Gesellschaft für Anlagen-
und Reaktorsicherheit
(GRS) mbH

Proceedings of the 1. OECD (NEA) CSNI - Specialist Meeting on Instrumentation to Manage Severe Accidents

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Annotation

Proceedings of a Specialist Meeting organized by OECD / CSNI - Principle Working Group 4, Task Group on Containment Aspects of Severe Accident Management (CAM) in Cooperation with the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Cologne (F.R.G.)

Keywords

Kernkraftwerk, Unfall, AM, Sicherheitsbehälter, Expertensystem, Instrumentierung, Konferenz

PREFACE

OECD member countries have adopted various accident management measures and procedures. To initiate these measures and control their effectiveness, information on the status of the plant and on accident symptoms is necessary. This information includes physical data (pressure, temperatures, hydrogen concentrations, etc.) but also data on the condition of components such as pumps, valves, power supplies, etc.

In response to proposals made by the CSNI - PWG 4 Task Group on Containment Aspects of Severe Accident Management (CAM) and endorsed by PWG 4, CSNI has decided to sponsor a Specialist Meeting on Instrumentation to Manage Severe Accidents. The knowlegde-basis for the Specialist Meeting was the paper on "Instrumentation for Accident Management in Containment". This technical document (NEA/CSNI/R(92)4) was prepared by the CSNI - Principle Working Group Number 4 of experts on January 1992 .

The Specialist Meeting was organized and hosted by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH at Cologne on March 16th - 17th, 1992.

The Specialist Meeting was structured in the following sessions:

I. Information Needs for Managing Severe Accidents,

II. Capabilities and Limitations of Existing Instrumentation,

III. Unconventional Use and Further Development of Instrumentation,

IV. Operational Aids and Artificial Intelligence.

CONTENTS

[1] remarks to this paper are enclosed in the summary of session IV

SESSION IV - Operational Aids and Artificial Intelligence

[2] remarks to this paper are enclosed in the summary of session III

SUMMARY and RECOMMENDATIONS of the SPECIALIST MEETING

First CSNI-Specialist Meeting onInstrumentation to Manage Severe Accidents

The First CSNI Specialist Meeting on Instrumentation to Manage Severe Accidents was held at Cologne, Germany on 16th and 17th March 1992. It was hosted by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. About seventy experts attended the Specialist Meeting from thirteen countries and two international organizations; these included an expert from the Czech and Slovak Federal Republic. Twenty-two papers were presented in four sessions. The proceedings will be published by GRS under separate cover.

The Specialist Meeting concentrated on existing instrumentation and its possible use under severe accident conditions; it also examined developments underway and planed. Desirable new instrumentation was discussed briefly. The interactions and discussions during the sessions were helpful to bring different perspectives to bear, thus sharpening the thinking of all. Questions were raised concerning the long-term viability of current (or added) instrumentation.

It must be realized that the subject of instrumentation to manage severe accidents is very new, and that no international meeting on this topic was held previously. One of the objectives was to bring this important issue to the attention of both safety authorities and experts. It could be seen from several of the presentations and from the discussions that this kind of work is still in a planning phase. The following conclusions and recommendations must therefore be seen as preliminary.

1. To make decisions which are appropriate and effective to control and mitigate an accident, it is essential to have the clearest picture possible of the accident and its progress. This can be obtained by accumulating information from as many sources as is practical.

2. It is important to use a systematic approach to evaluate accident sequences, information needs and instrument capabilities in severe accident conditions.

3. It should be confirmed that instrument performance will be sufficient to give the information needed to manage a severe accident. In some cases the instruments may function beyond their specifications.

4. Important lessons can be learned from the TMI and LOFT-FP-2 measurements from the instruments, in particular for instruments giving new information (e.g. Source Range Monitor (SRM) information about vessel water level).

5. All participants agree on using the full instrumentation and accident management capacity of the plants. All are focusing on making full use of post-TMI-2 safety enhancements and instrumentation additions already in place.

6. Most participants agree on the types of measurements which will prove useful. Various means are being pursued to think ahead and interpret plant status, such as computer codes and calculational tools.

7. An important conclusion is that there is a need for additional work on unconventional use of existing instrumentation under severe accident conditions.

8. This work will identify areas where existing instrumentation can indirectly contribute to the information needs in severe accident situations and areas where it cannot, thereby giving indications on desirable new developments.

9. The question of new accident management instrumentation was raised. The current perspectives are based on national objectives, and dependent on the optimism or pessimism of participants over the longer term viability of instruments. It is clear that efforts to ensure the long-term viability of instruments are being pursued by all (with a reasonable "common sense" attitude). In fact, the pessimistic view is "conservative" and leads planners to make prudent provisions to manage the accident with any instruments that may be available.

10. Some new instruments are being developed; their possible usefulness under severe accident conditions needs to be further qualified.

11. In spite of the different purpose, some instruments used in experiments can be evaluated and qualified also for current nuclear power plants.

12. The papers presented at this meeting clearly showed that most approaches of expert systems are still in a conceptual phase. Some applications transferred from other fields are under development for use in the severe accident domain. Only those systems that offer a set of less sophisticated tools can be said to be readily available for limited purposes.

13. Expert systems may be of help to plant staff and external experts, but cannot substitute for them.

14. There will not be a single expert system for severe accidents (i.e. a general problem solver) but rather a set of simpler systems devoted to specific goals in situations that can be clearly identified.

15. Expert systems should have the capability to verify plant conditions and assumptions made by the operating personnel.

16. Expert systems used in this domain must be even more explanatory and transparent to permit verification of their conclusions by the personnel .

17. Expert systems should, if possible, also be used during normal plant situations to increase operating personnel confidence.

SESSION I

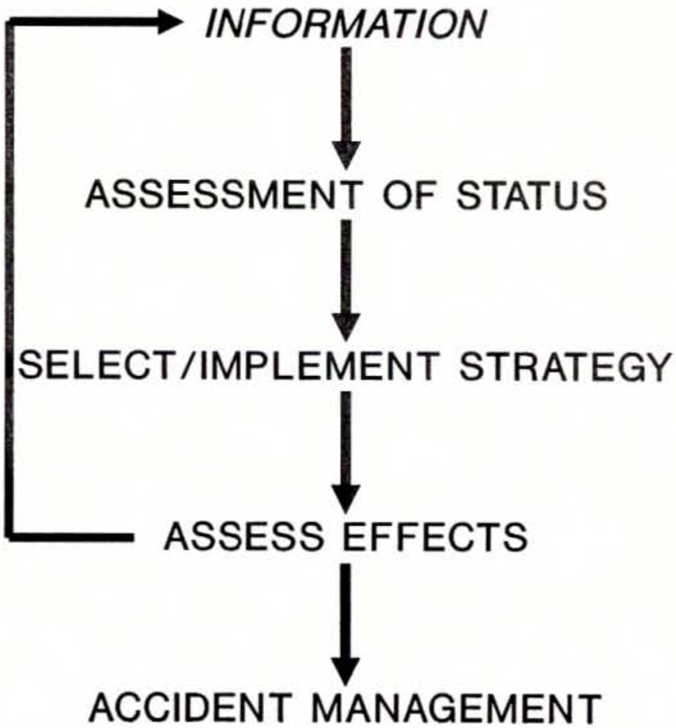
Information Needs for Managing Severe Accidents

Chairman: R.N. OEHLBERG

INSTRUMENTATION FOR ACCIDENT MANAGEMENT IN CONTAINMENT

Task Group on Containment Aspects
of Accident Management

ACCIDENT MANAGEMENT PROCESS



ACCIDENT MANAGEMENT INFORMATION NEEDS

ACCIDENT MANAGEMENT PLANNING

REQUIRES

ACCIDENT MANAGEMENT INFORMATION PLANNING

SYSTEMATIC PLANNING FOR ACCIDENT MANAGEMENT

REQUIRES

SYSTEMATIC ASSESSMENT OF INFORMATION NEEDS

INEL METHODOLOGY FOR SYSTEMATIC ASSESSMENT OF ACCIDENT MANAGEMENT INFORMATION NEEDS

SAFETY OBJECTIVE

↓

SAFETY FUNCTION

↓

CHALLENGES TO SAFETY FUNCTION

↓

MECHANISM/PHYSICAL PHENOMENA INVOLVED

↕

POTENTIAL STRATEGIES TO RESTORE SAFETY FUNCTION

SYSTEMATIC ASSESSMENT OF ACCIDENT MANAGEMENT INFORMATION NEEDS

FOR EACH MATCH OF CHALLENGE AND STRATEGY:

- ASSESS AVAILABLE INSTRUMENTATION
- CONSIDER EFFECT OF ACCIDENT ENVIRONMENT ON INSTRUMENTATION
- TABULATE OTHER SOURCES OF INFORMATION
- ASSESS NEED FOR SUPPLEMENTAL SOURCES OF INFORMATION

TYPICAL RESULTS
OF INSTRUMENTATION ASSESSMENT

SAMPLE APPLICATION OF INEL METHODOLOGY INDICATES:

- INSTALLED INSRTRUMENTATION IS USUALLY QUALIFIED FOR DBA CONDITIONS ONLY

- PERFORMANCE BEYOND DBA IS GENERALLY UNKNOWN (WITH EXCEPTION OF A FEW SANDIA TESTS BEYOND DBA)

- FOR MANY STRATEGIES, OTHER SOURCES OF INFORMATION NEED TO BE TAPPED

OTHER SOURCES
OF ACCIDENT MANAGEMENT INFORMATION

OTHER SOURCES OF INFORMATION ABOUND
-- BUT MAY BE DIFFICULT TO INTERPRET:

- INSTRUMENT READINGS BEYOND QUALIFICATIONS
  (If bias is known, readings can be corrected)

- INSTRUMENT FAILURE MODE may give clue about conditions which caused failure

- SYSTEMS STATUS (OPERATIONAL/FAILED)
  (Operability/failure may indicate conditions)

OTHER SOURCES -- continued

- COMPONENT FAILURES
  (Failure mode may indicate cause of failure)

- UNINTENDED USE OF INSTRUMENTATION
  (e.g. use of neutron instrumentation
  as water level indication at TMI-2)

- PORTABLE/REMOTE INSTRUMENTATION
  (e.g. use of portable rad. monitor to obtain
  Containment radiation levels from outside)

- SYNTHESIS OF SYSTEM/COMPONENT FAILURE STATUS
  (May give composite picture of plant conditions)

MEETING OBJECTIVE

IT IS THE HOPE OF THE "CAM" TASK GROUP THAT THIS EXPERT MEETING WILL PROVIDE THE OPPORTUNUTY TO PURSUE NON-ROUTINE AND CREATIVE METHODS FOR OBTAINING, INTERPRETING, SYNTHESIZING, AND APPLYING INFORMATION FOR ACCIDENT MANAGEMENT.

SWEDISH APPROACH TO INFORMATION NEEDS IN SEVERE ACCIDENT SITUATIONS

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ABSTRACT

In Sweden, systems for mitigating severe accidents have been installed at all plants and procedures have been implemented for accident mangement. This work has included the assessment of needs of information and the survivability of existing instrumentation during the various phases of an accident scenario.

The approach has been pragmatic and based on existing knowledge of accident phenomenology and MAAP code calculations together with plant staff experience of detailed plant design and installation.

During the early phases of accidents, which is defined to remain up to maximum fuel temperatures in the order of 800 °C, the ordinary instrumentation is to a great extent useful. The reactor vessel level measurement is however identified to be weak in BWRs as soon as the core is partly uncovered. This has lead to the development of a Core Cooling Monitor, which is presented by Professor Becker in a separate presentation.

In later phases of accident scenarios, the general basis has been that no instrumentation inside the containment can survive. It has been analysed what information is strictly needed. It has been found that detailed information of the status inside the pressure vessel is of little importance after vessel penetration.

Certain important information needs have been identified, that was not safely accessible from existing instrumentation. This has lead to complementary installations, using instruments inserted into the containment through protected guide tubes. Also for sampling of gas and water complementary installations have been made.

In conclusion, several improvements in accident mangement instrumentation have been made or are underway. However, also problems have been identified, which call for further attention. Likewise, more systematic analyses that are underway in Sweden and abroad may indicate that we have overseen something

Introduction

The Swedish nuclear utilities have all implemented severe accident mitigation systems and accident management procedures for their 12 nuclear units. This means that they have had to face also the information needs in different accident situations. It does not, however, mean that there is nothing more to be done.

The time schedule imposed on the utilities by the Swedish Government, to have finalized the "first round" of actions by the end of 1988, has made it necessary to make the best out of the information available and thus to find solutions that are "good enough". The accident management strategies and an overview of Swedish works going on in the field was reported at the OECD/NEA Specialist Meeting in Rome last autumn (1).

This report will give a short overview of the Swedish severe accident management organization and strategy and take this as a basis for the further discussion. The general idea is to put the reader into the position of the 'accident manager' and to discuss the needs from a pragmatic point of view rather than to make a full scope analysis of all possibilities.

To make the report tangible, the Forsmark 2 plant will be used as an example. There certainly are differences in approach between the Swedish plants, but there are more similarities.

Swedish Severe Accident Management Organization

A basic idea in Sweden is that accident management starts as soon as a plant disturbance has occurred that **has, or should have, caused a reactor scram.**

This approach means that accident management starts already at the first precursor of an accident. It also means that the staff has to apply the Emergency Operating Procedures, EOP, at each scram, and thus get acquainted to their use. In the early stages of an accident, it is a pure operator responsibility to handle the siruation.

Coming further into an accident state the plant management and the Technical Support Center, TSC, may be available for support. In case of rapidly developing scenarios this may, however, not be possible. The operators thus need to have authorization to handle even severe accidents without tedious

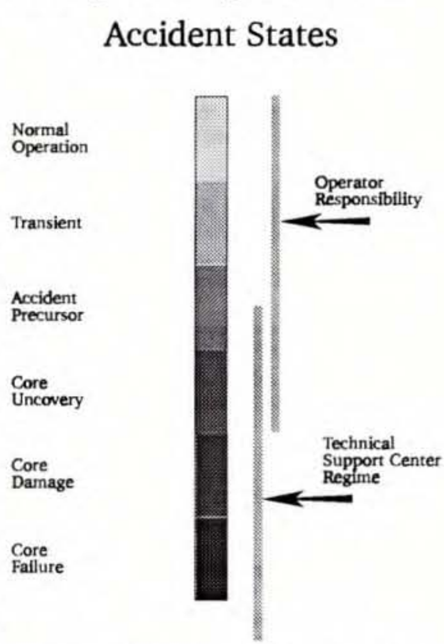


Figure 1: Accident states and responsibilities

contacts *e.g.* to get permission for use of filtered venting. This is also one of the important basic elements of the Swedish legislation. The plant itself is authorized to take all decisions concerning the accident management at the plant, including the decision to relief radioactive gases, if this is necessary to mitigate the plant state.

The shift crew consists of a shift supervisor, a reactor operator and a turbine operator plus two station technicians and a shift electrician, the last two working mostly outside the control room. The two operators are responsible for the operation of the plant. They have event oriented procedures also to handle all kinds of disturbances. The shift supervisor is normally working in his office adjacent to the control room.

In case of a reactor scram, the shift supervisor is to apply the EOP, which is functionally oriented to four critical safety functions: Reactivity, Core cooling, Heat sink and Activity barriers. The work in the control room thus, to a certain extent, represents redundancy and diversification. The instrumentation and information system is, however, the same for the opertors and the shift supervisor respectively.

The EOPs cover, as indicated in figure 1, accident states rather far into severe accident states. It is assumed that the TSC and plant mangement is available for support further into the accident. At this stage, they will also have access, and time to apply, their set of background information, the Technical Handbook for Plant Management with the Swedish acronym THAL. Whereas the EOPs are distinct procedures the THAL contains a more elaborate background material but calls for assessment of the situation on a broader basis.

The EOPs contain protocols for reporting the situation to TSC and Plant Management. These reports will be very important, as the credibility of instrumentation as a whole will be questionable if an accident develops far into the severe regime.

Basic Information Needs in Severe Accident Situations

When designing severe accident mitigating systems and Emergency Operating Procedures, the basic strategy was that existing instrumentation should not be taken credit for, as it was not designed and installed taking severe accident conditions into account. The analysis thus was started from scratch and from "the back end", *i.e.* in a situation when the reactor vessel is penetrated by the core melt and part of the core is already in the bottom of the containment. In this situation information from the vessel itself may be of help, but is not central. The list of necessary information then was found to cover

1. Containment pressure
2. Containment water levels
3. Temperature in drywell
4. Radiation level in drywell
5. Activity in water- and gas phases

This basic list of information needs naturally is insufficient to give a picture of the containment and reactor accident state, and it will be necessary also to analyze the availability of existing instrumentation in different accident situations. The result of this analysis is contained in the THAL handbook.

Availability of Existing Instrumentation

Reactor instrumentation

Pressure- and level instrumentation is located outside the containment, generally divided in four subdivisions and connected to the reactor with instrument pipes. Cables to temperature sensors *etc*. are qualified for Design Basis Accident conditions. Within design the instrumentation thus should be credible as a whole. The pressure transducers are believed to be operable also far into the severe accident regime.

Already following a severe reactor pressure transient, *e.g.* caused by a pipe rupture, the reactor water level measurement has to be checked. The procedures then include control of reference leg cooling and crosschecking of instrument subdivisions etc. Uncertainty in this verification leads to priority for water injection to the pressure vessel.

The water level indication in the vessel is poor when the water falls below the top of core level, 1 single instrument measuring a range of some 13 m. The THAL handbook contains information on how the neutron flux measurement is expected to react during core uncovery. This can help to give complementary information, but needs care in interpretation. This is also one of the reasons for the Swedish interest in developing the Becker Core Cooling Monitor, BCCM, which is presented in a separate presentation.

Following core melt, most of the pressure vessel instrumentation will be affected by extreme temperatures. Also deposition of aerosols and activity may contribute to plugging or superheating of instrument pipes and to difficulties to interpret instrument recordings.

Containment instrumentation

The containment instrumentation has been analyzed for a broad range of possible accident states, also the late state when the containment is reflooded up to a level in range of the reactor core. Several of the existing instruments do not meet requriements for severe accident states.

Containment pressure	is insufficient in range and may be covered with water during containment flooding
Water level in lower drywell	is insufficient in range. Risk for damage by missiles and susceptibility to high temperatures
Water level in wetwell	is believed to operate also in accident situations
Temperture in wetwell	is designed only for the range 0-60 $^{\circ}C$
Temperature in lower drywell	is designed only for the range 0-60 $^{\circ}C$. Also subject to missiles and excessive temperture
Temperature in upper drywell	Some of the instruments will be drowned when flooding the containment
Radiation monitoring	Some of the sensors are not suitably located with respect to flooding and temperature

Thus, both reactor and containment instrumentation may be used under certain conditions but respect has to be taken to the specific accident situation and the plant operational state. This is difficult to include in a systematic way into the EOPs, which have to be simple and straight on. A lot of the information is, however, included in the THAL handbook.

Instrumentation for Severe Accidents

When design was made for the severe accident mitigation systems, special instrumentation was included to supervise the containment state also in severe accident situations. The instruments now built into the plant take their power supply from a separate battery powered system. It has capacity to run the instrumentation for 24 hours and has connections prepared for recharging from a mobile generator. All the equipment is qualified for severe accident conditions, including earthquake, with instrument pipes and radiation sensors protected for missiles and excessive temperatures. All electronic devices are located outside the Containment. Instrument panels are located in the central control room and also in the two emergency control boards and in the filter system local control panel. The instruments installed are shown in figure 2.

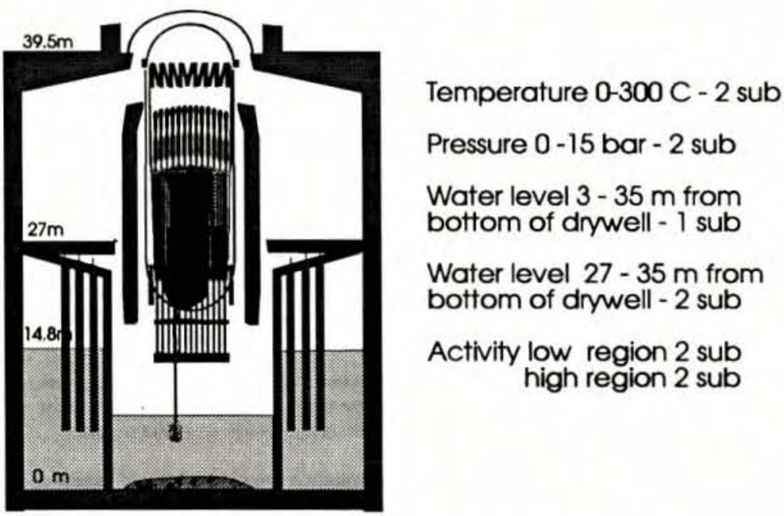


Figure 2: New instruments for severe accident conditions

Also the measurment ranges are designed to severe accident conditions and to the possibility of flooding the containment in the late stage of recovery. The radiation monitors have ranges 10^{-4} - 10^{2} (low range) and 10 - 10^{6} Gy/h (high range) respectively. The water level gauges are of bubble type blowing nitrogen into the bubble tubes and measuring the pressure difference.

In addition to these instruments, the PASS, the Post Accident Sampling System, enables the staff to take samples of

1 reactor water (before meltthrough)

2 condensation pool water

3 lower drywell water and

4 containment atmosphere

The PASS system will be rather slow as it is a manual system and the samples will be highly radioactive.

From viewing the list of instruments above, it can be concluded that the information given by them will be very poor standing alone. The plant management and the Technical Support Center would have very limited capability to assess an accident situation on the basis of only this information.

Accident History Information Needs

When technical support staff arrives to the plant, the above instruments will be all they can rely on without knowledge of the plant state history. Thus the documentation of plant parameters during the accident is crucial for their ability to perform accident management.

In the Swedish plants this is taken into account in the EOPs by predesigned protocols to be filled out regularly during an accident - in this context it is an advantage to define the initiation of accident management as has been done for Swedish plants - and there are also a standardized procedure for what information is to be given to the support staff at initial handover of the accident state.

If the operators have had full knowledge of the development of the accident and followed their procedures, the historic reports thus will, together with instrument readings, give an acceptable basis for the further accident management. It is, however, a plausible part of a severe accident scenario that the operators have misinterpreted the situation one way or the other or have not realized the importance of some information. In such a case, the historic report will be misleading or at least incomplete. Examples of this can be found not only from drills but also *e.g.* from the TMI accident.

In any case, the fresh view of the support staff will lead to complementary questions concerning the sequence of events and of parameter values. The answers to these, if available, will be difficult to synchronize with the historic picture given in the report.

Again, the conclusion must be that the late state accident management would improve considerably in capability and precision, if the documentation of operating history could be improved in quality and in detail and still better would be if the operators could be relieved from making the routine part of this documentation and from answering questions that could be answered by the plant computer.

Naturally, such enhancement would be beneficial also for the operators in the accident precursor and early accident situations. That is why we in Sweden have put more effort in analyzing and preparing for accident management in the early accident stages than in late accident management. If we are successfull, the operators will have tools available to avoid the development of the accident into a severe state.

Possibilities to Enhance Accident Documentation

Modern computer technique using expert systems can be used for improving the information quality delivered to the operator. The improvement can be of many different kinds:

* reduction of alarm message flow
* supervision of critical safety functions
* elaborate verification of parameter values
* calculation of parameters that are not directly measured

Several of these tasks are more or less implemented in a number of plants, especially the first, signal reduction. The SAS II-project in Forsmark 2 has made a comprehensive research project on the second (2).

The SAS II project has applied an expert system shell to supervise all the four critical safety functions that are subject to control in the Emergency Operating Procedures. The presentation is also adapted to the EOPs. They are meant to be a support for the shift supervisor in his work with the EOPs. The SAS II system also contains logics, by the help of which the shift supervisor is able to track malfunctions within the critical safety functions back to the root-causes.

The system was developed using experienced plant operators for functional design. It has now been validated by a number of shift supervisors, using the SASII system operating on a plant simulator. It is being installed in the Forsmark 2 plant by the end of this year.

Another Project, the CAMS-system (3), is still in a preproject phase. The idea is directed towards accident management, but is actually a broader and more general approach, in which accident management is one of several applications. Each module in the system outline (figure 3) could be developed separately and only the data base would be the same.

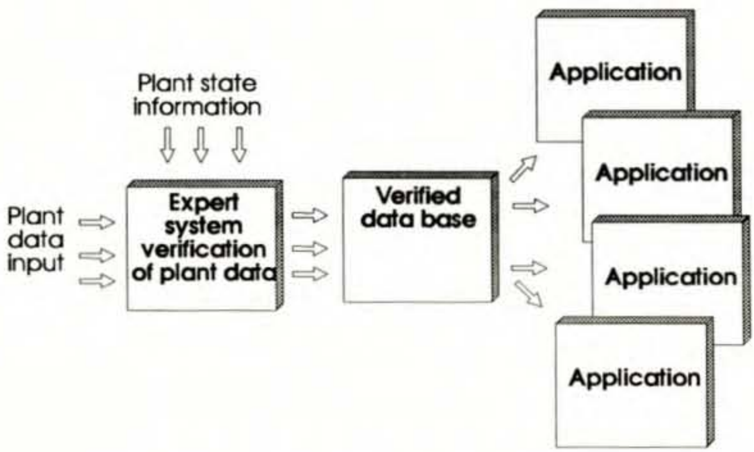
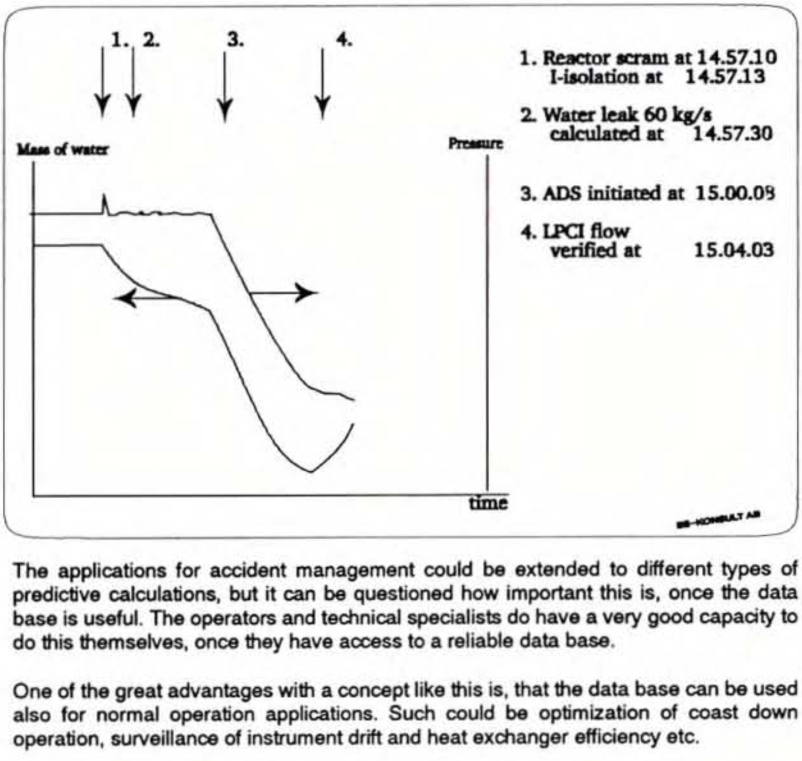


Figure 3: CAMS General System Structure

The verified data base already exists in a fragmentaric form in most plants. Median values of redundant channels are calculated excluding channels with excessive difference from the others, *etc.* More elaborate systems would include comparison of flowmeter indication with state of according pumps and valves. A full scope system could also use continuity conditions or even a full scope simulator, which calculates faster than real time and checks the validity of several measured value. It could also calculate parameter values that are not directly measured. During shutdown and accidents the decay power and the core two-phase water level are examples of useful information.

Having established a data base, the applications should be built separately and adapted to each specific user. Thus the operator will need the information presented at one level of detail and adapted to the control room procedures and practices, whereas the TSC staff would need another set of presentations. An example of a useful TSC presentation is shown in figure 4.



The applications for accident management could be extended to different types of predictive calculations, but it can be questioned how important this is, once the data base is useful. The operators and technical specialists do have a very good capacity to do this themselves, once they have access to a reliable data base.

One of the great advantages with a concept like this is, that the data base can be used also for normal operation applications. Such could be optimization of coast down operation, surveillance of instrument drift and heat exchanger efficiency etc.

Conclusions

The accident management in a far developed accident with the reactor vessel molten through and part of the core in the containment outside the reactor vessel can never be efficient only on the basis of instrumentation that is reliable in that late accident state. It is strongly dependent on reliable information from the earlier part of the scenario.

Information from the accident initiation state can be improved by computerized support, aiming at verification of parameters and plant state information and at documentation of the parameters and sequence of events.

Such improvements will lead to double benefits:

* Improvement of the capability of the shift crew to interpret the information correctly and thus improve their possibility to recover a stable situation before it has developed into a severe accident state.
* Improvement of the basis for the Technical Support Center and Plant Management to assess the situation and to take the appropriate decisions, should a severe accident occur.

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Information and Requirements needed for Accident Mitigation

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ABSTRACT

This paper describes a methodology to determine the needed information and instrumentation as well as the requirements to the equipment for accident mitigation. The investigations are made for typical German PWR's with a large dry containment. The methodology presented comprises the following four steps:

- development of Safety Objective Trees for German PWR's
- determination of information needs and information sources
- assessment of instrumentation capability
- comparison of information needs and available instruments

The part of the developed Safety Objective Trees which characterizes the Safety Objective "Prevention of Containment Failure" will be used to demonstrate the application of the presented method. The main points discussed after that are the criteria to assess instrumentation capability and the results of the comparison between needed information and available instruments concerning accident mitigation measures, e.g. containment venting and measures to limit the H_2-concentration inside the containment.

Also, the general requirements to the equipment necessary to realize the presented accident mitigation measures will be demonstrated. Strong environmental conditions expected during severe accidents as well as criteria to minimize the release of radioactive substances are reasons for the high equipment requirements.

Finally, this presentation gives more detailed information about accident mitigation while the paper with the title "Information Needs for Accident Management - GRS Approach" presented at the CSNI specialist meeting on "Severe Accident Management Programme Development" held in September 1991 at Rome [8] gives more information about prevention measures and aspects.

Paper presented at the
CSNI Specialist Meeting on

Instrumentation to Manage Severe Accidents

GRS (mbH), Cologne, Germany
March, 16th - 17th 1992

1 Introduction

Accident management comprises the total of all measures to analyse, to control and to manage severe accidents as well as the prevention or mitigation of the conseqences to the environment. The planned actions and preparatory measures will enhance the safety, the capability, and the reliability of nuclear power plants.

Accident mitigation comprises three *Safety Objectives*:

- retention of the core in reactor pressure vessel
- prevention of containment failure
- limitation of fission product release.

The availability of all needed information with the help of adequate instrumentation is necessary to realize the prepared AM measures or plans. Without adequate diagnostic capability or instrumentation the operating personnel cannot reliably identify the plant status, cannot select the correct and most effective strategy and can not control the effectiveness of the selected strategy.

The safety-related instrumentation installed in nuclear power plants was primarily designed for the conditions of the DBA's. After the TMI accident the importance of the instrumentation increased and new requirements for safety-related and wide-range instrumentation were developed.

The GRS accident management program, which is sponsered by the BMU (Federal Ministry for the Environment, Nature Conservation and Nuclear Safety), comprises many different activities. One of them, a methodology to determine systematically the needed information for accident management, will be presented. An application of this methodology to a typical German Pressurized Water Reactor (PWR) with a large dry containment is given. Especially the information needs and requirements for the filtered containment vent strategy and the H_2-limitation measures will be presented.

Answers to the following questions should be given as a result of the investigations.

- Which Information is needed during accidents for understanding the status of the plant, for equipment diagnosis, for decision making and for control of the effectiveness of accident management measures, e.g. for filtered containment venting and H_2-limitation measures?
- Which are the general requirements to the equipment?
- Could the installed instrumentation supply the needed information for the realisation of the above-mentioned measures?
- Which instruments are able to function under conditions which have to be expected during severe accidents and which are the challenges to the normal function of the instruments?
- Which potential or additional instruments could be useful to achieve needed information and which are the general requirements for those instruments ?

2 Approach

Our approach to determine the needed information for AM and to assess the existing instrumentation includes 4 steps.

Firstly *Safety Objective Trees* for LWR were developed, e.g. for a 1300 MW PWR. The calculated design basis and severe accidents are the basis for the development of these *Safety Objective Trees.* The physical phenomena and mechanisms occuring at these events are considered. The safety or AM goal for all developed trees is the "reduction of beyond-DBA consequences". Four *Safety Objectives* were selected but only the last three of them are relevant for *Accident Mitigation*:

- prevention of core damage
- retention of the core in the reactor pressure vessel
- prevention of containment failure
- limitation of fission product release

Also the different *Safety Functions*, their *Challenges* and the *Mechanisms* occuring were determined. New *Strategies* to prevent the *Challenge* or to mitigate their consequences were evaluated.

Secondly, the *Safety Objective Trees* at every branch point were examined; than it was determined which information is necessary. For better understanding all this information is set into tables (see table I and II). We selected the needed information to maintain *Safety Functions* and to diagnose the *Challenges* and the *Mechanisms* . The information needed for the selection of *Strategies* is subdivided into two parts: criteria for selection and criteria for control of the effectiveness. The developed general requirements for the realization of the measures (e.g. requirements to the equipment) are taken into account. The selected information sources could supply the needed information directly (e.g. temperatures, pressures, position of valves, power supply) or indirectly (e.g. balance of heat generation and heat sink, status of the core or the fuel assembly, relocation of the core). Therefore the information sources were subdivided into these two parts. Finally the existing instrumentation of a typical German 1300 MW PWR was determined in connection with the selected information sources.

Thirdly, criteria for the assessment of the *Instrumentation Capability* were found out. The range and accuracy of measurements as well as the environmental qualification conditions and failure criteria have to be determined. It is not so easy to find out more simplified criteria to assess the function of instruments under severe accident conditions.

After comparison with conditions prevailing during severe accidents it can be identified which information will be supplied with the existing instrumentation and which additional instrumentation will be needed.

3 Application to a PWR - Containment Mitigation Measures

In order to distinguish between accident prevention and mitigation four *Safety Objective Trees* were selected . As an example for the demonstration of the method the tree which includes the *Safety Function* "Prevention of containment failure" and is relevant for the mitigation phase is presented in this paper. The maintaining of the containment integrity is very important, because it is the ultimate barrier for retention of fisson products. Some of the phenomena which challenge the containment safety function are presented. The containment venting strategy and the different H_2-reduction measures connected with these phenomena.The investigations are made for a PWR with a large dry containment.

3.1 Safety Objective Tree

The *Safety Objective Tree* presented in figure 1a and 1b comprises three *Safety Functions*

- "Pressure Control" (C1),
- "Temperature Control" (C2),
- "Maintain Containment Integrity" (C3).

These *Safety Functions* will be presented in greater detail in the following chapters.

3.1.1 "Pressure Control" (C1)

The challenges and mechanisms occuring will be characterized before the determined *Strategies* will be presented in chapter 3.2.

"Slow Pressurization" (C1A)

The *Safety Function* "Pressure Control" (C1) will be challenged e.g. due to "Slow Pressurization" (C1A). During different accidents in the early accident phase big steam mass flow rates released into the containment via the leak or through the pressurizer safety valves. In the later accident phase, if core cooling could not maintained and a reactor pressure vessel failure occures, continuous mass flow rates of steam will be produced in the case of melt-concrete interaction with or without sump water contact and/or as a result of evaporation of sump water by saturation conditions within the containment. Also, high concentrations of noncondensable gases as a result of the melt concrete interaction could challenge the *Safety Function* of the containment.

"Rapid Pressurization" (C1B)

In principle three different *Mechanisms* could lead to a "Rapid Pressurization" (C1B) of the containment. Two of them, "Direct Containment Heating (DCH)" (C1B1) and "Ex-Vessel Steam Explosion " (C1B2), often discussed in connection with other PWRs [1, 4, 5] are considered as not relevant for KWU-type PWR's in the German Risk Study Phase B [2] and therefore more detailed information is not treated in this paper. The increasing H_2-concentration inside the containment may lead to a H_2-detonation process if there are no measures to reduce the concentration before. This detonation has the potential to destroy the containment due to rapid overpressurization.

"Subatmospheric Pressure" (C1C)

The "Subatmospheric Pressure" (C1C) path describes the *Mechanisms* occuring in connection with the long-term operation of containment venting followed by steam condensation inside the containment and low concentrations of noncondensable gases.

3.1.2 "Temperature Control" (C2)

Two different phenomena are selected which may challenge the *Safety Function* "Temperature Control" (C2).

"Temperature in Atmosphere to high" (C2A)

The high temperature in containment atmosphere could be a result of the high energy input of a steam mass flow or of a H_2-deflagration process or as a result of a fire accident. The containment integrity will be challenged, e.g. in case of a failure of containment penetrations due to high temperature over a long time period.

"Melt-Concrete Interaction" (C2B)

The basemat of the containment especially the steel shell will be challenged if the melt / concrete interaction could not be stopped.

3.1.3 "Maintain Containment Integrity" (C3)

Two different phenomena are selected, which may challenge the *Safety Function* "Maintain Containment Integrity" (C3).

"Loss of Tightness of Containment" (C3A)

There are also two different *Mechanisms*, an "Isolation Failure" (C3A1) and a "Leak in Containment Shell / Penetration" (C3A2), which could lead to a loss of containment safety function. The "Isolation Failure" *Mechanism* has to different modes, any failure before or after isolation of the containment.

High atmospheric temperatures during several minutes or hours could be the reason for the penetration failure.

However not only the fact that there is a leak in the containment shell is enough for the loss of containment safety function; the diameter of the leak or the leak area is also important.

"Generation of Missiles" (C3B)

There are also three different *Mechanisms*, "Steam Explosion In-vessel" (C3B1), "H_2-Detonation" (C3B2) and "High Pressure Failure of RPV" (C3B3), which could lead to a loss of *Safety Function* of containment. But the probability that any of these mechanisms will occure is very different. The result of these explosions, detonations or rupture events are nearly the same. Some parts of the facility with high energy may destroy the containment immediately.

3.2 Strategies

As shown in fig. 1 there are possible many different *Strategies* to prevent or to mitigate the consequences of severe accidents. To select the special and most effective *Strategy* it is necessary to detect the *Mechanism* occuring without any doubt.

Not all of these measures are realised in German PWR's. They will be discussed and the general requirements to the instrumentation and the equipment for realisation will be determined [6, 7, 8].

This presentation will give only more detailed information about two of the possible strategies - containment venting - and the measures to limit the H_2-concentration inside the containment, e.g. by - catalytic recombination and ignition -.

3.3 Information Needs and General Requirements to the Equipment

The second part of our method comprises the determination of needed information and available information sources. The path C1A "Slow Pressurization" of the *Safety Objective Tree* will be used to demonstrate the method. The needed information and available information sources to detect the *Challenges* and the *Mechanism* and to select the *Strategies* are depicted in the tables I and II.

3.3.1 "Steam Production" (C1A1)

a) Information Needs

The needed information to detect the *Challenge* of the *Safety Function* is the time-dependent containment pressure, which is a direct information (see table I).The needed information to detect the mechanism "Steam Production" (C1A1) is the relation of heat production and heat removal and the steam concentration inside the containment. The first needed information will be supplyed indirectly by the time-dependent containment pressure and temperature. Instruments to measure the steam concentration are not available for the investigated plant. Information about the contact of the melt in the reactor cavity with the sump water in the later event phase is necessary to detect the reason for the steam production. An information source which could supply this information could be the decreasing sump water level or the increasing containment pressure at the beginning of the contact. But it is very difficult to detect this mechanism without doubt.

The necessary information to select the filtered venting strategy (strategy 6 in table I) is the time-dependent containment pressure and temperature and the concentration of noncondensable gases inside different rooms. Also the loss of tightness and the fission product invetory in the containment atmosphere or the time difference between the initiation of the event and the initiation of the filtered venting are criteria or needed information.

The following points gives more information about the criteria:

- The containment venting strategy has to be initiated if the pressure exceeds the design pressure of the containment.

- Also, high atmospheric temperature (about 145 °C) is a criterion to initiate containment venting strategy, to remove the heat and to reduce the loadings to the equipment, e.g. instrumentation, cables, penetrations etc..

- The containment venting strategy has to be initialised if an important leak will be detected or if the melt breaks through the basemat and the pressure is much higher than in the annulus or in the environment.

- It is not so easy to find out the best point in time for the first start of containment venting. If the fission product inventory in the containment atmosphere is to high and the containment venting will be initialised at this time then the fission products may lead to a high amount of heat generation in the filter system. In this case the venting system should not be initiated earlier than about three or four days after the initiation of the accident.

- Information about the concentration of the components of the atmosphere (air, H_2, CO + CO_2) is absolutely necessary before the venting strategy can be initiated. The concentration of H_2 has to be reduced to nonburnable concentration. If the H_2-concentration is too high the venting system may fail in case of a H_2-deflagration or -detonation and an uncontrolled release of fission products out of the containment may occur.

b) General Requirements

The general requirements depend upon the design of the system. Different filter systems and positions of the filters inside or outside the containment are possible. Also, different points to connect the system with the containment are possible, but it is not a task for this presentation to give a survey of different containment venting systems. Some global requirements will be given below [6]:

- connection of the venting system to existing parts of the NPP

- the possibility to control the valves of the venting device from the control room

- determination of the mass flow rate of the venting stream (it depends upon the NPP)

- determination of the filter factors for iodine and aerosols

- instrumentation to measure the following parameters

    - pressure, differential pressure of the filter unit

    - filter temperature (e.g. after closing the venting line)

    - temperature of the venting stream

    - water level inside the venturi filter (if used)

    - radioactivity behind the filter unit

3.3.2 "Noncondensable Gas Build-up" (C1A2)

a) Information Needs

The second *Mechanism* "Noncondensable Gas Build-up" (C1A2) as a result of e.g. the melt- concrete interaction also leads to a "Slow Pressurization" of the containment. The needed information and the available information sources are depicted in table II. Information about the distribution of the concentration of noncondensable gases inside the containment, the containment pressure and the steam concentration is necessary to detect the *Mechanism*. The steam concentration inside the different rooms of the containment is an important parameter.If there is more than 50% steam inside the containment then the hydrogen is not burnable because of the steam inertisation.

More information about the core melt process and the RPV-integrity during the different accident phases could be helpfull to detect the mechanism without doubt.

The needed information to use the *Strategies* "Recombination" or "Ignition" is discussed in the following part. In most cases "Recombination" will be a passive measure and therefore no information is necessary to initiate the measure. The needed information to initiate the ignition (global or local if possible) is the concentration of H_2, O_2 and steam inside the containment, the time-dependent temperature and pressure inside the containment and the status of the melt-concrete interaction. In most cases the initiation of the ignition automatically occure. The needed information to control the effectiveness of the measures is nearly the same as before: the concentration of H_2 and the pressure and temperature inside the containment.

b) General Requirements

To prevent early and late containment failure by hydrogen burning, the mitigation measures have to fulfill the following points [7]:

- Exclude large-scale detonation or a highly turbulent deflagration with the potential to reach failure-pressure of the containment.
- Prevention of local detonations which could lead to missile-generation.
- Prevention of high local hydrogen concentration.
- Mitigate the consequences of local, multiple deflagration, leading to high temperatures (failure of local equipment).

The determination of the most effective position inside the containment to install e.g. recombiners or/and ignitors is also a task for the near future. Another task is the design of recombiner devices against possible poisoning and aggressive media.

3.4 Criteria to assess the Cabability of Instruments

In our investigation the most important instruments installed in the investigated plant and the qualification range and other features were determined. The required safety-related and wide-range measurements for PWR according to KTA-3502 rules (Nuclear Safety Standards Commission [9]) and their measurement range are presented in table III. These required instruments include also the H_2-concentration and the containment pressure and temperature measurement.

Typical instrument systems consist of transducers, cables, electronics and other components, and it is not easy to determine all possible failure conditions. All safety-related and wide-range instrument systems were tested under specific environmental conditions depending upon their position inside the containment, outside in the annular space or in the valve compartment. Figure 2 shows examplarily the pressure, temperature and time-dependent test parameters for instruments positioned inside the containment [10].

For the assessment of the capability of the instruments we use at the moment four simplified criteria. The criteria are not totally equal to the criteria used in the literature [5].

1. Instrument performance will be degraded if the system is operated outside the range.

2. For instruments located in the primary circuit, the evaluation is focused on sensors because of the temperature or pressure conditions to which these sensors could be exposed during a severe accident.

3. For instruments located in the containment the possible strong environmental conditions, e.g. during hydrogen burning influencing the whole instrument system have to be considered.

4. The availability of electrical power supply if needed.

These simplified criteria reflect the principle behavior of the instruments under accident conditions. More detailed investigations about the capability of instrument systems especially in the case of longterm high radioactivity loads will be a task for the near future. Together with the improvement of the criteria the quality and accuracy of codes used to calculate the possible accidents in NPP's have to assessed .

A PWR accident sequence with total loss of feedwater with primary bleed as an accident management measure, the so called ND*-accident, was selected to demonstrate the application of the method [2, 8].The calculated accident parameters (see table IV) are used to assess the loads to the instruments.

3.5 Assessment of Instruments needed for Venting and H_2-Limitation Measures

a) Pressure Measurement

The measurement range as shown in table III is much higher than the failure pressure of the containment and therefore the pressure criteria to initiate filtered venting should be available. The expected parameters inside the containment during the accident are lower than the qualification limit (KTA-3505 rule [10]) of the investigated instrument if no hydrogen burns. Further investigations are necessary to assess the influence of local hydrogen burning.

b) Temperature Measurement

The calculated parameters of the DBA's are the basis for the determined measurement range for the temperature measurement inside the containment dome required in KTA-rules (20 ... 160 °C) as well as the range of the installed instruments of the reference station (20 ... 200 °C). The temperatures inside the containment calculated for the ND*-accident lies within the range of the installed instruments if no hydrogen burns. In this case the temperature criteria to initiate filtered venting will be supplied.

Any more investigations are necessary to assess the influence of global or local hydrogen burning processes, the temperature increasing in this case and the influence of heat production and temperature increasing if catalytic devices will be used to limit the hydrogen concentration. For example such questions as how much time does a hydrogen burning process need and which temperatures are expected are of interest to assess the availability of instruments much better.

c) Measurement of H_2-concentration

The installed H_2-diffusion instruments inside the containment of the reference plant can measure the H_2-concentration at eight different points at the same time continuously with a measurement range up to 10 Vol.%. The measurement principle of these instruments is the principle of temper color. The measurement system consists of a detector which recombines the hydrogen together with oxygen. The small amount of heat resulting of this process will be detected with a temperature-dependent resistor and a comparable temperature. It means that oxygen as a compound of the atmosphere is absolutely necessary for the measurement of H_2 with this method.

Further investigations are necessary e.g. to assess the influence of hydrogen combustion to the oxygen concentration of the air and therefore the normal function of the measurement system.

The qualification conditions for this H_2-measurement instrument are the same as for safety-related instruments installed inside the containment. Therefore the expected environmental conditions do not challenge the function of this instrument if no hydrogen burns.

If there are no measures available to limit the H_2-concentration or if the installed measures are ineffective then the measurement range of this instrument is too small to detect the expected H_2-concentration during severe accidents.

In the case of a leak in the containment shell than it is possible that the hydrogen leaves the containment. In this case the H_2 concentration inside the annulus, the room between primary and secondary containment, increases and there are no instruments installed up to now to detect hydrogen nor are any measures prepared to limit the concentration.

4 Conclusion

The presented method to determine the information needs and to assess the availability of instruments could be successfully used for PWR's as shown in our example. This paper gives only a short overview of the work being done in the field of accident mitigation in relation to "Prevention of Containment Failure". The developed *Safety Objective Tree* comprises all possible *Mechanisms* occuring during different severe accidents in our opinion. The other trees for the other accident phases or *Safety Objectives* will be completed and discussed in the future.

The assessment of the capability of the existing instrumentation shows that further investigations are necessary. It is not so easy to get information about the situation in different rooms inside-the containment e.g. about steam inerting or about H_2 concentration greater than 10 Vol.%. Higher concentrations are possible if severe accidents occure .

Further investigations are also necessary, e.g. to obtain information about the influence of local or global hydrogen burning.

As a result of the investigations demonstrated at the last CSNI meeting at Rome [8] it is necessary to get more information about:

- the status of the core during severe accidents,
- the location and relocation of material,
- the water inventory of the lower plenum,
- the integrity of the reactor pressure vessel and
- the melt / concrete interaction in an adequate manner.

Another big problem is the correct assessment of the function of instruments, e.g. the asumption that instruments, if they are working outside their qualification or measurement range for only a short period, will be degraded.

The extension of the investigation to a broader spectrum of events is necessary and can be performed in the future using the knowledge base of the existing PSA and other studies.

