thereby avoiding unnecessary evaluations.

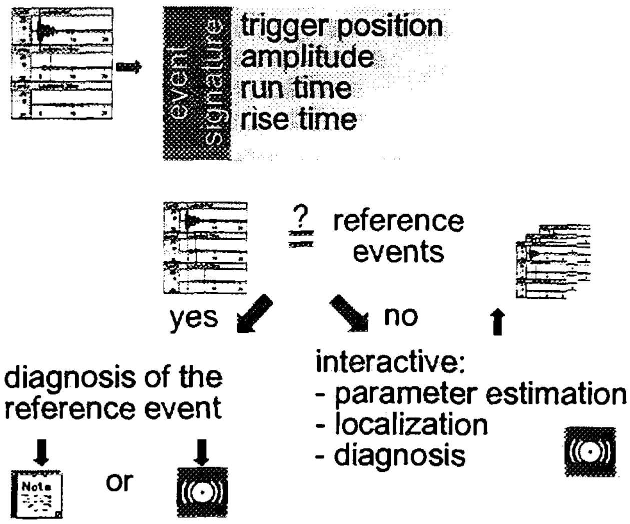

Figure 3 - Procedure of the KÜS '95 sensitive event classification to identify events

If the event is relevant or unknown, however, a KÜS system alarm is triggered. The operator can then either assign the event to one of the known classes of reference events in an off-line evaluation process, or can define the event as a new reference class and perform a corresponding diagnosis. The system incorporates all the analysis tools required for this process (Figure 4).

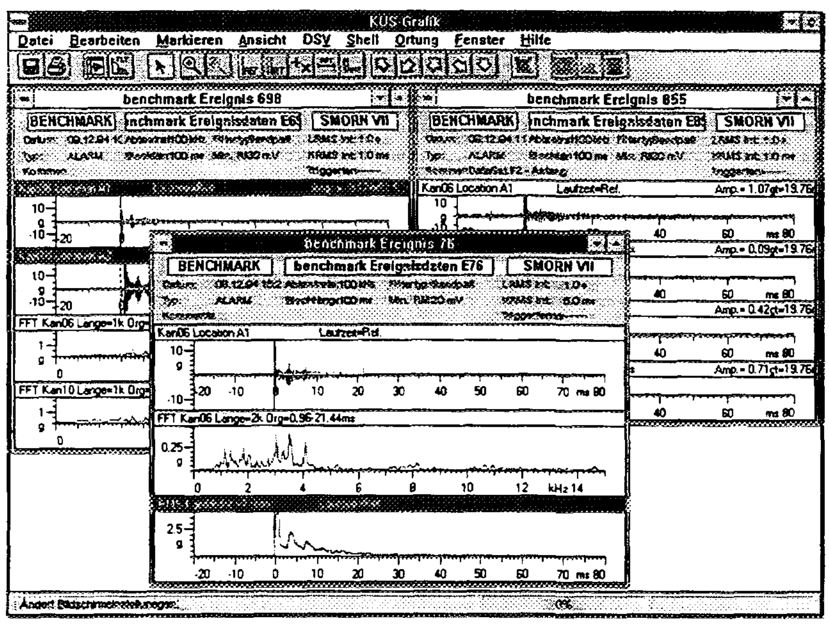

Figure 4 - Graphic analysis of an impact noise in the structure-born signal

KÜS '95 could prove its performance in the frame of the International Benchmark Test organized by the Nuclear Energy Agency of the OECD in 1995. The results were presented at SMORN VII (1995, Avignon). The data collected with KÜS '95 and Siemens evaluation methods (localization and mass estimation of different impacts) were outstanding amongst 18 participants (most of them being suppliers or developers of LPMS).

The KÜS '95 system was able to locate the points of impact to within a few centimeters. For this purpose the system measures the sound propagation time difference. It makes use of the fact that the burst induced by the impact propagates at the speed of sound, thus reaching the transducers at different times.

The structure-borne noise signals also contain information on the mass of the impacting object. The underlying physical principle is explained by Hertz's law, which states that the time during which two bodies remain in contact following an impact is a function of their masses and velocities. Siemens uses the frequency analysis method, which is based on the principle that the central frequency of the impact signal spectrum is indirectly proportional to the contact time (Figure 5). The KÜS '95 system was able to determine the mass to within approximately ± 30 percent - an accuracy which is perfectly adequate for practical applications.

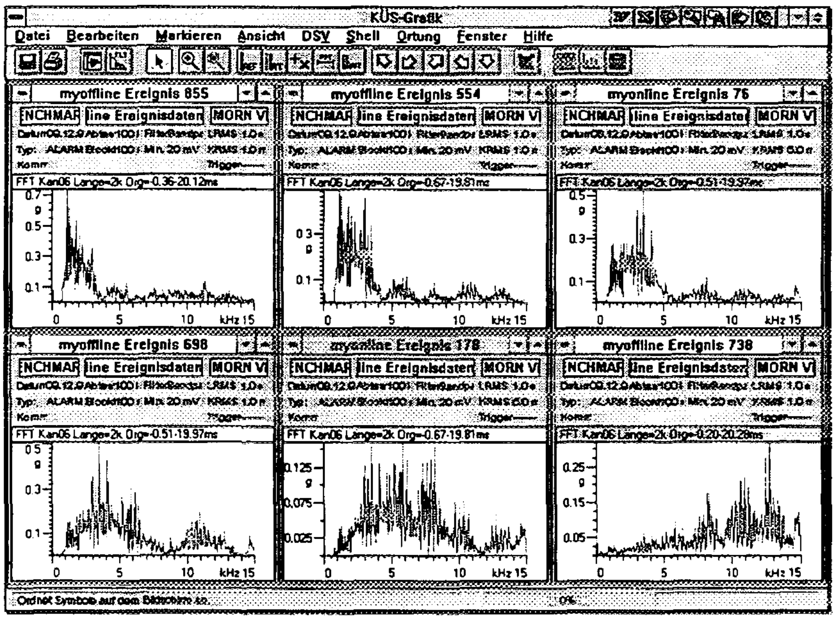

Figure 5 - Frequency spectra of impact noise of 6"-, 5"-, ... 1"-balls from the SMORN VII benchmark test - frequency spectra allow the mass estimation of the loose part

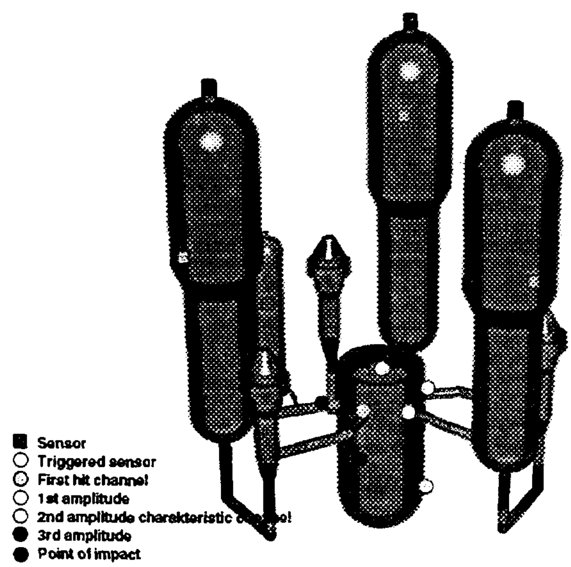

Figure 6. Depiction of event localization

KÜS '95 as well as the other modern diagnostic systems are designed for intuitive operation with clear presentation of monitoring results. The sequence in which individual channels are triggered is shown in a 3D display of the reactor coolant system which can be rotated stepwise in all three axes providing the user with virtual access to the reactor coolant system (Figure 6). The same 3D display is available for depicting the results of event localisation.

4 SÜS '95 - Complete Automatic Vibration Diagnostics

Licensing authorities require vibration measurements which must provide an early indication of changes in the vibration patterns of structures such as reactor pressure vessel internals and reactor coolant system components, as well as rotating machinery such as reactor coolant pumps. The expertise necessary to analyze the measured data has been systematically acquired and completed since the commissioning of these systems in Germany.

Previously, vibration measurements were time-consuming. The SÜS '95 vibration monitoring system now performs these necessary and routine tasks without operator intervention. The system is designed to comprise all features an automatic vibration monitoring system today can include:

- vibration monitoring of structures and rotating machines
- complete solution from the pickup through to the analysis system
- complete solution from the function test through to the results report
- complete solution from design through to service

The systematic SÜS '95 concept starts with the selection of pickups which were specially developed for the measurement of power plant components vibrations - e.g. the SAUM absolute displacement transducer - and optimized for their application in terms of sensitivity and frequency range (Figure 7). It goes without saying that they are qualified for use in the harsh conditions prevailing in nuclear power plants and can be remotely calibrated at any time.

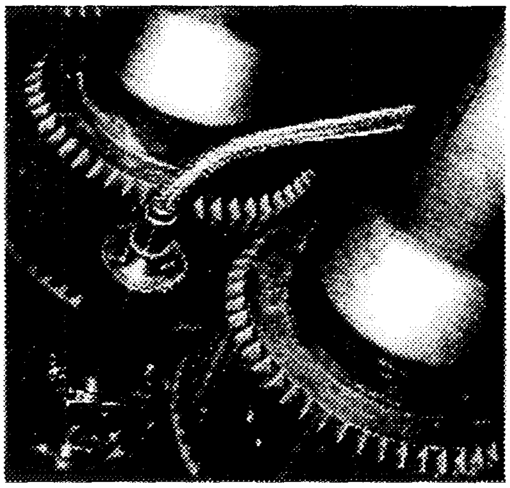

Figure 7 - The specially developed absolute displacement transducer SAUM for the measurement of power plant components vibrations.

SÜS '95 automatically performs a functional test of all instrument chains prior to each measurement, calibrates them and documents the current settings. This is the only possible way to implement a really automatic vibration monitoring procedure which excludes incorrect measurements and poorly adjusted instrument chains.The modular design of SÜS '95 guarantees service-friendliness as well as simple extensions to the system and its software.

SÜS '95 performs all the routine tasks required for vibration measurement single-handed:

- start-up of measurement at the required time (single or cyclical operation)
- selection and functional test of required instrument chains
- measurement procedure
- calculation of the required characteristic vibration variables, detection of changes and indication of threshold violations
- display of monitoring results in a 3-D image (Figure 8)

- recording of process parameters and allowance for effects of different operational conditions
- documentation of results in report form in accordance with applicable specifications (Figure 9).

The system provides comprehensive information when deviations are detected:

- designation of the affected component
- assumed cause of the change
- required action and scope of action.

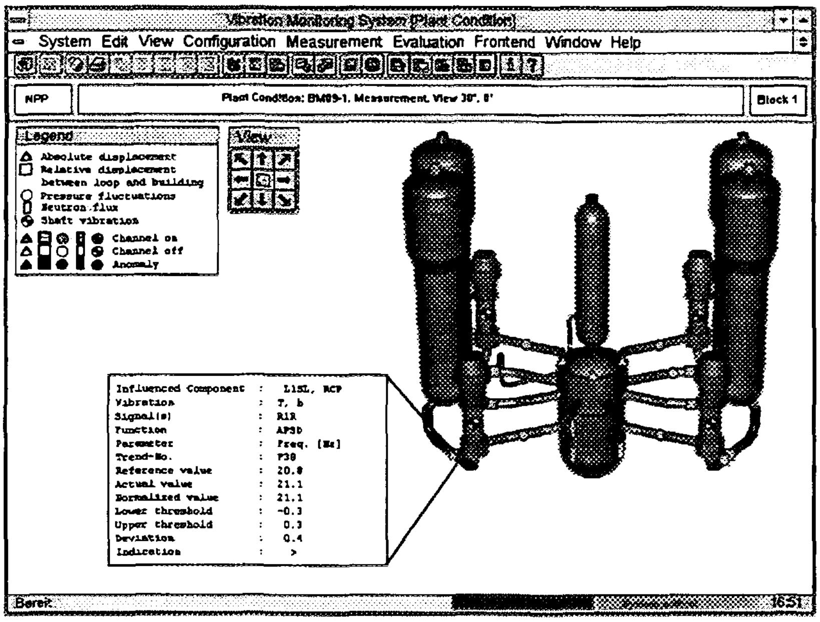

Figure 8 - SÜS '95 displays the location of an affected reactor coolant system component

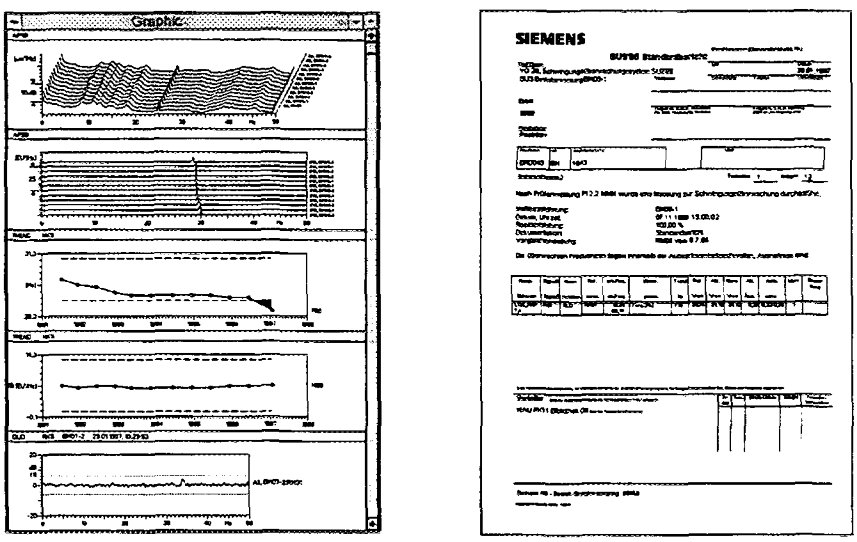

Figure 9 - SÜS '95 includes complex evaluation tools and performs automatically a standard report which presents the results of the vibration monitoring

SÜS '95 thus creates more scope for dealing with the actual plant and grants the user more time to attend to details before deviations develop into real problems.

The SÜS '95 vibration monitoring system is further enhanced by a comprehensive range of vibration monitoring services:

- concept and configuration of vibration monitoring
- development and production of systems: new systems/conversions/upgrades
- installation and commissioning
- system service and maintenance
- performance and evaluation of vibration measurements (reference measurements, repetitive measurements, special measurements)
- archiving of data, trend analysis and documentation suitable for independent experts
- dynamic tests in the vibration laboratory and in the power plant
- instruction, practical training and advice.

5 *FLÜS - Leak Detection by Monitoring Humidity*

The leak detection system FLÜS detects leaks in steam- and water carrying components, and also monitors the humidity of the ambient air. Special features of the new system include highly sensitive detection and leak localization which is accurate to within the meter range. The FLÜS system detects leaks in thermally insulated components at an early stage, and evaluates their development over time. Leakage from small cracks (leak-before-break criterion) is also detected.

FLÜS detects the humidity that builds up in the vicinity of a leak by means of a temperature- and radiation-resistant metallic tube filled with dry air and placed inside the insulation. The tube has diffusion points through which ambient humidity can pass at defined intervals. At predetermined time intervals, the air column is drawn through a moisture sensor (dew-point measurement). The moisture content of the air is measured for each diffusion point and the leakage rate is estimated based on these data. The location of the leak is determined based on the transit time and speed of the air column between the measuring location and the sensor (Figure 10).

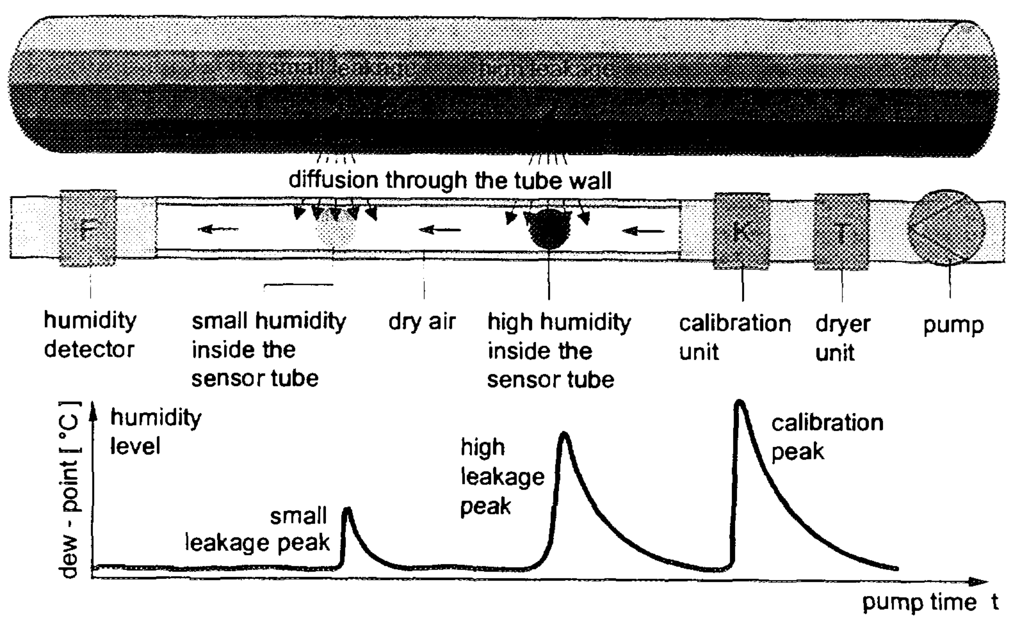

Figure 10 - FLÜS detects even small leaks in steam-carrying components and very accurately determines the location. Leaks are detected based on the moisture distribution in a sample air column into which the escaping steam locally diffuses.

The end of the air column sampled in each measurement is selectively marked by means of defined amount of steam. This test peak defines the reference values for transit time and moisture content and is also used for self-monitoring of the entire system (detector, pump, tube). A leak alarm is issued if a predefined reference curve is exceeded by a specific factor (relative alarm) or if a fixed threshold is exceeded (absolute alarm).

The modular measuring system is operated with a PC. The measured values are recorded periodically, monitored with respect to alarm criteria, and stored. If an alarm is issued, reports provide information on the leakage rate and location, and on the alarm itself.

The FLÜS measuring station, centrally located in an accessible compartment inside the containment building, can control as many as eight measuring loops, each up to 150 metres long with a spacing of approximately 0.5 metres between diffusion elements.

The FLÜS system has been applied for a German pressurized water reactor plant to detect possible leaks in the reactor pressure vessel closure head since 1995. During the qualification procedure for this application a mock-up was used to study the effects of leakage into the insulation of the reheater vessel. The results of this study confirm the excellent results:

- A leakage rate of 0.05 kg/h is reliably detected; measurable effects are already noticeable at 0.01 kg/h.
- There is a clear correlation between leakage rate and dew point. The leakage rate is quantifiable if the FLÜS indicator is calibrated for the respective insulation.
- The response time of the FLÜS monitoring system is as short as 15 minutes.
- A local increase in moisture can be localized within 1 m.

To monitor the reactor pressure vessel closure head, one sensor tube is placed inside the insulation and another outside it. The two tubes are combined, e.g., to decouple the leakage monitoring process from normal fluctuations in the ambient humidity.

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SUPERVISORY MONITORING SYSTEM IN NUCLEAR POWER PLANTS

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ABSTRACT

Monitoring of a power plant is one of the essential tasks during operation and the computer-based implementations are nowadays seemingly quite mature. However, presently these are still not satisfactory enough to meet the high standards of the licensing requirements and they are mostly not truly integrated to the plant's design-based monitoring system. This is basically due to the robustness problem as the majority of the methods are not robust enough for the monitoring of the safety parameter set in a plant or intelligent supervision. Therefore, a supervisory monitoring system (SMS) in a plant is necessary to supervise the monitoring tasks: determining the objectives to be obtained and finding the means to support and fulfill them. SMS deals with the changing plant status and the coordination of the information flow among the monitoring subunits. By means of these the robustness and consistency in monitoring is achieved. The paper will give the guidelines of knowledge and data management techniques in a framework of robust comprehensive and coordinated monitoring which is presented as supervisory monitoring. Such a high level monitoring serves for consistent and immediate actions in fault situations while this particularly has vital importance in preventing imminent severe accidents next to the issues of recognition of the monitoring procedures for licensing and enhanced plant safety.

1. INTRODUCTION

The goal of monitoring is to detect process changes and faults during normal operations and to take actions to avoid damage to the process or injury to human operators. Process supervision or monitoring in an operating power plant is essential in two main aspects. These are in the first place, to avoid the accidents and in the second place system availability. Monitoring contains the following tasks. Fault detection and diagnosis; fault evaluation; decision on operating state; fault evaluation. To enhance the monitoring activities early process fault detection and localization is required such that sufficient time is provided for fault elimination and prevention of further fault development. Fault diagnosis is a major part of a monitoring task. From this viewpoint many fault detection methods have been developed [Willsky 1976], [Pau 1981], [Iserman 1984], [Gertler and Singer 1985].

The implemented methods use deterministic as well as stochastic signals. However, these methods are still rather simple and consist of mainly limit value checking of some available single signals or derived statistical quantities. The most important monitoring functions are the alarm handling and protection. These are achieved by means of conventional instrumentation which are foreseen for licensing. In parallel with the technological developments new instrumentation's and methodologies are endeavored to be integrated to the monitoring systems for enhanced safety and cost effective operations. In this respect, computer technology and its derivative artificial intelligence (AI) can be referred in the first place. Due to this, a number of parametric, non-parametric methods and AI implementations are developed for fault diagnosis the outcomes of which are used in various ways. Among these methods and implementations mention may be made to fast Fourier transform (FFT) techniques, time-series analysis , hypothesis testing methodologies. Also, new information processing technologies may be mentioned along this line [Türkcan et al. 1996; Ciftcioglu and Türkcan , 1996]. Although all these implementations are nowadays seemingly quite mature, presently these are still not satisfactory enough to meet the high standards of the licensing requirements and they are mostly not truly integrated to the existing monitoring system so that they remain often as the secondary systems for operator's aid and are articulated as 'decision support systems'. This is basically due to the robustness problem as the majority of the methods are not robust enough for the monitoring of the safety parameter set in a plant or intelligent supervision. For instance, majority of time-series methodologies make use of residuals for fault diagnosis where reference residuals are defined precisely for each normal operational status and normal operational changes e.g., power level change, require re-calibration. In a sensor-failure scheme, a failed sensor is assumed to have no effect on the computation by analytical redundancy although to some degree it effects the accurate status determination. In AI category, similar problems are involved in the majority of the neural network (NN) approaches, for instance.

2. SUPERVISORY MONITORING TASKS

Referring to above mentioned shortcomings, a supervisory monitoring system (SMS) in a plant is necessary to supervise the monitoring tasks: determining the objectives to be obtained and finding the means to support and fulfill them. SMS tasks is quite different from the conventional monitoring processes since the adaptation of the SMS behavior or structure to deal with the changing plant status and the coordination of the information flow among the units are essential. By means of these the robustness and consistency in monitoring is achieved. The supervisory monitoring is performed through an accurate system model in multilevel form and it addresses higher level monitoring aspects. Modeling can be constituted by several components like static modeling, dynamic modeling and computational modeling.

Operating in real-time, the tasks of a SMS can be divided into two major categories:

1. Fault detection and diagnosis which includes optimal state estimation
2. Model management which includes simulation and learning

Fault detection and diagnosis performs the detection of incipient failures and causes of the failures. It should also report the failures to the operator. This pass of information should be done intelligently so as to help the operator focus on the current part of interest. Fault detection can be carried out by means of several ways; namely, by processing received alarms, by model referenced process verification, and by data and trend analysis. In each case, the diagnosis has to be done as fast as possible to avoid the obscurity of the real cause of the fault.

For *alarm handling*, an essential problem the alarm overwhelming, that is a situation when too many alarms are generated. This should be carefully considered by alarm gradation. For the *process verification* of the operation by the measurements at hand, the measured quantities should be compared regularly against the values from the models based on the first principles. Also it should include a reference model to predict the future operational values and verify if these predicted values match the actual measurements. When a difference is detected, a potential fault is detected and it should be possible to infer some prognostic information. In case of a model fault, an error is found in the model description which is subject to improvement. For the *data and trend analysis* sensory data from the plant should be validated prior to their use. Should there may be discrepancies between the incoming data and the model-based counterparts consistency and valid operational status must be established.

Model management maintains and exploits a process model reflecting the current state of the process. Measurement values, trends, failures and structural changes are recorded in a data base so that topological and behaviorally correct process model is always available for the other tasks of the SMS such as prognosis tasks. Hence the model management acts as a data base as well as it identifies current and future states and trends of the plant and evaluates the model. This is achieved by model and parameter updating and learning.

For *model and parameter updating*, all sensory data should be stored into the model so as to keep the model as accurate as possible. Any change in the plant dynamics should be reflected into the model. Also the model must be regularly aligned to the measured process state. This is because in case a discrepancy, to distinguish between model fault and operational fault is rather difficult. This process is called model alignment. As the plant model is in the form of several layer corresponding to shallow-knowledge and deep knowledge, there should be a systematic transfer of behavior of the lower layer to a higher level. This is called abstraction. Generally the lowest level is fed with the plant's sensory data. The higher levels should stay tuned to plant data by evaluating the abstraction relations.

For *learning* one can distinguish two modes: supervised and unsupervised. Supervised learning is initiated and coordinated by the predetermined means and it is performed in adaptive form. In contrast with this the unsupervised learning is executed automatically and autonomously by the system in an intelligent way.

Learning addresses quantitative values of the system parameters, validity of existing concepts or creation of new concepts. Learning of quantitative values is concerned with numerical aspects, like parameter estimation to update current parameter values or backpropagation for weight adjustments in neural networks. Learning with respect to existing concepts deals with updating current structure of process models with known modeling elements. It applies process identification techniques to determine the current structural properties of the process in terms of known concepts and updates the model if necessary. Leaning new concepts is necessary when certain process phenomena cannot be described with available concepts.

Learning some inherent features of the process can require unsupervised learning. It requires parameter estimation to adjust initial parameter estimation errors and to deal with parameter variations due to process aging (e.g. burn-up).

Learning should be performed on every layer of the hierarchical monitoring system. Once the model is constructed and put to operation, learning must be performed on-line fully automatically. The gained experiences by learning can be used for derivation of heuristic rules or cases to speed up future search for solutions. For example, symptoms could be stored together with the actually observed fault, as a new case in a case-based fault diagnosis system.

Learning is a necessary ingredient of intelligence and it is one of the main characteristics of the SMS. Learning can also address the improved use of solution procedures and of combinations of task methods. By analyzing the effectiveness of current combinations, new rules can be created for composing improved combinations for future use. Such a learning process benefits the monitoring process with its advanced inference mechanism called supervisory inference. This is depicted in Fig.1.

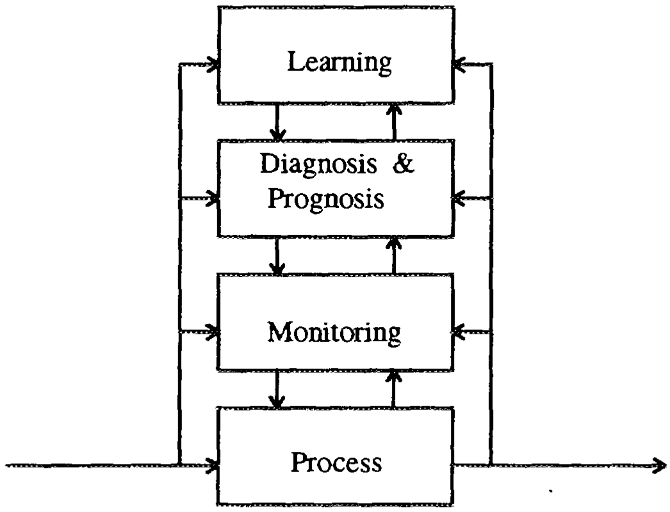


Fig.1. Supervisory inference

3. MAN-MACHINE SUPERVISORY MONITORING INTERFACE

The more powerful and flexible man-machine interface hardware and articial intelligence (AI)-based support tools become, the more emphasis needs to be put on the cognitive demands posed to a human operator. Since the operator is assumed to sustain the overall decision-making task in the supervisory monitoring loop, this imposes criteria with respect to the maximum cognitive load an operator can deal with properly. Human errors may be of different types. These are broadly categorized as *detection errors, cognitive errors and execution errors.* Therefore, taking an operator perspective while designing the man-machine interface is imperative.

From the above discussions, it is clear that the complexity of the supervisory monitoring systems should be hidden behind a simple, easy to understand interface. It is important to recognize that humans do not think numerically. When building operator interfaces for supervisory monitoring systems, this implies that data should be presented conceptually, that is symbolically or linguistically. The vast pattern recognition capabilities of humans should be exploited by applying new display techniques to create enhanced, data rich pictures. Linguistic labeling of data could be applied, using fuzzy classifications, to generate a natural language interface for better understanding of system messages.

The operator thinks on several levels of abstraction, from detailed low-level monitoring to aggregated overviews. The operator zooms into or out of a part of the operation to switch to a more specialized or higher category of decisions. Orthogonal to that, the operator is able to perceive multiple information flows in parallel, like a single screen containing several trends or momentary operational status values is viewed and interpreted at once.

4. KNOWLEDGE MANAGEMENT AND DATA HANDLING

4.1. Knowledge management

In the supervisory monitoring system, knowledge-based expert systems play the essential role. For knowledge representation the computational effectiveness is required. Solutions to problems in knowledge representation and inference should satisfy the real-time constraint. Also, it is necessary to identify and formalize inference structures appropriate for dealing with incompleteness and uncertainty. An AI system must be able of reasoning with incomplete and uncertain information. The current status of the expert systems are briefly described below.

