SPECIALISTS’ MEETING ON
INSTRUMENTATION FOR SUPERVISION OF
CORE COOLING IN FBRs

KALPakkAM, INDIA
DECEMBER 12-15, 1989
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INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH
KALPAKKAM, INDIA
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IAEA-IWGFR SPECIALISTS' MEETING ON
"INSTRUMENTATION FOR SUPERVISION OF CORE COOLING IN FBRS"

SUMMARY, CONCLUSION & RECOMMENDATIONS

1.0 INTRODUCTION

The Specialists' meeting on "Instrumentation for Supervision of core cooling in FBRS" was held at Indira Gandhi Centre for Atomic Research, Kalpakkam, India during 12-15 December, 1989. The meeting was sponsored by International Atomic Energy Agency (IAEA) on the recommendation of the International Working Group on Fast Reactors (IWGFR). The meeting was presided over by Mr. S.B. Bhoje (General Chairman) of India and was attended by 11 participants from Federal Republic of Germany, France, India, Japan, the United Kingdom, the United States of America and the IAEA.

The purpose of the meeting was to provide a forum to discuss instrumentation provisions required for the assurance of core cooling in all operating conditions covering needs for both global and local supervision. The presentations by participants were divided into four topical sessions.

During the meeting papers were presented by the participants on behalf of their countries or organisations. Each presentation was followed by an open discussion in the subject covered by the presentation. After the formal sessions were completed, a final discussion session was held and general conclusions and recommendations were reached.
The summary of the technical sessions and the conclusions and recommendations are given below. The agenda of the meeting & the list of participants are appended.

2.0 SUMMARY OF SESSION I

The first session in the meeting was customarily devoted to national position papers from the participating countries.

2.1 J.Ph. Girard from France presented details of the instrumentation system being used in SPX1 and evolving trend with respect to EPR. SPX1 has already logged about 100 EFPD of operation. The instrumentation system has run well ensuring high reactor availability. The 6 acoustic transducers in the reactor have been validated by cavitation noise in the dummy core; but their validation in the operating reactor using a whistling s/a had to be abandoned due to storage vessel leak. The French experience with neutron noise monitoring indicates that it may not be suitable for large cores.

The improvements contemplated over the existing SPX1 system are with regard to neutronic channels and the DND system. The neutron detectors at the bottom, below main vessel, require neutron guides in the core and impose certain constraints over core catcher leading to higher cost. As an improvement over this, high temperature fission chambers are proposed to be fitted in the above core structure. The DND system in the SPX1 has an average response time of 50 s. In order to improve upon it, it is proposed to locate high temperature DND detectors behind IHX in the inner vessel. This would reduce the response time to
25 s. This concept is being incorporated in the SPX itself for qualification.

Provision of acoustic boiling detectors in EFR is yet to be decided. This detection system is yet to be established to be included in the trip circuit.

2.2 W.O. Steiger from FRG reviewed the status of fast breeder programme in FRG. Two zero energy reactors STARK & SNEAK were built and operated and are already decommissioned. KNKII, a 20 MW(e) fast test reactor, in operation from 1978, has been shut down for the last 2 years for renewing the licence to extend the life time of core and also due to problems related to drive mechanisms of the shut down system. The reactor is likely to operate from Jan. 90. SNR-300, a 300 MW(e) prototype reactor, is ready for fuelling and the fuel elements are also ready. But due to licencing problems, the reactor is yet to start operating. The present FBR R&D activities in Germany are related to the common EFR project.

2.3 S. Govindarajan of India presented the rationale behind the choice of instrumentation system for core supervision of PFBR, a 500 MW(e) prototype fast breeder reactor, planned to be built at Kalpakkam during the next decade. The design limits of core components and various design basis events were discussed. PFBR will have monitoring of all fuel subassembly outlet temperatures by Chromel-Alumel thermocouples. Miniature flowmeters will be installed in 4 positions at core outlet to take safety action in the event of rupture of inlet pipe from
pump to grid plate. DND system will be used to prevent unsafe local blockages due to debris from failed pins. Acoustic boiling detectors and fast response T/C's will be provided for experimental purposes.

2.4 Masuo Sato of Toshiba, Japan presented the details of instrumentation system contemplated for a large scale breeder reactor proposed in Japan. For the Japanese prototype loop type FBR "Monju", temperature monitoring of outlet of all fuel subassemblies and part of blanket is provided. 34 of the positions also have flowmeters at the outlet. Other systems that complete supervision of core cooling are level detection in vessel, neutron detectors and fuel failure detectors. It was also mentioned that the localizing fuel failure by gas tagging takes a long time, about 10 h. A s/a outlet sipping method might be followed in future.

Advanced system contemplated for future FBRs are signal validation, on-line diagnosis for anomaly detection and operational guidance scheme.

2.5 I.D. Macleod of U.K. highlighted the work being carried out in U.K. on detection of anomalies in breeder region and on detection of boiling in core. R&D work on development of co-axial thermocouples with response time of 50 ms was also described. Details of analytical work on predicting the temperature noise at the outlet of subassemblies were also presented. The ultrasonic method developed for detection of temperature anomalies in blanket subassemblies was found feasible.
from the laboratory scale experiments carried out.

2.6 L.J. Christensen explained the efforts in USA to develop inherently safe fast reactors which would be economically competitive also. Current trend is to go for compact modular reactors of size about 400 MWe for each module for enabling factory fabrication, easy shipment etc. The other advantages of smaller sizes were also highlighted. The design details of Advanced Liquid Metal Reactor (ALMR) PRISM being proposed by GE were presented. The IFR program at Argonne was discussed along with the fuel reprocessing program using pyrochemical process. The ALMR will use metallic fuel with oxide fuel as an alternative. The present plan envisages commercial plant design update by the turn of this century. The cost of energy from ALMR is expected to be as cheap as that from LWR and lower than from coal. The metallic fuels have been used in EBR-II and the inherent safety of small size metal fuelled reactors has also been confirmed by various tests in EBR-II.

3.0 SUMMARY OF SESSION II

3.1 EBR-II paper stressed the need for replaceable flow and temperature sensors as several of these have failed during the past 25 years of operation. Unlike other reactors, there is provision only for temperature monitoring of selected subassemblies at the outlet as the fuel subassembly design rules out total blockage of a subassembly. EBR-II has programme to use digital pump speed as a parameter for flow trip and efforts are on to install two replaceable ultrasonic flow sensors in the
reactor low pressure plenum pipes. Several fully instrumented subassemblies with extensive temperature and flow measurement capabilities have been used in EBR-II for generation of data sets to validate thermal hydraulic behaviour of single subassembly and whole core behaviour. Through these experiments it was also shown that EBR-II core can withstand a loss of primary flow without scram and loss of heat sink without scram from full reactor power without core damage. EBR-II has a very simple RSS (scram) system. It will trip on 2/3 seismic detectors, 2/3 nuclear signals, 2/4 outlet temperatures and 2/4 loss of flow. The RSS has two independent sections (A and B).

3.2 The paper on instrumentation for supervision of core cooling for the proposed PFBR stressed the need for additional instrumentation for core monitoring. The use of fast response thermocouples, acoustic boiling detectors and eddy current flowmeters is recommended for PFBR; however these detectors are presently under development.

3.3 KNK paper highlighted the evolution of instrumentation system for core surveillance from KNK-I (thermal core) to KNK-II (fast core) for local and global safety of the core. KNK-II has two diverse shutdown systems actuated by parameters related to neutron flux, temperature, flow, reactor vessel level, leak detection system, DND and ratios of power and primary flow and primary and secondary flows. The paper also highlighted the number of scrams (54 nos.) encountered by the reactor over the past 10 years and their categorization into operator error, presence of gas bubbles and failure of mechanical and electrical
equipment. From this experience it was emphasized that there is also a need for threefold redundant measurement of component protection system, exact failure analysis to prevent recurrence of the event and performance of extensive tests, such as thermal cycling test, at the time of replacement of failed/defective components.

3.4 All the delegates agreed that as the reactors are built to last for a period of 30 to 40 years, there is a need to eliminate the use of non-replaceable sensors and qualify the sensors by rigorous testing for core cooling and supervision.

4.0 SUMMARY OF SESSION III

4.1 The first presentation discussed the R&D programme in U.K. to develop a method, using ultrasonic pulse echo techniques, for measuring the outlet temperature of breeder subassemblies. Measurement depends on the variation of velocity of sound in sodium with temperature. This ultrasonic technique is also capable of detecting temperature noise. Mr. Macleod described the experimental programme which consisted of tests in static sodium, turbulent water jets and flowing sodium and based on the experimental results he concluded that this method gives satisfactory accuracy for temperature measurement and seems to have potential for use in LMFBRs. During discussions, Mr. Macleod indicated that this concept will be tested in PFR.

4.2 In his presentation, Mr. Girard explained the subassembly outlet temperature measurement system provided in
Superphenix I. Thermowells at the outlet of subassemblies contain two Chromel-Alumel thermocouples and one intrinsic SS-Na thermocouple. Signal from one of the two Chromel-Alumel thermocouples is used in one of the two plant protection systems while the second thermocouple is connected to the second shutdown system to provide protection against fuel melting and excessive clad temperature. Mr. Girard briefly described the different computers involved in the evaluation of various parameters. He then explained the features of intrinsic SS-Na thermocouple system and the results obtained from the temperature noise measurements at SPX1. Due to mixing, noise at breeder subassembly outlet was found to be more. Comparison between Cr-Al thermocouple signal and low pass filtered intrinsic SS-Na thermocouple signal from a breeder subassembly was found to be good.

4.3 Mr. Macleod described the R&D programme at U.K. to apply acoustic tomography for the remote measurement of temperature, using an array of ultrasonic transducers. Basic measurement principle depends on the variation of velocity of sound with temperature. After explaining the principle of acoustic tomography, Mr. Macleod discussed the results obtained from a water tank experiment. Results confirmed the feasibility of the method for making steady state and transient temperature measurements. Mr. Macleod explained that this is a basic research work and further tests are necessary to confirm applications in LMFBR.
5.0 SUMMARY OF SESSION IV

5.1 Mr. Macleod explained about the computer based supervision of fast reactor core. The system is a fail safe microprocessor based system utilising low level multiplexing of the signals from fast response (50 ms) coaxial thermocouples. Dynamic logic is used and the correct functioning of the lines is continuously checked. This ensures failsafe operation. Test signals are interleaved with the true signals to produce a pattern which changes with every cycle of the multiplexer. A pattern recognition computer checks that the pattern is correct to confirm the correct operation of the multiplexer. Pulses from the pattern recognition logic are converted to DC signal which are routed to routing logic. The computer based temperature monitoring system (ISAT) is under test on reactors but is not yet used as part of a safety system in UK.

5.2 The second paper presented by Mr. Girard, France was part of the review presentation and focussed on core computer monitoring. The Diagnosis and Detection of Core Defects (DDDC) concept was explained. Two dedicated computers (TRTC) monitor the power generated in every fuel subassembly in the core. The multiplexer of the core thermocouple signals is placed in the reactor building. The digital information is sent to TRTC computer through data link.

A dedicated computer (CAROL) is used for calculating the reactivity balance. Noise analysis on core temperature signals and neutronic signals are being done on another computer (ANABEL). A central computer (CORA) receives information from
all other computers about the process signals. The graphic system in the control room are connected to the central computer (CORA).

5.3 Mr. I.D. Macleod explained the origin of the CRP and the test data used. Dr. Om Pal Singh presented IAEA CRP work on signal processing techniques for sodium boiling noise detection. He explained several conventional and new techniques tested and evolved during the execution of CRP. The techniques were evaluated in terms of their capability to indicate boiling with probability of spurious detection less than $10^{-1}$ per year. In a large core, the signal pattern from acoustic sensors should be memorised. The change in the signal pattern should be 'learnt' through experiments and stored as 'knowledge base' in the expert system.

5.4 IWGPR has agreed to extend the work in another CRP to include steam generator leak detection. USSR will join the existing group for this work.

6.0 CONCLUSIONS & RECOMMENDATIONS

6.1 The meeting provided a useful forum for frank exchange of ideas on 'Instrumentation for Core Cooling'. All participants agreed that good work has been carried out in this area and all the presentations reflected the experience gained. It was also expressed that more such specialists' meetings are desirable.

6.2 Detailed discussions on the various options available for safe and reliable instrumentation for core cooling indicated
the following views:

- The requirements of instrumentation depend on the type of fuel used. Complete analyses have to be done for all the DBAs on fuel, blanket, control subassemblies and fuel at in-vessel storage position to give conclusive requirements of instrumentation.

- An important safety consideration will be whether the melting of one subassembly will cause whole core accident. Additionally, one should consider if melting of one subassembly calls for permanent shutting down of the reactor, i.e. loss of plant investments.

- It was generally agreed that Cr-Al thermocouples with low response time for each fuel subassembly are desirable. These thermocouples should be replaceable and they are reliable to be put into the safety logic.

- Use of Na-SS thermocouple is not recommended for safety logic from considerations of lower sensitivity, noise problems and non-linearity. It can be used for experimental purposes and also for determining in-situ Cr-Al thermocouple response time. The co-axial Cr-Al thermocouple is better than the Na-SS.

- Flow monitoring by eddy current flowmeters is yet to be developed to be considered for inclusion in the safety logic. Their reliability for long term accurate measurement is at present not good enough. They are useful for detecting sudden flow variations.

- Boiling detectors can serve as an alarm to operator and as a trip only for large event like bulk boiling in core. Their usefulness for detection of local boiling is yet to be established. It was also mentioned that advantage must be
taken of the capability of boiling detectors to monitor component vibration, noise due to loose parts in vessel and also for leak detection in steam generators. Another area of application is to monitor breeder subassemblies.

- Ultrasonic method of temperature profiling at the outlet of breeder subassemblies is promising. However, the monitoring requirements of breeder subassemblies for temperature should be based on analyses of DBAs.

- It was expressed that DND system for global detection is required in addition to the localisation system.

- There was a consensus among participants that multiplexing of signals is acceptable. Location of signal processors on pile was also considered as suitable if processors are qualified for functional and environmental conditions.

- During the discussions on reliability of computer software, the participants expressed that this topic would by itself require a specialists' meeting.

- Monitoring of primary sodium flow through reactor was also discussed. It was concluded that direct measurement of primary flow is required as the pump speed is representative of primary flow only when the core configuration is unaltered and the integrity of the pipe as well as pump drive shaft from motor to impeller is ensured. Flow measurement by flowmeter is a more direct method. The replaceability of the flowmeter was stressed by all the participants.
PARTICIPANTS IN THE IAEA-IWGFR MEETING ON
"INSTRUMENTATION FOR SUPERVISION OF CORE COOLING IN FBRs"
KALPAKKAM, INDIA, DECEMBER 12-15 1989

FROM LEFT TO RIGHT—P. SWAMINATHAN, UMA SESHADRI, J.PH. GIRAED, K. RAGHAVAN, L.J. CHRISTENSEN,
MASUO SATO, OM PAL SINGH, C. PARAMASIVAM PILLAI, V. ARKHIPOV, S.B. BHOJE, I.D. MACLEOD,
R.K. VYJAYANTHI, W.O. STEIGER, G. VAIDYANATHAN, R. PRABHAKAR, S. GOVINDARAJAN, R.P. KAPOOR,
G. MURALIKRISHNA, R.D. KALE, AMITAVA SUR.

Titles of presentations

Session I

Chairman : Mr. S. B. Bhoje
Secretary : Mr. J. Govindarajan

National position papers:

1. Core parameter monitoring on French LMFBTR - Requirements, Current design and new trends, J. Ph. Girard et al. (France).
3. Instrumentation for supervision of core cooling in FBR's - Current approach in India, S. Govindarajan & S. B. Bhoje (India).
5. A review of work in U.K. on instrumentation for the supervision of core cooling in LMFBR's, I. D. Macleod (UK).
6. Advanced Liquid Metal Reactor (Prism), L. J. Christensen (USA)

Session II

Chairman : Mr. I. D. Macleod
Secretary : Mr. R. P. Kapoor

1. Experimental Breeder Reactor II (EBR-II) - Instrumentation for Reactor Core Surveillance, L. J. Christensen (USA).
2. Instrumentation for supervision of Core Cooling in FBTR and PFBR, Uma Seshadri et al., (India).
3. The KNK II Instrumentation for Global and Local supervision of the reactor core, W. O. Steiger (FRG).
Session III

Chairman: Mr. L.J. Christensen
Secretary: Mr. R. Prabhakar

2. Core Temperature monitoring in SPX-1, J.Ph.Girard (France) (part of the review paper presented in Session-I).
3. The Development of Acoustic Tomography for Temperature measurement in Fast Reactors, P.Olley (UK), I.D.Macleod.

Session IV

Chairman: Mr. J.Ph.Girard
Secretary: Mr. P. Swaminathan

2. Computer Core Monitoring in SPX 1, J.Ph.Girard (France) (part of the review paper presented in Session-I).

Session V

Chairman: Mr. W.O. Steiger
Secretary: Mrs. Uma Seshadri

Summary of Sessions, Conclusion & Recommendations.
SUMMARY

Monitoring of fast breeder reactors is aimed at safety and control purposes. In order to prevent consequences of main core accident, fast and reliable core parameter monitoring is needed.

Experience feedback is now available for the three reactor cores that are or were operated in FRANCE (RAPSODIE 40 MWh/loop, PHENIX 250 MWe/pool and SPX1 1200 MWe/pool). R and D work have been performed for the 1500 MWe SPX2 project and is under progress for the future European Fast Reactor.

The major features and outcomes of main core parameters are:

- Neutron flux monitoring: cold chambers outside of the plenum were used with success in the past, high temperature chamber were temporarily used on SUPER-PHENIX and are now in a qualification phase for continuous monitoring of core flux from handling state to full power. They should be placed in the upper plenum.

- Delayed neutrons emitted by the failed fuel fission products is performed today through a sodium sipping system. Though highly sensitive this method should be replaced in future core by neutron measurement system directly integrated on the pool. Simplification, lower cost and faster detection should result from this new concept.

- Temperature monitoring is performed through individual S/A control and automatic processing by a computer. 500 S/As temperature can be checked every second. Main efforts are today concentrated on economic basis with attempt to reduce wiring length.
- This main core parameter surveillance is completed by other measurements such as flowrate, acoustic monitoring, and using advance processing of the data to control the reactivity (reactivity balance meter), to calculate indirect core parameters and perform early detection through noise analysis.

Fast and reliable detection systems are needed. High availability, redundancy through duplication or diversity is required. With an equal safety assessment, lower cost equipment have to be design. Here are some of the aspects under consideration for running and future plants.

I - PURPOSE OF CORE PARAMETER MONITORING

Core parameters are both monitored for safety and operational purposes. Safety regulations lead to limits in radioactive release in the atmosphere in normal and adverse conditions. These limits are usually expressed versus probability of occurrence. Though it is the final consequence of a given sequence the maximum authorized release usually leads to limitation of core parameters such as:

- local and total power, power slope,
- clad temperature, usually before sodium temperature,
- linear power in individual pin,
- failed fuel detection.

Operational purposes, for instance to produce a given power at a given time and/or to extract the maximum power of a given S/A load, require also to monitor core parameters. Physical quantity that are of importance could this time be:

- total power including mean and increase of temperature of the primary loop in order to satisfy steam conditions,
- power distribution throughout the core to optimize power production.

Physical quantities to monitor, are elaborated using a limited number of transducers measurements. Missing informations could be fed to surveillance computer using simulation code in order to define threshold for automatic scram system or for pilot advice.

Main signals directly monitored are:

- instantaneous neutronic power,
- individual S/A temperature,
- inlet temperature,
- total flowrate - primary pump speed,
- delayed neutron emission.
Core parameters are then calculated taking into account computed individual S/A temperature and clad temperature coefficients (versus burn-up), individual S/A flowrate and so on.

II - INSTRUMENTATION FOR GLOBAL SUPERVISION

II.1 - Neutronic Power Measurement

Nuclear measurements are of course of main importance for global supervision of core. Both PHENIX and SUPER-PHENIX are equipped with cold neutronic chambers. For instance the neutronic monitoring of SPX1 core is done through neutron collector and chambers located under the main vessel and safety tank. Helium comptrollers are used for low power measurement, fission chambers for high level and an ionisation chamber for control room display and study of neutronic fluctuations. Monitoring has to be performed from 1 W (handling condition) to 3000 MW that is over $10^{10}$. An additional apparatus equipped with high temperature fission chamber was used to meet the very low compting rate of the start-up core (.5 c/s).

The core is monitored with two diversified systems of three sets of chambers. Automatic scram due to over-power, reactivity (positive and negative threshold) and neutron multiplication factor is done by 2 out of 3 on each system.

For the future reactors it is intended to use high temperature fission chambers directly installed in the hot plenum above the core. This should enable the designer to build a reactor with an anchored safety tank, withdraw the neutron guide and nevertheless control the power from a few watts to full power. Cross qualification of French and UK chambers is underway on the German reactor, KNKII, within the frame of the FBR European agreement.

II.2 - Failed Fuel Detection

This is a typical safety measurement as failed pin could induce in case of fuel release local S/A accident. Small core could easily be monitored by gas tagging techniques. For larger core global delayed neutron detection has been chosen. SPX1 core is thus controlled by fission product detection in sodium. Sipping of sodium is performed through two systems of pipes with, on each system, a diverse set of three chambers located over the foof. Sodium is fed to the measuring block by an electromagnetic pump equipped with a flowmeter. Scram is automatic with a 2 out of 3 vote for each channel. Response time of this system, due to the sodium transit from the S/A to the detector, is estimated to 50 s for a small rupture.
The availability of high temperature neutronic chambers leads to a new design of the failed fuel detection system. Called Integrated DND and tested on SPX1, this new design includes a set of three high temperature chambers "hidden" behind each intermediate heat exchanger. Less sensitive but faster this new design is the basic option for the next European breeder.

II.3 - Complementary whole core monitoring system

Cover gas monitoring, sodium flowrate, sodium level and acoustic measurement are also performed. In this field only few new concepts should be developed for future plants. Accuracy of flowmeter is around 2%, both permanent magnet and eddy current flow meters are used on French power plants.

III - LOCAL SUPERVISION OF THE CORE

Individual S/A monitoring is performed on existing and scheduled French and European plants by two different types of instruments:

- for safety and control purposes each fuel S/A is fitted with a set of thermocouples and,
- secondly, in order to be able to start rapidly again the plant after a clad failure each S/A is equipped with a pipe and a rotating sipping system to localize the failed S/A.

The thermocouples used are either thimble type TCs or immersed probes. Time constant of SPX1 type K thermocouple is 1 s, vote is done in 2 out of 2 for individual monitoring and 1 out of 2 for global monitoring of increase of temperature and instantaneous clad maximum temperature. The precision of the measurement is the intrinsic precision of the thermocouple and the process error due to fluid mixing at the S/A exit. Scanning of the 500 twinned thermocouples is performed every second with two different computers. Each probe equipped with two TCs is easily replaceable.

IV - DATA PROCESSING AND COMPUTER USED FOR PARAMETER MONITORING

The SUPER PHENIX core safety is organized around the so-called 'DDDC' Core Diagnosis and Defect Detection system including, apart from the plant computer a set of five interacting computers devoted to core surveillance.

The two first are safety computers linked to thermocouple measurements (Fast Temperature Processing Computer - TRTC), then the central Core Data Surveillance Computer (CORA) which receives all core
measurements, and calculates all individual S/A characteristics (S/A power, linear power, clad temperatures) together with complementary global quantity (temperature histogram, proportional DN level, failed fuel localisation processing). In addition this computer has a screen in the control room for operator information of the core status.

The two last computers elaborated not compulsory surveillance:

CAROL the reactivity balance meter and, ANABEL the parameter noise monitoring computer dealing with vibration, acoustic, neutronic and temperature fluctuations.

V - OPERATIONAL EXPERIENCE

V.1 - Availability

Redundancy of safety equipment, regulations on partial non-availability of equipment did not lead to any loss of production on either PHENIX and SUPER PHENIX. Spurious trips induced by operator error in the use of the system (during periodic maintenance for instance) have been only few for all systems over the 10 years of PHENIX and the first years of SPX1 start-up.

V.2 - Efficiency

Most scram algorithm are designed to prevent consequence of global or local overheating. An individual temperature monitoring equation protect the S/A from plugging.

Efficiency of protection system was proven during early failures of fuel and a small number of rod drops.

Monitoring of the margin towards plugging was shown to be efficient with the detected swelling of a prototype S/A on PHENIX, and a plug S/A during the start-up of SPX1. The use of a special differential algorithm on both reactors is very sensitive and both detections were performed before the automatic scram occured.
6 - CONCLUSION - FUTURE TRENDS

Few major innovative features are under development for future plants. Core monitoring rely mainly on nuclear and thermohydraulic measurements. With the same level of safety simpler and cheaper systems are now being researched.

As far as thermal monitoring is concerned the central computer implying a lot of wiring should be replaced by a network with localized data acquisition and treatment units directly on the ACS.

Neutronic and delayed neutron chambers should be located inside the main loop, directly in sodium, thus reducing all transfer devices from the core to controlling devices on top of the reactor roof for DN signal and the bottom of the reactor building for neutron monitoring probes.
Figure 2: CORE SURVEILLANCE AND PROTECTION SYSTEM OF SUPER PHENIX
Up to this day 4 fast nuclear reactors were built in the FRG:

STARK, SNEAK, KNK II, and SNR 300.

STARK and SNEAK, two zero-power reactors, built and operated by KfK, are decommissioned.

KNK II, ordered by KfK, built by Interatom and operated by KBG, started operation in 1978.

Due to licensing and technical problems the plant has been shut down the last two years. One year of shut-down was caused by the licensing procedure together with the life extension of the second core-loading from 455 up to 720 EFPDs.

Another year of stoppage was caused by torque increase problems of the driving mechanism as well of the 5 shim-safety-rods as of the 3 diverse safety-rods. The requalification of the driving mechanism was a very time consuming task.

The restart of the plant is scheduled for January 1990.

SNR 300 is ready for fuelling the reactor. The fuel elements were manufactured and the sodium systems are in operation. The licensing procedure however is in an impasse. The State Government of NRW, which is the licensing authority, is presently not willing to give the license for the next step, the fuelling of the reactor. There is a quarrel between the Federal and the State Government about the validity of the German atomic law together with plutonium breeding FBRs. A decision on this subject of the Supreme Court is expected in spring 1990.