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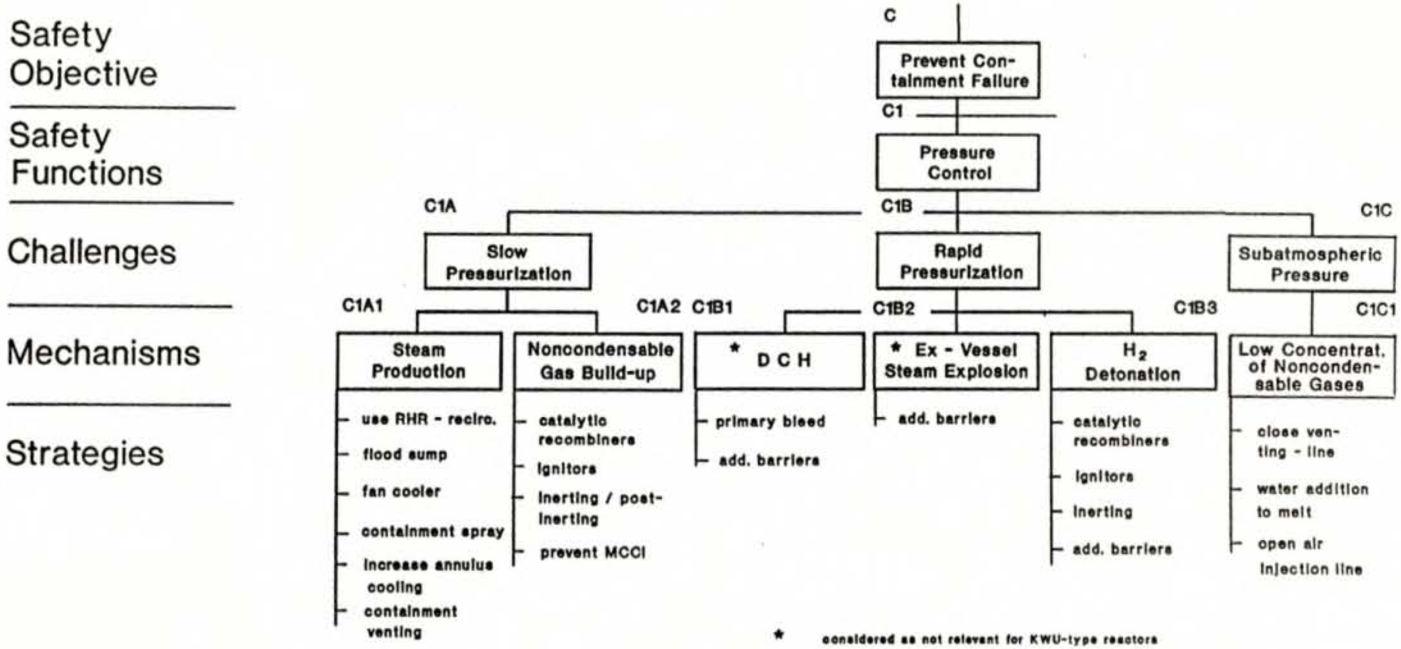

Figure 1a: Safety Objective Tree of PWR's for Accident Mitigation (part 1)

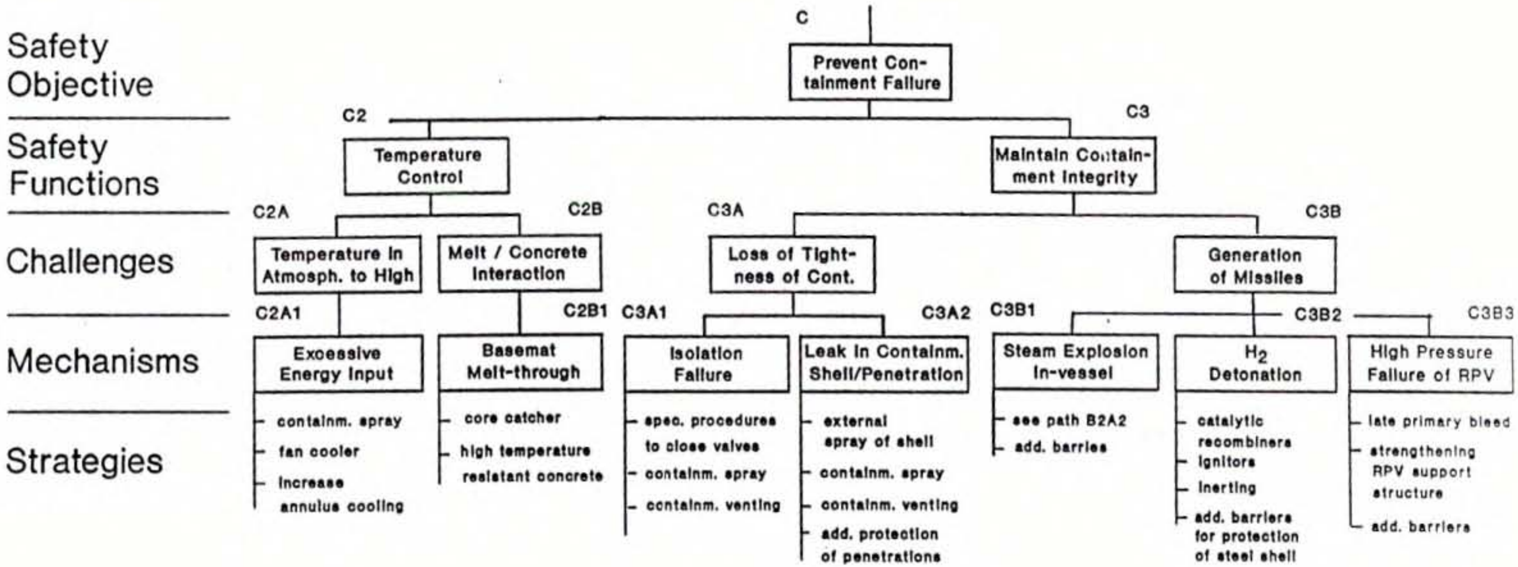

Figure 1b: Safety Objective Tree of PWR s for Accident Mitigation (part 2)

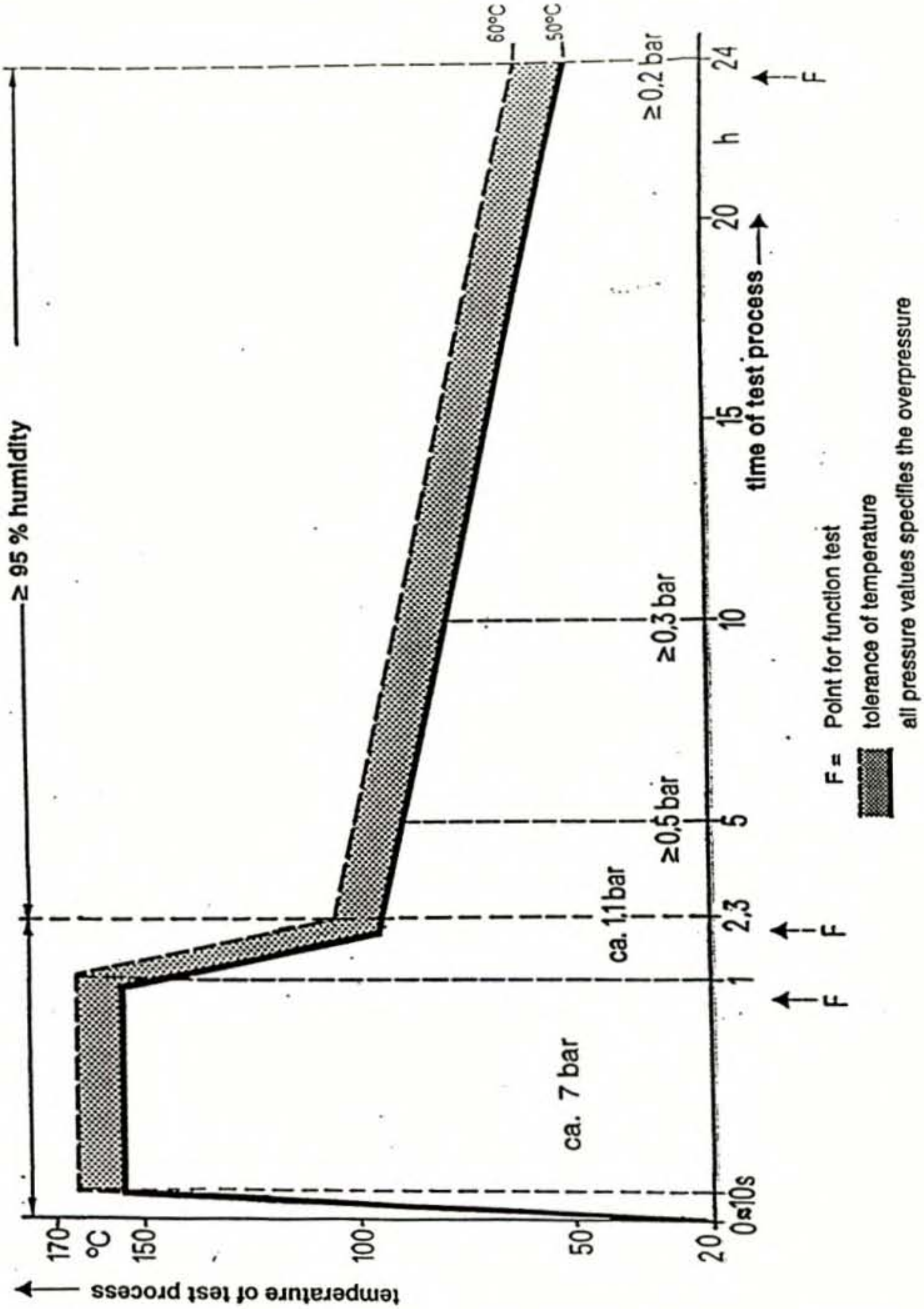

Figure 2: Time-dependent parameters for tests of instruments positioned inside the containment (KTA 3505, fig. 5-2)

Table I Information Needs for the Safety Objective - "Prevention of Containment Failure" (C)

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
<u>Challenge</u> "Slow Pressurization" (C1A)	time-dependent pressure in containment	pressure / diff. pressure containment / atmosphere	none	pressure / diff. pressure measurement	
<u>Mechanism</u> "Steam Production" (C1A1) (insufficient heat removal from containment)	<u>Indicator</u> relation of heat production and heat removal	none	time-dependent pressure in containment	pressure / diff. pressure measurement	
			time-dependent temperature in containment and sump	temperature measurement in containment dome and sump	
	steam concentration in containment atmosphere	steam concentration in containment atmosphere			steam concentration measurement in containment atmosphere
	contact melt with containment sump water	none	progress of sump water level	sump water level measurement	
			time-dependent steam concentration in containment atmosphere		steam concentration measurement

Cont. Table I

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
6. Strategy filtered containment venting	Selection Criteria time-dependent pressure in containment	time-dependent pressure in containment		pressure / diff. pressure measurement	
- requires nonburnable gasmixtures	time-dependent temperature in containment and sump	time-dependent temperature in containment and sump		temperature measurement in containment dom and sump	
- requires instumentation to detect e.g. H_2	availability of venting system system			parameter in the system	
	H_2-.concentration inside containment	H_2-concentration inside containment		H_2-concentration measurement	
	CO-concentration inside containment	CO-concentration inside containment			CO-concentration measurement
	fission product inventory in atmosphere	fission product inventory in atmosphere		sampling system	
	tightness of containment	tightness of penetrations	fission product inventory in outer containm.	activity / emission via stack	leak detection system
	Strategy Effectiveness time-dependent pressure in containment	time-dependent pressure in containment		pressure / diff. pressure measurement	
	time-dependent temperature in containment and sump	time-dependent temperature in containment and sump		temperature measurement in containment dom and sump	

Table II Information Needs for the Safety Objective - "Prevention of Containment Failure" (C)

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
Challenge "Slow Pressurization" (C1A)	time-dependent pressure in containment	pressure / diff. pressure containment / atmosphere	none	pressure / diff. pressure measurement	
Mechanism "Noncondensable Gas Build-up" (C1A2) (core melt process, melt concrete interaction)	Indicator H_2-concentration inside containment	H_2-concentration inside containment		H_2-concentration measurement	
	concentration of other gases inside containment	concentration of other gases inside containment			concentration of other gases inside containment
	steam concentration in containment atmosphere	steam concentration in containment atmosphere			steam concentration measurement in containment atmosphere
	time-dependent pressure in containment	time-dependent pressure in containment		pressure / diff. pressure measurement	
	status of melt-concrete interaction		temperature distibution in basemat		temperature measurement in basemat
	precursor informations about the status of the core or RPV-failure		coolant inventory in primary circuit	RPV-level probe (level in upper plenum)	
			core relocation status		radiation field outside RPV
			integrity of RPV	primary pressure	acoustic monitor

RPV - Reactor Pressure Vessel

Cont. Table II

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
1.Strategy catalytical recombination of H_2 - passive device - actually not installed in reference plant	Selection Criteria Most of the systems are passive system, therefore no information is necessary to initiate the system.				
	Strategy Effectiveness H_2-concentration inside containment	H_2-concentration inside containment		H_2-concentration measurement	
	time-dependent pressure in containment	time-dependent pressure in containment		pressure / diff. pressure measurement	

Cont. Table II

	Information Needs	Direct Information Source	Indirect Information Source	Available Instruments	Potential Instruments
<u>2.Strategy</u> catalytic or battery-powered ignitors - actually not installed in reference plant	<u>Selection Criteria</u> ignitability of the gas mixture inside containment or different rooms	H_2-concentration inside containment		H_2-concentration measurement	
		steam concentration inside containment			steam concentration inside containment
	time-dependent pressure in containment	time-dependent pressure in containment		pressure / diff. pressure measurement	
	time-dependent temperature in containment and sump	time-dependent temperature in containment and sump		temperature measurement in containment dome and sump	temperature measurement in rooms with ignitors
	availability of ignitors				
	<u>Strategy Effectiveness</u> H_2 concentration inside containment	H_2 concentration inside containment		H_2 concentration measurement	
	time-dependent pressure in containment	time-dependent pressure in containment		pressure / diff. pressure measurement	
	time-dependent temperature in containment and sump	time-dependent temperature in containment and sump		temperature measurement in containment dome and sump	temperature measurement in rooms with ignitors

Table III: Overview of Required Accident Measurements for PWR (KTA-3502 [9])

Nr.	Parameter	Safety Related	Wide Range
1	neutronflux	10^{-6} P_N to 10^{-3} P_N	-
2	boron concentration of sumpwater	50 to 2600 ppm	-
3	coolant temperature in loops	50 to 400 °C	-
4	core outlet temperature	100 to 1000 °C	100 to 1000 °C
5	level in pressurizer	1,8 to 11,4 m *)	-
6	level in SG	1,87 to 14,53 m *)	-
7	temperature of sumpwater	10 to 150 °C	-
8	level in containment sumpwater	1,8 to 3,9 m *)	1,8 to 3,9 m *)
9	subcooling	50 to 0 K	-
10	temperature of water in fuel pool	10 to 150 °C	-
11	pressure in reactor coolant system	1 to 250 bar	1 to 400 bar
12	pressure of SG-secondary	1 to 150 bar	-
13	pressure in containment (Δp)	-0,5 to 5,5 bar	-1 to 15 bar
14	pressure in annulus (Δp)	-0,5 to 5,5 bar	-
15	H_2-concentration in containment	0 to 4 Vol.-%	0 to 10 Vol.-%*)
16	temperature in containment dome	20 to 160 °C	-
17	dose rate in containment	10^{-1} to 10^{6} R/h	10^{-1} to 10^{7} R/h
18	emission via stack	10^{-7} to 10^{2} Gy/h	-
19	emission with waste water	$2x10^{-3}$ to $1x10^{9}$ Gy/h	-
20	level in fuel element pool	16,7 to 21,7 m *)	0 m to maximum level

*) range of the instrumentation in the reference plant (1300 MW PWR)

Table IV: Values of key parameters for the ND*-accident - total loss of SG-feed water supply with primary bleed

Value	1. Phase up to core uncovery	2. Phase up to core slump	3. Phase up to lower head failure	4. Phase ex-vessel
time period	0 ... 285 min	285 ... 360 min	360 ... 410 min	410 ... min
<u>Primary Circuit:</u> average core temperature	< 600 K	2200 K		
max. core temperature	< 850 K	2700 K		
gastemp. of upper plenum	< 600 K	< 1500 K	< 1000 K	< 1000 K
max. pressure of reactor	16,3 MPa	5 MPa	5 MPa	1 MPa
min. pressure of reactor	~ 1 MPa	~ 1 MPa	(0,1 MPa)	0,1 MPa
average H_2-mass inside the primary circuit	0 kg	0 ... 500 kg	500 kg	
<u>Containment</u> average temperature (with contact melt/sump water)	30 ... 110 °C	110 °C	110 ... 120 °C	... 150 °C
average temperature with continous H_2-burning	30 ... 110 °C	110 ... 350°C	120 ... 350 °C	< 160 °C
average pressure (with contact melt/sump water)	0,1 ... 0,25 MPa	0,25 ... 0,35 MPa	0,35 MPa	0,35 ... > 0,6 MPa
average H_2-mass inside the containment	0 kg	> 0 kg	> 0 kg ... 500 kg	< 1350 kg

INSTRUMENTATION NEEDS AND DATA MANAGEMENT BY THE FRENCH PROTECTION AND NUCLEAR SAFETY INSTITUTE FOR THE DIAGNOSIS AND PROGNOSIS OF THE RELEASE DURING AN EMERGENCY ON A PWR.

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Abstract

IPSN in conjunction with EDF has been developing for the last few years an approach for the diagnosis and prognosis of the Source Term during an accident on a PWR. Intended for the off-site emergency teams, this methodology is implemented with dedicated manual and computerized tools within the frame of the SESAME project.

It is necessary to have access during the accident to various information dealing with the state of the plant. These information needs and the various means available to pick up data from the plant are described in this paper.

Emphasis is given on the analysis of data that is needed to avoid any failure in the assessment of the state of the safety barriers and functions. This analysis deals with :

the quality of the information depending on the environmental conditions and on the availability of the supply systems,

the cross-check between measurements of same type,

the cross-check between measurements of different types.

OECD(NEA) CSNI specialist meeting on instrumentation to manage severe accidents. Cologne, Germany, 16th-17th March 1992.

Introduction

The instrumentation needs in case of accident on a PWR are twofold. They cover on the one hand the accident management by the operating team and, on the other hand, the follow-up of the accident by the national crisis teams. After recalling the tasks and resources of the IPSN Emergency Technical Center (ETC), this paper reviews the information necessary to the crisis teams.

The measurements - existing and to be developed - essential to provide a diagnosis and a prognosis of the state of the installation are described.

Finally, the needs in matters of management of the data provided by instrumentation are defined, as regards their acquisition, their organization, their control and their use within the IPSN ETC.

1.INSTRUMENTATION NEEDS ANALYSIS FOR THE CRISIS TEAMS OF THE FRENCH PROTECTION AND NUCLEAR SAFETY INSTITUTE.

1.1. The national emergency organization.

In case of accident in a nuclear installation, it is necessary to evaluate the situation and in particular to forecast its possible consequences in terms of release into the environment. In France, this information is elaborated by a national emergency organization and provided to the local government representative (the head of the Prefecture) who takes it into account in order to implement the decisions concerning the protection of the population.

The national emergency organization consists mainly of a decision-making level (the Emergency Managing Centers) and a reflection level (the Emergency Technical Centers of the utility - one located in the plant, the other in the Paris area - and of the IPSN).

The IPSN Emergency Technical Center (ETC) is organized round a management unit receiving analysis data from two working parties, one studying the situation within the damaged plant (Plant Assessment Unit) and the other concerned with assessing the radiological consequences of the accident (Radiological Consequence Unit).

1.2. The Plant Assessment Unit.

This paper focusses on the instrumentation necessary for the work performed by the Plant Assessment Unit. During an emergency, the experts working in this Unit have to face the challenge of making, in real time, an operational synthesis of the available information. In particular, they have to make the discrenpancy between essential and subordinate information, detect the errors and raise the judicious questions at the right time. Finally, their synthesis aims at providing a diagnosis and a prognosis of the situation.

Although this synthesis is elaborated by the IPSN ETC on its own side, it is periodically confronted through a phone conference network, with the diagnosis and the prognosis performed by the ETCs of the utility. In order to structure the dialogue between the three ETCs, a think grid has been jointly designed (fig. 1). According to that grid, the surveyed items are : the physical state of the safety barriers (fig.2), the availability of the safety systems and the margins to critical states.

Various means are available to pick up the information needed for filling up the think grid :

- the terminals existing in the plant are duplicated in the ETC. These tools allow the access to logic and analogic signals available for the operators. This information is structured within synthesis images describing the state of the plant. These data are transmitted through the French national network TRANSPAC.

- specific messages are used in case of unavailability of the plant computer. In this case, the information is transmitted to the ETCs by fax or through phone conference.

These data being obtained are structured in a pre-formated think grid designed with the national operator. This message describes the state and the evolution of the safety barriers, of the safety functions and of the safety-related systems. The answers to the quantitative questions are calculated with manual or computerized tools developed within the frame of the SESAME project (fig.3).

2. NECESSARY MEASUREMENTS.

2.1. Diagnosis of the plant.

The diagnosis provides information on the state of the three safety barriers. It gives also the state of each safety function and associated systems. Fission product releases out of the fuel, activity suspended into the containment, leak path and releases in the environment are thus estimated.

2.1.1. First barrier.

The state of the first barrier, the fuel, is determined by two measurements :

- the core exit temperature,

- the dose rate in the containment.

The subcriticality safety function is followed up using the intermediate nuclear measuring channels.

The water-inventory safety function is followed up using the reactor vessel, containment sump and pressurizer levels with the saturation margin.

2.1.2. Second barrier.

The state of the second barrier, the primary circuit, is determined by two measurements :

- the containment pressure, the saturation margin, a primary circuit mass balance and the dose rate in the containment for a primary break within the containment,

- the same parameters associated with the activity measurements in the surrounding buildings for a break out of the containment on a connected circuit,

- the activity and steam generators blowdown measurements associated with their levels for a steam generator tube rupture.

The heat removal safety function is followed up using the temperature and the pressure of the primary circuit.

2.1.3. Third barrier.

The state of the third barrier, the containment, is determined by the follow-up of :

- position switch on the isolating valves of the mechanical penetrations which are useful to detect a leak path,

- the activity through the stack, in the auxiliary sumps and buildings

  for the assessment of the containment integrity,

- the state of the steam generator and condenser steam dump valves

- the damaged steam generator pressure and the mass balance

  for an accident with steam generator tube rupture and steam pipe rupture.

The associated function, the confinement, is followed up by the containment building pressure and an activity balance between the containment and the environment.

2.1.4. Quantifications.

The filling-up of the synthesis grid needs the quantification of various parameters or phenomena. This is done with correlations or computerized tools using simplified models.

The **BRECHEMETRE** software is dedicated to the assessment of the primary break size by means of a mass balance based on the pressurizer level and the input and output flow rates. The water inventory is compared with critical flow rate correlations to assess the break size dealing with this thermohydraulic environment.

The criticality margin is assessed with the boron concentration measurement and the thermohydraulic characteristics of the coolant by means of the **CRAC** software. A mass and concentration balance is used in case of unavailability of the boron direct measurement to determine the anti-reactivity associated with the burnable poison.

The clad rupture or core meltdown fraction is estimated with the exit core temperature or the containment dose rate. This is done using correlations introduced in the **SINBAD** software.

The containment leak rate is assessed from a leak size and the containment pressure using the **SINBAD** software. The **ALIBABA** expert system provides an early diagnosis of containment leakage using the instrumentation associated with the containment isolation and the activity measurements.

The containment building pressure and temperature measurements associated with particular assumptions such as steam saturation or core meltdown fraction allow the hydrogen risk assessment using the **HYDROMEL** software.

The release into the environment during an accident without containment bypass is quantified in the **PERSAN** software. It is obtained from other softwares results and particular information such as the use of the spray system or of the sand-bed filter.

The release into the environment during an accident with containment bypass and without core damage is quantified in the **RTGV** software using simplified models of thermohydraulic and of fission product transfers.

2.2. Prognosis on the accident progression.

The prognosis on the state of the plant deals with the state of the safety functions and with the availability of the associated functions. The necessary information can be provided directly by the instrumentation such as it is the case for reliable or foreseeable losses due to defaults on the supply systems. The information can also be picked up through the control team for the procedures to be followed or through the on site operating teams for the repair or substitution of devices.

Some quantifications are necessary as for the diagnosis.

The **SCHEHERASADE** software deals with the delay before core uncovery by means of a mass and energy balance. The necessary information is the thermohydraulic environment and the input and output flow rates in the primary and secondary circuits.

The predicted time between core uncovery and clad rupture or core meltdown is calculated with the residual power using the **SINBAD** software.

The prognosis on the releases calculated with the **PERSAN** software does not need additional information from the instrumentation.

2.3. Measurements to be developed.

Some limitations exist in the available instrumentation dealing with the implementation of the diagnosis-prognosis method :

- the dose rate measurement in the reactor building does not allow an accurate assessment of the activity of the various families suspended in the containment.

- the direct hydrogen measurement is not available in all foreseeable conditions.

- finally, the containment leak is not directly measured.

Specific phenomena dealing with the accident progression in case of core meltdown such as core slump, bottom head failure and corium-concrete interaction cannot be followed up by an appropriate instrumentation.

3. DATA MANAGEMENT.

3.1. Data collection.

A· described under 1.2., various means are available in order to catch the required information. Manual and automatic tools have to be separately mentioned :

- the local treatments allow the storage of 100 measurements important for safety every minute. When they are available, these data allow the diagnosis of the plant state (mainly safety barriers and functions). It is automaticaly reassessed with the reception of a new set of values. The advantage of this device is the exhaustivity while its main disadvantage is the leak of reliability which demands the results validation by an expert. The pre-defined messages sent by fax are used in place of this tool while the plant computers are unavailable.

- the other means allow to pick up more precise but less general information which deals with specific questions.

3.2. Data organization.

The set of data necessary for the implementation of the diagnosis-prognosis method in the IPSN ETC is structured in three main parts :

- the data dealing with a standardized plant series. They describe the buildings and main equipment geometrical characteristics ; these basic data are not modified during the crisis,

- the data dealing with the initial conditions of the unit such as power history, fuel features and various information on particular aspects of the plant before the accident,

- the data changing during the accident, mainly the thermohydraulic data of the primary and secondary circuits and of the reactor building. They are re-assessed automatically when local treatments are available or manually when periodic messages sent by fax are used.

The use of this organization allows the crisis team to follow up the accident in a better way by taking into account the new values more quickly.

3.3. Data control.

Each data has to be checked in order to assess its validity. The reception of an alphanumerical data means that the sensor, the transductor and the processing functions are available with their supply systems. If not, a specific code is sent meaning that this measurement is unavailable. The sensor qualification, its physical environment and electrical supply are described. In case of availability, the value is compared with its up and down limits.

The consistency between the measurements of the redundant chains or of the same type of sensors is checked. Such controls are :

- the cold and hot leg, core exit and between legs temperatures ,

- the pressurizer, reactor vessel, refueling water storage tank and sump levels,

- the various activity measurements.

Finally, the more sophisticated control deals with the cross-check of measurements of different type. Such controls are :

- exit core temperature with containment dose rate,
- reactor vessel level and saturation margin,
- refueling water storage tank level and safety injection flow rate,
- reactor vessel and pressurizer levels with the temperature before the pressurizer valves,
- boron concentration and intermediate nuclear measuring channel.

This analysis allows to re-assess a data or to detect a particular event such as a bubble at the top of the reactor vessel or a primary breach through the pressurizer valves.

This set of control is structured by means of rules dealing with periodically received values.

3.4. Data processing.

The collected, structured and controlled data are used in different ways :

- position and displacement in the state grid,
- use in the various SESAME softwares,
- filling-up of the think grid.

This data management is semi-automatic. Each diagnosis proposal has to be validated by the crisis team of the Plant Assessment Unit.

Conclusion

The present direction of the IPSN tasks in matters of accidental instrumentation deals with a more efficient follow-up of a severe accident by the crisis teams. However, the instrumentation is assessed to be fully available under the qualification conditions and unavailable beyond. The next step will deal with the behaviour and response beyond these environmental conditions and with the complementary instrumentation needs.

REFERENCES.

/1/ C.CHAULIAC, L.JANOT, A.JOUZIER, B.RAGUE. Diagnosis and prognosis of the source-term by the French Safety Institute during an emergency on a PWR. (OECD/NEA specialist meeting on severe accident management program development. September 1991).

FIGURE 1 THINK GRID

	DIAGNOSIS			PROGNOSIS	
	SYSTEM AVAILABILITY	STATE OF SAFETY FUNCTIONS	STATE OF SAFETY BARRIERS	SYSTEM AVAILABILITY	STATE OF SAFETY FUNCTIONS
SUB-CRITICALITY			FUEL		
RESIDUAL POWER REMOVAL			RCS		
CONFINEMENT			CONTAINMENT		

FIGURE 2: PHYSICAL STATE OF THE SAFETY BARRIERS

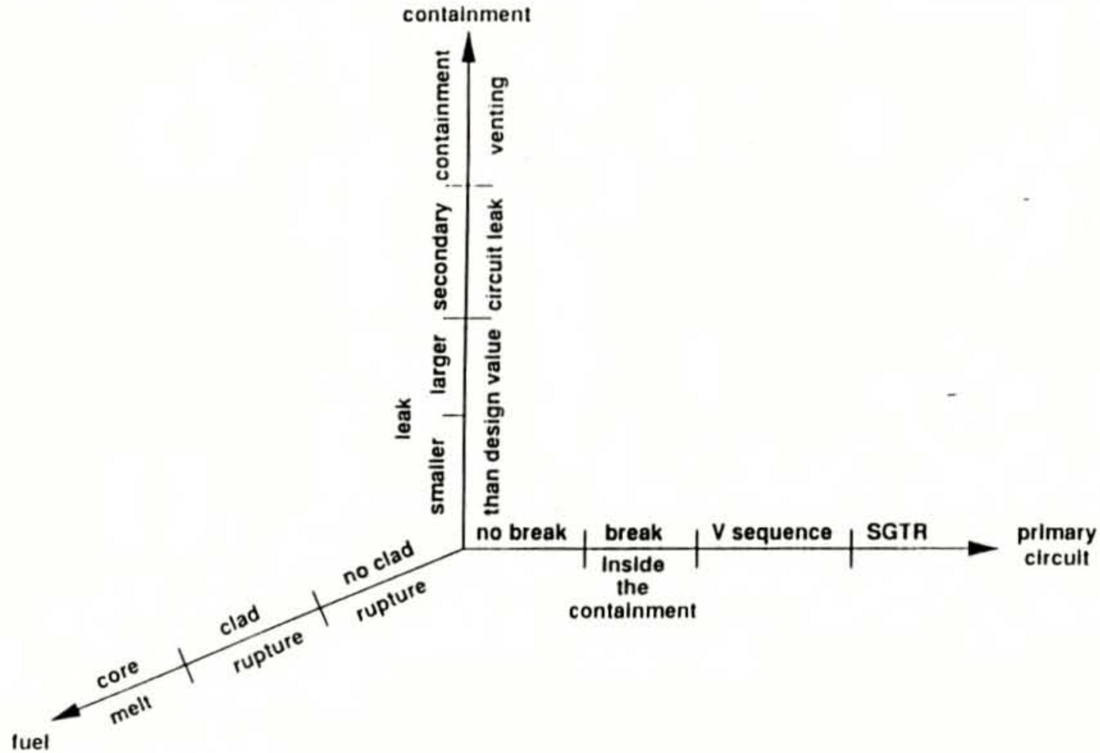
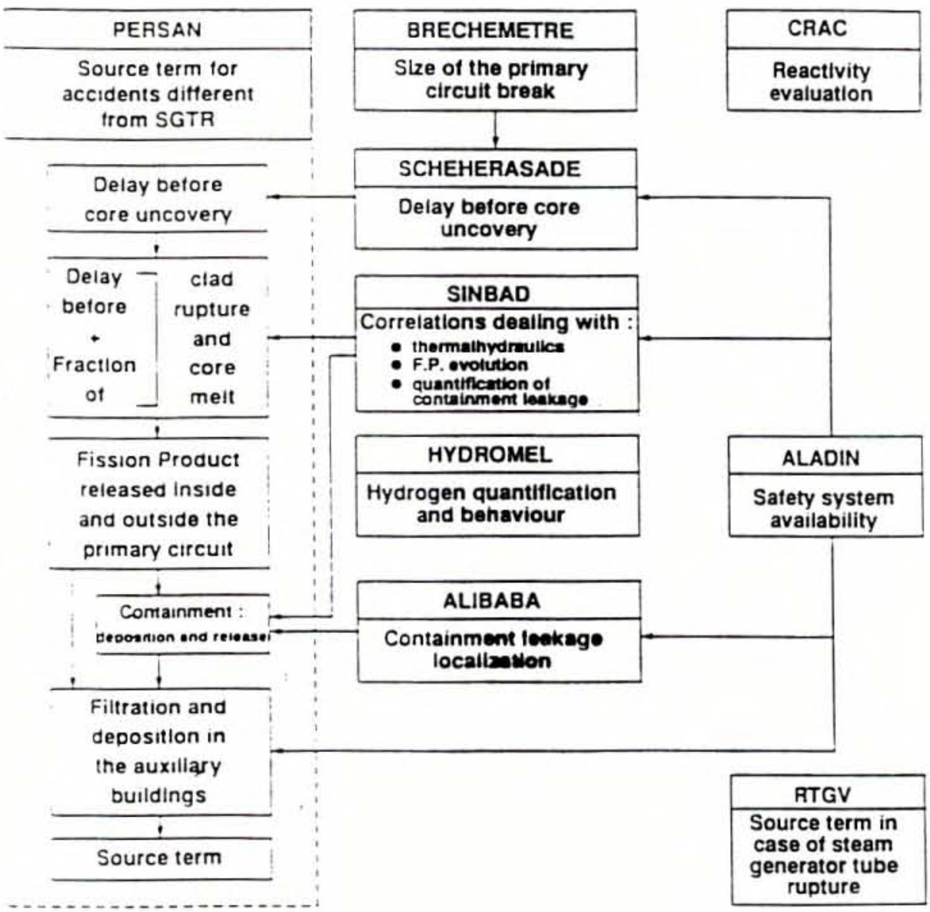


FIGURE 3

THE SESAME PROJECT

ORGANIZATION OF THE SOFTWARES USED BY THE PLANT ASSESSMENT UNIT



Instrumentation and Severe Accident Plant Status Interpretation

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ABSTRACT

An effective accident management program involves identifying those actions necessary to assure that all available utility options are considered and understood so that, in the event of an accident, maximum benefit can be obtained from the effective and timely utilization of such capability. To assure that such capability is effectively utilized, it is appropriate to identify those plant parameters that can (1) be utilized to determine the necessity for various actions to be taken, and (2) provide feedback on the effectiveness of such actions.

Therefore, a critical element of the process for determining the appropriate guidance for operators in any postulated accident conditions involves the identification of available plant information sources and the degree to which plant instrumentation can be utilized to determine plant functional status.

A report by an NEA Group of Experts states: "In the face of the specific loads and requirements imposed during severe accident sequences the existing instrumentation may not be adequate and may have to be improved and perhaps supplemented."[†] A report[††] prepared by INEL for the U.S. NRC mentions several examples of existing instrumentation that may not be available under certain circumstances.

The Electric Power Research Institute is conducting a project related to instrumentation and severe accident plant status interpretation, which may provide a balance to the views expressed in the two reports cited above. The project will recognize the facts that (i) instrument responses during severe accidents do not need to be as accurate as during normal operation, and (ii) not all instrument loops will see a severe environment. In particular, the proposed work is to provide technology to get the most information from the existing instrumentation under severe accident conditions by developing (1) calculational aids to determine actual plant parameters based on severe-accident-affected instrument readings, and (2) means to utilize indications from operational instruments to infer parameters values for failed instruments, or where no instrument may exist.

[†] "The Role of Nuclear Reactor Containment in Severe Accidents", Report by an NEA Group of Experts, April 1989, NEA-OECD

[††] "Accident Management Information Needs", NUREG/CR-5513, April 1990, INEL

Specific deliverables for this project are (i) an instrumentation data base that will include both instrumentation failures and successes under severe conditions, and contain instrument performance information from both nuclear and non-nuclear industry situations; (ii) methods to assess the validity of instrument signals and estimate the performance of individual instrument loops; and (iii) calculational aids to estimate and interpret instrument readings under severe accident conditions, including the ability to extrapolate readings from functioning instruments to locations where instruments have failed.

INTRODUCTION

A key to achieving regulatory closure of the severe accident issue, for operating nuclear power plants, is the commitment by nuclear utilities to assess and enhance their existing accident management capabilities. A number of activities are currently underway that will assist utilities in their understanding of where current accident-management provisions can be enhanced. Currently, utility owner's groups are developing generic severe accident management guidance (SAMGs) specific to each of the major Nuclear Steam Supply System (NSSS) designs. These SAMGs will identify and develop accident management strategies in response to expected conditions associated with severe accidents.

A vital link in the accident management process is successful interpretation of the plant instruments' response to the accident. Previous work[1] has used Regulatory Guide 1.97 for providing a boundary for the adequacy of existing plant instrumentation for severe accident management applications. The Electric Power Research Institute (EPRI) is currently conducting a project to develop techniques to interpret instrument signals during a severe accident independent of the artificial, Regulatory Guide 1.97 limitations. This project builds on the existing work in this area by examining TMI-2 instrument response, instrument performance limits, and degraded performance more closely.

The ultimate results of this project will provide an approach for interpreting instrument responses during severe accidents and for developing methodologies, including calculational aids, to understand instrument responses during severe accidents.

[1] Accident Management Information Needs, NUREG/CR-5513, April 1990, Idaho National Engineering Laboratory

PROJECT APPROACH

The approach chosen for this project is being applied at two specific U.S. nuclear plants: A General Electric BWR (Mark II Containment) and a Westinghouse PWR. The project is being performed in the following steps:

- Identify severe accident conditions for which mitigating accident management strategies may be desired.
- Define information to arrive at plant status which allows anticipation and/or identification of associated severe accident conditions.
- Identify selected specific instruments that could fulfill this role.
- Develop an understanding of how identified instruments perform under degraded conditions and when they fail.
- Develop calculational aids where necessary to correlate measured data and/or trends of instruments to parameters of interest.
- Develop matrices relating plant status and plant status trends with associated accident conditions and relating instrument availability/performance.

Identification of Severe Accident Conditions

The first step is identifying conditions for which mitigating actions should be taken. Conditions correspond to different phases of different accident types. A key reference source for identification of these conditions is the "Accident Management Guidance Technical Basis Report"[2] . This report discusses, in detail, basic phases of severe accidents with respect to core status and containment status; and it also discusses the effects of a range of mitigating actions that might be taken. Examples of important accident conditions include core unrecovery, cladding oxidation, fuel melting, and vessel failure.

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[2] EPRI report, in preparation

Identification of Parameters of Interest

Successful anticipation and/or identification of a plant condition or plant status requires definition of criteria which define the existence of that condition or presage the onset of that condition. Qualitatively, criteria are parameters such as core temperature, vessel water level, containment pressure, RCS system pressure, etc. The second step of the project is, therefore, to identify parameters of interest that allow anticipation and/or identification of severe accident conditions, and provide information with regard to response of the plant to the severe accident strategies that are being implemented by plant staff. It is important to note that less stringent instrument accuracies may often be sufficient to fulfil operational needs during a severe accident. In some cases, instrument/parameter trends may be adequate. Selected cases will be pursued.

Identification of Instruments

Based upon the parameters of interest identified in the prior task, instruments are identified that fulfill those needs under severe accident conditions. Diverse instrument sources will be considered to provide some redundancy in fulfilling a given operational goal. Plant drawings, equipment lists, and other sources are used to identify potentially useful instruments. Where possible, instruments that can directly measure a key parameter of interest are identified (e.g., source range monitors or cavity temperature sensors). Instruments that measure a secondary effect which can be correlated to a key parameter (via a calculational aid) are also identified (e.g., process radiation monitor which can correlate to core damage status). Both non-safety and safety related instruments are considered. The project is taking into consideration the fact that many instruments can provide useful data over a wider range of conditions than those over which the instrument is required to perform.

Characterization of Instrument Performance

A selection of instruments identified as potentially useful in a severe accident will be chosen, and, then, investigated to determine under what conditions the instrument's performance begins to degrade and ultimately when the instrument fails. This determination involves review of vendor test data, vendor contact, industry studies on instrument survivability, and other sources. The project is attempting to characterize degraded instrument performance in terms of decreasing accuracy and the ability of the instrument to continue to trend the measured variable. Support system failure is considered in assessing overall instrument failure, and can also be used for supplying information about the accident.

Calculational Aids

A very important step in this project is development of calculational aids which permit determining the value for, or trend of, a key parameter by correlation with another monitored parameter. These "aids" consist of analyses which calculate the relationship between two plant parameters, e.g., pressurizer water level and core water level. These calculational aids help to quantitatively establish criteria for when severe accident management mitigating actions should be taken. They also enable the interpretation of a wider range of instruments for severe accident management in a timely manner.

Project Results

The project will produce matrices which relate information sources to their associated severe accident conditions and relate specific instruments and instrument performance to information sources. The format of the information source/accident conditions matrix is shown in Fig. 1, while the format for the instrument/information source is shown in Fig. 2.

Where possible, a given parameter in the information source/accident conditions matrix will include an accuracy judgement. Also, the same parameter may appear more than once in the information source/instrument matrix, if it can be derived from different instruments.

The project started in November 1991 and will be completed by the end of 1992. Periodic review meetings with operations and engineering staff from the participating plants will be held to assure the practicality and correctness of project results. Similar reviews are planned with staff from a B&W PWR and a

Combustion Engineering PWR to identify the degree to which project results are applicable to those NSSS designs.

Conditions (e.g. uncooled core, vessel failure, etc.)	Oxidizing Core (OX)	Badly Damaged Core (BD)	Core Ex Vessel (EX)	• • •
Severe Accident Type (e.g., LOCA, ISLOCA, ATWS, etc.)				
		Information (e.g. core water level)		

Figure 1

	Instrument ID ➡			
Information Source ⬇				
		Instrument Availability/ Performance Code*		

* Where Codes Are: A = Available
D = Degraded Accuracy (specify)
T = Degraded Trending Only
F = Failed

Figure 2

CONCLUSION

A report by an NEA Group of Experts states: "In the face of the specific loads and requirements imposed during severe accident sequences the existing instrumentation may not be adequate and may have to be improved and perhaps supplemented".[3] This project is intended to support a basis for showing the extent and adequacy of existing instrumentation for use with severe accident management guidance. It is, however, recognized that this project is limited in the scope of the instruments to be examined, and therefore, it is not intended to be drawing generic conclusions about the overall adequacy of instrumentation in today's plants.

[3] "The Role of Nuclear Reactor Containment in Severe Accidents", Report by an NEA Group of Experts, April 1989, NEA-OECD

CONSIDERATIONS ON MONITORING NEEDS OF ADVANCED, PASSIVE SAFETY LIGHT WATER REACTORS FOR SEVERE ACCIDENT MANAGEMENT

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ABSTRACT

This paper deals with problems concerning information and related instrumentation needs for Accident Management (AM), with special emphasis on Severe Accidents (SA) in the new advanced, passive safety Light Water Reactors (PLWR), presently in a development stage.
The passive safety conception adopted in the plants concerned goes parallel with a deeper consideration of SA, that reflects the need of increasing the plant resistance against conditions going beyond traditional "design basis accidents". Further, the role of Accident Management (AM) is still emphasized as last step of the defence in depth concept, in spite of the design efforts aimed to reduce human factor importance; as a consequence, the availability of pertinent information on actual plant conditions remains a necessary premise for performing preplanned actions. This information is essential to assess the evolution of the accident scenarios, to monitor the performances of the safety systems, to evaluate the ultimate challenge to the plant safety, and to implement the emergency operating procedures and the emergency plans. Based on these general purposes, the impact of the new conception on the monitoring structure is discussed, furthermore reference is made to the accident monitoring criteria applied in current plants to evaluate the requirements for possible solutions.

1. Introduction.

This paper discusses some possible improvements in nuclear safety of new advanced, passive safety light water reactor (PLWR), for most of which conditions exist for a better exploitation of their safety potential.
PLWR are intended to achieve improvements in areas such as reliability of safety functions, human factors, occupational doses, environmental impact, emergency preparedness, amplitude of accident sequences spectrum they can face. However, details of design solutions or accidents analysis results are not yet available.
Even if no nuclear plant commissioning is foreseen in the short term in Italy, ENEA/DISP and other public and private italian organizations are devoting efforts in analysing PLWR design proposals and related issues for possible future installations.
In this paper, considerations on new monitoring objectives and requirements arising from new design conceptions, especially in Accident Management (AM) areas, are reported and discussed; AM is considered in a global view, so not only Severe Accident (SA) conditions are investigated.
Obviously, monitoring has a direct interface with AM, because of the information needed for operator actions; an indirect interface comes from the observation that more enhanced monitoring, controls, protection and automation prevent or reduce the frequency of some accident conditions, or (together with improved process design margins) make longer time intervals available for operator interventions.

2. The conception of advanced, passive safety reactors.

The most important objectives and features of PLWR designs (at hardware and software level) can be synthesized as follows:

* plant simplification, both at system and at component level,
* application, as extensive as possible, of the inherent and passive safety principles,
* low accident progression rates (i.e. critical limits of response parameters are delayed with respect to perturbations),
* capability to cope with any considered event for a predetermined "grace period" (typically three days) without reliance on human action,
* improved reliability of the safety functions, with particular reference to the reduction of human factors importance and of the avoidance of technological faults ,
* assured protection against a set of events that includes severe ones,
* simplification of emergency preparedness requirements.

The further objective of the overall safety structure optimization implies the integration of safety principles pertaining functions and systems in a comprehensive set of requirements at plant level. In particular, requirements coming from AM and related monitoring needs should contribute to, and should be impacted by the overall safety design structure.

The PLWR designs considered in this paper put reliance on enhanced prevention of core melt, for an enlarged set of events, and on effective mitigation features in melt-down conditions.
This family of plants is the most interesting from AM viewpoint due to the largest room to mitigative actions in the largest spectrum of plant conditions.
Among the proposed designs that respond to the above characteristics there are, for instance, the Advanced Pressurized Water Reactor (AP-600), developed by Westinghouse Electric Corporation and the Simplified Boiling Water Reactor (SBWR), developed by General Electric.
The above quoted designs consist of medium power (about 600 electric MWatts) reactors and are beeing developed in the United States with the sponsorship of the Department of Energy.

3. The Accident Management applied to Advanced, passive safety LWR - Information needs.

For the present generation NPPs, Accident Management is considered an important step in the defence in depth strategy. Experience has been gained, during the last decade about rules, guides and procedures for Accident Management; many improvements have been implemented to cope with beyond design bases conditions, taking advantage of the margins already existing in their designs.
AM requirements affect important areas of plant design such as supervision, manoeuvre and information means (man-machine interface system).
The completeness, qualification and understandability of the information to the operator have been more and more improved; so, in this area, a consolidated basis of applications is available for future implementation in advanced, passive safety designs.
Nevertheless, some adjustments are needed to take into account the peculiarities of new designs. A special mention has to be devoted to the reduction of human factors importance (par.2); in addition to the "Grace Period" requirement, a large spectrum of possible design improvements is pursued, including the following:

* enhanced capability of the plant to withstand operator errors in the frame of plant procedures,
* reduced testing/maintenance needs or complexity,
* enhanced quality of information to assess plant and environment conditions during accidents,
* optimal automation level (optimization between workload and understanding ongoing processes),

*restrictions to operator interventions in selected plant conditions (i.e. bypass of critical systems during accidents),
*extended autonomy of safety systems,
*capability of easily detect abnormal plant conditions and operator errors.

Even if these requirements tend to reduce the importance of the human intervention, it can be certainly asserted that AM still plays an important role in the concerned plants for the following purposes:

+to furtherly reduce event consequences, both to the plant and to the environment, even in the Grace-period (achievement of additional safety margins),
+to face unexpected event evolutions, by setting up possible actions to perform when specific conditions take place,
+to perform plant recovery.

For the reasons above, adequate information must be provided in new plants and the related monitoring needs must be considered.

The information important for AM and for the assessment of accident scenarios is delineated in ref.5, concerning the present generation plants. The purpose of the information is to perform preplanned manual actions (e.g. emergency procedures entry conditions), to verify the accomplishment of safety functions and the integrity of barriers to radioactive releases, to control the availability of the concerned plant systems, and for the evaluation of the amount of releases to the environment. <u>It is an authors' opinion that the same general information purposes are applicable to the advanced, passive safety plants.</u>

Coming to the assessment of the information needs, guidelines of Ref.5 are no longer applicable because systems, processes and reference conditions of PLWR differ from those of current plants. As starting point for the new assessment, three different Accident Configurations asking for AM intervention can be distinguished:

*inside Design Bases conditions (DBAs),
*Severe Accidents (SA) , and
*recovery after accidents.

The strategies and the available features are different in each configuration; related information needs are described in the following paragraphs, with particular emphasis on SA.

Concerning conditions inside DBA's, the PLWR has to be designed and optimized for the most frequent and deterministically characterized conditions (operation, transients and DBAs), during which a complete and optimally organized set of information must be available, as in ref. 5 for present generation plants.

Concerning SA, special provisions at hardware and software level are included in PLWR designs. In particular, the instruments providing for the information needs, should survive in severely degraded conditions.

In view of defining monitoring needs and also for emergency procedures set up, it is useful to further subdivide Severe Accidents into two groups:

* those coming from Initiating Events (IE) consisting essentially of process disturbances, followed by further failures,
* those resulting from IE that produce relevant common cause failures (Station Black Out, earthquake, fire, external man-made events).

The differences between the two groups above mainly come from:

* different diagnostic capabilities; for instance, Control Room information can be heavily impaired by the second group of events,
* different availability of equipment devoted to AM interventions (e.g. Energy sources, plant systems), and
* different sets of actions (procedures) to be implemented (directed toward the minimization of the IE effects of the second group of events), possibly in different areas of the plant (in remote control centers if the control room is no longer available).

Concerning the recovery phase, it is assumed to begin when a stable condition of full plant control is reached; the main AM objectives during this phase are to bring the plant to more safe states (less energetic, less prone to new phenomena generation and so on) and to provide long term water and energy sources needed. Data about possible systems to put in operation could complete the set of information already available for the assessment of plant conditions, in order to manage the long term plant control.

4. Impact of the new conception on Monitoring requirements.

4.1 Application of passive safety concept on monitoring.

According to the passive safety conception, any SA should be prevented and mitigated by means of systems not requiring neither external signal/energy feed, nor human intervention. The degree of implementation of this general requirement depends, of course, on the characteristics of each system (e.g. feasibility constraints) and on the required function.
Possible cathegorizations of passive systems/components are reported in Reference 1 and 2; the cathegories are essentially related to fluid or mechanical motions allowed and to internal energy sources and signal processing equipment needed. The spread of passivity degree ranges from a most stringent level, in which safety functions accomplishment is based only on material properties (e.g. fuel cladding), to a least stringent, in which safety functions are accomplished by means of fluid and mechanical natural motions initiateded by a logic and control subsystem with adequate reliability levels.

The design of initiating subsystem above could be based on the development and implementation of the "self acting" conception. This conception is aimed to provide adequate reliability levels, comparable to those of totally passive systems, by means of fail safe design and by using dedicated sensors and energy sources; signal exchanges with other systems should be reduced in order to assure independence and, furthermore, the required initiation function should be maintained, despite of operator error; fig. 1 shows an example of a simple circuit avoiding complete bypass of an essential function. In addition to the improvements in the initiating/actuation functions, the implementation of the self acting conception could have a considerable impact on the overall monitoring structure.

Plant staff manipulations needs could rise when plant information system displays conditions that require change /restoration of systems line-up (e.g. primary /secondary fluid interface damage), or improvement of systems performances or avoidance of automatic actuations that could worse the accident sequence. Adequate equipment devoted to actuate and clear bypasses and to provide information on bypassed systems come as an important item to face.

In general, passive systems pose specific monitoring problems, due to their specific characteristics; capability and uncertainty problems rise up together with limits on automation.
The capability of satisfactorily monitoring the motion of large masses of water, through branched and distributed volumes, could be invalidated in some process scenarios, in which small differential pressures act as driving forces. Such conditions could raise also concerns of flow measurements uncertainty in density or gravity driven circulations.
The limits on automation come from the requirement that the operator should always recognize the ongoing phenomena.

A special mention has to be given to the defence from spurious actuations of systems required for the subsequent operation of other passive systems. These spurious actuations can be more stressing or dangerous in PLWR than in present generation NPPs (e.g. ADS in drywell for SBWR, with suppression pool bypass possibility, flooding following ADS for AP-600).
In spite of possible design provisions to face this problem (e.g. set points choice, signal diversification, component quality), conditions could arise calling for timely operator intervention. In this case special care should be dedicated to provide reliable and timely information to support operator action.