A first generation expert system is a shallow expert system which consists of a knowledge processing unit and a heuristic knowledge base. An expert shell contains no a priory knowledge. It has to be filled with domain knowledge prior to its use. Domain knowledge is captured in production rules. The production rule paradigm is a model for human reasoning. It captures an expert's experience and casual reasoning strategy. It is a representation paradigm where knowledge can be captured in the form of rules. The rules consists of compiled associations of facts and phenomena with solutions and actions. The knowledge base containing these rules is a large set of recompiled chunks of deep knowledge ready to use rather than a collection of shallow knowledge in the form of if-then rules.

In the first generation expert systems, two fundamental forms of reasoning process is involved. These are forward chaining and backward chaining. Both strategies work on production rules, but complementary. Forward chaining works from antecedent to conclusion, while backward chaining works from conclusion to antecedent. There are several ways for the improvements to increase search speed, performance. However first generation expert systems contain shallow knowledge. Expert systems based on purely shallow knowledge cannot give satisfactory explanations about their behavior and show abrupt degradation at the edge of their knowledge domain, since no compiled knowledge about cases that never have occurred before is available. Moreover, since knowledge elicitation depends on subjective human experts addressing only a limited number of cases, the expert system's knowledge domain is incomplete and possibly inconsistent of format and meaning. The disadvantages and limitations of first generation expert systems are summarized below in three categories, namely concerning human-computer interactions, problem-solving flexibility and extendibility-maintainability [Keravnou, 1990].

- Incoherent sequences of questions
- Redundant questions
- Historical information on a case is not maintained, requiring the user to enter it again for each consultation on that case.
- Inflexible user interface where information is required to be entered in very specific terminology's and formats, otherwise information is ignored.
- User is neither allowed to revoke an answer nor to pursue the effects of an alternative answer.
- Explanations do not cover al the explanation needs of the user.
- Performance degrades dramatically when dealing with rare case.
- Inability to recognize that a problem case is at the periphery or outside of its area of expertise.
- Difficult to modify the system's knowledge. Consistency checks are not facilitated
- Inability of the system to evolve on the basis of its experiences in problem solving

Causes of disadvantages and limitations above are explained as follows.
- Shortcomings of reasoning knowledge that it is not complete. The generic tasks and strategies are implicit,

- Shortcomings of domain-factual knowledge due to its structure which is not compatible to the way human experts model their knowledge.

The causes all originate from the differences and incompatibilities between human and expert systems knowledge representation and processing.

Referring to the shortcomings of the first generation expert systems, the second generation solutions are summarized below.

In second generation expert systems, knowledge is derived from the first principles introducing generality. However this does not imply that the resulting model precisely describe the physical model because the first principles are not detailed enough for the complexity of the real world.

Designing a second generation expert system with the objective to overcome a specific first generation limitation without solving this limitation in the context of others, is prone generating local, non-robust solutions (Keravnou, 1990). For sound improvements in second generation expert systems, the limitations in the first generation counterpart must be well understood. Hence, the architecture should then be designed from the perspective of the root causes and not of their effects. Then, the architecture will provide a global and thus effective solution.

Integrating first and second generation expert systems should make it possible to use heuristic knowledge to decide when to carry reasoning back and forth from heuristics to first principles. This is important when experience fails or is lacking or when the domain model is incomplete.

Efficiency and the ability to reason progressively ensure that model-based reasoning is performed in time. This is very important for critical situations where response time should be small to avoid imminent accidents. A progressive reasoning mechanism generates a preliminary answer using only a very small knowledge base. While time is available , gradually larger knowledge bases are accessed to refine this answer step by step. After some time, the current inference takes the precedence. This way, a subsequently more accurate answer is always available at any point in time.

4.2. Data Handling

To process the massive amount of low-level data from the plant, two approaches can be used. These are parallelism and hierarchy. For parallel data handling neural networks for solving pattern recognition and minimization problems is of particular interest. Here simple procedures are carried out on all data items concurrently by means of simple processors. There are many such processors and there is no need for complex data structures. In contrast with parallelism, in the hierarchical approach the data are structured as efficiently as possible in order to concentrate processing where it is needed. The two approaches are not mutually exclusive as they are approaches for the same problem. However hierarchical approach is more systematic where the data items are grouped into higher level categories. By varying the category level data may be viewed at various levels of detail.

With respect to supervisory monitoring, two aspects of data manipulation are of importance. These are trends and uncertainty processing which are described in the followings.

4.2.1. Trend analysis

Trends of process measurements are an important source of information. They indicate behavioral properties, such as oscillations and monotonic increase. By analyzing trends, it is possible to quickly extrapolate future events, crossing alarm boundaries for example. Several trend representations are available. Each of them has its own expressive power. Among the simplest methods are linear and non-linear regression analysis, least squares, exponential smoothing or moving average. These methods all process raw history data of limited time window length, generating mean behavior.

To represent oscillations as well, Fourier analysis is used, expressing signals as a sum of sine waves. In principle, the signal should be of infinite length for the Fourier analysis to work well. Therefore Fourier analysis methods produce results under the assumption that a signal will behave in the future as it has done in the past. An adaptation is the short-time Fourier transform, used to transform a fixed length part of a signal. A disadvantage of this method is its sensitivity to noise.

On the highest level of expressive power are wavelets, representing a signal as a series of predefined, scaleable wave shapes [Daubechis, 1992]. Wavelets allow for scaling of detail over time, because they are self-similar representations that compress or expand in time as frequencies of the original signal increase or decrease, respectively. Gabor type representations retain the shape of the envelope, while the frequency is changed. In wavelet representations the frequency is retained, while expanding or compressing the shape of the envelope. This allows to preserve all information contained in the original signal, in contrast with Gabor type representations where these are of fixed duration, forcing information to be thrown away when they are scaled. These are represented in Fig.2.

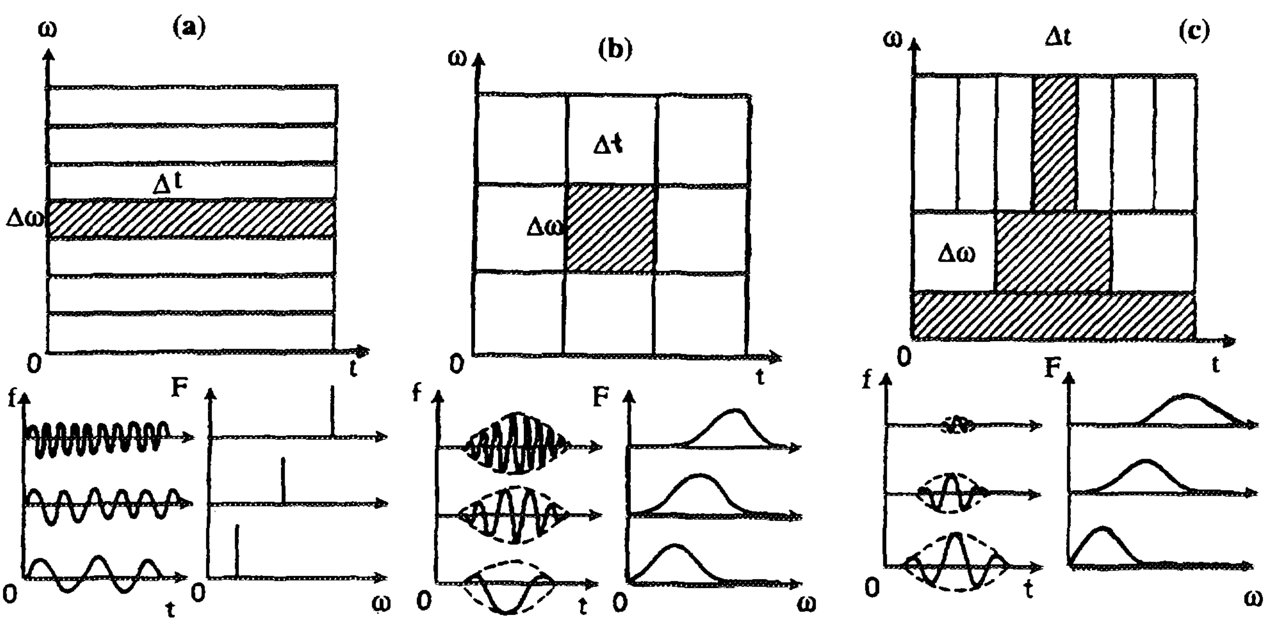
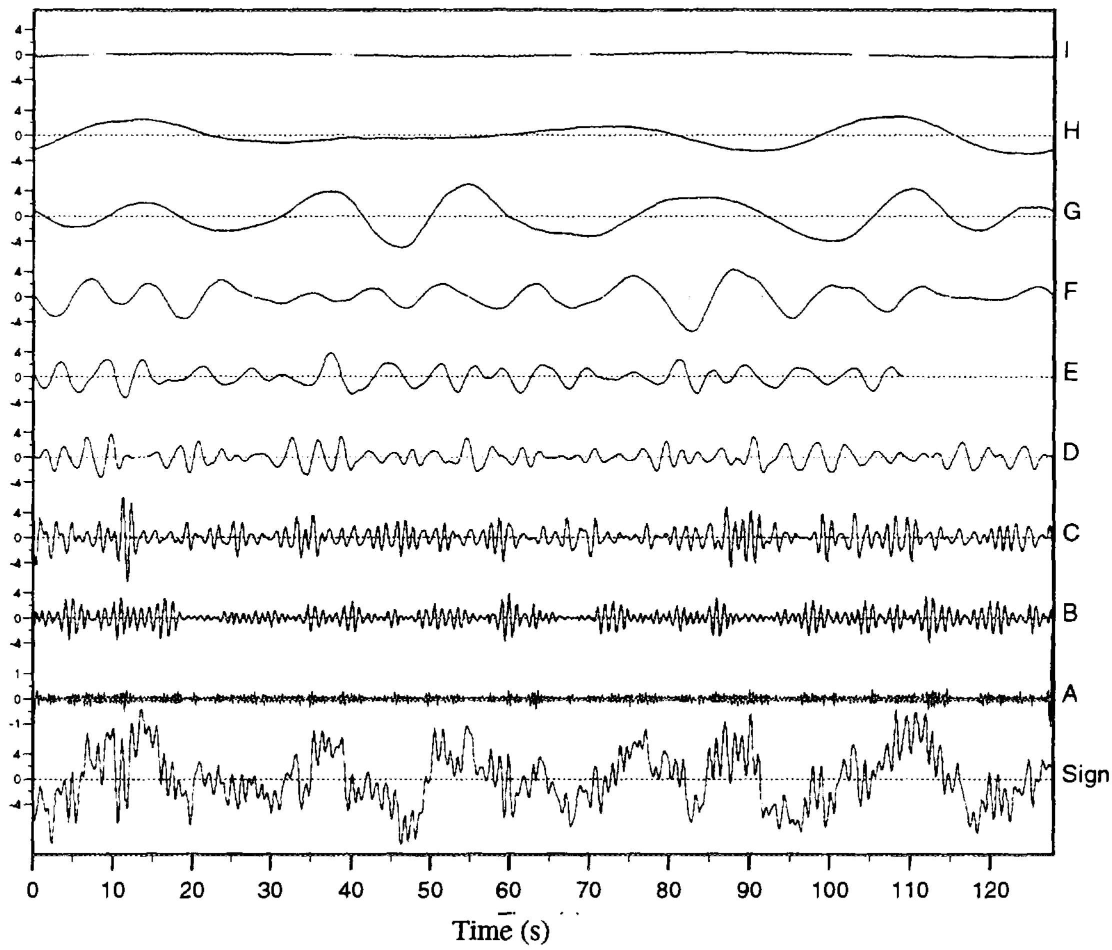


Fig.2 : Fourier (a), Gabor (b) and wavelet (c) representation where $\Delta\omega$ is frequency resolution, Δt time resolution. Note that each level of representation is orthogonal to each other, so that summation of the represented variations yield the variation from the sensor, i.e. perfect reconstruction *Such a representation does not assume periodicity of the data so that it is superior to FFT type analyses.*

The multi-resolution representation capability of wavelet analysis is represented in Fig.3.

Signal and Decomposed Signals



Signal: AC signal ; sampling rate 8 samples/second		
A: decomposed signal for 4.	to 8.	Hz
B: decomposed signal for 2.	to 4.	Hz
C: decomposed signal for 1.	to 2.	Hz
D: decomposed signal for 0.5	to 1.	Hz
E: decomposed signal for 0.25	to 0.5	Hz
F: decomposed signal for 0.125	to 0.25	Hz
G: decomposed signal for 0.0625	to 0.125	Hz
H: decomposed signal for 0.03125	to 0.0625	Hz
I: decomposed signal for 0.015625	to 0.03125	Hz

Fig.3 : Multiresolution signal decomposition by wavelet transformation
"Transient Detection by Wavelet Transform", Ö Ciftcioglu and E. Türkcan, SMORN VII, 19-23 June 1995, Avignon, France

When informative power of a trend representation technique is plotted against the scale of representation, each of the techniques takes a position in this space. The more a technique is situated in the upper right part of the figure, the more it is apt for supervisory monitoring as depicted in Fig.4.

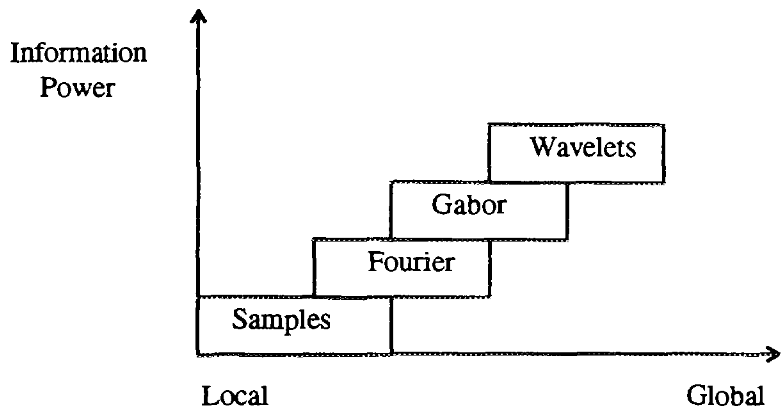


Fig.4 : Informative power of data representation

4.2.2. Uncertainty

Uncertainties can be handled by means of two different paradigms. These are analytical paradigm and rule-based paradigm. The analytical paradigm concerns the probability density of the uncertain quantity and the associated confidence levels. This type of treatment is rather conventional and well known. It operates on the measured values from the sensors the probability density being in most cases gaussian.

The rule based paradigm concerns the fuzzy sets and associated logic where fuzzy set theory was first introduced by Zadeh [1965]. A concept that plays a central role in fuzzy logic is the concept of a linguistic variable. The concept of a linguistic variable enters in the characterization of dependencies through the use of fuzzy 'if-then' rules. With fuzzy sets, a numerical value is classified into one or more linguistic labels, each with an associated membership value. This results in a multi-value representation since the membership functions, representing the numerical strength of linguistic labels for the domain of classification, overlap. An input value intersects with one or more membership functions of the input classification and it is classified by as many linguistic labels. Before entering a fuzzy system, numerical values are fuzzified. This is usually done by an input classification, matching input values against a chosen set of linguistic labels. These labels partly overlap so that a numerical value can be classified into more than one label, each with an associated membership value. Inference is performed by evaluating fuzzy production rules. Propagation of fuzziness is linear with respect to arithmetic operations. Logical combinations are performed by T- and S-norms for conjunctions and disjunctions, respectively. T- and S-norms have to fulfill four criteria, namely, they should be non-decreasing functions in each argument, be commutative, be associative and they should have an identity value.

Since a numerical value can be classified into more than one linguistic value, more than one rule might be triggered, producing several answers. This multiple answer is defuzzified to obtain a crisp numerical value.

The fuzzy approach can be supported by neural network approach as well. The relation between these two approaches referring to the supervisory monitoring is depicted in Fig.5.

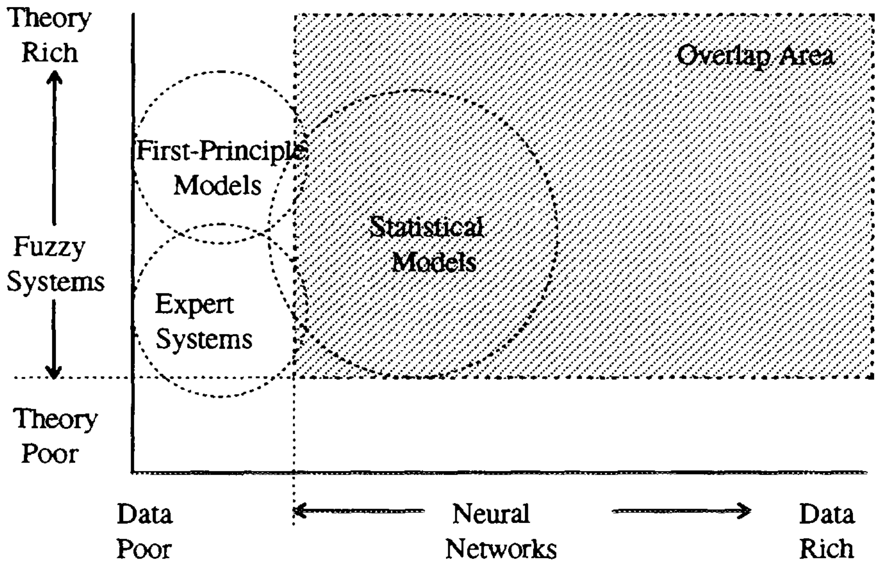


Fig.5 : Fuzzy versus neural networks in supervisory monitoring

5. CONCLUSIONS

The paper shows that supervisory monitoring requires high-level programming approaches to manage the increased problem complexity of large-scale, plantwide process monitoring applications. Modularity and symbolic processing are key issues for solving problems associated with supervisory monitoring. Modularity is needed in a hierarchical approach to deal with the high degree of large scale process complexity. Symbolic processing offers improved reasoning flexibility needed to handle a large variety of expected and unexpected situations.

Improved data representations are needed to effectively represent trends and anomalies of process signals. The detected and isolated trends are the source of motivation supplying information to higher layers of the supervising system. Not all signal analysis techniques are able to do this in sufficient detail, because they assume a certain structure of the signal, which is seldomly present in all signals of a process. Therefore, more expressive representation and analysis techniques should be used, which are general enough to capture the diversity of shapes and preserve the main signal characteristics in higher decision levels.

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TECHNICAL COMMITTEE MEETING
NUCLEAR POWER PLANT DIAGNOSTICS-SAFETY ASPECTS AND LICENSING

23 - 26 JUNE 1997

PORTOROŽ, SLOVENIA

SAFETY AND LICENSING ASPECTS ON NUCLEAR POWER PLANT DIAGNOSTICS IN SLOVENIA

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ABSTRACT.

Nuclear Power Plant incidents or accidents are initiated very often by a component failure even if it is not regarded as a safety related component. Plant safety, reliability and cost effectiveness can be enhanced and what is most important, the hazards for the environment can be reduced significantly through appropriate use of some specific Early Failure Detection Method or System.

In the present paper the current trends in Nuclear Power Plant Krško and licensing aspects introducing monitoring and diagnosis systems are described, to improve NPP reliability and safety.

Most of the nuclear power plants are currently under preparation of the program - a systematic approach to implement different Early Failure Detection Systems. Possible contents of these programs is described in detail in several technical papers. Usually, monitoring of specific nuclear power plants parameters, components and systems shall be considered as a tool of integrity.

Reasons for monitoring are contained in regulatory requirements and in operation experience, on the other hand, the licensee is highly interested for safe and reliable operation.

The operational events and failures are usually the first initiator for systematic approach to specific phenomena observation and its analysis.

The Slovenian Nuclear Safety Administration follows the efforts of NPP Krško in all above mentioned fields including training of personnel, adequate procedures, related QA Program and experience from other NPP utilities and Regulatory Bodies.

The licensing aspects contain all the specific steps as they are required for specific plane modification. As a guide, the methodology and guidelines from the US code 10CFR 50.59 for safety evaluation are taken into account. The safety review process is a complex activity of all departments at the Slovenian Nuclear Safety Administration. In the case of an unreviewed safety question an engineering evaluation and a through understanding of the design basis of the system are essential.

In the conclusion the possibility of the early failure detection operator support systems introduction into NPP Krško regarding existing systems is discussed.

1. INTRODUCTION.

The need for diagnostics systems introduction:

- Nuclear power plant decision (different events)

- Worldwide practice and experience

- Regulatory authority requirements

- NPP designer proposal

The introduction and permanent using some or more of the diagnostics systems, the NPP operator is pursuing mainly two of the following objectives:

- assure the highest degree of safety by continually observation of the important parameters obtained by these systems,

- the cost effectiveness of the NPP will significantly increase

On the one hand, the monitoring systems working during the plant operation make it possible to assess the overall conditions of the plant at any given moment and on the another hand, the time independent evaluation and analysis will be possible.

It is very important to choose the proper technical concept for monitoring system construction and design, that will ensure the highest degree of reliability and will correspond to all licensee requests on:

- component integrity

- damage mechanisms

- specific components

- chosen measurements locations

- type of measurements chain

- method of data verification and evaluation

- regulatory position

2. PRESENT STATUS IN NPP KRŠKO.

On the recent operating experience, in NPP Krško currently the following diagnostics activities are in use:

- motor operated valves diagnostics measurements

- thermography on electrical and mechanical components

- vibration monitoring of specific rotating equipment

All above mentioned diagnostics methods are used in the case of their need, there is no permanent system on line placed on some system or component.

Some of the monitoring systems were the part of the original power plant design (vibration monitoring of the main turbine-generator set, for example) and currently there is no specific need to introduce some new ones.

In the last few years, as a type of operator computerized support system was introduced, called Process Information System. It is a type of data acquisition system with few hundred of process variables on the computerized system, which allows operators or other plant personnel to follow on line chosen variables.

3. LICENSING ASPECTS OF NPP DIAGNOSTICS.

The licensing of new diagnostic system is a process which shall contain all the activities requested by regulatory authority (legislation) requests and also the requests from international codes and standards.

The following parts of licensing process are the most important:

- the impact on safety in different operational modes

- qualification of the systems with appropriate methods and procedures

- acceptance criteria for methods used in the process of qualification

- determination of the threshold values for different actions

- regulatory authority reporting requests

- affects on technical specifications, procedures, safety analysis reports and other documents

The Slovenian Nuclear Safety Administration follows the worldwide practice and experience. Introduction of the new diagnostic system will be the case of safety important modification in the situation, that the diagnostic system will be in the coincidence with some safety related function of specific safety feature.

The methodology and guidelines from the US code 10CFR 50.59 are followed in the process of safety evaluation. In the case of an unreviewed safety question an additional engineering evaluation of the diagnostic system design basis will be necessary.

Important consideration is that a change to non-safety related equipment not described in Final Safety Analysis Report can indirectly affect the capability of equipment important to safety.

Regulatory authority communicate with the licensee through asking relevant questions based on the engineering and safety evaluation of the diagnostics system to be installed at the NPP.

Examples:

- change convert of feature that was automatic to manual or vice versa

- introducing an unwanted or previously unreviewed system interaction

- change affect the seismic or environmental qualification of the system

- new electrical loads introduced

- maintenance, testing, personnel qualification

4. SELECTION OF THE SYSTEMS (COMPONENTS) TO BE MONITORED

The decision of the nuclear power plant intention to introduce new diagnostics monitoring system is usually based on the indicators from:

- transients, damage, failures

- operational history of the components important to safety

- boundary conditions obtained from stress and fatigue analysis

- experience from inspection

- life extension (ageing phenomena)

The monitoring system is basically global or local and each shall be continuous or not continuous.

It is important to properly locate the measuring points, choose the data type to be monitored and which measurement system will be in use, on the other hand, the decision should be made on damage mechanism that will be observed in which operation modes.

5. CONCLUSIONS

The decision for introduction of the specific monitoring system into nuclear power plant is a complex task. First of all, all the operational events and power plant performance indicators should be the basis for such decision. Using the

international experience and databases is necessary. The computerized diagnostics monitoring systems should be considered as a helpful operators support tool.
Such systems aid them to diagnose the abnormal behavior of certain parameter before the normal alarm limits are reached. The monitoring and diagnostics function is to alert the operators to the safety status of the plant.

Applications of the monitoring and diagnostics systems are already in operation or under development in many countries. They are sometimes integrated with other operator support system .
Slovenian Nuclear Safety Administration strongly supports all activities for achieving higher level of nuclear safety through introducing diagnostics systems.

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INTERNATIONAL ATOMIC ENERGY AGENCY

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„Nuclear Power Plant Diagnostics -
Safety Aspects and Licensing“

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APPLICATION OF MODEL-BASED AND KNOWLEDGE-BASED MEASURING METHODS AS ANALYTICAL REDUNDANCY

by

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Abstract

The safe operation of nuclear power plants requires the application of modern and intelligent methods of signal processing for the normal operation as well as for the management of accident conditions. Such modern and intelligent methods are model-based and knowledge-based ones being founded on analytical knowledge (mathematical models) as well as experiences (fuzzy information). In addition to the existing hardware redundancies analytical redundancies will be established with the help of these modern methods. These analytical redundancies support the operating staff during the decision-making.
The design of a hybrid model-based and knowledge-based measuring method will be demonstrated by the example of a fuzzy-supported observer. Within the fuzzy-supported observer a classical linear observer is connected with a fuzzy-supported adaptation of the model matrices of the observer model.
This application is realized for the estimation of the non-measurable variables as steam content and mixture level within pressure vessels with water-steam mixture during accidental depressurizations. For this example the existing non-linearities will be classified and the verification of the model will be explained. The advantages of the hybrid method in comparison to the classical model-based measuring methods will be demonstrated by the results of estimation.
The consideration of the parameters which have an important influence on the non-linearities requires the inclusion of high-dimensional structures of fuzzy logic within the model-based measuring methods. Therefore methods will be presented which allow the conversion of these high-dimensional structures to two-dimensional structures of fuzzy logic. As an efficient solution of this problem a method based on cascaded fuzzy controllers will be presented.

1. Application of analytical redundancies

1.1 Introduction

The safe operation of nuclear power plants requires the application of analytical redundancies in addition to the existing hardware redundancies for the normal operation as well as for the management of accident conditions. Analytical redundancies will be established with the help of modern and intelligent methods of signal processing. This modern and intelligent methods will be used for the support of the operating staff during the decision-making.

The spectrum of application of analytical redundancies includes:
- the monitoring of the process state (estimation of non-measurable state variables)
- the diagnosis of the plant operational function (failure detection)
- the control (application as controller)
- the limitation (application within a limitation system).

The knowledge basis is given by:
- knowledge (generated by experiments),
- analytical knowledge and simulation results,
- experiences.

The monitoring of the process state (estimation of non-measurable variables) with the help of model-based and knowledge-based measuring methods will be emphazised in the paper.

1.2 Conventional model-based measuring methods as analytical redundancy

Most applications of analytical redundancies were realized with the help of conventional Model-based Measuring Methods (MMM) like observer or Kalman Estimator.

The Model-based Measuring Method uses measured input and output variables of the process. It consists of the mathematical model of the process and the feed back of the error between the measured and calculated output variable beyond a correction matrix. The estimated and real output variables approach by an appropriate dimensioning of this matrix (Figure 1).
The quality of the estimated state variables depends on the quality of the state space model. The classification of MMM is depending on the mathematical model which is used (linear model, and non-linear model).

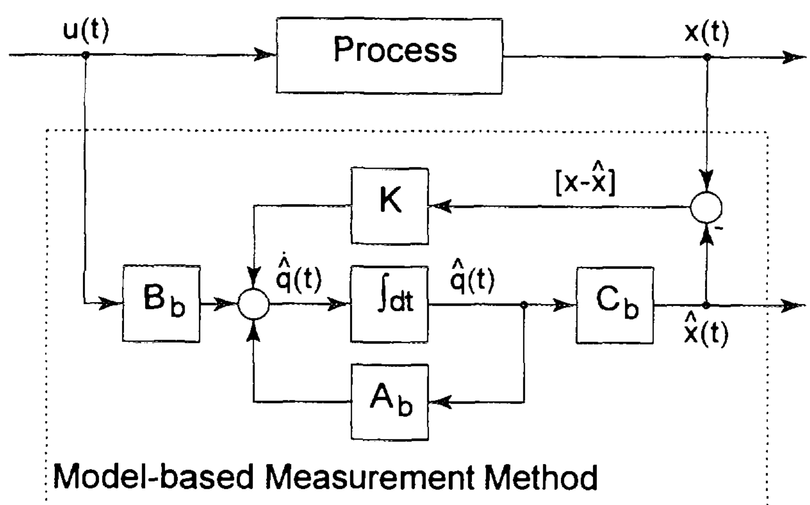

Figure 1: Structure of a linear Model-based Measuring Method (Observer)

Variables of Process		Variables of MMM	
$\mathbf{u}(t)$ -	input variables	$\hat{\mathbf{q}}(t)$ -	estimated state variables
$\mathbf{x}(t)$ -	output variables	$\hat{\mathbf{x}}(t)$ -	estimated output variables

The design methods of linear MMM like Luenberger Observer are well known and are characterized by general validity and simple design algorithms. The disadvantage of the linear MMM are the limited range of validity (exact estimation around the operating point only).
An improvement of the quality of estimation can be realized by non-linear MMM based on non-linear model statements. Non-linear observers require special design algorithms (without general validity) and were often applied for special processes only.

For the description of non-linearities a new type of MMM was developed using also algorithms of fuzzy logic.

1.3 Fuzzy-supported Observer

The developed idea of the application of fuzzy logic in connection with model-based measurement methods is the following:

> *Combination of a linear model-based measurement method (using the advantages of simple design, global ranges of stability and observability) with a fuzzy adaptation in order to expand the range of validity of the linear model statement.*

The developed method was realized in form of a fuzzy-supported observer consisting of a linear observer and a fuzzy-supported adaptation of the matrices of the state space model. The observer as well as the fuzzy adaptation is feeded by the input and output variables of the process (Figure 2).

The fuzzy-supported observer was developed with the help of the simulation tool DynStar with Fuzzy Shell which allows the simulation of MMM as well as fuzzy algorithms.