For the future the FRG supports the Common European projects for an EFR. The R&D programs of KfK are already mainly directed on this project. So Mr. Marth, the former head of the FBR-project within KfK, has become Executive Director of the Management Group for R&D in this European Cooperation. The activities of KfK in this field are merged now with R&D programs in the field of light water reactor safety. Thus a new KfK project Nuclear Safety Research (PSF) will be formed. (More details on EFR were given by Mr. Marth during the 22nd IAEA-IAEA Annual Meeting, April 18-22, 1989, Vienna).
1. INTRODUCTION

Safety in Nuclear Power Plants is ensured by adopting a 3-level defence-in-depth approach. The first level involves adequate design of the systems with margins of safety. The second level concerns monitoring of the plant during operation and taking appropriate safety actions to prevent accidents and damage to components. The third level attempts to mitigate consequences of accidents. It is obvious that the three levels are inter-related and the requirements of instrumentation and control will have to be tailored to fit into a comprehensive plan for safety.

Apart from safety requirements, instrumentation is also required as an aid to operator to monitor the health of a system and to provide valuable data to the designer to verify his design and to improve upon it.

As far as core cooling is concerned, the major objective of monitoring devices is to prevent large scale core-melt down due to loss of flow, resulting in a major radioactivity release from the primary containment. Though the concern is mainly with regard to a global event, the possibility of
fault - propagation from local event necessitates monitoring local malfunctions too.

It is recognized that core cooling cannot be judged in isolation without reference to heat generation. The adequacy of core cooling in fact means proper balance between coolant flow and heat generation.

While considering the requirements of instruments to supervise core cooling it is necessary to consider events involving power increase in addition to those involving flow reduction.

While the main requirement is detectability of an event by an instrumentation system, reliability and replaceability without impairing primary containment integrity are also important.

2. DESIGN BASIS EVENTS

The first step in defining instrumentation requirements is to consider the design events that affect the core cooling. These events can be classified into two large groups - global events and local events. Under each category they can be further divided into four categories according to probability of occurrence and consequences. A typical list of events is shown in Table I.
Table I: Design Basis Events for core cooling

Global:

Normal

Upset - Inadvertent control rod withdrawal
- Argon bubbles in core
- Oil leakage into primary circuit
- Pump slow down

Emergency

Faulted - Primary pump seizure
- Loss of main heat transport loops
- Guillotine rupture of inlet pipe from pump to grid plate

Local

Normal

Upset - Local blockages from fuel debris from failed pins

Emergency - Fuel loading error
- Orifice assembly error
- Gross enrichment error

Faulted - Local blockage from external debris
- Local enrichment errors
- Gross inlet or outlet blockage of a S/A

3. CRITICAL REGIONS IN THE CORE

A typical fast reactor core consists of several regions and a large no. of subassemblies. The core lay-out of the 500 MW(e) Indian Prototype Fast Breeder Reactor proposed to be built during next decade is shown in fig.1. There are 181
fuel subassemblies and 12 control rods in the central core region surrounded by blanket, reflector and inner shielding subassemblies. Last of rows of subassemblies cooled by coolant flow from pumps is reserved for spent fuel storage. The outer shielding subassemblies are cooled by natural convection.

The large no. of subassemblies makes it impossible to monitor each subassembly for local events. The priority with regard to local monitoring goes with reference to the probability of a local event propagating to large scale fuel melt down. In this respect fuel subassemblies alone require monitoring. Coolant boiling and fuel melt down in the fuel region could lead to reactivity increase due to positive void coefficient and core compaction, resulting in global power increase and large scale fuel melt down. Therefore it has been decided to confine local monitoring only to fuel region in PFBR.

4. DESIGN LIMITS

The following design limits are imposed to define instrumentation requirements with regard to the design basis events.

1. No fuel melting or coolant boiling in fuel region for any event

2. Clad hot spot temperature < 800°C for upset events

3. Clad hot spot temperature < 700°C for normal events
5. **INSTRUMENTATION FOR GLOBAL EVENTS**

Global power increase and flow reduction result in coolant temperature increase at the core outlet. However, detection of the event by temperature measurement is a slow method because of large response time of thermocouples. Detection of power increase by neutronic channels and of flow reduction due to pump slow down or seizure by pump rpm monitoring should be the first level defence and temperature measurement can be at best the second level defence. However, the guillotine rupture of inlet pipe from pump to grid plate cannot be detected at the pump level and will require flow monitoring at the core outlet. It is proposed to install miniature eddy current flow meters at the outlet of a few selected subassemblies to monitor this event.

6. **INSTRUMENTATION FOR LOCAL EVENTS**

Events like incorrect unsafe loading of a subassembly, gross enrichment error in a subassembly, orifice assembly error in a subassembly and BOL gross blockage in a subassembly can be safely detected by monitoring the individual subassembly outlet temperature at reduced power level. Conventional Chromel-Alumel thermocouples are adequate for this. However, local enrichment error in a subassembly or local blockage in a subassembly cannot be detected by monitoring the outlet temperature by slow response thermocouples. A temperature noise analysis system using fast response thermocouples is called for to detect such local anomalies. Though this
method looks feasible, enough experience has not been gathered on this as yet.

Fuel debris from failed pin can be detected by DND system before they accumulate beyond allowable limits. The likelihood of external debris resulting in local blockage has to be realistically assessed. The local enrichment error should be made impossible by proper manufacturing procedure.

Gross blockages during operation are highly unlikely because of design provisions like radial coolant entry, multiple hole entry etc. Any blockage can at best reduce the coolant flow in a subassembly only to a limited extent. For example a blockage of 50% of entry reduces the flow only to an extent of 10%. Thermohydraulic studies done for PFR have shown that very large sudden blockages can be tolerated even for slow response thermocouples. For a slowly developing gross blockage very large flow reduction can be permissible. Table 2 summarises the results of thermohydraulic studies.

Table 2: Gross blockage thresholds for design limits

<table>
<thead>
<tr>
<th>Event</th>
<th>Clad hotspot C</th>
<th>Max sodium temp C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum permissible flow reduction % for sudden blockage</td>
<td>32 (37.5)*</td>
<td>71 (77.5)</td>
</tr>
<tr>
<td>Maximum permissible rate of reduction (%/s) for 96% flow reduction</td>
<td>5 (15)</td>
<td>16.6 44</td>
</tr>
</tbody>
</table>

*Numbers in parantheses correspond to nil response time in detection system.
It is seen from table II that the advantage in having a fast response thermocouple is not very significant over conventional slow response thermocouples. Considering improbability of large gross blockages, slow response thermocouples are considered adequate to take care of these events.

7. BEYOND DESIGN BASIS EVENTS

The discussion so far covered only events that have probability of 10^-6 or more for occurrences. It has to be estimated whether total instantaneous blockage of a subassembly can be completely ruled out or should be accommodated in the design. Such an event can be detected by miniature flow meter at the outlet of each subassembly or by having acoustic boiling detectors. An indirect method would be to detect the increase in outlet temperatures of adjacent subassemblies due to inter-subassembly heat transfer. Recent Scarsbee experiments have shown that one could detect such an event by DND also. Decision is yet to be taken on whether to provide for TIB in the design and in such an event what detection method should be employed to contain damage.

8. SUMMARY

Instrumentation for supervision of core cooling in PFBR will be as follows:

i. Subassembly monitoring will be confined to fuel region

ii. Individual subassembly outlet temperature will be monitored by conventional chromel-alumel thermocouples.
iii. Local blockage due to fuel debris originating from pin failures will be detected by DND system.

iv. Global core cooling will be monitored by providing miniature flow meters at the outlet of 4 subassemblies in the outer fuel region in addition to the thermocouple.

v. Acoustic boiling detectors will be provided as an experimental measure to study the feasibility and gain experience.

vi. Fast response thermocouples are proposed to be provided one for each subassembly along with 2 conventional thermocouples in order to study the efficacy of such a system to detect local blockages or local enrichment error.
<table>
<thead>
<tr>
<th>SYMBOL</th>
<th>TYPE OF SUB ASSY.</th>
<th>No.</th>
<th>MASS PER ASSY., Kg.</th>
</tr>
</thead>
<tbody>
<tr>
<td>FUEL (INNER)</td>
<td>91</td>
<td>245</td>
<td></td>
</tr>
<tr>
<td>FUEL (OUTER)</td>
<td>90</td>
<td>245</td>
<td></td>
</tr>
<tr>
<td>CONTROL (TYPE I)</td>
<td>9</td>
<td>200</td>
<td></td>
</tr>
<tr>
<td>CONTROL (TYPE II)</td>
<td>3</td>
<td>200</td>
<td></td>
</tr>
<tr>
<td>BLANKET</td>
<td>100</td>
<td>320</td>
<td></td>
</tr>
<tr>
<td>REPELLEATOR</td>
<td>78</td>
<td>355</td>
<td></td>
</tr>
<tr>
<td>B,C SHIELDING (INNER)</td>
<td>78</td>
<td>185</td>
<td></td>
</tr>
<tr>
<td>STORAGE LOCATION</td>
<td>100</td>
<td>245</td>
<td></td>
</tr>
<tr>
<td>STEEL SHIELDING</td>
<td>869</td>
<td>330</td>
<td></td>
</tr>
<tr>
<td>B,C SHIELDING (OUTER)</td>
<td>426</td>
<td>265</td>
<td></td>
</tr>
<tr>
<td>IN VESSEL TRANSFER POSITION</td>
<td>1</td>
<td>---</td>
<td></td>
</tr>
</tbody>
</table>

Fig. 1. PFBR Core Plan
Development Status of Japanese Instrumentation for Supervision of Core Cooling

Masuo Sato

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Isogo Engineering Center
Toshiba Corporation
1. Introduction [1]

For safe and reliable operation of fast breeder reactor, the monitoring and control system should support the operator with regard to following works.

1. to monitor and understand the plant status
2. to diagnose the plant status
3. to perform corrective actions, if necessary
4. to confirm the recovery of the plant

Safety parameters are important relating to the operating safety of the nuclear plant, providing the operators with following informations on the reactor safety status relating to the core cooling.

1. Reactivity control integrity
2. Fuel integrity
3. Core cooling integrity
2. Supervision scheme of core cooling

[2],[3],[4],[5]

2.1 Instrumentation features of core cooling

According to classifications by installation sections, fast breeder instrumentation for supervision of core cooling consists of (1) fuel sub-assembly outlet temperature sensor, (2) in-vessel level meter, (3) flowmeter, (4) neutron detector, and (5) fuel failure detection system. These instrumentations are classified into reactor protection system, control system and monitoring system.

In order to perform these functions, these instrument signals are processed in a computer system for plant control and monitoring. As these instruments dominate plant operational performance and its safety in the course of instrumenting various plant process values, high accuracy, high sensitivity, high reliability and easy maintenance are required. Furthermore, quick response and on-line calibration ability are required, especially for reactor protection system instruments.
2.2 Advanced monitoring and anomaly detection scheme

(1) Signal validation function [6]

The information provided through operator-machine interface should be qualified and validated to avoid mis-indication or mis-understanding caused by sensor failure. The following techniques are applied for input signal validation.

- Signal range check
- Status monitoring of signal input modules
- Signal comparison among redundant sensors
- Signal validation by analytical redundancy
- Signal fluctuation monitoring

By proper installation of above functions according to the importance and characteristic of each parameter, the reliability of informations through operator-machine interface can be much improved.

(2) On line diagnosis for anomaly detection [6],[7],[8]

A nuclear power plant contains various inherent noise sources. Process variables, excited by these noise sources, fluctuate through the plant response characteristics. Hence, these fluctuations contain information about the plant operating state. Noise analysis techniques can extract this information in the form of APSDs (auto-power spectral densities), coherence functions, etc. These noise analysis results respond sensitively to the change in plant operating state and
show various patterns in each state. If the relation between the noise patterns and the plant operating states were known, the present plant state can be identified by comparing the newly obtained sample noise patterns with reference noise patterns.

Among the various methods, the transit time measurements by using the inherent temperature noise or flow noise are considered to be most convenient because of the direct measurements with simple equipments.

In addition, the thermocouples are widely used sensors in nuclear power plants and therefore the transit time measurements by temperature noise can be applicable to various cooling systems.

The temperature transfer function between two measuring points can be obtained from a simple time-dependent heat balance equation.

Fuel assemblies used in an LMFBR are exposed to thermal-hydraulically severe environment and there is a possibility that an anomaly in core assembly will cause a fuel failure accident. So it is important to detect such an anomaly in its early stage.

It is known from many ex-core experiments that local core accidents such as local blockages or local boiling raise the amplitude of coolant temperature fluctuation at the outlet of the assembly. It is necessary for highly sensitive detection to reduce the background component independently of operating conditions. Advanced method for reducing background component from outlet temperature fluctuations are developed.
2.3 Operational guidance scheme [1],[9]

Based on the monitoring capability, we have come to the stage to make the operation guide to recover the plant status to the normal status in correspondence to three different plant operational conditions, i.e.,

1. Mode I: Normal operation
2. Mode II: Operation in case disturbances happen
3. Mode III: Operation after the plant trip

In the operation mode I, the main goal of the plant operation is to continue the present normal operation, keeping various operational limits.

In the operation mode II, the goal of the plant operation is to prevent the disturbance propagation so that unexpected plant trip should not occur.

In the operation mode III, in case the plant trips, the most important task of operators is to maintain the plant not to exceed the various safety limits and bring it back to the safe shutdown condition.

It is very important for a nuclear power plant to take proper and quick action under abnormal condition. In such a large scale complex system, many alarms are usually induced from a single causal alarm. Operators must identify the cause from among the many alarms, and take proper and quick action. But this is one of the most difficult jobs for operators, because it requires much operation experience and skill. It is very effective especially for a LMFBR plant to introduce a plant operational guidance system to cope this situation, because for LMFBR's sodium (Na) which has
large thermal inertia is used as coolant, so
we have enough time to identify the cause of alarms and
take operation before the abnormal condition propagates
from the sodium cooling loop to the other part of the
plant.

An operational guidance system for a LMFBR plant to
improve the plant operability by using a knowledge
engineering method have been developed.

The major objective of this system is to quickly
supply operators with information concerning the cause
of alarm (fast hit alarm) and with guidance indicating
proper steps to be taken when an alarm has occurred in
the LMFBR plant. The main function of the systems are
given below:

1. Monitor process
2. Inference diagnosis
3. Output and man-machine process
Advanced man-machine interface scheme for supervision [1]

Recent advance of electronics has brought many new products for operator-machine communication. Among many new products of interfacing tools, the following devices are considered as attractive candidates to augment the operator-machine communication capability.

1. Fullgraphic CRT
2. Large size display
3. Voice recognition device
4. Voice announcement device
5. Touch sensitive screen

In a nuclear power plant, an inherent safety feature would be verified if the decay heat after reactor trip were removed by natural circulation flow without the need for any electric power supply. To evaluate this natural circulation, the Power Reactor and Nuclear Fuel Development Corporation (PNC) decided to perform verification tests using the experimental fast breeder reactor JOYO.

These tests were intended to verify that decay heat after reactor trip would be removed by natural convection flow showing that the LMFBR plant has an inherent safety feature in continuous natural convection cooling of the core fuel subassemblies.
4. Conclusion [1]

For a LMFBR, new concept of the monitoring and control technologies are proposed. The main features of this system are as follows.

(1) Computerized and CRT based information presentation
(2) Introduction of Safety Parameter Display Function
(3) Introduction of Plant Operation Guidance System
(4) Advanced man-machine console with new MMIF devices
(5) Introduction of the optical fiber links
Fig. Advanced control and monitoring system
5. Reference


A Review of Work in UK on Instrumentation for the Supervision of Core Cooling in IFR's

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For Presentation at the IAEA/IWGFR Specialist Meeting on Instrumentation for the Supervision of Core Cooling in IFR's at Kalpakkam, INDIA
12th to 15th December 1989
INTRODUCTION

In the UK protection against overheating in the core is obtained from robust design backed up by thermocouple instrumentation for temperature measurement. Much work has been done to improve the performance and reliability of thermocouples and the associated instrumentation and data handling systems.

In recent designs of Fast Reactor concern to avoid thermal cycling has led to design changes in the Above Core Structure (ACS) which makes it difficult to position thermocouples to measure breeder element outlet temperatures. There has therefore been a programme of development of remote methods of measuring subassembly outlet temperatures aimed at monitoring breeder elements.

In addition to temperature measurement acoustic techniques for Boiling Noise Detection are also under development as a back up method of detecting severe overheating.

The work on these various systems is described briefly in this paper.

MEAN TEMPERATURE MEASUREMENT

Thermocouples

To improve the reliability of mean temperature measurement the coaxial thermocouple was developed which is described by Thompson et al.[1]. This thermocouple is 1mm diameter and has a time constant of about 50 ms. It is significantly more reliable than the conventional mineral insulated thermocouple and is incapable of sustaining an undisclosed operational fault. These thermocouples can be wound into a rope around a central 3mm thermocouple to give a robust assembly for insertion in the reactor. The sensing end of the rope is protected by a perforated metal tube.

Safe operation of fast breeder reactors (FBR) requires that the outlet temperature of each subassembly be monitored accurately. In the European Fast Reactor (EFR), as in its immediate predecessors, the thermocouples are supported from the Above Core Structure so that each core assembly is associated with a thermocouple located centrally and above the core outlet level.

For hydrodynamic reasons, modern FBRs are characterised by a low baffle design of ACS. Consequently, flow exiting a core subassembly is forced to turn rapidly through 90° and thence flow radially outwards in a flat core exit jet. This behaviour places an upper limit on the distance between the core outlet and thermocouple heights. Moreover, flow at large radius is contaminated with flow from the inner core regions and it is necessary to investigate how accurately the 'apparent temperature' (ie that measured by the thermocouple) follows the true subassembly exit temperature.

At NRL (Risley) investigation of core outlet thermocouple response (both generic and design specific) has been undertaken for a number of years. Sensitivity to changes in subassembly power output and flowrate have been investigated. Future work will address the efficacy of the EFR ACS thermocouple instrumentation by means of scaled models using air as a simulant fluid.
Data Processing

In the UK the signal processing system considered for fast Reactor mean outlet temperature is the Individual Sub-Assembly Temperature trip system. This is described in a specialist paper to this meeting[2], and also in [3].

The ISAT system is a failsafe microprocessor based trip system utilising low level multiplexing of the signals from fast response (50ms) coaxial thermocouples. The multiplexing reduces the amount of equipment required to monitor the large number of thermocouples on a large power reactor.

Remote Measurement

In some Fast Reactor designs it is difficult to place thermocouples in a position from which the outlet temperature of Breeder Subassemblies can be measured accurately. To overcome this we in the UK have been developing remote methods of measuring temperature using ultrasonics. This technique is described in a specialist paper to this meeting[4] and has been found to have the potential for development into an accurate fast response method of measuring temperature directly at the outlet of the subassembly so reducing errors due to the effect of cross flow. A further development to measure the temperature distribution across the core top using tomographic techniques is described in another specialist paper[5].

Temperature Noise

Work in the UK has concentrated on the development of modelling techniques to predict the temperature noise signals from developing blockages in the subassembly over a representative range of operating conditions. Alongside the attempt to obtain analytical solutions there has been an attempt to predict real-time fluctuations to enable the evaluation of statistical techniques for decision making. This note updates the work presented at the IWGFR Specialists meeting in 1984[6].

The CDFR coaxial thermocouple clusters[1] have been tested[7] over a range of flow conditions in the BNL No 4 sodium loop. Using a temperature noise method it has been shown that the thermocouples in the clusters have time constants covering a range of 33 to 70 ms, dependent on the flow, compared to 32 ms for an individual thermocouple. This performance together with the ISAT[2] processing system enables the monitoring of a useful fraction of the temperature noise spectrum.

Work is currently in progress to assess the use of ultrasonics as a means of measuring the outlet temperatures of fuel subassemblies. Since the method is very fast, several hundred measurements per second are possible, the technique is suitable to measure rapid temperature transients and high frequency temperature noise. Recent work at BNL, reported at this meeting[4], demonstrates that fluctuations in line averaged values of temperature exist in turbulent flow and can be detected with the ultrasonic technique.

The transport equation for fluctuating temperatures has been solved in 2-dimensions by a new approach using an extension to the eddy diffusivity hypothesis[8]. The complete solution has been obtained for a "top-hat" temperature profile, which represents the position at the pin bundle exit as the hot liquid from sub-channels behind a blockage mixes with cooler liquid from nearby unblocked channels. This analytical solution reproduces many of
the features observed experimentally. However, attempts to extend this analysis to a full 3-dimensional solution have proved unsuccessful due to the complexity of the resulting transport equation.

The multi-particle Monte Carlo code STATEN[9] has been developed and tested against data from pipe flow experiments over a wide range of Reynolds numbers and under the axially changing turbulence conditions found in a divergent jet[10]. The axial changes are modelled by a series of linked steps in which homogeneous conditions are assumed. Dissipation processes are represented and both mean and fluctuating temperature fields are generated. Further verification of STATEN has been performed against the KfK sodium and water experiments performed in the TEFLU rig, and against the similar jet block experiment at UKAEA Risley[11]. This has enabled the small scale grid turbulence downstream of the pin bundle to be modelled correctly.

The STATEN code has been used to predict the temperature fields generated in the outlet region of a CDFR subassembly, from the pin bundle exit to the thermocouple location[11]. The varying turbulence structures produced by the venturi shield and orientation bar have been simulated. Different temperature profiles at the pin bundle outlet have been considered. These profiles represent normal reactor conditions by a cross-subassembly tilt of up to 50°C with cooler edge channels, and local increases due to coolant blockages within the pin bundle.

Similar modelling of the Superphenix subassembly has been performed within the collaboration[12]. This has recently been extended to predict the signals from the internal blockages[13] and to evaluate the noise of line averaged values; to establish the viability of ultrasonic detection methods in a reactor.

STATEN simulated temperature signals at the sensor location have been processed to derive statistical parameters (for example skewness, kurtosis, spectral form) for use as inputs for pattern recognition processing. Data sets for progressively increasing blockages and normal conditions have been used[14,15], to test an Adaptive Learning Network (ALN) approach. It appears possible to design a suitable network of the flow area irrespective of location in the bundle or cross-subassembly temperature tilt. This would permit the use of a single detection algorithm for all core locations. The simulated test data sets could provide a useful basis for comparing different decision making systems.

The early work has been extended to generate an improved test data set which has then been used to compare another pattern recognition method, Cluster Analysis (CA) with the ALN technique[16].

ACOUSTIC BOILING NOISE DETECTION

In fast reactors the maximum temperature of the sodium in normal operating conditions is some 300°C below the boiling point and boiling can only occur when there is severe overheating. The acoustic noise from the boiling process is therefore an indication of insufficient cooling at some point in the region monitored.

In the UK acoustic techniques for the detection of boiling noise are being developed as a candidate system for the monitoring of breeder element outlet temperatures. This work has been described[17-20]. The Acoustic Boiling Noise Detection (ABND) system has an economic advantage over thermocouple protection
in that a much smaller number of detectors is required however, its most probable application is to monitor conditions in breeder elements where the installation of thermocouples is difficult.

In any protection system the achievable sensitivity has to be balanced against the tolerable rate for spurious alarms. This is discussed in detail in the work of the CRP on Signal Processing for Boiling Noise Detection[21]. In a reactor the requirement for spurious trips is severe, the probability must not be more than $10^{-1}$ per year. This means that there is a region of signal level between normal operation and the level at which a trip can be generated with confidence in which the acoustic signal could be an indication of an incipient fault. For this reason ABND is seen as a two level system generating a trip on large signals and functioning as a diagnostic warning device on lower signals. In this latter role ABND could well be part of a larger computerised protection system making use of signals from a range of detectors covering several parameters.

A number of techniques have been developed to aid the discrimination of the boiling signal from the background noise. The most promising methods have been a) Location Techniques and b) Pattern Recognition.

Location Techniques

Location methods are of value for two main reasons. first it is obviously of interest to the operator to know the position from which the boiling indication is coming. This enables the indication to be checked against other signals, eg thermocouples, in that area. Also this information is clearly of value after shutdown as aid to correcting the problem. Secondly if the acoustic detectors can be focused on a particular region then the background noise from elsewhere in the system is excluded. This improves the signal to noise ratio and is a valuable aid to detection.

Pulse Timing

When the signals are clear, for example a large boiling pulse, location can often be determined by a direct comparison of the signals from the detectors to determine the difference in the transmission time of the signal to each of a pair of transducers. In a two dimensional system this enables a semi-hyperbola to be drawn, with the detector which received the signal earlier as focus, on which the source must lie. A second pair of transducers allows a second semi-hyperbola to be drawn and the source will be located at an intersection of these. Additional pairs of transducers allow ambiguities to be clarified. An example of this location method from an experiment carried out in an experiment on a model of part of the PFR core is shown in Fig 1. Six transducers located above the outer corner subassemblies were used to locate the source of an electrically generated signal. This technique is appropriate when there is good signal to noise ratio. The method can be extended to three dimensions by plotting hyperboloids. A minimum of four transducers is required.

When the signal cannot be readily discriminated from the noise by inspection a correlation method is required to determine the relative delays. This is in effect the basis of beamforming techniques.
Delay and Sum Beamforming

In the simplest beamformer, the delay and sum beamformer, the correlation value is obtained by calculating the distance from the point of focus to each of the hydrophones. These distances are divided by the velocity of sound to give the transit times which in turn are divided by the sampling interval used in the digitising process to express the transit times as numbers of time points in the digitised records. The records can then be shifted by these amounts so that a signal emanating from the focus will appear at the same timepoint in all records. The records are then added and squared to give a measure of the acoustic power from that point. This technique has been applied to data obtained on the half scale model of the PFR core Fig 2. A plan view of the model showing the position of the hydrophones is shown in Fig 3. A transducer driven from an electrical oscillator was used as the source and the results in the form of Isometric plots are shown in Fig 4. It is seen that the source is always located correctly in the central subassembly and that the sharpness of the location improves with frequency. At high frequencies the wavelength becomes less than the space between transducers and spurious locations are indicated due to aliasing. The spurious indications vary with change in frequency and so can be distinguished from the true location.

An advance on the above technique can be obtained using adaptive methods as explained by Pirth[20].

Adaptive Beamforming

Adaptive array processing can significantly suppress the contribution from noise with high spatial correlation. Unpredictable physical features such as entrainments of gas or vapour within the reactor core may cause high signal attenuation at some sensors. We require the additional capability to suppress contributions from sensors with low SNR.

An adaptive array processing strategy which satisfies these requirements is the orthogonal beamforming or eigenvalue decomposition technique (see for example Owsley[22], Bienvenu and Kopp[23]). This technique is summarised below.

Define the following quantities:

a. Estimated cross spectral density matrix $B$. In practice there are a finite number of matrices $R$ at discrete frequency values $f$. Each matrix has dimensions $n \times n$ (where $n$ is the number of sensors) and is positive definite. The matrix is subject to statistical variability due to the finite length of sample used to estimate it.

b. Estimated cross-spectral density matrix $N$ for the noise (again positive definite).

c. Steering vector $\phi$. This vector contains the phase lag information corresponding to the current array focal point. It is a function of frequency.