Furthermore, as a prevention of the additional degradations of plant conditions due to the above spurious actuations, enhanced status monitoring of interfacing systems and components should be provided (e.g. vacuum breakers position monitoring in SBWR, in case of spurious ADS).

4.2 Reference Events.

The design of the information system for conditions inside DBA can take advantage of appropriate balance between significant experience gained in operating plants and the ongoing technological improvements. Instead, the enlargement of reference events to SA has a large impact on the monitoring structure (e.g. different phenomena and environments, systems availabilities and safety objectives).
A general design objective should be the comprehensive optimization of the monitoring system for the whole set of conditions, rather than to assess the availability of DBA monitoring instrumentation for more degraded conditions.
At this regard, it must be assumed that SA behaviour will be realistically evaluated by means of models to be timely, fully implemented and tested in Computer Codes. The uncertainty bands should be estimated to be applied for environmental qualification, for ranges definition and for support systems availability evaluation.

The characterization of new information needs can be derived by considering some significant milestones in AM development.
The establishment of symptomatic/function oriented procedures brings to group similar Plant States (i.e. sets of systems configurations and process parameters values) occurring in different sequences. The Plant States pertaining to a group exhibit similar conditions for the selection and the application of the pertinent emergency procedures, whithout any need for a complete event and scenario diagnosis. Some tens of states can be identified in the first phase of the accident, before core melt, because of the large number of initiating events and of involved system failures. As the accident proceeds to Core Melt, the number of significantly different Plant States decreases. This reduction is more conspicuous for PLWR, in fact specific plant conditions are needed and have to be created (e.g. by ADS actuation) for the subsequent intervention of passive, low pressure injection systems. Unfortunately, the similarity of the foreseen plant configurations (e.g. primary system at containment pressure) during SA is counter-balanced by a spread of physical, chemical and thermal-hydraulic conditions because of the variety of the possible phenomena evolutions (e.g. corium and fission product interactions). This spread makes difficult to foresee well characterized Plant States to select the appropriate Accident Management measures. Furthermore, potentially beneficial

actions could result in adverse effects; for instance, adding water to a damaged core could rise Hydrogen production, adding water on a corium could increase containment pressure rise.

In most cases, primary system and containment become strictly linked (e.g. primary system open to the flooded containment) and the containment remains the main environment to monitor.

From the considerations above, the following conclusions for the information system come out:

A) the need of sturdy monitoring equipment, as independent as possible from environmental conditions and from supporting systems potentially affected by accident conditions,

B) the importance of available and reliable information to perform appropriate AM actions and for diagnostic purposes,

C) the need to move the attention from the reactor coolant system to the containment, also for those strategies in which actions to preserve the vessel integrity are addressed (external vessel cooling).

In the following discussion further bases and implications of the conclusions above are examined.

The experience gained from the analysis of SA in current plants, points out two main categories of SA sequences:

- those for which the primary system is intact and at high/medium pressures when severe fuel damage occurs, and
- those for which the primary system is open to the containment before severe fuel damage occurs.

While in the present generation NPP's the first cathegory can contribute significantly to Core Melt Frequency, for PLWR it can be considered residual due to special design efforts (i.e. improved depressurization capabilities). This is true if the scenarios calling for automatic systems bypass by the operator are very few. This consideration reinforces statement C above[1]; furthermore, it appears impractical and not feasible to include severe conditions such as in vessel steam explosions or severe reactivity accidents, for which containment integrity could be still maintained, among in vessel monitoring equipment design

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Note[1] Neverthless, also in this kind of designs, actual scenarios could differ from the anticipated ones, because of operator intervention aimed to lead the accident evolution in a different way from the automatic protection systems would do, if the bypass of the automatic functions will be made possible by design. In such conditions, in which the operator has decided to directly manage the accident, further automatic protection in case of subsequent failures would be no longer available or effective. To overcome this risk, such operator actions should be clearly addressed (by the emergency guides strategies) only in conditions that could not lead to unexpected core damages , and for which adequate monitoring can be assured.

conditions; on the other hand, it appears not conceivable to postulate long term primary system integrity after the above class of accidents.
A possible approach to face monitoring requirements coming from the mentioned problems could be to identify few different sets of instuments, for which different qualification requirements and reference conditions apply; the combination of these sets should be able to satisfy the monitoring needs of the plant and the information requirements of the plant staff in different conditions. An adequate assessment in this area is important because different design options influence the overall monitoring structure (e.g. different independence constraints between different sets of instruments) and arrangements (e.g. influence of environmental conditions on the lay-out of panels). For instance, the SA monitoring could rely on a set of instruments, different from DBA monitoring instrumentation, in order to give, with adequate margin, unambiguous, detailed information in conceivable scenarios at containment level. Anyway, the need to assure meaningful measurements during SA, also inside primary system, is a further condition to be adequately fulfilled.
The best confidence on the overall information could be achieved by means of dedicated sets of instruments capable to provide complete information for AM in a large spectrum of degraded conditions.
The information concerning the primary system remains important to allow AM strategies similar to those applied in current generation plants (e.g. secondary side feed and bleed in PWR), in conditions potentially leading to high pressure SA. On the other side, the containment dedicated instruments could be useful to provide information when plant conditions are particularly degraded.

Functions/parameters whose supervision is needed for AM in Emergency Operating Guidelines of the present generation plants are:
- reactivity,
- reactor coolant system pressure and temperature,
- reactor coolant inventory and chemistry,
- residual heat removal,
- steam generator level (for PWRs),
- hydrogen/Oxigen concentration in containment,
- in containment radioactivity.

Possible additional/specific monitoring areas for SA and recovery management include:
- the heat rejection to the environment (or, at least the existence of conditions in which it can take place),
- the presence of water levels at critical elevations,

- the set up of different plant damage states at primary system and at containment level (e.g. corium in the reactor cavity, Hydrogen generation) and their evolution toward the worse or the better direction (e.g. containment pressurization or depressurization).
- external energy and water inventories possibly needed,
- key systems availabilities.

4.3 Information System Requirements.

In the following, considerations about the problems related with actual AM information systems (from the monitoring to the elaboration and presentation subsystems) are presented, including:

* adequacy and completeness of information contents,
* performance, and
* reliability requirements of instrumentation.

About the completeness of the information content, it is necessary to refer to AM procedures design, including the general purpose to give the Plant Staff all the elements to assess the plant and environmental conditions.
The structure itself of the procedures allows an importance categorization for the required information at different levels, typically: entry conditions to the procedures, "if" gates at plant or system levels, warnings on interfaces. In principle, the emergency procedures set up strictly interacts with the monitoring system design.
Even if the information addressed in the symptomatic procedures cover all the needs from the point of view of possible actions, information is also needed for assessment of scenarios and of their evolutions. This kind of assessment could be useful for different purposes, including: estimate of actual plant conditions (e.g. integrity status) to optimize the effectiveness of the procedured actions, forecast of plant behaviour in order to correlate its timing with possible external emergency actions.
The above quoted needs, together with feasibility and adequacy constraints, permit to generate the list of plant variables to be monitored.
About the adequacy of the information content, special care should be devoted to the optimal choice of actual parameters to be measured. Some factors affecting this choice include:

- capability of direct and quick indication of the addressed conditions and phenomena,
- low disturbance levels,
- meaningfullness of the measured value for different purposes,
- uniqueness of the process condition in which that parameter assumes the threshold values for the required actuations/actions.

The feasibility of measurement for some parameters can be conditioned by instrumentation reliability and performance constraints (especially at sensor level). These constraints are depending on technical limits of the available instruments; to this regard a special attention must be devoted to critical operating conditions which can affect the monitoring equipment in particular applications, such that an impairment of physical and functional integrity of the equipment could result. The limits above can be of particular concern for SA monitoring where unusual environmental and/or process condition are likely to affect the instrument operation.

About the performance requirements of the overall AM information system, main aspects to be considered include, as usually, measuring range, accuracy and response time. They are important during the implementation of the AM procedures to allow the operator to check, with precision and timeliness, the conditions for the execution of the planned manual actions. The measuring range can have a major relevance for the assessment of plant conditions and scenarios diagnosis, especially in the case of severe accident, for which process and environmental parameters can exhibit the largest excursions.
Sensors are mainly involved to meet the requirements above, in fact a little contribution to inaccuracies and delay times can be expected from the elaboration and presentation equipment downstream.
These last parts are aimed, as usually, to other functional aspects, for instance:

* removal of signal spikes and reduction of noise components;
* electrical isolation of signals to allow their use for less critical purposes (e. g. long term data logging, plant supervision and control);
* autodiagnostics of failures occuring in the information system;
* validation of information to be displayed to the operator;
* correlated and unambiguous presentation of information;
* data recording.

Digital processing and human factor constitute relevant technological areas of the information system. Their application is expected to raise problems of software reliability and man-machine interface validation. These problems could be of particular concern if operator aids based on Artificial Intelligence techniques will be developed to support procedures implementation and scenarios diagnosis.

The reliability requirements are important to assure the integrity of information to be provided to the operating staff against possible errors and losses. Degraded information can result as a consequence of:

* unforeseen process conditions causing credible but faulty signals at the sensor output;
* random equipment failures (including support systems);

* adverse environmental conditions, causing a loss of functional and physical integrity of the monitoring equipment.

Main provisions against the risk of the above degradation factors are discussed below, they include: functional diversification, redundant architecture of the information system, and equipment qualification.

Functional diversification requires the existence of different physical parameters related to the same phenomena (e. g. process, system operation); the diverse information sources can be used for validation purposes; furthermore, a diversified source can be used as backup to supply the required information if the primary source is not available.

Redundancy is usually a stringent requirement for the monitoring of critical variables during DBA's, as well as the independence of redundant monitoring channels, and associated support systems. The redundancy requirements related to SA should be discussed on a case by case basis, taking into account probabilistic aspects; for instance redundancy could be recommended for the monitoring channels involved with the verification of entry conditions of AM procedures.

Similar considerations apply also to equipment qualification. In particular it does not appear feasible to provide full qualification for SA monitoring equipment because of the extreme excursions of process variables and of environmental conditions. DBA qualification margins are expected to give limited assurance about the availability of the monitoring equipment in SA: the initial phase could be covered at the most, then the qualification limits would be exceeded. To this regard various provisions are conceivable, but their effectiveness is not obvious. An analytical approach could be used to verify if equipment remains available also during SA, provided that the qualification limits are not exceeded by large amounts. Instead, if the qualification limits are strongly overcome by peak values and durations occuring in SA, other provisions could be applied, such as relocation of instruments and connecting cables in less harsh environment, or shielding of components with respect to the challenging environmental factors. As part of the systematic conception and design effort of the PLWR, a qualification program for the monitoring instruments should be addressed, if the monitoring needs in SA are not achievable with the above provisions. To this regard various considerations could be made. First of all, the profile of SA parameters (i.e. process and environment) to employ as a deterministic reference for qualification, can be reduced significantly if mitigation effects of passive systems are trusted (e.g. avoidance of large hydrogen explosions or of direct containment heating). Furthermore some relaxation could be introduced in the qualification criteria and procedures, based on results of research and development activities (ref.7); for instance, simultaneous simulation of reference conditions for qualification could be replaced with sequential simulation.

The plant design life extention to sixty years is another factor to be considered in the qualification requirements; anyway, most component can be replaced during plant life, so a shorter qualified life is allowed for them.

5. Approaches to a methodology.

Different methodologies have been proposed for systematically approaching the information need problems in the present generation plants (ref.4 and 6). The main purpose of these methods is to verify the availability of existing plant instruments during SA and to identify the need for modifications and for additional instrumentation.
Many steps of the proposed approaches can be applied to PLWR; the main modifications are induced by the advantage of beeing in the design stage. A tentative approach applicable to PLWR is shown in fig. 2, where the following steps are identified:
STEP 1 - Consider foreseen accidents and potential evolutions, as deriving by the probabilistic safety studies, accident analysis and design basis, then categorize Plant States encountered in severe accident paths. This approach could be preferred to the functional one, addressed in ref. 4 and 6, because all conceivable scenarios should be addressed in the design stage, therefore information is needed about reference conditions and support systems availability. Functional approach could be a further assessment tool to be used in Step 5 below.
STEP 2 - Identify the AM strategies for the above Plant States, related systems and plant staffing, characterize also the expected environmental conditions;
STEP 3 - Define information needed to safely perform the AM actions and to assess the general Plant Conditions and Scenarios, identify the support systems that are needed;
STEP 4 - Identify monitoring instrumentation requirements and, possibly, subdivide it in subsets to be available in homogeneous conditions; in this phase the purpose to avoid, to the possible extent, harsh environments for SA instruments should be pursued;
STEP 5 - Build up the overall information structure to assess the overall adequacy (e.g. redundancy, diversification, information correlation).

6. Conclusions.

The new advanced, passive design conceptions on one side arise some new monitoring and control problems, on the other side offer the opportunity to design optimal monitoring structures, able to provide adequate responses to information needs. The general purposes of the information to be presented to the Plant Staff, as identified for current generation plants, are still applicable to the advanced, passive safety plants.

The identification of the information needs comes from the interaction with many design activities; discussions have been addressed to the feedbacks coming from the enlarged set of events to be considered, from the passive safety systems conception and from the possible AM strategies. Possible design requirements and problems concerning the implementation of information systems have been considered and discussed.
The potential exists to satisfactorily resolve all the issues with adequate design and research efforts. If the same innovative mind, that brought to new designs conception, is maintained in more detailed design, further safety improvements could be achieved.

7. References.

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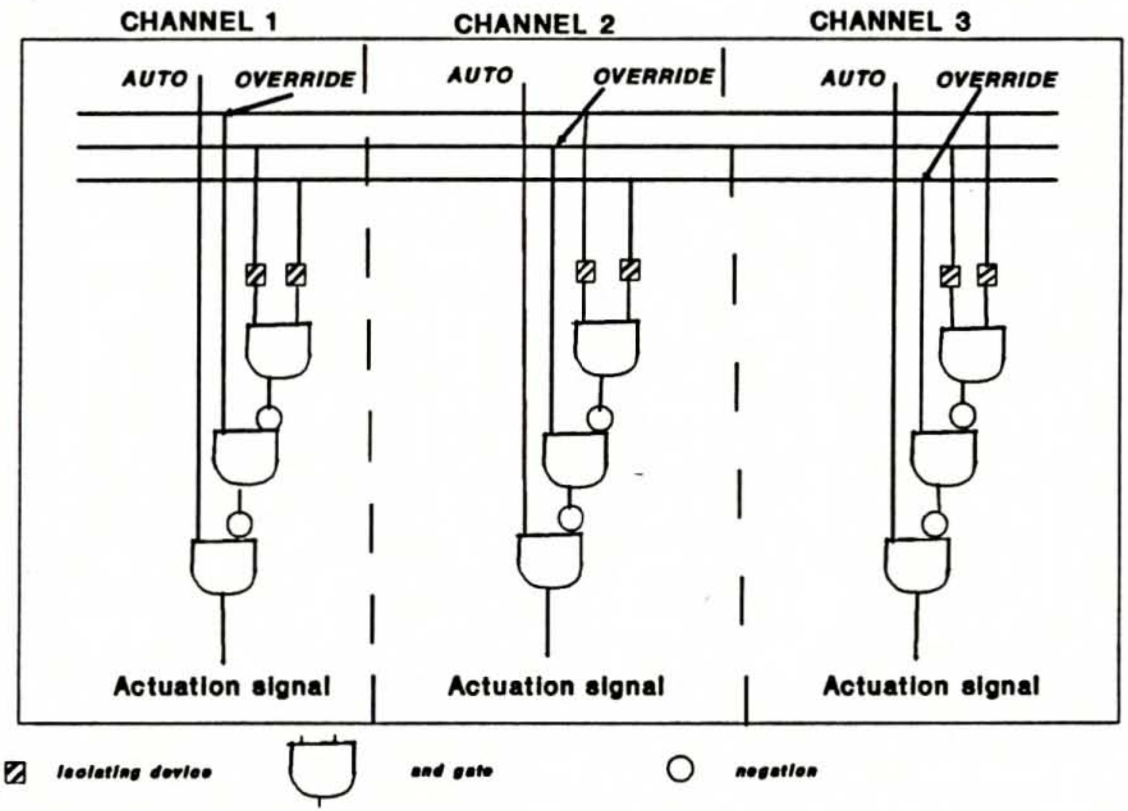

Fig 1 Possible, simple solution to limit by pass capability

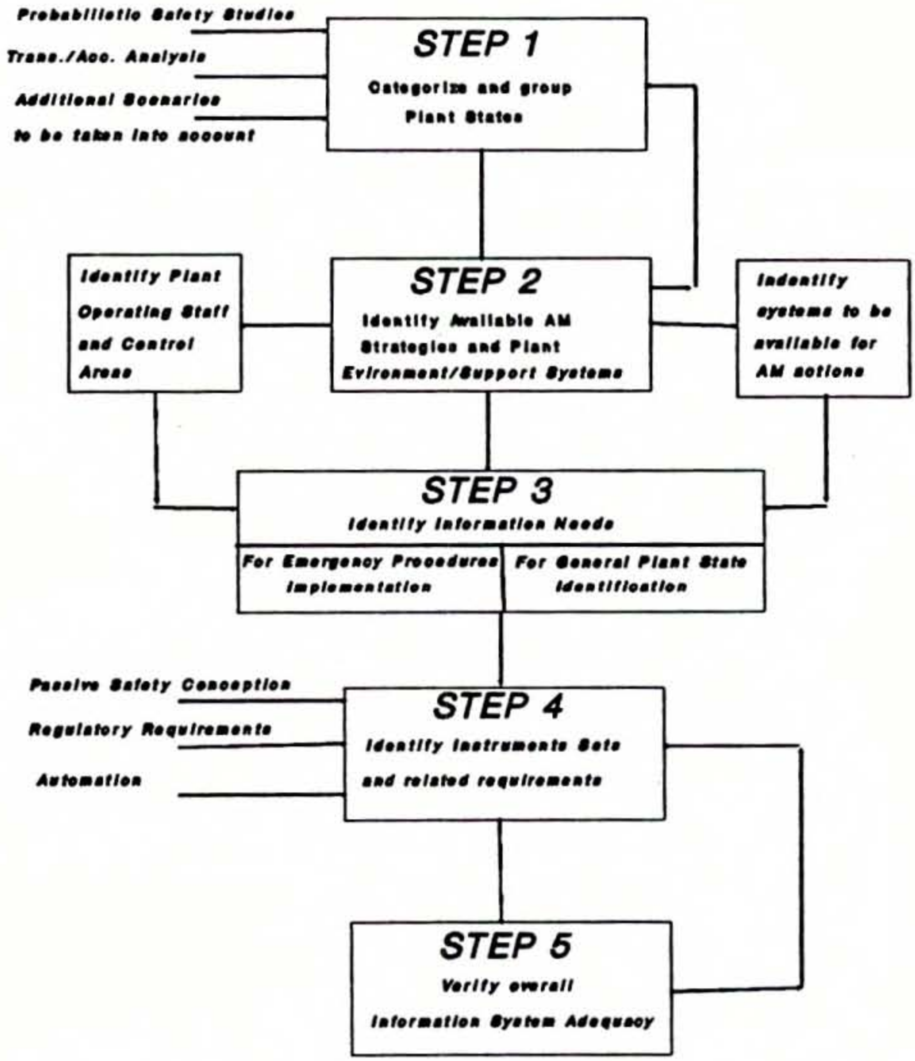

Fig.2 Methodology forInformation System Set up

SUMMARY of SESSION I

Dr. Patricia Worthington (USDOE) presented the report titled "Instrumentation for Accident Management in Containment prepared by CSNI-PWG-4's Task Group on Containment Aspects of Severe Accident Management. The INEL and NUMARC/EPRI self-assessment methods for utility use were referred to. Each has instrumentation as one component of the assessment. It was suggested that creative information gathering include portable instrumentation, system status (operational/ failed), component failures, and unintended uses of instruments. It is the hope of the Task Group that this meeting facilitate creative approaches to accident management information/ instrumentation. This paper provided an excellent kick-off to the meeting.

Dr. Eric Södermann (ES-KONSULT AB, Sweden) discussed the Swedish approach to information needs in Boiling Water Reactor (BWR) severe accident situations. Swedish Emergency Operating Procedures (EOPs) include mitigative steps. During a severe accident the operator would uses EOPs even to the point of being allowed to release radioactive gases (through scrubbers) without requesting situation-specific state/regulatory permission. Normal operator responsibility is expected to a 800 °C core temperature. At that point the Technical Support Center (TSC) should be available and assume lead responsibility. Accident management reactor instrumentation consists of pressure (reliable), water level (not so reliable below top of core), and power range monitor (complicated to interpret). For the late stages of accident management, it is judged that none of the Reactor Pressure Vessel (RPV) instruments will work. Containment instrumentation (for late severe accident stage) consists of pressure, water level, and temperature for wetwell and drywell, and radiation monitors.

Accident history information is considered important to interpreting current plant status. A system called SAS-II has been developed to perform oversight of the critical safety functions as an aid to the shift supervisors. A currently unfunded project concept, CAMS (Computerized Accident Management Support), is intended for the use of plant staff. It would eventually include an expert system judging validity of its plant signals. A great advantage of the concept is that the data base can be used for normal operation applications. The conclusion was that we can never make late-stage instrumentation sufficient, thus accident history is important to understanding especially for late stage plant status. Additionally, capturing early accident history for review by the TSC and crisis response team is important so they can correctly begin to deal with a severe accident.

Dr. Martin Sonnenkalb (GRS, Germany) presented "Information and Requirements needed for Accident Management". The GRS approach was influenced by the NRC/INEL approach but was independently arrived at. It starts with four safety objectives:

(1) prevention of core damage
(2) retention of core material in vessel
(3) prevention of containment failure
(4) mitigation of fission product release.

These are supported by safety functions which deal with various challenges. The challenges are caused by physical mechanisms and have various accident management strategies identified to deal with them. A detailed Pressurized Water Reactor (PWR) example was discussed. Filtered containment venting was discussed for PWRs. Requirements are that it be operated from the control room, that the system has to be protected from effects of hydrogen combustion by additional countermeasures, and that instrumentation and control of the system include temperature and pressure variables together with an activity measurement, which is important.

Information sources for prevention of containment failure were outlined. RPV integrity and core melt process information would be useful to anticipate and respond in a timely manner. Requirements to limit hydrogen concentration were identified. Measurements of importance include pressure, temperature, radioactivity, and hydrogen / steam / oxygen concentrations.

In conclusion, the safety objective tree approach was successful. All relevant mechanisms were considered. A question during discussion raised the issue of determining hydrogen concentration based on observing if the ignitors are functioning, for those ignitors requiring electricity to function. Another question identified the potential information need for a carbon monoxide detector. Another question was how Direct Containment Heating (DCH) and steam explosion were eliminated from consideration for German plants. The answer was that the probability of high loads from steam explosion was very low because there would be no water in the cavity, and that DCH was eliminated because of primary bleed action and cavity design. Questions asked relative to the use of recombiners for severe accidents brought out the intent of Germany to look aggressively at catalytic (passive) hydrogen recombiners.

Mr. Bruno Rague (IPSN, France) presented instrumentation needs and data analysis for the diagnosis and prognosis of the source term by the French Institute for Protection and Nuclear Safety (IPSN) during an emergency in a PWR. Making such a prognosis at an early stage, in terms of time, duration and intensity of the potential release of fission products into the containment, will provide civilian authorities with an sufficient advance warning for off-site emergency plan implementation. The French national emergency organization was described. The IPSN crisis team which reports to the French Safety Authority, but also periodically compares its analyses with those of the utility emergency technical support team, basically uses a computerized data acquisition system including plant parameters (safety panel, plant computer outputs) and radiation monitoring in the vicinity of the plant. Such a system is backed up by telefax and an audioconference system when appropriate. Necessary measurements for the diagnosis of the plant status include core exit temperature, containment pressure, saturation margin, primary circuit mass balance, and similar parameters in the auxiliary building in the event of a containment bypass. Additionally, activity and steam generator blowdown (level-related) measurements are also important. The status of containment isolation, generator and condenser steam dump valves, and steam generator pressure and mass balance are also useful. The various software tools used to help with accident prognosis are part of the SESAME project: they include BRECHEMETRE (assessment of primary break size and other parameters), SINBAD (containment leak size), ALIBABA (expert system to identify containment leakage), HYDROMEL (containment temperature and pressure influenced by core melt progression, and hydrogen risk), PERSAN (containment bypass analysis) and RTGV (thermohydraulic and fission product transfer in case of steam generator tube rupture).

Data acquisition, organization and control were discussed. Procedures / analysis necessary for the use of French sand bed filters have been developed. Code qualification has been closely looked at. On-line data is available from the plant to the national crisis center.

Dr. Jason Chao (EPRI, USA) presented "Instrumentation and Severe Accident Plant State Implementation". EPRI's current program is aimed at getting the most out of current instrumentation/information to allow actions to be taken, and feedback on the impact of those actions. Instrument response does not need to be as accurate as during normal operation. Not all instrumentation loops will see a severe environment. Calculational aids to interpret instrument response and to predict conditions where there are no instruments will be developed. TMI-2 information was considered and studied. The six project steps to obtain the deliverables were described. Methods to interpret instrument response and survivability will be developed. The project will identify instruments of interest and select some for further study. The project identified conditions which instruments are likely to see. Interpretation of instrument response is the goal. Calculational aids will be developed to determine the relation of instrumentation readings to actual plant parameters. Additional aids will help interpret instrument errors. Deliverables include an instrumentation data base including failures and success under severe conditions, methods to asses validity of signals from instrument loops, and calculational aids.

Mr. Fausto Zambardi (ENEA/DISP, Italy) spoke about "Considerations on Monitoring Needs of Advanced Passive Safety Light Water Reactors for Severe Accident Management". A brief introduction to advanced passive reactor design was given. Information needs are influenced by the more robust (safer) design aspects of the advanced plants, such as operator error tolerance, reduced complexity, and the capability to easily detect abnormal conditions. Consideration in the design will be given to the design basis, severe accidents, and recovery after accidents. The concept of passive safety systems is to prevent accidents with limited human intervention. This potentially decreases the reliance on instruments and human intervention to insure safety. Instruments are still important to monitor plant states. Functions to be monitored for EOPs include reactivity, Reactor Cooling System (RCS) pressure and temperature, residual heat removal, steam generation (PWRs), hydrogen/oxygen concentration in containment, and in-containment radioactivity.

Possible additional functions to be monitored include heat rejection to the environment, water level in containment, external energy and water inventories, and key systems availabilities. Instrument system requirements will include adequacy, completeness, performance, and reliability characteristics. A possible method to determine instrumentation needs for advanced reactors is embodied in the following five steps:

STEP 1: Using probabilistic risk analyses, considering foreseen accidents and potential evolutions characterizing various plant states.

STEP 2: Identifying strategies for the plant states.

STEP 3: Defining information needed to perform these strategies.

STEP 4: Identifying monitoring instrumentation requirements.

STEP 5: Building in redundancy, diversification, and information correlation.

In discussion, it was suggested that these five steps could be applied to current plants also.

SESSION II

Capabilities and Limitations of Existing Instrumentation

Chairman: G. LÖWENHIELM

BWR Instrument Availability During Severe Accidents[a]

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ABSTRACT

The ability of plant personnel to successfully manage severe accidents is strongly influenced by the availability of timely and accurate plant status information. The United States Nuclear Regulatory Commission (USNRC) recognized this relationship by making instrumentation one of the five elements of its accident management framework. This paper describes the results of research sponsored by the NRC to evaluate the availability of plant instrumentation during a range of possible severe accidents at a BWR with a MARK I containment design.

Instrument availability is evaluated based on environmental conditions for a range of possible severe accidents that could occur at a boiling water reactor (BWR) with a MARK I containment design. Based on this evaluation, the principal challenge to instrument availability is the severe pressure and temperature environments in the containment and reactor building during an accident. These conditions can develop before or after core damage, depending on the sequence. For accidents with operating containment cooling systems and failure of emergency core cooling systems, performance of many instruments located in the containment can degrade but only after significant core damage has occurred. For accidents involving an ATWS, performance of instruments located in the drywell, wetwell, or reactor building can degrade prior to core damage. Severe conditions in the reactor building can result from containment failure or failure of the vent system during containment venting.

Detailed results from the evaluation of BWR instrument availability during severe accidents is published in NUREG/CR-5444[1].

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INTRODUCTION

The capability currently exists to manage a broad range of accidents at nuclear power plants in the United States. Consequently, severe accidents at nuclear power plants will occur only if there are multiple failures of safety related equipment, serious human errors, or some combination of these two conditions. To manage this complex severe accident behavior, plant personnel must successfully diagnose the occurrence of an accident, determine the extent of challenge to plant safety, monitor the performance of automatic systems, select strategies to prevent or mitigate the safety challenge, implement the strategies, and monitor their effectiveness. One of the areas affecting the capability of personnel to effectively carry out these actions is the availability of timely and accurate plant status information. Plant instruments are relied upon to supply this information.

Safety-related instrumentation installed in a nuclear power plant is primarily designed and qualified for preventing and mitigating design basis accidents. The capability of the instrumentation to supply the information needed for severe accident management has not been comprehensively investigated for conditions typical of a broad range of severe accidents.

In this paper, severe accident conditions that influence instrument availability and performance are identified and the availability of plant instrumentation for a wide range of severe accidents is assessed for a boiling water reactor (BWR) with a MARK I containment design. For this assessment, instrumentation data and severe accident results applicable to the Peach Bottom Atomic Power Station are used. The Peach Bottom station has two General Electric boiling water reactors (BWR-4) each with a rated thermal power output of 3293 MW_{th} and housed in a Mark I containment.

APPROACH USED TO EVALUATE INSTRUMENT AVAILABILITY

The approach used to evaluate instrument availability is described in the following steps:

Step 1: Identify Severe Accident Sequences

The types of possible severe accident sequences that represent the spectrum of accident types which have principal impact on risk for a BWR with a Mark I containment are identified in this step. The probabilistic risk assessment results presented in NUREG-1150[2] for the Peach Bottom Atomic Power Station were used to identify the types of severe accident sequences that are significant contributors to risk. The NUREG-1150 results are the most recent evaluation of all credible types of accidents that will dominate core damage frequency and risk to the public. Although the results are specific to the Peach Bottom plant, the sequence categories identified are sufficiently broad to apply to most BWRs with MARK I containment designs.

The following plant damage states identified in NUREG-1150 and used in this assessment are:

1. Station blackout (SBO)
2. Large- and small-break loss-of-coolant accidents (LOCAs)
3. Anticipated transients without scram (ATWS)
4. All other transients except SBO and ATWS

Step 2: Determine Expected Conditions

The expected conditions within the reactor coolant system, containment (drywell and torus), and reactor building for the identified severe accident sequences are determined in Step 2. Thermal hydraulic conditions for the plant damage states and accident progression bins identified in Step 1 are based on results presented in BMI-2104[3] and NUREG/CR-4624[4] analyses. The principal parameters of interest for evaluating instrument availability from these analyses are the temperature and pressure in the areas in which instrument components are located. These areas include the reactor coolant system, containment (drywell and torus), and the reactor building for all sequences. These analyses are used because most of the important events expected during a severe accident, from core melt through lower head failure and beyond, are considered, including possible containment failure modes.

A tabulation of the maximum value of various thermal hydraulic parameters reached during the important accident phases is presented in Table 1. For this evaluation, 100 percent relative humidity is assumed for all accidents. It is judged that the range of thermal hydraulic conditions expected for any plant damage state and accident progression bin combination are adequately reflected in the BMI-2104 and NUREG/CR-4624 analyses.

Table 1
Summary of Maximum Value of Key Parameters
for the Severe Accidents Used in this Evaluation

Parameter	Initiation to Core Uncovery	Uncovery to Start of Melt	Meltdown to Core Slump	Core Slump to Head Failure	After Head Failure
Within the Reactor Vessel					
Average Core Temperature	965 K (1277°F)	1496 K (2233°F)	2126 K (3368°F)	2358 K (3784°F)	N/A[1]
Core Exit Gas Temperature	N/A[1]	1228 K (1750°F)	2061 K (3250°F)	2061 K (3250°F)	N/A[1]
Max RPV Structure Temperature	N/A[1]	561 K (550°F)	1644 K (2500°F)	1644 K (2500°F)	N/A[1]
Max Reactor System Pressure	8.28 MPa (1202 psia)	7.51 MPa (1090 psia)	7.93 MPa (1150 psia)	8.05 MPa (1168 psia)	N/A[1]
Primary Containment					
Pressure	0.89 MPa (129 psia)[2]	0.89 MPa (129 psia)[2]	0.21 MPa (30 psia)	0.90 MPa (131 psia)	0.91 MPa (132 psia)
Drywell Temperature	435 K (324°F)[2]	324 K (324°F)[2]	409 K (276°F)	1384 K (2031°F)[3]	1028 K (1391°F)
Pool Temperature	449 K (349°F)[2]	449 K (349°F)[2]	373 K (212°F)	373 K (212°F)	373 K (212°F)
Reactor Building					
Temperature	394 K (250°F)[2]	394 K (250°F)[2]	922 K (1200°F)[4]	394 K (250°F)	1644 K[4] (2500°F)

Notes:

1. N/A - Not Applicable
2. Conditions predicted for an ATWS event with containment failure due to overpressurization from pool heatup.
3. Conditions predicted due to the rapid generation of hydrogen during core slumping during a large break LOCA.
4. Temperature spike due to predicted hydrogen burns.

The effect of radiation conditions is evaluated by comparing the integrated dose resulting from various radionuclide release scenarios based on release data presented in NUREG-0737[5] and radionuclide distribution data from BMI-2104. The data in NUREG-0737 assumes release of 100 percent of the noble gas, 50 percent of the halogen and 1 percent of the particulate (solid) radionuclides from the fuel for LOCA events that depressurize the reactor coolant system. The BMI-2104 report presents estimates of the releases of the fission products and other aerosols from the fuel during core melt and core-concrete interaction. The magnitude of the iodine and particulate releases is the principal difference between the BMI-2104 and NUREG-0737 data.

Step 3: Assess Instrument Availability

Instrument availability during the identified severe accident sequences is assessed in Step 3 based on the location of the instrument components and conditions that would influence instrument performance. Instrument availability is evaluated based on: the general location of the instrument; the range; and the qualification limits for temperature, pressure, humidity, and radiation levels. The instrument evaluations presented are based on the Regulatory Guide 1.97[6] review for the Peach Bottom Atomic Power Station[7]. This information provided the measurement ranges and the qualification level of each instrument required for DBA events.

The qualification limits for plant instrumentation included in this evaluation are summarized below:

- For Sensors Located in the Drywell:

Temperature Limit	-	431 K (317°F)
Pressure Limit	-	0.44 MPa (64 psia)
Radiation Limit	-	4.4×10^7 rads
Humidity Limit	-	100 percent

- For Sensors Located in the Reactor Building:

Temperature Range	-	322 - 394 K (120 - 250°F)
Pressure Range	-	0.10 - 0.12 MPa (0 to 2 psig)
Radiation Limit	-	3.5×10^4 rads
Humidity Limit	-	100 percent

The evaluation of instrument availability focuses on the location of the sensors with consideration given to electronics, cabling, splices and other components. The assessment of instrument availability assumes that instrument performance will be degraded if the pressure, temperature, or radiation conditions in the vicinity of the instrument exceeds the specified qualification limits, or if the parameter being measured is outside the instrument range. This definition includes the possibility of instrument failure. Degraded instrument performance means that the indicated magnitude or trend of the measured parameter is in error. This error may cause the operator to take inappropriate action, cause premature termination of the operation of an automatic safety system, or alternately start the operation of an automatic safety system when it is not required. An example would be termination of the operation of the high pressure coolant injection system (HPCI) due to an false indication of high reactor vessel water level.

INSTRUMENT AVAILABILITY EVALUATION AND RESULTS

The principal environmental challenge to any instrument is the occurrence of severe pressure, temperature, and radiation conditions in the vicinity of the instrument, resulting in degraded instrument performance. As used in this evaluation, severe conditions means that conditions in the vicinity of the instrument have exceeded the specified qualification limits. Severe conditions will occur within the reactor coolant system for any accident resulting in significant core damage or core meltdown. Severe conditions can also occur in the containment (drywell and torus) and in the reactor building prior to the occurrence of core damage for accidents initiated either by an ATWS with standby liquid control system (SLCS) failure or for transient initiated accidents with successful actuation of core cooling systems but where containment heat removal systems have failed. In either case, continued heat rejection to the suppression pool will cause drywell and torus pressurization. Severe conditions can occur in the containment (drywell and torus) due to the release of hot steam and hydrogen from the reactor system for accidents where the reactor system is at high pressure at lower head failure. After head failure, severe containment conditions can also occur due to generation of non-condensible gases during core-concrete interaction. If containment failure occurs or if duct failure occurs after the containment is vented, then release of steam and hot non-condensible gases can cause severe conditions in the reactor building.

The instrument availability evaluation based on pressure and temperature conditions is summarized for the following situations:

- o Severe conditions only in the reactor system
- o Severe containment conditions before core damage
- o Severe containment conditions after core damage
- o Severe reactor building conditions before core damage
- o Severe reactor building conditions after core damage

This approach is used because of the possibility of severe conditions in the containment and reactor building prior to core damage during an ATWS or accidents where the containment heat removal systems have failed. A summary of the instrument availability evaluation is presented in Table 2.

Radiation could affect instrument availability in the longer term (days or weeks) if core melt occurs. Instrument components located in the reactor building could be particularly susceptible since these instruments are generally qualified to an integrated dose limit of 3.5×10^4 rads. This integrated dose could be exceeded in a few hours in a core melt accident where containment failure occurs. For instruments located in the containment, the radiation qualification limit is generally 4.4×10^7 rads. The length of time required to exceed this dose is on the order of a few weeks, assuming a realistic amount of fission product retention in the suppression pool.

Instruments Located in the Reactor Coolant System

The only instruments located in the reactor coolant system used for accident management are the detectors for the source range monitor, intermediate range monitor, local power range monitor, and average power range monitor systems. These systems would provide important information during a severe accident because they would be used to monitor the reactivity safety function.

Severe conditions will develop in the reactor coolant system if core uncovery occurs and core damage starts. Degraded performance and ultimately failure of the detectors for the above systems will occur due to temperatures approaching 2200 K (3500°F) as core damage progresses and core meltdown starts for any severe accident. As discussed in the following section, there is the possibility that the performance of these systems would degrade before core damage occurs as a result of severe conditions in the containment or reactor building.

Table 2
Instrument Availability Assessment

A : Instrument Available
-D-: Degraded Performance Possible

Instrument	Severe Conditions Only In The Reactor System	Severe Containment Conditions		Severe Reactor Building Conditions	
		Before Core Damage	After Core Damage	Before Core Damage	After Core Damage
suppression pool water temperature (on torus shell, category 1)	A	-D-	-D-	-D-	-D-
source range monitors (drywell, category 1)	-D-	-D-	-D-	-D-	-D-
intermediate range monitors (drywell, category 1)	-D-	-D-	-D-	-D-	-D-
average power range monitors (drywell, category 1)	-D-	-D-	-D-	-D-	-D-
drywell sump level (drywell, category 1)	A	-D-	-D-	-D-	-D-
primary containment area radiation monitor - high range (drywell, category 1)	A	-D-	-D-	-D-	-D-
primary containment isolation valve position (drywell, category 1)	A	-D-	-D-	-D-	-D-
primary containment isolation valve position (reactor building, category 1)	A	A	A	-D-	-D-
reactor vessel pressure (reactor building, category 1)	A	A	A	-D-	-D-

Table 2
Instrument Availability Assessment

A : Instrument Available
-D-: Degraded Performance Possible

Instrument	Severe Conditions Only In The Reactor System	Severe Containment Conditions		Severe Reactor Building Conditions	
		Before Core Damage	After Core Damage	Before Core Damage	After Core Damage
reactor vessel water level (reactor building, category 1)	A	A	A	-D-	-D-
suppression pool water level (reactor building, category 1)	A	A	A	-D-	-D-
drywell pressure (reactor building, category 1)	A	A	A	-D-	-D-
containment and drywell oxygen concentration (reactor building, category 1)	A	A	A	-D-	-D-
containment and drywell hydrogen concentration (reactor building, category 1)	A	A	A	-D-	-D-
drywell atmosphere temperature (drywell, category 2)	A	A	A	-D-	-D-
primary containment safety relief valve position (drywell, category 2)	A	-D-	-D-	-D-	-D-
vent stack effluent radioactivity (reactor building, category 2)	A	A	A	-D-	-D-
suppression chamber spray flow (reactor building, category 2)	A	A	A	-D-	-D-

Table 2
Instrument Availability Assessment

A : Instrument Available
-D-: Degraded Performance Possible

Instrument	Severe Conditions Only In The Reactor System	Severe Containment Conditions		Severe Reactor Building Conditions	
		Before Core Damage	After Core Damage	Before Core Damage	After Core Damage
drywell spray flow (reactor building, category 2)	A	A	A	-D-	-D-
reactor core isolation cooling flow (reactor building, category 2)	A	A	A	-D-	-D-
high pressure core injection flow (reactor building, category 2)	A	A	A	-D-	-D-
core spray system flow (reactor building, category 2)	A	A	A	-D-	-D-
low pressure core injection flow (reactor building, category 2)	A	A	A	-D-	-D-
standby liquid control system flow (reactor building, category 2)	A	A	A	-D-	-D-
standby liquid control system storage tank level (reactor building, category 2)	A	A	A	-D-	-D-
residual heat removal system flow (reactor building, category 2)	A	A	A	-D-	-D-
residual heat removal heat exchanger outlet temperature (reactor building, category 2)	A	A	A	-D-	-D-

Table 2
Instrument Availability Assessment

A : Instrument Available
-D-: Degraded Performance Possible

Instrument	Severe Conditions Only In The Reactor System	Severe Containment Conditions		Severe Reactor Building Conditions	
		Before Core Damage	After Core Damage	Before Core Damage	After Core Damage
reactor core isolation cooling room temperature (reactor building, category 2)	A	A	A	-D-	-D-
high pressure core injection room temperature (reactor building, category 2)	A	A	A	-D-	-D-
emergency ventilation damper position (reactor building, category 2)	A	A	A	-D-	-D-
status of standby power and other energy sources important to safety (reactor building, category 2)	A	A	A	-D-	-D-
common plant or multipurpose vent flow (reactor building, category 2)	A	A	A	-D-	-D-
control rod position indicator (drywell, category 3)	A	-D-	-D-	-D-	-D-
primary loop recirculation flow (reactor building, category 3)	A	A	A	-D-	-D-
reactor building or secondary containment area radiation monitor (reactor building, category 3)	A	A	A	-D-	-D-

Instruments Located in the Containment (Drywell or Torus)

Instrument sensors located in the drywell, as listed on Table 2, include the drywell sump level, primary containment area radiation monitor, and drywell atmosphere temperature. Instrument sensors to monitor suppression pool temperature are located on the torus shell. The motorized drives for the movable detectors used in the source range monitors (SRM) and intermediate range monitors (IRM) are also located in the drywell. Some BWR plants may have additional equipment located in the drywell such as reference legs for the reactor vessel level system.

Severe Conditions In The Containment Before Core Damage

Degraded performance of instruments in the drywell and torus is possible during accidents where containment pressurization occurs prior to core damage. The principal challenge to instrument availability is high pressure conditions generated from continued heat rejection to the suppression pool during an ATWS or resulting from failure of the containment heat removal systems as explained earlier. Pressurization resulting from these postulated accidents would reach 0.69 - 0.79 MPa (100 to 115 psia) before containment venting is initiated. For Peach Bottom, this is almost twice the instrument pressure qualification limit of 0.44 MPa (64 psia). If the containment is not vented, then higher pressures approaching the mean containment failure pressure of 1.1 MPa (165 psia) are possible. The mean containment failure pressure of 1.1 MPa (165 psia) was used in the NUREG-1150 evaluation of Peach Bottom. The primary containment area radiation monitor may be particularly affected by pressures above the qualification limit since a gas filled detector tube is used which could be affected by pressure changes. Containment temperature would also rise above the instrument qualification limit as the mean containment failure pressure is approached.

Temperature conditions in the containment resulting from an ATWS or from the failure of containment heat removal systems will principally affect the suppression pool water temperature indication since the upper limit of the range of this instrument will be exceeded. In the case of suppression pool temperature, the upper limit of the instrument range is 383 K (230°F). This limit would be exceeded by 310 K (100°F) or more during an ATWS with SLCS failure.

Degraded performance of the motorized drives used for the SRM and IRM systems is also possible due to severe conditions in the drywell. As a result, the ability to monitor core power during an ATWS could be affected.

Severe Conditions in the Containment After Core Damage

A review of the BMI-2104 and NUREG/CR-4624 results show that drywell and torus pressure and temperature spikes are predicted to occur suddenly due to the release of steam and non-condensible gases from the reactor coolant system upon vessel failure for accidents where the reactor coolant system is not depressurized. A typical drywell temperature spike from the Battelle analysis is 755 K (900°F) at the time of lower head failure with a corresponding rise in pressure to 0.69 MPa (100 psia). Instruments located in the drywell could experience temperatures and pressures well above the qualification limits for brief periods of time. Exposure to these conditions could result in degraded instrument performance. Containment hydrogen burns are not considered in this evaluation since the containment is inerted with nitrogen.

Both pressure and temperature in the containment will rise after vessel failure due to the release of hot steam and hydrogen from the reactor system for accidents where the reactor system is at high pressure at lower head failure or due to the generation of non-condensible gases from core-concrete interaction. Degraded performance of the drywell atmosphere temperature, suppression pool temperature, or containment area radiation monitor instruments would not be expected until the temperature or pressure increased beyond the qualification limit in the containment.

Instruments Located in the Reactor Building

Severe conditions in the reactor building will have the greatest effect on instrument availability during a severe accident. The principal reasons are because components of many instrument systems are located in the reactor building as seen from Table 2 and because the qualification limits are generally lower when compared to instruments located in the containment.

Severe Conditions in the Reactor Building Before Core Damage

The principal challenge to availability of instrument located in the reactor building is the flow of high temperature steam that would be released to the reactor building if the containment fails. Containment venting could also release high temperature steam to the reactor building since venting system failure is likely if vents other than the hardened vent system are used during an ATWS with SLCS failure or if a hardened vent system is not installed. This steam could cause the temperatures in many reactor building locations to approach 394 K (250°F), which is above the temperature qualification limit for most instruments located in the reactor building. As a result, degraded performance of many instruments with components in the reactor building is expected.

An additional challenge to instrument availability in the reactor building is the effect of steam condensation on instrument components, particularly electronic components. Condensation on component surfaces could cause failure due to electrical shorts.

Severe Conditions in the Reactor Building After Core Damage

If both core damage and containment failure occurs, severe temperatures and high steam concentrations will occur in some areas of the reactor building causing degraded performance of the instruments in those areas. In addition, there is the possibility of hydrogen burns in the reactor building. These hydrogen burns can cause temperature spikes in excess of 1366 K (2000°F). It is noted that the reactor building is compartmentalized and that the effect of hydrogen burns on instrument performance could be localized.

Some testing has been conducted to assess the effects of hydrogen burns on typical nuclear reactor instrumentation system components. Results from these tests indicate that a single hydrogen burn would not fail either the transducers or cabling of the tested systems. However, both transducers and cabling failed when multiple hydrogen burns were used in the tests. Based on these results, degraded performance of the instrument systems in the reactor building is assumed when multiple hydrogen burns were predicted. It is recognized that the general assumption that multiple hydrogen burns will degrade performance of all instruments is conservative since the extent of the failures would be dependent on the building design, the amount of hydrogen released, and instrument system hardware.

Instrument Availability During A Station Blackout or Loss of a DC Bus

The Regulatory Guide 1.97 category (cat 1, 2, and 3) for each instrument is listed in Table 2. Category 1 instruments require onsite (standby) power.

Onsite (standby) power does not necessarily mean that the power source has a battery backup. Category 2 requires only a high reliability power source (not necessarily standby power). Category 1 and Category 2 instruments are required by Regulatory Guide 1.97 to have battery backup power only when momentary interruption of the instrumentation is not tolerable. Category 3 requires only offsite power.

The availability of instrumentation during a station blackout or loss of a DC bus is dependent on the plant design. If a battery backup is provided for all Category 1 equipment, then these instruments would be available at the beginning of the station blackout. The duration of the instrument availability depends on the battery design, size, load, and load shedding. Generally, Category 2 or Category 3 instruments would not be available, although some plants have some Category 2 or 3 equipment on battery backup. Instrument availability during a severe accident initiated by a station blackout or loss of a DC bus must be evaluated for a specific plant due to differences in instrumentation design.

During a station blackout, systems that are used to obtain and analyze samples of reactor coolant, drywell or torus atmosphere, and suppression pool water may not be available. As a result, information needs requiring sampling information may not be met.

CONCLUSIONS

The principal challenge to instrument availability during severe accidents in a BWR with a Mark I containment is concluded to be severe pressure and temperature environments in the containment and the reactor building. These severe conditions can develop either before or after core damage, depending on the accident sequence. For sequences with operating containment cooling systems and failure of emergency core cooling systems, performance of many instruments located in the containment can degrade but only after significant core damage has occurred. For accidents involving an ATWS, performance of instruments located in the drywell, wetwell, or reactor building can degrade prior to core damage. Severe conditions in the reactor building can result from containment failure or failure of the vent system during containment venting. Steam condensation on instrument components in the reactor building could also affect instrument availability.

Radiation would become important when instrumentation is required for monitoring that may extend weeks or months beyond the initiating event. Instrument components located in the reactor building could be particularly susceptible since these component are qualified to relatively low dose limits. These dose limits could be exceeded in a few hours during a core melt accident where containment failure occurs.

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DIFFERENT METHODS FOR CORE DAMAGE ASSESSMENT (CDA)

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P-O Aronsson, B Bjurman, G Löwenhielm

Background

Due to national regulations, all Swedish nuclear power plants are provided with filtered containment venting. Strategies for severe accident management have also been developed and implemented. A question of greatest importance for adequate accident management is the extent of core damage. The answer to that question may influence actions to be taken, in the short term as well as for long term actions, and it is also essential when estimating environmental consequences of an automatic or deliberate containment venting.

Approach

The objective of the strategy for core damage assessment (CDA) is to obtain information as soon as possible and to continue the assessment until stable conditions are established.

It is recognized that methods available immediately after the onset of an accident are mainly indicative and fairly imprecise. More detailed and accurate information can be obtained in the longer time span.

CDA-methods

a) **Process parameters**

Information on process parameters is readily and immediately available in the main control room. The instrumentation is mainly intended to inform the operator about the status of various process and safety systems. It will, however, also indicate if conditions that may result in core damage have prevailed or not.