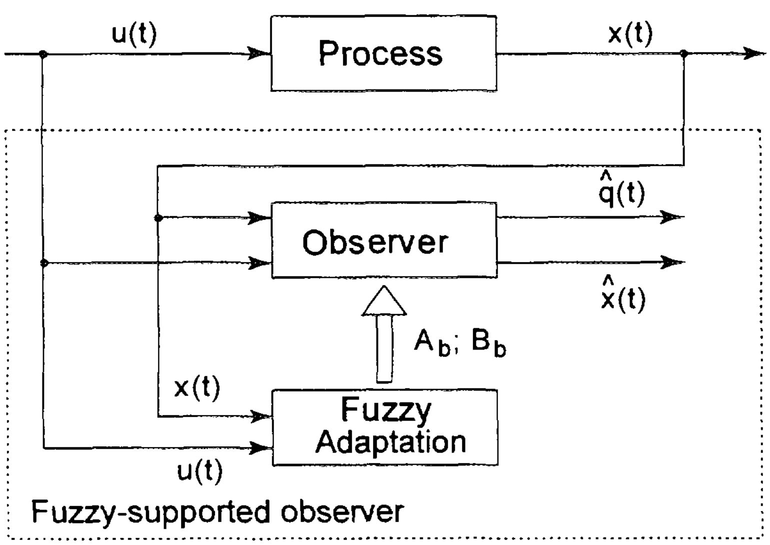

Figure 2: Structure of the fuzzy-supported observer

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2. Determination of non-measurable variables in pressure vessels

2.1 Description of the process

The application of MMM was realized for the estimation of the non-measurable variable mixture level. The mixture level is a safety-related variable for pressure vessels with water-steam mixture like pressurizer, steam generator and reactor pressure vessels. The monitoring of the mixture level is very important during accidental conditions (depressurizations as a result of a leak). The Figure 3 shows the different levels within a pressure vessel.

The following levels within the pressure vessel can be classified:

⇒ hc - collapsed level of the pressure vessel
⇒ hc_{bF} - collapsed level between the fittings of the narrow range measuring system
⇒ hc_i - collapsed level indicated by the measuring system
⇒ hm - mixture level of the pressure vessel
⇒ hc_{lF} - collapsed level below the lower fitting
(characterizes the steam content in the volume below the lower fitting)

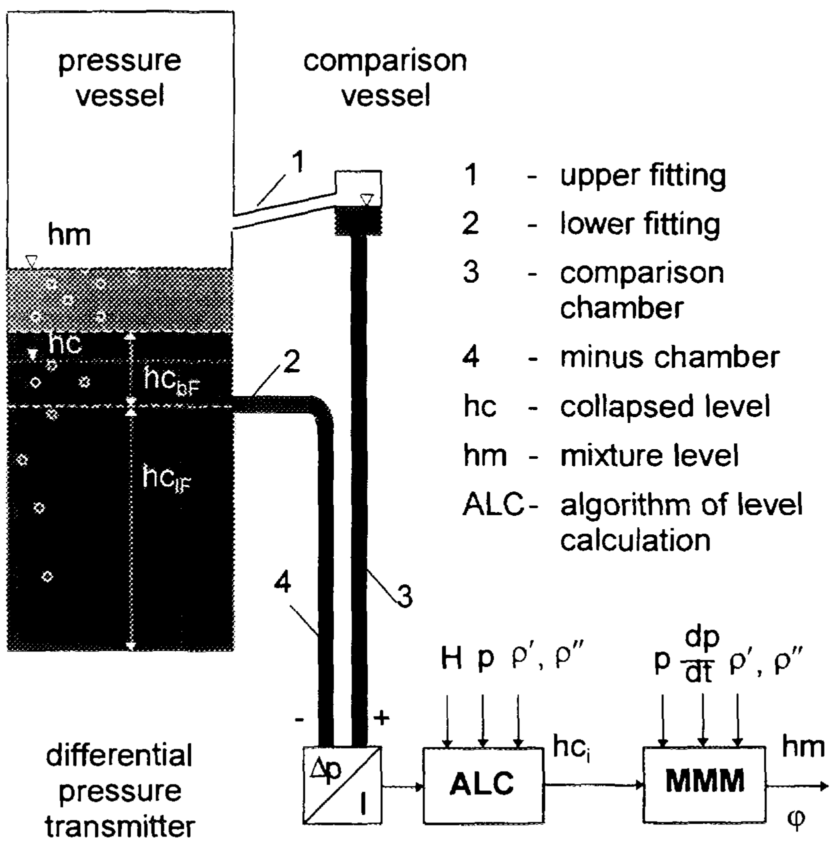

Figure 3: Pressure vessel in connection with a hydrostatic level measuring system

The global aim is the estimation of the non-measurable mixture level hm during negative pressure gradients $hm = f\left(p,\ dp/dt,\ hc_{bF},\ hc_{lF}\right)$. In the first step it was necessary to estimate

the collapsed level below the lower fitting hc_{lF} which is an input variable for the calculation of the mixture level.
For the realization of the estimation of the non-measurable collapsed level below the lower fitting hc_{lF} the measurable process parameters

- pressure p (as a result the pressure gradient dp/dt)
- collapsed level between the fittings of the narrow range measuring system hc_{bF}

are used.

2.2 Non-linearities and fuzziness of the process

If only the influence of the depressurization is considered the non-linear model statement for the description of the collapsed level below and between the fittings is characterized by the following dependences:

$$hc_{lF} = f\left(p,\ dp/dt\right) \tag{1}$$

$$hc_{bF} = f\left(p,\ dp/dt,\ hc_{lF}\right) \tag{2}$$

The non-linearities of the presented process can be classified as follows:

⇒ <u>algorithmic non-linearity</u>

non-linear terms of the state equations in form of products of state variables and input variables: $\hat{\mathbf{q}}(t) \cdot \mathbf{u}(t) \mathrel{\hat{=}} hc \cdot \dfrac{dp}{dt}$

⇒ <u>non-linearity depending on thermodynamic properties</u>

density and enthalpy depending on pressure: $\rho(p),\ h(p)$

⇒ <u>non-linearity depending on process state</u>

- steam temperature above the saturation temperature: $\vartheta_{St} \geq \vartheta_{Sa}$
- disturbances: feed, bleed, spray

The fuzziness of the described process can be characterized by:

⇒	description of phase separation:	velocity of the steam and water phase
⇒	description of heat transfer conditions:	heat transfer coefficient

2.3 Linear MMM for the estimation of the collapsed level Δhc_{lF}

For the estimation of the non-measurable state variable collapsed level below the lower fitting hc_{lF} a state space model for a linear observer was developed. The model is based on a nodalization of the volume of water-steam mixture. The simpliest nodalization is the subdivision in two nodes:

- the zone below the lower fitting of the measuring system
- the zone of water-steam mixture between the fittings of the measuring system.

As a result of this simplified nodalization a state space model of second order was generated which is characterized by the following variables:

⇒ input variable: pressure gradient dp/dt

⇒ state variables: deviation of collapsed level below the lower fitting of the narrow range measuring system Δhc_{lF}

deviation of collapsed level between the fittings of the narrow range measuring system Δhc_{bF}

⇒ output variable: deviation of collapsed level between the fittings of the narrow range measuring system Δhc_{bF}

The state equations are:

$$\begin{bmatrix} \frac{d\,\Delta hc_{lF}}{dt} \\ \frac{d\,\Delta hc_{bF}}{dt} \end{bmatrix} = \begin{bmatrix} a_{11} & a_{12} \\ a_{21} & a_{22} \end{bmatrix} \cdot \begin{bmatrix} \Delta hc_{lF} \\ \Delta hc_{bF} \end{bmatrix} + \begin{bmatrix} b_1 \\ b_2 \end{bmatrix} \cdot \begin{bmatrix} \frac{d\Delta p}{dt} \end{bmatrix} \tag{3}$$

$$\Delta hc_{bF} = \begin{bmatrix} 0 & 1 \end{bmatrix} \cdot \begin{bmatrix} \Delta hc_{lF} \\ \Delta hc_{bF} \end{bmatrix} \tag{4}$$

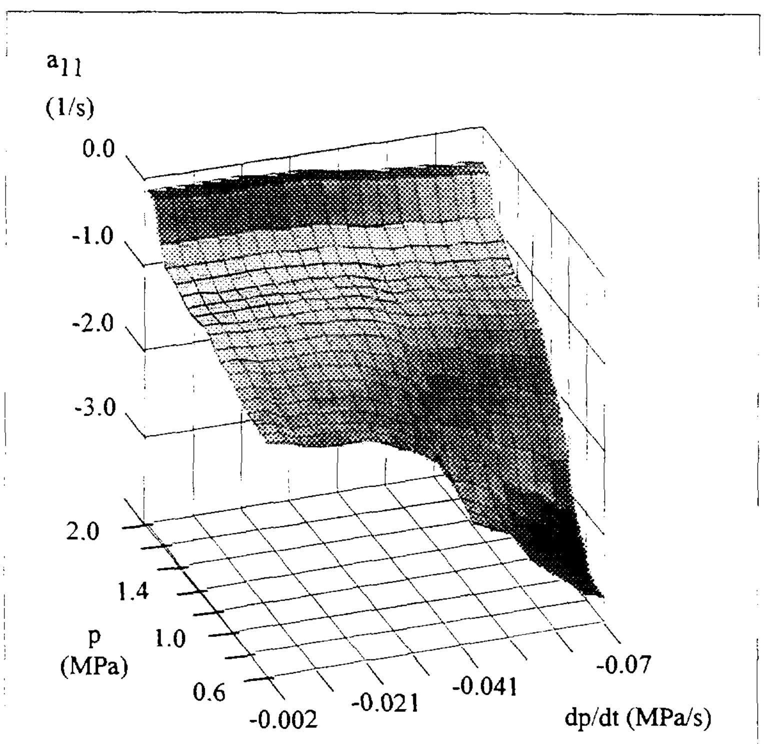

Diagram (x = p, y = dp/dt, z = a_{11})

Figure 4: Non-linear characteristic field of the matrix element a_{11} of the system matrix depending on the pressure p and the pressure gradient dp/dt

The model matrices are characterized by the following dependences:

⇒ elements $a_{11}, \ldots, a_{22}$ of the system matrix depending on pressure p and pressure gradient dp/dt

⇒ elements b_1, b_2 of the input matrix depending on pressure p

Figure 4 demonstrates the influence of the pressure and the pressure gradient on the matrix element a_{11} of the system matrix.

The non-linear character of the elements of the model matrices requires an adaptation of this model matrices depending on the changing of the process state. The adaptation was realized on the basis of knowledge which was generated by experiments and simulations.

3. *Generation of the knowledge basis*

For the investigations it was necessary to generate a data basis with the following aims:

- analysis of the non-linearities which must be considered in the analytical redundancy
- verification of the state space model
- verification of the designed model-based measurement method
- generation of the knowledge for the knowledge-based algorithm.

The realization was carried out with the help of test facilities as well as the simulation code ATHLET, which is a complex simulation code for thermofluid-dynamic processes.

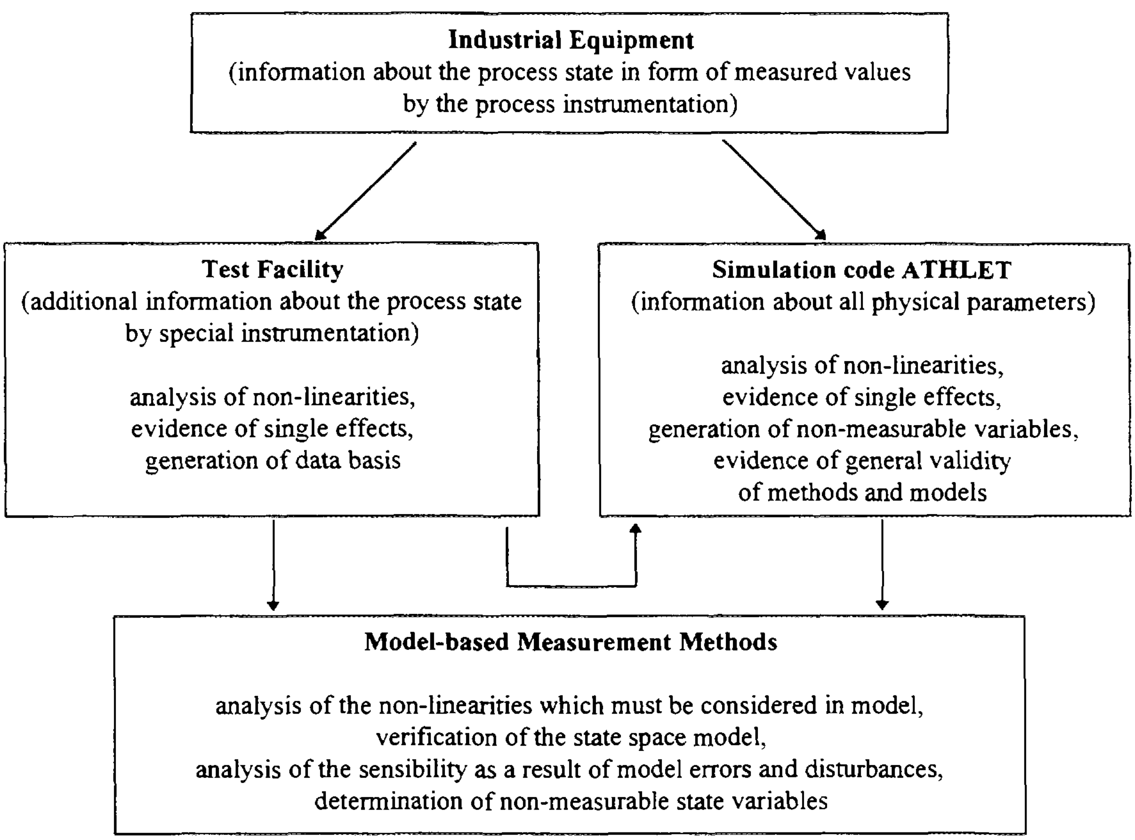

Figure 5: Verification of model-based measurement methods

Blow down experiments were carried out on the pressurizer test facility of the IPM equipped with additional measuring systems. The experiments were post calculated with the help of the simulation code ATHLET.The identity between the measured experimental datas and the calculated parameters by ATHLET resulted in the conclusion that the ATHLET-data set describes the real process in the right way. In the next stage the simulation code serves as a compensation of the real process with the advantage that all parameters (measurable and non-measurable) were provided by ATHLET. So the possibility was given to verify the model as well as the Model-based Measuring Method (Figure 5).

4. Design of the fuzzy-supported observer

Within the developed fuzzy-supported observer each element of model matrices which must be adapted was calculated by a fuzzy controller. On the basis of the chosen example the aim is the adaptation of the elements a_{11}, a_{12}, a_{21}, a_{22} of the system matrix $\mathbf{A}_b$ and b_1, b_2 of the input matrix $\mathbf{B}_b$ depending on the change of pressure and pressure gradient.
The design of the fuzzy-supported observer can be simplified if the amplification gain is designed for the complete range of parameter changing (range of changing of the elements of the matrix $\mathbf{A}_b$). In this case the adaptation of the elements of the proportional feedback $\mathbf{K}$ is not necessary.

The practical applications often require more than two input parameters for the fuzzy controller which resulted in high-dimensional structures of fuzzy logic. In this case a reduction of these high-dimensional structures to two-dimensional structures is advantageous.

Fuzzy Sets

The range of parameter changing was subdivided into representative parts. As a criterion of the choice of the partial ranges the changing rate of the pressure was used. For the above mentioned example the range of pressure was described by seven characteristic points. Each point of pressure was represented by a value of the linguistic variable (Table 1).

Representative	**value of the linguistic variable**
p_{AP}=0,6 MPa	Very Small (VS)
p_{AP}=1,0 MPa	Small (S)
p_{AP}=1,2 MPa	Medium Small (MS)
p_{AP}=1,5 MPa	Medium (M)
p_{AP}=1,6 MPa	Medium Great (MG)
p_{AP}=1,8 MPa	Great (G)
p_{AP}=2,0 MPa	Very Great (VG)

Table 1: Values of the linguistic variable pressure

In Figure 6 the distribution of the fuzzy sets characterized by triangle-type membership functions is represented for the linguistic variable pressure.

For the pressure gradient and the elements of the model matrices values of linguistic variables were defined in the same way.

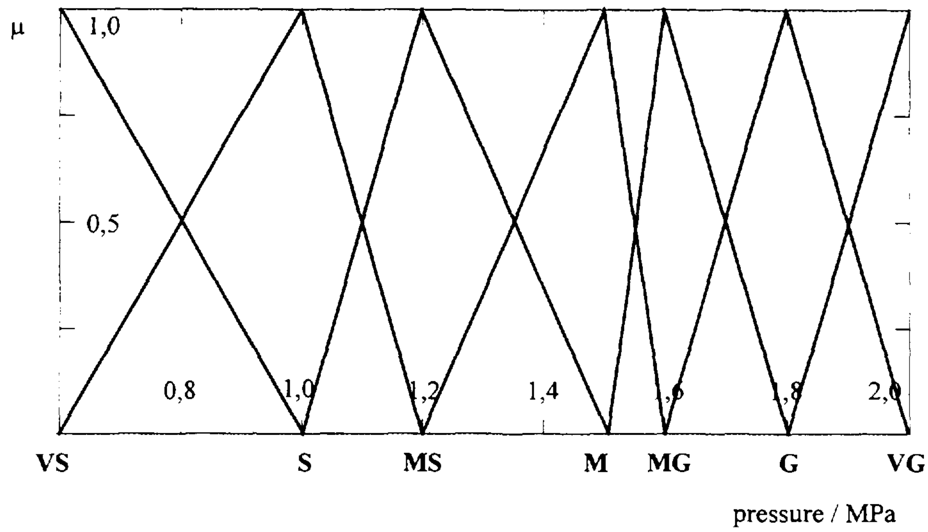

Figure 6: Fuzzy sets of pressure

Rules

The relationships between the linguistic values are described in form of IF - THEN - rules. For the matrix element a_{1_1} the following rule can be generated:

IF ' *pressure* ' = ' Very Small ' AND ' *pressure gradient* ' = ' Very Great Negative '

THEN ' *matrix element* a_{1_1} ' = ' Very Great ' (5)

Table 2 presents the matrix of rules generated for the matrix element a_{11} depending on the linguistic variables pressure and pressure gradient.

dp/dt \ p	VGN	GN	MGN	MN	MSN	SN	VSN
VS	VG	VG	VG	VG	VG	VG	G
S	VG	VG	VG	VG	G	S	S
MS	VG	VG	VG	G	S	S	S
M	VG	G	S	S	S	S	S
MG	G	S	S	S	S	S	S
G	S	S	S	S	S	S	S
VG	VS	VS	VS	VS	VS	VS	VS

Table 2: Rules of the matrix element a_{11} of the system matrix

In the same way the rules of the other elements of the matrices were generated.
With the help of these algorithms of fuzzy logic the description of the non-linear behaviour of the matrix elements was realizable.

5. Results of the state estimation

The results of the state estimation were demonstrated on a blow down experiment which was realized at the pressurizer test facility at the IPM and postcalculated with the help of the ATHLET-Code. The pressure reduction was realized from 1.8 MPa to 0.9 MPa in the period $t = 70 \ldots 250\,\mathrm{s}$.

Input variables for the model-based and knowledge-based methods

The Figures 7 and 8 show the response characteristics of the pressure and the pressure gradient during the experiment.

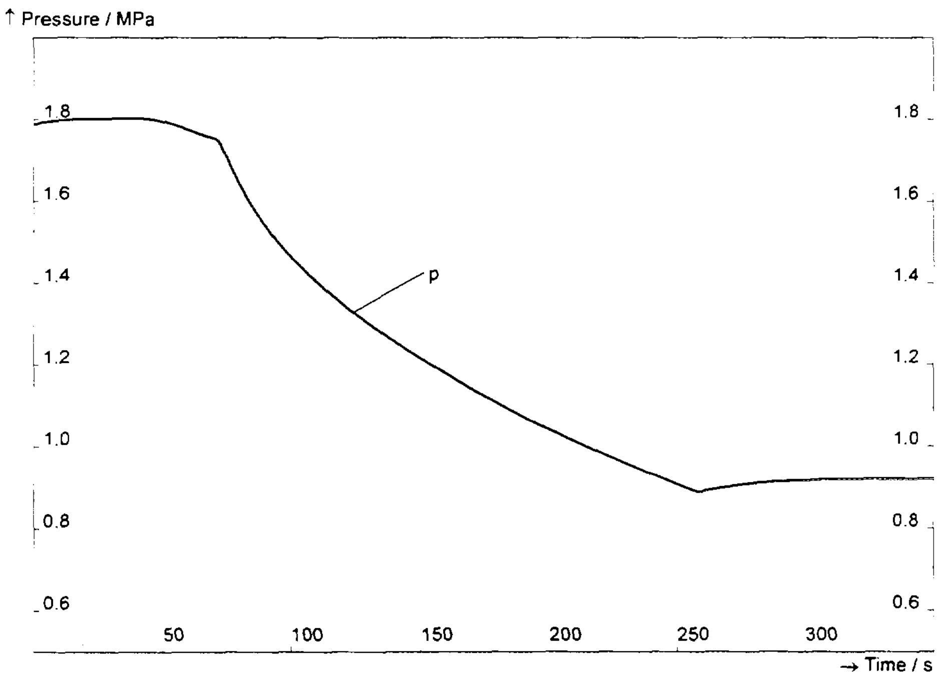

Figure 7: Response characteristic of the pressure during a blow down experiment

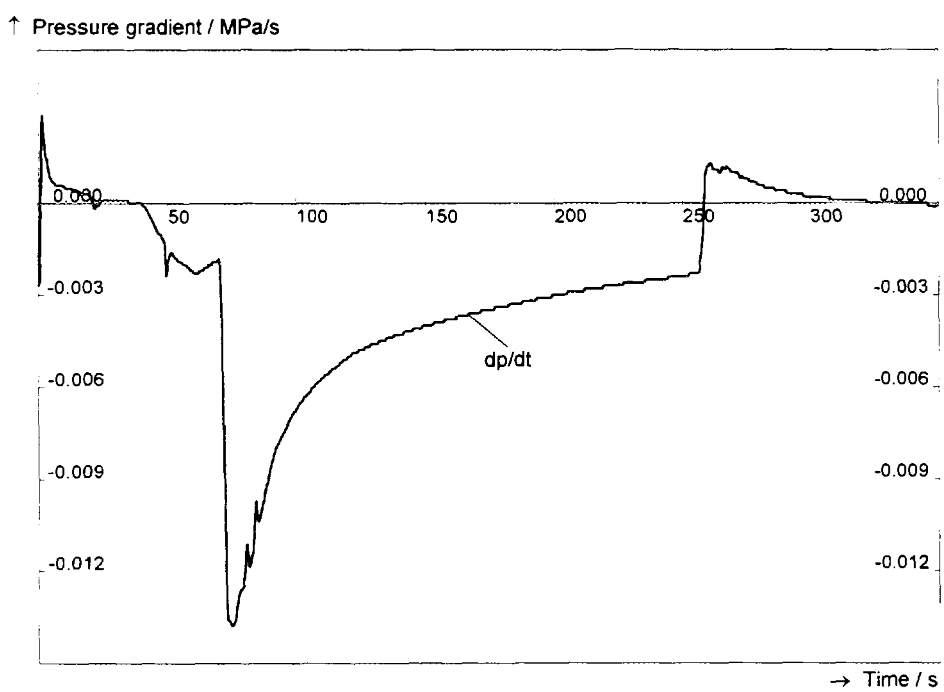

Figure 8: Response characteristic of the pressure gradient during a blow down experiment

P5

The following kinds of analytical redundancy were compared with the process:
- linearized parallel model (linear state space model)
- conventional observer (based on the linear state space model)
- fuzzy-supported observer (based on the linear model in connection with adaptation of model matrices).

Results of state estimation of the measurable state variable

The measurable state variable deviation of collapsed level between the fittings Δhc_{bF} was exactly estimated by all kinds of observers. The produced identity between the measurable output variable of the process (ATHLET) and the calculated output variable of the observer is the result of the feedback of the estimation error beyond the correction gain (Figure 9).

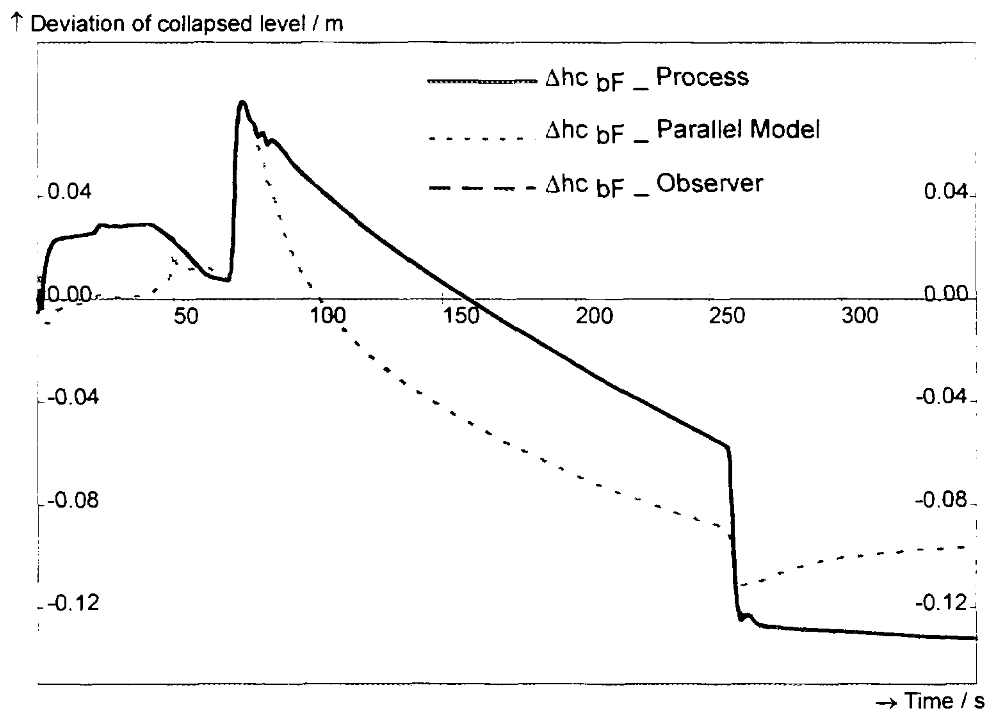

Figure 9: Response characteristic of the measurable state variable Δhc_{bF} of Process, linear Parallel Model and the Observers (conventional Observer, fuzzy-supported Observer)

The calculated state variable of the linear parallel model is characterized by great differences during the depressurization in comparison to the process as a result of the influence of the non-linearities. A good reconstruction was realized only around the operating point of the linear model characterized by the initial pressure before the depressurization (t ≈ 70 s).

Results of state estimation of the non-measurable state variable

The non-measurable state variable deviation of collapsed level below the lower fitting Δhc_{lF} was estimated with different quality by the different kinds of observers (Figure 10).
The quality of reconstruction can be improved by the application of the conventional observer in comparison to the parallel model. As a result of the influence of the non-linearities the estimation is characterized by a non-negligible estimation error at the end of the blow down (t ≈ 250 s).

The best estimation of the non-measurable state variable was realized by the fuzzy-supported observer.

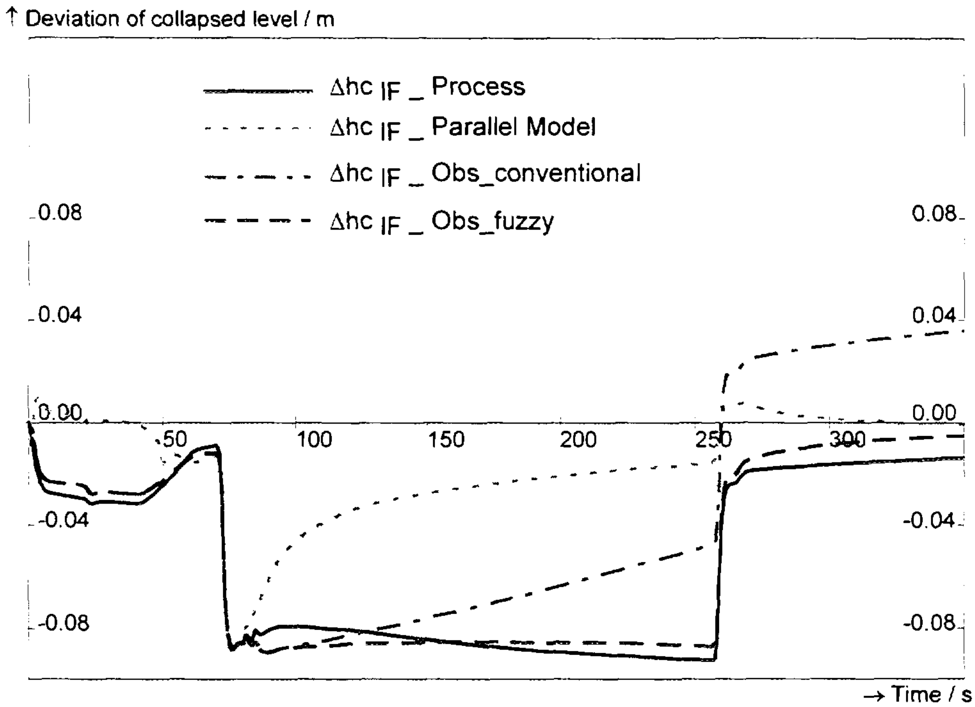

Figure 10: Response characteristic of the non-measurable state variable Δhc_{IF} of Process, linear Parallel Model, conventional Observer and Fuzzy-supported Observer

The mixture level will be calculated on the basis of the estimated state variables.

6. Cascading high-dimensional fuzzy controllers

The implementation of fuzzy logic in Model-based Measuring Methods for the description of strong non-linear and complex processes often requires the consideration of many input variables for the fuzzy controller. Generally that leads to a multi-dimensional structure of the controller.