The array output power at frequency $f$ for a non-adaptive beam former may be expressed as

$$ P(f, \phi) = \phi^H R_p $$

(1)

54
where the superscript $H$ denotes the conjugate transposed.

Generally adaptive methods replace expression (1) by

$$P(f) = \mathbf{v}^H \mathbf{R} \mathbf{v}$$

(2)

where $\mathbf{v}$ is a "weight vector". The relative amplitudes and phases of $\mathbf{v}$ are now chosen to optimise some measure of the array performance subject to some constraint on $\mathbf{v}$.

In the orthogonal beamforming approach we require to maximise the total array output power subject to the constraint that the array response to the "noise" is unity, ie

$$\mathbf{v}^H \mathbf{R} \mathbf{v} = 1$$

(3)

The maximum value of $P(f)$ may now be shown to be given by

$$P_{\text{max}}(f) = \lambda_1$$

(4)

where $\lambda_1$ is the maximum eigenvalue of the equation

$$\mathbf{R} \mathbf{v} = \lambda \mathbf{N} \mathbf{v}$$

(5)

The optimum weight vector is the eigenvector $\mathbf{v}_1$ corresponding to $\lambda_1$.

If there is no "signal" present and furthermore $\mathbf{R}$ and $\mathbf{N}$ are estimated exactly then $\mathbf{R} = \mathbf{N}$ and all of the eigenvalues of equation (5) are equal to 1. This does not happen in practice because of sampling errors and in this case the eigenvalues are clustered randomly in the neighbourhood of 1. If a single source "switches on", one eigenvalue emerges from this cluster and assumes some larger value. Thus $\lambda_1$ may be used in the "detection" role and some detection threshold set.

To "locate" a single source the correlation

$$\gamma(r_f) = |\mathbf{E}^H \mathbf{v}_1|^2 / \left( |\mathbf{E}^H \mathbf{N}^{-1} \mathbf{E} | (\mathbf{v}_1^H \mathbf{N} \mathbf{v}_1) \right)$$

(6)

may be tabulated or plotted over all candidate source positions $r_f$ (NB $r$ depends on $E_f$). The maximum is located in order to give an estimated source position.

An experimental result of the application of these techniques to data from the FFR model in the case where the signal to noise ratio on each of the transducers used was as stated in the table ie about -9db on average, is shown in Figs 5 and 6. The benefit of the orthogonal beamforming technique in distinguishing the new source from the normal background is clearly seen.

**PATTERN RECOGNITION**

This is a powerful method of distinguishing between a signal and the background noise. Essentially it consists of finding a combination of the features of the signal which is distinct from the background. The UK work in this area is demonstrated in the CRP on Acoustic Signal Processing [\ref{ref}]. An important property of this technique is that it enables the probabilities of failure to detect boiling and spurious trip to be estimated within the
statistical accuracy possible with the data records available. In practice to
give an assurance of a very low spurious trip rate a long examination of
background data is necessary.

The current work in NRL is aimed at combining the above techniques to achieve
a sensitive acoustic boiling noise detection system with a low spurious trip
rate. This in turn could eventually be part of a multiparameter diagnostic
system.

CONCLUSIONS

Thermocouples are a reliable method of measuring outlet temperatures of core
subassemblies and instrumentation has been developed to a high degree of
reliability. Ultrasonic techniques offer a promising method of measuring
temperature remotely which could be valuable where it is difficult to place
thermocouples. The ultrasonic method is likely to be less affected by cross
flow than thermocouples.

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Dr D Firth        NRL UKAEA, Risley
Mr G Spence       NRL UKAEA, Risley
REFERENCES


Fig. 1 LOCATION BY DRAWING HYPERBOLAE

1/2 Scale Model of P.F.R. Core

Transducer Positions

Position of Source
HYDROPHONE POSITIONS
A, B SOURCE POSITIONS

FIGURE 3: HYDROPHONE ARRAY AND SOURCE POSITIONS
ISOMETRIC PLOTS OF THE OUTPUT FROM THE DELAY AND SUM BEAMFORMER FOR SIGNALS OF VARIOUS FREQUENCIES

FIGURE 4
FIGURE 5: PLOTS SHOWING THE OUTPUT FROM THE
TIME DOMAIN BEAMFORMER FOR A BROADBAND SIGNAL
IN THE PRESENCE OF BROADBAND BACKGROUND NOISE

FIGURE 6: PLOTS SHOWING THE OUTPUT FROM THE
ORTHOGONAL BEAMFORMER FOR A BROADBAND SIGNAL
IN THE PRESENCE OF BROADBAND BACKGROUND NOISE
TABLE 1: AVERAGE SIGNAL TO BACKGROUND NOISE AT EACH HYDROPHONE FOR A BROADBAND NOISE SOURCE WITH BROADBAND BACKGROUND NOISE.

<table>
<thead>
<tr>
<th>HYDROPHONE NUMBER</th>
<th>AVERAGE SIGNAL TO BACKGROUND NOISE (dB)</th>
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</thead>
<tbody>
<tr>
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<td>-9.7</td>
</tr>
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<td>2</td>
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INTRODUCTION

The main physical parameter determining a deviation from normal operation of a reactor is the coolant temperature. It is this temperature that mainly determines the performance of in-pile components and structures. In the end, any perturbations of other parameters result primarily in a temperature variation. The coolant temperature is functionally related with flow rate through subassemblies and their power. Therefore, all the existing methods of core state monitoring are related to recording of abnormal values of neutron flux, coolant flow rate and its temperature. Specific methods of recording these deviations may differ. For example, besides direct temperature measurements, the acoustic methods of coolant boiling detection are used which practically indicate a considerable increase of the coolant temperature; the same can be also said about sudden increase of gas activity or delayed neutron precursors appearance.

The core supervision system can be divided into:

a) a subsystem of the core monitoring as a whole which is recording neutron flux, flow rate through the reactor, the coolant temperature at the reactor inlet and outlet whose values are used for the reactor control and safety system;

b) a subsystem for monitoring of a single subassembly condition the principal purpose of which is to detect any failure in a subassembly and to produce warning or safety signals.

The single-subassembly state monitoring methods are subdivided into the local methods when each subassembly has its measuring channel, and integral ones when with the use of some detectors much less in number than the number of subassemblies, an abnormal state...
of some subassembly is detected. The ratio between local and integral methods in various countries is formed in different ways according to design features of subassemblies and in-pile structures. Besides, at the choice of some or other methods of monitoring also affect such features as reliability, performance, response time, etc. The economic problems are also of importance.

In the USSR, from the very beginning of fast reactors introduction the subassembly design with the lateral coolant outlet through a number of windows has been developed. Though the head of such a subassembly has a central hole, nevertheless, due to mixing of the coolant passing through these windows with the coolant from adjacent subassemblies, the medium above a single head subassembly is already not its individual characteristic. Therefore, the readings of any detector located above the subassembly head will be affected by adjacent subassemblies. Besides, as a result of using subassemblies with a relatively small flat-to-flat size (98 mm, 94 mm) when in the core of, e.g., the BN-600 reactor there are more than 500 subassemblies it is rather difficult for structural reasons to afford the location of detectors above each subassembly and communication channel penetrations through the upper plug that by its function is also neutron shield. As a result, in the operating BN-350 and BN-600 reactors and in the BN-900 reactor design the local, in the full sense of the term, methods of single subassembly monitoring are not used, and the integral methods are preferred. In the BN-1600 reactor design under development now a version of a larger-size subassembly with an open head and with monitoring the state of each of them is considered. It should be mentioned, however, that the advantage of the local methods of monitoring as compared to the integral ones is not obvious, and considerable efforts are still needed for choosing an optimum combination of different monitoring methods.

Methods for monitoring of subassembly in operating and designed fast reactors are considered below.
1. TEMPERATURE MONITORING.

BN-350
To monitor the coolant temperature inside the reactor vessel two groups of thermocouples are used. One of them, of 4 in number, is located in the central rotating column above the subassembly heads, another eight - in the upper reactor plenum. The thermocouples being enclosed in protective jackets, their time constants are rather considerable and make up 10 sec. Of course, if to use directly the readings from thermocouples with such time constants in the temperature safety channel of the reactor then it will prove to be too delayed in action. That is why a special temperature safety device is used which processes the thermocouple readings taking into account their transfer functions and recovers real coolant temperature values, the safety signal being generated by the 2/3 logic. This device has had long - term operating experience and has shown a good performance. For the temperature safety channel to perform efficiently it is necessary to have such time constants of thermocouples or of a device processing their readings that do not exceed the fuel-pin time constant that for the BN-350 is 2 sec.

BN-600
In this reactor 20 thermocouples are placed above subassembly heads and 8 thermocouples are in the upper plenum. The thermocouples time constants are 20-25 sec. Their readings are also processed by the temperature safety device providing an acceptable response time of the safety channel.

During the years of the BN-600 operation much material on natural noises of various parameters was obtained. Information on natural noises is required both for an evaluation of disturbances in the operation of safety and automatic control system and for studying physical characteristics of the reactor, for the development of the control and diagnostics methods.

In addition to standard thermocouples, two thermoprobes were manufactured especially for the BN-600 which were inserted into subassembly heads and incorporated four chromel-alumel thermocouples and one sodium-steel thermocouple. The time constants

67
of the chromel-alumel thermocouples were ~1 sec. Recording and analysis of reactor parameters noises are carried out at 50% rated power in the period before the reactor reaches the medium steady state and at rated power after the reactor goes to the medium steady state. Experiments have shown that the highest temperature noise is observed in the upper reactor plenum. The temperature fluctuation band reaches 8.5°C and RMS - 1.4°C. This hinders high-quality operation of the temperature regulator at using in it a signal from thermocouples located in the upper plenum. The fluctuation band for thermocouples above the subassembly heads and for thermoprobes is within a range of 2-5°C and RMS of 0.3-0.7°C. With increasing of the reactor power and passing to the medium steady-state conditions there were observed both an increase of the fluctuation band because of increased coolant heating and a decrease resulting from energy tuld flattening in the core. A principal part in the temperature noise at the subassembly outlet is played by turbulent mixing of coolant at a non-uniform temperature distribution over the subassembly cross-section and a less part-by a deviation of integral parameters (of power and flow rate). All these factors should be taken into account at the development of systems for temperature monitoring at subassembly outlets. Particular care should be taken in the analysis of sodium-steel thermocouple readings. For them it is important to take particular care in providing thermostabilization of the "cold" junction. Otherwise, taking into account its inertialessness the fluctuation of its readings may be of a false character. In the experiments the sodium-steel thermocouple signals dispersion reached 2°C. The correlation analysis of noises in this case has revealed that dispersion of a high-frequency spectrum portion which is determined by turbulent fluctuations of coolant velocity and by non-uniformity of temperature distribution over subassembly cross-section makes up ~70% of the total one. Thus the high-frequency temperature fluctuations dispersion was 1.4°C.

This is a background noise as a result of the additional temperature disturbance at the subassembly outlet due to the discrepancy in heat removal inside the fuel subassembly. Single
subassembly monitoring by their outlet coolant temperature can be efficient only based upon the thermocouple noise analysis because an increase of the deterministic temperature value by 15°C may be a consequence of normal technological processes or of a redistribution of the energy field due to the control rod movement. An increase of the temperature high-frequency natural noise dispersion in this case will not exceed 20% of the initial one whereas the dispersion due to blockage inside the fuel subassembly causing the same increase of the mean temperature increases several times. Calculations have shown that for the BN-600 reactor conditions the minimum size of blockage in the subassembly active part centre resulting in increased temperature dispersion at the fuel bundle outlet up to 2°C makes up 10% of the flow area. As to the requirements on processing of a signal from a thermocouple used for subassembly monitoring, then based upon the time required for failure detection and reactor shut-down which to our opinion should not exceed 15-20 sec, the relative error in the calculation of RMS of a random process during this time in a frequency range of 1-20 Hz does not exceed 7% that is quite satisfactorily. Of course, the best variant for measuring temperature fluctuations is the use of practically inertialess sodium-steel thermocouples. At using conventional thermocouples in technological jackets the processing should be carried out with the use of the thermocouple transfer function which should be obtained in advance. It should be said, however, that from the technical and procedure points of view the single subassembly state monitoring with the use of thermocouples followed by a signal output into the safety system is far from introduction.

BN-800

The temperature monitoring at the core is carried out with the use of 24 thermocouples located in the upper reactor plenum at the coolant inlet to the intermediate heat exchanger and of 111 thermocouples combined in groups of three in separate jackets and located in the central rotating column at a distance of 100 mm above subassembly heads. They are arranged so that they cover the
whole core. The time constants of thermocouples are 5 sec. As the
design of the BN-800 fuel subassembly is similar to that of the
BN-350 and BN-600 reactors then here there are also the same
problems related to placing thermocouples above each fuel
subassembly and interpretation of their readings. The thermocouples
located above fuel subassemblies can characterize only the state of
those subassemblies above which they are located and also
temperature deviations as a whole above different throttling zones
as a result of the reactor integral parameters variation due to the
refuelling or control rod movement. The safety system over the
coolant temperature (and over other parameters as well) is realized
by 10 independent safety rods which are operated by two independent
subsystems each of which includes a device for logic processing of
discrete signals. Independence of the sets is provided by mounting
them in various rooms, by laying communication lines through
different cable routes and by connecting to independent electric
power supply sources. Emergency signals arrive at each subsystem
through three channels and are processed by 2/3 logic. The device
for logic processing of signals also consists, in its turn, of
three channels whose signals are processed by 2/3 logic. Such
arrangement of the emergency safety scheme will permit to increase
reliability and to minimize false trips.

BN-1600

At present it is too early to speak about the details of
subassembly monitoring with thermocouples but as the main variant
is considered that with an open subassembly head the flat-to-flat
size of which is 153 mm and with several thermocouples placed above
each of them. The possibilities of using the readings of all these
thermocouples for the detection of emergency situations in single
fuel subassemblies with putting out signals into the safety system
are still to be carefully studied. Here it is essential to assure
reliability both of their readings and of the system of these
readings processing, the more so that only the coolant temperature
noise analysis may be most informative.

In conclusion of this section it would be desirable to mention
one of possible means of the thermocouple trouble detection. At a loss of tightness of a thermocouple sheath a leak-in of coolant inside it and shorting of electrodes take place that create the so-called "fictitious junction" in the area of the leaking-fluid surface layer. This "fictitious junction" being agitated as a result of leaking fluid leads to a displacement of the temperature measurement point. One of the methods of detection of this phenomenon is the use of a three-electrode thermocouple manufactured after the USSR inventor's certificate 857735. The main point of this method is in changing the calibration characteristic of the thermocouple and, accordingly, of temperature indications, due to shorting by leaking fluid of the end of the third conductor freely suspended in the region of measurement with the two working thermoelectrodes. This leads to shorting of some part of one of the working thermoelectrodes and to corresponding sharp changing of the thermal emf. From this change in the thermal emf the presence of the fault is determined.

2. FLOW-RATE MONITORING.

In the fast reactors in operation and being designed in the USSR at their power operation the integral monitoring of coolant flow-rate through the reactor and/or through the loops is carried out. Monitoring of coolant flow-rate through each fuel subassembly during reactor power operation presents certain difficulties though some investigations in this direction are underway. Such monitoring is carried out only on a shut-down reactor with the use of an electromagnetic flow-rate measuring device which is aimed in turn at individual subassemblies making a tight fit. These measurements are aimed at a verification of conformity to designs of the real hydraulic characteristics of reactor subassemblies, operational monitoring of coolant flow-rate through fuel subassemblies, the determination of the effect of fuel-pin and subassembly-duct deformation upon the flow-rate magnitude. The flow-rate measuring device includes a number of electromagnetic sensors which not only
provide duplication of readings but with their use a principle of correlation measurements is realized. The correlation method allows to control the electromagnetic flow-meter readings to an accuracy of 1%. Calibration of both the correlation method and of the flowmeters themselves is carried out on special metrological rigs. Long-time operation of flowmeters have shown their high efficiency.

Let us now turn to the consideration of the integral reactor monitoring methods.

**BN-350**

This loop-type reactor has 6 loops. For measuring flow-rate in each loop, two electromagnetic flowmeters are mounted on 500-600 mm diameter pipes at the primary pumps inlet and outlet. Due to a considerable fluctuation in indications of the flowmeters mounted at the pumps outlet their signals cannot be directly used in the safety system. That is why they are subjected to processing in a special device. Coolant flow-rate monitoring is also performed by measuring pump revolutions. Before their mounting on the loops the flowmeters were subjected to calibration at a special circulation rig.

**BN-600**

In this reactor for sodium flow-rate monitoring an indirect measurement of flow-rate is provided through a by-pass on which 6 electromagnetic flowmeters are mounted. The flowmeters were calibrated by rig tests. Besides direct measurements a correlation principle of flow-rate measurements was realized. With its use one can also monitor at regular intervals the flowmeter indications. During start-up and adjustment work, prior to bringing the reactor to power operation a set of tests on determining the thermohydraulic characteristics of the reactor primary circuit was carried out that allowed to determine the correspondence of indications of flowmeters at the by-pass loop, of sodium flow-rate through each subassembly and the primary circuit of the reactor to the pump rotation speed. The obtained dependence of the by-pass loop flowmeter indications on the reactor flow-rate is of almost
linear character.

The by-pass flowmeter indications fluctuate within a 5% band of nominal flow-rate. An analysis has shown that the flowmeter signal noise can be approximated by a sum of two components - of a low-frequency one and a broadband non-correlated one the weight fraction of which is predominant. The low-frequency component represents the actual change of flow-rate and the broadband component is connected with the turbulent perturbations of the velocity field in the flowmeter cross-section.

**BN-800**

In this reactor, in addition to the measurement of flow-rate through the by-pass loop as in the BN-600 reactor the measurement of flow rate in loops is provided. For this purpose at the outlet of each pump a flowmeter is placed that is similar to that used for measuring flow-rate in single fuel subassemblies at a shutdown reactor. Besides the direct measurement of flow-rate with the use of magnetic sensors, in these devices it is expected to use time-of-flight and frequency-correlation methods of monitoring.

For flowmeter calibration there are used: a circulation metrological flowmeter rig with a measurement limit of 0-12 m$^3$/hr using a volume-space method with an accuracy of calibration of 0.1% and a comparison method with an accuracy of 0.3%; a rig including a loop providing also with the volume-space method the calibration within 0-20 m$^3$/hr to an accuracy of 0.3%; and a circuit with flow-rate up to 100 m$^3$/hr containing a measuring comb with the help of which using the comparison method the calibration to an accuracy of not less than 1% is carried out. A rig with a flow-rate of up to 100 m$^3$/hr is being mounted and a flowmeter unit with a flow-rate up to 1600 m$^3$/hr is currently designed realizing a principally new pulsed method of calibration. This rig is extremely compact for its parameters and has small specific metal content. It will allow to calibrate flowmeters to an accuracy of 0.5%.
3. MONITORING OF ACOUSTIC AND NEUTRON NOISE

One of the systems which can provide monitoring of core subassembly is the sodium boiling detection system. Such a system based upon the neutron and acoustic noise measurement is now under development for the BN-800 reactor.

A subsystem for sodium boiling monitoring by neutron noise is being created based upon the use of ex-vessel ionization chambers located inside common cells. In this case a noise component of the signal is isolated and fed at the input of an instrumentation unit being specially developed now.

The physical basis of the operation of the neutron-noise sodium boiling monitoring subsystem is the variation of the core physical characteristics at the appearance of sodium vapour in a single fuel subassembly. In this case the algorithms involve the dynamics of signal variation with the vapour quantity variation in the process of boiling. Really, the presence of the sodium void effect of reactivity causes corresponding response in neutron flux. Due to a small volume of boiling, however (a case of sodium boiling in one subassembly) the total reactivity can vary insignificantly and lie below the noise level of measured signals. At the same time, as was shown by experiments at the BOR-60 and BN-350 reactors, at frequencies of 0.3 Hz to 10 Hz the sodium boiling signal level becomes higher than the background noise level and, according to our estimates, can be well detected by measuring devices.

It should be noted that in the BN-800 reactor core there are regions with a zero void coefficient of reactivity. This somehow reduces the efficiency of the neutron-noise boiling detection method. Besides, due to a small value of the void coefficient of reactivity at the periphery of the core some doubts arise as regards a possibility of boiling detection in this region of the core. It is possible that some part of this zone which is on the side of neutron chamber position can be monitored by the local component of neutron noise caused by passing of neutron flux through vapour space (the effect of exposure to neutrons).

To our opinion, the subsystem of boiling detection in the core by neutron noise will permit to detect an event of boiling and, may
be, a trend of development of an emergency process causing boiling.

The acoustic-noise sodium boiling monitoring subsystem incorporates specially designed waveguide acoustic detectors and a pertinent set of equipment. For the BN-800 reactor it is envisaged to mount 6 waveguide detectors uniformly arranged on the circumference above the core. For this purpose the large rotating plug is provided with special penetrations for these detectors.

The acoustic-noise coolant boiling monitoring subsystem operates on the basis of recording acoustic signals generated by vapour bubbles at their collapse in underheated sodium. Experiments at BOR-60 and BN-350 have shown the sodium boiling signal to be manifested most effectively in a frequency band of 20 kHz to 200 kHz. To determine the point of the vapour bubble collapse a well known triangulation method is used. This method is based upon the measurement and analysis of the time of signal arrival at spaced-apart detectors. The use of this method allows not only to determine the point of boiling but also to carry out practically spatial filtration that, in its turn, increases the reliability of boiling event detection and selectivity of the measuring system.

Soviet specialists believe that both above systems of coolant boiling monitoring in the nuclear reactor core will allow to record sodium boiling in the core at an early stage of abnormal situation development, and a correlation of signals obtained from these system will allow to increase the reliability of monitoring.

4. OTHER METHODS OF MONITORING.

Operating experience of both experimental and power reactors has shown high efficiency of the delayed-neutron monitoring method. The efficiency of any method is determined, first of all, by its sensitivity and speed of response. So far in the USSR only integral methods of delayed-neutron monitoring are used. However, they also have proved to be highly efficient.

Sufficiently promising is a method of monitoring based upon balance-of-reactivity calculations. Experimental sample tests at
various reactors have shown that this method allows to record an anomalous reactivity variation as a result of coolant boiling in a single subassembly or of molten fuel displacement in it.

CONCLUSION.

1. Up to now, in the USSR thermocouples have been used only for monitoring the conditions of core subassembly cooling as a whole including the distribution of a coolant temperature over the core radius at the subassembly outlet.

An experience of using them in the system of monitoring core subassembly conditions in the safety system has revealed that the time constant of the thermocouples themselves or of a device for their indications processing should be ~1 sec. An analysis of indications noise of thermocouples located above subassembly head pieces and in thermoprobes allows to draw a conclusion about a possibility of isolating a signal characterizing a deviation in subassembly operation from normal conditions. Work on development of the BN-1600 design version with thermocouples mounted above each fuel subassembly is conducted now.

2. The use of flowmeters at the reactor by-pass and in the loops is an efficient means of monitoring for core cooling. Favourable experience of their operation at the plants in operation has been obtained. A method of monitoring coolant flow-rate through individual subassemblies at a shut-down reactor has fully justified itself.

3. An integral method of delayed-neutron monitoring has proved to be a sufficiently effective method of individual subassembly monitoring.

4. Experimental verification of neutron and acoustic-noise monitoring methods has proven their prospects and they are introduced at the BN-800 reactor.

5. Reliability and efficiency of single subassembly cooling monitoring can be provided only at a combined use of various methods. An optimum set of monitoring systems is still to be determined. When choosing these means of monitoring one should apparently prefer the integral methods.
Experimental Breeder Reactor II (EBR-II),

Instrumentation for Core Surveillance

by

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Introduction

The Experimental Breeder Reactor-II (EBR-II) is a small but complete liquid-metal-cooled reactor (LMR) power plant. It has a nameplate peak thermal power of 62.5 MWt and a corresponding electrical output of 20 MWe; it has been operated over a range of power including a peak of 70 MWt, as experimental program needs have dictated, for 25 years. This facility is located at Argonne-West site of the Idaho National Engineering Laboratory, state of Idaho, USA.

Description of EBR-II plant

The EBR-II plant consists of a primary system and a secondary system both using molten sodium as the coolant and a rather conventional steam system, as shown in figure 1. The primary system is located in a large double-walled tank, as shown in figure 2. The tank contains about 340 m$^3$ of sodium at 371° C under normal operating conditions. The two primary pumps take their suction directly from the tank and deliver a combined flow to the reactor of 485 kg/s at a
power of 62.5 MWt. The corresponding mixed-mean temperature of the sodium coolant leaving the reactor is 473°C.

The reactor consists of an array of hexagonal subassemblies 5.817 cm across flats on 5.893 cm centers. The core region extends out through row 7, the reflector region through row 10, and the blanket region out another 5-1/2 rows. The core region may be smaller than the full seven rows. There are two safety rods (fueled-bearing) in row 3. There are 12 control rod positions in row 5, although only nine contain control rods (fuel-bearing) at present. The remaining three control rod positions are available for special in-core instrumented subassemblies, such as those used for the thermal-hydraulic (T-H) testing. The current fuel for EBR-II (Mark II) is a uranium alloy initially containing 5% simulated fission products, enriched to 66% in $^{235}\text{U}$. The fuel is a pin 0.3302 cm in diameter in type 316 annealed stainless steel cladding, 0.4420 cm outer diameter × 0.0305 cm wall. Sodium is used as a thermal bond between the pin and its cladding. Each such fuel element is wrapped with 0.1245 cm diameter spacewire. There are 91 elements in a regular fuel subassembly and 61 in a control or safety subassembly.

The primary flow to the reactor splits into two streams, one entering the high-pressure plenum that feeds the first seven rows and the other entering the low-pressure plenum that feeds the remaining rows. About 84% of the flow goes into the high pressure plenum and the remaining 16% goes into the low pressure plenum. The high and low pressure sodium streams mix in the reactor outlet plenum, go through the outlet pipe to the primary auxiliary pump (a DC electromagnetic device), and then to the intermediate heat exchanger (IHX) in which it transfers its heat to the secondary sodium. The primary sodium leaving the IHX dumps directly back into the primary tank. Power to the auxiliary pump is provided by a rectifier backed up by a battery floating on the line. In case of a power failure, an emergency 480 volt diesel generator provides
power to the rectifier. If the emergency power system fails, the battery takes over the load. Recent testing and analysis indicated that the auxiliary pump is not needed as an engineered safety feature for operation of EBR-II.