To correlate values of important process parameters with various degrees of core damage is not easy or straightforward. In this project we have relied upon judgement of experienced accident analysts. A conservative approach has been chosen, where values on process parameters are given, which will not lead to core damage with a reasonably high degree of probability.

The main parameters of interest are the water level in the reactor vessel and the coolant flow through the core.

In BWRs both of these parameters are monitored. As an example for BWRs it is assumed that if at least half of the main circulation pumps are working and the water level is at half of the core height, or higher, then the core is cooled and the risk of severe core damage is small.

In the Swedish PWRs the water level in the reactor vessel is not monitored and therefore the monitored level in the pressurizer tank is used as an indication. If at least one reactor coolant pump or two safety injection pumps have been in operation the risk of severe core damage is assumed to be a small .

For PWR the core temperature, monitored through the Core Exit Thermocouples (CETC), can be used as an indication of possible core damage. CETC will give quick and direct information on the temperature at core outlet. This will indicate if temperatures has been reached which can result in core degradation. The CETCs may also be capable of indicating "hot" regions in the core. If the CETC displays more than 900 °C, and at least parts of the core is suspected to have been uncovered for more than 5 minutes, a severe core degradation can be at hand.

b) **Doserate in the containment**

The containment radiation monitors (CRM) measure total dose rate from radionuclides within the containment. During severe accident conditions, with considerable core damage, the main contribution to the recorded dose rate comes from airborne activity. They can be used for estimation of the fractional release of gaseous activity from the fuel. The measuring range is adequate even if all gaseous activity is released from the fuel. To estimate the fractional release of noble gases, using the CRM reading, the total core inventory as well as the relative abundance of different gaseous nuclides and the detector response for each nuclide must be known in advance.

The core inventories of fission products are calculated for full power steady state conditions. If the power level differed from this case prior to the accident, a power correction factor must be applied to maintain the accuracy of the method.

In practice the CRM reading, multiplied by a power history correction factor, is compared with precalculated data in a diagram showing the CRM reading as a function of time after scram for four different fractional releases from fuel. Each type of detector has it's own diagram.

Any aerosol or particulate activity deposited on the detector shielding, or on a surface nearby, can cause faulty readings and an overestimation of released noble gases.

c) **Activity content**

In Swedish LWRs a post accident sampling system (PASS) is installed. Samples can be drawn from the primary system (liquid phase) and different compartments in the containment (gas and liquid phase). The activities of various radionuclides of interest are measured to give their activity concentration in the sample volume. To determine the activity content in the sampled compartment the total mass (or volume) from which the sample is drawn also must be evaluated. Knowledge of certain plant parameters, e g the water level in the containment, is essential in doing this.

Before the fractional relaease from the fuel is calculated some corrections are made. The core inventory, based on an equilibrium core at the end of a cycle, is corrected due to actual power history. The results from the nuclide specific activity measurements are corrected due to ingrowth of activity from parent nuclides and corrected for decay to give the activity at the time of scram.

Radionuclides to be measured are grouped in advance according to type of core damage and time period when they are present. Based on theoretical calculations and experimental results found in the litterature, the degree of core damage correlating to the measured fractional releases of different radionuclide groups is established in advance. This may be used as a rough guideline when interpreting the actual result of the PASS measurements.

d) **Hydrogen concentration**

Above temperatures of approximately 800 °C, the zirconium in the core region reacts with the water vapor to generate hydrogen. By measuring the hydrogen concentration in the containment (either by online monitoring with gas chromatograph, taking gas samples for manual analysis or by continuous monitoringing with hydrogen sensors) it is possible to estimate the total amount of released hydrogen. Knowing the total inventory of metallic Zr in the core region and assuming that all gaseous hydrogen comes from Zr oxidation, the fraction of the Zr inventory oxidized can be calculated.

The fractional amount of Zr oxidized is not a direct measure of the degree of core damage, but can give additive qualitative information on the status of the core.

The MAAP code has been used to obtain a relation between generated hydrogen and fractional release of noble gases for some typical accident sequences. The correlation between the release of noble gases and the generation of hydrogen varies for different types of accidents. By comparing the release of noble gases given by hydrogen data and the direct measurement of noble gases within the containment additional (and hopefully supporting) information may be obtained.

Implementation

The doserate levels in the containment, measured by CRM, and the activity content, using PASS, are the two main approaches for CDA. The other two methods, monitoring of process parameters and measuring of hydrogen concentration, can be regarded as complementary. Each method has been transferred into step by step instructions, specific for each reactor unit.

The main methods are aiming at determining the released fraction from the core of different radionuclides (or group of radionuclides). This quantitative information can then be translated to qualitative information on the degree of core damage.

The core inventory of different nuclides is usually given in tables, valid at the end of a full power operating cycle. Thus the inventory has to be corrected for the actual operating history prior to the accident.

One goal has been that all calculations and corrections should be possible to execute by hand. In this way the CDA will be independent of the electrical power supply and computer equipment. It is also easier for the personnel to check for consistency of the data during the evaluation process. Therefore relatively simple correction methods have been developed, without adding unacceptable uncertainty to measured values.

A future development will be to implement the CDA-methods on a Personal Computer which will facilitate easier and faster data handling and provide a handy and comprehensive display of results.

CONTAINMENT RADIATION MONITORS

Each monitor consists of:
1 Self Powered Gamma detector (SPG)
1 Ion Chamber (IC)

Measuring range:
SPG $10 - 10^{6}$ Gy/h
IC $10^{-4} - 10^{2}$ Gy/h

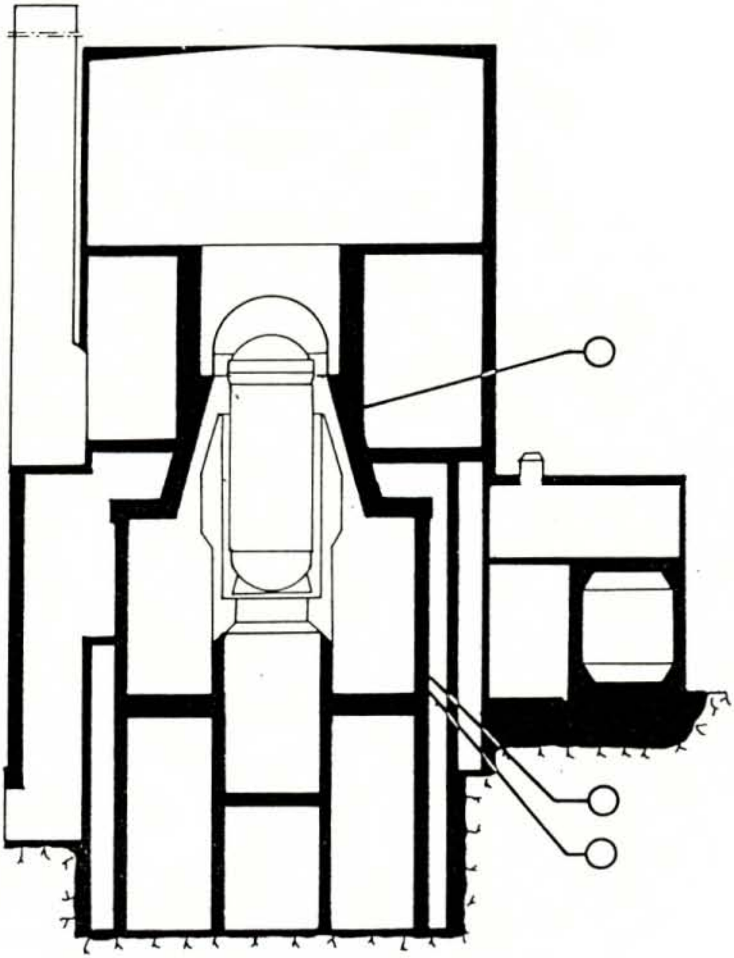

FIGURE 1
Containment Radiation Monitor positions in a BWR.

CONTAINMENT RADIATION MONITORS

Each monitor consists of:
1 Self Powered Gamma detector (SPG)
1 Ion Chamber (IC)

Measuring range:
SPG $10 - 10^{6}$ Gy/h
IC $10^{-4} - 10^{2}$ Gy/h

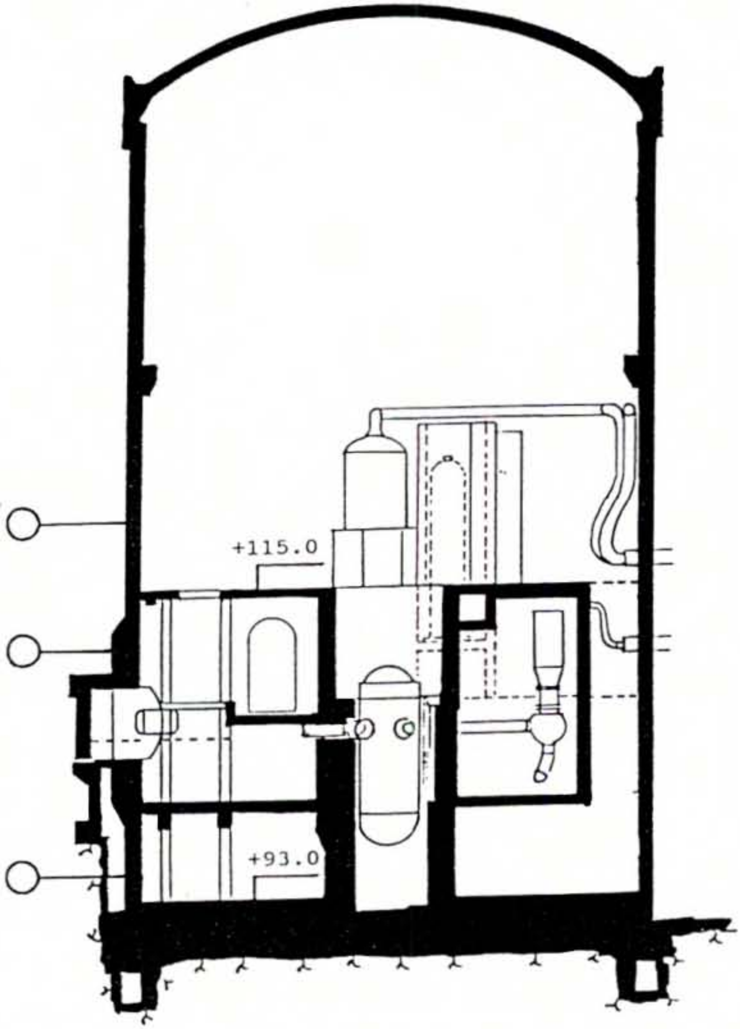

FIGURE 2
Containment Radiation Monitor positions in a PWR.

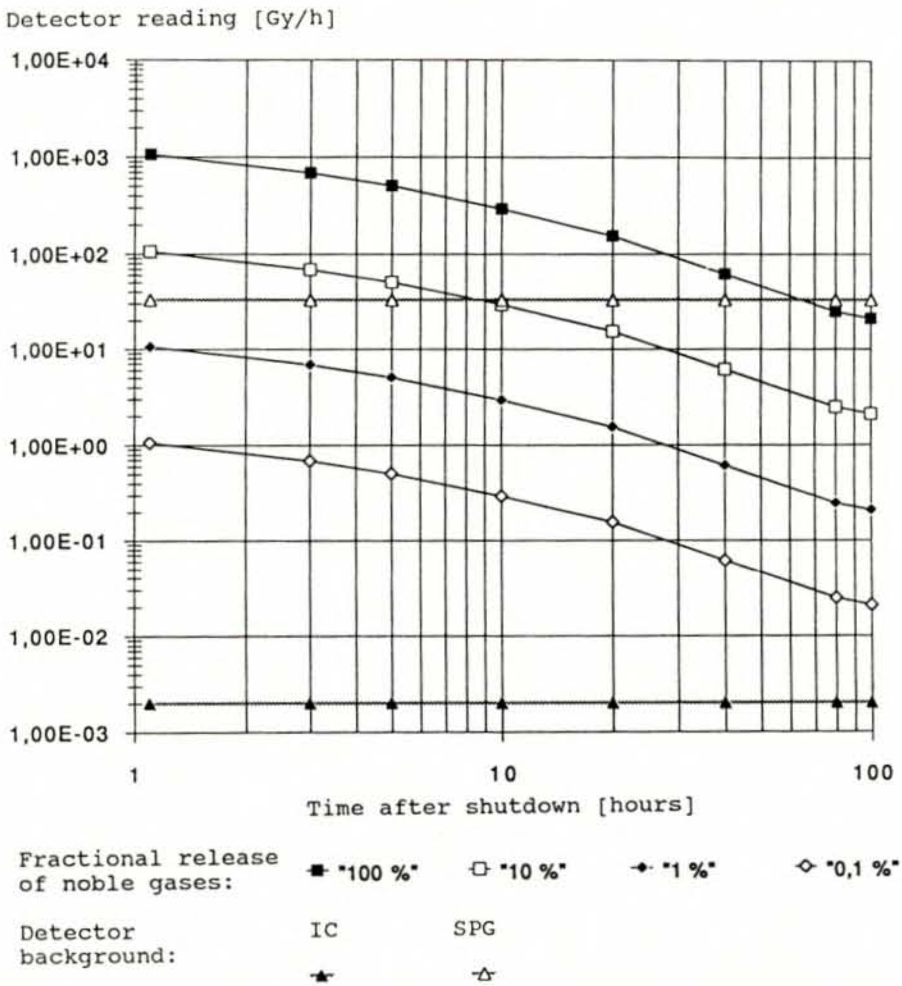

FIGURE 3
Precalculated CRM readings for different fractional releases of noble gases. The actual CRM reading are plotted in the diagram in order to assess the magnitude of the noble gas release.

POST ACCIDENT SAMPLING SYSTEM (PASS)

Ⓦ = WATER SAMPLE Ⓖ = GAS SAMPLE

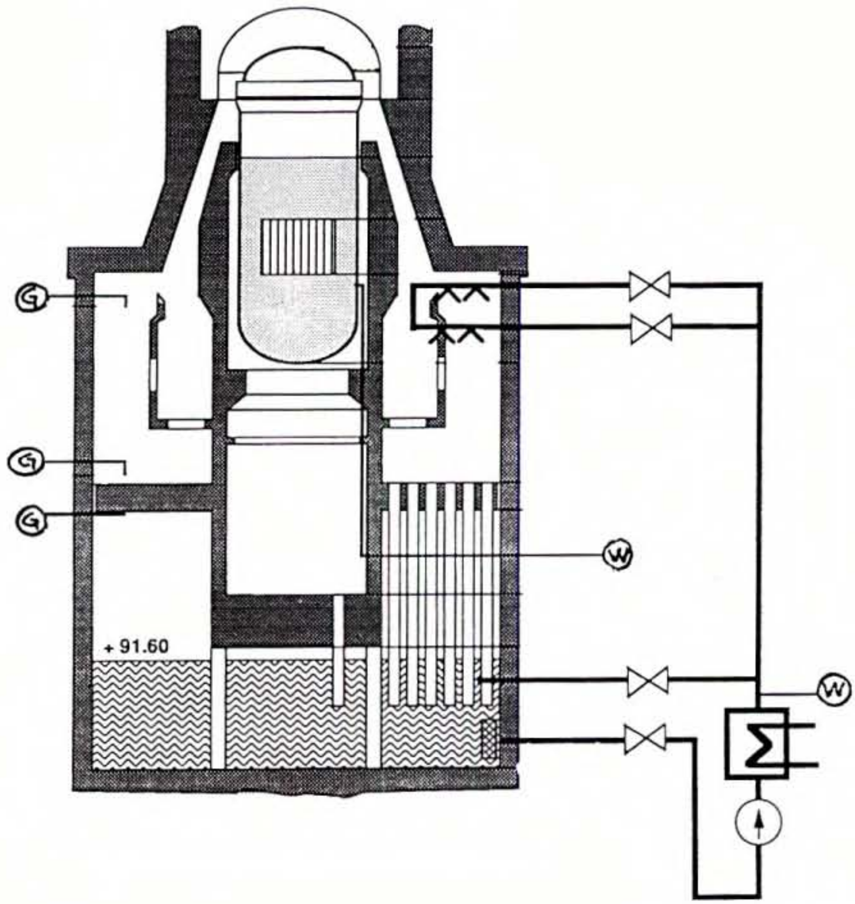

FIGURE 4
Sampling points in Post Accident Sampling System in a BWR.

POST ACCIDENT SAMPLING SYSTEM (PASS)

W = WATER SAMPLE G = GAS SAMPLE

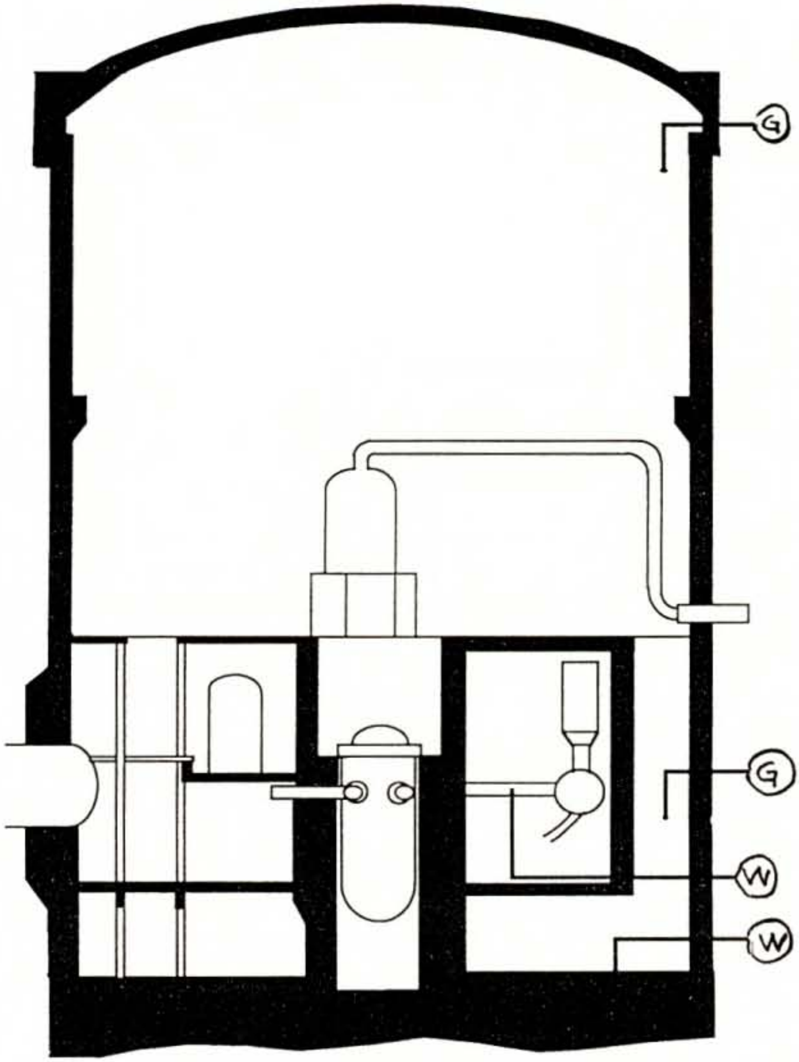

FIGURE 5
Sampling points in Post Accident Sampling System in a PWR.

ACTIVITY IN THE CONTAINMENT

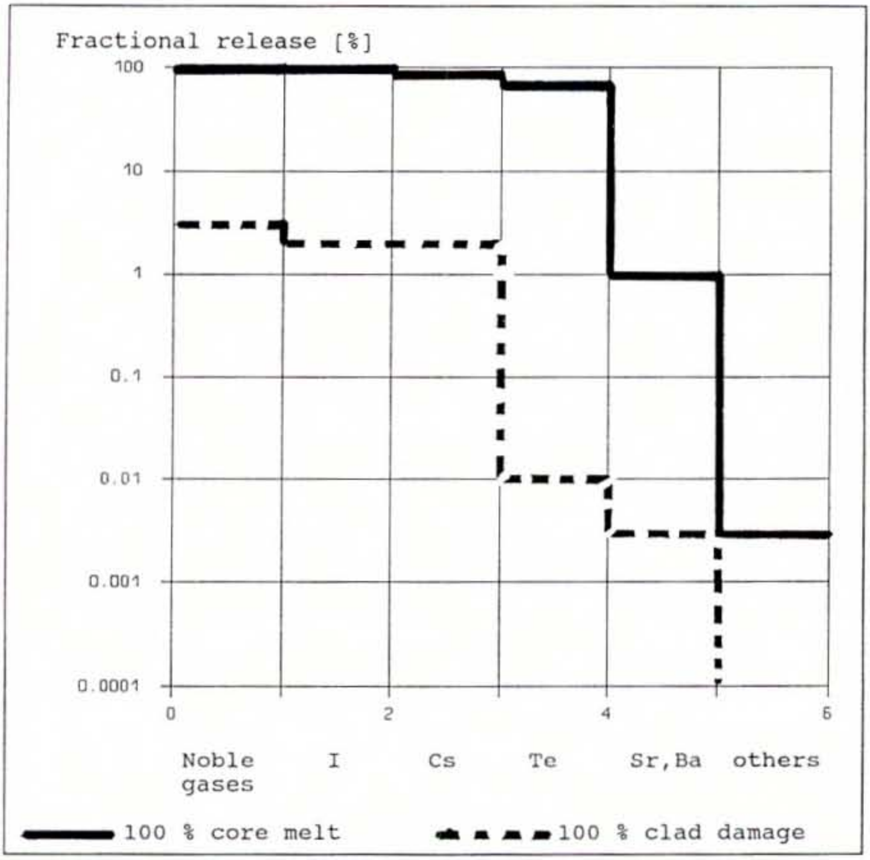

FIGURE 6
Anticipated fractional release for different groups of radionuclides. The calculated fractional releases (from measurement of PASS samples) are plotted in the diagram in order to assess the magnitude of core damage.

ACTIVITY RATIOS IN FUEL

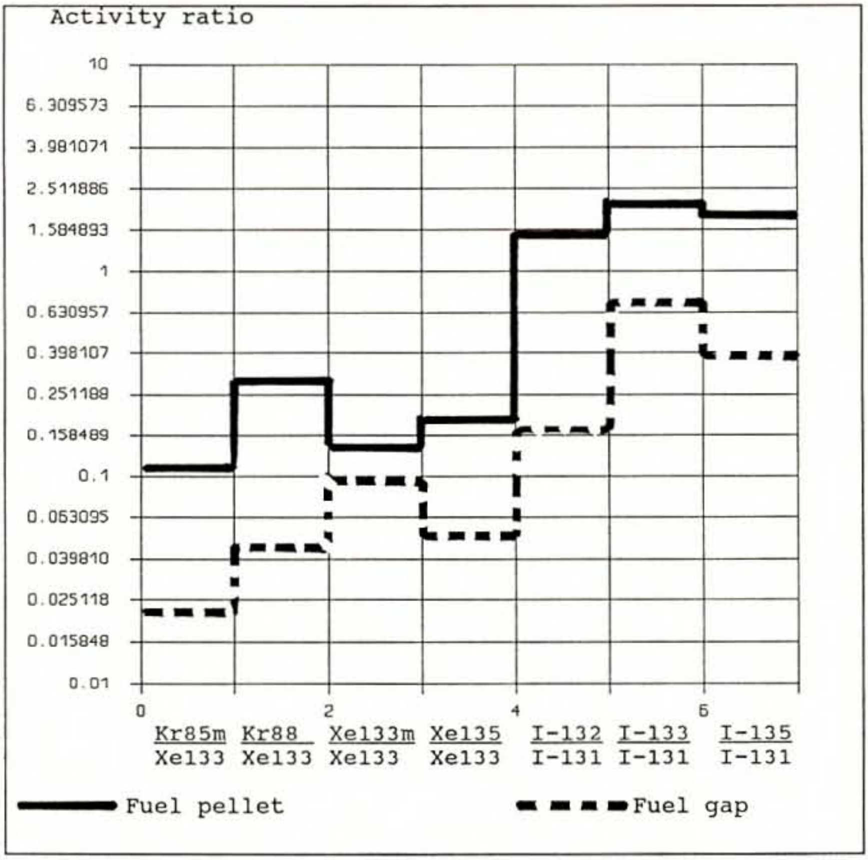

FIGURE 7
Anticipated activity ratios within noble gases and iodines existing in the fuel pellets and the fuel gaps under normal operation. The calculated activity ratios (from measurement of PASS samples) are plotted in the diagram in order to assess the type of core damage.

HYDROGEN SENSORS

Each monitor consists of:
1 Hydrogen sensor
1 Temperatur sensor

Measuring range:
0 - 10 % volume
0 - 200 °C

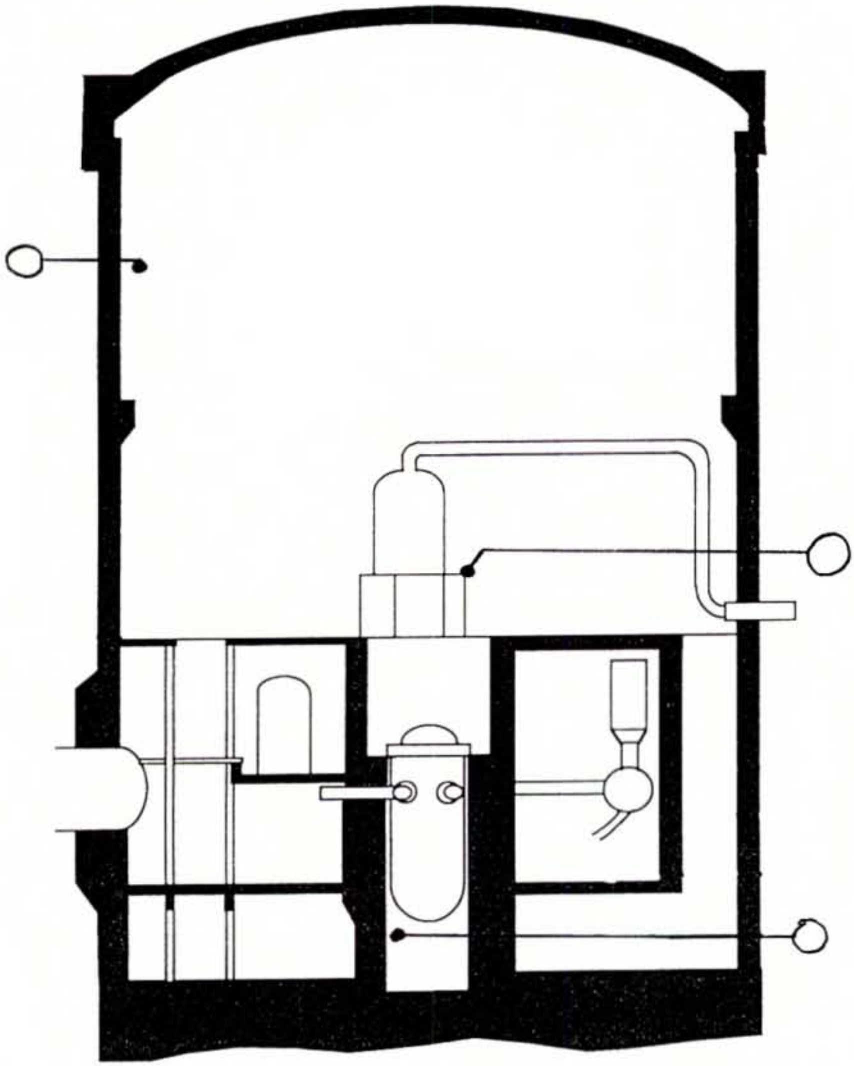

FIGURE 8
Hydrogen sensor positions in a PWR.

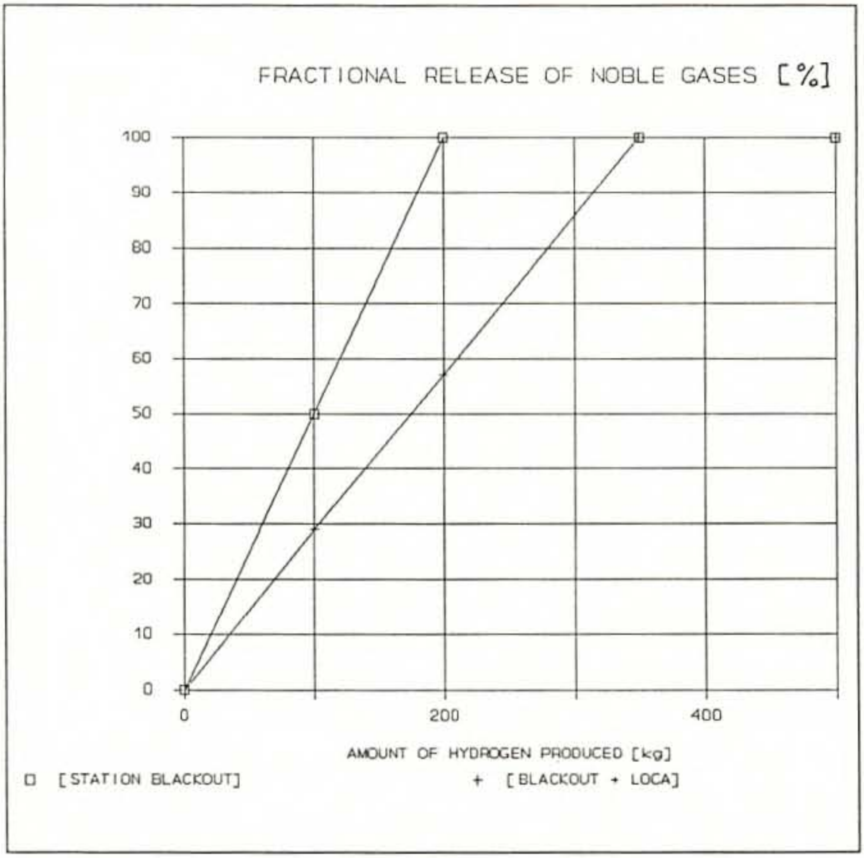

FIGURE 9
Fractional release of noble gases as a function of hydrogen produced for two types of accidents, station blackout and station blackout + LOCA. The calculated value for hydrogen content in the containment (from measured hydrogen concentration by PASS or hydrogen sensors) is plotted in the diagram. The corresponding value for fractional release of noble gases can be compared for consistency with measured fractional releases.

<u>OECD (NEA) CSNI SPECIALIST MEETING ON</u>
<u>INTRUMENTATION TO MANAGE SEVERE ACCIDENT</u>

GRS, Cologne, Germany
16th 17th March 1992

INSTRUMENTATION CAPABILITIES DURING THE TMI-2 ACCIDENT AND IMPROVEMENTS IN CASE OF LP-FP-2

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1. Introduction:

The accident in Three Mile Island Unit 2 (TMI-2) from March 1979 have shown clearly the need for special accident instrumentation. During the first two hours of the accident there was enough time and possibilities to bring the reactor better under control. Nevertheless the accident evaluation program conducted in the USA together with CSNI efforts have led to a better understanding of severe accidents and how to prevent or mitigate them. Accident management procedures now developed or under development depend to a great extent on reliable measurements and other operator aids during the accident.

In this paper a brief account is given on the behavior of the instrumentation during the TMI-2 accident and lessons learned and applied to prepare, conduct and analyze the OECD-LOFT severe core damage experiment LP-FP-2.

2. TMI-2 Measurements

About ten years after the TMI-2 accident reliable results of the accident evaluation program carried out by EG&G Idaho were published. Analysis of available measurements, plant inspections and examination of material specimens were used to determine a consistent scenario of the accident as descreibed by J.M. Broughton et al. /1/.

The TMI-2 joint task group of the CSNI (Committee on the Safety of Nuclear Installations) coordinated an international TMI-2 material examination task and a TMI-2 analysis exercise which started in 1985. The TMI-2 analysis exercise final report was published in 1991 /2/ and included 12 different calculations from 8 OECD countries. The data summary report /3/ prepared by EG&G, Idaho for the analysis exercise included a list of about 300 recorded measurements which were selected out of 3000 measurements available. The selected measurements were found to be of interrest for the analysis exercise. Only 100 measurements were reviewed and commented

- 35 temperature measurements located in different components of the plant but no one inside the pressure vessel. Most of these temperature measurements are qualified. There were 50 in core thermocouples recorded but they were not reviewed and consequently not published. Information deduced from the behavior of these thermocouples is used in reference /1/ in predicting the accident scenario as shown in fig 2.
- 32 nuclear radiation measurements most of them were commented as trend measurements. Three source range power level measurements were qualified with a given uncertainty.
- 9 flow rates 8 of them were estimated values and only one measurement (letdown cooler volumetric flowrate) was qualified with an uncertainly of ±25%.
- 6 presure measurements which can be seen as the most reliable measurements available from TMI-2. They are a part of the 24 parameters monitored by the reactimeter which were used as a reference baseline for other data sources.
- 3 level measurements (pressurizer, SG-A and SG-B) which are also a part of the reactimeter parameters.
- 6 binary measurements corresponding to the 4 primary pumps and two pressurizer valves.

3. Important Events Indicated by the Measurements:

More than ten years after the accident and through the huge effort spent to analyse measurements and materials from TMI-2, it is now easy to deduce some events from the measurements available as given in reference /1/. From the system pressure behavior together with five binary measurements, it is easy to identify in figure 1, the initial phase as a Small Break Loss Of Coolant Accident (SB-LOCA).

The rapid increase in pressure after closing the leak (block valve closed at 139 min.) was due to a core heat up phase. Accident management started in TMI-2 with the pump transient at 174 min. (nearly three hours after reactor scram) and stopped the rapid increase in pressure. This was followed by a cooldown phase through a primary bleed and feed using the pressurizer block valve and the High Pressure Injection (HPI).

The influence of the pump transient to stop the initial core heat up is verified through the quench of the core outlet thermocouples as shown in figure 2.

The relocation of molten corium into the lower plenum at 224 min. is indicated by the increase in the count rate of the source range monitor (SRM) shown in figure 3. The response of the SRM is a direct indicator of neutron leakage which increased suddenly as the molten fuel has dropped into the unschielded lower plenum. The increased system pressure and cold leg temperatures may be also due to the increased steam production at that time.

The long time behaviour of the measured SRM during 25 hours after scram is shown in figure 4 which indicates redistribution of molten corium after the relocation at 224 min.

No direct indication for hydrogen generation is available from the published TMI-2 measurements

The calculations carried out in connection with the international analysis exercise have shown large uncertainties in the start of H_2 generation, rate of generation and total amount. Calculation results published in reference /2/ varies, depending on code and also on code user. The total amount calculated varied between 50 and 480 kg while the uncertainty in timing is about ±10 min.

Henrie and Postma spent big effort to analyse recorded data and examined damaged as well as surviving components to determine hydrogen generation, hydrogen concentration and quantities in the different components. Results of their estimations are summarized in figure 5 which was published in reference /4/. The total value of 460 kg agrees with the value calculated using MAAP-DOE /2/.

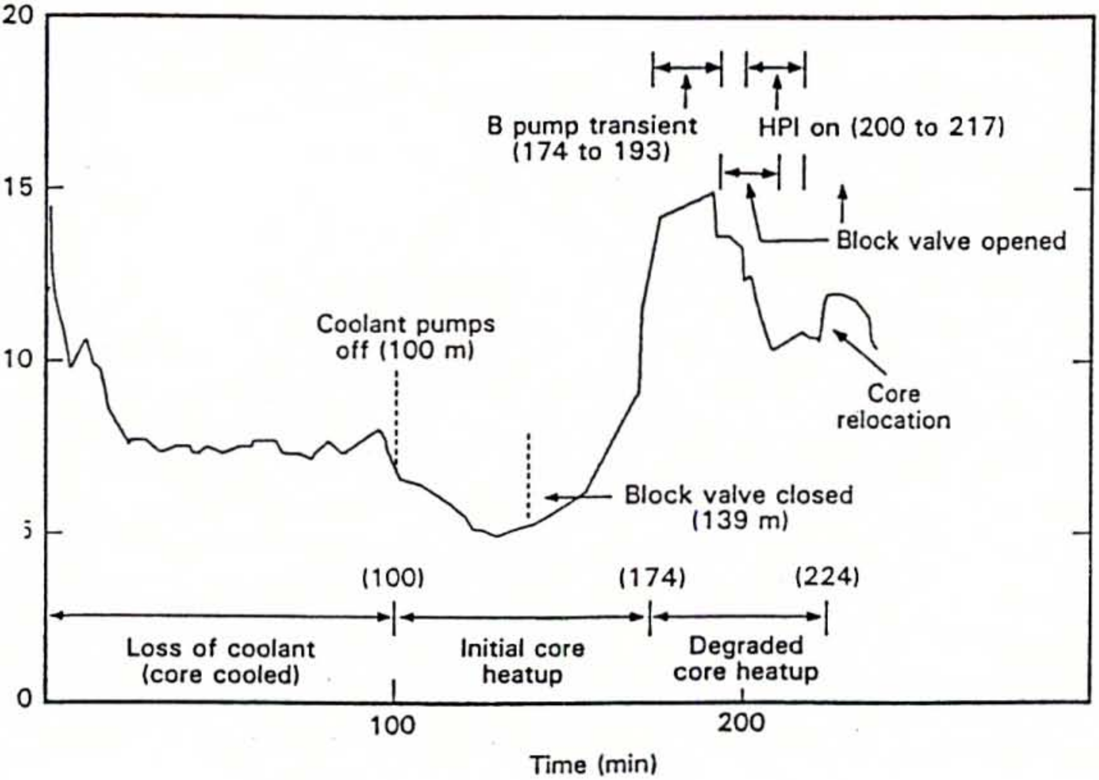


Fig. 1: The TMI-2 RCS pressure history

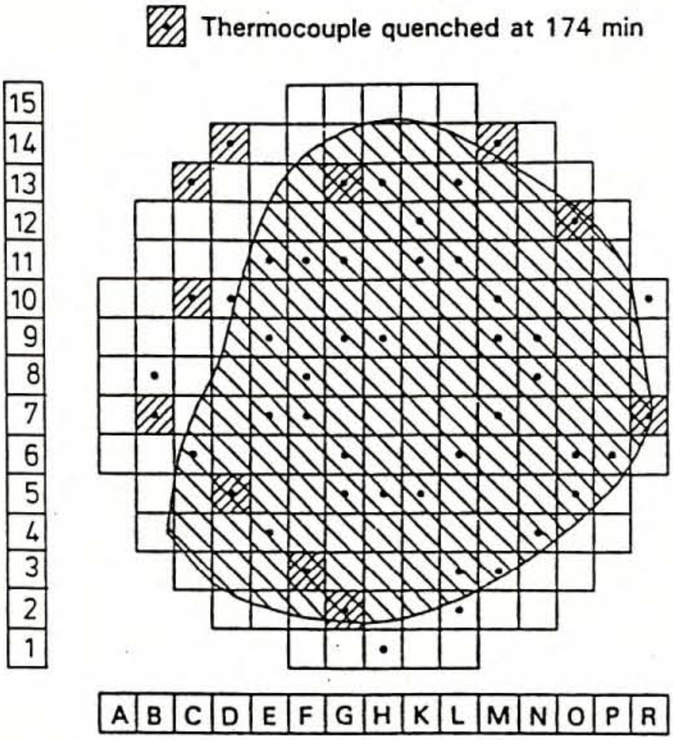


Fig. 2: In-core thermocouple after pump transient (175 min).

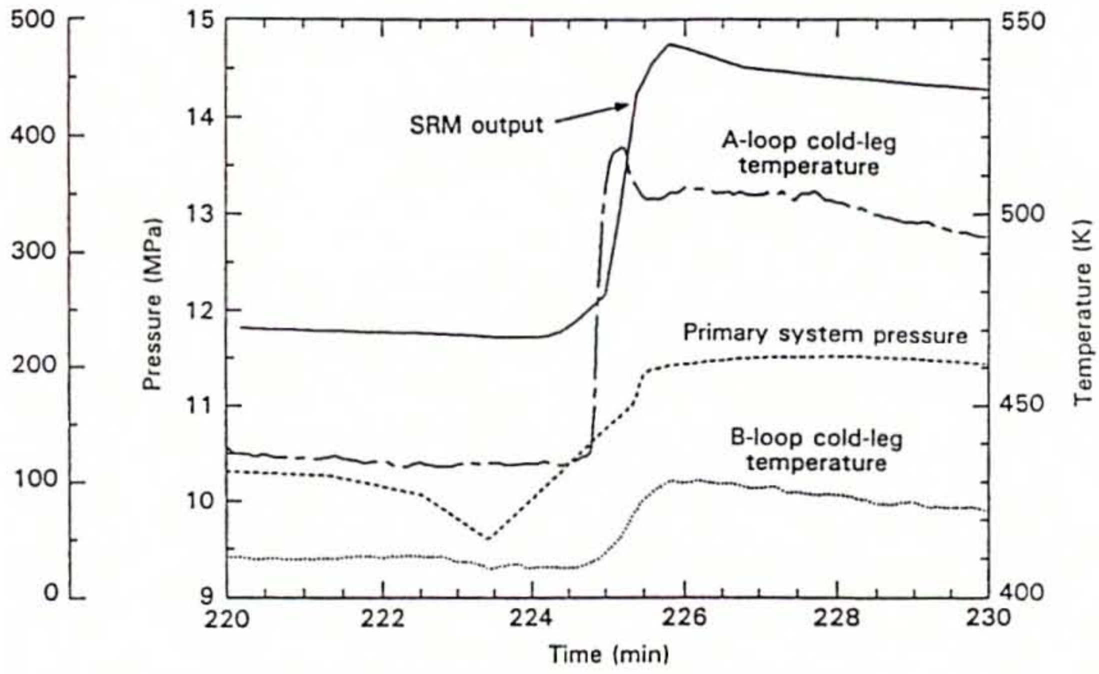


Fig. 3: Neutronic and RCS instrument response between 220 and 230 min, indicating core relocation between 224 and 226 min.

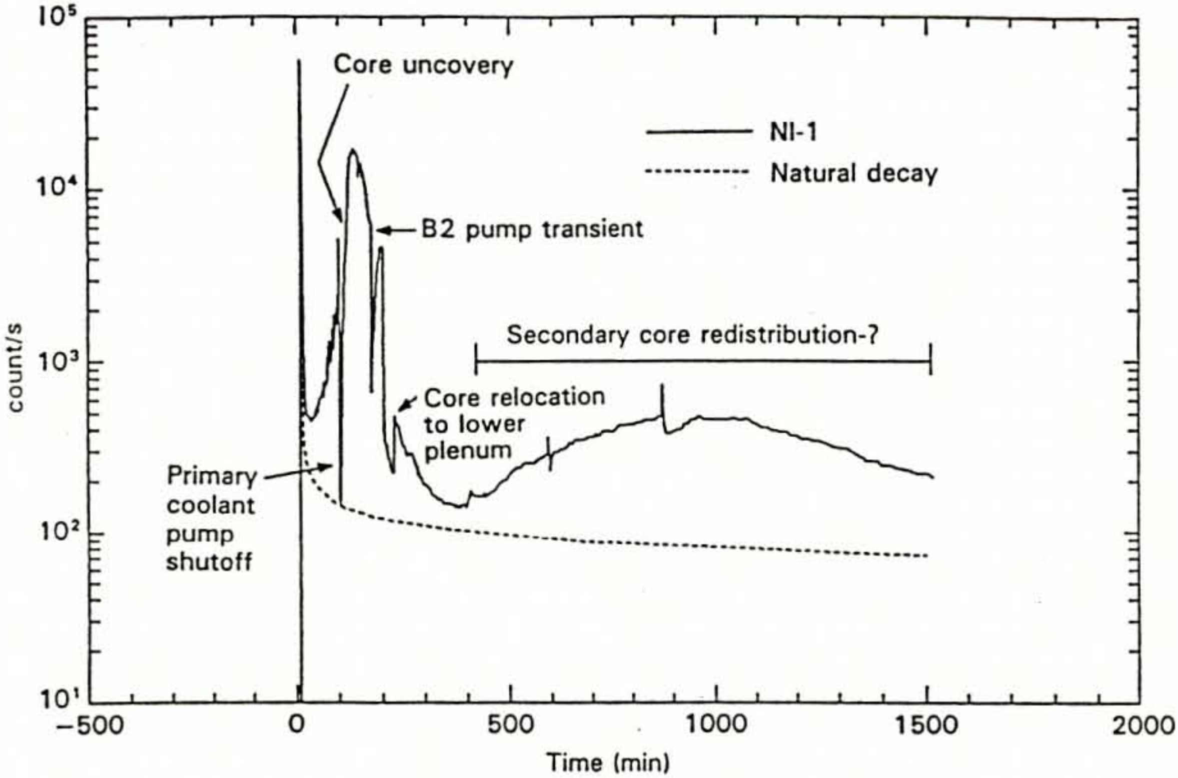


Fig. 4: Core source range neutron flux history

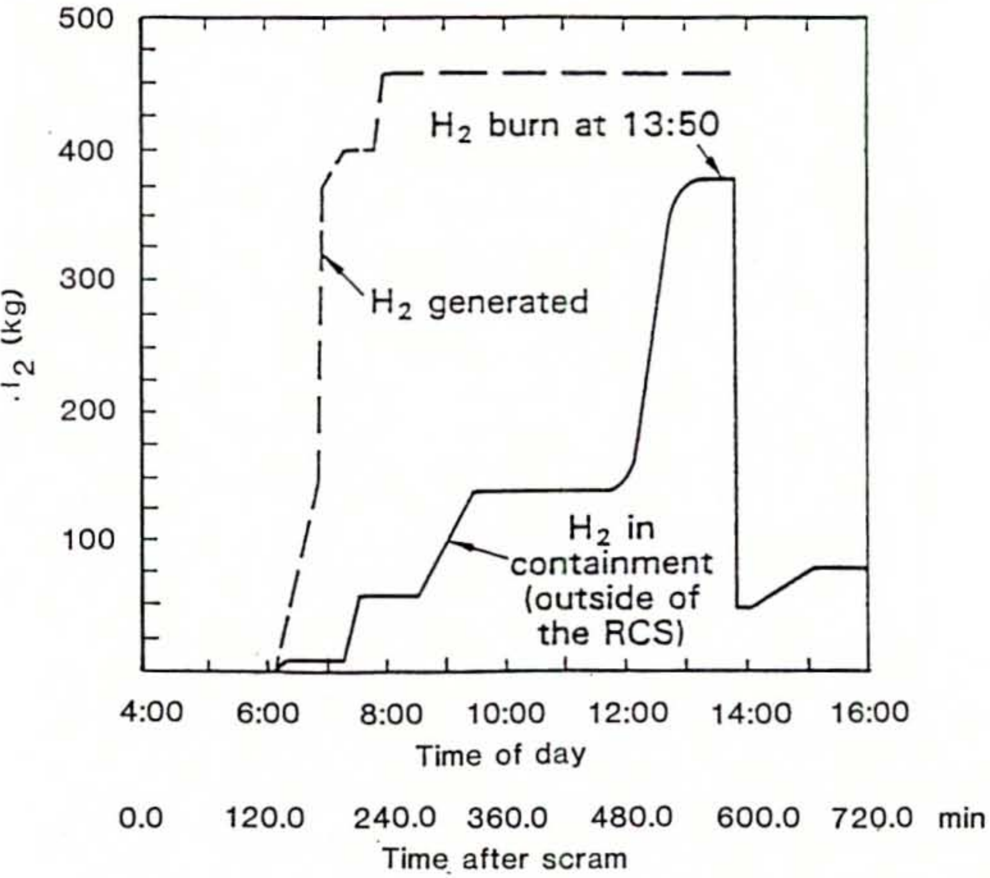


Fig. 5: TMI-2 hydrogen generation and accumulation in containment as estimated by Henrie and Postma /4/

4. LP-FP-2

The last experiment in the OECD LOFT Project LP-FP-2, conducted on Juli 9, 1985, was a severe core damage experiment. It simulated a LOCA caused by a pipe break in the Low Pressure Injection System (LPIS) of a four-loop PWR as described in /5/. The central fuel assembly of the LOFT core was specially designed and fabricated for this experiment and included more than 60 thermocouples for temperature measurements. A Fission Product Measurement System (FPMS) beside the normal LOFT instrumentation was designed and fabricated for the detection, identification and collection of radioactive isotopes and hydrogen in the LOFT Primary Coolant System (PCS), LPIS and Blowdown Suppression Tank (BST). The FPMS system consisted of three basic subsystems:

a) Four gamma spectrometer systems and one gross gamma detector.

b) A deposition sampling system and

c) Filter sample systems.

The filter sample systems included three sample lines F1, F2 and F3 which were designed to provide a continuous sample of the vapor and aerosols generated during the heat up phase of the experiment. For example the F1 sampling line (fig. 6) consisted of the following major components:

1. Sample line probe placed above the CFM
2. Argon dilution gas supply
3. Dual cyclone separator/isolation valves
4. Dilution filter
5. Virtual impactor
6. Collection filters
7. Infrared moisture detectors
8. Hydrogen recombiner.

This example shows the complexity of the FPMS. Experience available in EG&G Idaho from TMI-2 analyses and from the PBF severe fuel damage scooping test conducted in October 1982 were utilized in the design, conduction and analyses of this experiment.
LP-FP-2 costs was $ 25 million out of $ 100 million for the whole OECD LOFT project. Not

all phenomena occured was explained, futher studies could still lead to new interpretations of the data.

More details about different measuring systems and their behavior during the experiment are given in reference /5/.

In order to achieve the predefined thermal-hydraulic boundary conditions for the first few minutes of a core damage accident an unusual LOCA consisting of two simultanious breaks (ILCL and LPIS) and discharge through the PORV was conducted in LP-FP-2. The principal sequence of events during the experiment is shown together with the primary system pressure history in figure 7. Analyses of the experimental data carried out and published in the Experiment Analysis and Summary Report (EASR) /5/ can be used to show the different stages of a severe accident. Some phenomena can be identified through redundant and diverse measurements as summarized in the following section.

5. Identification of PWR sever core damage phenomena from LP-FP-2 measurements

5.1 Movement of Water Level

Several measuring methods were developed specially in LOFT to indicate the formation of a water level and some of them can be used to follow the movement of this level in the pressure vessel. These measurements can be used to initiate appropriate accident management measures.

The first indication of void formation in the primary loop was the decreas in fluid density measured in the intact loop hot leg and cold leg at 50 an 100 S after scram as shown in figures 8 and 9. In figure 9 the behavior of all three γ-beams crossing the cold leg are shown. The upper beam C indicates that steam is acumulated at the upper region of the tube. The continuous loss of fluid through the leak caused an increase in void in the upper plenum as indicated by the behavior of the thermocouple TE-3UP-011 (Fig. 10) which was located at the same horizontal level of hot and cold-leg connections. This thermocouple indicated departure from saturation at about 475 S. after scram. Further decrease of the water level inside the core is determined by recording the time of departure from saturation of thermocouples at different locations in the core as plotted in figure 12. A diverse method for measuring the water level in the core region can be dedluced from the behavior of the Self Powered Newton Detectors (SPND) which were installed in LOFT in the fuel assembly 4. Fig. 11 shows the behavior of the SPND located nearly in the middle of the core.