The amount of the individual rules increases with the number of the input variables and the corresponding linguistic values complicating a real time processing. The formulation of the rules for more than two input variables is complicated and their representation not clear. In this case, it is difficult to set up the rules from the experience knowledge about the dynamic behaviour of the system. The parameterization and optimization of the fuzzy controller is hardly possible because of the many degrees of freedom of the fuzzy controller.
A possibility of the simple implementation of the experience knowledge despite the high amount of the input variables and the real time capability is the structure optimization.

In [2] was already pointed out that a fundamental problem of the parameterization and optimization of the fuzzy controller is the amount of the individual rules.
To reduce the number of these rules it was proposed to use cascaded controller structures.

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In addition, it is to prove that the associative law is valid for the basis-rule:

$$X1 \cap X2 \cap X3 = (X1 \cap X2) \cap X3 \quad (6)$$

With n input variables result n-1 two-dimensional fuzzy controllers and n-2 virtual linguistic variables which do not have to have an absolutely physical meaning.

The cascaded controller structure has fewer rules than a multi-dimensional controller. For example, with 3 linguistic values (Fuzzy - Sets) per input variable result 81 rules for a controller with 4 input variables, while the cascaded structure with 3 (=4-1) two-dimensional fuzzy controllers and 2 (=4-2) virtual linguistic variables contains only just 27 rules.

The equation (6) is described by the following signal flow diagrams (Figure 11).

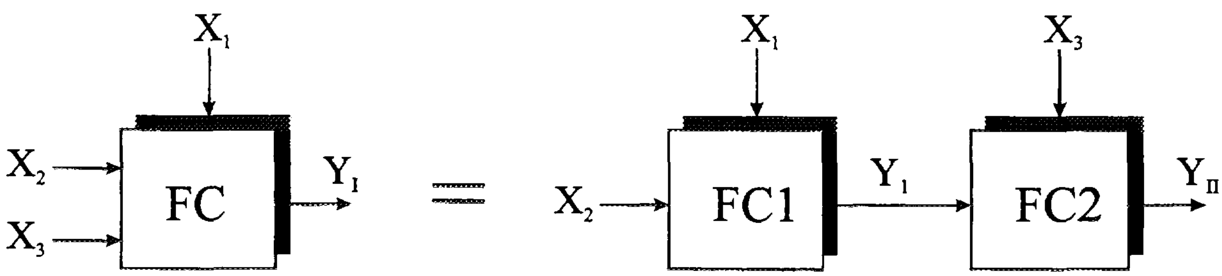

Figure 11: Cascading of a 3-dimensional fuzzy controller

First, it is assumed for the further investigations that equation (6) is fulfilled if, for a given definition of the number of the sets for X and Y, the linguistic values for Y_I and Y_{II} agree (Figure 11).

To carry out this proof, the following consideration must be placed before:

For high dimensional fuzzy controllers it is expedient to define
dominating input variables (X1 , X2)
and
non-dominating input variables (X3).

With the dominating input variables the base shape of the characteristic field is formed and with the non-dominating input variable results a deformation of the base characteristic field (Figure 12).

a) Characteristic curve from the base characteristic field (X_2 = Const.)

b), c) by X_3 caused deformation of the characteristic field

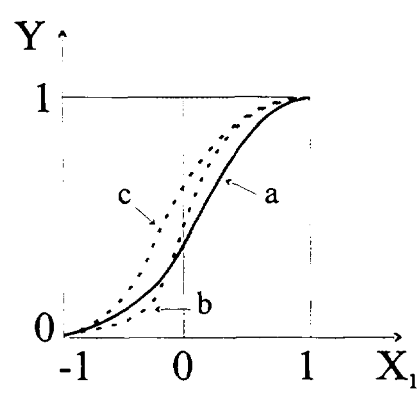

Figure 12: Examples for possible deformations of the base characteristic field

For example, the deformation in Table 3 is thereby realized that for

X3 = L a displacement in L - direction

X3 = H a displacement in H - direction

follows in the result matrix Y.

One can denote such an establishment also as an adaptation rule. The adaptation rule applied for Table 3 is implemented in the cascaded structure in the FC2-matrix (Table 4). The rule matrix FC1 corresponds to the matrix in Table 3 for X3 = N.
The marked errors in Table 3 of the rule matrix show which individual rules are different. With an increasing number of the input variables and fuzzy sets, the number of the mismatches increases.
By introducing a larger number of sets (5) for the virtual variable Y1, this lack can be removed. With the given rule matrix in Table 5 for the cascaded controller, the complete correspondence with the complete rule matrix in Table 3 is obtained.

For monotonous symmetrical characteristic fields it could be shown that the mismatches do not or only slightly have an effect on the numerical values for Y. Further investigations are necessary for asymmetrical characteristic fields. As a result, rules should be derived for the structure optimization of multidimensional fuzzy controllers.

Y_I				X_1		
				L	N	H
X_3	L	X_2	L	L	L	L
			N	L	L	N
			H	L	N	H
	N		L	L	L	N
			N	L	N	H
			H	N	H	H
	H		L	[redacted]	N	H
			N	N	H	H
			H	H	H	H

Table 3: Complete rule matrix for the three-dimensional fuzzy controller FC (Figure 11)

Y_1		X_1		
		L	N	H
X_2	L	L	L	N
	N	L	N	H
	H	N	H	H

a) FC1

Y_{II}		X_3		
		L	N	H
Y_1	L	L	L	[redacted]
	N	L	N	H
	H	N	H	H

b) FC2

Table 4: Complete rule matrix for the cascaded fuzzy controller
(FC1 and FC2 in Figure 11)

Y_I		X_1 L	X_1 N	X_1 H
X_2	L	VL	L	N
	N	L	N	H
	H	N	H	VH

a) FC1

Y_{II}		X_3 L	X_3 N	X_3 H
Y_I	VL	L	L	L
	L	L	L	N
	N	L	N	H
	H	N	H	H
	VH	H	H	H

b) FC2

Table 5: Rule matrix for the 3-dimensional cascaded fuzzy controller with 5 sets for Y_I

VL very low   L low   N normal   H high   VH very high

7. Conclusions

The monitoring and diagnosis of the actual process state especially in the case of arising accidental conditions as well as during accidents require the application of intelligent methods and high performance algorithms of signal processing (e.g. Model-based Measuring Methods).
The advantages of such methods of signal processing are:
- more information about the actual process state,
- realizable in existing control systems,
- real time application.

The determination of non-measurable process parameters is realizable by the application of Model-based Measuring Methods (MMM). The combination of conventional MMM with knowledge-based algorithms (fuzzy logic) improves the quality of the state estimation.
The advantages of such hybrid methods are:
- description of complex multi-variable systems,
- description of complex non-linearities,
- description of the fuzziness of systems.

The generation of the knowledge requires specific investigations of the process (experiences, experiments, complex simulations).

A fundamental problem of parameterization and optimization of fuzzy controllers is the high number of the rules. To reduce the number of these rules, a controller structure was proposed. As the associative law is valid for monotonous symmetrical characteristic fields, the multi-dimensional fuzzy controller structure can be transformed in a cascaded one. The dominating input variables determine the base shape of the characteristic field, while the non-dominating input variables generate the necessary deformation of the base characteristic field.

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POTENTIAL OF ACOUSTIC MONITORING FOR SAFETY ASSESSMENT OF PRIMARY SYSTEM

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Abstract—**Safety assessment of the primary system and its components with respect to their mechanical integrity is increasingly supported by acoustic signature analysis during power operation of the plants. Acoustic signals of Loose Parts Monitoring System sensors are continously monitored by dedicated digital systems for signal bursts associated with metallic impacts. Several years of ISTec/GRS experience and the practical use of its digital systems MEDEA and RAMSES have shown that acoustic monitoring is very successful for detecting component failures at an early stage. Advanced powerful tools for classification and acoustic evaluation of burst signals have recently been realized. The paper presents diagnosis experiences of BWR's and PWR's safety assessment.**

1. Introduction

The classical concept to assess the integrity of components by stringent quality control measures during design, construction, and commissioning, combined with repetitive, discontinuous tests, and inspections during plant operation, is more and more supplemented by integral on-line status assessment concepts. Efforts to advance the safety of nuclear power plants (NPP) using modern computer technology have led to powerful new solutions for more automated fault diagnosis systems.

One of the main integral on-line diagnosis methods in NPPs is (besides vibration and process noise signature monitoring) acoustic monitoring by means of so-called loose part monitoring systems (LPMS). They are capable to detect component failures at an early stage /1-4/. Basic work has been sponsored by the German Federal Ministry for Environment, Nature Conservation and Nuclear Safety, which is highly acknowledged.

In the German RSK-guidelines /5/ the use of adequate measures is required in order

- to detect free and captive loose parts within the pressure retaining boundary and
- to localize loose parts as well as possible.

Piezoelectric accelerometers working in the acoustic frequency range have been found to be very effective for this task.

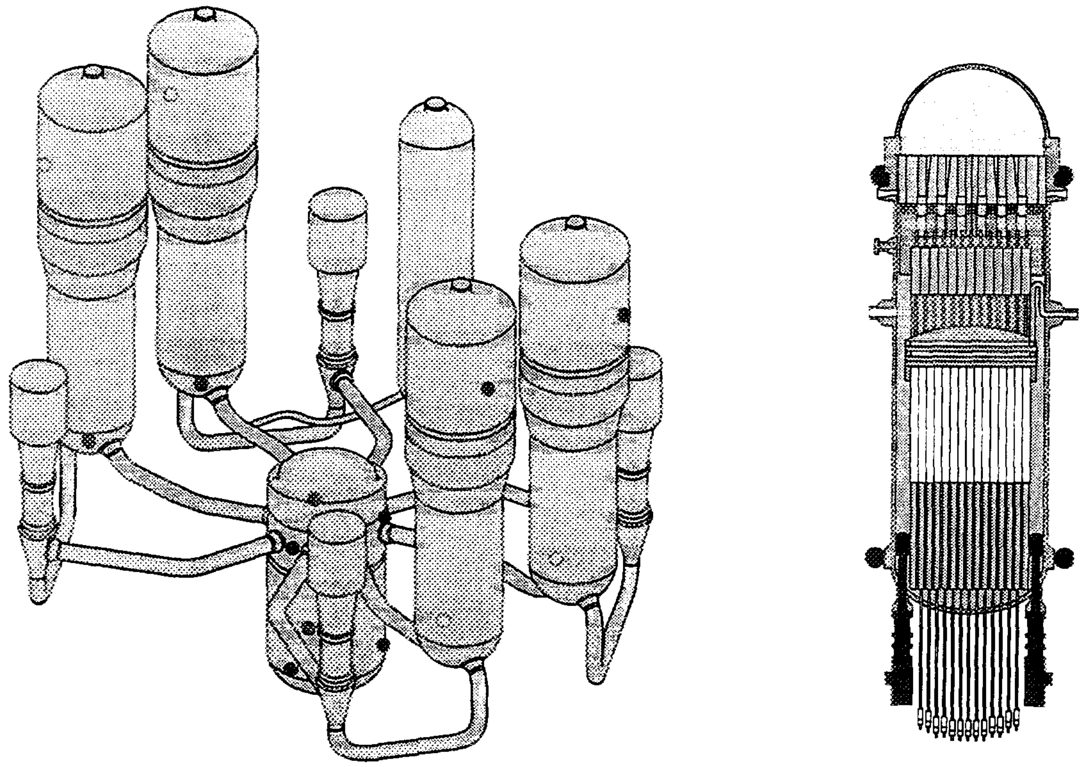

Fig. 1: Sensor positions of loose parts monitoring systems for PWR and BWR

Fig. 1 shows the sensor positions of LPMS for a 1300 MW Pressurized Water Reactor and a 1300 MWe Boiling Water Reactor. The sensors are mounted as near as possible to the surface of the monitored structure. They are positioned in different levels at the reactor pressure vessel and steam generator. ISTec/GRS has collected extensive acoustical data sets from reactor internals and primary circuit components.

2. Acoustic signal analysis systems

Loose parts monitoring systems are installed in all German NPPs. The systems work on-line and monitor the reactor coolant system for detached or loose internal or foreign parts. The technical status of the systems is different according to the different period of their construction. They have been developed now to a satisfactory status. In recent years more and more utilities have replaced old analog systems by modern digital systems. Current investigations concentrate on the improvement of on-line diagnosis methods and trending of components status. ISTec has recently developed two software modules for classification and acoustic evaluation of LPMS signatures /6,7/.

Basic requirement for monitoring of active and passive components at Light Water Reactor (LWR) primary systems is the qualified analysis and detailed burst data interpretation of acoustic signals. A fast digital off-line burst processing system (MEDEA-system) has been realized in the ISTec laboratory. Ist major components are a 16 channels transient recorder, a UNIX- workstation, fast data storage facilities, interactive analysis software and a burst data base, see fig. 2. Acoustic signatures from measurements and analyses of 13 German LWRs with more than 11000 bursts are available. In a burst pattern data base they are characterized by the event type, measurement and signal parameters.

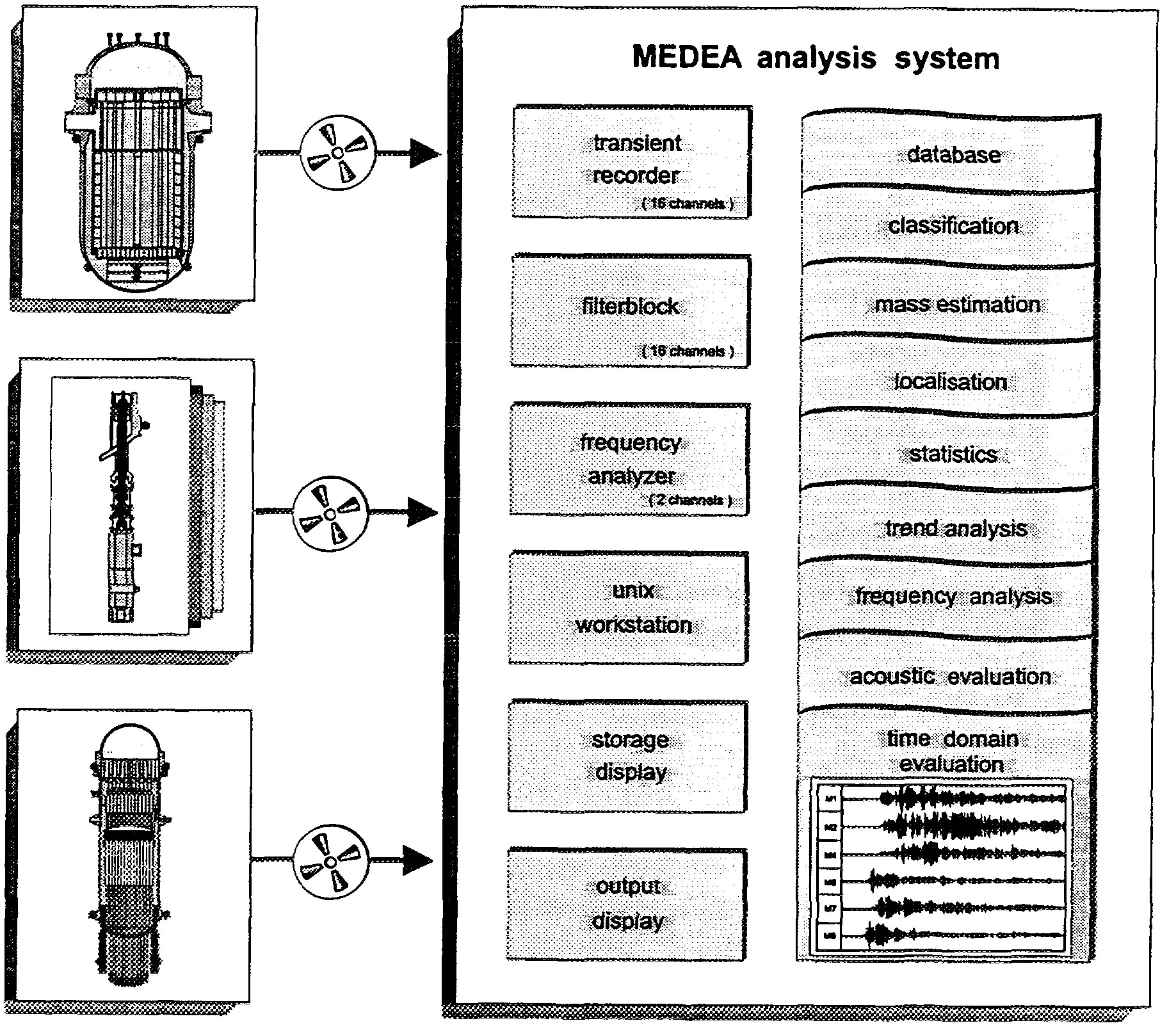

Fig. 2: MEDEA-system for off-line acoustic analysis

For on-site digital signal recording and diagnosis support to the plant a Remote Acoustic Monitoring and Signal Evaluation System (PC-RAMSES) has been developed by ISTec. It consists of six PC-based transient recorder storage modules, a graphic display and a modem. Software packages have been established for storage of the data to the on-site computer after event detection, for remote controlled settings of the transient storage modules via the telephone line and for data transfer of qualified signal patterns to the ISTec laboratory.

3. Safety Assessment by Acoustic Analysis

The common idea of safety assessment of reactor primary system by acoustic analysis is to use acoustic signatures for an integral component status assessment. Surveillance task is always a prevention of failures or a specific precaution against damage. A survey of successfully applied acoustic signal analysis methods is presented in fig. 3 as a reference taken from the practical use of the methods at the plants by ISTec/GRS. It demonstrates which status information has been gained by acoustic signal analyses, which sensor positions are especially suited for the specified surveillance task and which verified evaluation methods have been used for actual occurrences.

Surveillance Task	RPV sensor	SG sensor	SG add-on sensor	Reference
Surveillance of reactor vessel lower plenum and SG inlet plenum for free loose parts	●	●		burst analysis / strip chart record / acoustic evaluation
Surveillance of reactor vessel internals and SG internals for captive loose parts	●	●		burst analysis / localization / time interval analysis, acoustic evaluation
Surveillance of feedwater ring integrity (BWR + PWR)	●	●	●	pulse sequence analysis, tube-bundle vibration analysis
Recognition of impact events in the SG U-tube area		●	●	localization, pulse sequence analysis, acoustic evaluation
Irregularities at diffuser plate of main coolant pumps	●	●		pulse form analysis, acoustic evaluation
Loose screw connections at internal axial pump	●			pulse sequence analysis, acoustic evaluation
Core barrel monitoring for extensive screw failure	●	●		pulse sequence analysis, correlation with neutron flux
Surveillance of feedwater check valves (bearing and seat)		●	●	burst form analysis, acoustic evaluation
Recognition of impact events in heat removal system	●	●		burst analysis, correlation with operational measures
Surveillance of mechanical integrity of fuel elements during loading/unloading	●			correlation of acoustic signals with hoist force indications

Fig. 3: Possible status informations of primary system components gained by acoustic analysis

Acoustic monitoring systems contain an alarm processing unit with absolute and/or variable (floating) thresholds. Due to the high sensitivity of acoustic monitoring, the detection potential for impact occurrences is comparatively high. Low energetic and minor relevant events are indicated and could be seen as precursors of real failures. Too frequent unnecessary alarms can reduce the confidence to this monitoring technique and should be avoided by appropriate evaluation. On the other hand, they are of great value as status indicators, if they can be used for safety assessment of primary system components. In the following, experiences of status evaluation for safety assessment is presented.

4. Experiences of Status Evaluation for Safety Assessment

Emphasis of current activities on acoustic monitoring at the primary system of BWR's and PWR's is put on the enhancement of the knowledge base for the interpretation of acoustic signatures. The detailed evaluation of acoustic events requires extensive know-how and analysis efforts; interpreted case studies and well known reference patterns are the basis for the evaluation of extraordinary acoustic signatures. Plant independent registration and analysis of operational experiences, which have been gained on site, provide status information for components assessment.

ISTec/GRS has been involved with a large number of occurrences for analysis and for clarification of the problem by acoustic analysis. The advantage of its use has become obvious in numerous cases. Their bandwidth reaches from

- status assessment of the primary system and its components over
- support and optimization of repair measures with reduction of radiation dosage to personal up to
- flanking and accompanying measures for worst case considerations for plant operation

In the following some representative cases of the number of occurrences will be presented.

Status Assessment of Internal Axial Pumps after Replacement of Hydrostatic by Hydrodynamic Bearings.

In recent years consecutively in the three German BWR plants of construction series 69 with 800 MWe respectively 900 MWe electric power the hydrostatic bearings of internal axial pumps (8 respectively 9) have been replaced by hydrodynamic bearings; the pump shafts have been modified and exchanged. The task for the acoustic analysis was to perform a very sensitive surveillance of the pump area for low energy impact or contact occurrences as an additional, time-limited measure. In all three plants the RAMSES system of ISTec has been set up in parallel to the installed LPMS systems during the startup-phase for a period of typically six weeks with relatively low alert levels for an integral status assessment of the internal axial pumps. By a modem connection from the plant to the ISTec laboratory it was possible to follow the trend of the acoustic occurrences and to transfer registrated burst signals to the ISTec laboratory for analysis.

For a 800 Mwe plant, the results of the measurement campaign are shown in fig. 4. The measure was part of the start-up program after the shaft replacement, accepted by the authoritiy. Measurement phases A, B and C with additional FM tape recording are marked in the pump speed diagram. Measurement phase B contains test impact recordings for upper and lower signal evaluation region. Measurement phases A and C show recordings of cold testing and hot testing of the pumps. The analyses were supplemented by the recently developed acoustic module and classification module of ISTec /6,7/. The registered signatures delivered no specific occurences of mechanical impact or contact type. The experience with the new shafts and bearings of the internal axial pumps is positive.

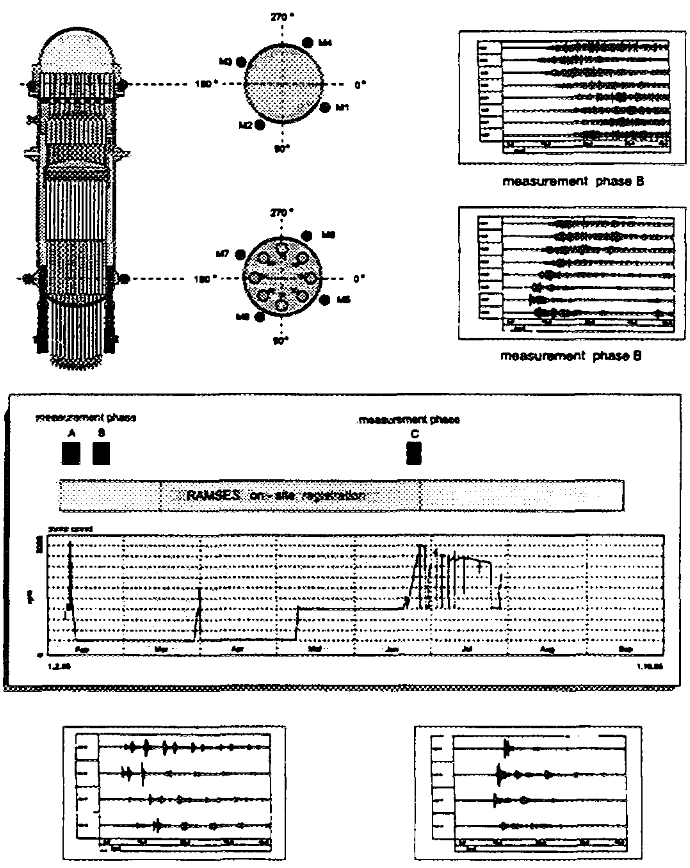

Fig. 4: Status assessment of internal axial pumps after replacement of bearings

Support and Optimization of Repair Measures at Steam Generators

In the Swedish plant Ringhals 2 (a Westinghouse type PWR) as a repair measure the steam generators have been replaced by new steam generators of Siemens/KWU. Unexplained acoustic indications occurred in a steam generator at high power levels. Clarification was required by the authorities before granting licence for full power. The acoustic signals, stored on magnetic tape were analyzed in ISTec laboratory. Fig. 5 shows the sensor positions and a measured burst pattern. The signatures could be interpreted as being caused by fluid induced, position invariant impacts of a loose metallic part with a mass between 100 g and 250 g within the feedwater ring at a specified angular position of 260 and 280 degrees. The localization graphs drawn on the development of the steam generator are shown in fig. 5.
A steam generator measurement phase with different water levels and constant feed water flow showed differently frequent impact signals in accordance with the results, see fig. 5. The following surveillance program, which had been accepted by the Swedish authority, was then established to monitor the steam generator until the regular end of the fuel cycle. A later inspection of the feedwater ring confirmed the ISTec diagnosis. A metal disc for inert gas provision during pipe welding, fixed by a wire, was found exactly in the specified region of the feedwater ring of the steam generator.

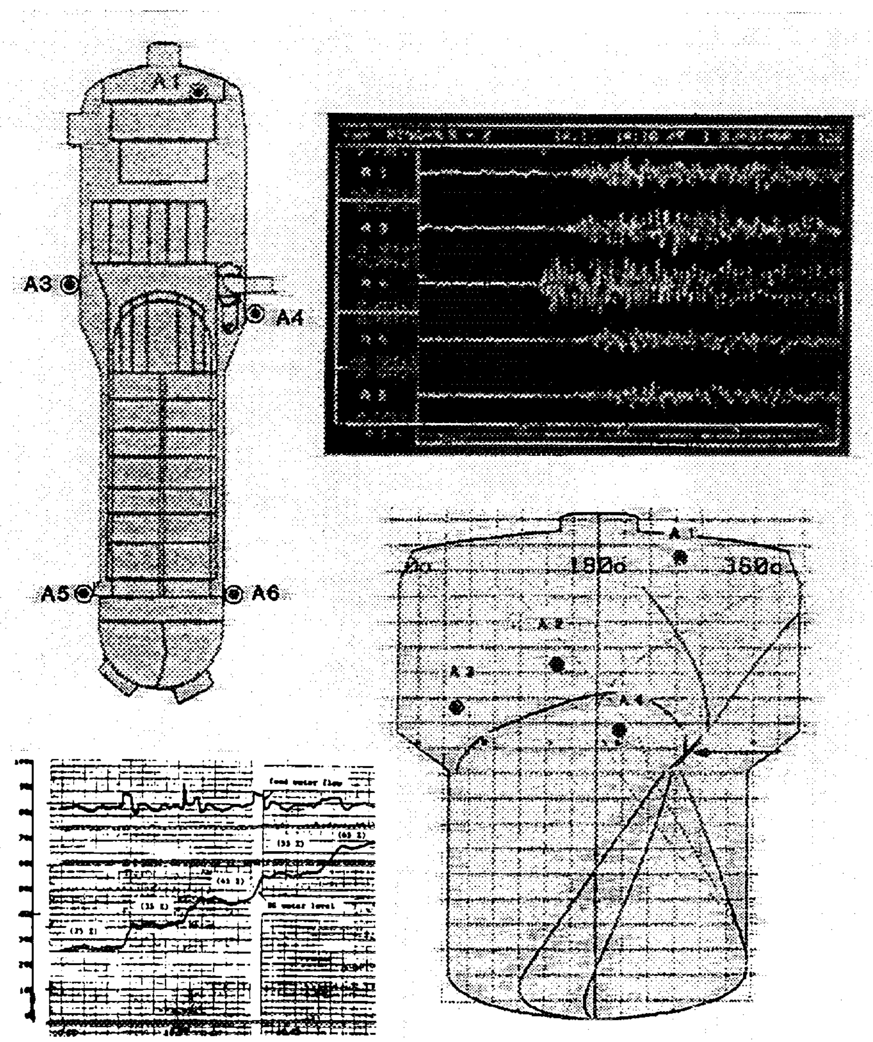

Fig. 5: Support of a repair measure at a steam generator

In a 1300 MWe PWR as a repair measure the steam generator tube bundles in the four steam generators have been fixed in the U-tube area with a comb-like construction by movable metallic swords. A combined instrumentation and monitoring program has been established in acceptance of the authority and TUEV. In the height of the U-bend area at each steam generator two accelerometers, shifted by 180 degrees, have been installed. The signals of each first sensor have automatically been monitored by alert levels, the second sensor was used as a passive measurement channel. Test impacts with an extended accelerometer instrumentation within the steam generator have been performed and recorded as reference signatures. Fig. 6 (left part) shows the observed signal pattern of a test impact to the comb construction with an aluminium rod. Vibrational measurements have been carried out under full power operation, too. Fig. 6 (right part) shows the APSD of a tube bundle sensor. Frequency peaks can be attributed to eigenfrequencies of the tube bundles and pendular vibrations of the steam generator.