The secondary system sodium is driven by a single electromagnetic pump at a flowrate of 315 kg/s. The heat in the secondary system is transferred to the steam system in seven evaporators and two superheaters, all of the Argonne double-wall heat exchanger tube design.

The steam is used in a conventional turbine-generator to produce electricity. At full-power (62.5 MWt) the superheaters deliver 32 kg/s of steam at 438°C and 8.70 MPa to the turbine.

A more detailed description of the EBR-II plant is given in Reference [1].

Figure 3 shows the instrumentation associated with the primary coolant system when the plant was built. Of the original 10 flow sensors, only three are still in operation today. All three of these sensors are part of the reactor shutdown system (RSS). An instrument probe has been installed through the reactor cover into the outlet plenum region that provides an additional flow sensor in the form of a delta pressure measurement. This sensor signal is also part of the reactor RSS flow trip circuit. The scram contacts are arranged in a two of four trip logic. Engineering effort is presently ongoing to qualify digital speed sensors mounted on the primary pump motor shafts as flow trips. A toothed nut on the shaft produces pulses in a magnetic sensor. This pulse signal is proportional to pump speed which in turn is proportional to flowrate.

Figure 3 also shows the location of two ultrasonic flowmeters located in the low pressure plenum inlet piping at the throttle valves. These flowmeters are recent additions to the flow instrumentation and are in final checkout before being placed in operation. Figure 4 is a diagram of a ultrasonic flowmeter assembly. The system consists of a sensor assembly and an electronic control subsystem. The sensor assembly is suspended from the valve plug of the
throttle valve so that the full sodium flow in the four inch pipe flows axially in line with the sensors. The sensors are both transmitters and receivers. The ultrasonic travel time in sodium at 371°C is about 2377 m/s. The travel time for 35.6 cm is 150 ns with a difference of 1.0 ns difference between the up time and the down time. To resolve the travel time to approximately ±2 ns requires at least 100 pulse times be counted and averaged. Fast electronic circuits and a control computer are used for this purpose.

Figure 5 shows the placement of the subassembly outlet thermocouples that monitor the temperature of selected subassembly coolant streams. These temperature signals provide a diverse and independent backup to the loss of flow trips, and four of the signals are part of the RSS and trip on high temperature from subassembly flows.

Thermal-hydraulics testing at EBR-II

The thermal-hydraulic testing program at EBR-II, initially conducted to support the continued safe and reliable operation of EBR-II, has evolved into an experimental and supporting analytical program contributing to the design and performance assessment of advanced liquid-metal-reactors, with special emphasis on inherent safety. These efforts, which essentially started in 1974, have been primarily directed towards understanding the detailed response of EBR-II to a wide variety of upset conditions and utilizing this knowledge to validate general purpose thermal-hydraulic-neutronic computer codes for application to new plant designs. Initial emphasis was placed upon reactor and primary heat transport system phenomena, and more recently, the focus of the work has been on whole-plant dynamic behavior. The success of this program has been immeasurably aided by the availability of fully-instrumented and calibrated in-core fueled and non-fueled assemblies, XX07, XX08, XX09 and XX10 [2,3,4]. These assemblies give direct, real time measurements of in-core
sodium temperatures. These measurements were a prime basis for
"bootstrapping" from test to test and for use in an overtemperature scram
circuit. These assemblies, with their extensive temperature and flowrate
measure capabilities, have permitted the generation and documentation of
comprehensive data sets that have been used to validate codes modeling single
and multiple assemblies, and whole core behavior.

The in-core instrumented assemblies XX07 and XX08 have been previously
described in [3,4] while XX09 and XX10 (latest probes) were discussed in [2].
However, due to the importance of these instrumented probes to the conduct
and interpretation of the testing, XX09 and XX10 will be discussed in detail.
The fueled assembly, XX09, contains 61 elements, 59 of which are Mark-II
metal fuel and 2 serve as hollow conduits for below-core instrumentation leads.
There are 28 thermocouples measuring the three-dimensional temperature field
throughout the assembly, ranging from below core to assembly outlet; these
include thermocouples within the fueled region and in the inter-assembly
bypass flow region. Two permanent-magnet flowmeters are located in tandem
within the assembly below the core and have been calibrated over a flowrate
range covering rated conditions down to both upward and downward natural
convective flow (i.e., from -0.3 to +3.2 l/s). The normal operating conditions are
468 kW (25.6 kW/m, peak), 3.14 l/s, and 136° C coolant temperature rise. It is
orificed for a flow of 9340 kg/h and the mixed mean outlet temperature is 519° C
(based on an inlet temperature of 371° C).

In XX10, there are 19 elements, 18 of which are solid stainless steel (type
316) and 1 serving as a hollow conduit for below-core instrument leads. As with
XX09, there are 2 below-core permanent-magnet flowmeters calibrated over the
expected flowrate range from -0.04 to +0.44 l/s. A total of 26 thermocouples are
included providing full coverage from below-core to assembly exit regions. The
particular choice of materials and dimensions used for XX10 was based upon
standard thermal-hydraulic scaling laws so that the relative dynamic performances of XX09 and XX10 would closely approximate that of fueled and blanket assemblies in a large reactor [5].

Subassembly Design Considerations

The presence of an instrumented driver fuel subassembly in the core of EBR-II played an important part in the loss of flow without scram (LOFWS) tests and loss of heat sink without Scram (LOHSWS) tests, as it provided the ability to measure coolant flows and temperature at different axial elevations during each test. Thus, the accuracy, reliability, and range of operation of the instrumentation had to meet required specifications.

However, there was another consideration, a conceptual functional requirement in which XX09 had to represent a prototypical subassembly. That is, XX09's T-H characteristics were designed to be similar to those of other liquid metal cooled reactor (LMR) core designs so that the data could be reasonably extrapolated to these reactor designs. Previous experience in EBR-II [6] indicated the importance of determining the whole-core behavior during natural convective transients, and the need to holistically describe the entire core, which includes both driver and blanket subassemblies.

An analysis was performed on a conceptual design subassembly, applicable to either a driver or blanket, and nine nondimensional parameters were established. Of these, six were used in a steady-state comparison between two large LMR designs and EBR-II. As a result of this comparison, two instrumented subassemblies were designed, one a driver type (XX09) and the other a blanket type (XX10), with significantly different T-H characteristics. These were built for use during the Shutdown Heat Removal testing (SHRT) program [7]. Details of the conceptual design basis, the large plant comparison, and description of the blanket subassembly are given in [8].
The instrumented subassemblies XX09 and XX10 were designed to fit into control rod positions located in the fifth row of the core. The full assembly consists of three components: the subassembly, an extension tube, and a terminal box. The XX09 subassembly is representative of a standard EBR-II driver. The extension tube connects the subassembly to outside the primary tank. It provides a protected conduit for the instrument leads and permits movement of the subassembly during fuel handling. The terminal box provides a junction area from which the instrument leads are connected to the DAS, which records, processes, and stores the test data. A schematic diagram of the XX09 assembly configuration is shown in figure 6.

The lifetime of XX09 is dependent upon the time-to-breath for the Mark-II elements, and that generally has occurred at a burnup of about 9 at\% (2 years in the reactor).

There are two flowmeters located in the lower shield below the active core region. Both flowmeters were calibrated; the flow inaccuracy was less than 2% in the operating range of 390 to 9800 kg/h. The flowmeters were also calibrated for reverse flows up to 980 kg/h.

The spacer-wire thermocouples are contained in a metal sheath of 316 stainless steel and are made of Type K chromel/alumel with magnesium oxide insulation. The thermocouples are butt-welded to the wire wraps, and the length adjusted so that the thermocouple junction is at the specified core elevation when the wire wrap is spirally wound onto the elements. They have been fabricated to the U.S. Reactor Development and Technology (RDT) standards. The standard deviation was 0.7° C for an out-of-core calibration in the operating range between 427° and 593° C.

There are a total of 28 thermocouples. There are two located in the flowmeters near the inlet, five at the core midplane (MTC), thirteen near the top-of-core (TTC), four above the core (14TC), two at the subassembly outlet
COTC), and two in the thimble region of the subassembly (ATC). Fig 7 gives the specific instrument loading.

A T-H analysis was performed for XX09 on steady-state data obtained during EBR-II run 129C (June 1984). The T-H code THI-3D [9] was used to predict the coolant temperatures. THI-3D is a steady-state, single-phase, multi-channel code that uses the laws of mass, energy, and momentum conservation to calculate the temperature field in both the axial and radial direction. The turbulent interchange, radial thermal conduction, and wire-wrap forced flow between subchannels are considered explicitly. The model consisted of a total of 149 radial nodes, with 120 nodes for the XX09 subassembly and 29 for the surrounding annular thimble region between XX09 and the outer control-rod hex can. There are 49 axial nodes, each 12.7 mm long, modeling the entire Mark-II element length. The THI-3D code was used previously in thermal-hydraulic modeling and worked very well, especially when the annular thimble region around XX09 was included. Also, modeling of the adjacent subassemblies was not necessary, as they have a similar power-to-flow ratio, and the effects of intersubassembly heat transfer are negligible. The agreement is very good, with most of the data being within 5° C of the predicted temperature. Temperatures at the midplane were found to be higher than expected. However, this discrepancy can be explained by a number of factors, of which, a wire-wrap hot-spot effect appears most plausible. Details on the XX09 subassembly and thermal-hydraulic analysis are given in [10].

The most exciting T-H tests were the LOFWS test 45 that was conducted in April, 1986 [11]. The loss-of-flow-without-scram tests involve bypassing the normal loss of flow scram function and tripping the main coolant pumps. All of the tests demonstrated passive power reduction caused by reactor feedback mechanisms. Figure 8 presents pretest predictions and temperatures measured with a representative thermocouple in XX09 near the top of the core for a LOFWS from
100% power. A companion test, loss-of heat-sink-without-scram (LOHSWS) test was conducted the same day. LOHSWS involved a loss of normal means of transferring heat from the sodium pool to the balance of plant where electricity is generated. Figure 9 shows the tests results. The key observation from these test is that EBR-II core can ride through two severe transients without damage.

Conclusions

EBR-II has operated for 25 years in support of several major programs. During this time period, several of the original, non-replaceable, flow sensors, RDT sensors and thermocouples have failed in the primary system. This has led to the development of new sensors and the use of calculated values using computer models of the plant. It is important for the next generation of LMR reactors to minimize or eliminate the use of non-replaceable sensors.

EBR-II is perhaps the best modeled reactor in the world, thanks to a dedicated T-H analysis program. The success of this program relied on excellent measurements of temperature and flow in subassemblies in the core. The instrumented subassemblies of the XX series provided that measurement capability. From this test series, EBR-II calculations showed that the core could withstand a loss-of-flow without scram accident and a loss-of-heat sink without scram accident from full reactor power without core damage. From this, reactor designers can now design with confidence, inherently safe reactors.

EBR-II is an integral part of Argonne's IFR Program (Integral Fast Reactor) and will continue the T-H program. Additional series XX in-core instrumented subassemblies will be required.
References


5. P. R. Betten et al., Conceptual Design Basis and Temperature Predictions in a Simulated Instrumented LMFBR Blanket Subassembly, Third Intl. Meeting on Reactor Thermal Hydraulics, Newport, RI, October 15-18, 1985 (submitted for publication)


Fig. 1. Schematic of EBR-II plant.
Fig. 2. EBR-II primary system.
SODUM LEVEL

AVERAGE OF FOUR SUBASSEMBLY OUTLET TC's

REACTOR UPPER PLENUM PRESSURE SYSTEM

LOCATED IN UPPER PLENUM INSTRUMENT PROBE

ULTRASONIC FLOWMETER MOUNTED IN LOWER PORTION OF THRUTLE VALVE

REACTOR AT-CALCULATED FROM IXH HEAT BALANCE

ULTRASONIC FLOWMETER LOCATED IN LOWER PORTION OF THRUTLE VALVE

PRIMARY COOLING SYSTEM INSTRUMENTATION

Figure 3
EXISTING VALVE PLUG - P₁ & P₂ LOCATIONS

FLOWMETER LEADS

VALVE PLUG

FLOWMETER LEADS

FLOWMETER TRANSDUCERS (2 PAIRS)

SECTION A - A

T/P-C-10222-X

THROTTLE VALVE FLOWMETER VALVE PLUG AND FM TRANSDUCER SUPPORT ASSEMBLY

Figure 4
Figure 6
Figure 7
Figure 8
Figure 9

Temperature vs. Time into Transient

- ○ XX09 Measurement
- ● NOM XX09 Coolant
- --- Reactor Inlet
- ▲ Reactor Inlet Measurement

Time into Transient, s

Temperature, °F

- 1500
- 1000
- 950
- 900
- 850
- 800
- 750
- 700
- 650
- 600
1.0 INTRODUCTION

Indian Nuclear Power programme envisages setting up a series of Fast Breeder Reactors in the early part of the next century. A 40 MW thermal (13 MWe) Fast Breeder Test Reactor (FBTR) has been built with French collaboration and became critical in Oct. 1985. We are planning to set up a 500 MWe Prototype Fast Breeder Reactor (PFBR) at Kalpakkam and the design for this is in an advanced stage. A detailed project report is being finalised for obtaining approval of the Government. In this presentation, the design features of Instrumentation for core monitoring of FBTR is described. Also is presented the philosophy adopted in the design of the Instrumentation for Core Supervision of PFBR.

2.0 FBTR:

In FBTR, instrumentation for core supervision has retained the philosophy followed in Rapsodie with modifications, wherever felt necessary. 85 positions in the reactor core have been provided with 2 nos. of finite time constant thermocouples except the central one which is provided with 3 nos. of T/Cs to measure the outlet sodium temperature from each subassembly. Stainless steel sheathed magnesium
oxide insulated Chromel Alumel T/C's have been the obvious choice because of their fairly linear EMF vs. temperature characteristic and stability at 900 deg.C. Besides, type K T/C's are known to withstand nuclear radiations of the order of $10^{14}$ of 10 n/cm²/Sec. as will be encountered in Fast Reactors. Ungrounded hot junction thermocouples were preferred as failure due to thermal shock is minimum for such type. The Chromel Alumel T/C's also offer the added advantage of higher signal strength (>15 mV) at operating temperatures, facilitating the use of conventional amplifiers for processing the signal. 2 nos. of 1 mm T/Cs are assembled on a probe called "Probe for core temperature measurement". The probe is introduced into a thermowell tube arrangement, so that the probe does not come in direct contact with sodium. The probe assembly has been designed to provide good thermal contact between the T/C's and T/W even in the presence of possible differential expansion. The probes are inserted through 6 T/C passage tubes in the small rotating plug (SRP). Each tube accommodates 14 nos. of T/C probes. The probes are provided with disconnectable lemo connectors at the cold end, to facilitate replacement of the T/C's. Since temperature measurements directly relate to the safety of the reactor, the response of these temperature sensors is required to be fast. A time constant of 2 secs. with minimum dispersion in the response time of any two T/C's was specified. Tests in sodium were conducted, simulating the reactor core conditions. Response time observed were of the order of 4 to 6 secs and the maximum dispersion of 1.8 secs.
was observed. Tests were repeated with Indium filled in the well. Response time improved to values between 0.6 to 1 sec. and also the dispersion between the two time constants was found to be very small (<0.2 sec). However, the probes have been installed only in dry thermowells as reports indicated interaction of Indium with the Stainless steel of the thermowell. The response time was also found to vary with the circumferential position of the gripper in the dry thermowell.

These signals are scanned by the Central Data Processing System (CDPS) which is an on-line dual computer system. Each thermocouple is scanned once in a second. The healthiness of the thermocouple is checked by the computer by comparing the outlet temperature of fuel sub-assembly with inlet temperature of the core. Faulty thermocouples are deemed to have crossed the safety limit. They do not take part in the calculation of mean temperature. The operator is informed about the faulty thermo-couple through display and print out.

The average outlet temperature of the fuel sub-assembly is calculated ($\Theta_m$). This will be compared against alarm, LOR and scram limits. The global temperature rise ($\Delta \Theta_m$) is also calculated by the computer and checked against alarm, LOR and scram limits.
For every sub-assembly, the expected temperature rise is calculated by multiplying the global temperature rise \( (\Delta \Theta) \) with a sub-assembly constant \( (a_i) \). Initially, sub-assembly constants are calculated taking into account thermal power distribution across the reactor core. On demand, computer can also calculate the sub-assembly constants at different power levels. The actual temperature rise along every fuel sub-assembly is also compared with expected temperature rise. The difference is compared with expected temperature rise. The difference is compared with alarm and scram limits. Colour graphic terminal is available for the operator to view the temperature distribution across the reactor core. Operator can view "on-line trend" of the core temperature signals in the colour CRT.

Plant simulator is used to check and validate the core temperature supervision software. On-line diagnostics programme checks the functioning of the core temperature supervision programme and other safety related programmes. If any safety related programme is not operational, on-line diagnostics will order switch over to the stand by computer.

Flow reduction in individual subassemblies is possible only due to blocking of the flow passage. Designs adopted rule out the possibility of a total blockage in FBTR. Even if it were to happen the detection and safety actions are expected to limit crossing transient hot spot clad temperature. Besides, global incidents involving power or flow changes are detected and safety actions initiated. For fuel failure
which can lead to local blockages two methods are employed for early detection of clad rupture, one detecting delayed neutron precursors (DND) in sodium and the other detecting fission gases (GFPD) in the primary cover gas. The DND system actuates the reactor protection system in the event of a large DND signal. The GFPD alerts the operator thro' an alarm in Control room. The DND system adopted for FBTR is similar to Rapsodie with modifications incorporated to improve signal sensitivity. System response time is of the order of 30 secs. The response time of the GFPD system is of the order of 4 to 5 minutes taking into account decay of $^{23}$Ne activity before detection.

3.0 PFBR:

The approach with regard to subassembly accident protection to be adopted for PFBR are broadly enumerated below:-

(a) Conventional Chromel Alumel T/C's in dry well will be sufficient; Signals should be scanned once a second.

(b) Safety actions should be taken based on absolute mean outlet temperature and temperature rise through the subassembly.

(c) Fast response T/C's will be preferred if they are replaceable. Response time for these type of T/C's should be less than 100 ms. However, fast response T/C's are not mandatory since credible blockages due to external debris are not expected to cause sodium boiling.
(d) With fast response T/C's noise measurements for detection of local blockage will be possible. But considering insufficient experience to date, it would not be feasible to take safety action based on noise measurement.

(e) For failed fuel detection, both cover gas monitoring and DND need to be provided.

(f) From reactor availability considerations fast localization of failed fuel will be required.

(g) To limit consequences of BDBA events, acoustic boiling detectors are recommended and safety actions are to be taken based on these measurement.

(h) Flow meters to take care of total flow blockage is recommended.

(i) Detection system should have redundancy and voting logic to ensure a reliability of 99.9%.

The above requirements were reviewed and instrumentation for Supervision of Core Cooling in PFBR is planned on the following lines.

Temperature will be measured at each of the 216 subassemblies using 2 Nos. of conventional Chromel Alumel thermocouples and one number of zero time constant SS-Na T/C. The conventional T/Cs will be replaceable. They will be fixed on probes and inserted into thermowell tubes. The intrinsic T/C's will be integral with the thermowell and non-replaceable. Such a requirement calls for probe design
distinct from FBTR. Conventional T/C's will be MgO insulated stainless sheathed, Chromel-Alumel 1 mm sensors mounted on the probe for core temperature measurement. The probes will slide into the thermowell tube from the attic of the control plug. Probe and tube designs will aim to have the lowest response time possible. Preliminary studies indicate that in order to avoid interferences with shroud tubes some T/C tubes may have 3 bends in different planes. A full scale mock up is being planned towards perfecting design of a probe for core temperature housing the two types of T/C's within a single well.

Scanning speed of 1 sec will be retained as in FBTR. The scanning speed is decided taking into account the time constants of detecting devices, heat propagation constants and the hypothesis of plugging conditions. Each set of 216 T/C's will be scanned and processed by three separate surveillance computers.

Signal processing will be on similar lines as in FBTR with added features to improve accuracies through advanced telemetry techniques. Only signals from conventional T/C's will be used for safety actions.

In order to detect gross blockage of flow through subassemblies, flow measurements are carried out at 4 locations on the outer periphery of the fuel subassemblies. Power to flow ratio will be representative of the temperature rise across the reactor core and will be used as a gross flow blockage signal.
A combined probe type eddy current flow meter (ECFM) with 2 conventional T/C's is under development. ECFM's are good candidates for detecting local boiling, considering their fast response (better than 10 sec) and higher sensitivity (approx. 175 microvolts/m/hr). But for their higher cost, they would have been ideal detection instruments for individual SA flow excursions. Accoustic Boiling detectors are also planned to be provided for early detection of sodium boiling which is presently under development at this centre. However, it is not considered mandatory for the safety of the reactor.

Detection of failed fuel pins will be by GFPD and DND in sodium as in FBTR. Fast localisation of the failed fuel element is planned in PFBR in addition to GFPD and global DND. Considering the large response time (order of a few minutes) no safety actions are envisaged from GFPD signal and it will function only as a pre-alert signal. DND method consists of sampling sodium from eight zones in the pool and monitoring for presence of delayed neutrons in the case of a fuel pin failure. Samples are collected by e.m. pumps and routed to DND blocks installed on the rooftop. The DND localization method involves collection of samples from heads of each individual subassembly with the help of three selector valves provided for this purpose. Each selector valve can select sample from one of the 72 subassemblies.
with the help of a control circuit. Provision to sample one or a group of subassembly is also available. The sample selected is monitored in a separate DND block located on the control plug. By this method a 1 cm equivalent fuel pin rupture can be identified within a maximum period of 15 minutes. At an earlier point of time, when the evolution of a break in a fuel element cladding was not well understood, early removal of the failed sub-assembly from the reactor was considered desirable. Presently, continuous operation of the reactor without its removal is considered permissible. This may allow some redesign to be effected from economic considerations.

4.0 CONCLUSION:
It is recognised that monitoring of sub-assembly outlet temperature is an important feature in Fast Breeder Reactors. Development efforts are underway in this Centre for the design of probe for core temperature measurement, eddy current flow meter and acoustic probes needed for core monitoring. Reliable instrumentation which includes advanced microprocessors with self-diagnostic features and validated software is expected to be available indigenously by the time PFBR construction is taken up.
REFERENCES:


THE KNK II INSTRUMENTATION FOR GLOBAL AND LOCAL SUPERVISION OF THE REACTOR CORE

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ABSTRACT:

After an introduction into the KNK plant itself, their historical development and their present situation, the instrumentation of the global and local supervision of the KNK II-core as well as the main safety-related i and c-systems will be described. Special emphasis is laid on the instrumentation of the reactor protection systems and the shut down systems.

After that some practices are reported about instrumentation behavior and lessons learned from the operation and maintenance of the above mentioned systems.

At last follows a short description of the special instrumentation for the detection of failed fuel subassemblies and of the plant data processing system.

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1. INTRODUCTION / SUMMARY PLANT REVIEW

The Compact Sodium Cooled Nuclear Reactor KNK is the only Fast Breeder Research and Prototype Plant in operation in the Federal Republic of Germany since more than 12 years (See also IAEA Reference Data Series No. 3, facility ref. No. DE 27). It is mainly used for the testing of fuel elements, components and different instrumentation devices in view of the design and construction of the SNR 300. The latter is ready for operation since several years, but the licensing is in an impasse due to diverging opinions between the Federal and State Government. So the fuelling of the reactor has not started.

KNK was ordered in the beginning of the 60ies and is owned by the Nuclear Research Center Karlsruhe (KfK). It was designed and built by the German company Interatom (IA) between 1966 and 1971 and is operated by Kernkraftwerk-Betriebsgesellschaft (KBG), a subsidiary company of the local utility Badenwerk, on the basis of a contract with KfK.

KNK was originally designed as a prototype Nuclear Power Plant to produce superheated steam for electricity production. Thus the heat of the reactor is transmitted on a high temperature level (about 525°C) by respectively two primary and secondary sodium loops to the water-steam-circuit. The thermal output of the plant is 58 MW, the electrical output 20 MW.

According to the goal of the plant the first core was a moderated one (KNK I). Circumion hydride was used as moderator and UO$_2$ with 6.75% enrichment as fuel. Operation started in 1971 and was finished in 1974.

Between the years 1974 and 1977 the reactor was converted into a fast breeder prototype (KNK II) by the original construction company IA and again by order of KfK.

The first KNK II loading (KNK II/1) was designed for 400 effective full power days (EFPD). It started operation in spring 1978 and reached this target in summer 1982.

The second loading (KNK II/2) was designed for 455 EFPD, operation started in summer 1983, the designed burn-up was reached towards the end of 1987.

Due to the good irradiation behaviour of the fuel the lifetime of the second core will be extended up to 720 EFPD. The license to do so was given by the authority on December 1988. The licensing procedure itself lasted 4 years.

The long shut-down period of one year (1988), however, caused a torque increase of the driving mechanism of the control and shut-down rods. Consequently the authority has not allowed to
start the plant so far.

During 1989 a cleaning procedure for the driving mechanism by wetting respectively immersion the driving mechanism into clean hot sodium (400°C) was developed in order to dissolve the oxidated sodium deposits. By this way a very useful and effective cleaning method was found.

Now the plant is ready for start-up, but once again we are waiting for the agreement of our local authority, the State Government of Baden-Württemberg. There should be no more obstacles because in these days the Federal Reactor Safety Committee was in favour of our measures taken to monitor the control resp. shut-down rod driving mechanism and to prevent future oxidation of deposited sodium aerosols (Fig. 1: KNK plant photograph).

2. GLOBAL SAFETY RELATED REACTOR CORE INSTRUMENTATION

The KNK II Core is monitored as a whole by measuring the following parameters:

- neutron flux $\phi$
- primary sodium flow rate $F_p$
- reactor vessel sodium level $L$
- reactor vessel sodium inlet temperature $\theta_{in}$
- delayed neutron emitters $\phi_{d,e}$
- primary vessel leakage $L_v$
- common primary sodium inlet duct leakage $L_{leak}$

All parameters are measured redundant beginning with the sensor up to the signal processing within the reactor protection system. Besides the twofold neutron flux start-up measurement all other measurements are threefold redundant.

The electrical supply-system is just as threefold redundant as the measuring systems.

The KNK II Reactor is equipped with two redundant and diverse shut-down systems (5 respectively 3 shut-down rods) and, since the conversion from KNK I to KNK II, with two diverse protection systems. The first (older) system was built by the German company AEG, the second one by Hartmann and Braun (H+B).

The tables No. 1 through 9 in the appendix identify the different channel groups, their equipment, the trigger signals and levels, the trip signals and special features.
3. LOCAL SAFETY RELATED REACTOR CORE INSTRUMENTATION

The KNK II fuel subassembly (S/A) arrangement consists of two zones (see Fig. 2): the inner zone with 7 test S/A (test zone), containing mixed uranium-plutonium oxide, surrounded by 24 driver S/A (driver zone), containing enriched uranium oxide.