Further diversification of water level measurements in LOFT was the conductivity probes as shown in figure 13.

In commercial PWR´s two methods are now in use which are based on differential pressure measurement and heated sensor measurement /6/. The two methods are reliable enough to be used in safety-related systems. In BWR the pressure differential method togehter with thermo comples are in use. Other methods like ultra-sonic /7/ and core cooling monitors /8/ are under development.

5.2 Metal-Water Interaction:

From the analyses of core temperature measurements in LP-FP-2, the rapid increase in temperature shown in fig. 14 was a result of the oxidation of zircalory which became rapid at temperatures in excess of 1400 K. Further examination of such high temperatures measured by thermocomples gave rise to the detection of a cable shunting effect which is defined in reference /5/ as the formation of a new thermocouple junction on the thermocouple cable due to exposure of the cable to high temperature. Experiments were designed and conducted by EG&G Idaho to examine the cable shunting effect. The results of the these experiments indicate that the cladding temperature data in LP-FP-2 contain deviations from true temperature due to cable shunting after 1644 k is reached. This temperature is whithin the range when rapid metal-water reaction occurs. An example of such temperature deviation due to cable shunting is shown in fig. 15.

An important idication of metal-water interaction is the hydrogen generation rate. Two thermal conductivity gas detector were installed in the F1 sample line (fig. 6) for hydrogen measurements. Both instruments registered "over-ranged" and could not be used to determine the hydrogen concentration in the F1 sample line. During the post-experiment phase, the hydrogen concentration in the vapor space of the BST was measured. This measurement was based on the examination of grab samples which were taken 28 days following the experiment. Calculated total amount of H_2 in the BST based on the grab samples was 250 ±11 g. This amount of H_2corresponds to an oxidation of 11.6% of the zircaloy available in the Central Fuel Modul (CMF), however, this result is too small to account for the observed oxidation of the CFM (58%) based on the Post Irradiation Examination (PIE) results mentioned in /9/. An explantation of this discrepancy was found to be a further oxidation of zircaloy during the reflood phase as explained in /10/ and discussed in section 5.3.

5.3 Further Events

More phenomena were detected from the analyses of the recorded behavior of the 60 thermocouples in the CFM together with other tehermocouples and measuring systems in the LOFT nuclear reactor.

After the first indication of metal water reaction at 1430 S. several instruments indicated a common event at 1500 S. These instruments included gross gamma monitor, momentum flux meter in the downcomer, upper tie plate and guide tube thermocouples. This event is believed /5/ to be the rupture of the control rod cladding.

The behavior of the upper tie plate thermocouples after reflood shown in figure 16 was a clear indication of an exothermal reaction between the injected water and the partially molten material in the CFM. Post test analysis /10/ have shown that the oxidation in the melts and other components of the upper tie plate was responsible for the generation of further 818 g of H_2 after reflood. This value was varified by an energy balance and calculated volumes of noncondinsibles in the PCS which determined that 819 ± 364 g of hydrogen was accumulated in the PCS following reflood. The total amount of hydrogen produced during LP-FP-2 experiment was estimated to be 1024 ± 364 g as given in /9/.

Another important phenomenon which was indicated and followed by the available LP-FP-2 instrumentation was the natural circulation in the PCS. A slow secondary cooldown was initiated at 2530 S. PCS loop natural circulatin did not start until 12115 S as shown in figure 17. Narural circulation began when the noncondensible gas bubble in the top region of the steam generator tubes had reduced in size from cooling to allow spillover to occur. The temperature in the SG-outlet plenum started to decrease at about 6000 S. before spillover happened which indicated a reflux mode of cooling in the time between 6000 and 12000 S. When spillover occurs the temperature in the outlet plenum decreases rapidly. Primary to secondary heat transfer also begins and the primary side outlet coolant temperature approaches and follows the secondary coolant temperature as clearly shwon in figure 17.

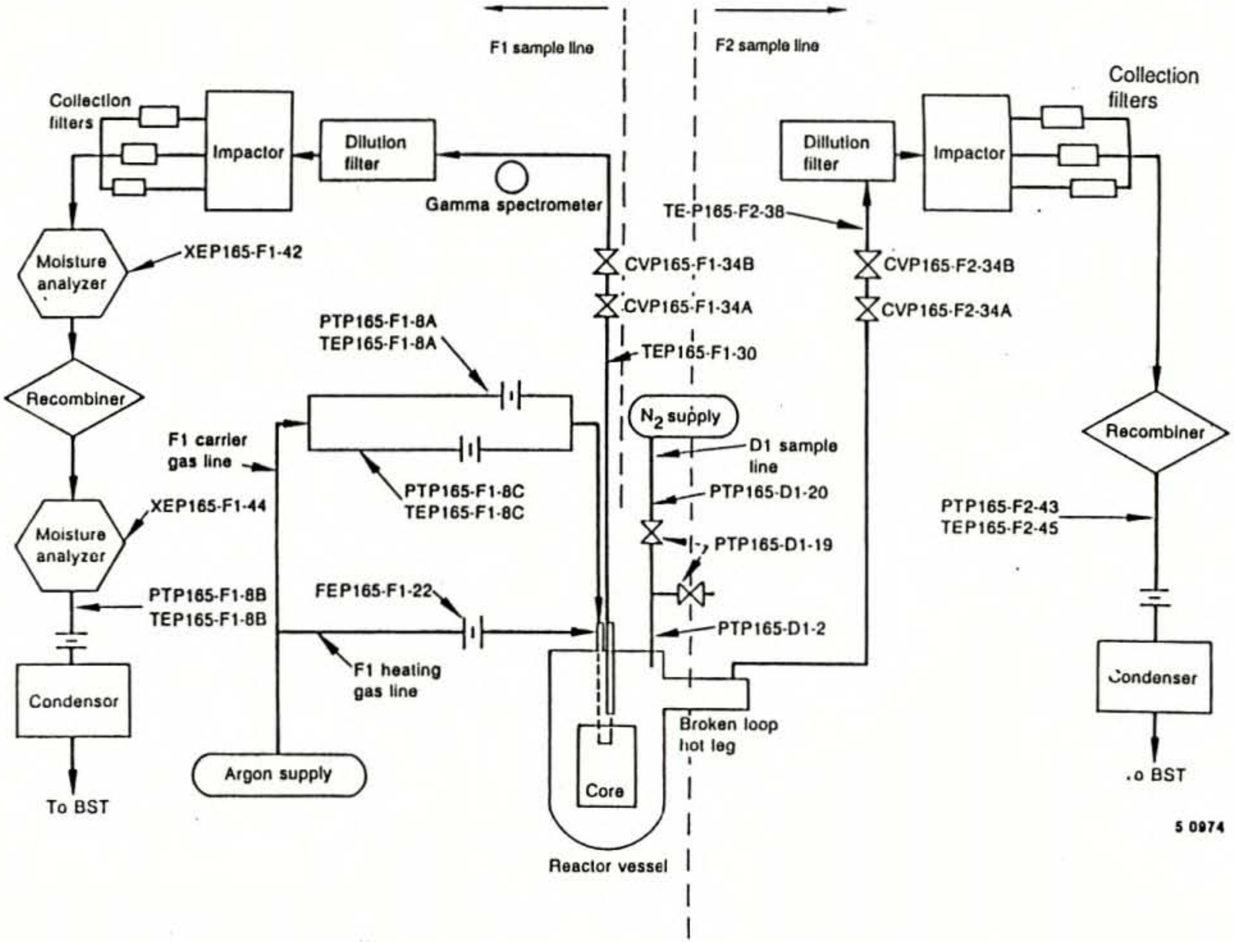


Fig. 6: Schematic of F1 and F2 aerosol sample systems

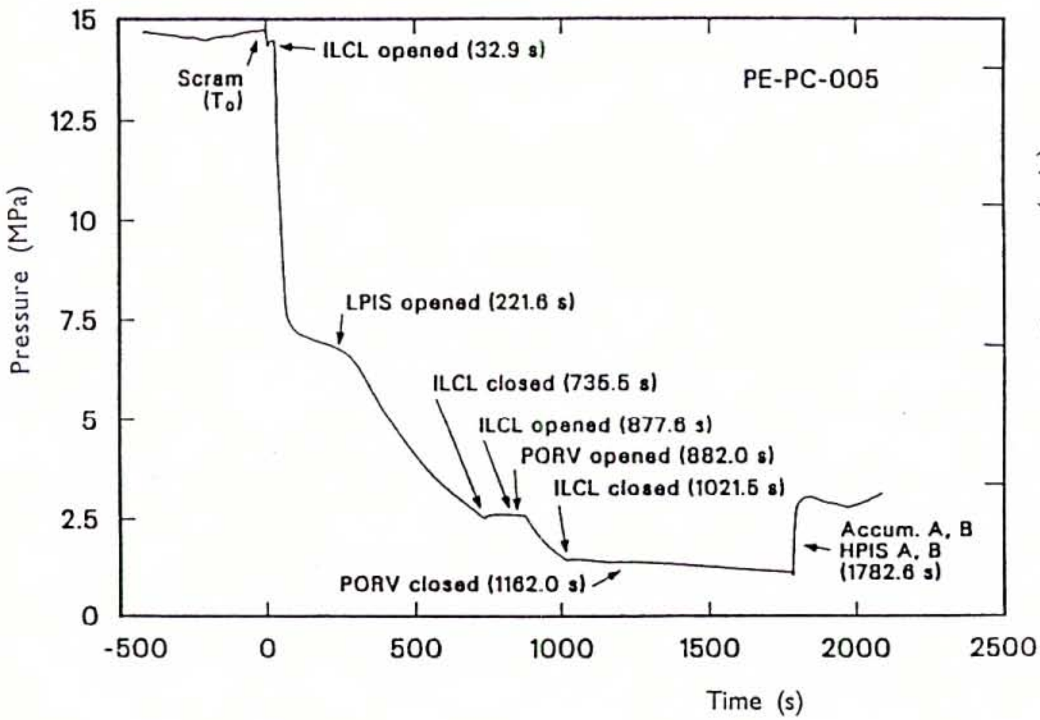


Fig. 7: LP-FP-2 System Pressure

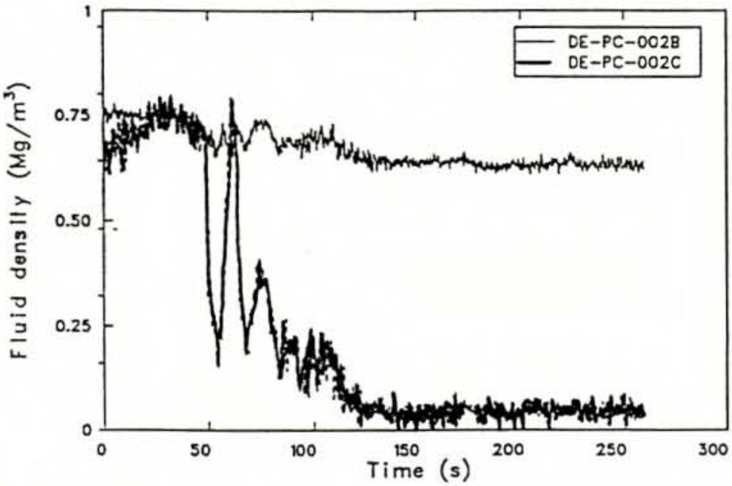
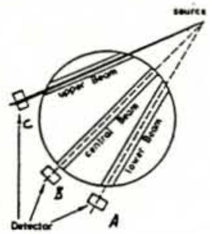


Fig. 8: Intact loop hot leg densities



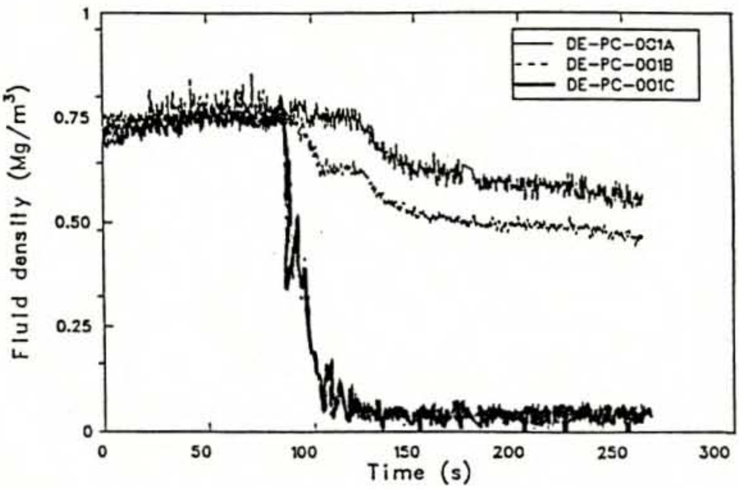


Fig. 9: Intact loop cold leg densities

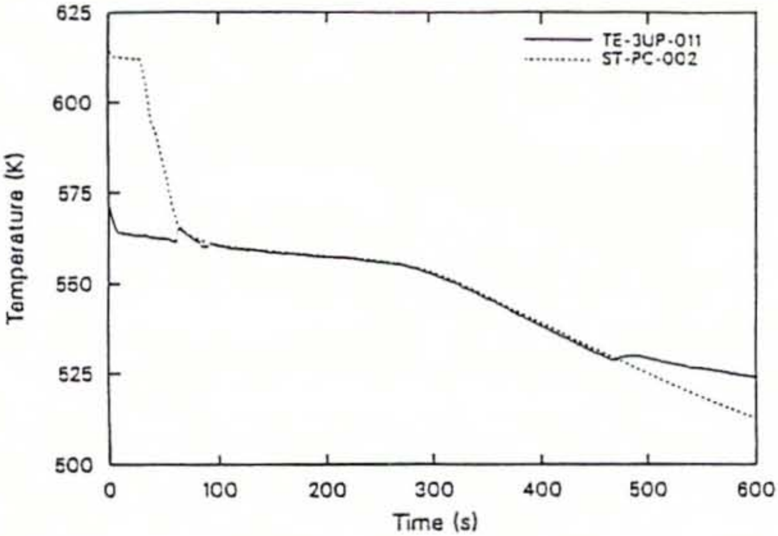


Fig. 10: Comparison of upper plenum fluid temperature with saturation temperature

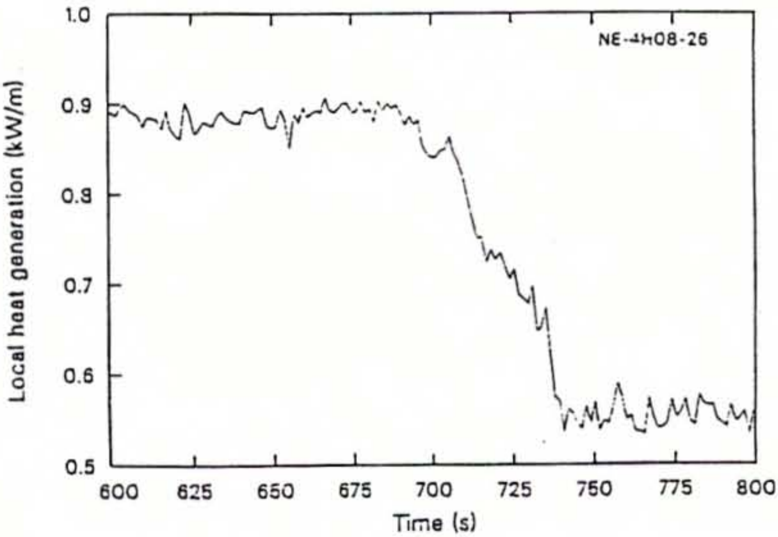


Fig. 11: Response of the SPND in fuel assembly 4 during core uncovery

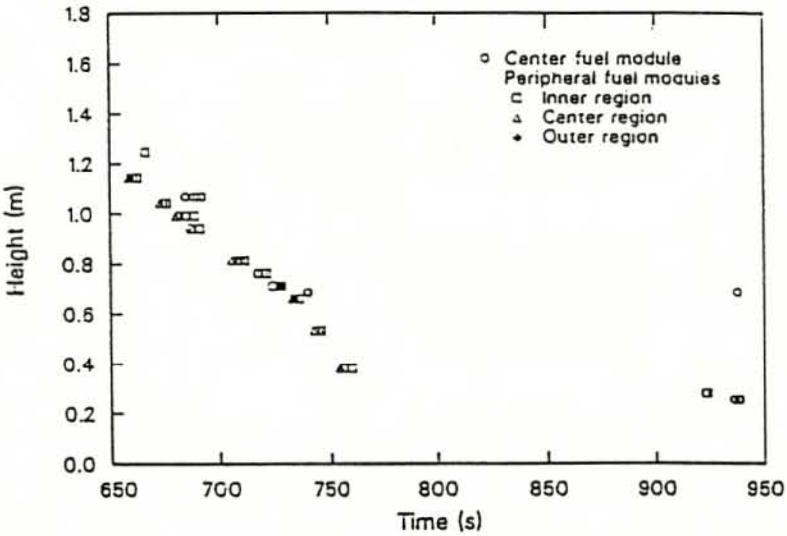


Fig. 12: Core uncovery in LOFT Experiment LP-FP-2 as indicated by thermocouple departure from saturation temperature

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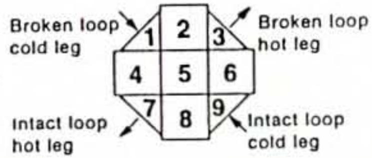

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CHARACTER RANGE TABLE			
( )	1.000	≤ N <	999.000
(O)	.010	≤ N <	1.000
(X)	-999.000	≤ N <	.010

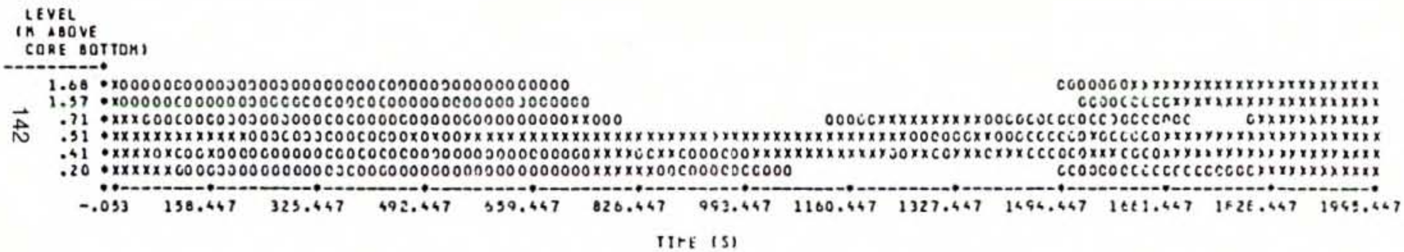

Fig. 13: Conductivity level probe response in Fuel Assembly I

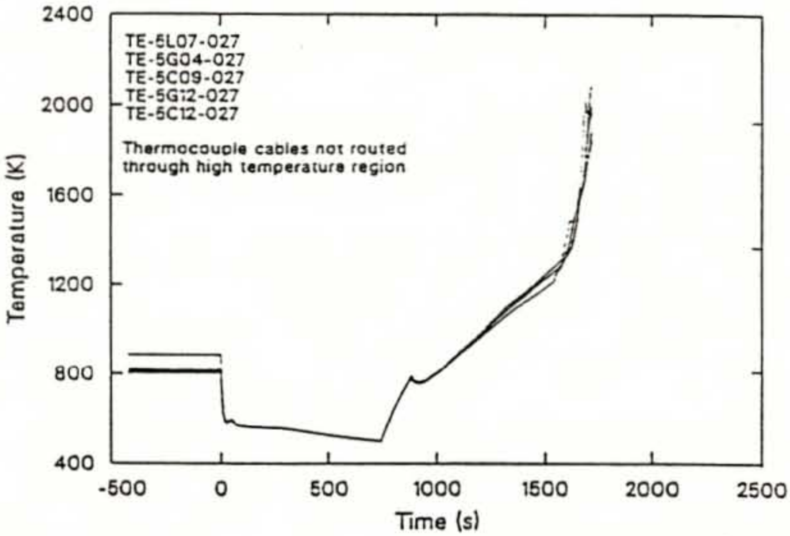

Fig. 14: CFM fuel cladding temperature at the 0.686 m (27 in) elevation

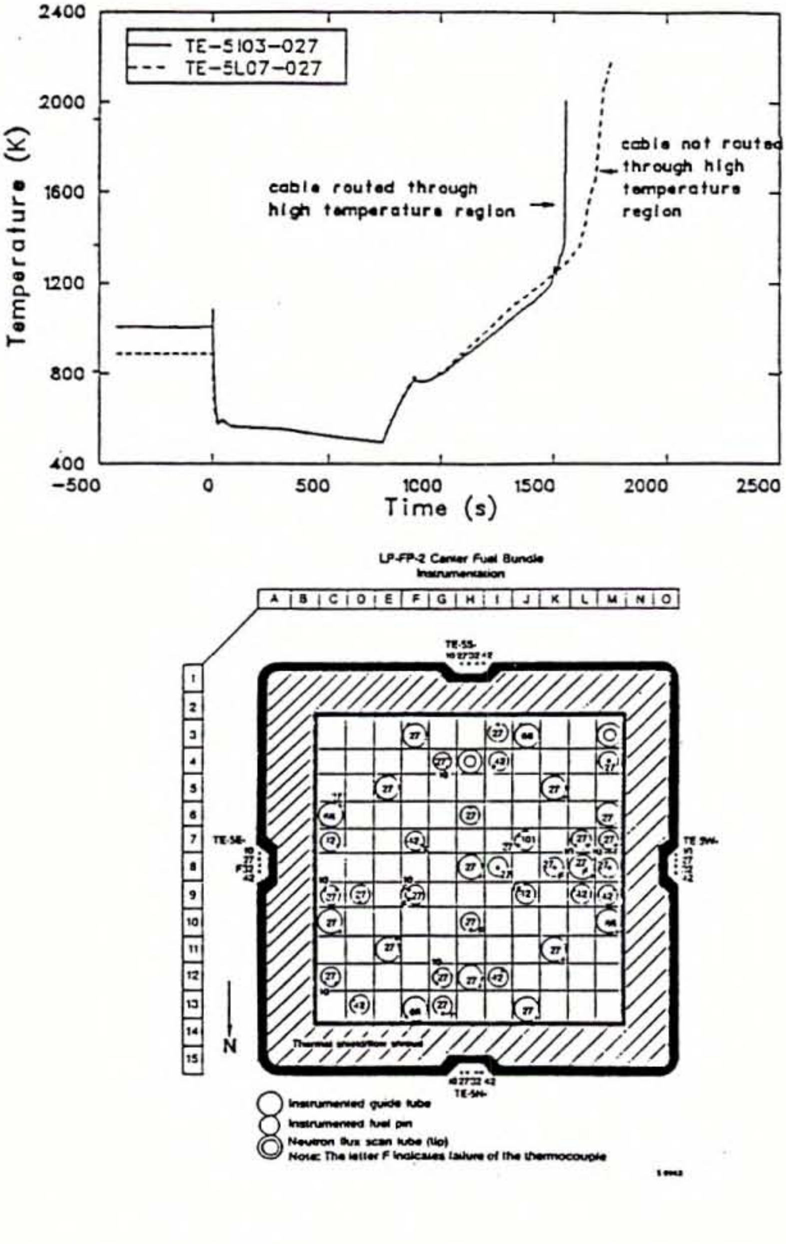

Fig. 15: Comparison of temperature data with and without cable shunting effects at the 0.686 m (27 in.) elevation in the CFM

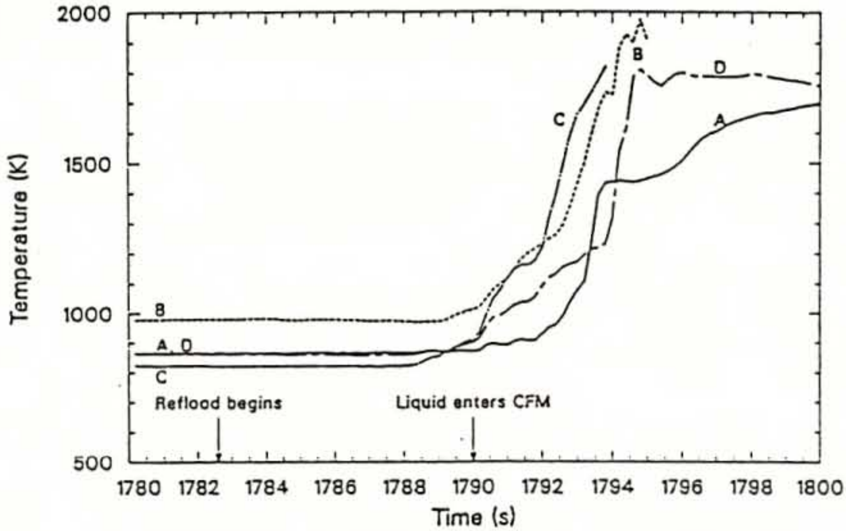

Fig. 16: Upper tie plate temperatures during reflood

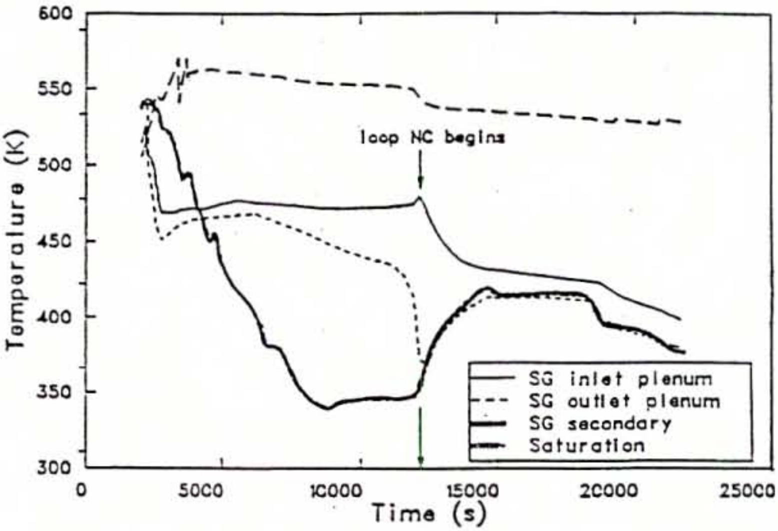

Fig. 17: Initation of natural circulation at 12,115s

6. Conclusions

Both in TMI-2 and LP-FP-2 only few types of sensors were able to withstand the consequences of severe accidents and were able to deliver information for post accident analysis. These were pressure sensors, thermocouples and radiation monitors. Advanced instrumentation technology have proven to be able to utilize these three types of sensors in redundant and diverse instrumentation of Light Water Reactors (LWR) to manage severe accidents.

One important phenomenon in LWR which have to be reliably monitored under accident conditions is the water level in the pressure vessel. Several methods are known and some of them are in use like:

1. Differential pressure method which is in use since several years in many operating plants specially in BWR´s
2. Heated and Unheated resistance thermometers are developed as explained in reference /6/ specially for use in PWR.
3. In core thermocouples which is proven to be realiable as shown in case of LP-FP-2 (fig. 12).
   It is planed to use analogous method in German BWR for AM purposes. Methods based on the use of heated differential thermocouples named BICOTH (Binary Coding Thermocouples With Heater) or TRICOTH (Trinary Coding Thermocouples With Heater) are in use and under development as explained in reference /11/.
4. Other methods based on:
   SPND behavior (fig. 11) are not yet available
   Conductivity probes (fig. 13) are not reliable enough.
   Ultra-Sonic liquid level monitoring systems are under development /7/
   Core cooling monitors /8/ are also under development and will be presented in session IV of this meeting.

The second important phenomenon in severe accidents in case of LWR is the core melt progression and the hydrogen generation. Both TMI-2 and LP-FP-2 had shown the importance of these phenomena and the urgent need for reliable measurements. In case of thermocouples precautions have to be considered in connection with the behavior under high temperatures. The cable shunting effect (fig. 15) found in case of LP-FP-2 have to be taken also into account.

No measurements of hydrogen were available in TMI-2 and the monitoring of hydrogen generation was not successful in case of LP-FP-2. Nevertheless there are several successful hydrogen measuring techniques available one example is explained in reference /12/ which can be applied to Nuclear Power Plants.

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/9/ M. L. Carboneau, "Highlights of the OECD LOFT LP-FP-2 Experiment Including Hydrogen Generation, Fission Product Chemistry and Transient Fission Product Realease Fractions"
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/10/ S. M. Modro, "The LP-FP-2 Severe Fuel Damage Scenario ..."
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/12/ G. Schumacher, R. Huber, H. Malauschek and I. Schub "Gas Analysis in CORA Experiments. Description of Analysis System and Results of CORA-29 &-31 Experiments". CORA-Workshop, Karlsruhe, Sept. 23-26 1991.

OECD NUCLEAR ENERGY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATION
SPECIALIST MEETING ON INSTRUMENTATION TO MANAGE SEVERE ACCIDENTS

Köln, 16 - 17th March, 1992

"Instrumentation for Integral Severe Accident Simulating Experiments"

by
R. ZEYEN *
and B. CLEMENT **

PHEBUS FP PROJECT at CEN Cadarache

* Commission of the European Communities
Joint Research Centre, ISPRA
seconded to CEN Cadarache
** Commissariat à l'Energie Atomique
Institut de Protection et de Sureté Nucléaire
Département de Recherches en Sécurité

ABSTRACT

The PHEBUS Fission Product FP Project is concerned with LWR severe fuel damage and fission product source term experimentation. Highly irradiated fuel pins are run in accidental condition until large UO_2 melting occurs and structural materials and fission products aerosols and vapours are swept into a scaled-down primary circuit. A circuit leak is simulated and a reactor containment provided for long term FP behaviour study, mainly emphasising iodine radio-chemistry and behaviour. This effort is done in order to improve the understanding of the phenomena taking place in a "beyond design basis accident" followed by a transfer of this knowledge into the computer codes used for the establishment of accident hazards.
This paper treats instrumentation and methods used to extract experimental data from PHEBUS FP tests.

The fraction and nature of radioactive fission products that reach the containment, and how it behaves in the containment is critical to accident source term estimates.
Particularly iodine radiochemistry is emphasized, oriented and monitored in the containment vessel over about one week: iodine in the gas phase and in the liquid phase of the sump is monitored on-line (gamma-spectrometers and selective samplers) and off line through numerous adsorption and sampling devices.

Calibration and quality control of instruments and analytical methods is a most important step to achieve perfect control of these uncommon tests and to have confidence in data on FP releases in severe accidents.

Finally those instruments which provide the best data, and reply to both criteria of simplicity and reliability, have a good chance for being considered later in accident managing reactor instrumentations.

PHEBUS FP is a CEC and CEA-IPSN co-sponsored project operating in the existing Cadarache based PHEBUS experimental reactor, with the participation of several non-european countries.

1. INTRODUCTION

The PHEBUS Fission Product (FP) Project is concerned with LWR severe fuel damage accidents using highly irradiated fuel pins. UO_2 melting occurs and structural materials, control rod and fission products aerosols and vapours are swept into a scaled-down primary circuit, and finaly into the simulated containment.

The amount of radioactivity released to the environment as a fraction of the initial core inventory is referred to as "Source Term", the major objective to be studied in the PHEBUS FP programme.

The behaviour of fission product iodine deserves special attention due to its potential radiological impact and to the complexity of the iodine chemistry. Many experimental programmes have focused on this behaviour and have been reported in review papers like [Ref.1] and [Ref.2,3].

Iodine escapes from the fuel in accidental conditions, driven by temperature, insignificantly as elemental I_2, but mainly in the form of CsI vapour or aerosols. In low pressure conditions only part of the CsI will deposit in the primary circuit, the remaining part will reach the containment and will deposit onto the sump surface. Readily soluble in water it will then immediately dissociate and hydrolyses into non-volatile I^- depending on concentration, temperature, pH and other factors. Iodates can then be formed in the sump through decomposition of HOI. The important value in reactor safety studies is the partitioning coefficient i.e. aqueous over gaseous iodine concentration, responsible for eventual release from a broken containment vessel. This coefficient is affected by radiolysis in the sump, increasing the I_2 concentration, very much depending on the sump pH, but also by the formation of organic iodine compound by reaction of I_2 with organic contaminants or in contact with paint; volatile methyl iodide CH_3I is the most common organic species. Its concentration and equilibrium depend again on factors like temperature, pH, radiation dose rate and heat transfer coefficients of liquid surfaces. The presence of many other FP species and compounds in an accidental reactor sequences does not make understanding easier.

There are numerous areas where PHEBUS FP can help explain and model fission product behaviour in the release phase, circuit transit or in the complicated long term containment scenarios. Reliable and complete instrumentation is the necessary condition for assuring sufficient data extraction for input into the accident modelling codes.

2. TEST OBJECTIVES

The experiment is designed to study various phenomena occuring during the course of a severe LWR accident, supplying new data for physical-chemical models. Examples of analytical content of the models are:

- release rates of more or less volatile fission products from overheated fuel,
- interaction of FP vapours with structural material aerosols,
- aerosol depletion in specific primary circuit components and in containment by-pass scenarios,
- influence of condensation, pool boiling, containment spray, etc. on the potential source term,
- influence of primary system pressure on the chemical form of fission products,

• iodine interaction with paint, radiolysis and re-volatalisation in the containment.

The last item is of particular importance and will be discussed in more detail in the instrumentation chapter.

Six experiments are foreseen, one per year starting in 1993, reproducing different accident scenarios [Ref.4]. More specific separate effect studies around the given problem area are foreseen in in-pile and out-of-pile experiments in periphery of the PHEBUS FP facility (e.g. iodine retention on paints, paint comparison studies, aerosol thermal resuspension, etc.).

3. FACILITY DESCRIPTION

The PHEBUS FP installation is a 1/5000 replica of a 900 MW PWR reactor, comprising the reactor core using highly burn-up fuel in a 20 rod array, the simulated circuit with a test matrix specific reactor component (steam generator, pressurizer, relief tank, etc.) and the containment with sump (see Fig.1). The existing PHEBUS reactor is used for the fuel irradiation and degradation process, already successfully performed in the LOCA and SFD tests. The main part of the simulated circuit following the fuel is actually been assembled inside a large stainless steel container, built in the new building adjacent to PHEBUS [Ref.5].

3.1. In-pile test section containing the fuel bundle:

Fig.2 gives a cross section of the test section at the level of the fuel. 20 PWR fuel rod, irradiated up to 32 GWd/t in the BR3 reactor, are assembled by 2 AFA zircaloy grids around one central steel clad Ag,In,Cd control rod, insulated by a variable density zircaloy shroud, held inside the pressure tube.

The bottom coolant inlet is closed during the experimental transient by a hydraulic "foot valve", during this time a small controlled steam flow, later replaced by uncondensible gases, carry the fission products and structural material aerosols up through the tubular section with a large temperature gradient into the horizontal part of the primary circuit.

3.2. Primary circuit (Fig.3):

Due to geometric constraints of the reactor, the primary circuit has to start with a 5 m long horizontal line, followed by the measurement zone «C», the steam generator U-tube and the measurement zone G, shortly prior to the containment inlet (Fig.3 shows the circuit inside the safety steel "caisson"). In the first scoping test FPT 0 the circuit temperature is trace heated to 980 K, with a transition zone down to 425 K in the steam generator.

The circuit tubing will be fully replaced between each test, with the experimental component, steam generator, replaced by another component or, by a minimum line into the containment.

LOCA LOOP SCHEMATIC

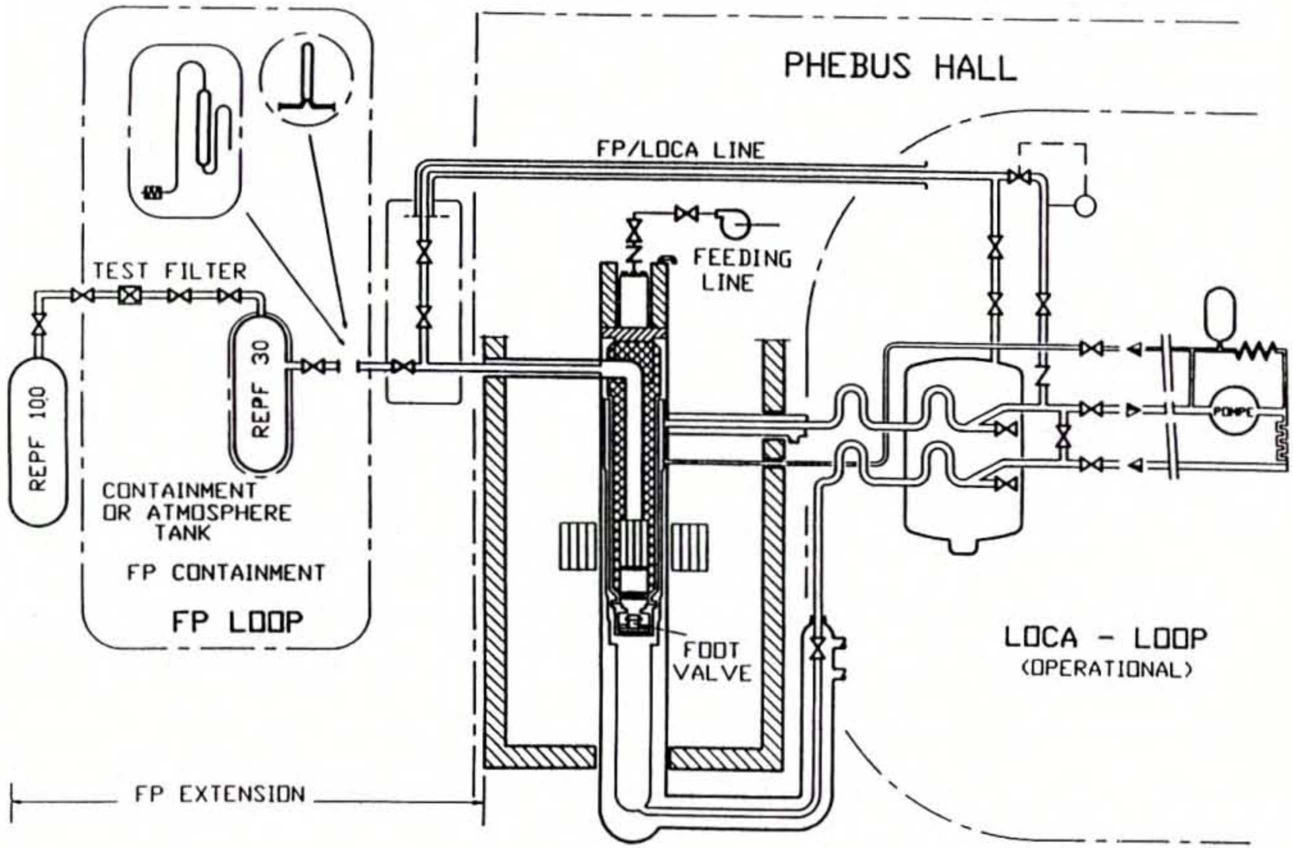


Fig : 1

IN-PILE SECTION OF FPT0:
CROSS SECTION ON DRIVER CORE MID-PLANE

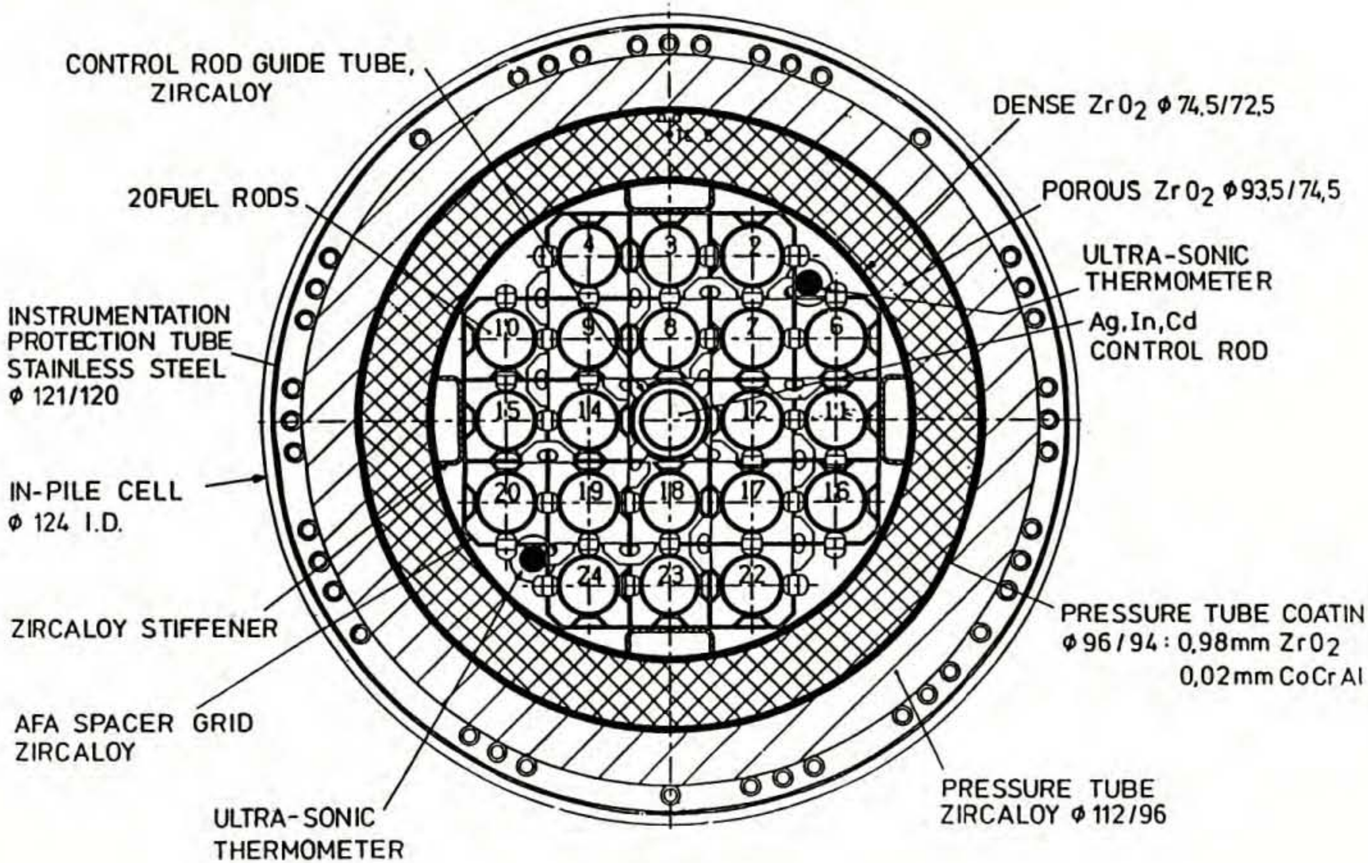
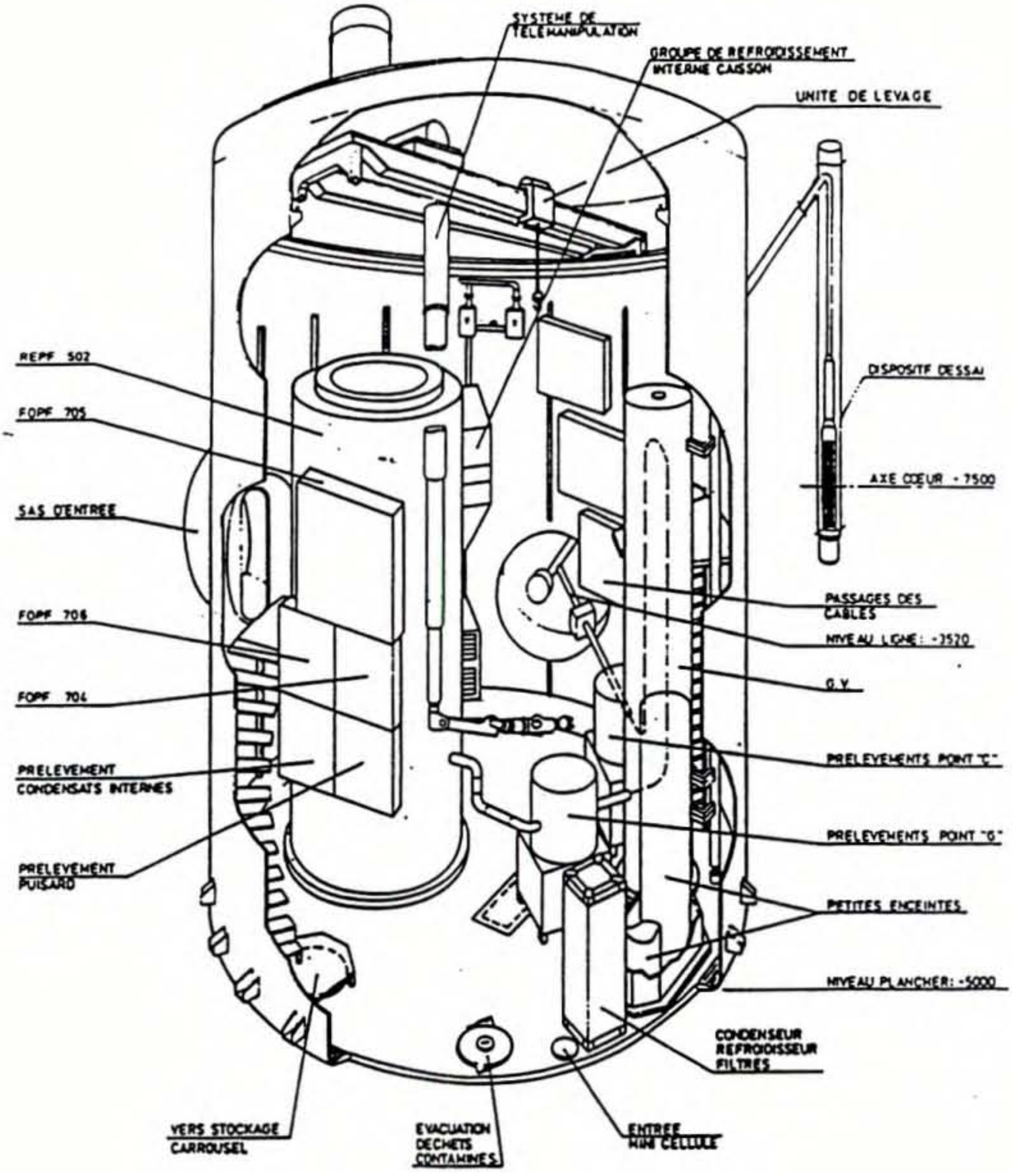


Fig : 2



ISOMETRIC VIEW OF THE FP CAISSON

Fig : 3

3.3. Containment (Fig.4):

The reactor containment is simulated by a 10 m^3 vessel with a 0.1 m^3 sump having reduced free surface, for representativity reasons. Three cylindrical condensers are designed to keep the containment walls dry, condensing out the surplus steam from the bundle, producing interactions of its painted surfaces with the containment gas phase and recovering the condensate for flow monitoring and chemical analysis. Three organic coolant loops control temperatures of the condensers, the containment walls and the sump.

Following effects can be simulated during the experiments (not exhaustive list):

- aerosol behaviour and settling,
- iodine chemistry checking the combined influence of the inventory, sump pH, gas and sump temperatures, radiolysis, mass transfer and surface absorption.
- interaction with paint,
- containment spray,
- depressurisation.

The containment is designed for mild hydrogen deflagration; its inner surfaces have to be decontaminated between each experiment.

3.4. Effluent system:

The containment depressurisation can become an experimental feature, if enough containment sampling instrumentation is available at the end of the aerosol and chemistry phase. The ultimate destination of the gaseous effluents is a 100 m^3 "atmosphere" tank, initially filled with nitrogen and whose gas volume is used for recirculating /diluting the hydrogen containing mixture from the containment. The atmosphere tank will later be equipped with a palladium plate-catalyser designed by GRS Köln and tested at KfK Karlstein, for automatic H_2 depletion before a final release to the stack.

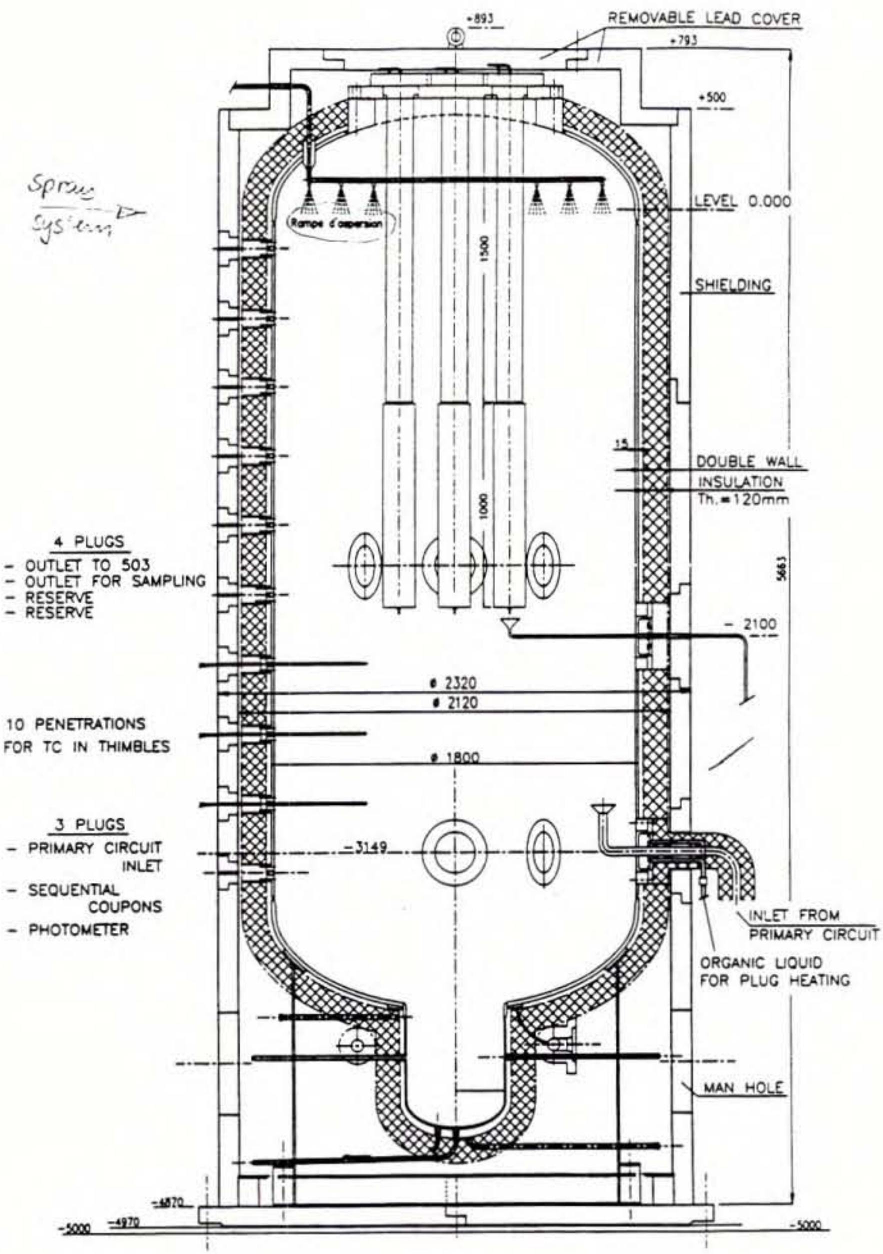
4. INSTRUMENTATION

The "instrumentation" term is generically used here to define devices and methods fullfilling the three main functions of process control and safety, thermal hydraulic measurements and specific fission product/activation product measurements. We will concentrate here on:

- on-line instrumentation for thermal hydraulic and FP data,
- sequential sampling of FP-containing gas, liquid and solids associated with post-test analysis of these sampling products,
- post-test analysis (PTA) of cicuit components and of gaseous and liquid effluents,
- post-irradiation examinations (PIE) of the degraded in-pile test bundle.