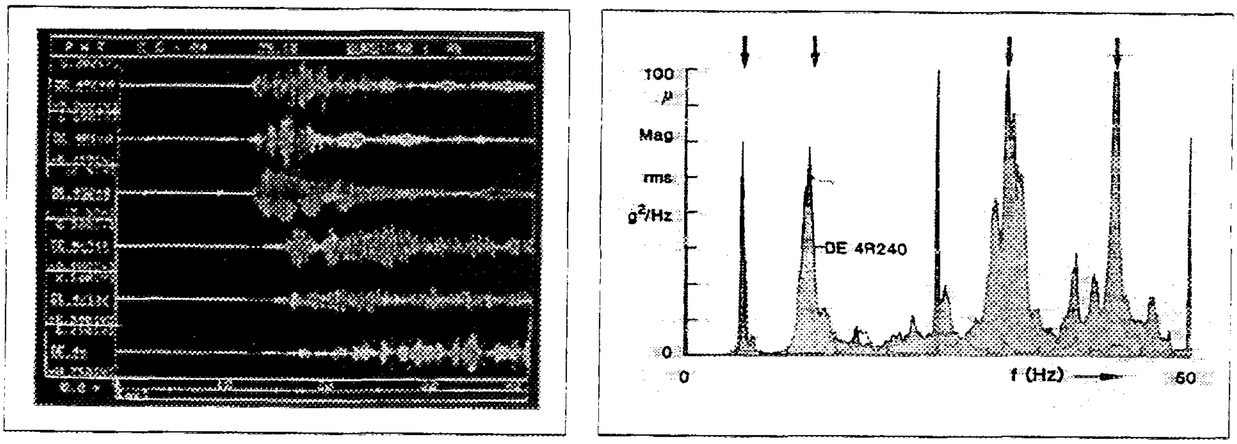

Fig. 6: Test impact pattern and APSD of steam generator instrumentation

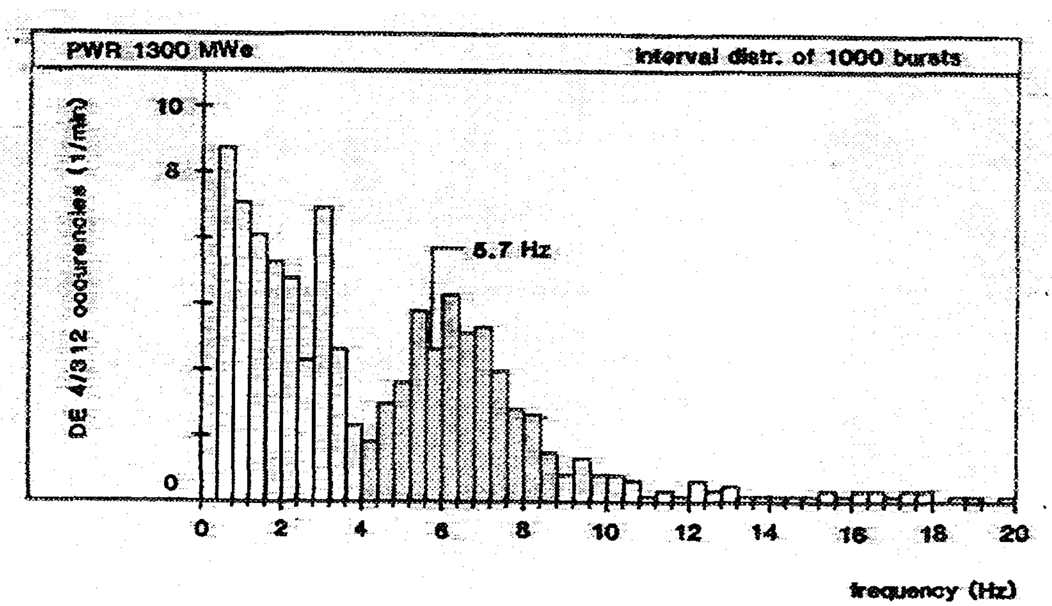

Fig. 7: Distribution of burst time intervals of tube bundle impact signals

During normal power operation of the plant characteristic burst signals from one sensor with a high sequence rate could be observed, yielding metallic impacts originating from one fixed position. Interval time analysis of fig. 7 showed a distribution with a characteristic shape and a dominant peak, which could be correlated to measured vibrations of steam generator tube bundles. This finding was supported by the fact that the impacts were subjected to operating conditions of secondary side (feedwater flow, asymmetric fluid load). Impacts of this kind could be avoided during subsequent operation by appropriate operating measures. Trending of the measured signals showed the success of the measures taken; the experience with the repair measures are positive.

Flanking and Accompanying Measures Needed for Plant Operation

An example for a flanking measure of acoustic analyses will be given below. Inspection of the core barrel of a 1200 MWe PWR showed indications for defective austenite core barrel screws, which should be replaced by ferritic screws then in the next plant revision. If screw failure should happen to a major extent, then by a worst case consideration it should be assessed that it could be recognized at an early stage by an appropriate measuring program.
A theoretical estimate of the acoustic evaluation showed that expected reactor vessel acoustic accelerometer amplitudes of a possible loose and impacting form sheet would reach signal noise ratios of more than 3:1 for the reactor vessel sensor and should therefore be detectable.
A combined acoustic and vibration monitoring program was performed till the end of the fuel cycle. The RAMSES system was installed on the site, the signals of four acoustic sensors of the reactor vessel have been used as monitor channels and were measured with relatively low alert levels. In the left part of fig. 8 the sensor positions are shown, in the right part the number of alarms per day in a 20-days period for the four sensors is presented. Digital signals of registered occurrences were transferred by telefone to the ISTec/GRS laboratory. The signal patterns could be evaluated as being not specific of the core barrel. No indications of ongoing screw failure of core barrel could be observed for this fuel cycle. This finding was confirmed later by inspection.

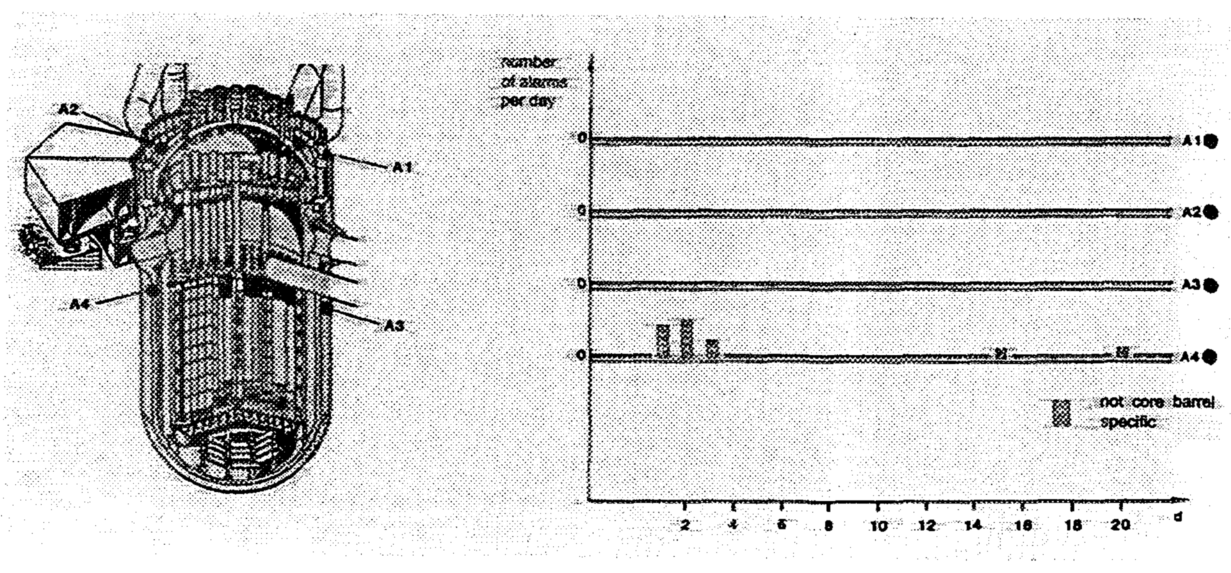

Fig. 8: Acoustic monitoring of possible screw failure of form sheet of core barrel as a flanking measure

5. Conclusions

Experiences during recent years proved that - more than the mere detection of loose parts - acoustic signal analysis has high potential for safety assessment of the primary system and its components with respect to their mechanical integrity. Basic requirement for a reliable diagnosis is the availability of a knowledge basis for the interpretation and evaluation of acoustic signatures. Successful applications of acoustical analyses have been described and illustrate the comprehensive potential of an active on-line monitoring and alarm processing of acoustic signals in nuclear power plants. The high cost of unplanned shut-down of the plant can be reduced and the safety of the nuclear power stations can be improved by applying such methods. Safety assessment of primary system and ist components with respect to their mechanical integrity can be performed now during operation of the plant.

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Safety Aspects of Neutron noise diagnostics and Loose parts monitoring in WWER reactors

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Introduction

There are several methods making use of fluctuating processes and acoustic noises which have been developed during the part twenty years to serve diagnostics of malfunction of nuclear facilities or equipment of nuclear power plants. Some of those might be of academic interests but many of them are really useful for operation and maintenance purpose of nuclear power plant management. There are also quite a few methods which might have direct impact on the safety of nuclear power plant, but most of the other methods have also impact on safety if we consider safety in wide sense.

In this paper our aim is to give very short introduction on different types of well selected noise diagnostics methods and then mentioning their occurrence in WWER reactors we analyse what impact they might have to operational safety and for ageing (which also affects on safety of the installations).

We do not deny, that one of our main aim is to call the attention of management staff of NPP, which deals with safety, safety culture, maintenance and operation proving, that such methods and system can give not only benefit to economy but also impact on safety of nuclear installations.

Core barrel Motion Detection

Soviet built WWER reactors suffer very frequently from core barrel motion. Their is no question about the safety aspects of core barrel motion monitoring, since it is clear

[1] Earlier working at Atomic Energy Research Institute of CRIP on the same subject

that if the core barrel[2] is going to have a large pendular movement it may touch the reactor vessel, which is dangerous. One of the most prominent case reported until now was the core barrel motion registered in NORD NPP (Greiswald) in 1988. This occurrence has been largely analysed in details (see the following publications [1],[2]) The analysis of this event has not been ended yet. Researchers from ZFR have just published their finite analysis on the subject[3]. From detailed description it was clear a hold down spring was the cause of the problem This lead to a near pendular motion of the core barrel. Core barrel was swinging and touching the guide lug(s). One of the guide lug was worn to half during the fuel campaign. One of the role of these guide lugs is to prevent that core barrel would directly knock the wall of the reactor vessel. From the detailed description of the occurrence it becomes clear that noise diagnostics methods based on frequency analysis of neutron flux fluctuation measured by state excore neutron detector were able to notice, the register the beginning of the motion, then to analyse its measure and to follow it during the whole fuel campaign, proving that there are still some part of the guide lug which prevent the motion from the free motion, therefore it is still tolerable from the point of view of safety of the reactor vessel. This is an outstanding example of the usefulness and power of the noise diagnostics methodology.

The methodology which is used to detect and measure the pendular core barrel motion based on spectral analysis of the measurable neutron flux fluctuation. Usually the excore neutron detectors are used (see Fig.1.), which are part of the safety channels as well. But in addition some other sensors (like in-core neutron detectors or accelerometers positioned on the reactor vessel) are also involved [2],[4].

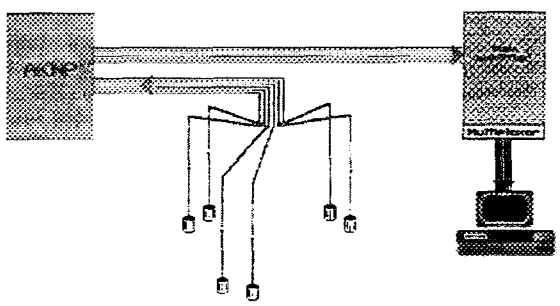


Fig.1. Electronics using state excore ionisation detectors of the AKNP safety system for core barrel monitoring in WWER type reactors

In Hungary two Units of Paks Nuclear Power Station have been equipped with reactor noise diagnostics system containing a half automated core barrel monitoring software package [5], [6]. Here we give some examples of their display (Figs 2a and 2b).

It is well known, that core barrel motions have occurred in several WWER[3] reactors. One of them was reported earlier from Rheinsberg at IAEA Meeting [7]. But we have heard about core barrel motion from Khemnitsky, from Kola, and from other sites as

[2] It is important to clarify that in this respect the terminology might differ due to different translation. We understand under core barrel the holder, which is hanged from the upper flange (Russian terminology is: шахта)

[3] We always use terminology WWER instead of VVER, since it serves for abbreviation of Water cooled Water moderated Energy Reactors

well. We have reported about core barrel motion from a WWER-1000 type reactor as well [8]. Originally there used to be a test for core barrel motion prescribed by main constructor bureau during the installation process for all WWER reactors. But this was carried out before the first load yet using accelerometer. Unfortunately those records were partly secret, partly they were not fulfilled for each WWER Unit in spite of the regulation. Nevertheless there is only one well investigated and easy to access method to build a continuos monitoring for this purpose and this is based on spectrum analysis of neutron flux fluctuations.

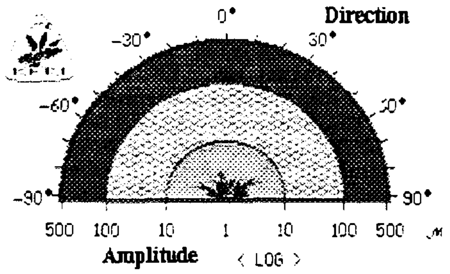


Fig.2a. Core barrel motion less then 10 micron (no motion) was observed

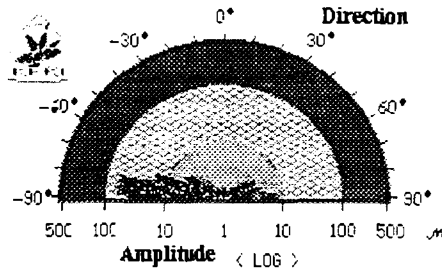


Fig.2b. Core barrel motion larger then 10 microns (less then 100 microns) was detected into 80 degree direction

This methodology is healthy not only for soviet built WWER reactors but for other types as well. Since they construction is different they have less tendency to dangerous motion. Usually they do not knock directly the walls. But usually the have core support and sometime they have also measuring tubes inserted from the bottom of the reactor vessel. Therefore in spite the fact that the amplitude of their motion is usually less, then those of the WWER reactors (since the core support serve as a damping), these smaller amplitudes are enough to damage those guide tubes for measuring installations. We have heard about such occurrences in German reactors as well as in French pressurised water reactors. Breaking those tube can lead to leakage and what is more dangerous one can loose those measuring channels

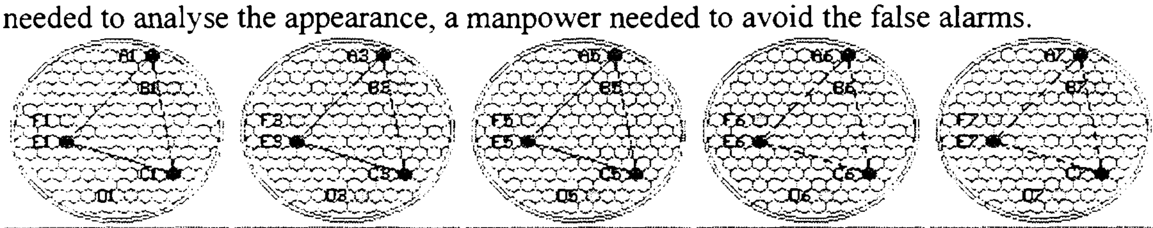
Therefore we believe that core barrel motion monitoring is one of the methods to be introduced to automatic noise diagnostics systems.

In-core vibration monitoring

WWER-440 type reactors have absorber assemblies. Their driving apparatus might be jammed. Such case was reported in 1995 from Paks NPP. Their excessive vibration can cause failure of their moving apparatus as well as damages of neighbouring fuel assemblies. Reports of excessive vibration of such control elements have been published earlier [9].

There are not accelerometers available in the core of WWER reactors. The first such test has been published just recently in one of German reactors, but still their burn out rate is too high, they can withstand radiation only for a couple of months. Today the only reliable information on in-core vibration can be achieved using in-core self powered neutron detectors, which are state composite of WWER reactors. In WWER-440 reactors there are 36 strings containing 7 spnd above each others in the core. In WWER-1000 type reactors there are 56 strings in a core. As it has been proved by numerous successful observation each string can see it own fuel assembly excellently, good visibility was reported from the neighbouring fuel assemblies, while poor visibility was reported for transport effect from further assemblies but still acceptable for horizontal correlation. The latter means that in case of an excessive vibration one of the control assemblies acceptable coherence and phase was achieved [9] between far standing spnd on distance of 4 to 6 fuel assemblies (each assembly has diameter of 144 mm).

The main sign of a control assembly vibration was the appearance of opposite phase with certain coherence between horizontally placed in-core neutron detectors. The task is obvious: we need software with automatic control of phase. Experience shows, that all combination between in-core detectors should be checked first. Once antiphase has been detected a manual package is needed to analyse the nature of that effect. Only owning with such manual package an expert can say something about the nature of that antiphase. The expert should take into account all previous lessons learnt during normal operation. Antiphase can appear due to erroneous connection of cables, due to transport effects, due to bad statistics, due to transients, etc. In principle it is possible to built in a knowledge based support system for experts. This system would contain knowledge and lessons learnt from previous experiences, still an expert is needed to analyse the appearance, a manpower needed to avoid the false alarms.



Уровень 1.	Уровень 3.	Уровень 5.	Уровень 6.	Уровень 7.
A1-C1 FCSA1C14.42S	A3-C3 FCSA3C34.42S	A5-C5 FCSA5C54.42S	A6-C6 FCSA6C64.42S	A7-C7 FCSA7C74.42S
1.4 Гц : 14.8 %	1.6 Гц : 27.9 %	1.5 Гц : 25.4 %	1.6 Гц : 34.1 %	1.6 Гц : 31.7 %
2.3 Гц : 15.1 %	2.2 Гц : 28.5 %	2.1 Гц : 11.6 %	2.1 Гц : 43.9 %	2.4 Гц : 41.5 %
3.4 Гц : 7.5 % 4.6 Гц : 167.6 гр.	3.5 Гц : 18.0 % 4.4 Гц : 162.0 гр.	3.1 Гц : 9.9 % 4.5 Гц : 120.8 гр.	3.1 Гц : 29.1 %	4.3 Гц : 31.6 %
A1-E1 FCSA1E14.42S	A3-E3 FCSA3E34.42S	A5-E5 FCSA5E54.42S	A6-E6 FCSA6E64.42S	A7-E7 FCSA7E74.42S
1.2 Гц : 8.4 %	1.1 Гц : 23.2 %	1.1 Гц : 27.6 % 1.5 Гц : 131.6 гр.	1.1 Гц : 41.1 %	1.1 Гц : 43.0 %
2.3 Гц : 13.6 %	2.5 Гц : 13.2 %	2.7 Гц : 11.7 % 2.2 Гц : 139.3 гр.	2.6 Гц : 28.7 %	2.4 Гц : 43.8 %
3.9 Гц : 9.6 % 4.4 Гц : 172.6 гр.	4.2 Гц : 8.5 % 4.4 Гц : 158.2 гр.	4.4 Гц : 13.0 % 3.3 Гц : 147.2 гр.	3.5 Гц : 29.0 %	3.5 Гц : 31.7 %
C1-E1 FCSC1E14.42S	C3-E3 FCSC3E34.42S	C5-E5 FCSC5E54.42S	C6-E6 FCSC6E64.42S	C7-E7 FCSC7E74.42S
1.4 Гц : 12.4 %	1.5 Гц : 14.2 %	1.6 Гц : 19.2 %	1.6 Гц : 25.3 %	1.4 Гц : 38.3 %
2.9 Гц : 10.2 % 2.4 Гц : 130.7 гр.	2.8 Гц : 25.0 % 2.4 Гц : 170.6 гр.	2.1 Гц : 10.2 % 2.6 Гц : 120.9 гр.	2.1 Гц : 38.1 %	2.2 Гц : 41.5 %
3.6 Гц : 13.2 % 4.2 Гц : 173.6 гр.	3.7 Гц : 11.4 %	4.3 Гц : 21.4 %	3.2 Гц : 24.5 %	3.4 Гц : 26.0 %

Fig.3. An example of the expert support display. Solid lines show in phase dash lines show antiphase behaviour at different elevations between spnd signals at different frequencies (see table)

Conclusion is that since several in-core vibration were reported in WWER type reactors and also since their detection and analyses was solved successfully with existing system it is recommended to have such system for safety purposes. Since many aspects of data analysis cannot be automated such systems cannot be a part of the safety system but it can be an important part of the operator support systems or information machines.

Reactivity coefficient monitoring

Reactivity coefficients are regarded as important parameters for operation and safety of nuclear reactors. Safety margins for the reactivity temperature coefficients are not monitored today on-line at all. Only the sign of that is checked during the start up process. We wish to underline the importance to follow the changes of that parameter since its large negative value is also dangerous for reactor operation in case of emergency halt. Earlier this was not considered as a safety point but today it became clear that in case when the emergency system inject cool water into the reactor core it can response with positive reactivity jump.

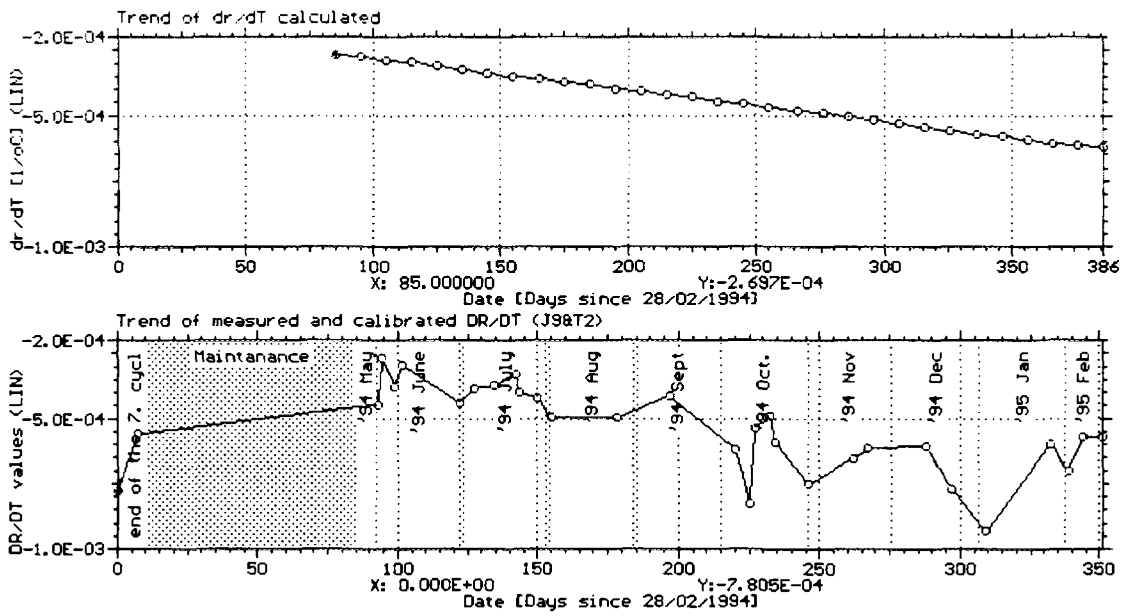


Fig. 4. Estimated and actually measured (using noise method) values of temperature reactivity coefficient during a fuel campaign

To set safety margins for reactivity coefficient during normal reactor operation becomes possible only if one can continuously monitor the changes of that important parameters. Nobody would like to introduce a thermal jump each day to check if that parameter is still inside the allowed margins. Noise monitoring methods are good in that sense that they do not disturb the normal operation at all.

There are two methods which have been reported to follow successfully the changes of reactivity coefficients during normal operation in a whole fuel campaign [10], [11]. They are only partly independent, but they contribute to each other rather well. It has been demonstrated the general change of the reactivity coefficient could be followed rather well during a fuel campaign in spite the fact that there were even disturbing effects. Consequently we can state that such computer based reactivity coefficient follower can be built in as an operator support measuring device in each Unit. Once we have an equipment which is capable to follow the changes of that parameter we can request limits for the given parameter. This is a task now for national authorities.

Temperature sensor positioning

Top of core thermocouples are very important sensors for core monitoring and diagnostics of WWER reactors, but hey are equally important in each PWRs. Their malfunction as well as their bad positioning would affect very much on reactor calculation as well as on detecting hot spots or other effects important for safety of the reactor core. Noise diagnostics technique is capable to test the correct displacement of the thermocouple and also to monitor its ageing or malfunction. We can reach this goal combining two methods.

The first method is based on phase between the temperature sensor and one of the neutron detectors [12].The second method uses the autoregressive modelling of the single output of the tested thermocouple. Using the autoregressive model one can estimate the thermocouple response on single step. From this curve one can estimate the time constant of the thermocouple in situ during normal operation without disturbing the operation or function of the reactor or that of the thermocouple. Time constant of the thermocouple is informative for its position as well as for its ageing, corrosion etc. therefore it can be used for the validation of the thermocouple both its positioning and correctness. This methodology has been applied fruitfully in several nuclear power plant and today simple versions built in a PC are available. We can recommend that for thermocouple diagnostics and signal validation.

Safety aspects of loose parts found in primary circuits and in the core

Loose parts (forgotten objects) found in WWER reactor has been told not too dangerous from direct safety point of view for WWER. Recently a rather large (about 8 kg) object has been forgotten in the steam generator of Unit 2 of Paks NPP. This steel plate was broken into small pieces during a year of operation when impacting the wall of the collector of the steam generator. The surface of the collector was just a little damaged. Tube inlets were notice to be worn. Some small pieces from that steel object was stuck in the tubes of steam generator. This lead to a tremendous work during maintenance period at the end of the fuel campaign. Thos cost of this maintenance work was rather huge and it caused also a delay in restarting the Unit. But in the first plan these are minor consequences from the point of view of safety of the nuclear installation. In the same time small broken pieces were swept away toward the reactor vessel.

It shows on the careful planning and good material of WWER rectors that no real damage was found either in the main coolant pump or on main closing valves. But also small fragments of forgotten object have been found in the reactor core and on the top of the reactor core. since it was difficult to wash them due to high radiation about 41 fuel assembly was removed from the core. No real damage of fuel pins was observed. But during the operation one of the regulating rod was stuck in a middle position. It has not been proved that these two effects are in direct correlation, but one suspect that this was one of the possible causes of the failure of the movement of the control rod. And this has a direct impact on the safety of the nuclear installation.

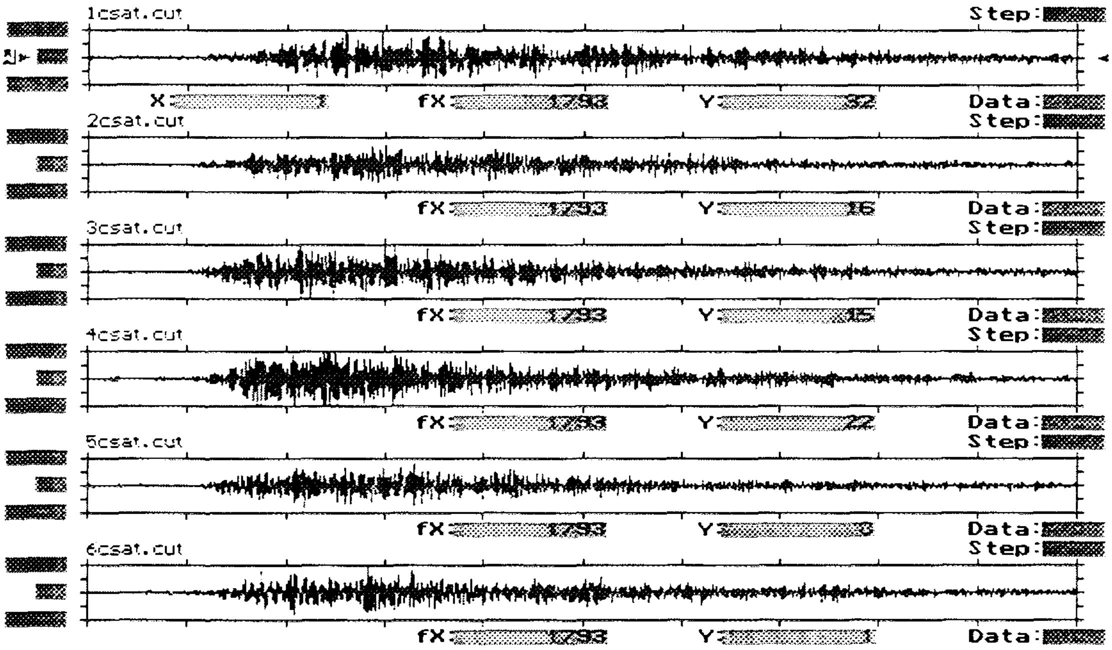


Fig. 5. Screen of the HELPS loose part monitoring system during analyse of the event

Consequently even if we can conclude that WWER reactors were design to withstand rather large loose parts from material point of view, still loose parts monitoring during the start up (and during normal operation as well) is advisable to avoid such cases, which might have direct impact on safety. Since we are talking about continuous monitoring process we believe that such analysing picture as it is shown on Fig.5. is good for experts when the event has to be found [13]. But for continuos monitoring we use an expert system which gives recommendation directly (see Fig.6. which was borrowed from HELPS -Hungarian Expert Loose Parts system [14])

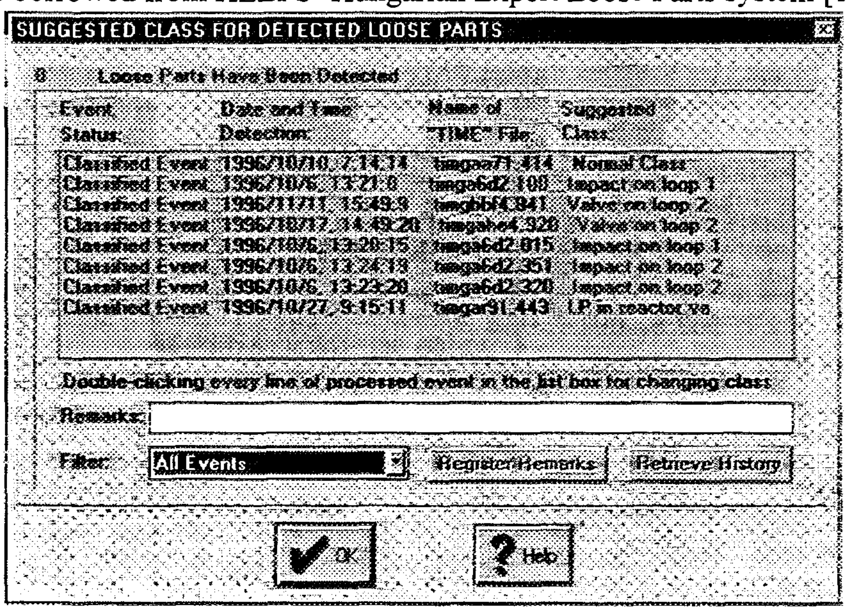


Fig. 6. Example of the expert system display of HELPS loose parts monitoring system

How to organise the diagnostics in NPP

There are different point of view on diagnostics of nuclear power plants. Some people claim that diagnostics is only the normal checking which is prescribed by manufacturers of the main components of the nuclear power plant. they disregard the developments in data collection systems and computer aided expert diagnostics systems. Some other people tends to order a ready made, rather sophisticated, fully automated, intelligent, expert system, which would give them guarantee that if their is a beginning of malfunction then it gives warning, later alarm, and if it says everything is OK then it is 100% OK. they do not like such terms as missed alarms or false alarms. They are used to safety systems, and they believe that they should get a system which acts automatically.