The coolant flow through each S/A is calibrated by a throttle nozzle in the lower grid plate according to the expected power distribution.

The main coolant flow is automatically controlled by the reactor power output in the range of 30 through 100% rated power. Thus the temperature difference along the fuel S/A in the mentioned range is constant.

Each S/A as well of the test as of the driver zone is equipped with 3 mineral insulated thermocouples mounted above the coolant outlet within an open guide tube. Before reaching the TCs the coolant flow passes a mixer unit and an inverse trumpet-like tube (see Fig. 3). By this way the temperature difference between the three TCs is minimized and the TC response time is as fast as possible. The TC itself has an outer diameter of 1.5 mm and is 5 to 6 meters long.

12 TCs respectively are connected with one reference junction of 50°C.

The tables No. 10 through 12 in the appendix identify the different crossing details of TC circuits to form channel groups for the reactor protection system.

4. COMBINED SAFETY RELATED REACTOR CORE INSTRUMENTATION

Besides the global and the local safety related reactor core instrumentation there are two combined instrumentations, where the trip signals are derived from the ratio of two single measurements.

More details of these instrumentations are given in tables 13 and 14 in the appendix.

5. PLANT PROTECTION SYSTEMS

Reactor plants are usually protected against incidents and/or accidents by the following protection installations:
- the reactor protection instrumentation (system), usually threefold redundant, protects the reactor core and its associated components against damage and the environment against hazards,

- the main component protection instrumentation, usually not redundant, protects in general the main components like pumps, heat exchangers, steam generators, feed-water systems, turbine, turbine-generator etc. against damage, and

- the plant process system protection instrumentation, usually twofold but out of two different measurements, protects not only the reactor process systems but also the environment against hazards.

5.1 THE KNK II REACTOR PROTECTION SYSTEM AND ITS ACTIVE SAFETY FEATURES (SAFETY SYSTEM)

The aim of the reactor protection system within the plant safety system is the initiation of all measures to keep the effects of an incident or accident within predetermined limits. Thus the main purpose is the prevention of the release of radioactive substances. These substances are normally kept back by

- the cladding of the fuel pins
- the primary envelope (primary sodium coolant and argon cover gas systems)
- the containment building.

In order to keep these barriers entire, the protection system

- shuts and keeps down the nuclear reaction
- maintains the post heat removal
- initiates the closure of the containment building.

As already mentioned the KNK II has two independant protection systems. The first system controls the whole plant, the second system controls only the reactor area i.e. the reactor core and its associated systems (the second system was fitted back during the conversion of the KNK into a fast reactor).

The logic of the first protection system is a dynamically pulsed and self-checking solid-state system (AEG - logipuls). The logic of the second protection system is a quiescent current relay system with a superimposed automatic solid-state checking system (H+B). Thus the two systems are redundant (because threefold) and diverse.

The first shut-down system actuates 5 shim-safety rods with a magnetic coupling device in the upper part of the rod (shut-
down reactivity 17 $). 

The second shut-down system actuates 3 flexible (articular) safety rods with a coupling device in the lower part of the rod, i.e. under sodium. The release mechanism is mounted in the upper part of the rod (shut-down reactivity 6.1 $).

The normal absorber material in both systems is boron carbide (B\textsubscript{4}C).

The drop time of all shut-down rods is in the range of 700 ± 100 ms. The separation of the coupling device and the coupling release of the second shut-down system allows the measurement of the release time. The measured values are in the range of 40 ± 10 ms. By this way the release mechanism is checked for proper operation.

The trip signals of the global and local instrumentation to shut down the reactor are already discussed in the previous chapters (tables 1 through 14).

Besides these trip signals there exist still 4 trip signals initiated by the sodium level measurement within the two primary pumps. All these signals are 1 out of 1 due to lack in space. Two of the signals are derived from the continuous level measurement, the two other signals come from the discontinuous alarm signal:

- sodium level within the primary pumps low
  (2 times 1 out of 1 per pump)

Furthermore there exist 3 additional trip signals from the secondary sodium and the water-steam system. These trip signals are:

- pressure in the expansion tanks of the two secondary systems low, 3 pressure measurements, trip signals (2 times 2 out of 3, trip level p ≤ 4.2 bar)

- one of the four rupture disks in the two secondary loops broken (leakage detectors, 4 times 2 out of 3, trip levels L ≥ 0)

- steam pressure low and channel deviation high (3 pressure measurements, trip signal 2 out of 3, trip levels p ≤ 60 bar and |Δp| ≥ 4 bar)

Besides the actuation of the shut-down rods the protection system initiates the following additional measures:

- shut-down of the primary pumps by trip signals from the channel groups 4.2, 5, 8, and 9 (primary flow rate variation, reactor vessel sodium level, primary vessel leakage and sodium inlet duct leakage).
- shut-down of the faulty loop (closing of valves) by trip signals

- from the channel group 4.2 (primary flow rate variation) together with a signal "sodium within the primary cell"

- from the channel group 5 (reactor vessel sodium level low) together with the same signal as before

- from the channel group 5 (reactor vessel sodium level high) together with the signal "sodium level secondary expansion tank low"

- closing of the containment building by a trip signal from the channel group 3.1 (neutron flux)

- together with alarm signals from radioactivity measurement in the containment

- together with the smoke monitoring system of the containment.

5.2 THE KNK II MAIN COMPONENT PROTECTION INSTRUMENTATION

As already mentioned the task of this instrumentation is to protect essential (and expensive) components against damage. This is usually done by monitoring the proper function of these components and their auxiliary systems i.e. lubrication of bearings, fluid levels in essential tanks, electrical data of the different supply systems etc.

In this case KNK is equipped like similar plants.

5.3 THE KNK II PLANT PROCESS SYSTEMS PROTECTION INSTRUMENTATION

KNK is equipped with several measuring devices indicating system failures (i.e. sodium leakage) and coincident alarm signals which initiate protective actions. These actions are:

- trip of the reactor

- trip of primary and secondary pumps

- activation of different valves to enclose the defective main component and to start the emergency core cooling by natural convection.
Tabel 15 in the appendix shows the initiating events for these functions in a matrix.

6. SAFETY RELATED INSTRUMENTATION BEHAVIOR

Within the past 10 years of KNK II operation - this are the years 1978 through 1987 - we recorded totally 54 "unwanted" scrams.

The time distribution is shown in Fig. 4 in the appendix.

Figure No. 4 reads as follows:

15 out of 54 trip signals were caused by negative reactivity changes due to gas bubbles travelling through the core after some fuel-element and grid-plate modifications together with the conversion of KNK into a fast reactor (KNK II). The problems could be solved by changing the grid plate throttle device. Additionally the licensing authority could be convinced to bridge this scram signal because of the redundant individual and integral temperature instrumentation of the test S/As by the coolant outlet measurement. (In the KNK II/1 core the signal was bridged from 30.04.81 through 31.01.82 and in the KNK II/2 core from 16.06.83 up today.)

15 out of 54 trip signals were "true" signals, very often caused in the first years of operation together with experiments or by the operators together with the testing of the equipment. (In the first years operators usually are not so familiar with the plant and their response to operator induced changes.)

Only 3 out of 54 trips were caused by mechanical failures of a turbine bearing resp. in the lubrication of a primary pump. The rest of 21 trip signals were caused by electrical and/or electronic equipment or component failures.

The analysis of these 21 trips shows:

- 9 trip signals were initiated by only 4 channel groups of the reactor protection system, mainly by the signal "channel deviation" and

- 12 trip signals were initiated by single alarm signals of different systems or component protection instrumentation.

The tables No. 16 through 18 in the appendix give more details of these trips.

The analysis of the different reactor trip signals allows the following statement:
The availability of the KNK II reactor protection system is very high, the system itself caused no spurious scrams.

Trip signals were very often caused by failures of the power supply respectively the speed control unit of the sodium pumps (9 failures resp. 42% of electrical failures).

Some trip signals were initiated by component failures of not-redundant equipment or by electrical interference problems.

Lessons learned are:

- If possible use threefold redundant measurements for the initiation of reactor trip signals i.e. level control of pumps and tanks.

- In case of equipment malfunction spend enough time for exact failure analysis and checking. (In many cases tables 16 and 17 show the same failure event repeated in a short time distance!)

- In case of replacement of defective items by new ones carry out extensive tests on the new component i.e. external temperature cycling.

7. THE INSTRUMENTATION TO DETECT FAILED FUEL S/As

Fuel S/As with cladding failures normally start with a release of a fraction of fission gas into the coolant, and subsequent into the cover gas. There the fission gases xenon and krypton are monitored. After some time the cladding failure increases and solid fission products are released into the primary coolant. This can be detected by monitoring delayed neutron emitters. This is done at the KNK as already described together with channel group 7.

The KNK has no further internal detection devices as described above. Thus the area of the failed S/A is defined by comparing the DN-signal levels of the two loops. By this way the core-half with the failed fuel S/A is defined. Later on the neutron flux of the reactor is tilted by withdrawing and insertion of single shim-rods with the reactor in operation with 15% rated power. During this procedure the DN-signal levels are observed. The magnitude of variation of the signals correlated with the temperature difference of the S/As in the neighbourhood of the moved shim-rod gives a "hit-list" for the potential defective items. At last the potential failed S/As are drawn into the fuel handling machine. There the gas flow and the pressure is lowered, the tempera-
ture rises and the defective item emits fission gas which can be detected by a \( \gamma \)-spectrometer. Up to now 7 failed fuel S/As were detected with this ex-core hot dry sipping method and replaced.

For the future the installation of an in-core hot wet sipping system was designed and built by INTERATOM in the framework of the R and D program of KfK and will be tested soon.

8. DATA PROCESSING AT THE KNK II

Together with the modification of KNK into a fast reactor (KNK II) a data processing system (AEG 60-50) was installed. It is a free programmable, on-line open-loop computer.

In the control room are installed:

- 1 control console
- 2 record printers (plus 1 line printer and 1 record printer in the computer room)
- 2 data displays (1 monitor for alphanumeric process information, 1 video display)

The process computer has no safety-related function, it is only used for the information of personnel in the control room.

The CPU has a 32 k core memory, the external data storage consists of 2 magnetic drums (512 k each) and 1 magnetic tape (800 bpi).

The system generates 6 types of protocols:

- **SSP** Malfunction and control procedure protocol. It is activated anytime and records nearly everything which gives a binary signal.
- **SAP** Malfunction and signal sequence protocol. This program is activated by special signals or coincidences of these.
- **STP** Hourly record of selected operating data.
- **MVP** Program for the recording of the transient behavior of up to 15 analogous signals with 9 possible scanning modes from 1 sec to 20 min.
- **MRP** This program allows to ask for analogous operating signals in retrospective. It also can be used as a "post mortem" protocol.
- **MVS** This program brings up to 12 analogous signals to the screen with a scanning mode of 10 seconds each.
Further there are some special programs realised:

DEPRO  A program surveying the steam generator behavior

BEPROR  A program for surveying the temperature of core S/As.

9. APPENDIX

Figures 1 through 4
 Tables 1 through 18
### TAB. 1: START-UP NEUTRON FLUX MEASUREMENT

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>1</th>
</tr>
</thead>
<tbody>
<tr>
<td>neutron flux measurement *, start-up range</td>
<td></td>
</tr>
</tbody>
</table>

**EQUIPMENT:**

twofold
2 BF$_3$ counters, log. amplifiers, differential amplifiers

**TRIGGER SIGNALS:**

1 out of 2: neutron-flux level high

**TRIGGER LEVELS:**

50% of last decade (5 decades)

**TRIP SIGNALS:**

first shut-down system (1. SDS)

**SPECIAL FEATURES:**

BF$_3$ counters are withdrawn during start-up
No withdrawal of control rods, if measuring signal ≤ 10 cps
The trigger signals are switched to "2 out of 2", if the high range channels are in proper operation (Φ ≥ 10% Φm)
**TAB.2: MEDIUM NEUTRON FLUX MEASUREMENT**

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
</tr>
</thead>
<tbody>
<tr>
<td>neutron flux measurement $\Phi$, medium range</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EQUIPMENT:</th>
</tr>
</thead>
<tbody>
<tr>
<td>threefold</td>
</tr>
<tr>
<td>3 boron coated, $\gamma$-compensated ionization chambers, log. amplifiers, differential amplifiers</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 out of 3 : 1) neutron flux level high</td>
</tr>
<tr>
<td>2) reactor period short</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER LEVELS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>1) $\Phi \geq 20% \Phi_n$</td>
</tr>
<tr>
<td>2) $T \leq 10 \text{ s}$</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIP SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>first shut down system (1. SDS)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SPECIAL FEATURES:</th>
</tr>
</thead>
<tbody>
<tr>
<td>The trigger signals are by-passed, if the high-range channels are in proper operation ($\Phi \geq 10% \Phi_n$)</td>
</tr>
</tbody>
</table>
### TAB. 3: HIGH NEUTRON FLUX MEASUREMENT

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>3</th>
</tr>
</thead>
<tbody>
<tr>
<td>neutron flux measurement $\phi$, high range (power range)</td>
<td></td>
</tr>
</tbody>
</table>

**EQUIPMENT**

sixfold
6 boron coated, uncompensated ionization chambers, linear amplifiers
3 ionization chambers are combined to one channel group respectively

**TRIGGER SIGNALS:**

2 times 2 out of 3: neutron flux level high
2 times: channel deviation in each channel group

**TRIGGER LEVELS:**

1) $\phi \geq 112\% \phi_n$ and $|\Delta \phi_n| \geq 20\% \phi_n$
2) $\phi \geq 120\% \phi_n$ and $|\Delta \phi_n| \geq 6\% \phi_n$

**TRIP SIGNALS:**

1) first shut down system (1. SDS)
2) second shut down system (2. SDS)

**SPECIAL FEATURES:**

The channels of either channel groups are continuously checked on each other. In case of substantial deviation a trip signal is caused.

All 6 channels are additionally equipped with reactivity meters which give only alarm signals (-5¢ resp. -7.3¢) but no trip signal.
Tab. 4: Primary Sodium Flow Rate Measurement

**Channel Group:**

| primary sodium flow rate measurement $F_p$ |

**Equipment:**

1. Magnetic flow meter respectively in either primary loops
2. Electrodes, cabling, and electronic devices threefold
3. Electrical summation of the two measurement signals
4. Differential amplifiers

**Trigger Signals:**

2 out of 3:
1) Flow rate low
2) Variation of flow rate high
2 times: channel deviation in each channel sub-group

**Trigger Levels:**

1) $F_p \leq 20\%$ and $|\Delta F_p| \geq 10\% F_p$
2) $|\Delta F_p| \geq 4\% F_p$
3) $|dF_p/dt| \geq 1 \%/s$

**Trip Signals:**

1) and 2): first shut down system (1. SDS)
   the smaller channel deviation ($\geq 4\%$) activates the
   second shut down system (2. SDS)

**Special Features:**

The channels of the channel group are continuously checked on each other. In case of substantial deviation a trip signal for both shut-down systems is caused.
### Tab. 5: Reactor Vessel Sodium Level Measurement

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor vessel sodium level measurement L</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EQUIPMENT:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Threefold</td>
<td></td>
</tr>
<tr>
<td>3 inductive sensors with continuous measurement and two independent level signals each (twin coils)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER SIGNALS:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>4 times 2 out of 3:</td>
<td></td>
</tr>
<tr>
<td>1) 2 times sodium level low</td>
<td></td>
</tr>
<tr>
<td>2) 2 times sodium level high</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER LEVELS:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>1) $L_1 \leq 1177 \text{ mm}$</td>
<td></td>
</tr>
<tr>
<td>$L_2 \leq 1127 \text{ mm}$</td>
<td></td>
</tr>
<tr>
<td>2) $L_1 \geq 2110 \text{ mm}$</td>
<td></td>
</tr>
<tr>
<td>$L_2 \geq 2160 \text{ mm}$</td>
<td></td>
</tr>
</tbody>
</table>

Level indication during reactor operation (500°C sodium temperature) equals to $1900 \pm 50 \text{ mm}$ and during shut down (200°C sodium temperature) to $1450 \text{ mm}$

<table>
<thead>
<tr>
<th>TRIP SIGNALS:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>$L_1$ initiates the first shut down system</td>
<td></td>
</tr>
<tr>
<td>$L_2$ initiates the second shut down system</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SPECIAL FEATURES:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>The trigger pulse $L_1$ comes from the independent level signal, the trigger pulse $L_2$ comes from the continuous level measurement.</td>
<td></td>
</tr>
</tbody>
</table>
### TAB. 6: VESSEL SODIUM INLET TEMPERATURE MEASUREMENT

| CHANNEL GROUP: | 
|---|---|
| reactor vessel sodium inlet temperature $\theta_{in}$ | 6 |

| EQUIPMENT: |
|---|---|
| threefold 3 thermocouples (TC) in each primary loop at the outlet of the intermediate heat exchangers |

| TRIGGER SIGNALS: |
|---|---|
| 2 times 2 out of 3: sodium temperature level high |

| TRIGGER LEVELS: |
|---|---|
| $\theta_{in} \geq 380^\circ C$ and $|\Delta \theta_{in}| \geq 70$ K |

| TRIP SIGNALS: |
|---|---|
| first shut down system |

| SPECIAL FEATURES: |
|---|---|
| The channels of either channel groups are continuously checked on each other. In case of substantial deviation a trip signal is caused. |
**TAB. 7: DELAYED NEUTRON MONITORING**

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>7</th>
</tr>
</thead>
<tbody>
<tr>
<td>delayed neutron monitoring $\phi_{\text{d}}$</td>
<td></td>
</tr>
</tbody>
</table>

**EQUIPMENT:**

- 2 times threefold for each loop
- 12 lead shielded He-3-counters (2 counters in parallel for each channel) mounted at the hot leg of the two loops within a polyethylene bloc,
- 6 amplifiers, discriminators and log. ratemeters

**TRIGGER SIGNALS:**

- 2 times 2 out of 3: count rate high

**TRIGGER LEVELS:**

- $\phi_{\text{d}} \geq 2000$ cps and $|\Delta \phi_{\text{d}}| \geq 800$ cps

**TRIP SIGNALS:**

- first shut down system

**SPECIAL FEATURES:**

- The channels of either channel groups are continuously checked on each other. In case of substantial deviation a 3 min delayed trip signal is caused.
- The trip levels correspond with about 80 - 90 cm$^2$ effective free fuel surface exposed to sodium.
- The travelling time from the core up to the detectors is less than 30 s under normal operation conditions.
### TAB.8: PRIMARY VESSEL LEAKAGE

| CHANNEL GROUP: |  
|----------------|--------------------------------------------------|
| primary vessel leakage detection \( L_v \) | 8 |

#### EQUIPMENT:

threefold
3 leak detectors (electrical conductivity of sodium) mounted in the lowest part of the safety vessel, in which the primary vessel is installed (double-walled-vessel)

#### TRIGGER SIGNALS:

2 out of 3: sodium in the safety vessel

#### TRIGGER LEVELS:

\( L_v \neq 0 \)

#### TRIP SIGNALS:

first shut-down system

#### SPECIAL FEATURES:

The cabling is terminated by a resistance to check discontinuity.
**TAB. 9: INLET DUCT LEAKAGE**

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>9</th>
</tr>
</thead>
<tbody>
<tr>
<td>common primary sodium inlet duct leakage detection L1a.</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EQUIPMENT:</th>
</tr>
</thead>
<tbody>
<tr>
<td>threefold</td>
</tr>
<tr>
<td>3 inductive leak detectors (twin coils) mounted in the lowest part of the double walled tube</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 out of 3: sodium in the system</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER LEVELS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>L1a ≠ 0</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIP SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>first shut-down system</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SPECIAL FEATURES:</th>
</tr>
</thead>
<tbody>
<tr>
<td>continuous functional test by measuring the transfer factor of the twin coils</td>
</tr>
</tbody>
</table>
### TAB. 10: S/A SODIUM OUTLET TEMPERATURE

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>10</th>
</tr>
</thead>
<tbody>
<tr>
<td>individual S/A sodium outlet temperature measurement $\theta_{S/A}$</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EQUIPMENT:</th>
</tr>
</thead>
<tbody>
<tr>
<td>threefold</td>
</tr>
<tr>
<td>3 TCs above each S/A of the test and the driver zone, reference junction, amplifiers</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 out of 3: coolant outlet temperature high (all S/A)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER LEVELS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\theta_{S/A} \geq \theta_{in} + f_{\Delta} \Delta \theta + \delta_{s}$ (sliding)</td>
</tr>
<tr>
<td>$\geq \theta_{in} + f_{\Delta} \left[ \frac{1}{n} \sum_{j=1}^{n} (\theta_{out_j} - \theta_{in}) \right] + \delta_{s}$</td>
</tr>
</tbody>
</table>

Two mean values of the difference temperatures $\Delta \theta$ as well of the test zone (7 S/A) as of the driver zone (12 selected S/A) are used to form the trigger levels.

The power distribution coefficient $f_{\Delta}$ is calculated according to the neutron flux distribution and the sodium flow calibration for each S/A. It may be corrected at 40%, 80% and 100% power operation due to the measured values.

The actuating distance $\delta_{s}$ is set to 30 K for normal S/A. It may be adjusted to experimental S/A if necessary.

<table>
<thead>
<tr>
<th>TRIP SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>first shut-down system</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SPECIAL FEATURES:</th>
</tr>
</thead>
<tbody>
<tr>
<td>The trigger levels $\theta_{S/A}$ follow the variation of the coolant inlet temperature $\theta_{in}$ and the mean value of the temperature difference of the coolant between inlet and outlet of the two groups &quot;test S/A&quot; and &quot;selected driver S/A&quot; respectively. Thus the trigger levels follow power variations (sliding trigger levels).</td>
</tr>
<tr>
<td>In the power range from 30% through 100% the temperature difference along the S/A is maintained nearly constant by automatic control of the main coolant flow according to the power output of the reactor.</td>
</tr>
</tbody>
</table>
**TAB. 11: TEST S/A SODIUM OUTLET TEMPERATURE**

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
<th>11</th>
</tr>
</thead>
<tbody>
<tr>
<td>individual sodium outlet temperature measurement of the S/As of the test zone $\theta_{\text{test S/A}}$</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EQUIPMENT:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>threefold 3 TCs above each S/A of the test zone (identical TCs to No. 10), reference junction, amplifiers</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER SIGNALS:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>2 out of 3: coolant outlet temperature high (test S/A only)</td>
<td></td>
</tr>
<tr>
<td>channel deviation in each channel group</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER LEVELS:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>$\theta_{\text{test S/A}} \geq \theta_{\text{calc}} + 30$ K (fixed) and $</td>
<td>\Delta \theta</td>
</tr>
<tr>
<td>$\theta_{\text{calc}}$ = coolant outlet temperature of S/A, calculated for 100% rated power of the reactor</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIP SIGNALS:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>first shut-down system</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SPECIAL FEATURES:</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>The trigger level stays constant if the power output is changed (fixed trigger levels).</td>
<td></td>
</tr>
</tbody>
</table>
### CHANNEL GROUP:

Integral sodium outlet temperature measurement of the test zone $\theta_{t\text{est}}$ and its variation $d\theta_{t\text{est}}/dt$

### EQUIPMENT:

Threefold
3 TCs above each S/A of the test zone (identical TCs to No. 10), reference junction, amplifiers, differential amplifiers. The three TCs are selected from different S/As and combined to one channel group.

### TRIGGER SIGNALS:

2 out of 3:
1) coolant outlet temperature high
channel deviation within the channel group high

2 times 2 out of 3:
2) variation of coolant outlet temperature high

### TRIGGER LEVELS:

1) $\theta_{t\text{est}} \geq \theta_{S/A \text{ select}} + 36$ K (fixed for 100% rated power)
   and $|\Delta \theta_{S/A \text{ select}}| \geq 40$ K

2a) $|d \theta_{t\text{est}}/dt| \geq 3$ K/min if $|\Delta \theta_{t\text{est}}| > 30$ K

2b) $|d \theta_{t\text{est}}/dt| \geq 2$ K/min if $|\Delta \theta_{t\text{est}}| > 20$ K

The trigger signals 2a) and 2b) are selected from 2 times 3 different S/As.