The establishment of a consistent instrumentation plan for the PHEBUS FP tests, as described in [Ref.6], has been supported by several shared cost action programmes sponsored by EEC:

with BATTELLE Frankfurt for thermohydraulic instrumentation [Ref.7,8],
with SIEMENS KWU for FP and PTA instrumentation [Ref.9],
with AEA Winfrith to review PTA methods [Ref.10].



CONTAINMENT VESSEL REPF 502

Fig : 4

4.1. ON-LINE INSTRUMENTATION and SEQUENTIAL SAMPLING

4.1.1. Test-Section Instrumentation:

Table 1 shows the instrumentation summary as foreseen for the first and second test sections. Relevant items on this list can be highlighted:

- High temperature thermocouples, W/Re wires, Re sheathed and Ir coated will be mainly used in the centreline of un-irradiated scoping-test fuel. In later experiments the use of fresh fuel rods in the bundle made up by highly irradiated fuel (up to 36 GWd/t) is still in discussion.
- ultrasonic thermometers in two bundle corner positions, both having 7 axial measurement zones over the mid-core area. These rather sophisticated devices, fabricated by the European Community Institute TUI at Karlsruhe, are supposed to measure up to fuel melting when W/Re thermocouples have stopped operating properly. The advantage of measuring axial profiles in one device is notable.
- neutron flux measurements will be performed in the driver core, but also axially along and close to the test bundle in order to survey fuel and AgInCd control rod material relocation.
- in follow-up tests a sequential coupon device will be installed in a gradient zone on top of the fuel for vapour deposit sampling at high temperature.

4.1.2. Primary Circuit Instrumentation:

Fig.5 shows a rough sketch of the circuit instrumentation as located on the primary circuit; again some relevant items can be highlighted:

- flow rates of steam and H_2 coming from the fuel will be gaged by several thin thermocouples and a correlation technique of small temperature variations propagating along the flow lines.
- the oxygen potential is measured at point «C», to get an understanding of the fuel chemistry.
- several γ spectrometers using high purity Ge detectors are located at relevant positions along the circuit line, most important of which is the first one at point «C», the closest to the accidented fuel bundle, thus the first insight into the γ emitting source term. At this point two spectrometers are dedicated to the on-line descrimination of moving versus deposited FPs in the circuit line. This is a difficult task considering the "moving" tubing (thermal expansion) linked to heavy shielding and collimator equipment. More γ spectrometers will survey gas and deposits in the steam generator's cold leg and prior to the containment inlet (source term).
- INEL Idaho Falls provide us with a redesigned version of their PBF aerosol light extinction photometer, a valuable device for both semi-quantitative information on special release events and to provide experimental guidance for sample taking.
- sequential gas capsules and filters together with impactors should give information of the gas species and aerosol nature, density and granulometry at various times. Samplers prior to the steam generator have to operate at 700°C, those after the SG

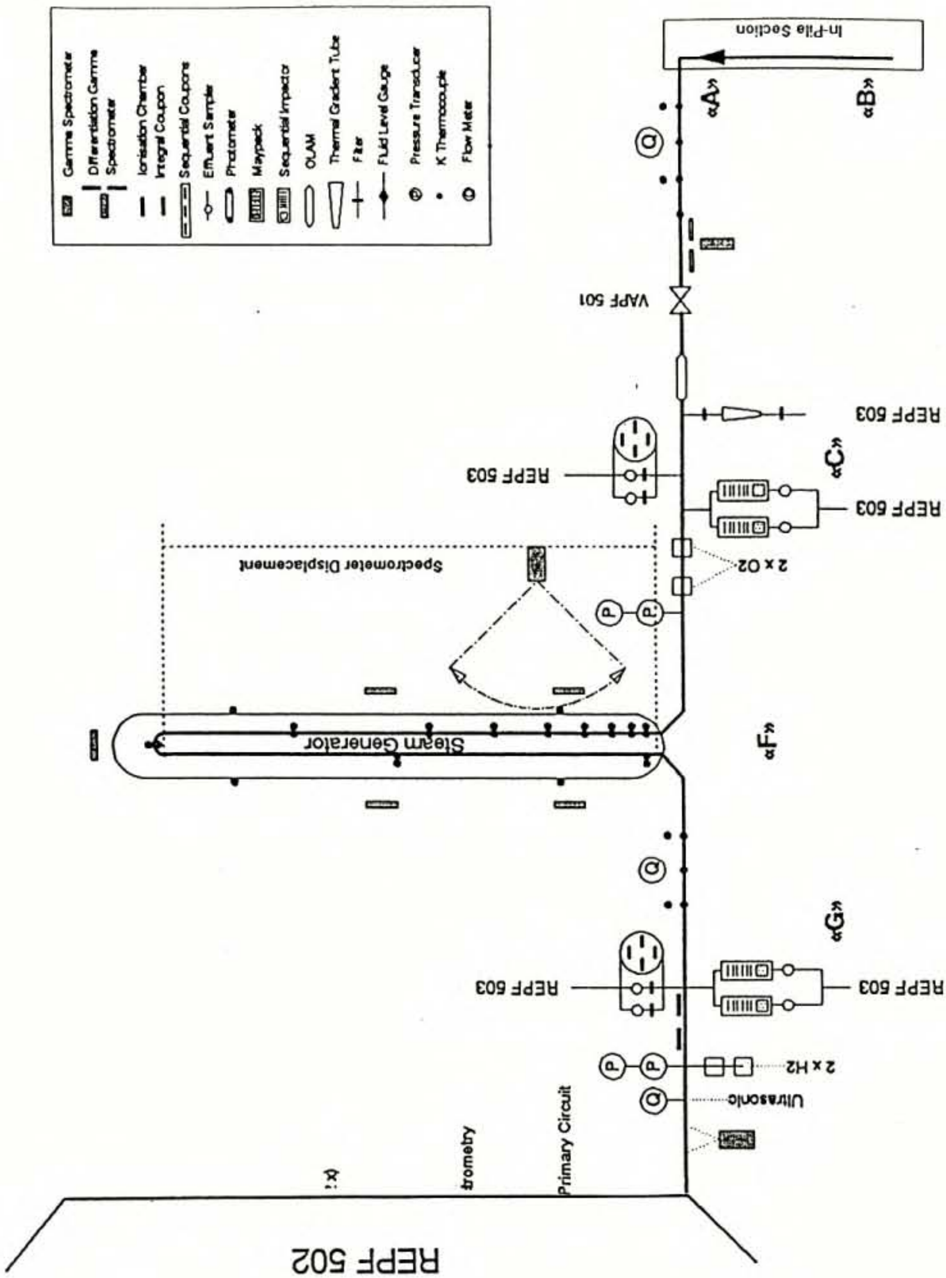
Table 1

Instrumentation Summary: TEST SECTION - POINT A+B

Parameter	Instrument	Location; Comments	FPT0	FPTx c)
Temperature	ultrasonic thermometers	two outer corners of the bundle, 7 axial positions	2 (14 measure-ments)	2
	K type TCs	in the lower part	2	2
	W/Re TCs	in the rod centreline	12	- 1)
	K type TCs	on AIC rod	2	2
	W/Re TCs	on guide tube	4	4
	W/Re TCs	on stiffener plates	8	8
	K type TCs	in the ZrO_2 insulator	12	12
	W/Re TCs	temp. gradient zone	1	1
	K type TCs	"	4	4
fluid	K type TCs	bundle inlet	2	2
"	W/Re TCs	temp. gradient zone	4	4
"	K type TCs	"	4	4
"	K type TCs	upper part	4 2)	4
	K type TCs	safety & heaters	10 & 11	≤ 21
Pressure	pressure transducers	in the gas feeding lines	2	2
	differential pressure transducer	bundle inlet/outlet	1	1
Flow rate 2)	hot wire flowmeters	injected quantity (H_2,He)	1	1
	pump rate, weighing	injected quantity (H_2O)	1	1
On-line γ activity	γ spectrometer (a)	at level -6500, at bundle outlet	-	1
Deposit	sequential coupons (a)	temp. gradient zone	6	6
Post test activity	γ spectrometer	post test examination: PEC	+	+
Fuel relocation	fission chambers (+ post test analysis)	along the bundle length in coolant water	4	4

a) Research in progress; b) Development in progress; c) Not decided.

1) Instrumented fuel pins under discussion.
2) two thermocouples could be used for flow rate measurement by TC correlation technique, in the test section upper part, if requested.



INSTRUMENTATION PLAN FOR THE FPT0 TEST
PRIMARY CIRCUIT

Fig : 5

at 150°C. The timing of these samplers can be adjusted during the experiment evolution according to the release phenomena. The sampler number might have to be increased in the future.

- thermal gradient tube (TGT) for the study of condensation patterns, between 700°C and 150°C, separating FP species like CsI and CsOH. Only one TGT will be used at point «C» because recovery for PTA and interpratation might be hazardous. Its temperature control with transient gas through-flow is a difficult task.

4.1.3. Containment Instrumentation:

The containment instrumentation can be found on Fig.6. Important on-line sensors are:

- wall and gas temperatures, pressure and sump level.
- moisture, H_2, and O_2 sensors in the gas phase, selective electrodes for I^- ions and pH in the sump. These sensors are all doubled for redundancy.
- the condensate flow is measured and lead out of the contaiment for analysis.
- on-line separation of I^- versus IO_3^- ions, their ratio co-responsible for volatile iodine formation, is envisaged for later tests.
- γ spectrometers continously detect the activity of certain isotopes in the containment gas phase, on the condensers' painted surfaces and in the sump.

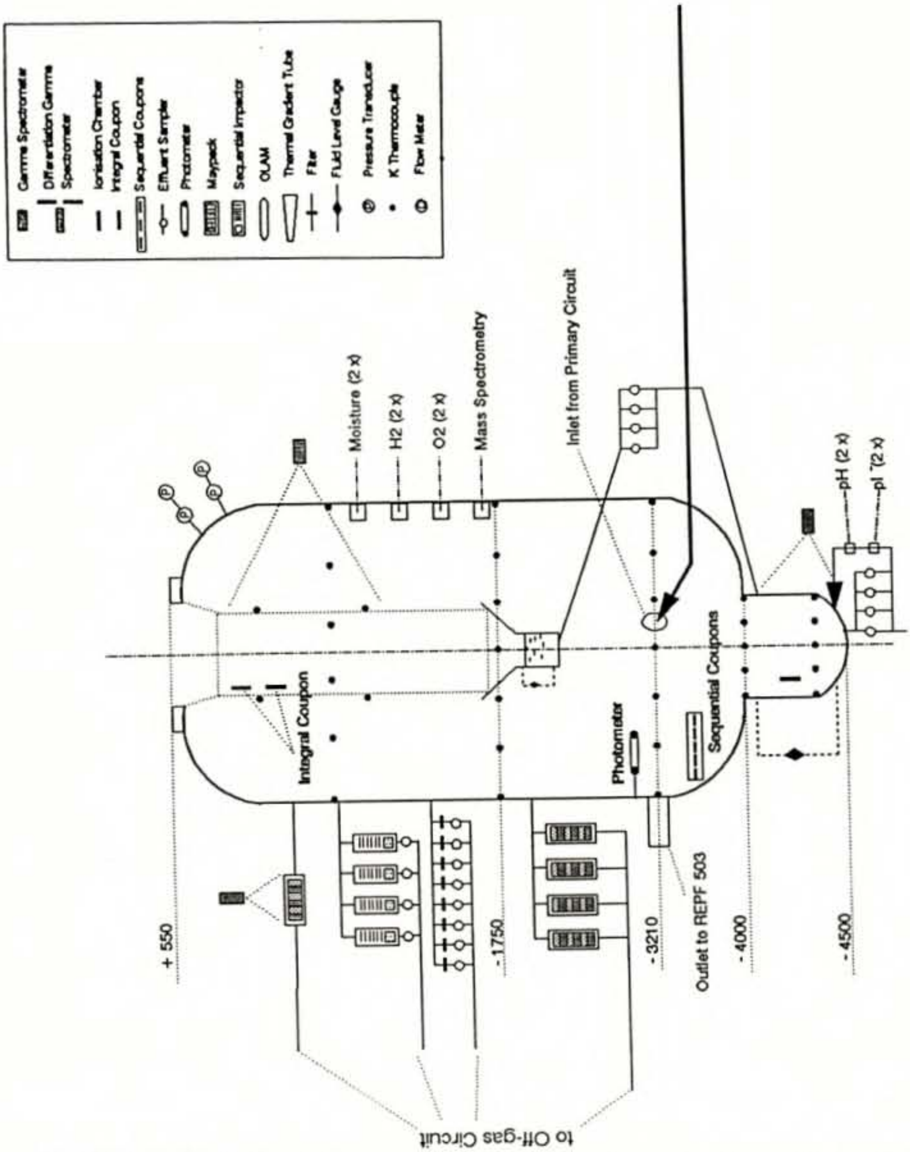
Sequential devices:

- a sequential deposition coupon device has been designed to recover 8 metal coupons at 8 different time windows. Its remote operated extraction after the test is not an easy task.
- sequential aerosol impactors of a modified ANDERSEN MKII type are foreseen for the first scoping test; real on-line classifying will be performed on a diffusion type ball bed stack in follow-up tests.
- a long series of filters and capsules will yield more data on aerosols (to be compared to impactors) and on gas composition for long life isotopes.
- gaseous iodine species monitoring is a specially treated item in PHEBUS FP: sequential selective filters are installed for quantitative measurement, an on-line selective filter equipped with a γ spectrometer diode will follow the species evolution over extensive periods of the long term chemistry phase of the experiment. More details of the filters and its qualification tests are given in the following separate paragraph 4.1.4.

4.1.4 Iodine Speciation Samplers' R & D:

Since the PHEBUS FP Project focuses on long term radio-iodine chemistry in the containment, an effort was made to determine those iodine species likely to come into contact with the population in the event of a containment leak. This is why the measurement of the total iodine concentration and its gas phase speciation was given particular attention.

Several experiments have made use of iodine selective filters, however not in the particular conditions of a PWR containment after a severe accident. Taking into account



INSTRUMENTATION PLAN FOR THE FPT0 TEST CONTAINMENT VESSEL

Fig : 6

this requirement and trying to avoid condensation, the filter operating conditions have been defined as the following: 160°C, 0.1 to 0.5 MPa, H_2 concentration 9%, high relative humidity.

The filter pack (Fig.7) is made up of an aerosol quartz filter paper, silver knit-mesh for molecular iodine and heavy metal doped silver zeolite for organic iodine. Each filter stage can be doubled or tripled according to the needs. These filter media have been selected through an experimental programm carried out at KfK-LAFII at Karlsruhe, using two existing facilities. Several absorber materials have been tested using both molecular and organic iodine (CH_3I): those showing the best retention properties have been selected and then tested in complete filter assemblies using iodine gas mixtures, and varying both its pressure and hydrogen content. The detailed results of this work will be published later.

The same absorber materials have also been used in newly designed "on-line iodine selective sampler" where each filter stage is scanned alternately by a γ spectrometer. The integrated aerosol concentration and the molecular and organic iodine contents can thus been followed over typically 10 to 20 hours of operation. More than one of these devices should be ideally installed and presented before the same γ detector. The problem of back ground noise from the high aerosol activity, compared to the iodine stages is not fully solved.

5. POST-TEST ANALYSIS (PTA)

PTA is a most important item in PHEBUS FP instrumentation, not least because of the shortcomings of on-line instrumentation.

The first major operation is the samplers' recovery and transfer (around 25) to the "CECILE" hot cell, below the main "caisson". After a first gross γscan the samplers are dismanteled and the sensitive components fine γ scanned again, before packaging and transport to different specialised laboratories. Iodine samplers have a general priority due to the ^{131}I isotope's short half life.

The following PTA analysis programme is a complex series of flow sheets from recovery to a series of analysis techniques ranging from scanning electron microscopy coupled to energy dispersive spectroscopy (SEM/EDS) and particle recognition equipment (PRC) up to many types of chemical techniques for element and species recognition and concentration determination. Neutron activation analysis (NAA) is an widely used accurate method for isotopic/elemental analysis of non- γ emitters, fissile particles and decayed iodine. A typical flow sheet for solid FP samples is shown in Fig.8. More details on the PTA working plan can be found in [Ref.11].

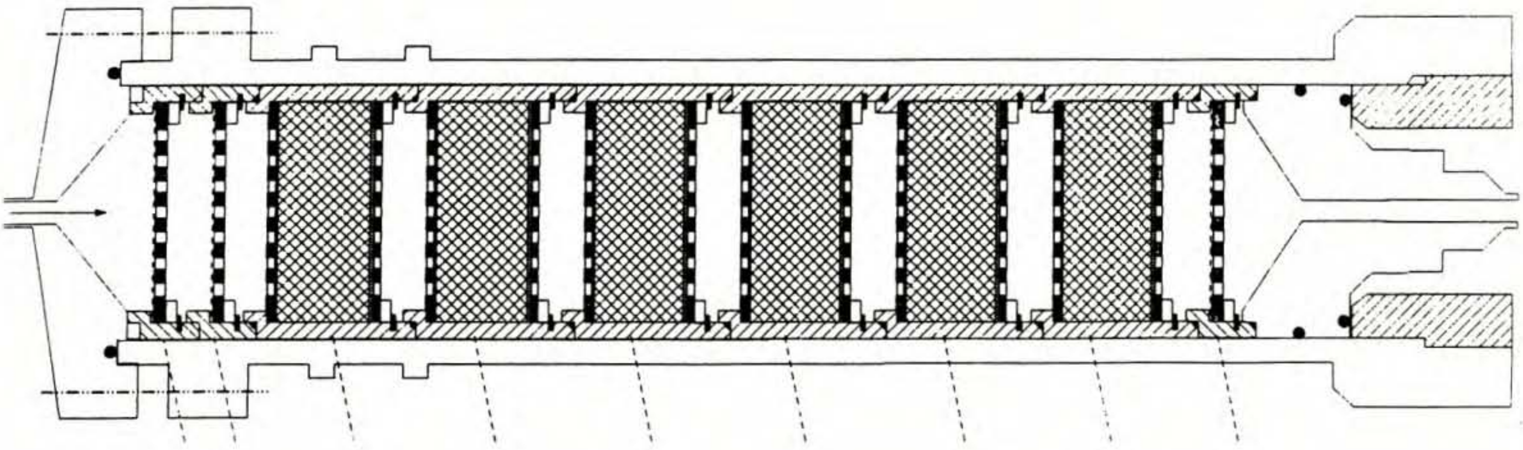
6. POST-IRRADIATION EXAMINATIONS (PIE)

Shortly after the irradiation experiment the test section will be un-coupled from the circuit and stored in a vertical examination and control station (PEC). This PEC houses a radial and axial γ scan and radiography facility with possibilities of computer tomography.

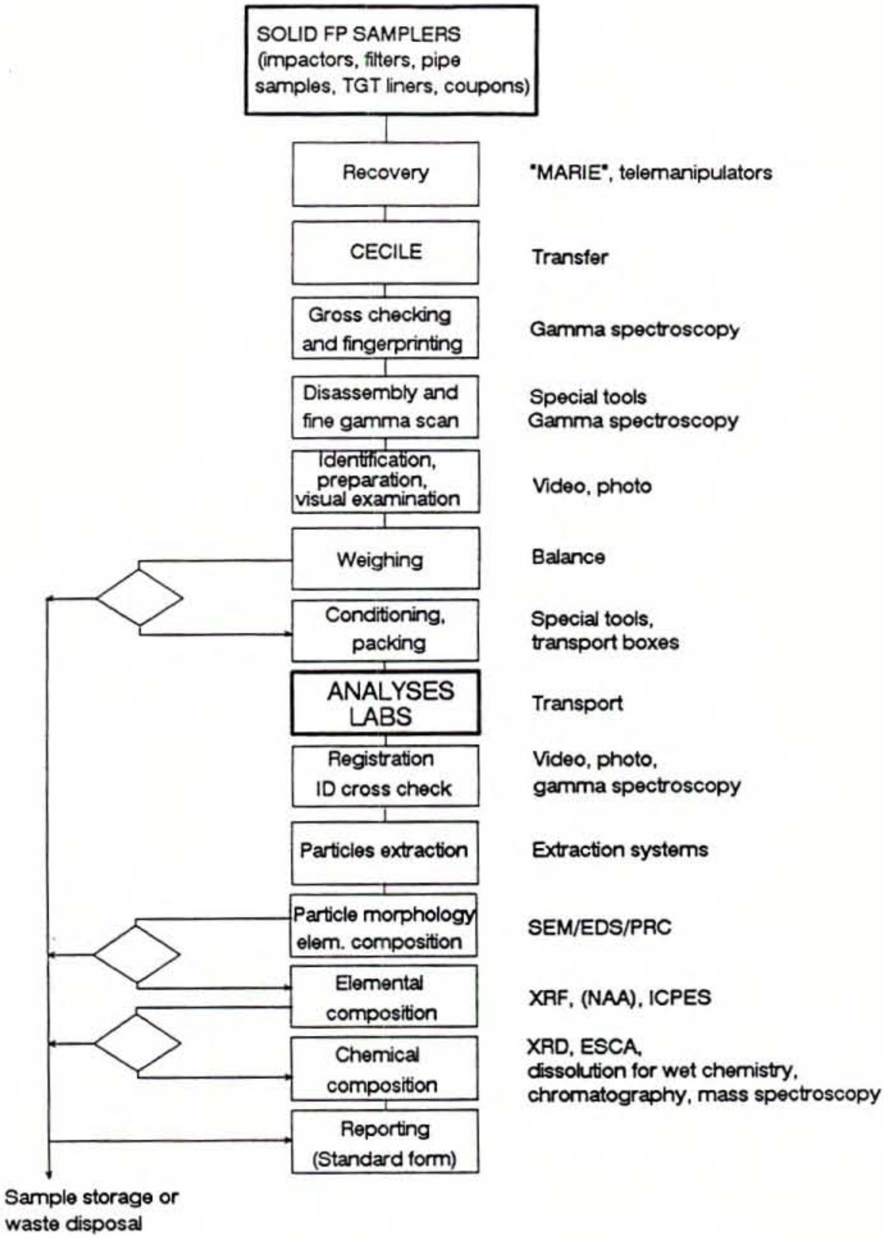
In a second phase, several months later, the test section will be transported in a specific shielded flask to a hot lab where horizontal and vertical cuts will be performed. These small sections will then be dispatched to outside laboratories for further detailed elemental analysis and FP inventory measurements.

The upper tubular plenum of the test section has a special interest: the surfaces of this

GASEOUS SELECTIVE IODINE FILTER



Fig



FLOW SHEET FOR SOLID SAMPLERS
(Filters, Coupons, Pipe Samples, TGT Liners)

Fig : 8

part have been located in a severe thermal gradient during the release phase and interesting deposits will have accumulated thereupon. This is why these tubular sections will be cut into small sections and treated, according to a PTA flow sheet, for detailed deposit analysis.

7. CALIBRATION and QUALITY ASSURANCE

A certain number of "classical" instruments, like low temperature thermocouples, pressure transducers, flowmeters are procured using quality procedures typical for reactor standards, their calibration is of industrial type or performed in existing laboratory equipment.

Another great number of instruments however cannot be handled in this way, or because they are "tailor made" for this Project and calibration devices have to be specially built for this purpose, or because qualification/calibration ask for a non-neglegible amount of know-how and equipment, not available at the Project, i.e. help is needed from european or overseas laboratories. Some examples of special procedures are given here:

* the γ spectrometers (8 to 10) require sophisticated mobile colimators (e.g. for disrimination of flying and deposited γ emitting isotopes) accounting for low and high count rates and scanning facilities which have all to be tested and calibrated on a purpose oriented calibration bench at Cadarache using different types of emitters. A test run during a similar experiment at Chalk River, Canada, is planned.

* the selective gaseous iodine filters had to undergo material choice- and operation testing in two KfK Karlsruhe test loops made available to the Project (see §4.1.4.). On-line iodine filter prototypes will have to be built and calibrated due to a severe collimation problems and due to the fact that the containment iodine concentration is slowly decreasing, the integrated signal does not change much at the end of the experiment.

PHEBUS FP results on iodine filters might be of particular interest for power reactor venting systems.

* impactor data is usefull only if the cutoff diameters have been previously verified in tedious laboratory calibration work (CEA Fontenay). The measurement range has to be particularly large since the pre-calculations on possible aerodynamic diameters are spreading widely.

* the light extinction photometer is a usefull relative particle number concentration device, which through precise calibration and mathematical treatments can also yield precious data on particle size or shape.

* off-line instrumentation described in the PTA paragraph needs obviously much qualification and calibration work on sophisticated analytical equipment applied to similar products from bench-scale experiments; much of this work has been done in associated laboratories in Europe and elsewhere.

Sometimes instruments have to operate in badly known conditions which cannot be reproduced in laboratory for qualification or calibration: an example could be the thermal gradient tube provoking vapour condensation at point «C». Much uncertainty remains as to the results obtained from this device.

8. CONCLUSIONS

PHEBUS FP is presently the largest experimental programme in the area of severe accident studies. It benefits of the interest of the international safety community and will contribute significantly to the quantification of source term and environmental impact of nuclear power plants.

Many instruments are used in this kind of Projects, some simple some sophisticated and expensive, but the PHEBUS FP tests will represent a well suited qualification facility for devices to be integrated in some safety features of a real scale commercial reactor.

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10. ABBREVIATION GLOSSARY

AEA	Atomic Energy Authority
CEA	Commissariat à l'Energie Atomique
CEC	Commission of the European Cummunities
LAF	Laboratorium für Aerosolphysik und Filtertechnik
INEL	Idaho National Engineering Laboratory
IPSN	Institut de Protection et de Sureté Nucleaire
FPT x	Fission Product Test N° x
SG	Steam Generator
PIE	Post-Irradiation Examination
PTA	Post-Test Analysis
SFD	Severe Fuel Damage
TUI	Transuranium Institute (CEC)

Instrumentation Used for Containment Experiments Under Severe Accident Conditions

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Introduction

To support the management of severe reactor accidents, the instrumentation of an LWR containment must satify more stringent requirements than for design-basis conditions. The additional requirements may involve either more extreme conditions or a greater number of quantities to be measured.

Some of these additional requirements are similar to those being fulfilled by the instrumentation used in large-scale reactor safety experiments performed, for example, in the HDR test facility or in the Battelle model containment (BMC; Fig.1). Therefore, the present paper will describe a selection of the instruments used in the HDR and BMC tests, and discuss their possible transfer to nuclear reactors.

However, when transferring experimental instrumentation to reactors the fundamental differences between the two applications must be taken into account: Reactor instrumentation for accident management purposes has to cope with high-dose radiation. Furthermore, it must remain undamaged in the accident, and must afterwards operate in inaccessible areas with 100-% availability. Experimental instrumentation, on the other hand, is preferably designed for complex and highly accurate measurements of high-transient effects.

Thermal-hydraulic Instrumentation

Thermal-hydraulic phenomena in a containment during a severe accident involve steam and gas releases into the containment volume, temperature and pressure transients, forced and natural convection flows, steam condensation, and heat transfer to structures. In most cases, experimental instrumentation for thermal-hydraulic phenomena is based on commercially available components that are individually adapted to the special requirements of the measuring problem.

In the experiments, *temperature* is conventionally measured by platinum resistance thermometers (high accuracy, low time resolution) or by chromel-alumel thermocouples (medium accuracy, high time resolution). Commercial Inconel sheathed thermocouples of 0.25 mm outer diameter yield a remarkably high time resolution (e.g., 5 ms time constant in hot water).
As low-transient temperature measurements are sufficient for accident management purposes, the existing containment temperature instrumentation can be used with minor changes in the measuring range.

Pressures and pressure differentials in containment experiments must be measured with high time resolution. To ensure good dynamic response, the respective transducers (piezoelectric or on strain gage basis) have to be installed in situ, i.e. without pressure transmitting tubes. As a consequence, the transducers must be equipped with a built-in temperature compensation and, in addition, must be protected against high-transient temperature shocks by thermal insulation.
For severe-accident management purposes it may be favourable to use only pressure transducers that are externally installed (with access for re-calibration and maintenance) and connected by pressure transmitting lines to the interior of the containment.

Local steam content and relative humidity of the containment atmosphere in severe-accident experiments are measured in situ, using a modified commercial sensor (Vaisala HMP 135Y) based on a capacitive method. The sensor modifications involve a special filter cap to guard the sensor against aerosol particles and

droplets, and an internal heater to remove condensate films from the sensor element after periods of supersaturation.
As it is now, the measuring system can be operated only after intense and continous maintenance between different experiments. Its reliabilty must be profoundly improved before it can be recommended for installation in nuclear rectors.

For slow *flow velocities* (between 0.2 and 20 m/s), turbine flowmeters are successfully used in containment experiments, even if high aerosol concentration should occur temporarily. So far, no experience exists on the survival of these turbine flowmeters in hydrogen burns and during large-break loss-of-coolant accidents (LOCA). High-velocity flows ranging between 20 m/s and the speed of sound are measured by Pitot-static tubes connected with in-situ pressure transducers. This kind of device was also successfully used for mesurements in hydrogen deflagration and large-break LOCA tests.
If flow velocity measurements should be needed for accident management in a reactor containment, both systems can be used after additional qualification.

Special Instrumentation for Severe Accidents

Severe accidents are accompanied by the generation of hydrogen and aerosol and their release into the containment volume. The main threat which hydrogen involves is a possible explosion causing additional pressure loads; the safety problems resulting from the aerosol are its strong radioactivity and its possible release in the long term, due to containment leakage.

For *hydrogen concentration* measurements, two systems were available in the HDR and BMC experiments. The transmitters of the first system, based on the thermal conductivity method, are installed outside the containment shell and connected by sampling lines to the various locations inside. The measuring devices of the second system use a catalytic method and are designed for in-situ installation. In HDR and BMC experiments, the first system proved to be the more reliable for accurate determination of the actual H_2 concentration inside the containment. As the

transmitters are easily accessible, additional checks during operation and, if necessary, re-calibration are possible by switching over to a hydrogen-free gas or a calibration gas with a defined H_2 concentration. In future BMC experiments the sampling lines will also be used by a second gas analyser (Fig. 2).
A similar sampling line system installed in a reactor containment and having additional safety features, e.g. against unintentional leakage, would form a very versatile setup for various measurements in case of unforeseen events.

The easiest and least error-prone approach at measuring the local concentration, material composition and particle size distribution of suspended *core-melt aerosols* is to take filter samples for later analysis by laboratory techniques. As the aerosol may be retained in the sampling lines, it is indispensable to locate the sampling filters in situ and to activate them by a remote-controlled valve. To provide a number of filter samples from the same location for different time intervals, so-called filter stations (Fig. 3) were used in the model containment aerosol experiments of the VANAM series. Each filter station contains a total of 12 filters on a turntable to be subsequently brought into measuring position, loaded by opening an inlet valve and sucking a defined gas volume through the filter. The steam portion of the gas volume is separated in a desiccator element (molecular sieve) outside the containment wall and measured by weighing, the non-condensable portion being determined by a flow controller (Fig. 3).
As the filters have to be removed by hand for further evaluation when the experiment is over, this system cannot be used under accident situations in nuclear reactors.

Better suited for use in reactors might be another filter sampling system that had been proposed for and partly used in the former DEMONA aerosol experiments in the model containment (Fig. 4): A sampling filter is fixed to the tip of a *long stalk*. This stalk is inserted into the pressurized containment through a small lock and removed when the filter has been loaded (Fig. 4). The same device can also be used for other sampling procedures or for inserting special instruments (e.g. a boroscope, a TV camera, or a radiation detector).

If such a system can be provided with the necessary safety features, e.g. against uncontrolled escape of radioactivity, it might be sucessfully used for various ad hoc measurements and diagnostics in reactor containments under accident conditions.

Another experimental aerosol instrument of the BMC facility is the *extinction photometer* (Fig. 5). Its optical part is located inside the containment and connected by glass fibres to the electronics outside. A remote-controlled pneumatic activator allows to shorten or lengthen the optical measuring length of the photometer and thus to eliminate any possible zero drift, e.g. due to aerosol contamination of the optics.
This system cannot be directly tranferred to a reactor plant either. However, some of its features may be useful when designing optical measuring or monitoring systems for reactor containments.

Conclusions

Considering the experience gained with conventional and special instrumentation in various large-scale severe-accident experiments in the non-nuclear HDR and BMC test facilities, some suggestions can be made for a possible application of similar instrumentation in nuclear reactor containments under severe accident conditions. Of course, the described test instrumentation cannot be directly transferred, but improvements, adaptions and qualification measures are needed first.

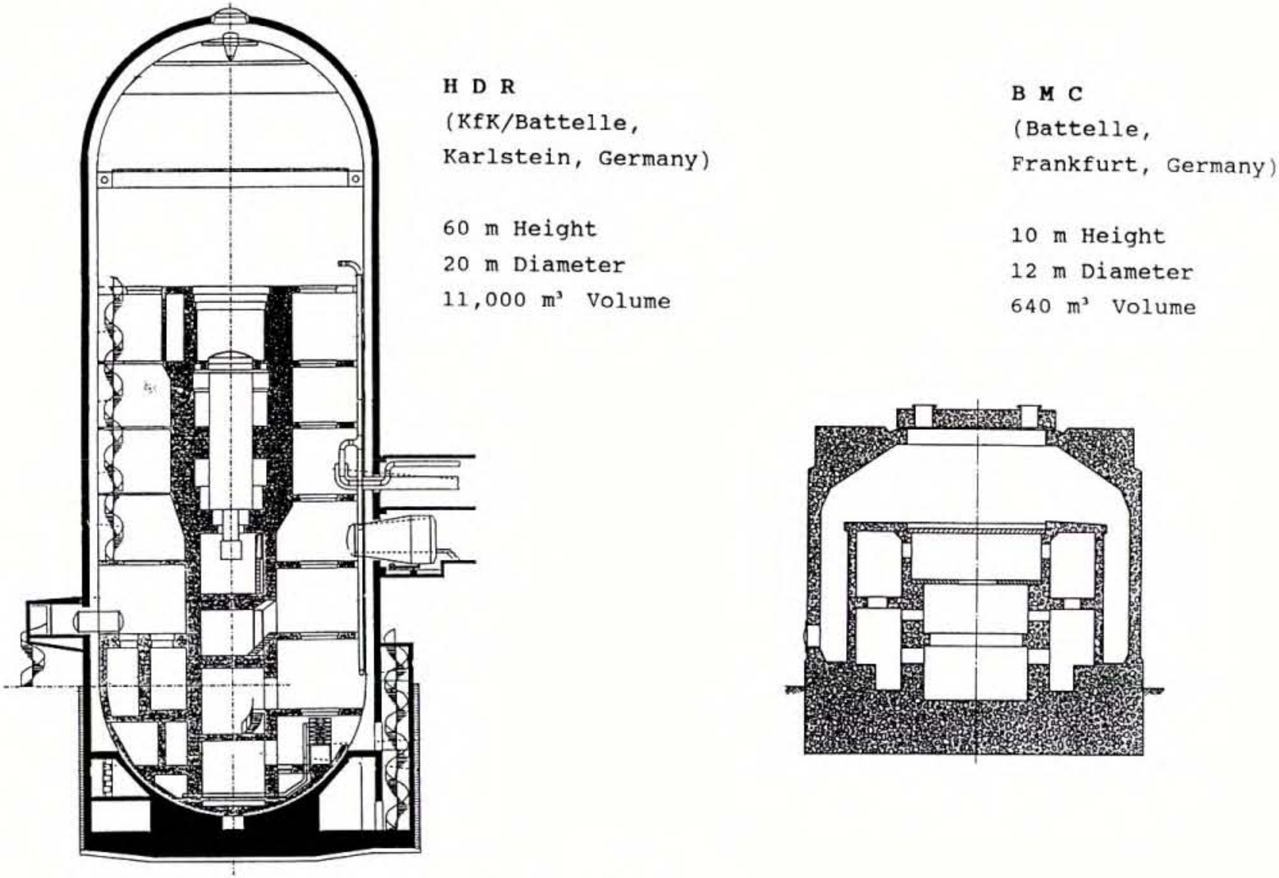


Fig. 1: Containment Test Facilities

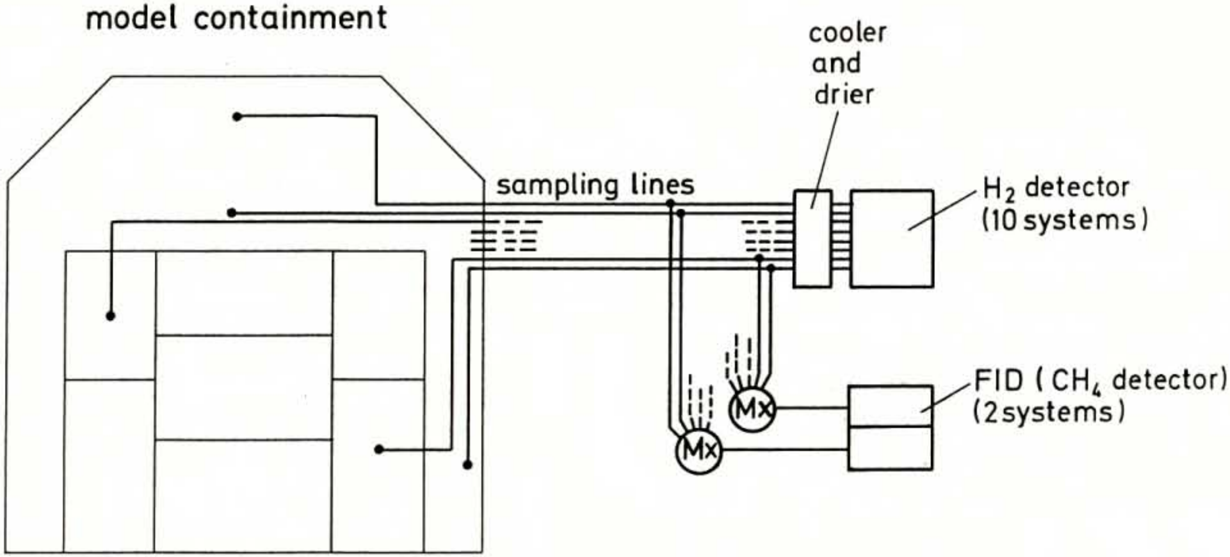


Fig. 2: Sampling Line System to Measure Local Gas Concentration

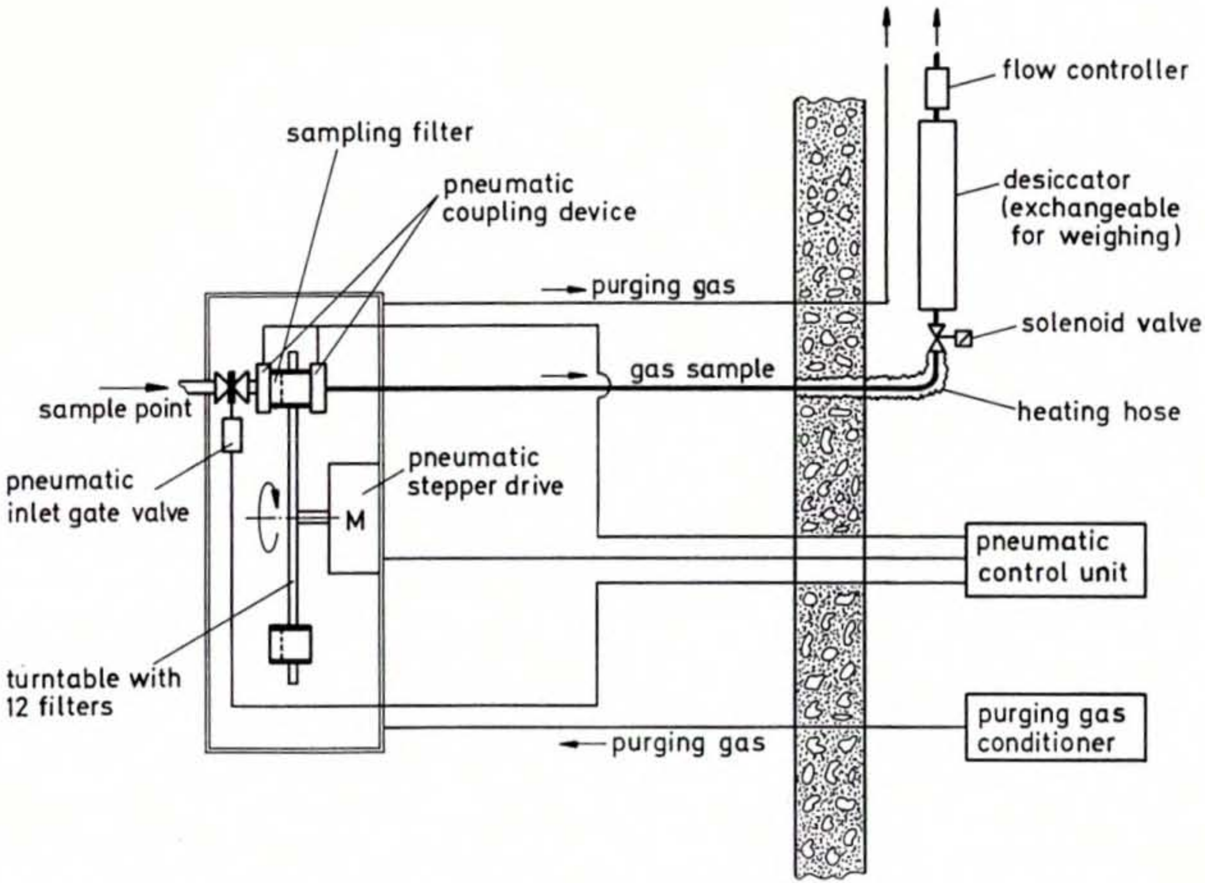
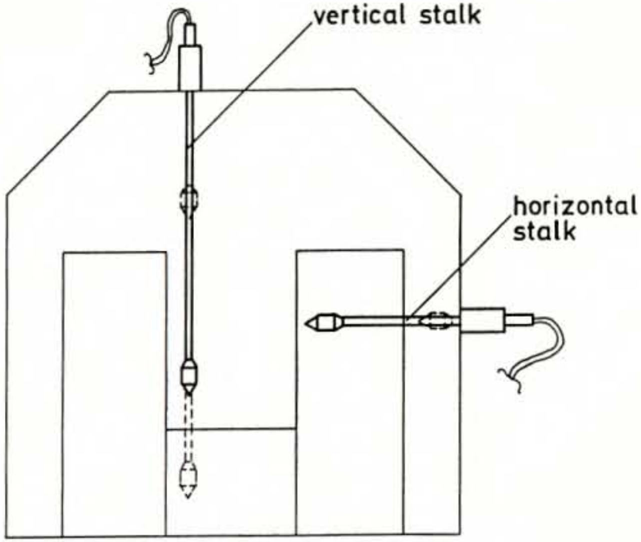


Fig. 3: Filter Station (for 12 filter samples)
to Measure Aerosol Concentration, Particle Size and Composition



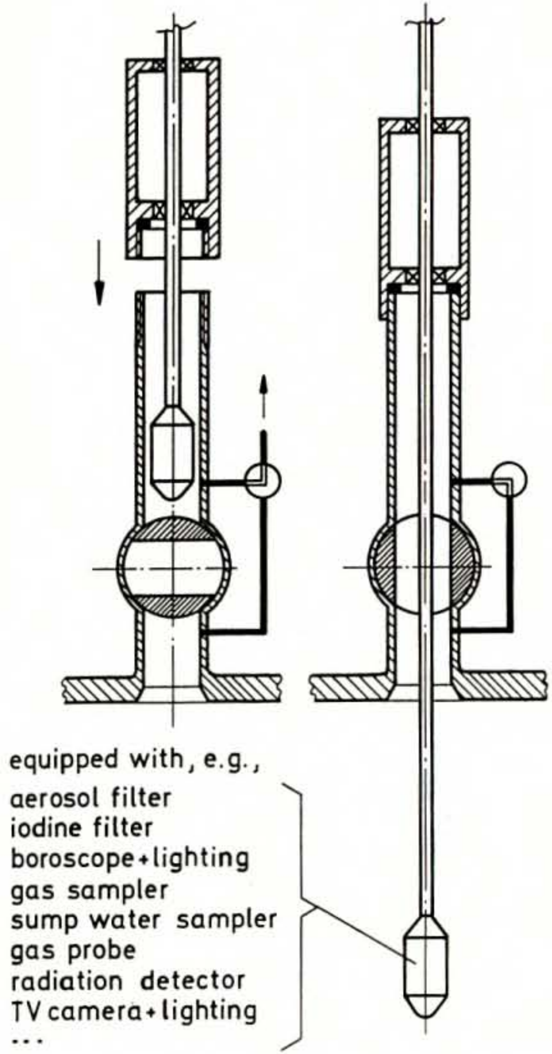


Fig. 4: Multi-purpose Stalk Probe

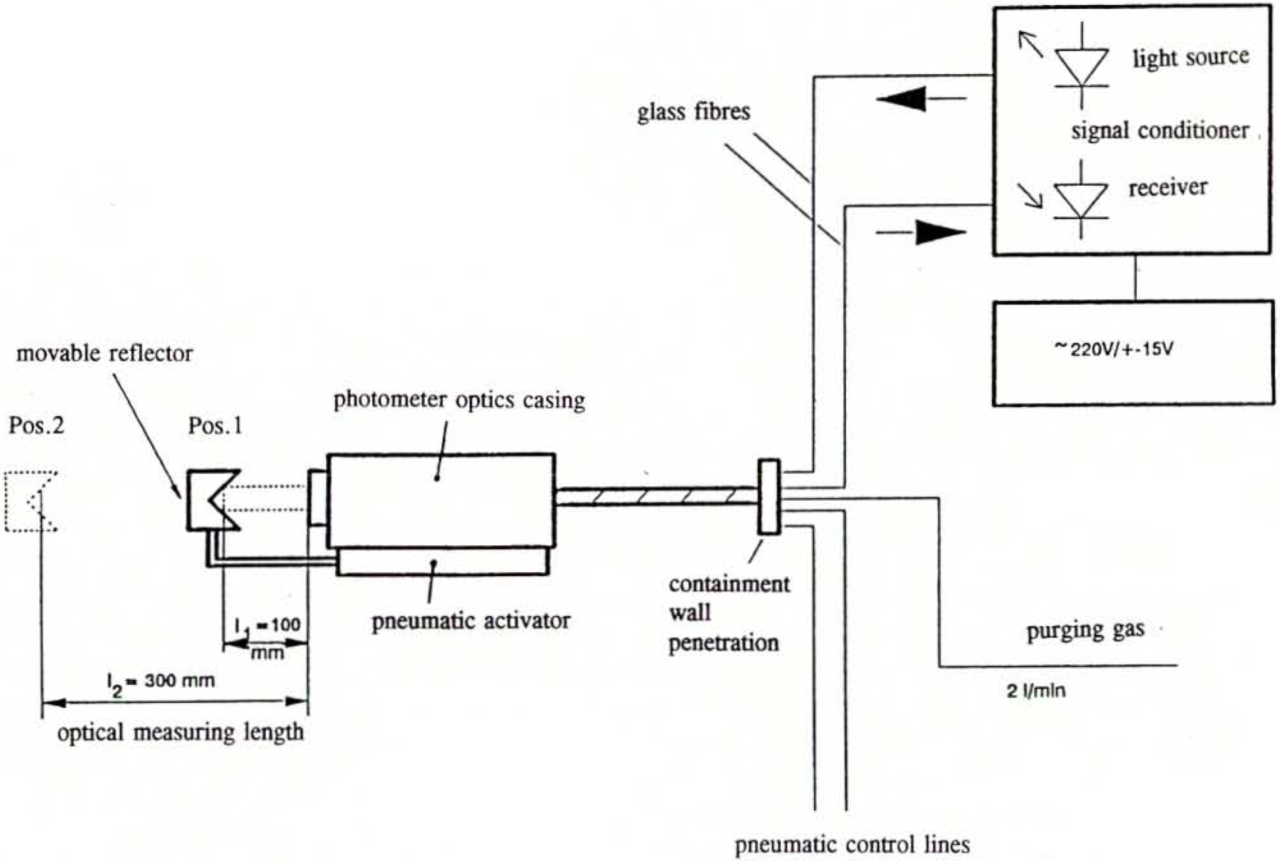


Fig. 5 Extinction Photometer

SUMMARY of SESSION II

The importance of a systematic approach to understand the capabilities and limitations of instruments. A step-wise procedure was demonstrated by **Mr. William Arcieri** (INEL, USA). The different steps include:

STEP 1: Identify Severe Accident Sequences
STEP 2: Determine Expected Conditions
STEP 3: Assess Instrument Availability.

From this information it can be found if needed instruments have the capabilities to give adequate information to find the best accident management strategy. Usually, the environmental challenge is temperature or pressure rather than radiation dose. It was pointed out in the discussion that many instruments may have an extended range beyond the specifications. Adequate instrumentation (extended range, environmental qualification) can also be found in non-nuclear applications.

A specific example of implemented procedures for Core Damage Assessment (CDA) was given from **Mr. Staffan Hennigor** (Vattenfall, Sweden). Methods to judge the core damage include:

- dose rate readings in the containment,
- post accident sampling system,
- process parameters,
- hydrogen concentration.

The first two methods are quantitative, in particular the post-accident sampling system. The process parameters can be used as a source of information if the core has been uncovered. The hydrogen measurements can only be used as a supportive argument confirming that zircalloy oxidation has occurred. It was pointed out in the discussion that the uncertainty on the amount of hydrogen generated was important, but that an equally important uncertainty was the possibility of hydrogen stratification in the containment. Thus, the importance of having a sufficient number of hydrogen sensors was stressed.

The instrument readings from the TMI accident and LOFT-FP-2 were demonstrated by **Dr. Adly Wahba** (GRS, Germany). A thorough examination of the instrument information from the TMI accident shows that water level in the primary system and core relocation can be deducted. The difficulty is, of course, to interpret instrument readings (e.g. SRM detectors) outside normal operation, in particular in a severe accident situation. In first place there is a lesson learned that some instruments may have a different interpretation, e.g. the SRM detectors indicating a water level decrease instead of increasing power. The second step to use this information needs further work. An interesting point was made from the LOFT-FP-2: a sharp temperature increase in a thermocouple was caused by cable shunting, i.e. the measurement system gives additional information if understood.