We believe that both attitude toward diagnostics is erroneous. What we need today is the following. We can and we have to purchase diagnostics systems based partly on fluctuation of different measurable parameters or on vibrations. Recordings and primary data evaluation should be done automatically. The most time consuming data analysis work as well as those expert analysis which can be automated should be automated. As a product of such system we get warnings that something is getting wrong with some indication what can be the cause, where to start further analysis, or we can get a message that everything was unchanged in the measured parameters.

Here starts the role of the experts which should analyse the warnings coming from the automated system with manual methods and comparison with the plant data. Such activity cannot be automated in the future as well since there is always a chance that one get something new, a new effect or new problem, which had not been before, consequently no way to learn it for any intelligent learning system. We believe that we cannot miss the careful analyse made by expert today and in the future as well.

The human factor has an overwhelming importance in diagnostics. They are the best to avoid false alarms. Also they are able to present the results in a form which is acceptable for others: for maintenance work, for management, for safety people etc. This is a second task, and it is as important as the first one. One of the biggest problem today is to make comprehensive report which would be east to understand by other specialists. If the expert tries to explain his or her conclusion using terminus of noise diagnostics, like spectra, coherence, autoregressive modelling etc. then the others cannot follow that and they will not accept the conclusion, they will not take actions. If expert gives only the final conclusion without explanation then nobody believes that plus it does not corresponds to the requirements of the quality assurance as well. This later gives the correct solution to this problem. According to the quality assurance program it is essential that the types of the report, their routes to the interested persons their contains and how to use it will be prescribes in details. Expert should have a detailed plan that in case of a loose part where and to whom make the report. What should be the contains of that report. It is clear that each case differs from another case. There must be a part, which is flexible. But this must be the smaller part of those reports (less than 20% of the total report). It cannot be the task of the expert to define to whom send the report. Also the action to be taken by addressee should be a part of the plant regulations. And this is a task for the plant management to organise now this part of noise diagnostics.

Conclusion

Technical part of noise diagnostics can be regarded as well analysed, mathematically solved. Many well organised noise measuring systems are working in different NPPs. They usually have more or less expert parts as well. We presented some areas, which have overwhelming importance for WWER diagnostics and which can be regarded as well understood, and also which have proved that methods used in noise diagnostics can contribute a lot to the safety of nuclear installation. We conclude, that most of the technical part has been solved or is in progress. The human factor and the reporting on the results of noise diagnostics is one of the most vulnerable part of this methodology today. The most important task for plant management and also for diagnostics today is to work out regulation, reporting system, limits and action plan for given plant which is in accordance with quality assurance programme.

Acknowledgements

Most of the reactor noise methods were developed while author was working at Atomic Energy Research Institute, Budapest, while loose parts system was elaborated in INT. The contribution of co-workers are highly appreciated.

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APPENDIX

Diagnostics Based on Fluctuation and Vibration Measurements in Nuclear Power Plants and its Impact on Nuclear Safety

Proposal for publication of a Review book of methods, systems and achievements

Edited by G. POR

Main

In the past twenty years noise diagnostics methods developed from laboratory tools to state system for nuclear power plants. The main area of these systems are as follows:

- Loose parts monitoring systems
- Acoustic based leakage monitoring
- Vibration monitoring of main components of primary loops
- Vibration monitoring of secondary loop components with main accent on turbine diagnostics
- Reactor diagnostics using neutron, temperature and pressure fluctuations
- Diagnostics of rotating machines

Methods includes:

- Conventional FFT spectra and their interpretation
- Alarms based on time signals and its moments (average, rms., etc.)
- Autoregressive modeling
- More advanced modeling methods: fuzzy logic, wavelet technique, neural network
- Interpretation models for different kind of events

Very valuable results have been presented in many areas of noise diagnostics and interpretation. Nevertheless in the present book we shall emphasize on achievements in practical use of the methods and analyses listed above since one of our main goal is to convince managers of nuclear power plants about usefulness of application of this methods and systems. Therefore contributions informing about the practical use of such methods and systems are most welcome.

Collecting information and papers

Reviewing book is considered to be the best collection of the present methodologies ready to be applied in nuclear power plants. Therefore the methodological part of the book is planned to collect by addressing the most prominent specialist in given subject. Selection will be made mainly on previous publication activity (surveying SMORN, IMORN and other Conferences /ANS, IMORN, NPIC&HMIT/ Meetings /like IAEA TC/). Nevertheless we shall also announce the preparation of the forthcoming book inviting anybody who wishes to contribute. Papers submitted for this book will be selected, accepted or rejected by a special committee.

To give information on his existing system(s) will be free for everybody, but the size will be limited. For longer information we shall have progressive charges. For a single page information we shall charge only 10 dollars but for each consecutive pages we double the charge. Systems will be surveyed thoroughly and in those survey we keep the rights also to criticize or to compare presented systems with existing other ones. Firms wishing to present their system should agree beforehand the critics without any right to oppose it on business base.

Editing

We request camera ready version in generally accepted worldwide PC based editors program like WORD for WINDOWS. Editing will be made by editor, but he will have an advisory group of specialist from different countries. All correspondence and discussion will be made via e-mail system.

Publishing

We consider to publish a book. Therefore it is desirable to involve a publishing company at the final period of the preparation of the book. IAEA would support this book but financing would come form publishing company (selling the book).

«Detection and localization of leak of pipelines of RBMK reactor. Methods of processing of acoustic noise»

The authors: Tcherkaschov Y.M., Strelkov B.P., Chimanski S.B. (RDIPE), Lebedev V.I., Belyanin L.A. (LNPP)

Introduction

Absence of reactor protection cap and the one-circuit type of a reactor cooling makes especially important to increase of operation safety of the equipment by the duly leak detecting of pipelines and exception of their instant destruction.

For realization of leak detection of input pipelines and output pipelines of RBMK reactor the method, based on detection and control of acoustic leak signals, was designed.

In this report the review of methods of processing and analysis of acoustic noise is submitted. These methods were included in the software of the leak detection system and are used for the decision of the following problems:

- leak detection by method of sound pressure level in conditions of powerful background noise and strong attenuation of a signal;
- detection of a small leak in early stage by high-sensitivity correlation method;
- determination of a point of a sound source in conditions of strong reflection of a signal by a correlation method and sound pressure method;
- evaluation of leak size by the analysis of a sound level and point of a sound source;

The work of considered techniques is illustrated on an example of test results of a fragment of the leak detection system. This test was executed on a Leningrad NPP, operated at power levels of 460, 700, 890 and 1000 MWe.

State of Problem

For the inspection of the RBMK reactor input pipelines (low water lines, pressure and distributing headers) and output pipelines (steam-water lines, drum separator and sections of downcomer piping) a method based on the detection and monitoring of acoustic signals of a leak propagating in the air atmosphere has been suggested.

The system operation is based on the principle of recording acoustic noise emitted by a leak, using the acoustic sensors.

The leak detection system, used this acoustic method makes an uninterrupted automatic monitoring of the reactor piping leak and has to solve the following problems:

- leak detection with sensitivity of about 230 l/h in the course of 1 hour;
- leak localization with an accuracy of approximately 1 m;
- quantitative evaluation of the coolant draining away;
- generation of warning signals to the power unit operator and alarm signals on reaching the threshold leak size.

The tests of a fragment of the leak detection system are carried out on a Leningrad NPP.

Object of inspection

Part of the RBMK cooling circuit piping located in rooms 033 and 505 within the reactor leak-tight area and not having thermal protection coatings is subject of inspection. The cooling circuit consists of two symmetric loops, controlled equipment within one loop contains the following components (Fig. 1):

a) Input pipelines of the reactor (room 033):

- pressure header (1 piece);
- group distributing headers (22 pcs.);
- low water line piping sections (846 pcs.);

b) Output pipelines of the reactor (room 505);

- steam water line piping sections (846 pcs.);
- steam drum separators (2 pcs.);
- downcomer piping sections (24 pcs.).

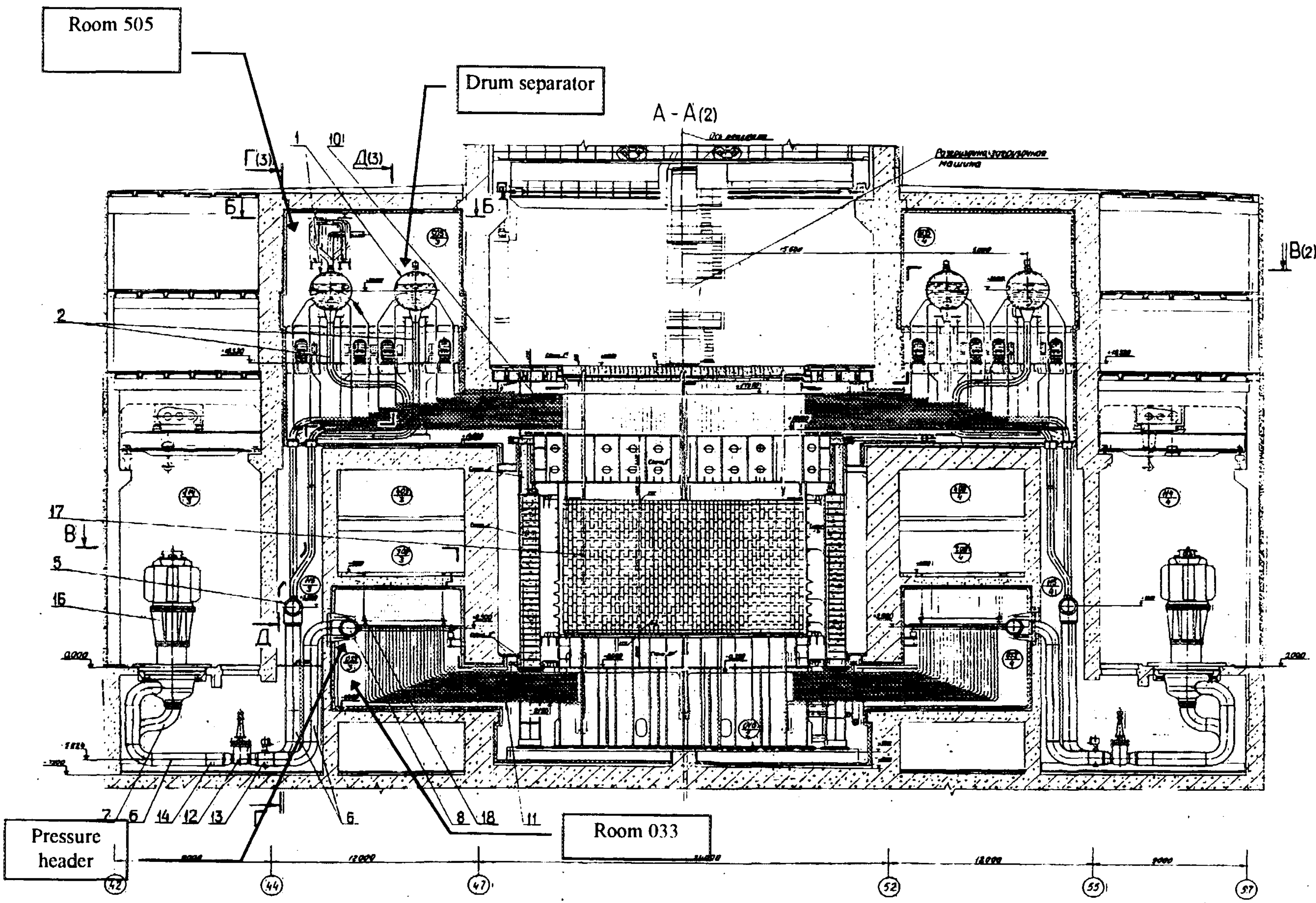


Fig. 1 The Unit 1 of Leningrad NPP with RBMK reactor

3. Methods of analysis

The algorithm of inspection comprises the following basic procedures (Fig. 2):

- «fast» detection of a leak;
- «early» detection of a leak;
- definition of location of a leak (localization);
- estimation of a leak flow rate.

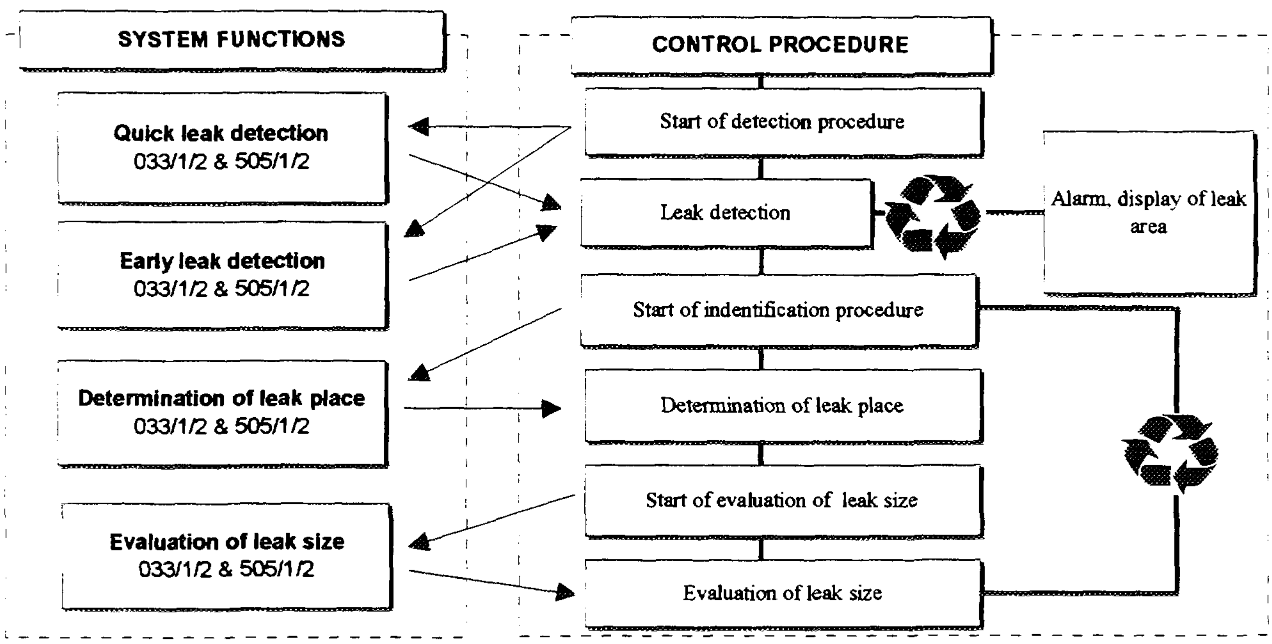


Fig. 2 The basic procedures of inspection algorithm

3.1. «Fast» detection of a leak by sound pressure method

The algorithm is based on comparison of sound pressure level in a preset frequency band with a threshold specified (the sampling cycle is about 2 min) and provides the inspection at signal/noise ratio ≥ 1.

So as to reduce the probability of «signal missing» and «false response» of the system the comparison is made for a preset time interval using three methods:

- excess of the absolute threshold for each microphone;
- excess of a relative threshold for each microphone;
- excess of a relative threshold, when comparing readings of different microphones.

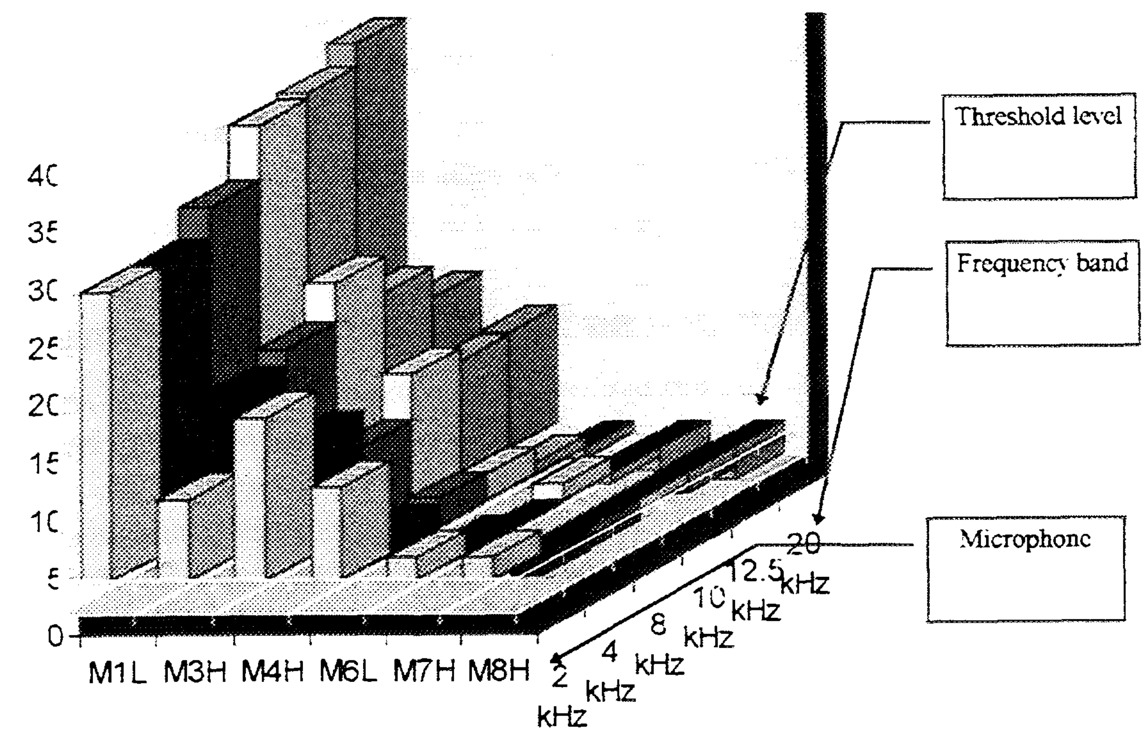


Fig. 3 Signal/Noise ratio (dB) of microphone signals of the leak: 800 l/h

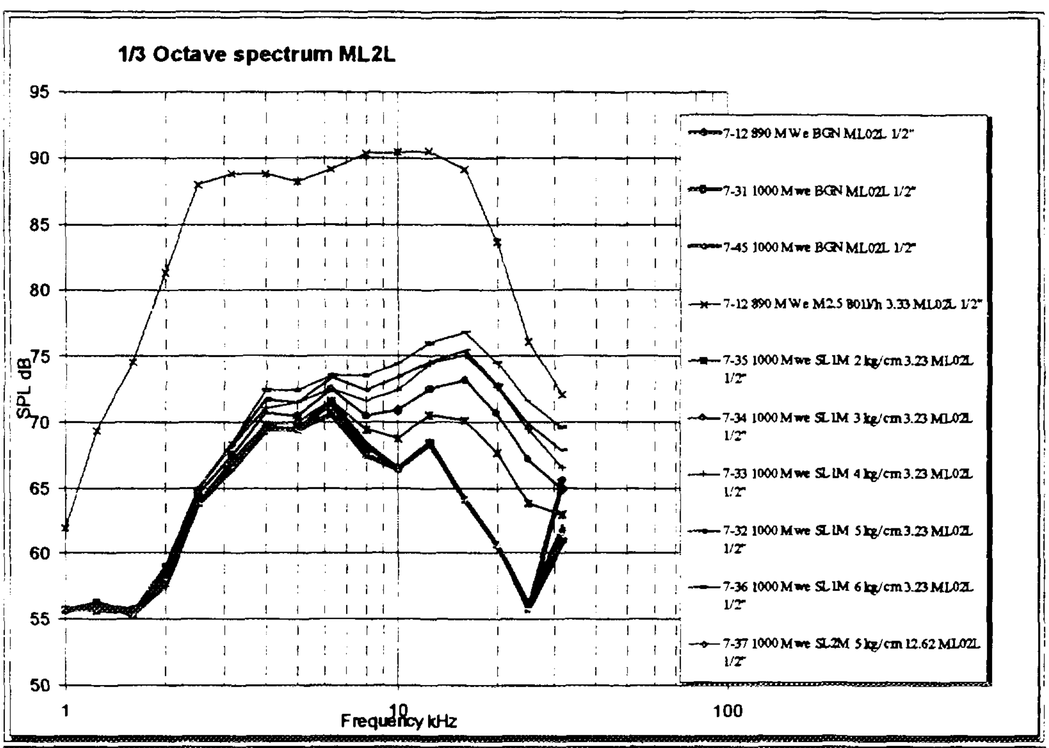


Fig. 4 The 1/3 Octave spectrum of microphone signals (SPL dB) for imitation of different leaks

As the evaluation results of the NPP test, the following results were obtained:

- the system sensitivity makes up to 0.067 m^3/h (at a distance up to 3.56 m from the source of sound) in rooms of low water pipelines;
- and up to 0.024 m^3/h (at a distance up to 4.36 m from the source of sound) in rooms of steam-water pipelines and drum separator;.

The threshold of detection was 3 dB.

3.2. «Early» detection of a small leak by correlation method

The algorithm is based on comparing cross-correlation characteristics of signals generated by a microphone pair with a preset threshold in the frequency band selected (the polling cycle is about 40 min) and it provides inspection at signal/noise ratio ≤ 1 (Fig. 5, 7, 8). To increase fast response of the method the polling is realized for the microphone pairs chosen in advance and for preset polling path.

(1) Analysis of coherence function

The coherence function is used for choice of an optimum frequencie band of the analysis and for detection of a local sound source.

To get diagnostic parameters, coherence function is smoothed out by the «moving averaging» method.

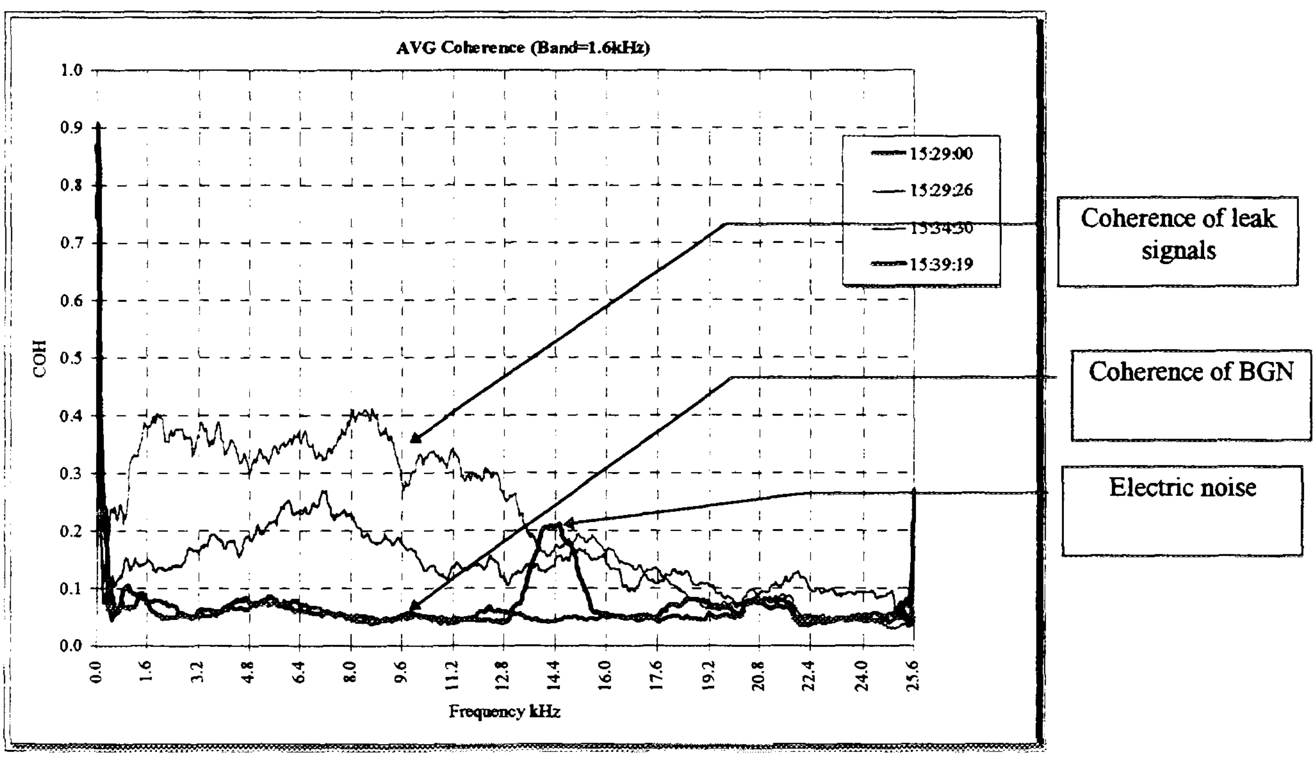


Fig. 5 Comparison of coherence functions, made before and after leak occurence

(2) Analysis of correlation function

The cross-correlation function is calculated by ZOOM method (frequency band is about 1.6 kHz) according to the following algorithm:

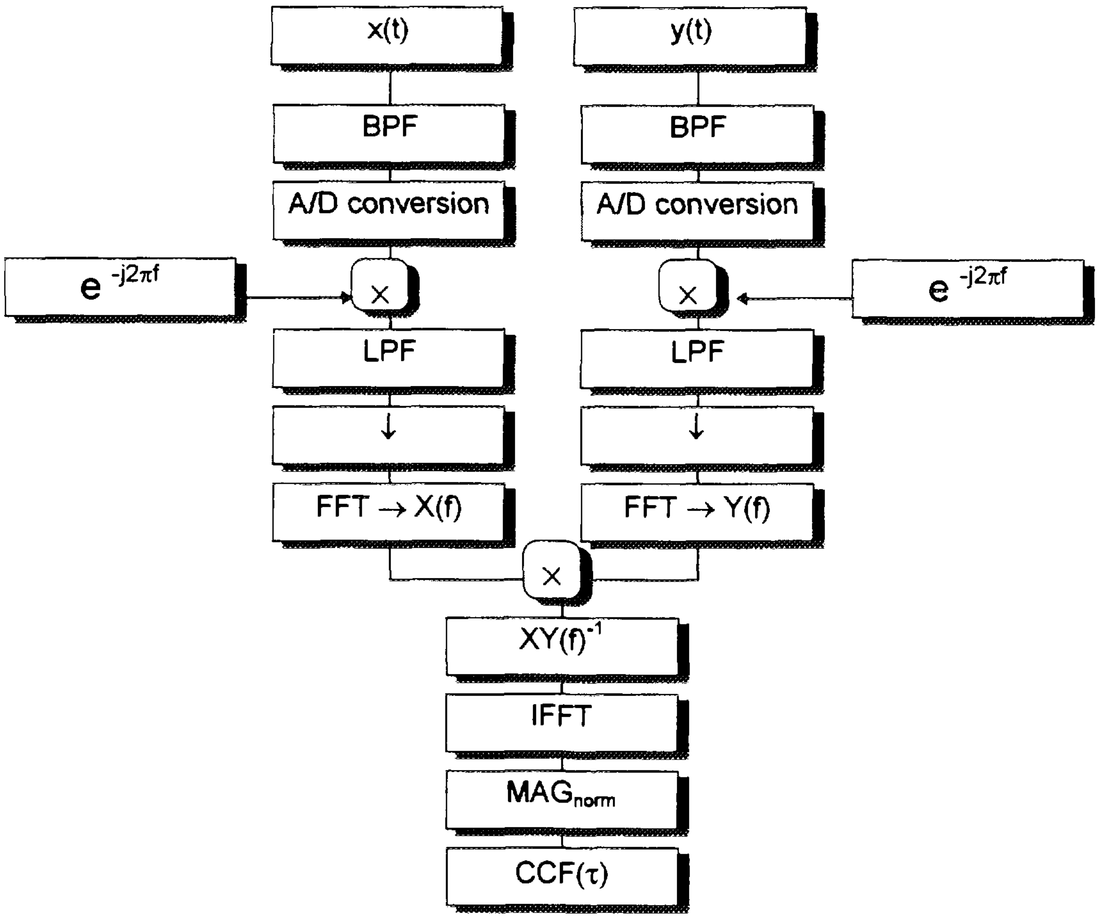


Fig. 6 The algorithm of calculation of cross-correlation function

The following diagnostic parameters of leak signal was used:

- maximum value of the normalized correlation function: Max[CCF]
- form factor of correlation function: Q[CCF]

The example is shown on Fig. 7, 8.