### TRIP SIGNALS:

1) and 2a): second shut-down system
2b): first shut-down system

### SPECIAL FEATURES:

To eliminate scrams caused by fast but in magnitude small changes of the outlet temperature the trigger signals to the first resp. second shut-down system are blocked for a small variation of magnitudes.
**CHANNEL GROUP:**
ratio high range neutron flux measurement vs. primary sodium flow rate measurement $\Phi/F_P$

**EQUIPMENT:**
2 times threefold ratio of measurements according to table 3 and table 4

**TRIGGER SIGNALS:**
2 times 2 out of 3: ratio disturbed

**TRIGGER LEVELS:**
1) $\Phi/F_P \geq 1.12$
2) $\Phi/F_P \geq 1.20$

**TRIP SIGNALS:**
1) first shut-down system
2) second shut-down system

**SPECIAL FEATURES:**
The primary flow rate measurement controls the trip signal of the neutron flux measurement.
### TAB.14: RATIO OF PRIMARY AND SECONDARY FLOW RATE

<table>
<thead>
<tr>
<th>CHANNEL GROUP:</th>
</tr>
</thead>
<tbody>
<tr>
<td>ratio of primary flow rate measurement vs. secondary flow rate measurement $F_p/F_s$</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EQUIPMENT:</th>
</tr>
</thead>
<tbody>
<tr>
<td>threefold</td>
</tr>
<tr>
<td>1 magnetic flow meter in either primary loops (see tab.4) and in either secondary loops</td>
</tr>
<tr>
<td>electrical summation of the signals respectively</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 out of 3: ratio disturbed</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIGGER LEVELS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>$</td>
</tr>
<tr>
<td>$F_p = 1.1 \cdot F_s$</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TRIP SIGNALS:</th>
</tr>
</thead>
<tbody>
<tr>
<td>first shut-down system</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SPECIAL FEATURES:</th>
</tr>
</thead>
<tbody>
<tr>
<td>The normal flow rate in the primary circuit is 10% higher than in the secondary circuit.</td>
</tr>
</tbody>
</table>
**TAB. 15: MATRIX OF COINCIDENT KMK II ALARM SIGNALS TO INITIATE PROTECTIVE ACTIONS**

<table>
<thead>
<tr>
<th>ALARM SIGNALS</th>
<th>SODIUM IN PRIMARY CELL</th>
<th>SODIUM LEVEL REACTOR VESSEL HIGH</th>
<th>SMOKE ALARM STEAM GENERATOR BUILDING</th>
</tr>
</thead>
<tbody>
<tr>
<td>VARIATION OF PRIMARY FLOW RATE HIGH</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SODIUM LEVEL REACTOR VESSEL LOW</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SODIUM LEVEL SECONDARY EXPANSION TANK LOW</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>PRESSURE IN SECONDARY SODIUM SYSTEM LOW</td>
<td>X</td>
<td></td>
<td>X</td>
</tr>
</tbody>
</table>

135
Tab. 16: Trip signals initiated by the reactor protection system due to external component failures

<table>
<thead>
<tr>
<th>DATE OF EVENT</th>
<th>INITIATING CHANNEL GROUP</th>
<th>CHANNEL INITIATION CAUSED BY</th>
</tr>
</thead>
<tbody>
<tr>
<td>21/06/78</td>
<td>channel deviation θout test S/A</td>
<td>defect of two TCs of different S/As within one channel group</td>
</tr>
<tr>
<td>26/05/81</td>
<td>channel deviation θ delayed neutrons</td>
<td>electrical interference signals on one channel group</td>
</tr>
<tr>
<td>02/06/81</td>
<td></td>
<td></td>
</tr>
<tr>
<td>28/07/81</td>
<td>change of flow-rate</td>
<td>failure in the electronic control unit of a primary pump (speed control), 2 x potentiometer defects, than 2 x transistor defect of an inductive unit which was used to replace the potentiometer</td>
</tr>
<tr>
<td>07/08/81</td>
<td></td>
<td></td>
</tr>
<tr>
<td>24/03/86</td>
<td></td>
<td></td>
</tr>
<tr>
<td>25/04/86</td>
<td></td>
<td></td>
</tr>
<tr>
<td>06/11/84</td>
<td>change of S/A outlet temperature</td>
<td>failure in the automatic power control of the plant (water steam-circuit)</td>
</tr>
<tr>
<td>15/11/84</td>
<td></td>
<td></td>
</tr>
<tr>
<td>DATE OF EVENT</td>
<td>TRIP SIGNALS INITIATED BY A SINGLE EVENT OR COMPONENT MALFUNCTION</td>
<td></td>
</tr>
<tr>
<td>--------------</td>
<td>---------------------------------------------------------------</td>
<td></td>
</tr>
<tr>
<td>25/08/79</td>
<td>fuse blown in primary pump supply, shut-down of primary pump</td>
<td></td>
</tr>
<tr>
<td>10/01/80 1)</td>
<td>interruption in the excitation circuit of a rotating dc/ac converter, power interruption of one channel: 1) trip signal of the neutron flux start up channel (1 out of 2) 2) coincident trip signals: level low expansion tank and smoke alarm containment</td>
<td></td>
</tr>
<tr>
<td>09/02/80 2)</td>
<td>component failure in the sodium level measurement of a primary pump (1 out of 1), shut-down of primary pump</td>
<td></td>
</tr>
<tr>
<td>24/03/81</td>
<td>failure of one BF$_3$-counter-tube of the start-up neutron flux measurement (1 out of 2)</td>
<td></td>
</tr>
<tr>
<td>04/05/84</td>
<td>failure in the electronic control unit of a primary pump, fuse blown, pump shut-down (see also 25/08/79 and table No. 16, 4 times similar event)</td>
<td></td>
</tr>
<tr>
<td>31/10/83</td>
<td>short circuit of a relay within the 2 out of 3 unit</td>
<td></td>
</tr>
<tr>
<td>21/08/86</td>
<td>failure in the level measurement of the feed water tank, feed water pump shut-down, trip by hand</td>
<td></td>
</tr>
<tr>
<td>DATE OF EVENT</td>
<td>INITIATING EVENT</td>
<td></td>
</tr>
<tr>
<td>--------------</td>
<td>------------------</td>
<td></td>
</tr>
<tr>
<td>01/04/79</td>
<td>delayed neutrons high: 1. fuel S/A failure</td>
<td></td>
</tr>
<tr>
<td>15/01/80</td>
<td>sodium level reactor vessel high (experiment with sodium level change)</td>
<td></td>
</tr>
<tr>
<td>23/03/80</td>
<td>variation of outlet temperature S/A high (gas bubbles?)</td>
<td></td>
</tr>
<tr>
<td>18/04/80</td>
<td>pressure expansion tank secondary sodium system low</td>
<td></td>
</tr>
<tr>
<td>19/08/80</td>
<td>delayed neutrons high: 2. fuel S/A failure</td>
<td></td>
</tr>
<tr>
<td>05/05/81</td>
<td>variation of outlet temperature S/A high (gas bubbles?)</td>
<td></td>
</tr>
<tr>
<td>20/12/81</td>
<td>primary sodium flow-rate low, variation high man-induced failure during emergency diesel supply testing</td>
<td></td>
</tr>
<tr>
<td>26/06/83</td>
<td>power range high: measuring range not adapted during start up of reactor</td>
<td></td>
</tr>
<tr>
<td>25/10/83</td>
<td>power interruption of 110 kV line</td>
<td></td>
</tr>
<tr>
<td>03/04/84</td>
<td>channel deviation integral temperature measurement test S/A man-induced failure during testing</td>
<td></td>
</tr>
<tr>
<td>12/04/84</td>
<td>steam pressure turbine low man-induced failure during start-up</td>
<td></td>
</tr>
<tr>
<td>10/10/85</td>
<td>variation of temperature outlet S/A high instability of automatic power control in low power range</td>
<td></td>
</tr>
<tr>
<td>29/04/87</td>
<td>delayed neutrons high: 7. fuel S/A failure</td>
<td></td>
</tr>
</tbody>
</table>
Fig. 3: Upper part of KNK 2/A with TC installation for coolant outlet temperature measurement.
Fig. 4: Number of unwanted reactor scrams 1978 - 1987
KNK II core design

Configuration Apr. 89

- test fuel elements
- unmoderated driver elements
- moderated driver elements
- control-safety rods (1. system)
- safety rods (2. system)
- breeder subassemblies (radial blanket)
- reflector subassemblies (stainless steel/ZrH$_x$)
- reflector subassemblies (stainless steel)
MEASUREMENT OF COOLANT FLOWRATE THROUGH THE FUEL ASSEMBLIES IN BN 350 AND BN 600 REACTORS

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Abstract

Methods of the primary circuit coolant flowrate measurement in BN 350 and BN 600 reactors are described. Flowmeter design and parameters are outlined. Flowmeter application during reactor conditions and the results of measurement are presented. Details of the modified flowmeter to be used in BN 600 reactor, that enables its verification during reactor operation by the correlation method have been briefly treated.
Coolant flow for the fuel assemblies (FAs) cooling is one of the major parameters of the nuclear power plant. It is evident that flow monitoring is a prerequisite of its safe operation. Flow measurement allows to form an additional signal required to provide the multilevel core emergency protection system from the FAs overheating. In addition there is a potential to obtain data about the FAs hydraulic characteristics variations and the entire primary circuit during the plant operation.

Monitoring of the bulk coolant flow to a great extent depends on the primary circuit equipment arrangement. In case of a loop layout (BN 350 reactor) it becomes more convenient to measure the flowmeter by way of mounting the flowmeter directly on the loops piping. For reactor plants (BP) with a 300 mm piping the acceptable accuracy of measurement can be provided by application of the conduction electromagnetic flowmeters. Yet, with the reactor power increase that results in the piping diameter growth, the flowrate measurement becomes more difficult due to the magnets weight difficulties of their calibration non-linearity of calibration characteristics.
During the flowmeter development for the BN350 reactor determination of the calibration characteristic has been experimentally accomplished comparing the flowmeter readings with the measurement results obtained by a standard flowmeter. However, the magnetic induction changes occurred during the flowmeter, shipment and installation, differences in the velocities profile on the test rig and in the plant resulted in the unacceptable errors of the coolant flowrate measurements. Therefore, the magnetic flowmeters readings are used to form the relative signals of the emergency protection.

In the plants with the integral equipment layout (BN600 reactor) application of the electromagnetic flowmeters in the primary circuit faced serious problems. That is conditioned by the difficulties of their installation and replacement and adverse operation. The flowrate measurement in the by-pass line is one of the solutions that allows to determine indirectly the primary circuit coolant flowrate. The by-pass line connects two circuit points that are under different pressure and can be arranged beyond the reactor vessel (RV), that provided favourable flowmeter operation conditions, maintenance and calibration. The measured flowrate is in a certain ratio to the primary circuit bulk total flow that depends on the paths hydraulic characteristics. With the integral layout the by-pass can be made either to the core (from the high pressure plenum to the low pressure plenum) or the main circulation pump (from the discharge piping to the pump suction plenum).

The by-pass hydraulic characteristic can be obtained with the adequate accuracy on the water test rig. If the core and pump hydraulic characteristics are chosen rather accurately, then it is quite possible to find out the primary circuit
actual flowrate and monitor it during operation.

This method is adopted in BB600 reactor. The core by-pass line has four electromagnetic flowmeters. The flowmeters readings are used to form the emergency protection signals in response to the primary circuit coolant flow changes. Flowmeters application revealed that they can be used for the reliable and efficient determination of the core hydraulic characteristic changes.

The major drawback of the flowrate measurement method adopted in BN-600 reactor is that the by-pass line goes beyond the RV and is a potential source of the radioactive sodium ingress in the rooms should the piping leak. More preferable in this respect is the by-pass line located in the primary circuit main circulation pumps that is confined within the RV. Coolant enters the by-pass line from the pump discharge piping and leaves it through the holes of the tube under the sodium free level in the pump tank. This design decision is adopted in reactor BN-800.

To enhance the reactor safety it is advisable to provide continuous coolant flow monitoring through each PA of the core. Yet, technically such measurements are rather difficult to accomplish especially for the FAs of BN 600 reactor which have the head with the lateral coolant outlet.

A routine flow control effected during refuellings and maintenance with the shutdown reactor is more practical. In this case there is no continuous flow monitoring through the FAs, yet it is always possible to verify the flowrate through the loaded FAs cross-section and its changes during operation.

A special flowmeter has been developed to carry out these in BN-350 and BN-600 reactors. Fig. 1 shows the flowmeter design.
The flowmeter comprises two major parts. They are flanged tube and movable part. The flowmeter housing is a flanged tube that functions as a guide for the movable part, a shield and also as a support for the movable part drive.

The movable part is a tube that has a built-in nut at the upper end. The nut is engaged with the drive's lead screw. Rotation of the screw built in the support flange results in a movable part 320 mm upward and downward movement.

The movable part has a built-in magnetic sensor and the extension. The magnetic sensor is a streamlined rod that contains a permanent magnet. At the movable part downward traveling the extension covers the FA head until its hard-faced edge finally touches the FA tapered part. Thus FA sealing is effected due to which the coolant goes up the extension past the sensor and then leaves the extension through the holes in the plenum above the core.

The sensor's magnet is operable at temperatures up to 600 C that provides reliable and extended long-term service at temperature range 250-350 C. Next to the magnet there are two built-in thermal couples the coolant temperature measurement at the FA outlet. The thermal couples and magnet's leads pass on the inner side of the movable part and finally are brought out through a helical compensation device area.

When reactor is shutdown for refuelling and maintenance the flowmeter is placed in a cell of the rotating plug. By rotating the plugs the flowmeter may be put on any FA of the core, blanket or storage. The measurements are accomplished when the primary circuit pumps are operated at reduced speed. The coolant flowrate range through the FAs measured by the flowmeter is 0.8-15 m/hour.
Taking into account that the flowmeter has a specified pressure loss the measured coolant flowrate through the FA with the flowmeter on it is different from the actual one. The difference depends on the ratio of the FA pressure loss and that of the flowmeter. On the basis of FA and flowmeter available characteristics the measured flowrate has been adjusted by 1.01-1.18 depending on flow orificing zones relationship between the flowrate through the FA and the potentiometer readings has been verified on a special sodium test rig. The sodium flowrate measurement has been accomplished by the volumetric method. Flowmeter application at BN 350 reactor has revealed its efficiency and ability for measuring the coolant flowrate through any type of FA. The measurements allowed to calculate the flowrate deviation in each flow orificing zone relative to the average value and also to find out the bulk flowrate in the primary circuit. The sodium flowrate deviations from the average value in the core FA have not exceeded \( \pm 2\% \) and corresponded to the water flowrate deviations obtained on the water test rig. More sizeable flowrate deviations from the average value for the given flow orificing zone have been observed in the side blanket FA. They have been caused by the leak between the FA nozzle and the header into the overlapped orifices. The effect of the coolant asymmetric feeding through the loops on the flowrate through the FAs has been also verified by the aid of the flowmeter.

All measurements have been accomplished with the pumps running at 250 rev/min and sodium temperature between 200 and 280 °C.

Now in BN 350 reactor the procedure of flowrate measurement through the FAs has been included in the reactor regulations and is accomplished not less than once in two years.
In BN 600 reactor the flowrate measurements through the FAs have been accomplished during commissioning phase and repeatedly during its operation. The measurements revealed that the coolant flowrate through the core FAs corresponds to the design values and is characterized by a small deviation relative the averaged ones in the given flow orificing zone. More sizeable deviations as well as in BN 350 reactor, has been observed in the side blanket and storage FAs.

The possibility of experimental reactor hydraulic characteristic determination on the basis of available primary pumps characteristics has become an important practical objective of the flowmeter application. The fact that the characteristics of two and three primary pumps have different slope at the point with the operation parameters with the same pressure difference over the core and hence, flowrate through the FAs has been used in the calculations (I). Thus if on the basis of the flowrate measurement results through the FAs the rotation speed of two and three primary pumps can be determined with the same flowrate through the FAs, then the point of their intersection would represent an adequate point of the primary circuit hydraulic characteristic.

Having in mind, that in this method of the primary pump speed determination the flowmeter e.m.f. is taken but not the actual flowrate through the FAs, the flow coefficient error is of no significance. Moreover the primary pump rotation speed measurement accomplished in different FAs enables to reduce the random component of the error in the primary circuit hydraulic characteristic.

Flowrate measurements through the FAs in BN 600 reactor that
have been accomplished during its operation revealed that the flowmeter did not always function properly. There were some cases when the flowmeter indicated the reduced flow through the FAs as compared to the actual one. That has been found out by way of multiple flowrate measurement including the FAs replacement if necessary. Flowrate reduction was caused by the inadequate flowmeter's extension sealing on the FA tapered part. The gap between the extension and FA head allowed part of the flow to avoid the magnetic sensors, that reduced the level of the signal to the potentiometer.

As is known, the applied magnetic transducers apart from the undeniable advantages such as lack of inertia and contact with the coolant have a substantial drawback. That is related to the problems of their metrological certification and calibration during operation.

The first flowmeters for BN 350 and BN 600 reactors have not been metrologically certified. They have been only calibrated in the sodium test rig to set the dependence between the signal from the magnets and flowrate. The accuracy of calibration was ± 2.5% for a wide flow range. Only for small flowrate values the error reached 5-8%. Yet, the sodium test rig where calibration has been accomplished failed to meet the necessary metrological requirements. In addition the flowmeter design prevented application of correlation for its calibration.

Therefore the flowmeter design has been completely revised to improve both mechanical and measuring parts. Two magnetic systems have been added to the measuring part, each having 6 leads used for the flowrate and correlation measurement. The signal from one of the magnetic systems is accepted as the major signal, while the other system remains redundant.
The coolant eddy patterns used as the marks in the correlation method of the flowrate measurement are induced by the eddy generator, which is a segment blocking one third of the extension cross-section. The eddy generator is welded to the extension wall at the inlet to the measuring section. Concurrently with the flowmeter design modification the test rig development has been underway that would have met all the necessary requirements on the accuracy of measurement during calibration. The test rig comprises the experimental section for installation of the correlation flowmeter of various types and configuration and equipment for the flow measurement by the volumetric method.

Presently, the flowmeter for BN 600 reactor has passed the metrological certification and tested in the test rig and reactor operations conditions. The flowmeter provides the flowrate measurement ensuring its metrological verification for the magnetic method in the range of 1... 20 m /hg, and for the correlation method in the range of 8... 20 m /hg. The relative measurement error is ± 2% of the flowrate value.
I. Шейнкман А.Г., Карпенко А.И., Лыжин А.А., Сапегин С.М.,
Теличко М.Т., Шабалин А.С. Экспериментальное определение гид­
равлических характеристик первого натриевого контура установки
БН-600 Белоярской АЭС. Теплофизика ядерных энергетических ус­
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Кирова, 1983г.
Fig. 1 Flowmeter
An Ultrasonic Technique for the Remote Measurement of Breeder Subassembly Outlet Temperature

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12th to 15th December 1989
Introduction

In the Liquid Metal fast Breeder Reactor (LMFBR) there has been concern to avoid positioning structural components in that region of the above core plenum which is directly above the core breeder boundary. This is because the temperature fluctuations in this region might cause fatigue. One consequence of this is that it is difficult to mount thermocouples in positions from which they can monitor the breeder element outlet temperatures accurately.

In the UK a joint R&D programme involving NNC, CEGB, and UKAEA has been launched to develop a method of measuring the breeder outlet temperatures from a remote sensor using high frequency sound waves. This paper explains the principle of the technique and goes on to describe experiments in static sodium which illustrate it in operation over the temperature range from 200°C to 600°C. The turbulent mixing above the subassemblies has an adverse effect on the transmission of the sound waves and experiments to investigate this effect are described. Finally proposals for tests of measurement of mean temperature and temperature noise in flowing sodium are discussed.

Basic Principle

The measurement depends on the fact that the velocity of sound in sodium is a function of temperature, it is a linearly decreasing function given by the equation:

\[ c = 2577.25 - 0.528 T \]

where \( c \) is the velocity of sound in metres/sec and \( T \) is the temperature in degrees celsius. If the velocity of sound can be measured the temperature can be deduced using the above equation. In the realisation of the technique proposed for use in an LMFBR (Ref 1) the velocity is determined by observing the difference between the time of arrival of two echo pulses, one from the front edge and the other from the back edge of a subassembly. This time difference is the time taken for the sound wave to traverse the diameter of the subassembly twice. Since the diameter of the subassembly is known, the velocity and hence the temperature can be deduced. The diameter of the subassembly will vary due to thermal expansion, an iteration of the calculation is necessary to allow for this. An important feature of the measurement is that the temperature is measured directly at the mouth of the subassembly whereas thermocouples have to be positioned some distance above the subassembly top.

The calculated temperature is a spatial average over the path of the sound beam across the subassembly. The ultrasonic method is capable of making up to 400 measurements per second and so is suitable for measuring the fluctuations of the temperature signal, temperature noise, which can give an early warning of localised hot regions within the subassembly.

Measurements in Static Sodium

The tests were carried out in a rig built at UKAEA Risley Laboratories to test acoustic transducers at temperatures in the range 550°C to 600°C. The test rig, see fig 1, consisted of the transducer mounted in a clamp which also held a reflector at a fixed distance, about 100mm, in front of the transducer. The surface of the reflector is concave towards the transducer with a radius of curvature equal to the distance from the transducer to the reflector. This makes the alignment of the transducer relative to the reflector less critical. The whole assembly was immersed in sodium. In the experiment the temperature of the sodium was varied between 200°C and 600°C.
by measuring the transit time for the ultrasonic pulse to make the double passage over the subassembly top it is possible to determine the velocity of sound and hence the temperature. Compensation must be made for any expansion of the supports of the reflector over the temperature range. A calibration measurement at 400°C was used to determine the effective path length.

Results from two experiments are shown in figures 2 and 3. Figure 2 shows the temperature determined by the ultrasonic method plotted against thermocouple reading. Figure 3 shows the error over the range. The maximum error was 5 degrees and the standard deviation 2.8 degrees.

Measurements in Turbulent Water

In the reactor the hot sodium from the subassemblies mixes with the cooler sodium coming from the gaps between the subassemblies. The sound waves of the ultrasonic measuring system have to traverse this region of turbulent temperature mixing twice in each measurement and will be liable to suffer refraction if they cross a region with a temperature gradient except when the isotherms are normal to the sound path. In order to study this effect a rig was constructed to simulate the turbulence and temperature conditions above the core using water as the fluid.

The velocity of sound in water increases with temperature over the range 10°C to 30°C whilst, as was stated, the velocity of sound in sodium decreases as temperature increases. It is therefore possible to simulate the refractive effect of a hot jet in sodium by a cold jet in water. Furthermore, since the magnitude of the gradient of the velocity-temperature curve in water, expressed as a percentage of the velocity, is ten times greater than that in sodium, a jet of water 1°C cooler than the surroundings will have the same refractive effect as a jet of sodium 10°C hotter. These facts mean that a useful experiment can be carried out using water at near ambient temperatures.

Figure 4 shows a schematic diagram of the rig, known as STRUM1, on which the experiments to investigate the effect of turbulent mixing were carried out. The rig consists of a tank representing the part of the upper plenum immediately above the subassembly outlets and eight jets fed from a header tank which represent a row of subassemblies. There is a heater and stirring device in the main tank so that it possible to produce a uniform temperature in it which can be up to 10°C above the temperature of the water in the header tank which supplies the jets. The header tank has sufficient capacity to maintain the flow of all eight jets for about 20 seconds which is ample time to record the data since up to about 400 measurements can be made in one second.

Figure 5 shows the distribution of the amplitudes of pulses received in the normal direction through eight jets and shows that though there is a reduction in amplitude for most pulses some 15% are received at the normal amplitude or greater. This illustrates the intermittent nature of the turbulence which allows some pulses to pass through gaps unattenuated. This is an important result since the speed of the measurement is such that adequate sensitivity can be achieved even if only a fraction of the transmitted pulses are received unattenuated. In addition many of the attenuated pulses should still be usable with good data processing.

The next experiment carried out on this rig was a demonstration of the ability of the method to follow a temperature transient. Ideally this would be done by varying the temperature of a jet at constant flow but this was not possible in this rig. The change of temperature was achieved by turning on the jets of water.
ata a temperature different from that of the main tank. The effect of the flow transient was shown to be insignificant by an experiment in which the jets and the pool were at the same temperature. Fig 6 shows a comparison between the temperature measured by ultrasonics and by a thermocouple placed in the jet. The graph shows the variation of temperature indicated by these instruments with time over a period of 5 seconds. After about 2 seconds the jets are opened producing the temperature change. The ultrasonic system was operating at 200 measurements per second and showed good agreement with the thermocouple. In the case shown the temperature difference was 6.1°C.

Tests in Flowing Sodium

A special experimental assembly, see fig 7, has been designed to enable ultrasonic temperature measurements to be made in the Jet-in-Pool test section of the PNL 4 sodium loop. In this loop the flow to the test section is split into two streams, the central jet flow being heated to 100°C above the main flow. The two flows recombine upstream of a sudden increase in pipe cross section giving the capability of producing a wide range of flow and temperature fields in the test section.

The high temperature ultrasonic transducer is used to transmit a beam of ultrasonic pulses down to a 45° deflector with a conical reflector on top. Part of the ultrasonic beam is reflected back to the transducer by the conical reflector while the remainder is deflected across the test section to another conical target from which it is reflected back to the 45° deflector and thence back to the transducer. The space across the test section represents the subassembly outlet region. A rake of 0.5mm thermocouples capable of being rotated and moved axially through a sliding seal is incorporated to provide comparative temperature measurements.

A typical ultrasonic trace is shown in fig 8. The time scale is 50 microseconds per major division. The three pulses from left to right are the main drive pulse generated in the transducer, the reflection from the first target after a delay time, and the reflection from the second target across the flow after a further delay time. The delay time, which corresponds to the transit of the pulse across the simulated subassembly exit is used to determine the temperature.

Mean temperature measurements have been made over the widest range of flow and temperature possible within the loop constraints ie 240 - 420°C and 2 - 9 L/sec. As shown in fig 9, good agreement has been observed between the ultrasonic measurements and the thermocouple readings.

In addition a range of temperature noise conditions has been generated by increasing the jet/annulus differential temperatures from 0 to 36°C. The resulting fluctuations were detected by the ultrasonic technique.

Conclusion

The test programme under way has already shown that the ultrasonic technique for measuring temperature is capable of good accuracy. The early results encourage the belief that a system to measure breeder outlet temperatures in a LMFBR should be feasible.

Reference

1. Concave reflector and transducer housing assembly.

2. Sodium temperature by ultrasonic measurement.
3. Ultrasonic measurement error as a function of temperature.

4. Schematic diagram of STRUM I rig.
Amplitude distribution of pulses received through central position.

Comparison between ultrasonic and thermocouple measurement.

\[ \Delta T = 6.1^\circ C \]
FIG. 7. Mechanical Details of Jet-in-Pool Installation and Schematic Diagram of Ultrasonic Signal Path.
FIG. 9. Typical Ultrasonic Trace with Pulse Time Differences
FIG 9. COMPARISON OF ULTRASONIC AND THERMOCOUPLE MEASUREMENTS IN FLOWING SODIUM.
The Development of Acoustic Tomography for Temperature Measurement in Fast Reactors

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12th to 15th December 1989
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6. CONCLUSIONS

7. REFERENCE
1 **INTRODUCTION**

Tomography is the name applied to the technique whereby some properties of a slice of a medium are deduced from observations of the transmission of signals through that area. The technique can be used with electromagnetic radiation eg. X-rays as well as acoustic waves. Acoustic tomography has been used to deduce temperature and flow patterns in the ocean.

A research programme is underway at NRL Risley to investigate the use of tomography for the remote measurement of the properties of a medium. The technique can be used to measure density, temperature, gas content or any other property which alters the acoustic transmission characteristics. This paper gives results which illustrate the application of the technique to the measurement of outlet temperature in an LMF BR.

2 **PRINCIPLE**

The region to be studied is divided into cells, the number of cells being determined by the spatial resolution required. In the case of a reactor the cells would correspond to subassemblies. A pulse of sound traversing the region of interest will pass through a number of cells and its velocity and the attenuation it suffers will be a combination of the velocities and attenuations in the cells through which it has passed.

The method therefore is to transmit a sufficient number of pulses across the core in different directions so that every cell is traversed and sufficient data is obtained to allow a consistent set of simultaneous equations to be assembled from which the velocity in each cell can be determined. Since the velocity of sound in most fluids is a function of temperature this velocity measurement can be used to deduce temperature.

3. **DESCRIPTION OF SYSTEM FOR TOMOGRAPHICAL SCANNING**

Initial experiments were performed on a small easily accessible tank to determine feasibility of various methods to perform a tomographic scan in water, and to indicate the degree of repeatability of transit time and transmitted rms measurements. The results of these investigations were brought together in the system outlined below.