Three papers dealt with instruments used in experiments, LOFT-FP-2, PHEBUS, HDR and the Battelle Model Containment. It should be emphasized that the purpose of instrumentation in experiments and nuclear power plants is different. Experiments usually have elaborate equipment to get as much information as possible from various phenomena and for code validation. The purpose of instruments for accident management is to understand reactor accident status and to select the best accident management strategy.

Mr. Roland Zeyen (CEC/JRC Ispra) presented the PHEBUS project, in particular the instrumentation and methods used to extract experimental data. Several sophisticated instruments measuring steam flow rate, hydrogen rate, oxygen potential, as well as an aerosol light extinction photometer, etc. to get a full understanding of the experiments being performed. Some of the containment instruments could be of interest for current reactors such as Maypacks (iodine measurements) and sequential sedimentation coupons.

Another presentation from an experimental facility, HDR and Battelle Model Containment was given by **Dr. Teja Kanzleiter** (Battelle, Germany). His experience of temperature, pressure, steam content, hydrogen and aerosol measurements from these experimental facilities and the possibility to use these instruments in commercial facilities was discussed.

In particular a sampling system using a long stalk could be used if it is possible to provide it with the necessary safety features.

It was also pointed out that not only instrument qualification could be achieved but also better understanding of the system response to different phenomena.

The discussion at the end of the session brought up interesting information needs, for example, how can the plant staff understand where the core is located: in place, in the lower plenum, in the cavity or somewhere else in the containment. This also implies the difficulty to judge it vessel melt-through has occurred.

SESSION III

Unconventional Use and Further Development of Instrumentation

Chairman: J. DUCO

UNCONVENTIONAL SOURCES OF PLANT INFORMATION FOR ACCIDENT MANAGEMENT

Organization for Economic Co-Operation and Development
Nuclear Energy Agency
CSNI Specialist Meeting of
Instrumentation to Manage Severe Accidents

March 16-17, 1992

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Albert Machiels
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TABLE OF CONTENTS

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ABSTRACT

An essential element to accident management is having as clear a picture as is practical of the plant status and thus of the accident and its progress. Effective, appropriate decisions to control and mitigate an accident are dependent on making this assessment of the accident. The objective of this paper is to stimulate consideration of unconventional plant information sources through discussion of specific examples.

A plant's condition during an accident can be characterized by plant parameters such as temperatures and pressures and by plant system operational status. For example, core damage is associated with increasing temperatures, pressures, and radiation levels in many different systems and plant areas. Reg. Guide 1.97 instrumentation exists to provide information to allow operators to take specified manual actions (Type A), to indicate whether plant safety functions are being accomplished (Type B), to indicate the potential for breach of barriers to fission product release (Type C), to indicate operability of individual safety systems (Type D), and to indicate the magnitude of radioactive material releases (Type E). Reg. Guide 1.97 instrument range requirements, with the exception of pressure instruments, address conditions up to design basis conditions. Pressure instrument range requirements exceed design basis conditions. During a severe accident, some instruments may not see conditions beyond their design basis.

Effective accident management includes the ability to establish a consistent picture of the accident by accumulating information from as many sources as is practical. Operability of systems and components, and non-safety related temperature, radiation, pressure, and water-level indication can be used to directly indicate, measure, or infer plant parameters which confirm, augment or replace those otherwise available. Innovative uses of information sources thus serve to increase the diversity and flexibility of accident data available. Both the value and rate of change of key

plant parameters can be directly measured or inferred from plant instrumentation. Even when the actual measured values for a particular instrument are less accurate than normally required, valuable information may still be obtained. Trends observed in the time-dependent behavior of that instrument can provide valuable information in how the accident is progressing. The failure of instruments themselves (especially those for which defined service condition requirements exist) can yield information about plant status.

Selected plant-specific examples of unconventional uses of plant information sources will be discussed in detail. Monitoring of secondary side cooling water system temperature can be used to infer core status and the ability to successfully remove heat to the ultimate heat sink. Containment/reactor vessel temperature can be used to infer core status and heat removal requirements. Tank pressure indication (e.g. pressurizer relief tank, waste gas surge tank) can be used to infer containment status. Tank level indication can be used to infer loss of inventory, system or containment overpressurization. Water temperature increases, pump discharge pressures and pressure drops across pumps can be used to infer containment status. Use of process or area radiation monitors can be used to infer core status and fission product releases.

Considerations regarding the use of unconventional information sources will be discussed. Plant-specific operator actions and procedural requirements may have significant impact on assessment of plant status. Instrument range capabilities may be significantly larger than the required/indicated range. Instrument accuracy requirements during an accident may be less stringent, allowing the use of marginally functional instruments. Information from locally indicated instrumentation may be unavailable due to harsh environments. (However, the presence of such environments would represent additional valuable information.) Correlation of instrument data to accident conditions such as temperature, pressure, core status, and magnitude or rate of radiation releases may require development of calculations and associated tables or curves (this is being addressed in an EPRI project described in a separate paper). Such analyses could be developed in advance.

In summary, this paper is intended to stimulate consideration of creative ideas for use of existing information sources to successfully manage an accident. Through the examples and discussion presented, it is intended to demonstrate that obtaining information to manage an accident can come from a variety of sources, some of which are unconventional.

INTRODUCTION

The goal of this paper is to initiate and facilitate further consideration and discussion of unconventional information sources that are useful in Accident Management. Accident Management consists of actions taken during the course of an accident by the plant operating and technical staff to prevent or minimize off-site radiation releases, gain control and return the plant to a safe state, and minimize damage to the plant. In addition, Accident Management consists of administrative and programmatic efforts to plan for and support actions taken by plant staff. Inherent in accomplishing these goals is obtaining as clear a picture as is practical of the nature of the accident and plant status. Development of a consistent and coherent understanding of the accident and plant status requires plant staff to evaluate and interpret data from a wide range of sources. This paper provides an overview of the breadth of information sources potentially available during an accident and discusses key issues related to evaluation and interpretation of the information from these sources.

Most plants already have some established sources from which information about the plant during an accident can be obtained:

- Plant Instrumentation, including Reg. Guide 1.97 instrumentation
- Information sources identified in abnormal operations or emergency operations procedures

However, the above information sources have some limitations. For example, past probabilistic risk analyses have shown that events involving loss of key electrical support systems can be significant contributors to core damage. Such events could jeopardize or degrade instrument availability. Plant-specific accident procedures and interpretation of instruments intended for design basis events may not be applicable in severe accidents. Information sources such as other NSSS and BOP instrumentation may be available.

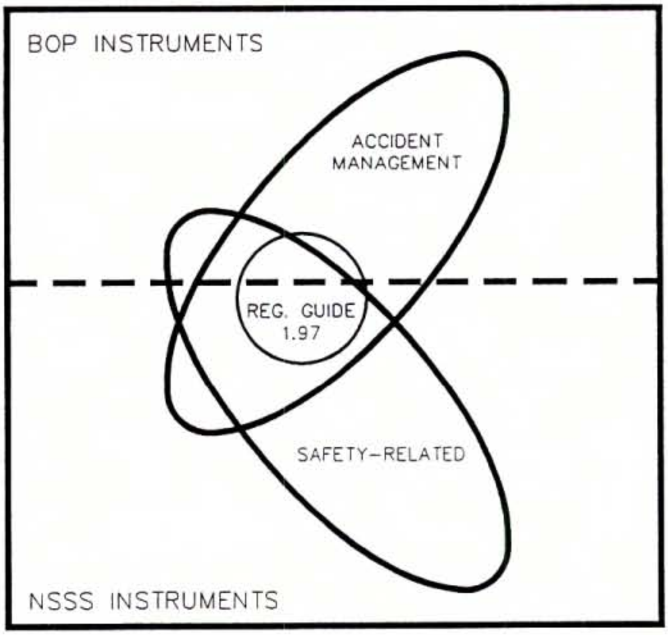
A diversity of other potential information sources exist which may yield valuable information regarding the plant status during either a design basis or severe accident. This paper examines some of these other unconventional sources of plant information and how they could be used in accident management. Examples include inferences of plant status from the operability condition and location of systems and components, measurements and trends from instrumentation not normally intended to function during an accident, and the presence of local harsh environments. Thus, the information resources available for accident management consist of Reg. Guide 1.97 instruments as well as many other potential sources.

Figure 1 depicts the conceptual relationship between the various instruments that comprise resources for accident management. It may be argued that the degree of overlap between accident management and safety-related instruments should be greater or that the relationship should be depicted differently. We have presented Figure 1 this way intentionally to provoke discussion. Safety related instruments are a subset of balance of plant (BOP) and NSSS instrumentation. Reg. Guide 1.97 instruments are a special subset of safety related instruments dedicated to accident monitoring for design basis events. Instruments useful in accident management include Reg. Guide 1.97, some safety related instruments, and some non-safety related BOP and NSSS instruments.

This paper provides an overview of some unconventional plant information sources with the objective of stimulating consideration of this aspect of accident management. This overview does not represent systematic evaluation of all potential information sources, but rather a sampling of some ideas gained from a brief investigation. As a means to validate the concept of innovative

Figure 1

SCOPE OF INFORMATION SOURCES
FOR ACCIDENT MANAGEMENT



uses of specific instruments, a detailed discussion of four specific ideas and their application at a plant is presented. A discussion of important issues to be considered in using more unconventional information sources is also provided. EPRI is conducting a systematic (though not comprehensive) evaluation of accident management information sources and their application in a separate project that is currently underway.

OVERVIEW OF UNCONVENTIONAL PLANT INFORMATION SOURCES

There are several ways in which additional information regarding plant status during an accident can be obtained. Any instrument that is still functioning may provide direct or indirect measurement of plant conditions from which valuable information regarding overall plant status and accident status can be obtained. Observations of system, subsystem, or component operability, inoperability, or failure may imply the presence or lack of plant conditions which confirm or clarify plant status. In addition, surveys taken to establish accessibility to key plant areas may identify locally harsh conditions which indicate plant status and accident status.

Examples of the type of information on plant status that is desirable in an accident include:

- Reactor vessel pressure
- Reactor vessel water level
- Core temperatures
- Fission and decay power level
- Containment pressure
- Containment temperatures
- Containment water level
- Containment radiation levels
- Containment relative humidity
- Reactor/Auxiliary Building room pressures
- Reactor/Auxiliary Building room temperatures
- Reactor/Auxiliary Building room water levels
- Reactor/Auxiliary Building room radiation levels
- Reactor/Auxiliary Building relative humidities

Although many plant instruments are normally intended for use only during non-accident conditions, these instruments may have the capability to measure a parameter over a much wider range of conditions.

Figure 2 conceptually depicts the potential applicability of many classes of plant instruments to accident management. Plant instrumentation is often required to function over a narrower range of plant conditions than those over which it is capable of operating. For example, pressure sensors similar to those found in fossil power plant boilers can measure pressures well in excess of normal process conditions. Many instruments may provide useful information under severe accident conditions.

The following subsections provide an overview of the range of potential sources of information that may be available during an accident. Examples of direct and indirect measurement of key plant parameters are given. Examples are given of how inferences can be made from observations of component operability/inoperability.

DIRECT MEASUREMENT OF KEY PLANT PARAMETERS

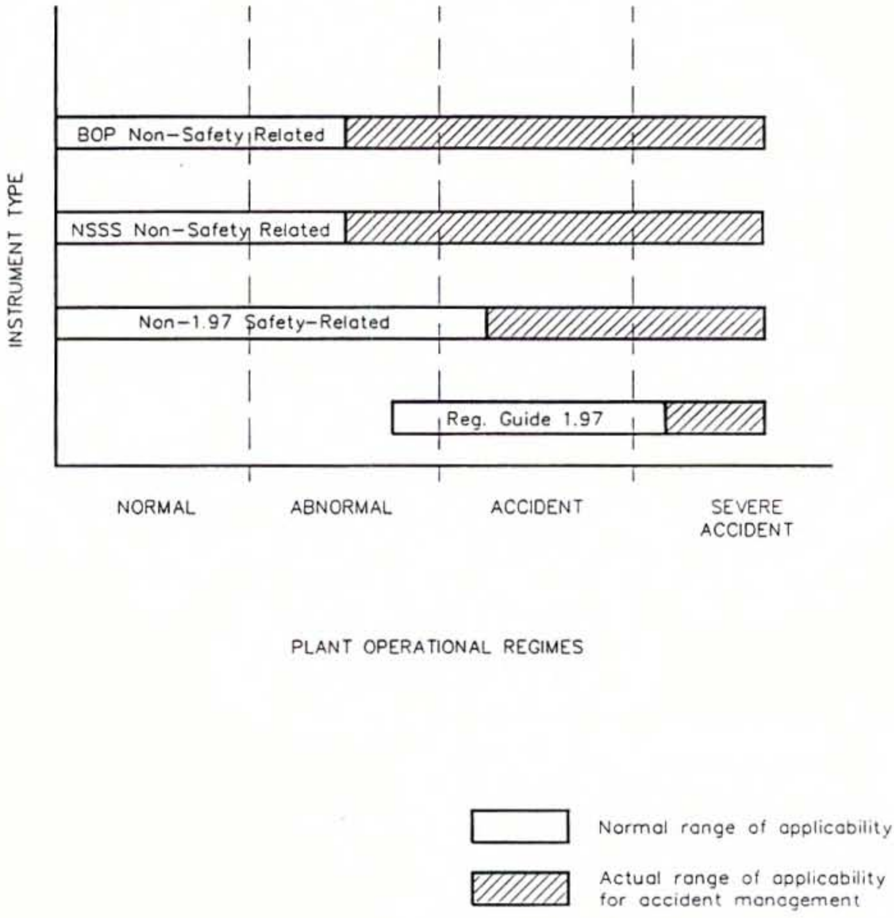
A variety of instruments may provide direct measurements of key plant parameters which indicate plant status. Some of these instruments are addressed in the scope of Reg. Guide 1.97:

Source Range Monitors

In PWRs, source power, intermediate, and full power range radiation detectors exist outside of the reactor vessel to track power during startup, shutdown, and normal operation. These devices are part of the Reg. Guide 1.97 scope and directly measure thermal neutron flux. An increase in

Figure 2

POTENTIAL APPLICABILITY OF INSTRUMENTS TO ACCIDENT MANAGEMENT



measured flux could indicate reduced water shielding due to decreasing core water level or increase in the void fraction in the core. This symptom could provide a very early indication of an accident. (This effect was observed in the TMI-2 accident, for example.)

Heated Junction Thermocouples (Reactor Water Level)

This is the reactor vessel level indication system used in most CE PWRs. It is part of the Reg. Guide 1.97 scope and directly indicates the coolant level in the core. For BWRs, Reg. Guide 1.97 specifies BWR Core Thermocouples as a diverse means of indicating water level. However, these are incore instruments and may not function during a severe accident.

Cavity/Reactor Vessel Temperature

Some plants have temperature sensors located in the reactor pressure vessel cavity or pedestal area as part of the cavity ventilation cooling system. In some plants, these sensors are located approximately at reactor vessel midplane and just below the bottom of the reactor vessel. Significant temperature increases or a rapidly increasing trend could indicate core damage or vessel failure.

INFERENCES OF PLANT STATUS FROM OTHER INSTRUMENTS

A variety of instruments may provide information from which inferences about the plant status can be made:

Heat Exchanger Outlet Water Temperature

Both PWRs and BWRs use service water systems to provide cooling to essential and non-essential heat loads such as reactor coolant pumps and containment air coolers. For example, PWR reactor coolant pump seals are sometimes cooled indirectly through service water or component cooling water heat exchangers. Therefore, an increase in heat exchanger outlet temperature could signify an increase in primary coolant temperature. A trend in heat exchanger outlet temperature could be correlated to the expected trend in primary coolant temperature resulting from various accidents.

Tank Pressures and Levels

Key tanks such as accumulator tanks, quench tanks, and condensate storage tanks are addressed by Reg. Guide 1.97. There are other tanks from which valuable information may be inferred. In some Babcock & Wilcox PWRs, the reactor coolant makeup tank is instrumented for temperature, pressure, and level indication. These parameters may be correlated to primary coolant conditions depending upon the type of accident and whether the tank is isolated. In some Westinghouse PWRs, the component cooling water system is equipped with surge tanks which are instrumented for level and pressure indication. An increase in surge tank pressure or tank level may indicate an interfacing system LOCA through thermal barrier cooling coils. A rapidly decreasing tank level may indicate a line break in the component cooling water system which would indicate a lack of cooling to safety related components cooled by the component cooling water system.

Process Radiation Levels

Reg. Guide 1.97 requirements include virtually all primary coolant radiation monitors, SGTS radiation monitors (BWRs), containment effluent monitors (PWRs), and condenser effluent monitors. However, process radiation monitors in other systems such as service water/component cooling water may indicate presence of radioactive material which may indicate some fuel damage or an interfacing system loss of coolant. Radiation monitors in the radwaste system (e.g. the waste gas system) may be used to indicate increasing levels of radioactivity in gaseous and liquid radwaste, possibly indicating fuel damage. Stack radiation monitors may be used to trend increasing radiation releases and correlate them to fuel damage.

Pump Inlet/Discharge Pressures

Changing pressure conditions at the inlet and/or discharge of a pump may indicate depressurization, loss of coolant, or increasing containment pressure. Many pumps have pressure indication on the discharge and/or inlet sides. For example, increasing temperature in the primary coolant and related components may result in an increase in component cooling water temperature, which will increase system pressure. This pressure effect will impact pumps in the cooling water system. In addition, changing pressure and temperature conditions in the reactor core and containment after an accident may be correlated to pressures seen at spray and injection pump discharges and inlets.

Ultimate Heat Sink Temperature

If the ultimate heat sink is not a natural body of water, then the temperature of the cooling ponds or cooling tower basin water may be an indirect indicator of increasing coolant temperature and increasing component temperatures. An increasing temperature trend may indicate a loss of core cooling or a loss of primary coolant to a secondary system (i.e. an interfacing system loss of coolant).

INFERENCES OF PLANT STATUS BASED ON SYSTEM/COMPONENT OPERABILITY

The observed operability/inoperability of systems and components and observations of plant conditions can provide information about the plant conditions during an accident. In this sense, the plant operators themselves constitute an extremely valuable source of information. Harsh environmental conditions related to the accident (radiation, temperature, humidity) may exist in certain areas where equipment is not functioning. The location and nature of the environment (obtained from local surveys or remote instrumentation) can provide information regarding other systems and components that may be affected and additional information about how the accident is progressing. A component or system may not operate due to system conditions caused by the accident, e.g. pressure reaching pump shutoff head, isolation of non-safety related electrical systems, etc:

- Failure of temperature measurement devices in or near the reactor (e.g. RTDs, heated junction thermocouples (HJTCs), core exit thermocouples (CETs) may indicate minimum temperature conditions in the core.
- This information would be valuable in estimating core damage and potential for subsequent releases.
- Successful operation of a component may indicate the lack of adverse conditions, e.g. continued function of components that are not expected to operate when submerged provides some indication of containment water level.

DETAILED EXAMPLES

The following sections discuss four specific examples of unconventional sources of plant information during an accident. Each example will be discussed in terms of how it would be applied at a plant. These examples are plant-specific in some cases and may not be typical of a given reactor type. The examples are as follows:

- Reactor Vessel Pressure and Level (BWR)
- Heat Exchanger Outlet Temperature (CE PWR)
- Cavity Temperature (W PWR)
- Pressurizer Relief Tank Pressure (B&W PWR)

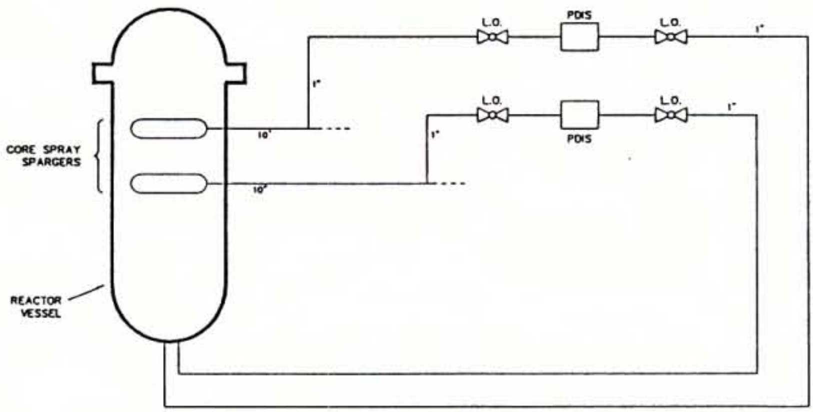
Reactor Vessel Pressure and Level (BWR)

Reactor vessel pressure instruments in the Core Spray System and systems with lines directly entering/exiting the reactor vessel at BWRs have potential applications to measure reactor vessel level and pressure. Core spray differential pressure indication as discussed below could be used to provide additional confirmation of the core water level. Use of RWCU pressure indication as discussed below can provide early indication of a loss of reactor pressure.

Reactor vessel level indication is possible via differential pressure indication associated with the Core Spray System (Fig. 3). Pressure sensors located on the core spray inlet lines to the reactor vessel measure the pressure difference between the core spray sparger elevation in the reactor vessel and the bottom of the reactor vessel. This enables detection of core spray line breaks, and in some BWR designs allows different core spray loops to be initiated or suppressed selectively. These instruments are typically indicated on Core Spray panels. In general, the measured

Figure 3

BWR CORE SPRAY SYSTEM
DIFFERENTIAL PRESSURE SWITCH CONFIGURATION

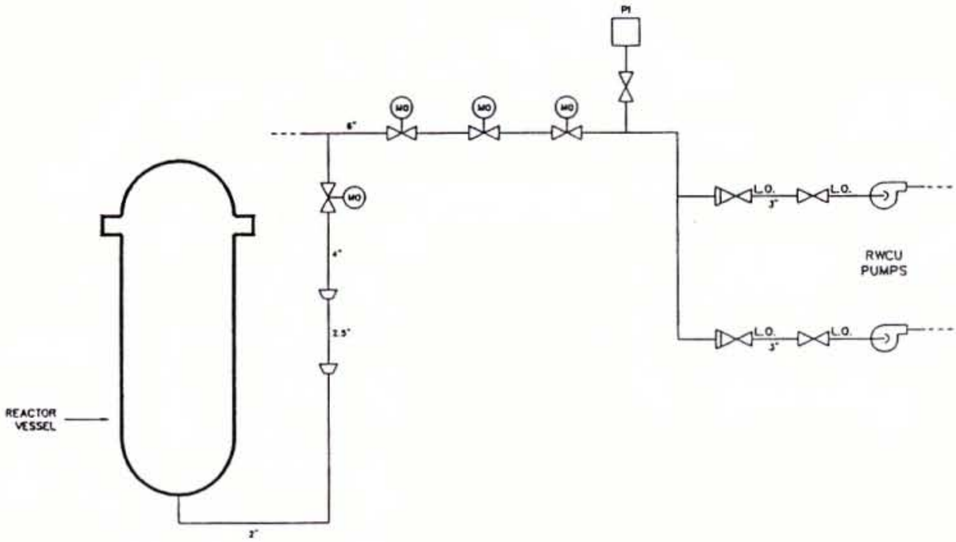


pressure difference will be the sum of the pressure differential due to flow through the jet pumps and the static head associated with the reactor vessel water volume. Depending on the type of accident and the source of makeup water entering the core, flow through the jet pumps may or may not exist. A knowledge of the flow rate entering the jet pumps could allow calculation of the pressure differential associated with the jet pumps. If the water temperature is known, the water density can be estimated. The static head can then be estimated as the difference of the measured pressure differential and the pressure change across the jet pumps. Knowledge of reactor vessel geometry and the static head can be used to estimate the reactor vessel water level. Many of these calculations could be prepared in a parametric manner to allow determination of reactor vessel level. Input parameters would be recirculation/injection flow rate and reactor vessel water temperature. Applicability of this particular idea is limited to BWRs with differential pressure indication on core spray lines.

Direct measurement of reactor vessel pressure is possible using Reactor Water Clean-up (RWCU) System pressure instruments (Fig. 4). Reactor vessel pressure indication is provided on the line from the reactor vessel to the suction of the Reactor Water Clean-up (RWCU) Pumps. This instrument is typically indicated on the RWCU panel. This line is isolated by motor operated valves. Since the RWCU system operates during full power operations, an event resulting in depressurization would be sensed by RWCU pressure indication prior to isolation. Subsequent to isolation, it would be necessary to open these valves and to close the RWCU pump suction inlet valves to maintain isolation. Since RWCU systems typically operate under full power conditions, the range of RWCU pressure instrumentation would be expected to include pressures in excess of 1000 psia. In addition, it is often the case that pressure sensor actual range capability is much greater than the range requirement (as discussed earlier in this paper).

Figure 4

BWR RWCU SYSTEM PRESSURE
INDICATION CONFIGURATION



Heat Exchanger Outlet Temperature (CE PWR)

It is possible to indirectly estimate trends in core temperature by monitoring the temperature of secondary cooling water systems which are providing cooling to components in contact with primary coolant.

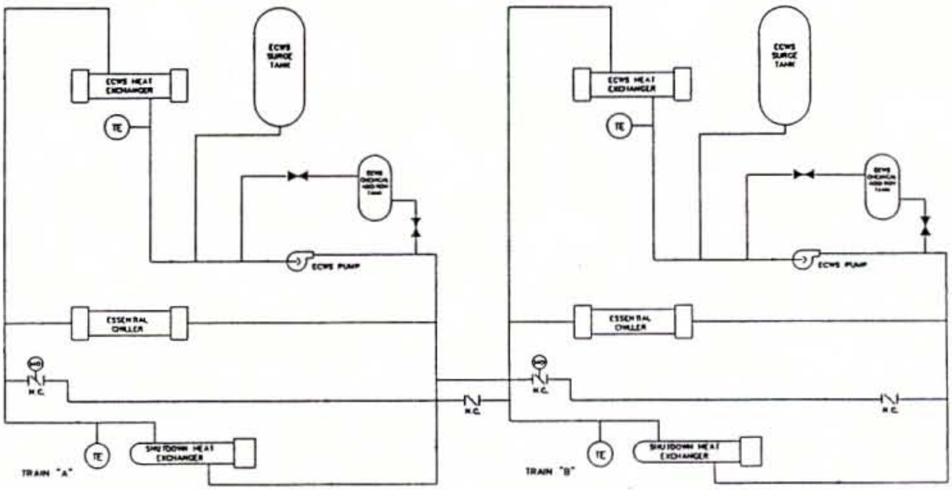
At CE PWRs, essential equipment is typically cooled by a dedicated service water system which is only used during an accident. This system is designated the Essential Cooling Water System (ECWS). This system is a closed loop cooling system which removes all heat necessary to safely shutdown the plant and which rejects this heat to an ultimate heat sink. Figure 5 depicts a typical ECWS configuration. The ECWS includes two heat exchangers which reject heat to essential spray ponds. Each ECWS heat exchanger is equipped with a locally indicated temperature instrument which is normally used to monitor heat exchanger performance. This instrument, if accessible, could be used to trend ECWS water temperature and thus indirectly trend primary coolant temperature.

Additionally, the essential cooling water temperature at the outlet of the shutdown heat exchangers is also monitored. The advantage of this instrument is that it is usually indicated both locally and in the control room.

This idea is generally applicable to most plant designs because all plants have an ultimate heat sink and a method of transferring heat from primary systems and components to the ultimate heat sink.

Figure 5

CE PWR ESSENTIAL COOLING WATER
SYSTEM CONFIGURATION



Cavity Temperature (W PWR)

Indirect indication of core temperature and its rate of change is possible through monitoring the cavity temperature. Core temperature trends can be important in determining the potential for core damage and radiation releases. Temperature elements associated with the cavity ventilation system are sometimes located in the reactor cavity adjacent to the reactor vessel below the inlet/outlet nozzle elevation and at the bottom of the reactor vessel (See Fig. 6). These temperature elements are used to sense cavity air temperature. Cavity ventilation systems must maintain cavity air temperature below certain design maximums. These temperature elements can be used to monitor trends in core temperature based on the assumption that the cavity air temperature is proportional to core temperature. The temperatures measured by these elements are indicated in the control room, so the information is readily available. Frequently, there are other containment ventilation systems with associated temperature elements that may also be used in a similar manner.

The applicability of this idea to other plants depends on the presence of a cavity ventilation system and associated temperature sensing devices. The location and distribution of temperature sensors is also important. However, many plants have other types of containment ventilation systems, any of which may have temperature sensors in useful locations.

Pressurizer Relief Tank Pressure (B&W PWR)

Relief tanks are typically installed in PWRs to accommodate steam releases from the pressurizer (See Fig. 7). A relief valve between the pressurizer and the relief tank provides a means to prevent overpressurization of the pressurizer. Relief tanks are designed to handle pressures on

Figure 6

W PWR REACTOR CAVITY TEMPERATURE INDICATION CONFIGURATION

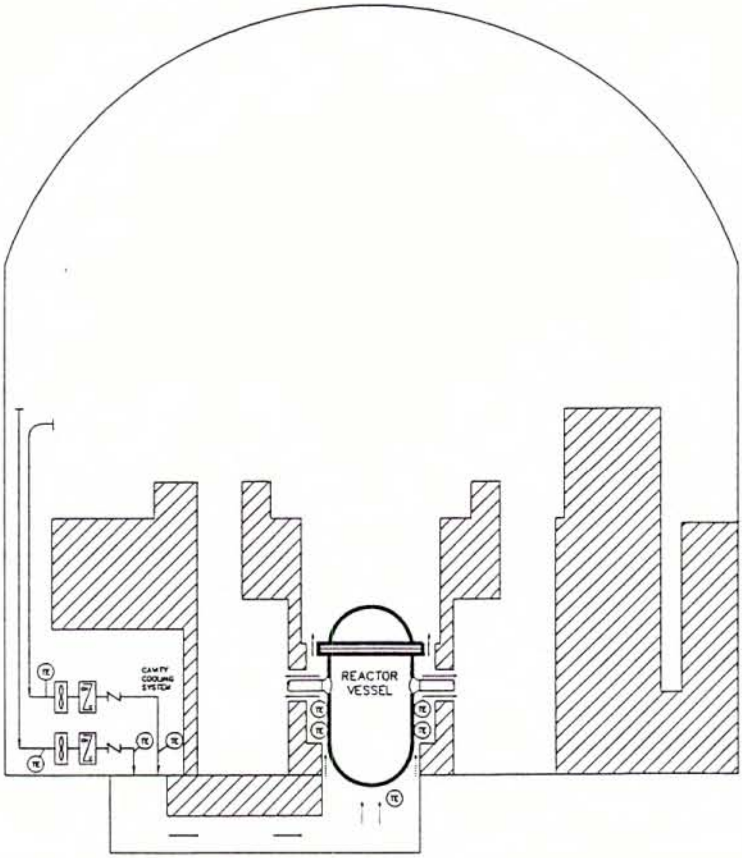
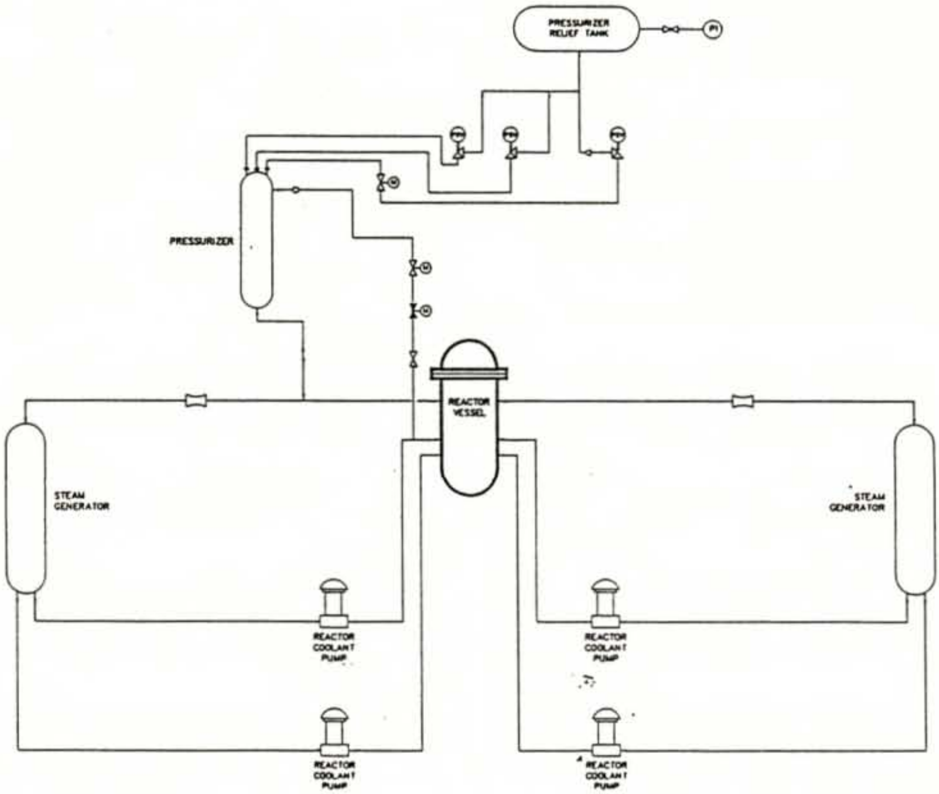


Figure 7

B&W PWR PRESSURIZER RELIEF TANK CONFIGURATION



the order of 100 psig. Monitoring relief tank temperature, pressure, and/or water level can help to detect events involving depressurization through stuck open pressurizer relief valves.

These relief tanks are also installed with rupture disks to relieve overpressure in the tank resulting from planned "feed and bleed" operations and to provide some overpressure margin. Although some rupture disks are designed to rupture at the preset pressure in one direction only, a failure of the tank rupture disk coupled with relatively low tank pressure indication may occur. Such a failure would indicate high containment pressure. Subsequent to such a rupture disk failure, the tank pressure indication would indicate containment pressure and could be used to monitor containment pressure. Thus, "inward" failure of the tank rupture disk is a kind of high containment pressure alarm and indicator.

This idea is generally applicable because pressurizers and associated relief tanks are present in almost all commercial PWR designs. However, the "setpoint" of the relief tank rupture disk may vary and is very important in interpreting the failure of a relief tank rupture disk.

CONSIDERATIONS

Application of the above ideas and similar concepts also requires consideration of several key issues related to the availability and applicability of the information obtained:

- Many of the instruments that may be used may have the capability to measure a parameter over a much wider range than that required, indicated, or specified by the vendor for nuclear applications. Therefore, a good understanding of actual instrument ranges and the accident range for the variable of interest would be advantageous.

- Instrument accuracy requirements during an accident may not be as stringent. Approximate measurements or the ability to measure a trend may be sufficient.

- The location of instruments must be considered with respect to expected environmental conditions resulting from an accident. Instruments providing useful data may not be located in accessible areas.

- Successful correlation of measured parameters to key plant parameters such as core temperature or containment pressure may require analysis and assumptions to obtain a useful correlation. Many of these analyses could be completed in advance in a parametric manner, facilitating their use.

- Operator actions may affect the assessment of plant status. Required operator actions during a particular accident should be considered when developing the expected conditions and expected instrument responses for a given accident.

- Portable instruments could be used to measure data over a wider range or to higher accuracy. Such instruments could be connected to existing instrument channels.

CONCLUSIONS

The purpose of this paper is to demonstrate the diversity of other information sources from which valuable information could be obtained regarding plant status during an accident. While the above examples do not represent a comprehensive list of possible information sources, they indicate the potential that exists for obtaining useful information during an accident from sources other than dedicated accident instrumentation.

It is hoped that further discussion and consideration will be given to the wide array of potential information sources available during an accident when developing accident management guidance. Plant specific characteristics and potential limitations must also be evaluated. This paper shows that several other potential sources of information exist to help confirm the understanding of plant status during an accident.

Discussions with other nuclear professionals in preparing this paper have convinced us that the nature and diversity of the ideas discussed in this paper indicate that such information sources exist at many plants. Strong plant specific knowledge of the plant response in various conditions coupled with a good understanding of expected conditions during accidents will allow the identification and application of a wide range of information sources to assist in effective accident management.

OECD NUCLEAR ENERGY AGENCY

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

SPECIALIST MEETING ON

INSTRUMENTATION TO MANAGE SEVERE ACCIDENTS

Cologne, Germany
16th-17th March 1992

Internal Gamma Activity Used For Water Level Indication

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Abstract: Many pressure vessels show strong internal gamma sources which can be used for water level measurement. Following two possibilities are descripted for such applications. Some experimental results are shown for one method .

1. Introduction

The monitoring of the water level in pressure vessels is endowed with high safety technical importance in all power plants with water cooled reactors. Such pressure vessels are the reactor, the steam generator and the pressurizer. Up to date the water level indication is realized with measuring systems which are based on the measurement of pressure differences. If there are boiling conditions then it exists ever the danger of relativ great measuring faults caused by the void generation in the comparision tube during great negativ pressure gradients. In order to undermine this faults in nuclear power plants there are some different systems measuring the pressure difference. But this isn't a methodical diversity. After the accident in TMI-2 worldwide activities there have been in order to develop other water level measuring methods which are showing the importance of the problems. /1/,/2/,/3/,/4/
One unconventional water level measuring method is based on the utilization of the internal gamma activity which exists in all pressure tanks reckoned up above. For the differ tanks are derived different measuring algorithms in reason of different properties of the gamma sources. But some principles are likewise valid for all applications.

- The aim of these researchs is the development of divers measuring systems for internal parameters of pressure vessels with the property that it should be used Out-core gamma detectors only. Those have the advantage of a smaller probability of destruction of the detectors in accidents in comparision to in-core detectors. Possible applications are the water level measurement in reactors and steam generators but the indication of core smelting too. For the fitting there is not the necessity of constructive modifications.

- The results of the gamma measurement are connected with results of a pressure measurement or a pressure and temperature measurement in the tanks to determine the mixture level.

- In order to eliminate the dependance on the gamma source strength on the reactor is used a composite detector. This detector is arranged at a point at which water level changes have not an influence on the detector signal.

- The measuring algorithms are based on measurement of the N-16 radiation from the primary water. This measuring system have a time limitation regarding availability because the Nitrogen N-16 radiation exists during power operation and some minutes after shutdown only. But such a system is available for the most and most important situations.

2. Water level indication on the steam generator with U-tubes

2.1. Description of the measured object

The gamma source is the radioactive primary water in the U-tubes. The source strength is a function of reactor power, the burnup condition, the concentration of oxygen O-16 in the primary water, the massflow of the primary water and the concentration of activated corrosion products and fission products in the primary water. The Nitrogen N-16 dominates absolutly in that gamma source. It is generated by the neutron capture reaction:

O-16 (n,p) N-16

Nitrogen N-16 radiates with two gamma energies, 6.1 and 7.1 MeV and have in the steam generator an average concentration about 100 $\mu Ci/cm^3$ during full power.
The level of the secondary water is the measured variable. The secondary water is a two phase mixture with an unknown density and an unknown density distribution. The water level is normally in the area of the water steam separator.

2.2. Characteristics of the water level measurement

The measuring algorithm is based on the transmission method similar other applications but without using a point source but a big cylindrical volume source. At first this method will be attributed to a point source problem how it is shown in fig 1.

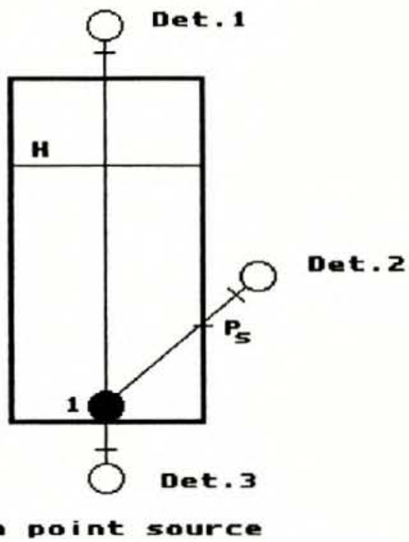


Figure 1: Water level measurement under point source conditions to representation of the measuring principle for a steam generator

The gamma flux density in the detector position can be calculated with the exponential law

$$\phi = \phi_o/4/\pi/l^2 \cdot K \cdot \exp(-m \cdot \varrho \cdot h) \quad (1)$$

ϕ : gamma flux density $[cm^{-2}s^{-1}]$
l : distance source detector [cm]
K : attenuation factor of construction material
m : mass absorption coefficient of water $[cm^2/g]$
ϱ : density of water $[g/cm^3]$
h : water level over the source [cm]

ϕ_o is known with using of the detector 3 and the product $\varrho \cdot h$ can be detemine with eq.(1). In order to calculate the water level H over the gamma source it is necessary to determine the water density. If the water in the single phase state then the density can be determined with the measurement of pressure and temperature but if the water in two phase state then the water density of the boiling-line can be determined only. Therefore the mass level only is definable. If the detector 2 is situated so that the point P_s is under the water level H then is the average water density definable and we can calculate the mixture level.

2.3. Measurement for a steam generator

With the axial detector should be measured the mass level over the U-tubes on principle, detector 2 is used for the mixture level detection and detector 3 is used for the elimination of the gamma source properties.

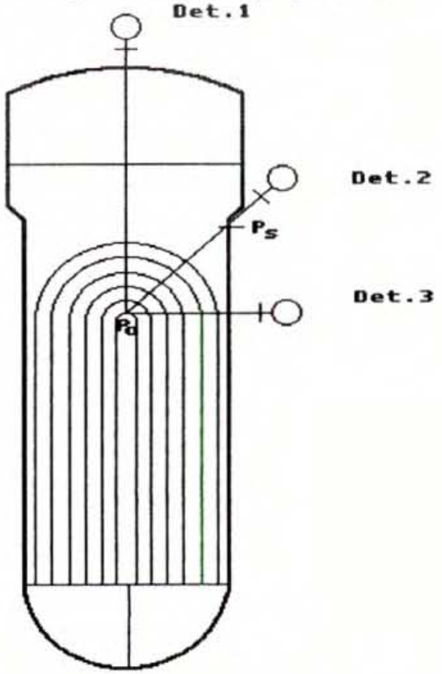


Figure 2 : Steam generator with U-tubes and the gamma detector system for water level measurement

Some principles and suppositions are valid for the detector arrangement.

1. Through the strong self attenuation of the U-tubes the detectors can't look very deep in the U-tube bundle. Calculations showed that the gamma flux at the detector 1 is caused to 99% by the sphere area of the U-tubes. In this reason we consider further the sphere only.

2. The detectors are so situated that their axises cross in the point P_0. Therefore the influence of the portion U-tube material on μ_q is undermined and μ_q is about equal for all detectors.

3. In the calculation is considered the uncollided gamma radiation by N-16 only. This reduction doesn't have an influence on the principle of this measuring method.

The source volume is composed by the three components primary water, U-tube materials and secondary water. This volume is homogenized for the following considerations regarding the gamma activity and the linear attenuation coefficient. We can suppose that the specific activity a is constant in this area. The linear attenuation coefficient which is descripting the self attenuation of the gamma source is calculated to

$$\mu_q = v_1 \cdot m \cdot \varrho_1 + v_2 \cdot m \cdot \varrho_2 + v_u \cdot \mu_u \qquad (2)$$

with

$$v_1 = V_1/V_q \,, \qquad v_2 = V_2/V_q \,, \qquad v_u = V_u/V_q \qquad (3)$$

μ : lin. attenuation coefficient [cm^{-1}]
v : volume portion
V : volume [cm^3]
index : 1 - primary water
2 - secondary water
q - source
u - U-tubes

We can calculate the three detector signals to

$$s_1 = \int_{V_q} a \cdot K_1/r^2 \cdot \exp(-\mu_q \cdot L_1 - \sum_i(\mu_{1i} \cdot L_{1i}) - m \cdot \varrho_{w1} \cdot L_{w1}) \, dV \qquad (3)$$

$$s_2 = \int_{V_q} a \cdot K_2/r^2 \cdot \exp(-\mu_q \cdot L_2 - \sum_i(\mu_{2i} \cdot L_{2i}) - m \cdot \varrho_{w2} \cdot L_{w2}) \, dV \qquad (4)$$

$$s_3 = \int_{V_q} a \cdot K_3/r^2 \cdot \exp(-\mu_q \cdot L_3 - \sum_i(\mu_{3i} \cdot L_{3i})) \, dV \qquad (5)$$

a : specific activity [$s^{-1} \cdot cm^{-3}$]
K : $K = E_d/4/\pi$ with E_d : detector efficency
r : distance from source point to detector [cm]
L : thickness of material layer [cm]
index : 1 - detector 1
2 - detector 2
3 - detector 3
w - secondary water
i - number of construction material sheet

Eq.(3) have four unknown variables a, μ_q, ϱ_w and L_w. This underdetermination requires a combination from relative value measurement and calibration. For this proposal the following signals are created.

$$S_1 = s_1/s_3 \qquad (6)$$
$$S_2 = s_2/s_3 \qquad (7)$$

Every change in the gamma source properties activity or self attenuation leads to about uniform changes in all three detector signals s_1, s_2 and s_3. Because in s_3 isn't a term for the gamma attenuation in a secondary water layer the quotient generation eliminates in eq.(6) and eq.(7) the properties of the gamma source. With that S_1 and S_2 depend on secondary water density and secondary water level only. With S_1 is determined the product $P_1 = \varsigma_B \cdot H_1$ from a calibration characteristic which was created in experiments in which the water level was changed at known water density. The mass level is determined with help of the product P_1. It can be written

$$P_1 = \varsigma_B \cdot H_1 = \varsigma_{w1} \cdot h_{w1} = S_1 = \varsigma_{m1} . h_{m1} \qquad (8)$$

H_1 : calibration water level for detector 1 [cm]
ς_B : comparision water density during calibration [g/cm^3]
h_{m1} : mass level [cm]
ς_{m1} : average density for mass level [g/cm^3]

However, h_{m1} is calculated to

$$h_{m1} = \varsigma_B \cdot H_1 / \varsigma_{m1} \qquad (9)$$

The value for ς_{m1} have to determined from pressure and temperature measurements. We have to distinguished two cases.

Case 1. if pressure $p < p_{boil}$ then $\varsigma m1 = f(p,t)$ (10)
Case 2. if pressure $p = p_{boil}$ then $\varsigma m1 = f(p_{boil})$ (11)

In order to determine the mixture level the quotient S_2 is analysed. If the assertion 'h_{m1} over P_s' true then the thickness of the secondary water layer is ever the same between the souce and the detector 2. Then it is possible to determine the water density with help of the calibration product P_2.

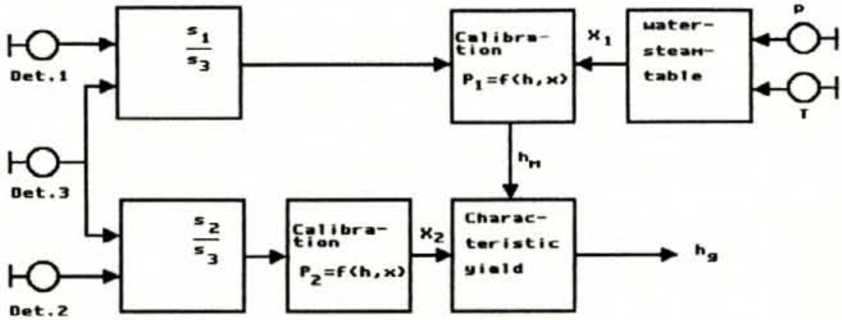
$$\varsigma_2 = S_2 / P_2 \cdot \varsigma_B \qquad (12)$$

Now the void fraction α_2 can be calculated to

$$\alpha_2 = (\varsigma_2 - \varsigma'') / (\varsigma' - \varsigma'') \qquad (13)$$

With it we get the two parameters h_{m1} and α_2. With the help of thermal hydaulic calculations it is necessary to generate a characteristic yield which is used for the determination of the mixture level h_g.

$$h_g = f(h_{m1}, \alpha_2) \qquad (14)$$



legend: x_1 – density of secondary water on the boiling line
x_2 – void fraction of secondary water for detector 2
h_g – mixture level
h_M – mass level

Figure 3: Measuring quantities and measuring algorithm for water level indication on steam generator with U-tubes

2.4 Experimental works

A part of the descripted algorithm was verified in some experiments at the zero power reactor ZLFR at the Technical University Zittau. On this reactor was realized a two detector system. The first problem was the generation of the calibration characteristic. It was solved with changed water levels over the core at a constant reactor power. The result was the characteristic in fig.4. /5/

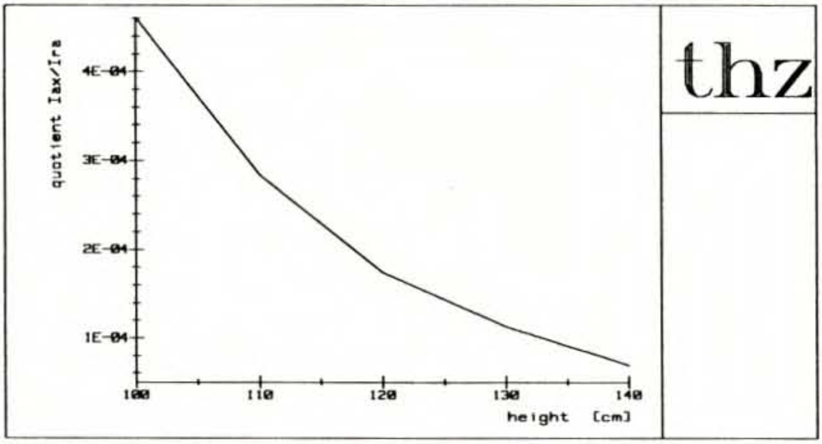


Figure 4: Calibration characteristic for the water level indication on the ZLFR

The second problem was the check of this characteristic. In these experiments the water level was constant and the reactor power was changed. The results are in fig.5 and fig.6.