/50

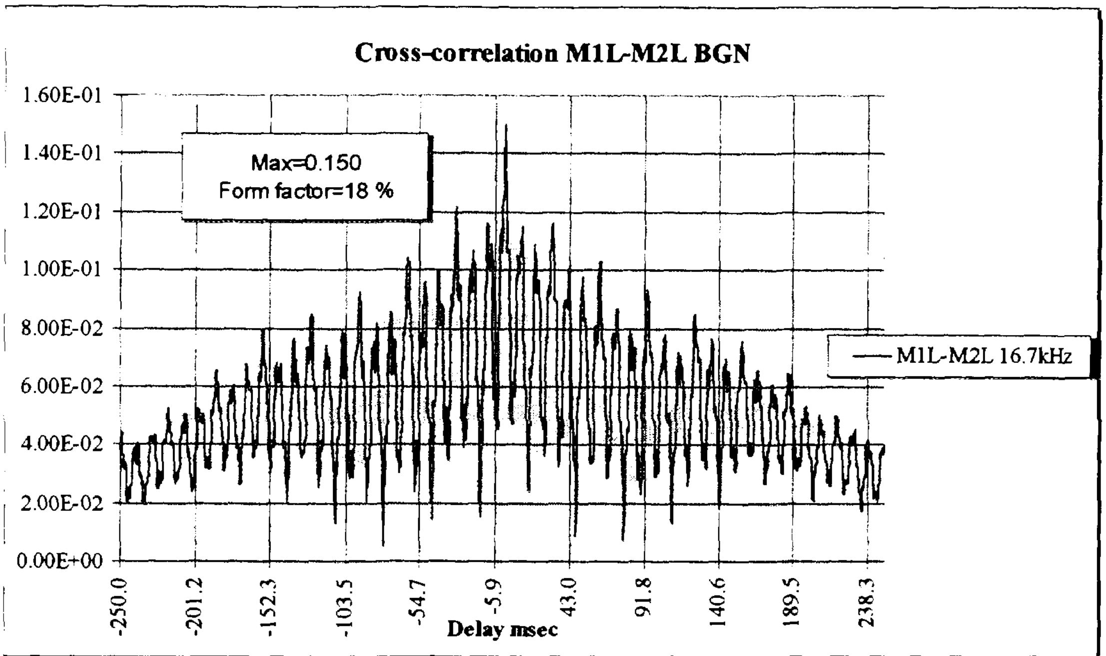


Fig.7 The cross-correlation function, made before leak occurence

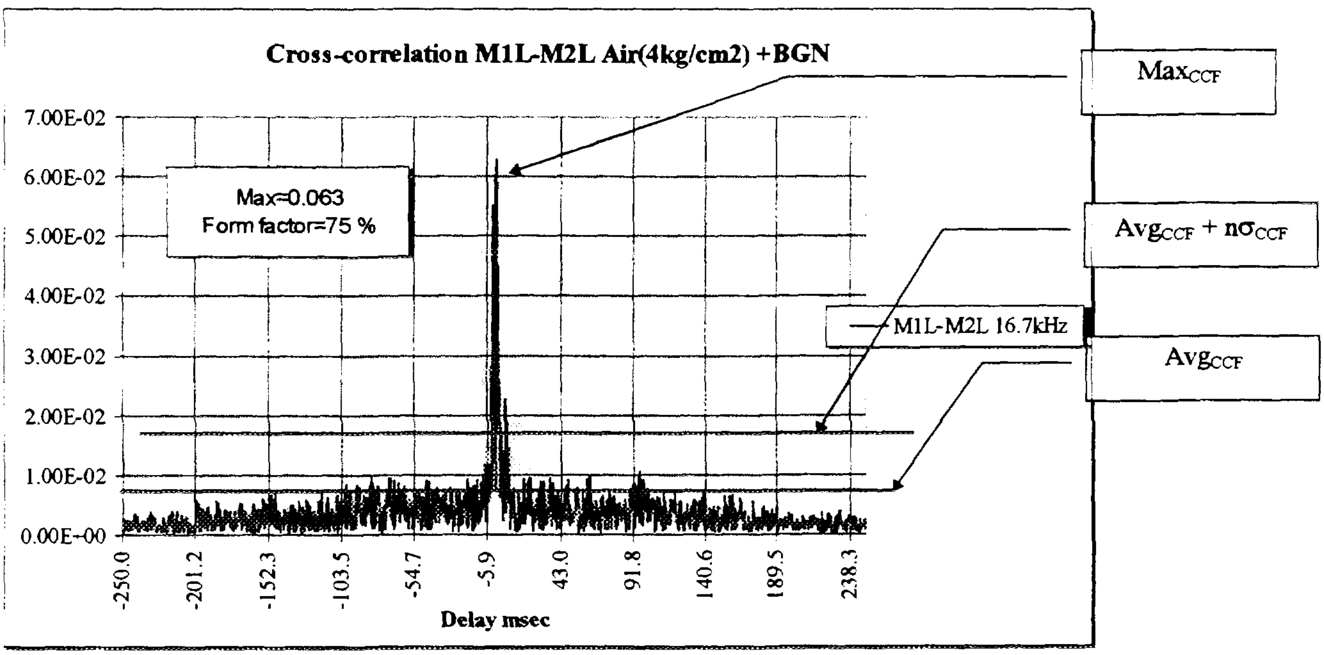


Fig.8 The cross-correlation function, made after leak occurence

As the evaluation results of the NPP test, the following results were obtained:

- it takes approximately 40 min to detect a leak by the correlation method;
- leak detection by form factor of cross-correlation function obtained by ZOOM method proved possible in the range of frequencies from 5 to 20kHz in the case, when: $(S+N)/N \cong 1$ and less;

3.3. Definition of location of a leak (localization)

The location of a leak is ascertained by the hyperbolic method (fig. 14) on the basis of the result of cross-correlation analysis of signals from a group of microphones (the sampling cycle is about 40 min, the error being about 1 m), as well as by the sound pressure method (fig. 10, when the signal/noise ratio ≥ 1). A group of microphones is defined when analyzing the sound level in the course of «fast» detection of a leak.

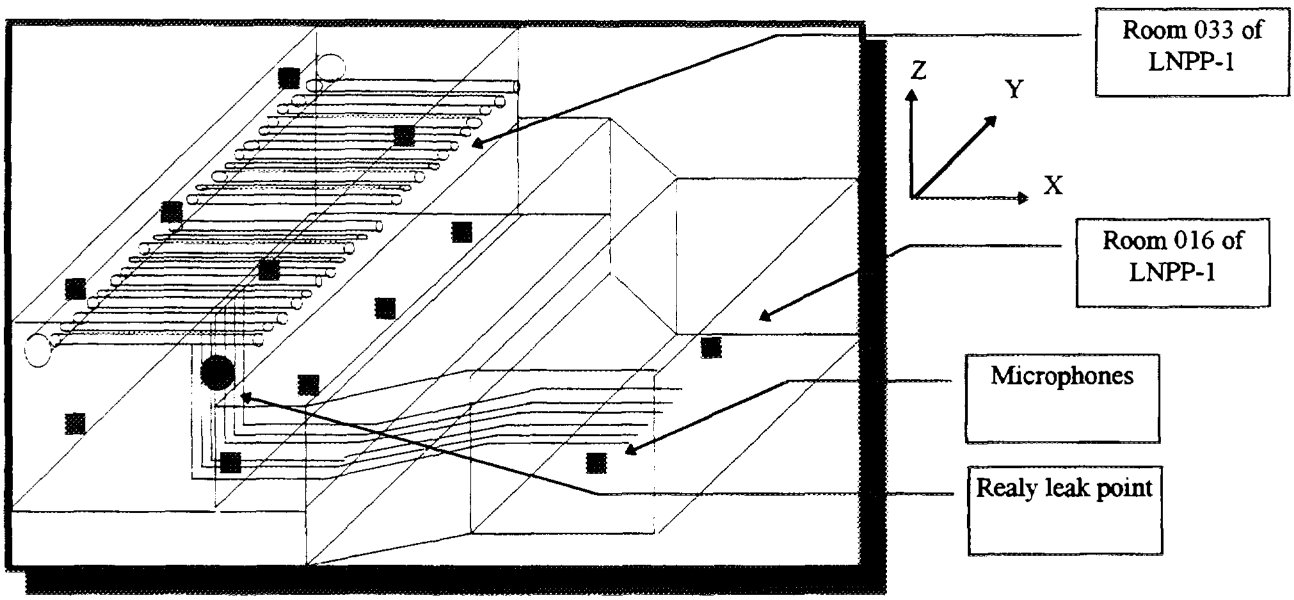
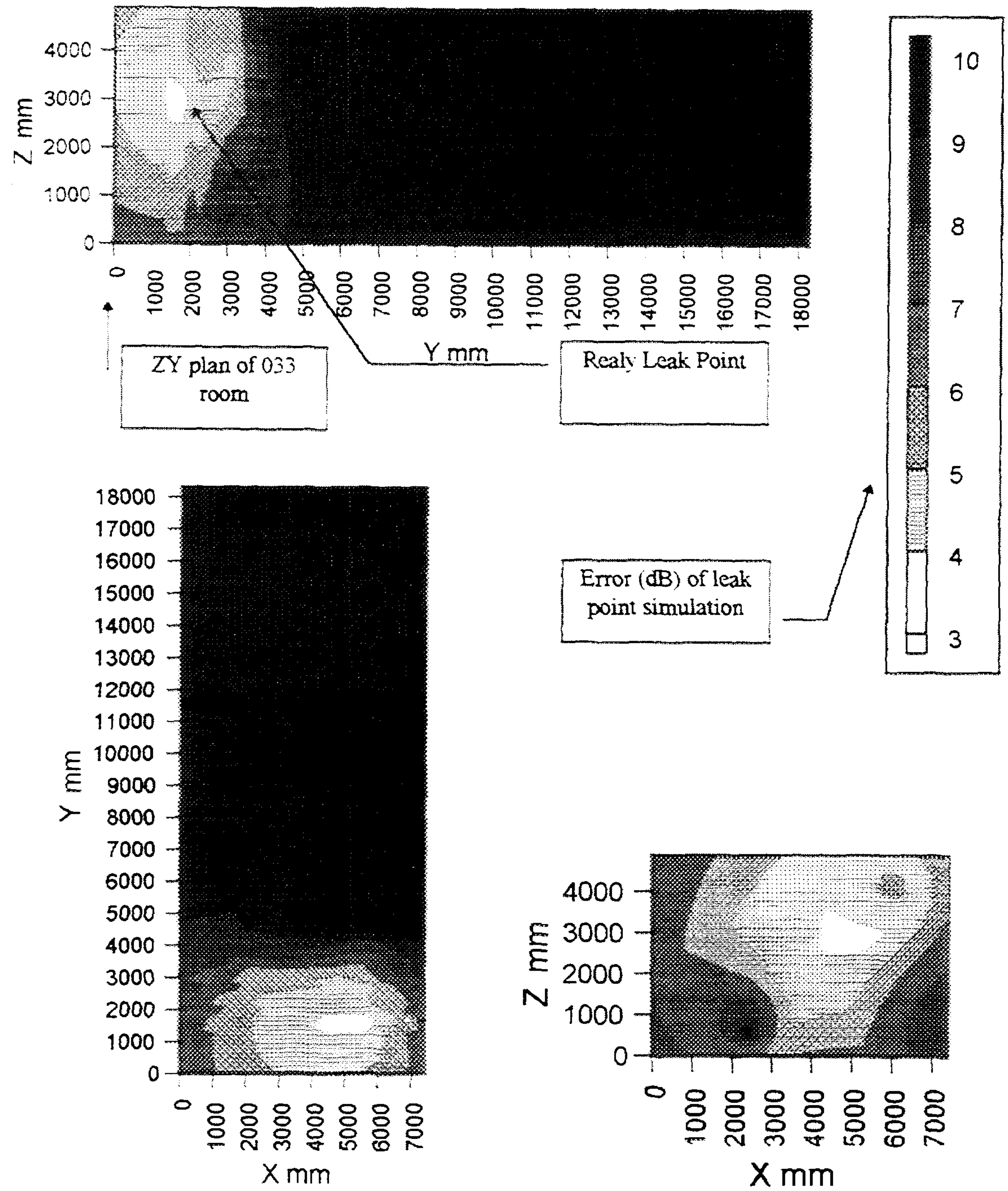


Fig. 9 The positions of microphones and leak imitator in NPP rooms

(1) Localization by sound pressure method

The definition of space location of a leak is made on the basis of analyzing sound pressure levels recorded by microphones after leak occurrence, making allowance for sound attenuation in the room.

The calculation method takes into consideration the presence of direct sound propagation areas and the effect of sound reflection, the characteristic of the microphones directivity and the influence of the background noise in the rooms.



Freq.= 8kHz, Mic. M1L, 2L, 3H, 4H, 6L, 7H, 12H
Real position: Xs= 4400, Ys= 1870, Zs= 2770
Estimated position: Xs= 5100, Ys= 1700, Zs= 3000, ERROR= 756

Fig. 10 Localization by sound pressure method

(2) Localization by correlation method

The location of a leak is made by the hyperbolic method on the basis of a times difference of arrival of signals on various pairs of microphones. The definition of time delay of signals is made by cross-correlation functions.

The cross-correlation functions is calculated in optimum frequency band, chosen by coherence function:

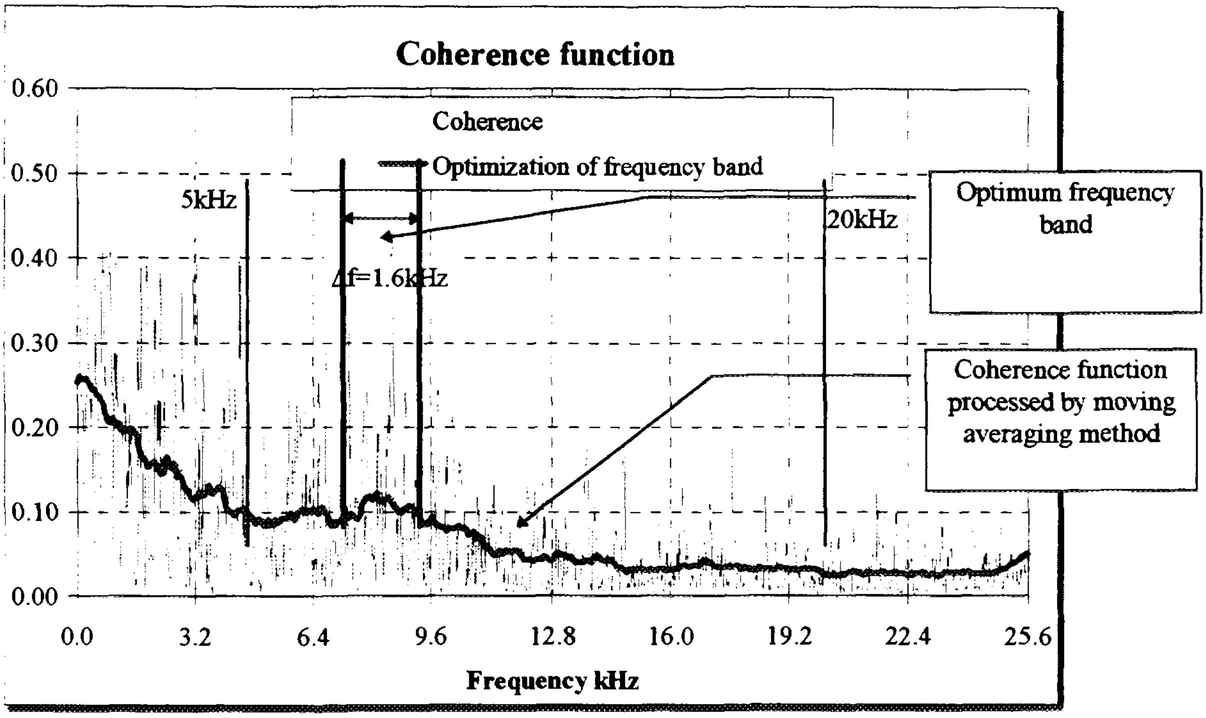


Fig. 11 The definition of optimum frequency band by coherence function

To increase the reliability of definition of signal delay, a several cross-correlation functions are processed in nearby frequency bands:

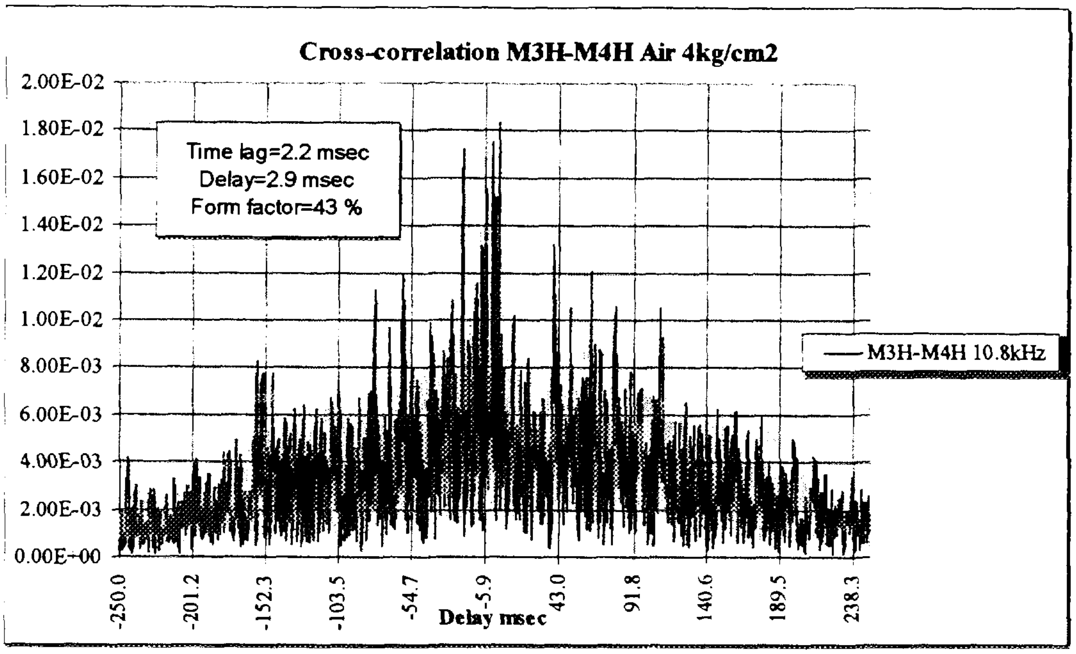


Fig. 12 The cross-correlation function is calculated before processing

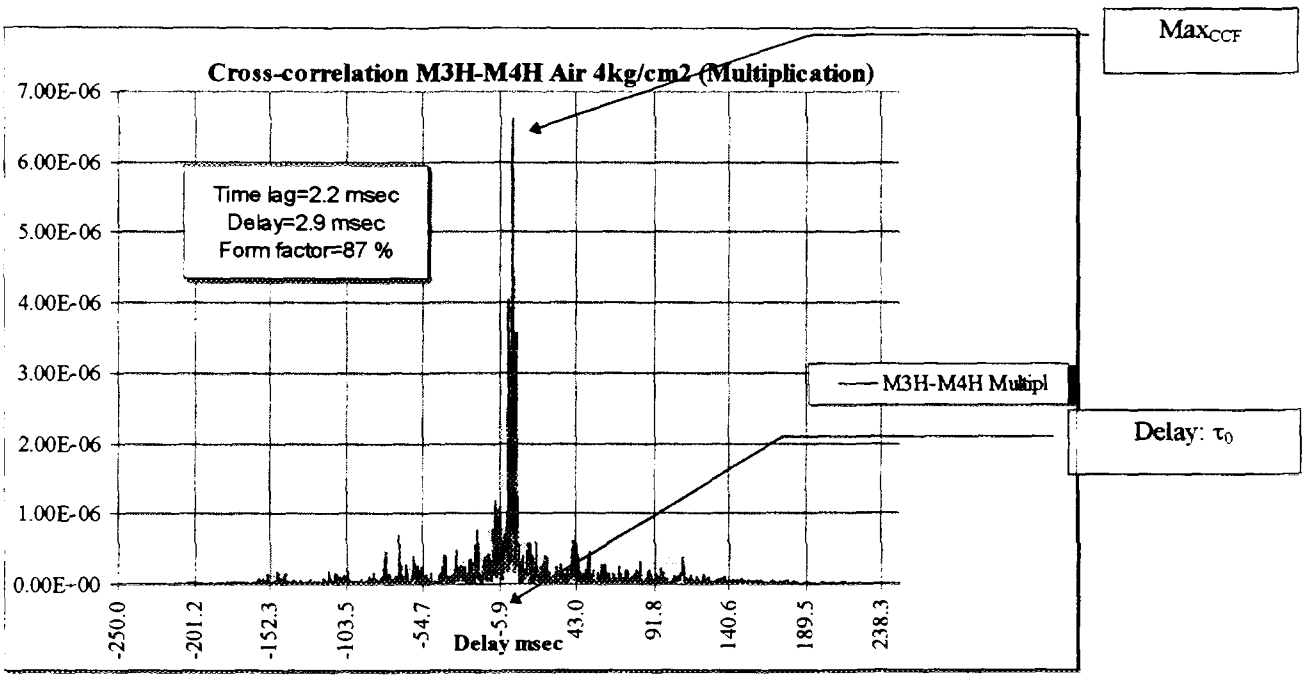
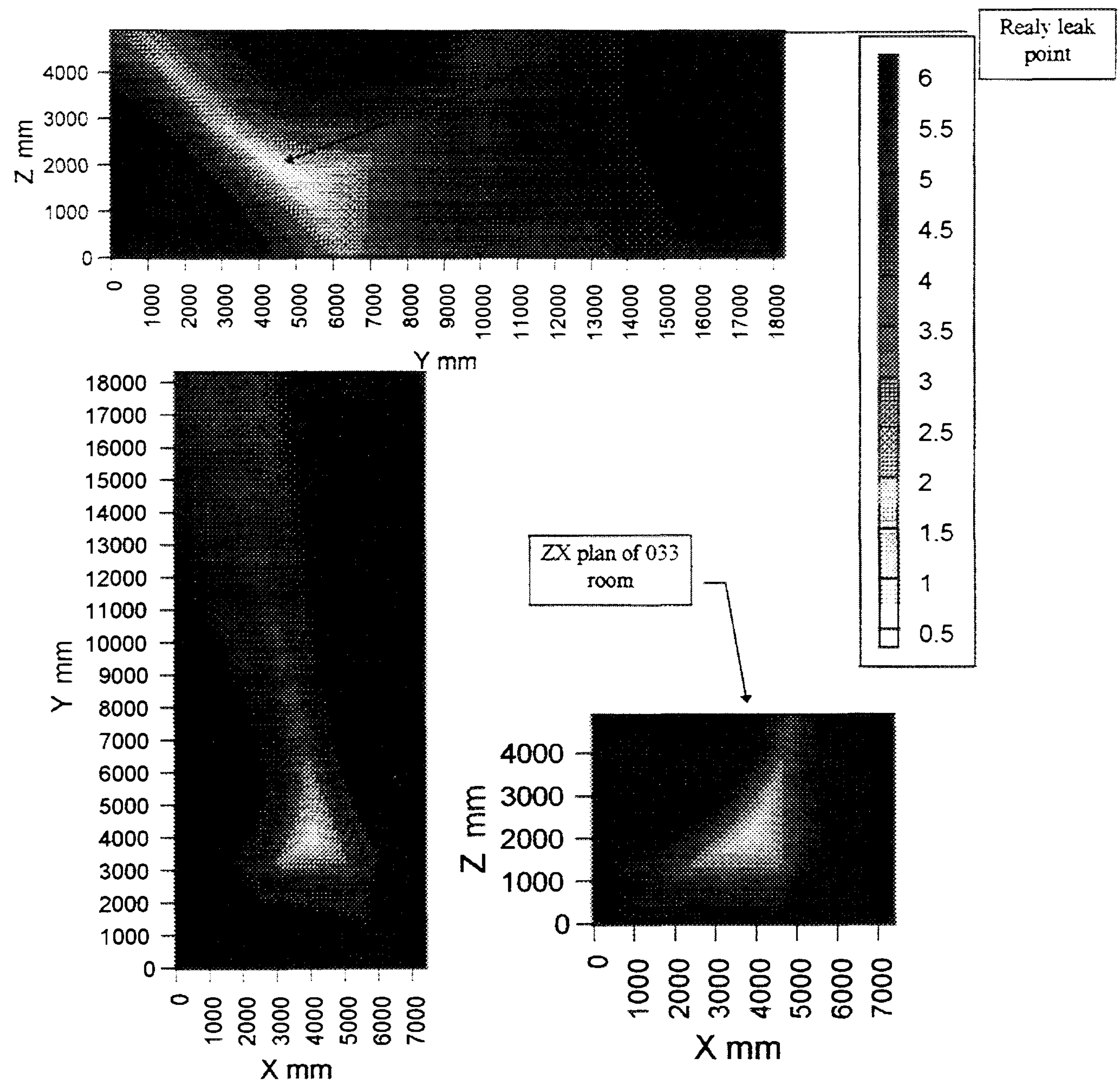


Fig. 13 The cross-correlation function is calculated after processing

The spatial location of a leak is made by the hyperbolic method (fig. 14) on the basis of a times difference of arrival of signals on various pairs of microphones:



Mic. M1L-M2L,M1L-M3H,M2L-M3H,M3H-M4H
Real position: Xs= 4450, Ys= 3100, Zs= 3000
Estimated position: Xs= 3700, Ys= 4400, Zs= 2400, ERROR= 1616

Fig. 14 The spatial location of a leak by the hyperbolic method

As the evaluation results of the NPP test, the following results were obtained:

- the accuracy of leak localization amounted to approximately 1.3 m using the correlation method
- and from 1 to 0.4 m using the sound pressure method

3.4 Leak location by the beam microphone technology

Beam microphone testing has been conducted in room 033/2 of LNPP-1 in the course of «cold» one-channel tests in case of turned on circulation pumps.

The measurements were made according to pitch-by-pitch change in the microphone position in a horizontal plane (horizontal angle) and in a vertical plane (helix angle - elevation).

The operation of main circulation pumps results in powerful background noise. The noise level decreases with the frequency growth, while major part of acoustic energy is concentrated in the range of frequencies up to 8 kHz.

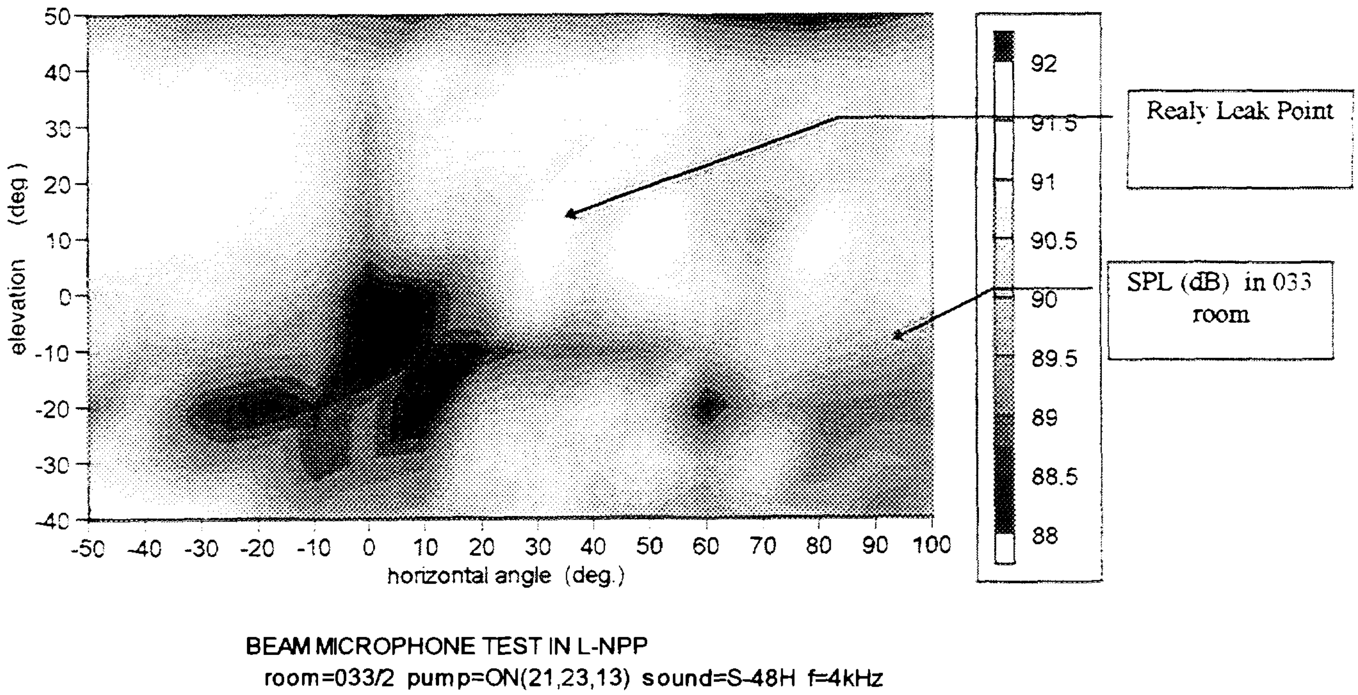
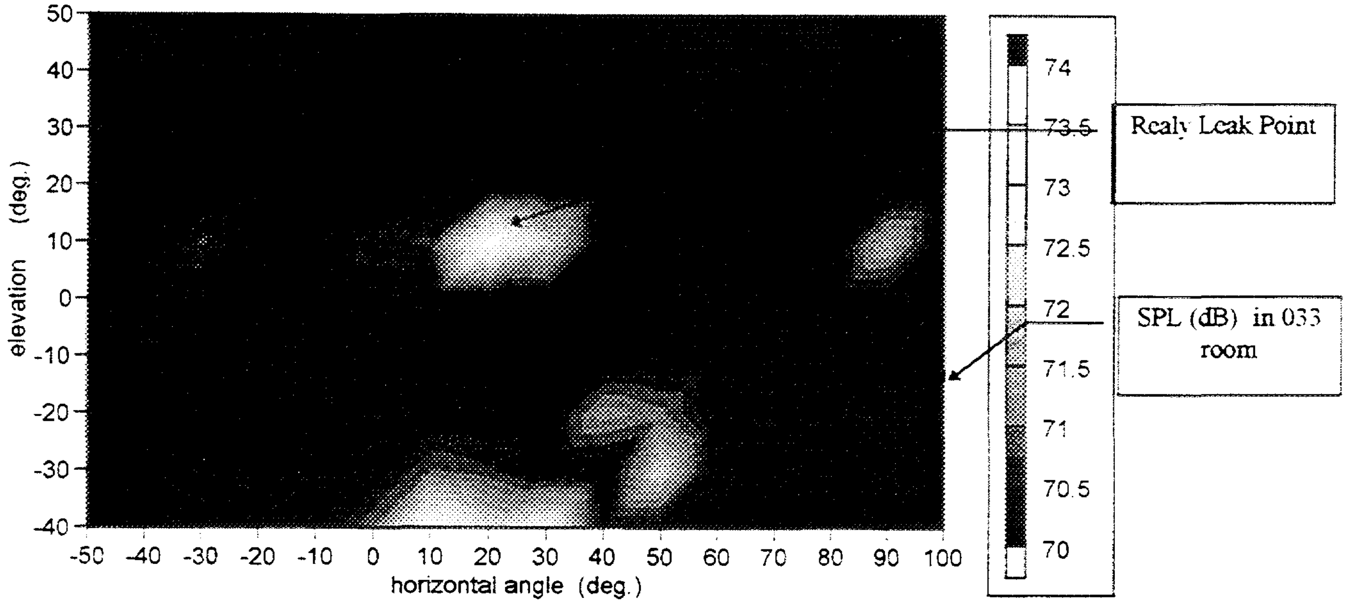


Fig. 15 Sound pressure, recorded on 4kHz

With an increase in the analysis frequency the influence of the background noise starts decreasing. Specifically, starting from the frequency of 8 kHz, the detection and localization of a leak is feasible. And at the frequency of 25 and 31.5 kHz the error in defining the leak angular coordinates does not exceed 50 (fig. 16):



BEAM MICROPHONE TEST IN L-NPP
room=033/2 pump=ON(21,23,13) sound=S-48H f=31.5kHz

Fig. 16 Sound pressure, recorded on 31.5 kHz.