3.1 **Transducer Array**

An array of transducers is positioned around an area of interest. An array of eight transducers (T1-T8) on a square, side length 375mm, is considered here, as shown in Figure 1. Each transducer in turn emits a burst of sound which is detected by the other 7. As each of 8 transducers can transmit to 7 other transducers, there are 56 paths - assumed to be straight lines, along which sound transmission can be measured. However the number of distinct paths is half this, 28, as there are two directions of travel between any given pair of transducers. In the experiment the transmitted and received signals are recorded in digital form, and the transit time along each path is measured. 28 values in all, and 28 values of received rms are calculated.
3.2 Theoretical

3.2.1 Requirements of Path-Cell ratio

For the solution of a set of simultaneous equations, there must be at least as many equations as there are unknown variables. In tomography there must be at least as many distinct paths between pairs of transducers, as there are cells.

For a system using N transducers there are \( \frac{N}{2} \times (N-1) \) paths and it may have up to that number of cells. As there are 2B paths between B transducers the area inside the array may be split into up to 2B notional "cells". Each cell, assumed homogenous, having definite, but unknown values for attenuation and velocity of sound. The chosen cell positions are shown in Fig 1, which uses 25 cells giving a small redundancy in the number of equations available.

3.2.2 Relating cell temperature to cell velocity

The relationship between acoustic velocity and temperature depends on the transmission medium. In water, velocity and temperature are related by the approximate formula:

\[
\text{v} = 1403 + 4.2 \times t - 0.028 \times t^2
\]

(\( v \) is velocity in m/s and \( t \) is temperature in °C). Fig 2 shows the plot of this function over the range 10°C to 40°C. In this range the function has no peak or trough, hence this is a feasible operating range for!a tomographic scanning system.

3.2.3 Relating cell velocities to transit time

The measured time delays are determined by the sound velocities in each of 25 cells. The time taken for the sound to travel between T1 and T2, for example, is the sum of the distance it travels through a given cell, divided by the speed of sound in that cell, summed for each cell it passes through. This leads to 28 equations linking 28 time delays to 25 velocities.

4 Experimental Test

4.1 Area of Scanning

An experiment to test the possible application of Tomography to remote spatial monitoring in a realistic situation has been set up as in Figure 3. Eight transducers (T1-T8) were placed around an area of interest in a water tank. The transducers were arranged in a horizontal square of approximately 0.4m side, 0.3m above a 1/4 scale model of the PFR reactor core. A heat source (a pipe from a large reservoir of hot water) was placed in one subassembly. The source position was below the area to be scanned.

The available manoeuverability of the transducers was limited by the design of the rig, and the position of the array relative to the subassemblies could only be measured to within 2 cm. However each source had one cell covering most of its area, with a single adjacent cell covering any significant remainder as shown in figure 3.
4.2 Scanning Procedure

Using the computer controlled hardware, a burst of sound was transmitted from transducer 1. The signal received on all transducers was recorded in digital form and stored. Then transducer 2 was pulsed, this procedure being repeated for each transducer in turn until signals transmitted along each of the 28 paths had been stored.

The signals were processed to extract transit times or transmission signal strength along each path. These values being used in the solution routine to calculate temperature or attenuation in each cell.

5 ILLUSTRATION OF RESULTS

5.1 Temperature Detection

To illustrate the use of acoustic Tomography in a geometry similar to the core outlet region in a Fast Reactor we used the half scale model of the PFR core and the experimental arrangement described in the previous section.

The heat source, which is marked on fig 3, was positioned inside the hexagonal subassembly chiefly covered by cell 23. Thermocouples were in place at the exit of this subassembly, and also 0.32m above it, which is 2cm higher than the level scanned by the array. This additional offset was to avoid affecting transmission.

When the heat source was turned on the lower thermocouple showed temperatures of up to 40°C, while the temperature at the thermocouple at the level of the array was 23°C, this was a rise of 7°C above the background temperature. Fluctuations in temperature caused by mixing of hot and cold water were apparent on both thermocouples.

Results

Steady State Measurements

Figure 4 shows a plot of temperature in the cells of the tomographic array in which shading is used to indicate the temperature measured by the tomographic technique. The results were obtained with the source turned on. The position of the source shows up clearly, giving a reading of over 20°C in the appropriate cell against a general background of 16°C. Two adjacent cells show a temperature rise of about 2°C while in all the remaining cells the change is less than 1°C.

Work on ultrasonic temperature measurement, ref 1, has shown that the sensitivity in sodium will be about a factor of 10 worse than in water. These tests suggest a sensitivity of 1 to 2°C in water and therefore 10 to 20°C in sodium.

Transient Measurements

The next experiment was to demonstrate the ability of the technique to follow transient changes in temperature. To do this 128 measurements were made for each cell at a rate of one every 4.25s, the total time covered was therefore 9 minutes.
In Fig 5 the calculated temperature for each cell is plotted against time during the 9 minute duration of the experiment. Initially the heat source was off. After approximately 3 minutes it was switched on causing a gradual rise in temperature mainly in cell 23 (cell numbers defined in Fig 3), and to a lesser extent in cell 22. There is a rise in 'noise' in all other cells. After a further few minutes the heat source was turned off causing an initial sharp drop in measured temperature in cell 23, followed by a steady decline to background levels.

In Fig 6 the variation in cell 23 is shown in detail along with the reading of the thermocouple above that cell over the same period. The thermocouple plot consists of 127 values, each value is the average of 7 readings taken during the acquisition of the transmission data. There is good agreement between the behaviour of the temperatures indicated by the two instruments.

Figure 7 shows a plot from a similar experiment in which the heat source was turned on for two periods of roughly 2 minutes each, with 2 minutes off between them. Again good quantitative agreement was observed.

Figure 8 shows a plot of cell 22, a cell covering part of the heat source and showing the next largest temperature rise. A thermocouple was positioned in the area of this cell for comparison. The thermocouple shows that the apparent temperature rise is genuine, and is in reasonable agreement with the tomographic results.

6. CONCLUSIONS

1. The experiments have shown that tomographic analysis of acoustic signals passing between transducers in an array can be used to measure temperature distribution within a plane area.

2. The results taken with other work on ultrasonic temperature measurement [1] indicate that the application of acoustic tomography in an LMFBR is feasible. Further tests on reactor scale and in reactor conditions are however required to confirm this.

7. REFERENCE

1. I D Macleod et al. An Ultrasonic Technique for the Remote Measurement of Breeder Subassembly Outlet Temperature. This Conference.
Fig 1 Positions of transducers and cell divisions for system
Fig 2 Temperature dependence of acoustic velocity in water.
Fig 3 Positions of transducers and cell divisions in water tank
Fig 4  Showing measured temperature for each cell with active heat source in cell 23.
Fig 5  Showing plots of temperature against time for each of the 25 cells in their relative positions.
Fig 6 Comparison of tomographical and thermocouple temperature due to single thermal transient, in cell 23.
Fig 7: Comparison of tonographical and thermocouple temperature in cell 23 due to switching heat source.
Fig 8 Comparison of tomographical and thermocouple temperature in cell 22 due to single thermal transient.
ABSTRACT

Some experimental data, obtained during temperature noise (TN) investigations at reactors BOR-60 and BN-600 and the results of sub-assembly (SA) outlet noise numerical modeling are given. Based on this analysis the requirements to dynamic characteristics of the sensors are formulated and some recommendations for increasing the reliability of anomaly detection are given.

INTRODUCTION

The thermohydraulic core parameters deviation from normal values can lead to significant temperature rise, coolant boiling and fuel pin clad destruction. The choice of physical methods and technical tools for anomaly detection is determined by reactor design and real operation conditions. The pin clad failure could be revealed by means of delayed neutrons and fission products in the coolant. The boiling could be detected by monitoring the signals of neutron, acoustic and temperature sensors. The integral methods, using a limited number of neutron and acoustic detectors, enable the reliable detection of the anomaly, which leads to coolant boiling across the whole SA section and throwing the steam into the subcooled sodium medium above the core /1/. This method is less reliable in detecting the local boiling and fails to detect the local blockage without boiling.
The individual temperature monitoring at the subassembly outlet seems to be promising. It should be mentioned, however, that the local blockage even resulting in local boiling, induces only negligible SA flowrate and outlet temperature deviation. Thus, the 10\% corner blockage, which causes a boiling, results in the flowrate deviation of some tenth of a percent, the mean outlet temperature deviation being 0.5±1K/2/. Independent of the fact whether the mean temperature or its fluctuating component is used as an anomaly pattern, the reliability of the anomaly detection depends on the relation between accidental parameter deviations and the background noise under normal operation. The background TN was investigated at the BOR-60 and BN-600 reactors. The signals of the temperature sensors located above the core and in the special assemblies were analysed. Based on these data, the fluctuation sources have been determined and the possibilities of improving detection systems have been estimated.

SOME EXPERIMENTAL DATA

The thermocouples (TC), installed opposite the central outlet hole (\(\phi = 6\text{mm}\)) at the 20 mm from the SA head were the main subject of investigation at the BOR-60/3/. The TC time constant initially evaluated is about 1±1.5 sec. The cross-correlation of TN signals with flowrate and neutron flux fluctuations shows that within the frequency band investigated more than a half of the temperature noise power is caused by the integral parameters mentioned. This fact agrees with a high correlation degree between the signals at the outlets of different SAs (\(\rho = 0.6\sim0.8\)). The values of TN rms are within the range of 0.27±0.3 K, the temperature difference between the core inlet and outlet being 120 K. It should be stressed that data mentioned correspond to the initial period of reactor operation. After some years of operation the remeasurements revealed a significant (2±4 ti-
rise of fluctuation amplitude in comparison with the initial value for the part of the TCs. This phenomenon was accompanied by a decrease of the cross-correlation degree between abnormal TCs and integral parameters signals. It could be accounted for the fact that due to numerous reloadings and possible mechanical damages the correspondence between the outlet hole axis and the TC position is broken. The flows from neighbouring SAs with different temperatures reach the monitoring TC and give rise to "mixing noise" of a large amplitude. Naturally, such noise is uncorrelated with flowrate and neutron power fluctuations. A similar picture was observed at the French SPX-1 reactor. The signals of TCs, located at the level of 100 mm above the core for SA monitoring, are not correlated with the integral parameters mentioned.

Some additional data on TN were obtained at the BOR-60 using a special instrumentated SA, equipped with a great amount of TCs with the time constant being ~ 0.1 sec. The TCs are placed in SA cross-sections at different heights (fig. 1). Within the frequency range investigated the signals of TCs at different positions are strongly correlated. Thus, the correlation degree between the signals in the section near the SA middle and in the SA head reaches the value \( \rho = 0.85 \) (fig. 1b). This indicates the presence of a common fluctuation source, most probably, a pulsation of the flowrate through the SA. This source masks the TN generated due to the blockage.

Thermonoises at the BN-600 were investigated with temperature sensors installed in a special probe. The hot junctions are located in the lower probe part, which is inserted into the SA head (fig. 2). The frequency band investigated is expanded by the use of intrinsic (sodium-steel) thermocouples in addition to usual Cr-Al sensors [4]. The advantage of the intrinsic TCs when monitoring dynamic processes is evident from comparison of the readings during the BN-600 reac-
tor shut down (fig. 3). In the initial state the reactor power was 20% and the flowrate was 30% of the nominal operation value. The intrinsic TCs prove to give a leading signal when recording the temperature fall. At the same time, due to some measurement method peculiarities, analysed in [5], the low frequency components are partly damped in the "sodium-steel" system.

At some reactor operating levels the TNs were recorded with fast temperature sensors. The corresponding rms value of fluctuations are given in the table; the correlation function for the nominal reactor power level is shown in fig. 4 as compared with the correlation function obtained by numerical modeling.

Table

<table>
<thead>
<tr>
<th>N, °C</th>
<th>80</th>
<th>95</th>
<th>100</th>
</tr>
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<tr>
<td>G, °C</td>
<td>88</td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>ς, K</td>
<td>0,26</td>
<td>0,34</td>
<td>0,36</td>
</tr>
</tbody>
</table>

SOME RESULTS OF CALCULATION

The numerical modeling provided a preliminary estimation of the outlet thermonoise statistical characteristics. The method and code based on the Monte Carlo approach and generally following Firth's work [6] were developed [7], modeling the random walk of a certain particle (the temperature mark in our case) in the turbulent flow. Using the "backward diffusion" principle, a backward path of the particle from the sensor to the section with the known temperature distribution (usually the upper boundary of fuel pins) is traced with a certain time step. The particle coordinates at any time step are determined taking into account three velocity components. The mean flow velocity is considered as a longitudinal speed component; the
particle drift in the transverse direction is characterized by a constant cross-stream velocity $v$ and a constant cross-stream correlation length $l$. These parameters define the space-temporal flow correlation. The stochastic angle deviation during the cross-stream motion is produced with the random number generator. The current temperature fluctuation is associated with the position, where the present particle crosses the initial section with known temperature distribution. The stochastic sequence obtained presents a temporal random realization, which can be processed by various statistical methods. The number of particles to be traced was up to 20 thousands. The calculation of mean values, rms, moments of the 3rd and 4th order, normalized distribution parameters (skewness, kurtosis), correlation functions and power spectral densities is foreseen.

Computations were made for the BN-800 subassembly model. The inner SA hexagonal section with the flat-to-flat dimension of 92 mm was replaced with a circle of the same cross-section. The coolant velocity varied within the values of $0.5\pm 2$ m/s ($Re=2.5\times 10^5$). The TN for different outlet temperature stationary profiles were investigated. The example of statistical characteristic at the axisimmetrical profile for the temperature drop between the center and periphery of a section being 15 K is given in fig. 4. The calculations show that skewness and kurtosis are sensible to temperature profile variations. For the nominal velocity value ($u=2$ m/s) the correlation time $\tau_K = \int_0^\infty \rho(\tau) d\tau$ is less than 0.1, the requirements to dynamic properties of the sensor following from this fact. Here rms is equal to 0.053 K. The difference between normalized experimental and calculated correlation functions can indicate the presence of some additional noise sources under real reactor conditions.
CONCLUSION

1. In the experiments at fast sodium reactors the background temperature noise under normal operation conditions has been investigated. Some noise sources are defined. Comparison with experiments at test facilities with blockages of different kind shows /8/ that the anomaly high (one order higher than the background) noise is observed only immediately behind the blockage. While going downstream the fluctuations attenuate and become comparable with the background.

2. The reliability of thermohydraulic anomaly detection can be increased in various ways, including:
- design measures, ensuring individual SA monitoring and preventing the influence of the neighbouring SA flows
- compensation of the thermoneise caused by integral parameter pulsations (similar to the reactivity balance method /9/)
- the noise components selection by a frequency filter
- analysis of high order moments

REFERENCE


meeting on "Instrumentation and diagnostic systems for fast reactors", Dresden, GDR, ZfK-519, pp.302-320


8. Proceedings of IAEA-IWGFR specialists' meeting on "Methods and tools to detect thermal noise in fast reactors", Bologna, Italy, 8-10 October 1984

Fig. 1  Sensors position in the instrumented SA (a) and correlation between the signals of $TC_{17} (\dot{c} = 0.39 \text{ K})$ and $TC_{35} (\dot{c} = 0.46 \text{ K})$ (b). $N_t = 49 \text{ MWt}, G = 1010 \text{ m}^3/\text{h}$. 
Fig. 2 Sketch of the thermoprobe
1 - cables with electrodes
2 - sodium gap
3 - metal body of reactor plug
Fig. 3  Transient sensors readings.

N - neutron flux, rel. un.
G - sodium flowrate, rel. un.
\( T_{\text{in}} \), TM - the temperature drop,
picked up by sodium-steel and Cr-Al thermoelectrical sensors
Fig. 4 SA sketch (a) and normalized correlation functions of temperature noise (b)
1 - stationary temperature profile; 2 - temperature sensor (z=200mm);
3 - calculated CF, $\xi = 0.053K$; 3 - experimental CF, $\xi = 0.36K$. 
COMPUTER BASED SYSTEMS FOR FAST REACTOR CORE TEMPERATURE MONITORING AND PROTECTION

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1 INTRODUCTION

The need for an effective system for monitoring the state of core cooling in fast reactor is regarded in the UK as being essential part of the safety system as well as providing important information to station operators. The UK has consequently spent considerable effort in developing temperature monitoring systems and incorporating them into highly reliable fail safe trip systems. One of the approaches that has been adopted is to make a direct measurement of the subassembly outlet temperature using specially designed coaxial thermocouples. The measured temperatures are then processed by a computer based trip system whose output goes to the guardlines for the primary shutdown system. The need for both the computer based trip system and the guardline to be fail safe, for safety, and highly reliable, for economics, has led to the use of dynamic logic to generate fail safety and redundancy to produce the required level of reliability. The resulting systems are both highly effective and have been proven in prototype form where they have successfully achieved the high data throughput and rapid response times which is particularly important for fast reactors where the time scales for many postulated accidented sequences can be very short, seconds rather than the tens of seconds of other reactor systems.

2 THE PULSE CODED LOGIC GUARDLINE

The traditional form of guardline structure used in the UK for reactor protection has been either relay based or used ladder. An alternative system that has been developed at Winfrith uses hardwire dynamic logic. The use of dynamic logic provides a continuous form of testing which is able to enhance availability as well as produce fail safety. The system can be configured many ways it has a set of M parallel guardlines, where M ≤ 4 usually, and any number of parameters can be present on the guardlines. An oscillator at the start of each guard line generates SET and RESET signals which are propagated down the guard line via the guard line logic for each parameter. Oscillators are also used to generate a coded signal, a different one from each oscillator, these signals are passed to the instrument voters via the instrument trip contacts marked A, B and C in Figure 1. If all the instrument inputs are healthy, or satisfy the voting, then the signal from the voter to the guard line logic lets the SET RESET signals pass down the guard to the guard line logic of the next parameter and eventually to a pulse to dc converter which powers the shutdown actuators via the guard line voting and if required a power amplifier. The pattern of the coded signals passed to the instrument voter via the instrument trip contacts are arranged such that the voter outputs a pulse signal provided the system is in a safe state. The mode of operation is best illustrated by considering an actual system, for example a two out of three system that could be employed on the system illustrated in Figure 1. The dynamic signals shown in Figure 2 are input to the parameter voting logic via the instrument trip contacts when all instruments are healthy this results in the uniform pulse pattern at the voter. Provided the pulse pattern is present the SET RESET
signals are propagated down the guard line. Should one instrument fail due to a fault on the trip state input it becomes stuck at 0. The pattern at the voter for line A stuck at 0 is shown in Figure 2a, it retains its pulse like shape so the SET RESET signals still propagate and no trip is initiated but it has extra detail which uniquely defines the fault. It is usually arranged to indicate this failure at the parameter voter. Should a second instrument go to 0 then the patterns of Figure 2b result, the voter outputs a static state and the SET RESET signals are no longer propagated down the guardline, a trip is initiated depending on the state of the other guard lines and guard line voting in use. If the instrument becomes faulty such it becomes stuck at 1 then then situation of Figure 1c results, the voter output is dynamic so no trip is initiated but the fault will be indicated and the pattern allows the fault to be identified.

Consideration of all other cases has been made, this includes the case of the failure of the pulse and coded signal generators. This situation immediately sets a guardline to the trip state and results in a stuck at 0 state on the instruments connected to that pulse generator, the system will trip on any further failure. The exhaustive investigations have shown that the system is fully fail safe and also allows faults to be quickly identified for repair or in exceptional circumstances vetoed the latter does however degrade the ability of the system to tolerate faults.

3 THE SELF TESTING TRIP SYSTEM

The trip lines that provide the input to the guardlines can also be arranged to operate in a similar manner, ie using dynamic logic to provide fail safety. However the systems incorporate computers to perform trip function analysis rather than the simple hardwired logic of the guardline. The simplest form of the system will be described as an illustration.

The plant signals are input to the multiplexer but interleaved among them in a carefully defined pattern are test signals Figure 3. These test signals are such the trip algorithm should output a fail, or trip, state if the trip computer is operating correctly. The multiplexer output passes through a polarity reverser, which switches the signal polarity on alternate sweeps of the multiplexer, through the analogue to digital converter to the trip algorithm computer where the trip calculations are performed. For example a system with four parallel trip lines might use 3 out of 4 voting. The resulting voter signals are input to a pattern recognition unit, which knows the pattern of trip and healthy states it should receive. If the pattern matches correctly the unit produces a pulse output which goes through a pulse to direct current converter and onto the guardline. If the patterns do not match, this could arise because of the plant moving to an unsafe or trip state or by failure (multiple) of the hardware then the pattern recognition unit outputs a static signal and power to the guardline contacts is lost.

The following points are noted, polarity reversal following the multiplexer can be included to check that the multiplexer is being
continually refreshed. The pattern of plant and test signals is carefully arranged to be unique for each set of eight inputs to the multiplexer so the multiplexer has to be fully refreshed, see Figure 4. The figure eight here arises from the use of an 8 bit pattern recognition sequence. The use of the computer to perform the trip calculation allows any plant signal and very complex trip algorithms to be employed. The vote algorithm computer is used rather than a hardwired voter such the signals from the parallel trip channels do not have to be synchronised. This marks a significant difference from the guardline structure where the codes are synchronised so hardwired voters can be used.

4 TESTING AND APPLICATION OF THE SYSTEMS

Both the pulse coded logic guardline and the self testing trip system have been tested and proved in operation under plant conditions. The pulse coded logic guardline was first tested in the laboratory then as a two parameter two out of three passive system on Bradwell (1976-1982) and a fourteen parameter two out of three passive system on Oldbury (1978-1982). The Oldbury system was then transferred to the DIDO reactor and following a three year trial operation was reworked and adopted for active service on the reactor. The system was found in both active and passive service to be extremely reliable fulfilling the reliability criteria set by the reactor operators. The systems have responded to all demand for trip action that have been made upon it and no fail danger states have been identified by analysis or have arisen in operation.

The computer based trip system has not received such extensive testing. The approach was first demonstrated in the laboratory and then a quasi commercial system was constructed and installed for testing in sodium loop at the Berkeley Nuclear Laboratories. This system contained four simple trip functions acting on subassembly outlet temperature, these were:

- absolute temperature too high;
- absolute temperature too low;
- rate of change of temperature too high;
- temperature difference from median of temperatures of six adjacent sub-assemblies too great.

The successful integration of the co-axial thermocouples, developed especially for fast reactor outlet temperature monitoring, and the trip system enabled it to be transferred to Dounreay and operated in passive mode on the prototype fast reactor. In operation only a small number of thermocouples have been used, dummy signals being supplied to the bulk of the inputs for convenience. The initial period of operation proved to be very disappointing as there were a large number of failures due to the very hostile environment, temperature and vibration, in which the system was placed. Following repair and relocation the system in an environment more appropriate for safety electronics has

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achieved the prescribed reliability. Further during operation the system has functioned correctly on demand on every occasion and no fail danger states have arisen. Analysis of the system has similarly failed to identify any fail to danger states.

5 SYSTEM DEVELOPMENT

The major thrust of development of the pulse coded logic guardline system is to meet demands of flexibility and cost that will enable the system to be made available for use on existing and/or new plants in a cost effective manner. To this end design work is underway to produce a standard set of modules that can be put together to produce a three or four guardline structure operating on two out of three, two out of four or three out of four voting and containing any number of parameters on the guardlines. The modules are to be made up of commercially available components that can be pre-qualified and which can be assembled quickly and economically to produce any desired guardline configuration.

The more general nature of the trip system which is based upon the use of programmable computers gives rise to a much greater potential for development but does give rise to some additional work associated with the software executing the trip function. The extension of the architecture of the trip system to replace the interleaving of the test and the plant signals by introducing the test signal onto the plant input lines by using summation amplifiers as shown schematically in Figure 5. This system requires the addition of a further computer to calculate the perturbation to the plant state, this additional computer must in effect invert the forward trip calculation. This inversion procedure has been achieved for simple trip algorithms but those algorithms that contain strong temporal aspects for example the effect of damping and time delays are much more difficult to invert. Care must also be taken, especially with the trip algorithms that contain temporal features, to ensure the disturbance of the system introduced by the test signal does not produce a state that effects subsequent cycles of the trip calculation.

The extension of the trip function beyond that provided by the simple temperature functionality used on the PFR has also been investigated through the vehicle of producing channel power and linear rating trips for a PWR system. This exercise continues and work on inverting the temporal part of the algorithm is progressing quite well.

Extensive research is being undertaken into the means of producing the software for the trip computers for both simple and complex trip algorithms. The correctness or at least fail safety of this software is crucial to the successful production and licensing of the systems for use. One of the major areas of research is into the use of formal methods for the specification stage following requirements capture, the development or reification of the specification to the coding level. Formal tools are also being investigated for code production. One of the main exercises that
should yield information about all aspects of this process is the CEC ESPRIT II DARTS project.

6 TEMPERATURE MONITORING

The temperature monitoring function of the self testing temperature trip system using thermocouples as the instrument has been described above. The UK have, however, been investigating other means of sensing the temperature in the core. The first system makes use of acoustic boiling noise to monitor temperature remotely within the core. The system as such can only indicate that the temperature is above or below the saturation temperature, however this is sufficient for reactor monitoring and protection purposes. The second system, which has been developed more recently, is an ultrasonic temperature measuring system that can be used to make a line of sight temperature measurement across the subassembly outlets at the top of the core. While both systems are not mature the results of experimentation, that has been undertaken in test loops and under test conditions in core, give a reasonable certainty that the approaches will function to detect anomalous signals. Aside from the problem of signal transmission, which also gives concern in the thermocouple based systems, there is some concern about the extensive nature of the signal processing electronics that are used in these systems. This concern takes two forms first is how to identify any degradation or failure of the signal processing electronics, second and particularly for the acoustic boiling noise system there is some concern as to whether the signal processing system will recognise/detect boiling. The latter point is not addressed here but the concern over system availability, particularly with the extensive use of signal processing, can be resolved for both acoustic and ultrasonics by incorporating them into a self testing system. Thus rather than the instrument feeding the signal processing electronics and then the trip electronics the instrument and test signals are multiplexed before signal processing and evaluation by the trip system, see Figure 6. This architecture thus provides continuous testing of the signal processing electronics b.t the system does not test the instrument and a fail to danger of the instrument is possible. This is also true for the thermocouple based system but thermocouple failure is more likely to lead to a fail safe rather than a fail danger state.

Test signal generation for both the acoustic and ultrasonic systems must either rely upon: use of pre-recorded signals that indicate boiling or a temperature above the trip temperature, or upon there being a test chamber in which sensors of the type used on the plant can be subject to conditions that result in a trip state. For example an acoustic microphone could be mounted on a boiling chamber. Such an approach has the grave disadvantage compared with the use of recorded signals that the acoustic, or equivalent ultrasonic, signal may not have the same characteristic as signals from the plant.