It should be borne in mind that detection and localization of a leak in the process of the tests were performed at a distance of 8.5 m from the source of sound. At this distance the value of 5^0 brings about an error in the spatial localization of about 0.7 m.

4. Conclusion

As the evaluation results of the NPP test, the following results were obtained:

- the reviewed methods are suitable for the detection and localization of leak in RBMK reactor even in the environment of high background noise level;
- and acoustic leak detection system can be successfully used on Leningrad NPP.

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EXPERIENCE AND PROBLEMS WITH IMPLEMENTATION OF DIAGNOSTIC SYSTEMS FOR VVER REACTOR PLANTS

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Justification and keeping of integrity and reliability of the equipment is one of the important task in the problem of NPP guaranteed safety. This task has been solving at all stages of NPP life cycle.

At the NPP design stage a complex of analytical and experimental investigations carries out for substantiation of equipment strength and durability. As a result, design requirements and limitations generate on operation conditions of NPP equipment and systems.

At the stage of comissioning extensive start-up tests and measurements carry out. Their task is to confirm that the design requirements and limitations are satisfied for the equipment of each concrete reactor plant. Such measurements are the tool of early equipment diagnosing as well, because they allow to reveal and eliminate possible non-design conditions before the beginning of commersial plant operation.

At the consequent commercial plants operation the equipment integrity confirms by periodic technical inspections which carries out at annual outages. Such inspections include, in main, control over equipment integrity by non-destructive methods.

In the last decade the equipment integrity monitoring during NPP operation was added to above mentioned kinds of monitoring reasoning from operation experience and appeared standard documents of authorities. Some specific plant equipment should be the object of the operation monitoring. First, it should be the equipment, the life time of which is less than plant age as a whole. Then, it should be an equipment on which are the most probable non-design conditions because of various reasons (NPP staff errors at equipment mounting during outages, changing of equipment fixing conditions during operation, substitution of some components of an equipment, etc). The effects of these and other similar factors can be rether serious for one cycle between outages, because of it the operation monitoring is necessary.

Main assignment of such operation monitoring realized through systems of operation diagnostics is to reveal non-design equipment states at on early stage of their ocurrence when I&C signals can not yet fix these states while non-destructive instrumentation can not yet be applied. Such systems allow to reduce equipment damage, to lower probability of incidents and, therefore, to ensure NPP safety. Besides diagnostic systems allow to estimate and predict a real ageing of the equipment so that to the beginning of the next outage to know "weak" equipment components first of all being subject to repair or substitution, i.e. to provide preventive maintenance. Solving this task the diagnostic systems represent themself as information support of non-destructive inspections, as the final confirmation of malfunctions and faults can be only obtained by instections.

The niche of diagnostic systems in the overall system of NPP reliability measures is shown in fig. 1 at the example of substantiation and keeping of equipment vibration strength.

Majority of diagnostic systems for operating VVER plants (fig. 2) is oriented to methods of noise and acoustic diagnostics, which are the most sensitive to early anomalies detection. At the same time it is obvious that equipment conditions diagnosis using such systems demands advanced techniques and algorithms of diagnosing, as such systems reveal abnormalies and malfunctions by indirect way on equipment response to technological noises, acoustic noises and impulses at appearance of anomalies, etc.

Therefore the development of methods of the analysis and interpretation of signals in diagnostic systems is an independent problem of diagnostics. The objective of this activity is to develop mathematical support of diagnostic systems as techniques and algorithms of diagnosis, diagnostic modes and setpoints.

For VVER plants the mathematical support of diagnostic systems is developed in some stages.

Before delivery of hardware at NPP's some minimum (base) diagnostic possibilities are provided as standard "portraits" of a response of the equipment at know effect as well as diagnostic indications of manifestation of abnormal conditions.

For leak monitoring system, such knowledge base is elaborated in test rigs conditions with simulation field parameters and with organization of inspected coolant leakages through artificially growed cracks as well as through untightness of flange connections. The examples of acoustic noises "portraits" at coolant leakage are presented in fig. 3.

Similary-artificial creation of abnormalies in test rigs conditions-knowledge bases are elaborated for control rod drives diagnostic system, and also system of loose parts detection.

For vibration monitoring system, base diagnostic possibilities are provided by means of a complex of analytical-experimental researches for definition of eigenfrequencies and modes of the equipment depending on its fixing conditions. As a result, there is possibility to identify the most characteristic peaks in the vibration spectra and estimate change of fixing conditions by the values of shiff of these peaks (fig. 3).

For the residual cyclic life time system, the analytical-experimental way provides interrelation between the most stressed components and indications of thermocouples, installed in characteristic points.

After mounting and adjustment of the hardware at NPP system adaptation is carried out for concrete unit condition. The objective is reached by special basic measurements, at which empirical dependences are defined between known entry effects and response of various systems sensors.

After these measurements operational documentation of diagnostic systems should be developed which should contain the list of achieved systems functionalities, the metrics for equipment normal conditions and diagnostic setpoints.

Experience of implantation of the first diagnostic systems at operating Russian VVER plants has shown necessity of defined organizational structure for operation and maintenance of the systems as well as for sequences of actions after diagnostic events.

As a result, the branch system of diagnosis is organized by now. Block diagram of it is represented in fig. 4. According to the scheme, the process of diagnosis and decision making is divided into two stages.

At the first stage this activity is carried out by NPP diagnostic staffs, which are organized at all Russian NPP with VVER by now. NPP diagnostic staff provides operation and maintenance of the hardware, carries out acquisition and express-analysis of the signals and makes preliminary conclusions at diagnostic events using operational documentation of the systems.

At the second stage the analysis of the information should be executed in branch Center of diagnostics with enlisting of various organizations experts. The enlisting of the experts of various profile (on strength and durability of the equipment, on process technology, etc) means qualitatively higher level of the analysis. Besides of that there is the possibility to carry out the analysis not only in time but also on an ensemble one-type reactor plants.

Except above mentioned functions the Center is also assigned for acquisition on diagnostic information from all NPP's, generalization of experience of diagnostic systems using, system operation support, training and checking of NPP diagnostic staff.

By now majority of Russian VVER reactor plants is equipped with systems of components of systems shown in fig. 2, therefore culture of operation raises, experience of their using is typed and in a number of cases abnormal conditions were revealed /1/.

In particular, at two units with VVER small leakages of the coolant were detected using of acoustic leak monitoring system.

Further, temperature monitoring of some characteristic plant components (injection nozzle of the pressuriser, feed water supply nozzles of SG) allows to eliminate non-stationary thermal loading which arose in transients because of NPP staff omissionss.

As the last and the most serious case of diagnostic events we shall consider results of vibration monitoring of reactor internals at unit 1 Khmelnitskiy NPP during comissioning /2/. Increased vibrations of the core barrel and adjoined components were revealed there during hot tests (fig. 5). It has required realization of the whole complex of measures, including:

confirmation of reliability of obtained experimental data;

additional analysis of vibration monitoring results for revealing of the reason of abnormal vibration conditions;

reactor disassembling and conducting of additional inspections and checks of core barrel fixing;

repair works on restoring of design conditions of core barrel fixing;

additional measurements of internal vibrations during start-up.

As a result, core barrel vibrations were lowered to the acceptable level.

However, it is necessary to mark, that some lacks occur during implantations of the first diagnostic systems at operating VVER's. So, monitoring scopes were determined not only by their impact on NPP safety but mainly by technical possibilities of suppliers. Further, the operational documentation on hardware did not fully meet to requirements for automized systems. Besides of that some of the systems were implanted without a due mathematical support.

It is obvious that development of normative base of diagnosing is necessary for elimination of marked lacks. So, concrete normative documents should be developed which should determine the order of diagnostic systems introduction as well as regulate conditions of their putting into industrial operation.

Application of I&C signals with their specialized processing and analysis is another direction of rising of diagnosis quality. On the one hand, it allows to support the indications of systems of noise diagnostics to provide entirety of the diagnosis. On the other hand, it allows to monitor NPP equipment and system at various operation conditions including transients with presence of malfunctions or emergency conditions.

This approach is realized at the moment at diagnostic system development for new VVER plants under design. According to this approach, the diagnostic system is developed as a complex frame, a lower layer of which are systems of noise diagnostics while operator support system is the upper layer. Applied I&C signals are conditionally joined in concept of process diagnostics. Depending on three possible states of the plant (normal, transient, emergency) I&C signals operation; discrete signals, which change plant mode operation; discrete signals resulting emergency means in operation.

The first group of I&C signals is used for early detection of anomalies. Some physical balance correlations (conversation of mass, temperature, energy) are made for these signals /3/. If imbalance occurs, temporary characteristics of signals are used as a image of malfunction (for example, temporary delays of one signal in relation to other, time of exceeding of diagnostic setpoints, etc). The above-mentioned systems of noise diagnostics supply the upper level of the diagnostic system with alarms on detected anomalies in standalone components or units of the equipment.

Besides of supporting of noise diagnostic systems, system of process diagnostics allows to monitor plant parameters, which are defined durability of the equipment, and transfer the lasts in the residual life time monitoring system.

Thus, the combining of noise diagnostic systems with I&C signals allows to describe all kinds and stage of malfunctions. And it is such complex system which allows to enhance and keep NPP safety.

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3. V.I. Pavelko. Statistics methods at preliminary stage of NPP diagnostics, "Issues of Atom Science and Techniques", 1990, №2.

Stages of justification and keeping of vibrational strength of VVER reactor internals

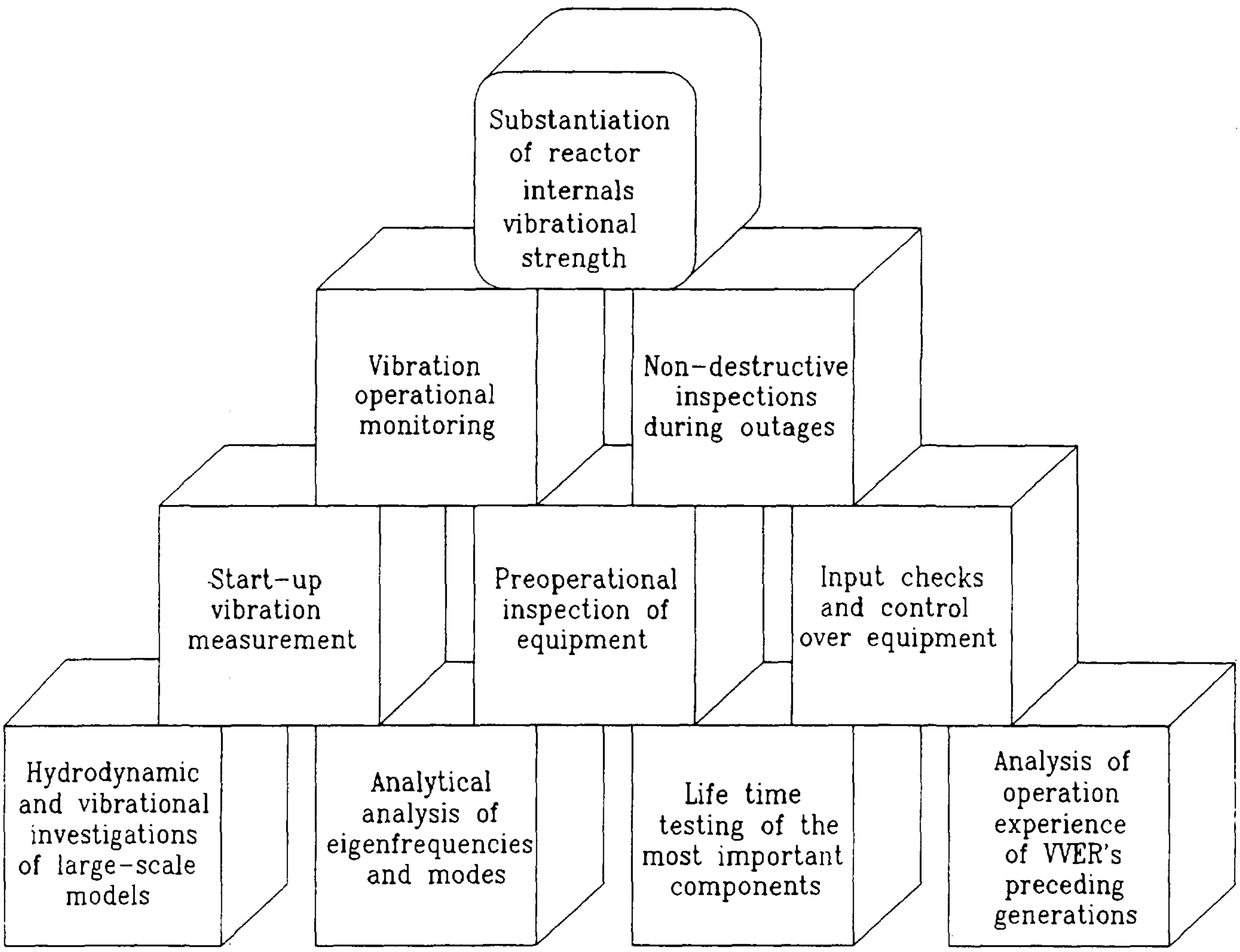
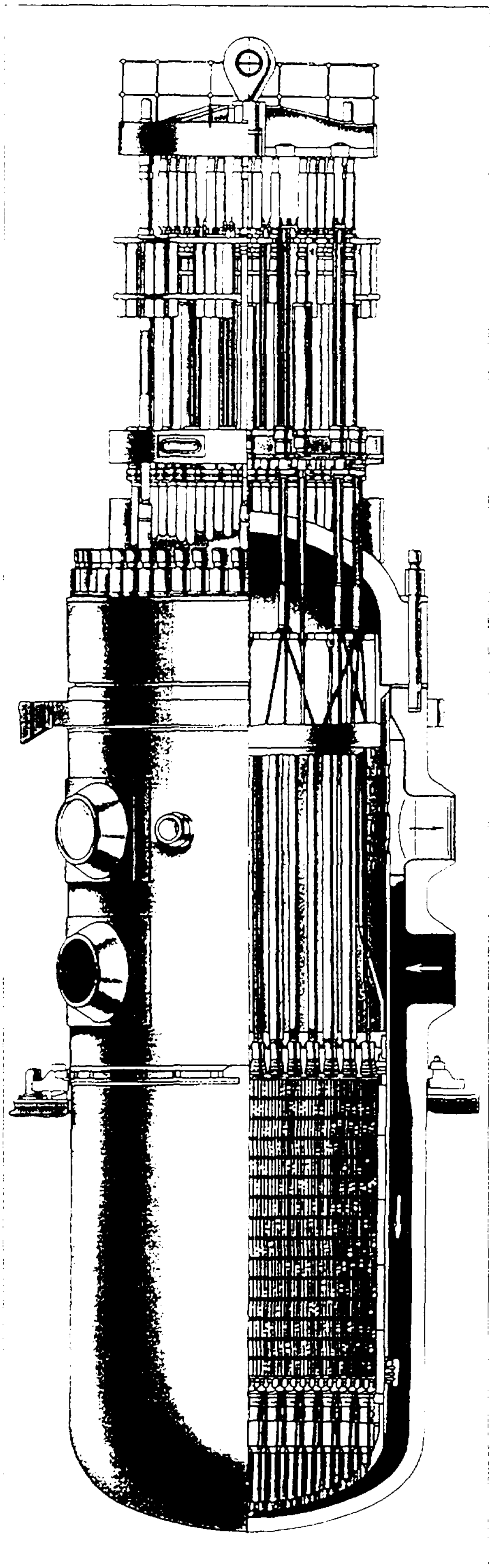


Fig. 1

VVER-1000 Reactor Monitoring and Diagnostic Systems



Leak Monitoring System

Vibration Monitoring System

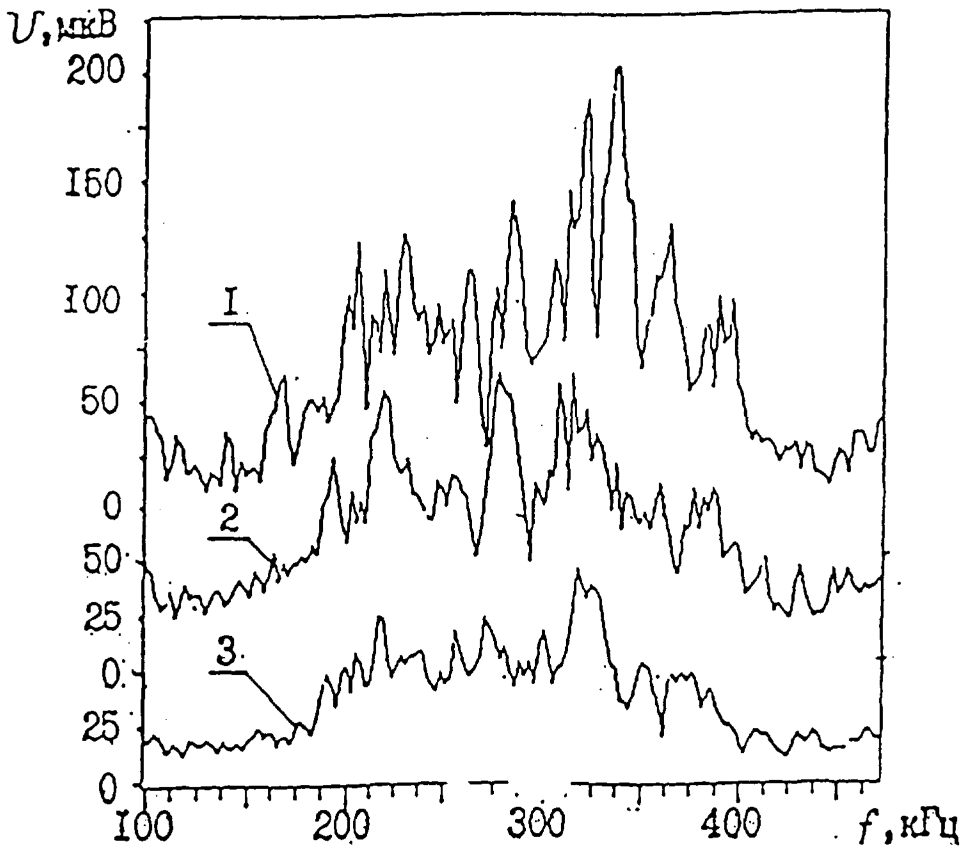
Control Rod Drive Monitoring System

Loose Parts Monitoring System

Residual Lifetime Monitoring System

Fig. 2.

Acoustic noise spectrums during coolant flow through cracks of different lengths at operation conditions (p=15,7 MPa, T=325⊕C)



1 - crack length 100 mm, Q=2 l/min
2 - crack length 70 mm, Q=1,4 l/min
3 - crack length 48 mm, Q=0,9 l/min

Vibration stresses spectrum at core barrel model

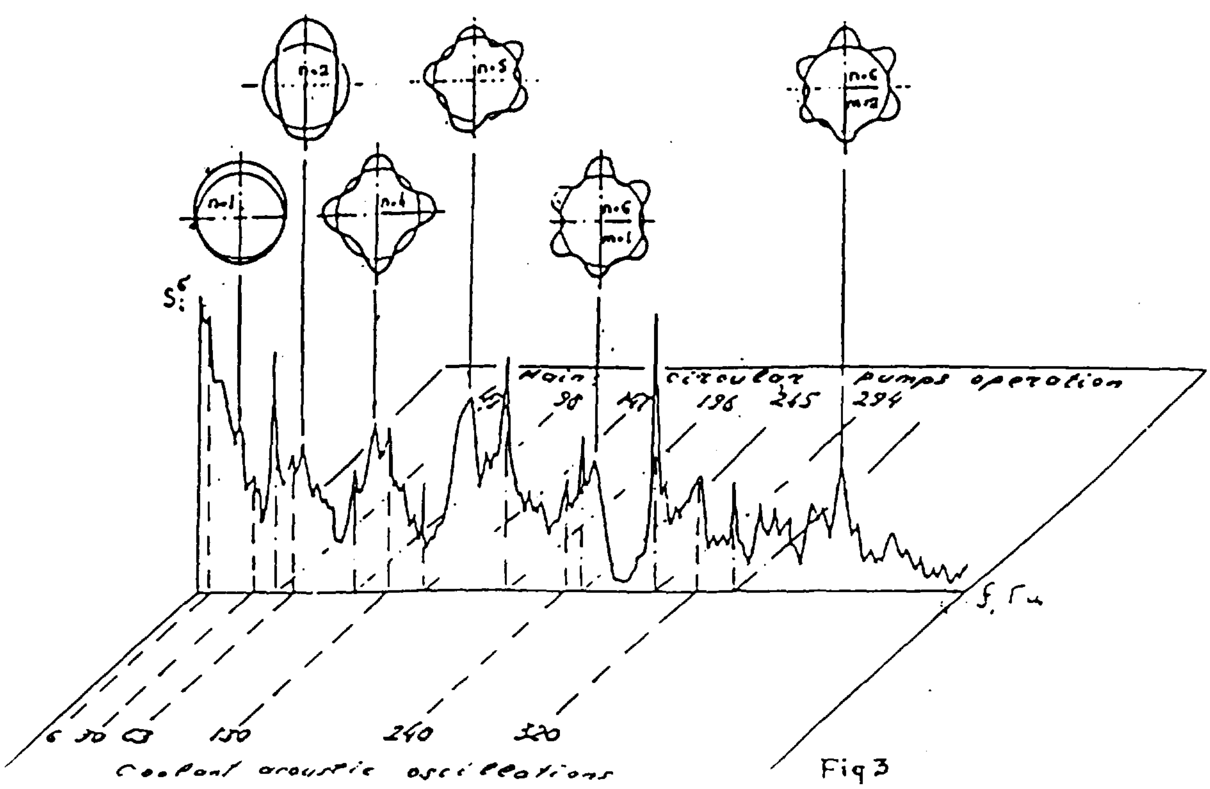


Fig 3

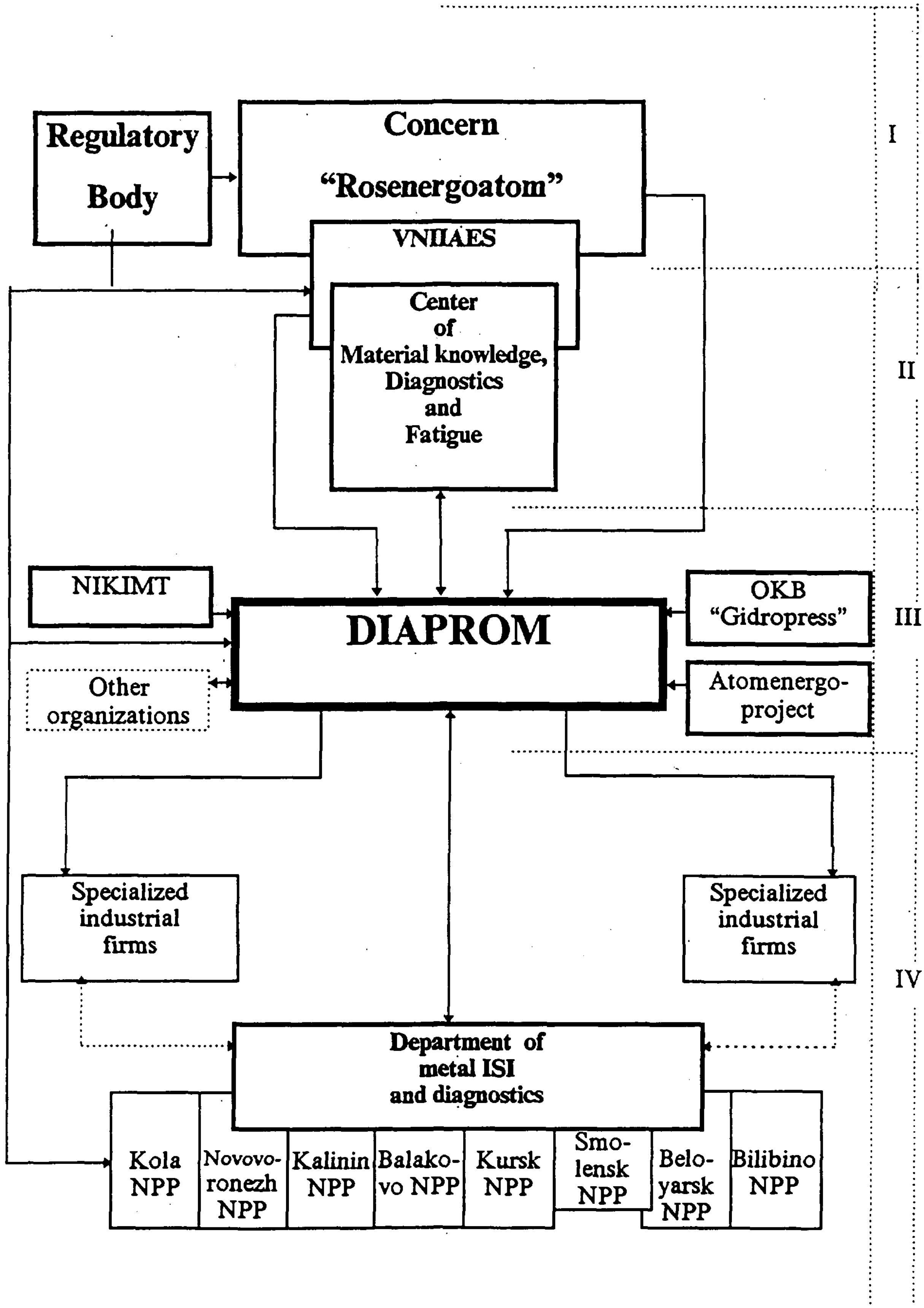
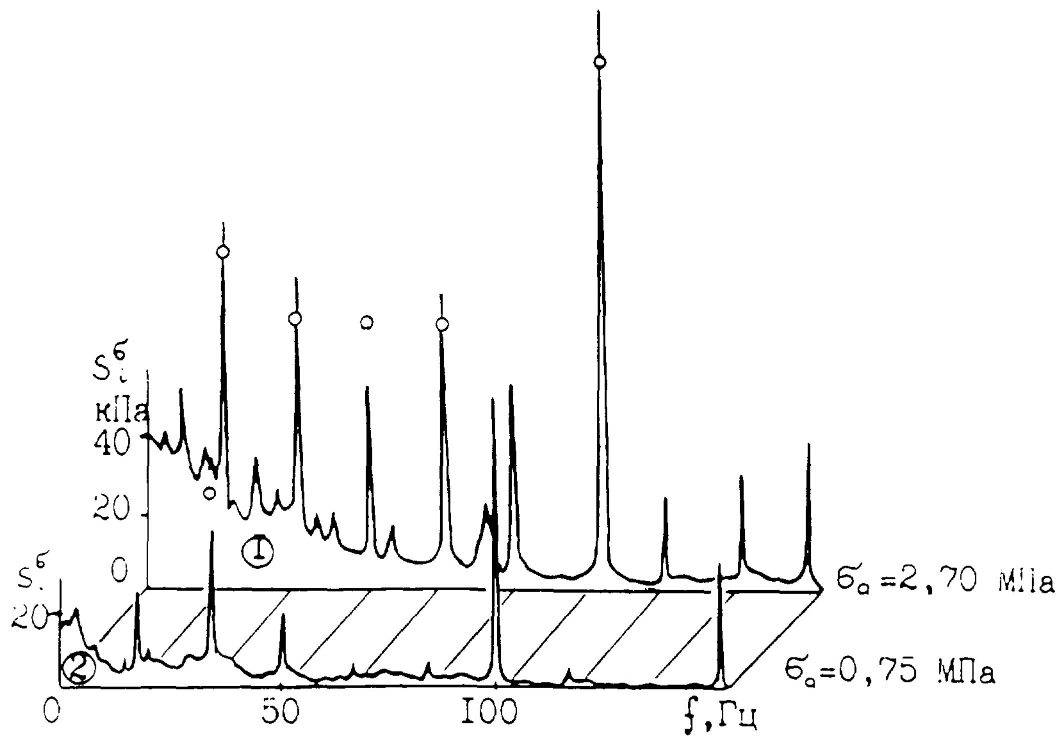


Fig.4 Structural diagramme of the industry's diagnostics system.

Vibration stresses spectrums at core barrel of khmelnitski-1 reactor



1 - measurements during hot tests

2 - measurements during start-up

o - project acceptable values

Fig.5

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ANNEX II:
LIST OF MEETING PARTICIPANTS

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