The ultrasonic system does, however, present other possibilities as the system works by making a sound speed measurement to obtain
the temperature by measuring the time of an acoustic pulse over a known distance. The trip signal for testing purposes could be produced by changing the effective distance in the sound speed calculation. This approach has the great advantage over using a pre-recorded signal in that the plant signal must be present thus this approach also provides a test of the ultrasonic transmission and reception equipment.

Work is continuing to investigate the use of self testing for these alternative systems and to produce a practical system for operation on the fast reactor. The system may well employ both forms of self testing indicated in the discussion above to produce a satisfactory system.

7 CONCLUSIONS

Self testing fail safe trip systems and guardlines have been developed using dynamic logic as a basis for temperature monitoring and temperature protection in the UK. The guardline and trip system have been tested in passive operation on a number of reactors and a pulse coded logic guardline is currently in use on the DIDO test reactor.

Acoustic boiling noise and ultrasonic systems have been developed in the UK as diverse alternatives to using thermocouples for temperature monitoring and measurement. These systems have the advantage that they make remote monitoring possible but they rely on complex signal processing to achieve their output. The means of incorporating such systems within the self testing trip system architecture are explored and it is apparent that such systems, particularly that based on ultrasonics has great potential for development. There remain a number of problems requiring detailed investigation in particular the verification of the signal processing electronics and trip software. It considered that these problems while difficult are far from insurmountable and this work should result in the production of protection and monitoring systems suitable for deployment on the fast reactor.
FIGURE 1. SCHEMATIC DIAGRAM OF PULSE CODED LOGIC GUARDLINE
FIGURE 2. INSTRUMENT AND SET RESET CODES

V₂/₃

A₆ + B₆ + A₇

FIGURE 2A. INSTRUMENT CODE, LINE A STUCK AT 0

V₂/₃

A₆ + B₆ + A₇

FIGURE 2B. INSTRUMENT CODE, LINE A AND LINE B STUCK AT 0

V₂/₃

A₆ + B₆ + A₇

FIGURE 2C. INSTRUMENT CODE, LINE A STUCK AT 1
FIGURE 3. ISAT SYSTEM FOR SUBASSEMBLY PROTECTION
### FIGURE 4. ISAT MULTIPLEXER INPUT SIGNALS AND PATTERN RECOGNITION LOGIC: STATUS PATTERN

<table>
<thead>
<tr>
<th>Multiplexer Input Signals</th>
<th>Trip Status Pattern</th>
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</thead>
<tbody>
<tr>
<td>01</td>
<td>1101101</td>
</tr>
<tr>
<td>02</td>
<td>1110110</td>
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<td>03</td>
<td>1111010</td>
</tr>
<tr>
<td>04</td>
<td>1111100</td>
</tr>
<tr>
<td>05</td>
<td>1111110</td>
</tr>
</tbody>
</table>

**Key:**
- • Thermocouples
- 0 Non-tripped State
- 1 Tripped State

**Test Signals:**
- H High
- R Ramp
- C Cold Junction

**Scan**
- Channel 1
- 8 Byte 1
- 121
- 12B
- 16
FIGURE 5.  ISAT SYSTEM USING INVERSION FOR TEST SIGNAL GENERATION
FIGURE 6. ISAT SYSTEM FOR ACOUSTIC AND ULTRASONIC SYSTEMS
REPORT ON IAEA-CRP ON SIGNAL PROCESSING TECHNIQUES FOR SODIUM BOILING NOISE DETECTION

(IGCAR, DDIAE, KFK, BNL - CONTRIBUTION)**

Ly

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* Paper for presentation in IAEA-IWGFR Meeting on 'Instrumentation for Supervision of Core Cooling of FBRs' December 12-15 (1989), IGCAR, Kalpakkam, Tamilnadu, India

IGCAR : Indira Gandhi Centre for Atomic Research, India
DDIAE : Darling Downs Institute of Advanced Education, Australia
KFK : Kernforschurys zentrum, Karlsruhe, FRG
BNL : Berkeley Nuclear Laboratories, CEGB, UK.
REPORT ON IAEA-CRP ON SIGNAL PROCESSING TECHNIQUES FOR SODIUM BOILING NOISE DETECTION

(IGCAR, DDIAE, KFK, BNL - CONTRIBUTION)

I. INTRODUCTION

In Liquid Metal Fast Breeder Reactors (LMFBRs), reliable detection of sodium boiling has been recognised as an important requirement for additional safety protection. The monitoring of acoustic noise emitted during sodium boiling is considered a promising method. However, for reliable detection, it is necessary to distinguish the boiling noise from background noise. Different methods are being developed for this purpose in several countries. An IWGFR Specialists meeting recognised the necessity for a benchmark test to make a comparative assessment of the existing signal processing techniques and identify optimum methods for a reliable on-line sodium boiling noise detection system. IAEA accepted the recommendation of IWGFR and sponsored the project. Australia, Japan, India, FRG, GDR and UK participated in the project. The project continued from 1984 to 1987. The results of the investigations are documented in IAEA report no. IWGFR/68.

The data for this benchmark problem was provided by IAEA from out of pile experiments performed on KNS I loop at KFK - Karlsruhe and in-pile experiments performed on BOR-60 in Russia. The details of the data set are presented by Macleod along with the highlights of the work of RNL (UK), Japan and GDR. In this presentation, first the IGCAR work is presented. It is followed by brief presentation of BNL (UK), Australia and FRG work.

II. HIGHLIGHTS OF CRP WORK

II.1 IGCAR WORK

The KNS-I and BOR-60 data were analysed using known methods and by searching new methods of signal processing. The tasks for both sets of data were to detect the onset of boiling by different methods and evaluating the quality of detection in terms of probability of spurious boiling detection (Ps) and missing the boiling detection (Pm) by choosing appropriate boiling threshold for each method. It was also checked whether the signal processing method under test can be set to achieve the spurious trip rate less than \( 10^{-5} \) per year (\( < 0.3 \times 10^{-5} \text{s/yr} \)) and if such a rate can be achieved what is the minimum averaging time needed in evaluating the features sensitive to boiling, what would be the probability of missing a boiling event and what is the level of discrimination between the boiling and non-boiling phases in signals.
At IGCAR, we investigated the suitability of several methods for sodium boiling noise detection. To calculate $P_s$ and $P_m$, a large number of estimates of the feature under test are calculated for background and boiling noise data and from that the probability density function (PDF) of features is determined for the respective data. The fraction of the area of PDF curve above a given threshold for background and below the threshold for boiling noise data gives respectively the $P_s$ and $P_m$. The values of $P_s$ and $P_m$ depend on threshold level. An optimum value of boiling threshold is selected by finding out the threshold level for which the product $(1-P_s)(1-P_m)$ has a maximum value. During the analysis, it was found that in most of the cases, particularly with the data with low signal to noise (S/N) ratio, $P_s$ and $P_m$ are well above the target values. This necessitated the development of new approach to reduce the $P_s$ and $P_m$ to target values.

**A. New Method for Achieving Target Limits of $P_s$ and $P_m$**

The new method is based upon the criterion that decision is not based upon a single estimate of the feature but on $N_o$ estimates evaluated successively and checking whether no or more estimates indicate boiling. If $P_s$ is the probability of false alarm for a single estimate crossing the threshold, the probability of false alarm when no or more estimates indicate boiling is given by

$$P_s^n = \sum_{n=N_0}^{N_o} \frac{N_o!}{(N_o-n)!} \frac{P_s^n (1-P_s)^{N_o-n}}{n!}$$

Similarly if $P_m$ is the probability of missing the boiling signal when single estimate does not indicate boiling, the probability of missing boiling when ($n_o-1$) or less estimates of $N_o$ estimates do not indicate boiling, is given by

$$P_m^n = \sum_{n=N_o-1}^{N_o} \frac{N_o!}{(N_o-n)!} \frac{P_m^n (1-P_m)^{N_o-n}}{n!}$$

From the above approach, it is always possible to achieve target probability values by selecting appropriate $n_o/N_o$ criterion for any values of $P_s$ and $P_m$ and is thus very attractive.

Different processing methods have been evaluated with above approach. The results are given briefly below.

**B. Analysis of KNS-I Data**

As reported by Macleod(2), the KNS-I data comprised of 12 - files. Each file consisted of three regions, namely flow
noise, initial boiling and intense boiling noise and each file with data of different signal to noise (S/N) ratio. The first file data is with 0.0 dB S/N ratio and the twelfth file with -17.5 dB. File 2 to file 11 data is with S/N ratio decreasing successively by 2dB. The objective of the analysis is to find the onset of initial and intense boiling with required Ps and Pm using different methods.

B.1 Detection of Onset of Boiling

Conventional methods

1. Visual observation: As a first step, the data were scanned and examined visually on oscilloscope. Onset of initial boiling could be detected in the first 9 files. However, onset of intense boiling could not be observed in visual inspection.

ii. Pulse counting: The number of pulses above a set threshold in a given time was counted and an increase in the counts was considered to be indicative of boiling signal. A pulse counter was built for this purpose. Onset of initial and intense boiling were detected in all the files. To evaluate Ps and Pm, the analog data were digitised with 50 microsecond sampling time. PDF was calculated for Pulse counts feature. It was found that for file 1, there was no overlap of PDFs in background and boiling regions leading to clear cut threshold and Ps = Pm = 0. Onset of boiling could be correctly predicted. However, in the case of twelfth file, there was overlap of PDFs and no/No criterion of 11/20 satisfied the target probability of $P_s = 10^{-10}$ and $P_m = 10^{-3}$. Applying this, onset of boiling was predicted 9 s after the actual inception point.

iii. Estimation of RMS: Onset of initial and intense boiling could be predicted with RMS estimates also. PDF of RMS for background noise data was more sharply peaked than the PDF of boiling noise data. This resulted in Ps to be more sensitive to threshold level. Unlike pulse counting method, PDF of RMS for background and boiling noise data overlapped for file 1 data also. For file 1, and 12, no/No criterion of 8/20 and 21/40 provided the target values of Ps and Pm. The onset of boiling predicted for file 1 was off (delayed) by 4.9 S for file 1 and 6.3 S for file 12. RMS - analysis was performed for squared data also. However, it proved of no advantage.

iv. Power Spectral Density (PSD): PSD - estimates for background and boiling noise data indicated that area under PSD curve increases above 35 kHz for sodium boiling. This area was treated as a feature. For file 1, onset of boiling was detected with a delay of 1.7 S and no/No criterion required was 7/13.
New Methods

v. Segmented Areas of PDF: From in-depth analysis of basic data, it was found that the area under PDF curve if segmented judiciously in three regions $A_1$, $A_2$ and $A_3$, then the areas are sensitive to boiling. An increase in $A_1$ and $A_3$ and decrease in $A_2$ clearly indicated boiling in file 1 and file 12. For file 1, $\frac{A_1}{A_2} = 5/10$ and for file 12, $\frac{A_1}{A_2} = 22/56$ criterion, the onset of boiling was as mentioned above. The detection was off (delayed) by 2 s for file 1 and 3.6 s for file 12. Other results are given in Table 1.

vi. Multivariate Pattern Recognition: In this technique, multiple features of noise signals are evaluated and considered simultaneously for detecting the onset of boiling. In the case of single features, the PDF of background and boiling noise data lie on a line and threshold is a point on this line. But for multiple $N$ features, the PDF would lie in $N$-dimensional space and boiling threshold would be a surface in this space separating the boiling and non-boiling regions. If all multiple features considered are independent of each other, then the probability of getting the spurious trip or missing the boiling would be simply the product of such probabilities for individual features. However, the features considered may be correlated. Then the PDF of features for any one set of data, would be

$$p(\bar{A}) = \left(\frac{1}{2\pi}\right)^{N/2} e^{\frac{1}{2}(\bar{A} - \bar{m})^T C^{-1} (\bar{A} - \bar{m})}$$

where $\bar{A}$ is the $N$-dimensional vector representing the $N$-features, $\bar{m}$ is the mean of vector estimates, $C$ is the covariance matrix and $|C|$ is its determinant. $T$ stands for transpose of a vector. Therefore, if $\bar{m}_1$ and $\bar{C}_1$ are the mean vectors and covariance matrix of $N$-features for background noise data, $\bar{m}_2$ and $\bar{C}_2$ the corresponding parameters for boiling noise data and $S$ is a surface in $N$-dimensional space separating boiling and non-boiling regions then

$$p_S = 1 - \frac{1}{(2\pi)^{N/2}|C_1|^{1/2}} \int_{S} e^{\frac{1}{2}(\bar{A} - \bar{m}_1)^T \bar{C}_1^{-1} (\bar{A} - \bar{m}_1)} d\bar{A}$$

$$p_m = 1 - \frac{1}{(2\pi)^{N/2}|C_2|^{1/2}} \int_{S} e^{\frac{1}{2}(\bar{A} - \bar{m}_2)^T \bar{C}_2^{-1} (\bar{A} - \bar{m}_2)} d\bar{A}$$

Threshold is fixed by maximising the $(1 - p_S)(1 - p_m)$ product as in the case of single features. This method was also applied to KNS-data. The areas $A_1$, $A_2$ and $A_3$, standard deviation, skewness and kurtosis were taken as features. The above procedure was applied to File 1 data. It showed that PDF of signals in 6-dimensional space were separated for background and boiling noise data. For file 12 data, the PDF...
overlapped and hence no/No criteria was applied to get target values of $P_s$ and $P_m$. For file 1, the predicted time of onset of boiling was 27.4 s (0.4s more than the actual) and for file 12, with 15/16 criteria the onset of boiling predicted was 30.1 s (2.3s more than the actual).

vii. **Adaptive Learning Network (ALN) Method:** ALN-method is applied to file 1 and file 12 data with two generations. Areas $A_1$ and $A_3$ and pulses above a set threshold, $n$, were used as the input features. In the first generation $A_1$ and $A_3$ are input and $y$ is the output. That is

$$y = q_1 + q_2 A_1 + q_3 A_3 + q_4 A_1^2 + q_5 A_1 A_3 + q_6 A_3^2$$

$y$ and pulse counts, $n$ are the input features to the second generation network. $y'$ second generation output is

$$y' = q_1' + q_2' y + q_3' n + q_4' y^2 + q_5' y n + q_6' n^2$$

File 12 data was used as training set. Coefficients in the equations were estimated by minimising square error function and with the output equated to zero in background noise region and one in boiling region. Network equations were then applied to remaining data in file 12 and also to data in file 1. Fig.1 illustrates the second generation output from file 1, clearly indicating boiling.

Table 1 compares the time of onset of boiling detected by different methods and the required no/No criterion to achieve target probabilities in each case. The values of onset of boiling time are also given at the bottom of the table.

**B.2 Conclusions**

a. A simple method like pulse counting technique is quite sensitive to boiling noise.

b. All individual estimates tested were found to detect boiling effectively and are suitable for multifeature pattern recognition also.

c. This analysis work identified new features like areas under the PDF curve, sensitive to boiling noise.

d. Maximisation of the product $(1-P_s)(1-P_m)$ is found suitable for fixing boiling threshold.

e. Use of binomial distribution approach as suggested makes it possible to achieve low probabilities of false alarm and missing the boiling detection.
f. Multivariate pattern recognition and ALN technique predicted onset of boiling very close to true values and appears to be suitable for reliable boiling detection.

C. Analysis of BOR-60 Data

In BOR-60 reactor, boiling was generated by gamma heating in tungsten rod bundles in two different test rigs. Boiling signals were measured with ionization chamber and several acoustic transducers of sodium immersible type and wave guide type. Details of the signals recorded from different transducers are given in table 1. Results of the analysis of these data are given below:

i. Preliminary Analysis: All the data were checked and scanned for RMS and pulse counting analysis by RMS-meter and pulse counter built in-house. The inception of boiling could be identified by both the methods for waveguide and pressure transducer signals in file 3 and 4 that contained both background and boiling signals.

ii. Detailed Analysis: Detailed analysis was performed for the signals from pressure transducer P2 and waveguide W1 for file 2 (background), file 3 (background and boiling) and file 7 (boiling) using different methods discussed above for KNS-data analysis. As decided in RCM, the results were prepared in the following form.

- File 2: probability of false alarm, Ps Vs averaging time
- File 3: probability that boiling exists, as a function of tape length
- File 7: probability of missing boiling, Pm Vs averaging time

The results of this analysis are given in detail in Ref. 1. Table 3 lists the no/No criterion and the detection time of onset of boiling in file 3 for different methods tested on waveguide data. Conclusions from analysis of BOR-60 data are:

a. Instruments like RMS-meter and pulse counter are able to detect onset of boiling.

b. Features like areas under PDF-curve, RMS-value, PSD which were found sensitive to boiling for KNS data were also found sensitive to inpile boiling signals.

c. Application of no/No criterion enabled achievement of low target probabilities.
d. All the methods evaluated have shown high boiling probability in the boiling region of file 3. $P_s$ and $P_m$ decrease to lower levels for averaging times greater than a few tens of milliseconds. However, these low levels are generally still higher than the target limits.

e. The multivariate pattern recognition technique is found to work well for BOR-60 data also. However, it is to be tested for more number of features and optimised with respect to computational time.

f. Two generation ALN method also responded well for BOR-60 data and is a simple method mounting further evaluation.

D. General Conclusions and Recommendations

Analysis of KNS and BOR-60 data has led to the identification of features, sensitive to boiling. Using these features, algorithms of multivariate pattern recognition and ALN have been evolved and appears to yield reliable detection capability. This analysis work also led to the establishment of a statistical criterian (no/No) to achieve low levels of probabilities for false alarm and for missing a boiling signal.

Time required for detection depends upon feature and on the number of estimates required to achieve statistical reliability. A good criterion to qualify a detection method would be to compare detection time with the time within which an anomaly must be detected. Latter time is a function of reactor operating conditions. It is recognised that detection time would be more at boiling inception (behind a small blockage) due to poor signal to noise ratio, but the time available for detection would also be more. It is expected that the methods tested would meet the requirement.

Analysis work carried out so far involved only the analysis of single channel data. The merit of duel channel analysis by cross-correlation should also be evaluated. The two multifeature techniques are to be refined further. A combination of hardware based instruments and software based systems will have to be used in a reactor for on-line boiling detection using the above techniques to achieve small detection times with reliability. The present research programme has identified features and techniques which have to be suitably evolved into the online system. This will form continuing phase of the present work.

II.2 BNL (UK) - Work

BNL, after an initial survey to determine if boiling could be determined by audio or visual means by human observer, concentrated their main effort on the use of ALN - technique. The input parameters to the network used are : mean, rms, skewness,
kurtosis, pulse counts and low frequency components of PSD. The ALN-technique was applied to both KNS and BOR-60 data and the results were obtained in terms of Ps and Pm. Ps for BOR-60 data was higher than the target but could be reduced by basing the decision on sequence of samples. The ALN-technique demonstrated the possibility of having a general detection method which can adapt to different signal conditions. Summary of results in terms of Ps and Pm for both KNS and BOR-60 data is given in Figs. 2 and 3.

III. DDIAE (Australia) Work

In Australia, advanced theoretical data processing techniques were tried out. Higher order differentiation technique suggested by Australia enhances the high frequency component in the signal. Sodium boiling noise being dominant in high frequencies thus get enhanced in the background of relatively low frequency flow noise. This improves the signal to noise ratio significantly. Further, Australia suggested the mean square estimate, mean square of the prediction error and mean square of the 4th order PDF cost function as other features sensitive to sodium boiling. The sensitivity of these features to sodium boiling is very well reflected in Figs. (4) to (7) for file 2 of BOR-60 data. The peaks in mean square quantities in these figures stems from the use of a short averaging time. The four features mentioned above were evaluated in terms of modified discrimination defined as,

\[
D = \frac{\text{Height of 1st significant peak after boiling onset}}{\text{Maximum peak value for flow noise only}}
\]

The comparison is given in table III.1. The 6th order difference and Prediction Error are seen as most effective features.

<table>
<thead>
<tr>
<th>Method</th>
<th>Discrimination (10 log D)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mean Square</td>
<td>0.37 dB</td>
</tr>
<tr>
<td>6th order difference</td>
<td>10.22 dB</td>
</tr>
<tr>
<td>Prediction Error</td>
<td>9.48 dB</td>
</tr>
<tr>
<td>PDF - Cost Function</td>
<td>8.42 dB</td>
</tr>
</tbody>
</table>
IV. KFK - FRG

At KFK, Dr. Scherer, developed pattern recognition method with statistical variables sensitive to boiling as components of it. The method is described in detail in ref. 1. The method involves three stages of learning, surveillance and decision. In the learning phase, features of the signal are chosen which give the best separation of boiling and non-boiling classes. The features investigated during the surveillance phase are listed in Fig. 8 along with a histogram showing their relative effectiveness for data from transducer P1 in the BOR-60 experiment. In the decision stage a five stage classifier was used comprising the present estimate and four estimates generated from previous feature values. The number of these classifiers which indicate boiling at any one time is taken as a measure of confidence which can be attached to the decision. Fig. 9 shows how the number of classifiers indicating boiling varies through the experiment for data from W1. Each figure represents a 25 ms interval.

III. Conclusions and Recommendations

CRP has led to the identification of new features and techniques sensitive to boiling by different countries. These should now be suitably incorporated in multi-features pattern recognition methods and optimized with respect to CPU - time of computer. Merit of dual channel cross correlation analyses should also be evaluated. The techniques should further be tested on new data with decreased signal to noise ratio. The methodology evolved should be tested and suitably modified for acoustic leak detection in steam generators of LMFBRs.

REFERENCES


Table 1

KNS I Data: Detection of onset of boiling (seconds from the beginning of file) and no/No Criteria

<table>
<thead>
<tr>
<th>Method Tested</th>
<th>File 1</th>
<th></th>
<th>File 12</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Time</td>
<td>no/No</td>
<td>Time</td>
<td>no/No</td>
</tr>
<tr>
<td>Pulse Counting</td>
<td>26.7</td>
<td>1/1</td>
<td>36.7</td>
<td>11/20</td>
</tr>
<tr>
<td>RMS-normal-data</td>
<td>31.9</td>
<td>8/20</td>
<td>34.1</td>
<td>21/40</td>
</tr>
<tr>
<td>RMS-squared data</td>
<td>31.1</td>
<td>10/20</td>
<td>35.9</td>
<td>22/40</td>
</tr>
<tr>
<td>PDF area A1</td>
<td>29.0</td>
<td>5/10</td>
<td>31.4</td>
<td>22/56</td>
</tr>
<tr>
<td>PDF area A2</td>
<td>32.7</td>
<td>3/40</td>
<td>42.9</td>
<td>73/131</td>
</tr>
<tr>
<td>PDF area A3</td>
<td>34.4</td>
<td>6/9</td>
<td>34.3</td>
<td>28/48</td>
</tr>
<tr>
<td>PSD area</td>
<td>28.7</td>
<td>7/13</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td>Pattern Recognition</td>
<td>27.4</td>
<td>1/1</td>
<td>30.1</td>
<td>15/16</td>
</tr>
<tr>
<td>ALN</td>
<td>27.95</td>
<td>---</td>
<td>29.9</td>
<td>---</td>
</tr>
</tbody>
</table>

True values of Onset of boiling time:

File 1: 27.0 S, File 12: 27.8 S
### Table 2

**Details of BOR-60 Signals**

<table>
<thead>
<tr>
<th>File No.</th>
<th>Type of signal</th>
<th>Channel Number</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>1</td>
</tr>
<tr>
<td>1</td>
<td>Reference</td>
<td>Sinusoidal calibration signal</td>
</tr>
<tr>
<td>2</td>
<td>Background</td>
<td>W1</td>
</tr>
<tr>
<td>3</td>
<td>Background boling rig 1</td>
<td>W1</td>
</tr>
<tr>
<td>4</td>
<td>Background boling rig 1</td>
<td>W1</td>
</tr>
<tr>
<td>5</td>
<td>Reference</td>
<td>Sinusoidal calibration signals</td>
</tr>
<tr>
<td>6</td>
<td>Background +</td>
<td>W1</td>
</tr>
<tr>
<td>7</td>
<td>Stationary boiling</td>
<td>W1</td>
</tr>
<tr>
<td>8</td>
<td>Rig II boiling</td>
<td>D1</td>
</tr>
<tr>
<td>9</td>
<td>Rig II boiling</td>
<td>D1</td>
</tr>
</tbody>
</table>

### Table 3

**no/No Criteria and Onset of Boiling for File 3 Channel W1 of BOR-60 Data**

<table>
<thead>
<tr>
<th>Feature</th>
<th>no/No criteria</th>
<th>Time of onset of boiling (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PDF area A1</td>
<td>18/30</td>
<td>119.6</td>
</tr>
<tr>
<td>PDF area A2</td>
<td>9/23</td>
<td>120.1</td>
</tr>
<tr>
<td>PDF area A3</td>
<td>10/15</td>
<td>119.6</td>
</tr>
<tr>
<td>RMS</td>
<td>19/40</td>
<td>117.4</td>
</tr>
<tr>
<td>Area PSD</td>
<td>9/20</td>
<td>108.6</td>
</tr>
<tr>
<td>Pattern Recognition</td>
<td>22/31</td>
<td>121.5</td>
</tr>
<tr>
<td>ALN</td>
<td>-</td>
<td>111.3</td>
</tr>
</tbody>
</table>
FIG. 1  NETWORK OUTPUT Vs FILE LENGTH (FILE 1 KNSI DATA)
FIG. 2 SUMMARY OF RESULTS.
FIG. 3 SUMMARY OF RESULTS.
FIG. 4 PLOT OF MEAN SQUARE
FOR DATA OF FILE 2 - BOR - 60

FIG. 5 PLOT OF 6TH ORDER DIFFERENCE
FOR DATA OF FILE 2 - BOR - 60
**FIG. 6** PLOT OF MEAN SQUARE PREDICTION ERROR FOR DATA OF FILE 2 - BOR - 60

**FIG. 7** PLOT OF 4TH ORDER PDF COST FUNCTION FOR DATA OF FILE 2 - BOR - 60
FIG. 8  QUALITY MEASUREMENT / SENSOR P1
### CURRENT CLASSIFICATION

|   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |
| 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0 |
| 1 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |

### CLASSIFICATION SUMMARIZED BY 5 MODELS

|   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |   |
| 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 0 |
| 0 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 0 | 1 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1 | 1 | 2 | 1 | 3 | 1 | 1 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1 | 0 | 0 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 2 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 7 |
| 4 | 4 | 3 | 3 | 4 | 5 | 5 | 4 | 4 | 4 | 5 | 5 | 5 | 5 | 5 | 5 | 4 | 5 | 5 | 5 | 5 | 5 | 5 | 5 |
| 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 |
| 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 |
| 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 |
| 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 | 5 |

**FIG. 9 CLASSIFICATION TABLE/SENSOR W1**