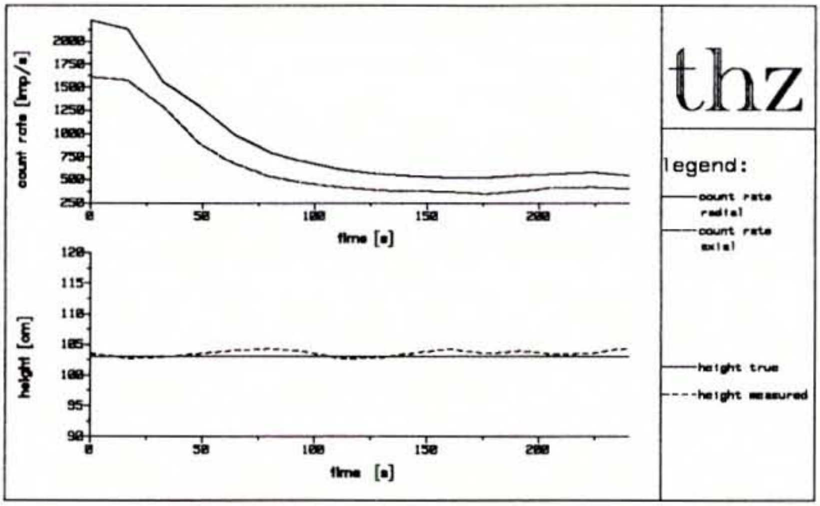


Figure 5: Water level measurement using internal gamma activity on the ZLFR at negative power gradient

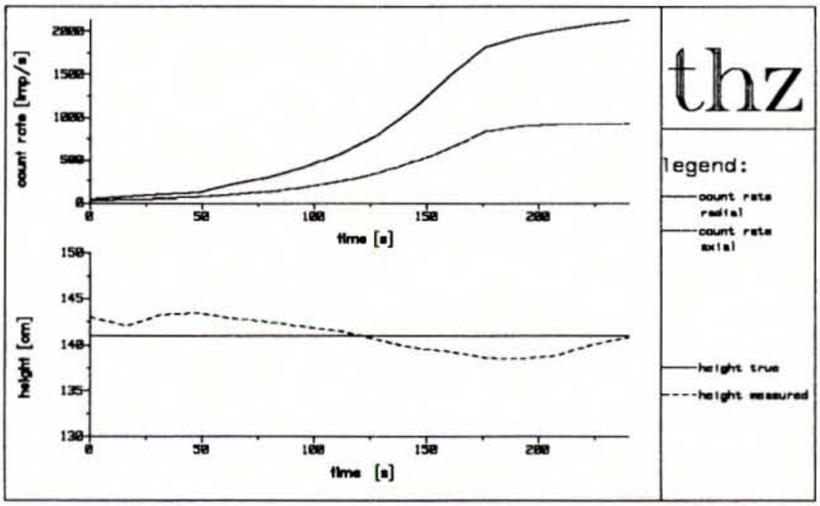


Figure 6: Water level measurement using internal gamma activity on the ZLFR at positive power gradient

The maximum divergence was +/-2.5 cm. The reason of these divergences is the occupation of the statistical measuring faults. In proportion to the measured point the statistical faults amount to between 0.6% - 5.5%. These experiments proved that the quotient S_1 is independed of the reactor power respectivly of the source activty.

3. Water level indication on Boiling Water Reactors (BWR)

3.1 Measured object description

The measured quantity is the primary water level. The permissible range is between one higher and one or two lower limits in the area of the water steam separator. The water is in two phase state too.
But in the BWR exist contrary to the steam generator two important gamma sources. The one is the core and the other one is the primary water itself. The primary water can be activated over distinguish ways.

1. transmission of fission products from the fuel elements
2. pull of activated corrosion products
3. activated elements in the primary water

During power operation the Nitrogen N-16, which is produced over way 3, generates the absolute greatest portion of the gamma flux out side of the vessel in the area of the water steam separator. This is caused by the high gamma energy too (6.1 and 7.1 MeV). That gamma radiation is relevant for the measurements in the area of the water steam separator.

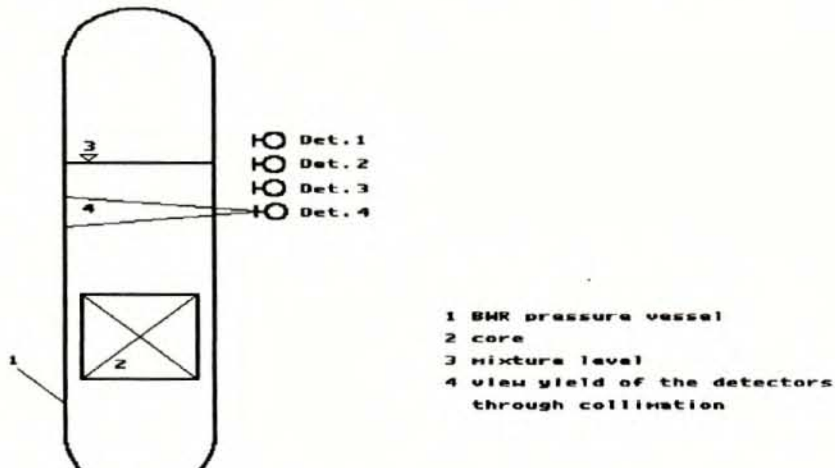


Figure 7: Boiling water reactor with a detector arrangement used for water level measurement

Because we find two distinguish sources it isn't possible to use the measuring method like on the steam generator, because it isn't possible to eliminate the gamma source properties with one comparision detector. Therefore we must use an other method.

3.2 Measuring sytem for a BWR

At first we can define two measured ranges one for reactor in operation time and one for the time after shut down in which isn't N-16 in the primary water.
The main task for the first measured range is the evidence though the mixture level within the limits or not and so it seems enough that it should be created a measuring system with the properties of a switch.

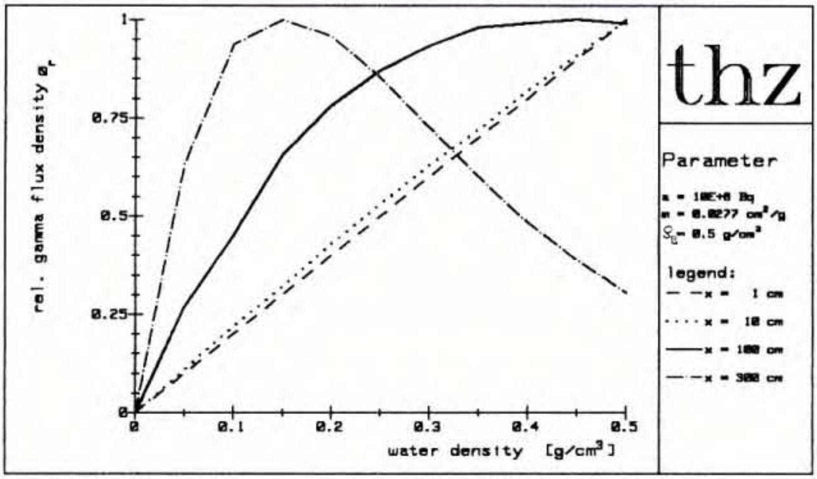


Figure 8: Relative gamma flux at the detector with the parameter distance x from a point source as a function by the water density wich is equal in the point source and on the way from source to detector.

If we consider the mixture level we see a distinction between the water density upper and under the mixture level. This distinction causes a change of the gamma source properties specific activity and self attenuation. If the water density in the source is increasing the specific activity is increasing also but the self absorption too. It is expected that this change causes a change of the detector signal of a detector upper and of a detector under the mixture level. But the direction of those changes is depended on the other parameters like distance from source to detector and the mass absorption coefficient.(see fig.8)
If we detect this differences we need several detectors one upon the other. These detectors have to perform some design principles.

1. The detectors have to show a wide measuring range and the same efficiency properties regarding gamma energy and radiation direction.

2. The direction perendicular in the vessel is favoured absolutly through efficient combination of collimation and inherent directional characteristic of the detectors . Every detector has his own separate source volume through the collimation.

3. All detectors must get the same geometrical conditions in their view yield. They have to be parallel each other. If there are constructive distinctions between the detectors correction factors have to been used.

4. The detectors are situated perendicular one upon the other.

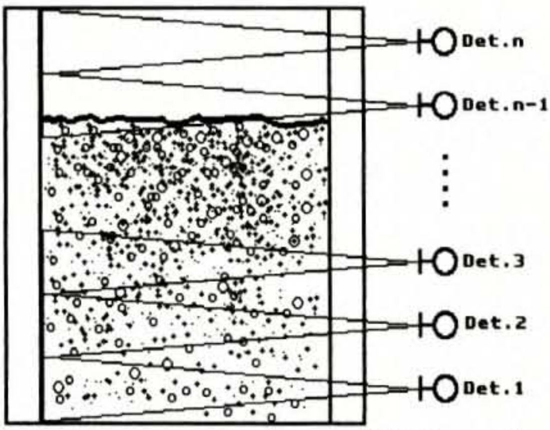


Figure 9: Detector arrangement on a BWR for water level indiction in the area of the water steam separator without internal construction

This detector arrangement is shown in fig.9. If we use ionization chambers then we get an analogeous current which is calculated for uncollided gamma radiation of the N-16 to

$$I_i = \int_{V_q} a \cdot K_i / r^2 \cdot \exp\left(-\mu_{qi} \cdot L_{qi} - \sum_j (\mu_{ji} \cdot L_{ji})\right) dV \qquad (15)$$

I : output current [A]
index : i - number of detector
j - number of material layer between source and detector
q - source

The detector signals I_i with $i = 1..n$ are the input for a measuring algorithm after flattening.
Firstly, it is looked for the maximum value of the detector signals. After that it is carried out a quotion forming .

$$S_i(h) = I_i/I_{max} \qquad \text{for } i = 1..n \qquad (16)$$

The quotients are independed of the specific activity in the primary water because the transit time from the first to the last detector is very short and the radioactive decay is hardly perceptible. Now we can approximate a function on the values S_i and we get

$$S(h) = f(S_i) \qquad (17)$$

It is expected that this function have in the height of the water level a distiortion. This distortion can be shown with forming the first derivation.

$$S'(h) = dS/dh \qquad (18)$$

If eq.(18) shows a jump then the water level is localized between this two detectors which are limiting the jump. The more detectors are used the better is the localization of the water level.
It is used the N-16 radiation and after shut down this activity is increasing quickly. That system is then not applicable but it is possible to switch to a second measuring system which is operating like the system on the steam generator (section 2.) using the core as gamma source only. (second measured range)

3.3 Calculations

In order to reconsideration of the above assertions some calculations was carried out. The basis method in this calculations is the Point Kernel Method for uncollided flux out side of a cylindrical source with cylindrical absorption materials. The geometrical proportions are similar the KWU-BWR. Because the uncertainty of the portion of water steam separator material for the homogenizing the calculation was carried out for some values for this portion. The axial void fraction distribution was approximated through one linear function for the water range and one other linear function for the steam range. The parameter of this two linear equations was variable. Radial changes of the water or steam density were not considered.
The result of this calculation is the uncollided gamma flux on the radial detector positions caused by N-16. This result was analysed and we got for example the curve like in fig.10 which is showing the jump in the first derivation of the S(h)-function very clear.

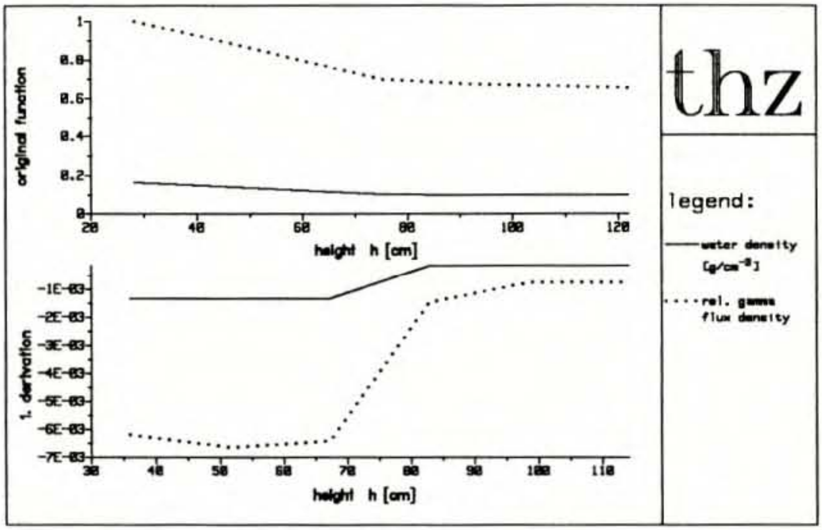


Figure 10: Water density and relative gamma flux density as function of the height in a reactor, the jump in the first derivation localized the water level

4. Conclusions

The water level measurement using the internal gamma activity is one way to get a divers measuring system to the difference pressure measurement. Through the elimination of gamma source properties through a comparision detector in section 2.2 or through the quotion forming in section 3.2 the chances for such a measuring system are very improved. Some experiments have been showed the aptitude of the comparision detector. The next works are going to carry out experiments on the zero power reactor for two phase state conditions and further on power reactors.

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SIEMENS

OECD (NEA) CSNI SPECIALIST MEETING ON

INSTRUMENTATION TO MANAGE SEVERE ACCIDENTS

Containment Atmosphere Measurements

Cologne, March 16 - 17, 1992

E 443 / B. A. Eckardt

Siemens AG, Power Generation Group (KWU)

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1 Introduction

In order to reduce the residual risk associated with hypothetical severe nuclear accidents, nuclear power plants in Germany have been backfitted with supplementary systems such as containment venting systems (Ref. 1). In conjunction with these measures the German Reactor Safety Commission (RSK) imposed the additional requirement that provisions be made for post-accident sampling of the containment atmosphere for the purpose of obtaining information on the condition of the core and on potential hazards to the environment. In addition it is planned to take measures to reduce the hydrogen concentration into account when considering severe accident scenarios with hypothetical core melt accidents. These measures comprise deliberate hydrogen ignition at low concentrations as well as the provision of a number of catalytic recombiners.

Measurement of the composition of the containment atmosphere is intended to allow assessment of the potential hazard for the area around the plant as well as providing additional information on the accident history, the plant condition and the effect of countermeasures. In addition to the measurement of atmosphere temperature and pressure, measurement of the gas composition e.g. hydrogen and, where applicable, CO_2/CO and airborne radionuclide concentrations are of particular interest. Depending on the accident sequence the postulated accident conditions can result in considerably higher concentration levels as compared to a design basis accident (DBA).

The following describes the functions of existing and newly-developed systems for measurement of:

- hydrogen concentrations
- airborne nuclide concentrations

as well as discussing the functions of these systems under severe accident situations.

2 Measurement of Hydrogen Concentration

Different systems are used at the various plants for measurement of hydrogen concentration. A basic distinction can be made between systems used for measurement inside the containment (in-situ monitoring) and extracting sampling systems where measurement is performed outside the containment.

2.1 Information Needed and Requirements

Special requirements have been stipulated for hydrogen monitoring systems for design basis accidents. The information required for monitoring severe accident scenarios depends largely on the existing measures already in force or planned for hydrogen control. The following therefore describes briefly the information required for BWR and PWR plants.

PWR

For PWR plants a hydrogen measurement range of 4 vol% has been stipulated for DBAs with a wide measurement range of 10 vol%. Severe accident scenarios which can lead to the release of larger amounts of hydrogen in the containment have been investigated for the PWR within the scope of the German Risk Study (Deutsche Risikostudie), Phase B through consideration of hypothetical core melt accidents. The action and type of equipment provided for hydrogen control have a decisive effect on the requirements regarding the measuring range and will therefore be described briefly in the following. The countermeasures recommended for German PWRs comprise the provision of a dual hydrogen reduction system.

These measures allow the buildup of hydrogen to be halted at low concentrations well below the detonation limit through deliberate controlled ignition and combustion, without posing any risk to the containment.

For this purpose Siemens has developed a catalytic igniter and a spark igniter as well as a catalytic recombiner, whereby the latter is suitable as a supplementary measure to reduce the hydrogen concentration (see Ref. 2).

For a typical German PWR the early ignition of gas mixtures close to the ignitability limit is a safety measure directed at limiting the hydrogen concentration in the containment and preserving the containment integrity. This capability has been illustrated in extensive representative test series.

In order to check the effectiveness of such measures for hydrogen control it may be necessary to measure increased hydrogen concentrations including measurements under various atmospheric conditions in the corresponding operating and equipment compartments. Data on oxygen concentrations could also provide information on the accident situation. Further special loads on the hydrogen instrumentation are the high radiation levels as well as temperature and pressure loads and atmospheric impurities.

BWR

For hydrogen control in BWRs of the 69 Product Line the containment is always inerted so that hydrogen and oxygen concentrations have to be measured in inert atmospheres. Requirements regarding the measurement of hydrogen and oxygen concentrations for DBAs stipulate a range of ≤ 4 vol% for hydrogen and ≤ 21 vol% for oxygen. Under severe accident conditions with postulated zirconium/water reactions much higher hydrogen concentrations are possible. The oxygen concentrations can for the most part be expected to remain in the range to be measured for DBAs.

2.2 Measurement of Hydrogen Concentrations in Atmosphere

The maximum hydrogen concentrations to be measured as well as the number of measurements depend on the reactor type and power plant. For design basis accidents, and on the basis of expected maximum hydrogen release rates, in-situ systems as well as extractive systems situated outside the containment have been developed and installed in various plants. These systems are also in all cases capable of supplying valuable information in the event of severe accidents.

Qualification of this instrumentation has been performed for design basis accidents in extensive tests taking into account requirements regarding temperature, pressure and dose rates, etc.

2.3 In-Situ Hydrogen Measurement

The Convoy plants and most of the other German PWR plants have been provided with hydrogen sensors developed by Siemens/KWU for installation in the containment.

In this way it is possible to monitor the area and time distribution of hydrogen concentrations after a loss-of-coolant accident continuously, simultaneously and without using a sampling system, i.e. without radiation exposure of the operating personnel. The measured values are displayed in the control room.

In PWR plants (Fig. 1), for instance, the hydrogen concentration is monitored in the lower and middle sections of the steam generator compartments and in the dome region of the containment. The sensors are connected to the signal processing units by electric cables. These telemetric cables pass through a number of separate cable penetrations out of the containment to the switchgear building. Here the signals from the sensors are processed so that the actual concentrations can be displayed and logged on a multichannel recorder in the control room.

This system has the following special features:

- Overview of the situation regarding hydrogen concentration and distribution in the different containment areas, especially during early accident phases, with simultaneous supply of data
- No opening or closing of containment isolation valves necessary
- No handling of radioactive gases outside the containment

Hardware Design and Operation

Figure 2 shows the various components of the instrumentation system. The LOCA-proof hydrogen sensor operates on the basis of catalytic oxidation of hydrogen on a heated filament. The atmosphere to be monitored diffuses into a measuring cell. Any hydrogen present is catalytically oxidized with ambient oxygen on a platinum element. The resulting temperature increase causes an increase in filament resistance producing a signal corresponding to the hydrogen concentration.

This measurement technique was qualified in the course of an extensive qualification program for accident conditions.

This measurement can also be used during severe accident scenarios. With sufficient stoichiometric excess oxygen, measurement of hydrogen concentration is possible up to 10 vol% allowing the effectiveness of the dual hydrogen control concept to be monitored. This system has not been employed to measure higher concentrations. This could however be performed through additional measures for example using the existing sampling lines. Such gas sampling with subsequent analysis in the laboratory would, however, have to take into account the contamination and shielding aspects of sampling as well as the requisite corrections for steam partial pressure.

After installation of sampling systems as illustrated in 3 below, the hydrogen and oxygen concentrations can additionally be measured in these systems.

2.4 Hydrogen Measurement through Extractive Sampling

All BWRs and a few PWRs are equipped with sampling systems having analyzers outside the containment for hydrogen (PWR) or hydrogen and oxygen (BWR). The function of such a system is described below using a BWR by way of example (see Fig. 3).

Functions

The function of the hydrogen/oxygen monitoring system is to monitor the volumetric concentrations of hydrogen and oxygen in the containment atmosphere at representative locations in the containment. The oxygen monitoring system is provided to perform the following:

- Monitor the distribution of inert gas (N_2) and the drop in oxygen content during inerting of the containment
- Detect rises in oxygen concentration following accidents
- Supply the information necessary for occupational safety and health when the containment is to be entered by personnel

The hydrogen monitoring system is provided to perform the following:

- Detect releases of hydrogen into the containment atmosphere during normal operation and LOCAs
- Monitor the effectiveness of hydrogen/oxygen recombining equipment

2.5 Operation/Description

The hydrogen and oxygen concentrations are monitored at a total of five locations inside the containment:

- 2 measuring points in the drywell
- 2 measuring points in the pressure suppression chamber
- 1 measuring point in the control rod drive compartment

The gas samples are withdrawn by a sampling gas compressor via sampling lines equipped with isolation valves. The gas sample is then cooled to a constant dewpoint temperature in a gas cooler/ dryer. The sample gas flow is then divided by pressure control equipment into two flows, the main flow being returned directly to the containment and a small part flow passing through the continuously-operating hydrogen and oxygen analyzers. All components involved in sample gas transport, sample gas conditioning and data acquisition are housed together in a sampling equipment cabinet.

The paramagnetic oxygen analyzer functions on the principle of a dumbbell suspended in a magnetic field.

The operating principle of the hydrogen analyzer uses the difference between the thermal conductivities of the gas sample and a reference gas. The design and configuration of the hydrogen/ oxygen monitoring system allow samples to be taken continuously from one gas sampling point at any time (drywell, pressure suppression chamber or control rod drive compartment). Changeover to a different gas sampling point is effected from the control room where data acquisition also takes place. Sampling, conditioning and evaluation of the gas are all automated.

This system is also capable of supplying information on hydrogen and oxygen concentration distributions in the containment during severe accidents.

The process used for measuring hydrogen allows concentrations far above 4 vol% to be measured. The measuring principles implemented in BWRs for oxygen measurement are capable of measuring all concentrations which may possibly occur under severe accident conditions.

In addition, correction of the measured values must in all cases be performed through determination of the steam content at the sample extraction point, e.g. through temperature and pressure measurement and steam condensation in the system must be taken into account.

In order to avoid unfavorably high radiation levels in the areas where measuring equipment is set up and in sampling lines, it is recommended to operate the system intermittently, particularly where high contamination of the atmosphere prevails.

3 Monitoring Containment Atmosphere Activity after Severe Accidents

In line with the recommendations of the Reactor Safety Commission (RSK) samples of the containment atmosphere should also be taken after a severe accident (e.g. with core meltdown) to provide information on the condition of the core and to indicate the potential hazard to the environment.

The potential hazard following releases to the environment largely depends on the aerosols and iodine present in the containment atmosphere. For this reason various systems have been investigated for their capability to detect these substances. Investigations showed that the systems already installed in power plants, as illustrated in Figure 4, exhibit considerable pipe factor problems under certain severe accident scenarios. Deposition rates determined on the basis of various experiments (Refs. 2, 3 and 4) when extrapolated for iodine in sampling systems with sampling line lengths of 30 to 50 m resulted in pipe factors of up to >> 10. Similar deposition problems with aerosols occur during particular accident situations where considerable aerosol fractions occur in the sampling lines. This results in requirements on the piping system which in some cases are completely contradictory, for example:

- high gas velocities in the sample extraction lines and large pipe diameter to reduce the iodine pipe factor

- low velocities and few bends when transporting large aerosols

- Avoidance of significant iodine pipe factor fluctuations in the event of, for example, organic atmospheric impurities in the sampling line caused by the accident.

As these problems cannot be solved satisfactorily with extraction pipe systems new system concepts have been worked out and evaluated with regard to measurement of the aforementioned two groups of substances as well as with regard to providing additional measurement of $H_2/CO/CO_2$ in the atmosphere.

A further consideration for a station for manual extraction of a sample for laboratory analysis is that it would have to be well-shielded from the containment and would have to be arranged such as to remain accessible following a severe accident.

On the other hand the sample extraction point in the containment should be representative of the atmosphere in the entire containment and in a PWR, for example, should allow extraction from the inner containment area. This results in the requirement that the system to be selected must still allow detection of the substances mentioned even with pipe lengths for example of 30 - 50 m. A possible arrangement for a PWR is shown in Figure 5. For BWRs transport lines tend to be longer with more pipe bends.

3.1 Function

The function of the sampling system is to extract a representative sample from the containment atmosphere which would provide information helpful in assessing the accident sequence, the plant condition and the potential hazard to the environment.

The following nuclides have to be detected:

- concentration of aerosol-bound radionuclides
- concentration of gaseous iodine and iodine compounds
- concentration of other substances present in gaseous form (noble gases)

Further important requirements:

- cross-contamination of one sample to another through "memory effects" should be avoided as far as possible
- measurement of other components of the atmosphere such as H_2, CO, CO_2 should also be possible

Further conditions to be taken into account are as follows:

Containment pressure/temperature		1 - 7 bar/50 - 160 bar
Conditions		saturated and dry atmospheres
Aerosols	dae	0.1 - 10 µm
Fog	dae	≤ 100 µm
Iodine		

Maximum concentration in containment during sampling		*Sample activity after dilution*
Noble gases	$2.7^{*}\ 10^{15}$ Bq/m^3	
Aerosols	$5^{*}\ 10^{15}$ Bq/m^3	< 10^9 Bq
Elemental iodine	$4.5^{*}\ 10^{15}$ Bq/m^3	
Organic iodine	" "	

3.2 Measuring Problems with Various System Concepts

Different sampling system concepts have been assessed on the basis of existing results from tests performed on different piping systems.

The results of these investigations showed that to cover such severe accident conditions, sampling systems as shown in Figure 5 would be required with sample transport through lines 30 - 50 m long and having several bends and this would lead to considerable falsification of measured values as a result of:

- significant deposition of aerosols, particularly for larger aerosol fractions
- significant deposition of elemental iodine, particularly where pipe wall contamination in the form of organic impurities, etc. exists as a result of the accident
- significant memory effects later in the accident sequence with reduced atmospheric contamination

Preference was therefore given to systems with in-situ sampling directly in the containment which would completely avoid the above problems.

In this connection reference was made to experience in aerosol sampling technology gained during the containment aerosol experiments DEMON and VANAM at Battelle in Frankfurt am Main as well as the ACE tests at Battelle Northwest in Richland, USA. In these experiments the sample filters were located and loaded directly in the containment atmosphere to avoid deposition in sampling lines. This allowed representative samples to be taken under considerably varying accident conditions with large fluctuations in aerosol particle size distribution, density of the atmosphere, moisture content and temperature, etc.

Based on calculations and on the knowledge gained during such experiments a pneumatic conveyor system with direct sampling in the containment atmosphere was considered to represent a favorable solution. The disadvantage was the amount of equipment required as well as active components in the containment.

For this reason further direct monitoring systems using scrubbers were designed and investigated. This solution was found to be preferable and is briefly described in the following (see also Fig. 6).

3.3 Pool Sampling System

In order to avoid significant sampling errors which can occur particularly during accidents where sampling conditions vary considerably, samples of substances likely to be deposited in sampling lines are taken directly in the containment.

The sampling unit which operates with liquid collects the most of the aerosols and elemental iodine. The nobles gas and organic iodide which do not form deposits are routed to dilution equipment outside the containment. This dilution equipment is used to extract a gas sample and dilute it to an activity concentration suitable for laboratory purposes. Following gas sample extraction the activities deposited in the inlet area are also recorded and transported to the external dilution system together with the aerosols and iodine present in the scrubber fluid and are also diluted to activities suitable for laboratory purposes.

After sampling, the equipment is decontaminated in order to eliminate as far as possible any memory effects and to flush the openings. This system has been recommended to the Reactor Safety Commission (RSK) for monitoring severe accident scenarios in addition to the equipment provided for design basis accidents.

3.4 Verification

Thermohydraulic function, separation and deposition properties were investigated for this equipment during tests which used realistic piping lengths of > 30 m. In order to determine transport losses in sampling equipment and transport lines tests were performed with solutions and suspensions. The results of some of these tests are shown in Figure 7 so that sampling losses can be minimized and pipe factors of < 1.5 can be obtained.

The results of a decontamination test on this equipment are shown in Figure 7. This shows that memory effects caused by sampling line contamination, particularly during measurements taken at a later stage in the accident sequence with considerably reduced activity concentrations can for the most part be avoided.

3.5 System Features

The following is a brief summary of system features:

- For most part representative sampling for enveloping aerosol spectrum of e.g. 0.1 - 30 µm and elemental iodine through in-situ sampling with low suction velocity and large suction inlet.

- For most part avoidance of pipe factor problems through
  - recording of deposits at sampling equipment inlet area
  - transport in liquid of substances likely to deposit
  - Regular flushing of suction inlet

- Insignificant memory effects through
  - "null" sample before the start of measurement and
  - system flushing (decontamination) after measurement/fouling

- Operation of equipment from well-shielded area possible e.g. via sampling line of 40 m or more

- Use of only passive components in containment, manufactured from temperature-resistant and radiation-resistant materials

- On-line $H_2/CO/CO_2$ measurement in sampling gas atmosphere possible.

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A.L. Wright
B.R. Fish
E.C. Beahm
C.F. Weber
Oak Ridge National Laboratory, Oak Ridge,
Tennessee 37831
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Philip J. Unrein, Charles A. Pelletier,
James E. Cline and Paul G. Voillequé
Science Applications, Inc.
Rockville, Maryland 20850
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M.J. Kabat
Ontario Hydro Health and Safety Division
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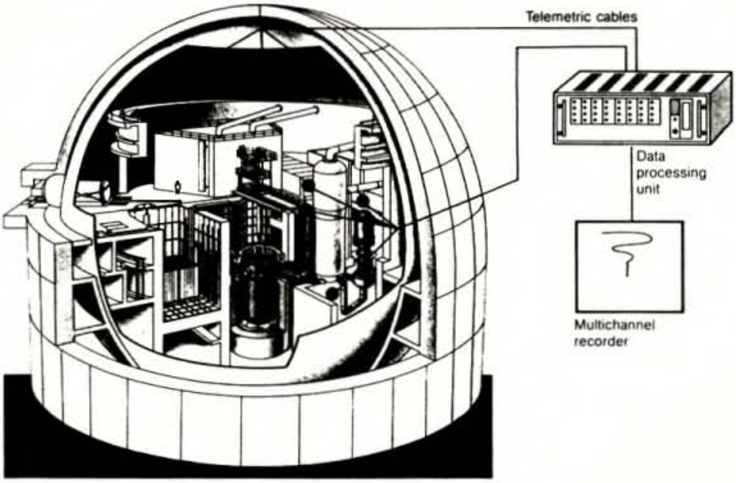
SIEMENS



Technincal data

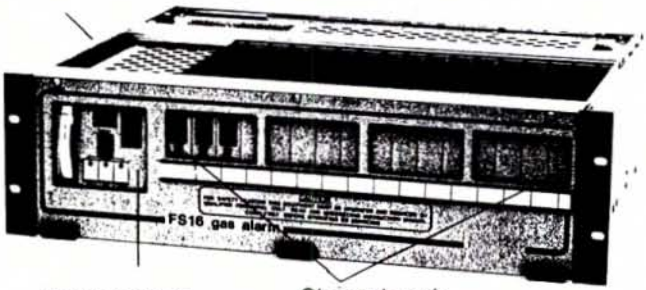
Hydrogen sensor	
Type	WS 85
Measuring principle	catalytic oxidation, thermal effect
Measuring range	0 to 10 vol.-% hydrogen
Design pressure	6 bar
Design temperature	160 °C
Radiation resistance	proven up to 250 kGy

Schematic Diagram of Hydrogen Monitoring System in a Siemens/KWU Pressurized Water Reactor Plant FIG. 1

S/KWU
Et/02.92
09-32-g

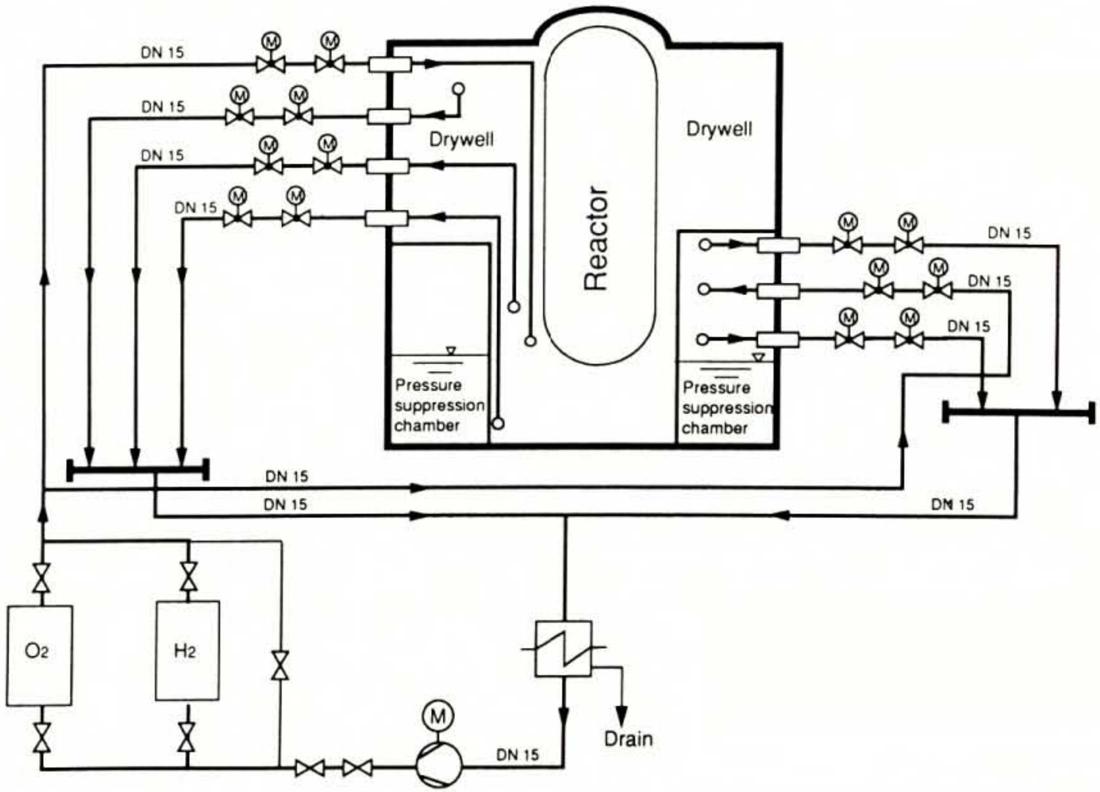
H_2 sensor

Control unit



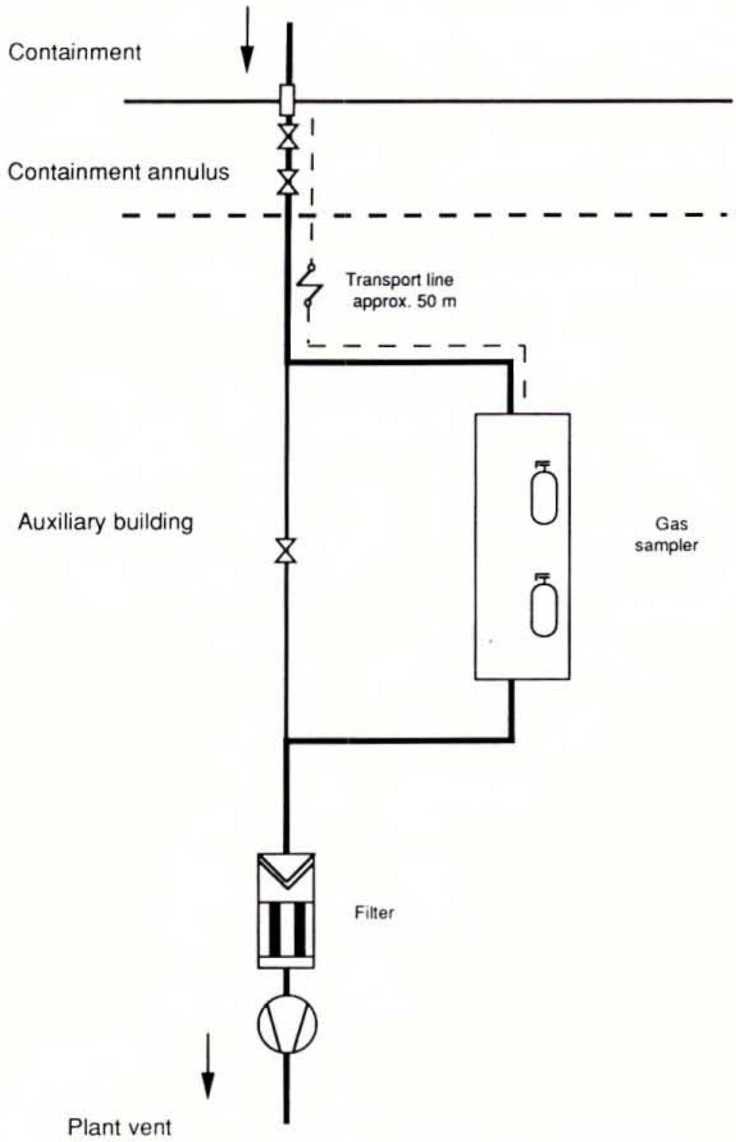
Components of Hydrogen Monitoring System
Fig. 2

SI KWU
E443
Fl/ 2.1992
13-001-D



H_2 / O_2 Monitoring System for BWR
FIG. 3

S/KWU
E 443
51-016-A



DBA Atmosphere Sampling System

FIG. 4

S/KWU
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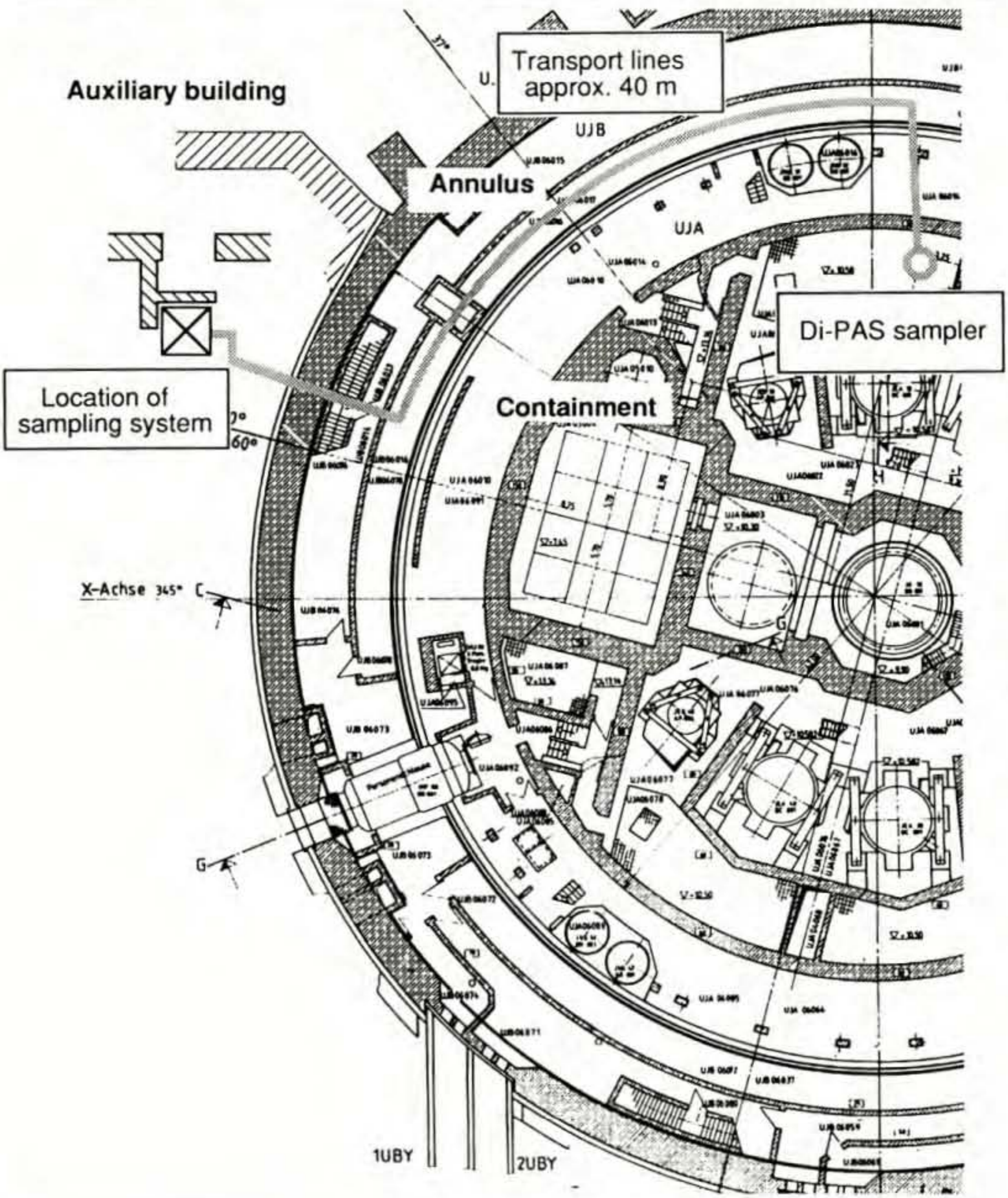
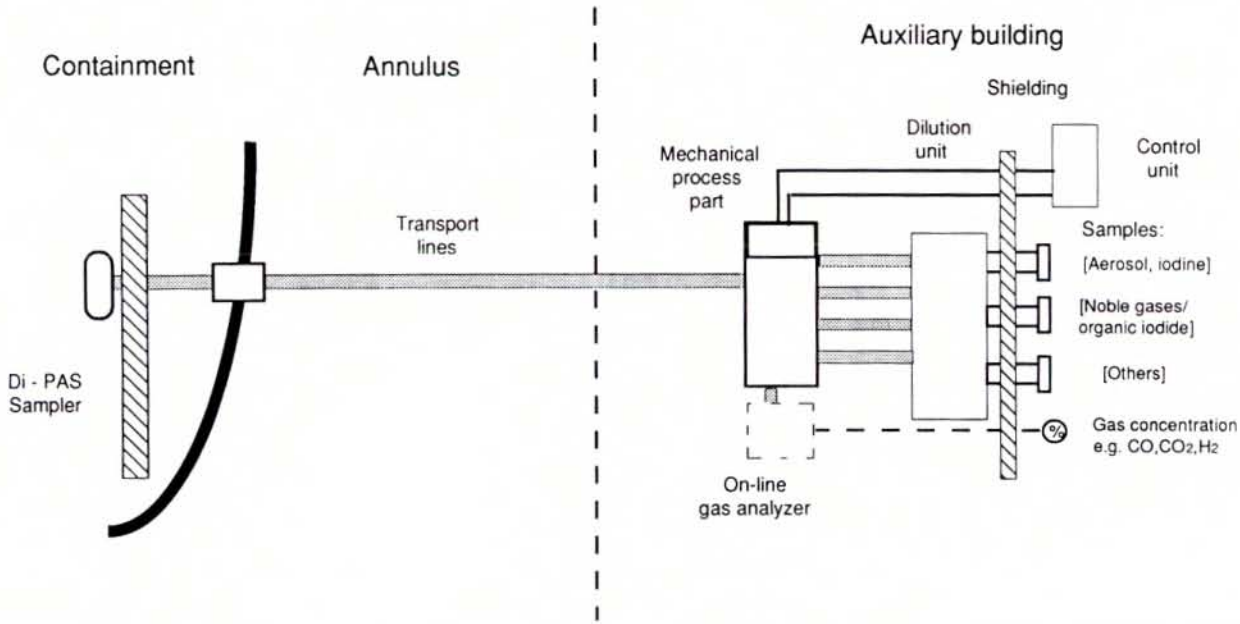
SIEMENS

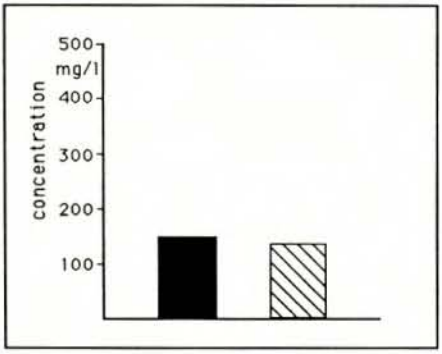


UB KWU
E 443
02.12.91
50-001-J

Arrangement of Di-PAS system for PWR
FIG. 5

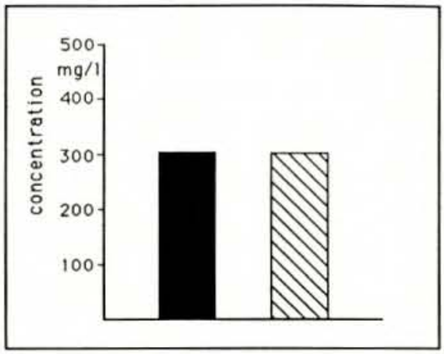


Schematic of Di-PAS SYSTEM
FIG. 6



Test I potassium iodide

Loss factor R = 1.03



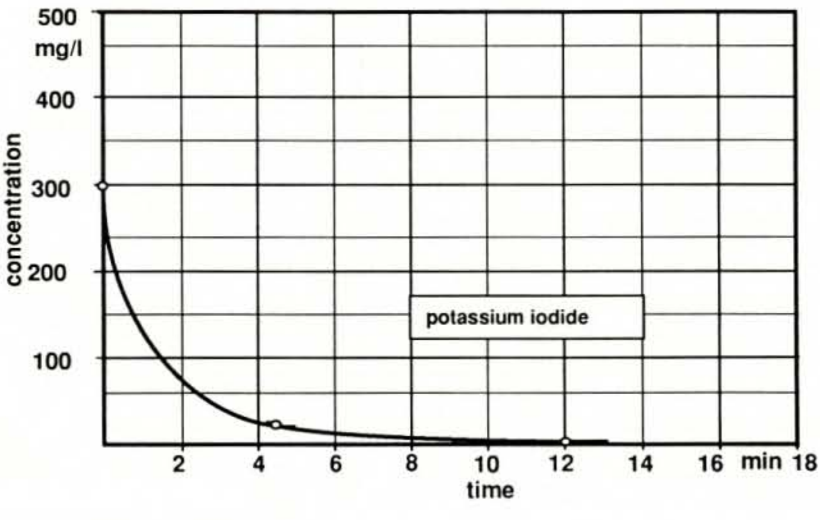
Test II potassium iodide

Loss factor R = 1.0

■ Measurement in pool

▨ Measurement in system after transport through line of approx. 40 m

<u>Test of system transport losses</u>



<u>System decontamination test</u>

Tests of System Transport losses and Decontamination
FIG. 7

S/KWU
Et/02.92
09-30-g

OECD NUCLEAR ENERGY AGENCY

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

SPECIALIST MEETING ON

INSTRUMENTATION TO MANAGE SEVERE ACCIDENTS

Cologne, Germany
16th-17th March 1992

Model-based correction algorithms improving the accuracy of hydrostatic level measurement on pressure vessels

by R. Hampel, W. Kästner, B. Vandreier and F. Worlitz, TH Zittau

1. Introduction

It is important to possess precise process informations for an optimised valuation of the plant process conditions. Especially these informations have a great priority as well as for the emergency operation and post accident management. The rapid and great transitions resulting from that are hardly to master by the used measuring devices. Spurious indications can occur the cause of which could be a modification of design conditions, specifical transients of process and the demage of the measuring instrument itself respectively during accidents.

Further more it would be desireable to get additional not measurable state variables in this situation.

For solving those problems modern methods and procedures of process identification, parameter identification and plausibility analysis comprising correction algorithms become more and more important.

These modern methods are used to solve the following problems

- diagnosis of the process state on the basis of combination by measuring variables, analytical redundancy and linguistic declarations,
- reconstruction of not directly measurable variables and parameters respectively
- detection and identification of process faults and instrumentation faults (diagnosis)
- reconfiguration of measuring signals (correction)

The reconstruction of process state is thus a combination of measured quantity, reconstructed state variables and analytical redundancy using model-based measuring methods.

The use of model based measuring methods has been investigated on the example of hydrostatic level measurement on horizontal steam generators.
The results of experiments on pilot plants as well as comparison with calculations of empowered programs for instants ATHLET and methods of parameter identification serve as a verification of methods and algorithms, which were developed.

The following describe the main facts of this work.

2. Application of modern methods of signal processing

2.1 Overview

Since the 60-ies the elaboration and application of so called "Modern Methods of Control Engineering" has taken place. It has been done to solve the tasks of process control and process monitoring better.

The development was accelerated at first by the necessity of effective and safety control of operational techniques and energetic processes and secondly by the availability of digital techniques in form of digital controllers, support systems and process computers.

These modern methods of control engineering are different from the conventional methods by:

1. More information about the controlled process
2. Extension of application fields
3. New algorithms of design.

It should be accentuated that these methods are based on a methodical conception, which is a general one, but they are applied the actual specific of the process.

In /1/ different modern methods of control engineering are represented and classified. The following table shows these methods and their target positions.

The increasing importance of the different methods for the control of normal operation as well as for the extraction of informations and the influence during and after accidents are explained in numerous publications.
An important method to realise the above mentioned target position is the method of the reconstruction (estimation) of the state of process by model-based measurement methods /2/.

method	target position
evaluation of characteristics	evaluation of not directly measurable characteristics with the help of known characteristics by algebraical and logical connection
Signal Filter	extraction of useful signals from disturbing measurements
state estimation model-based measurement methods . Luenberger Observer . Kalman-Filter	reconstruction of internal state variables and / or input variables
recurrent estimation of parameters	identification of parametric models of process
State Controller	inprovement of regulating quality by additional feed forwarding of state variables (Application of model - based measurement methods)
Inferential Control Internal Mode Control	control of processes with not directly measurable control quantities, which are influenced by slow variable disturbances or occasional step disturbances
adaptive control	- adaptation of control parameters - continuous adaptation of control parameters of time variable processes
Diagnosis Interpretation Monitoring	investigation of deviation from normal operation

Table 1: Modern methods of control engineering

2.2 Reconstruction (estimation) of state of the process

Dynamic processes are indentified by the registration of time variable and measurable values. The point is that it is mostly the result of influence and interaction of inherent system parameters, which are often not or only incompletly measurable.

But the information about the dynamic behaviour of these state variables gives the control engineer a lot of possibilities to take influence of the behaviour of the process in a wished way.

Thus the control engineer takes efforts to reconstruct (deterministic consideration) or estimate (stochastic consideration) the complete system state.
Methods, which apply a-priori-informations about the process besides the easily measurable values are used for this purpose. If these informations are available as a mathematical model the method is called model-based measurement method.

The mathematical model has the form:

$$\frac{d\mathbf{q}(t)}{dt} = \mathbf{A} * \mathbf{q}(t) + \mathbf{B} * \mathbf{u}(t)$$

$$\mathbf{x}(t) = \mathbf{C} * \mathbf{q}(t)$$

$\mathbf{q}(t)$ - vector of state variables
$\mathbf{x}(t)$ - vector of output variables
$\mathbf{u}(t)$ - vector of input variables.

2.3 Model-based measurement methods

The theoretic foundation of model-based measurement methods was done by Luenberger (Luenberger observer) and Kalman (Kalman-Filter).
A mathematical model of the process and the measuring device is arranged in parallel to the process. The input variables and the boundary conditions are known and are supplied to the model. The result of the mathematical redundancy are the estimated system state and the estimated measurement.
The difference between the calculated and measured values is a dimension for the deviation between model-based reconstruction and real process state.
This difference, which is emplified, is fed back to the process model. The reconstructed and real variables converge by an appropriate dimensioning of the amplification gain /2,3,4/.

The reasons of deviations can be:

- failure in model (incorrect simulation of the process),
- inaccuratly known starting conditions ,
- perturbing effects.

The supposition of application of the model-based methods are:

- knowledge of the process model
  (process model of normal operation and process models of hypothetical accidents with a high probability of occurrence),
- accurate registration of the input variables,
- process must be observable.

The importance of these methods for control of accidents is clear. Starting with a mathematical model of the undisturbed process disturbances produce deviations between the variables of the model and the measurable variables of the real process. By the application of suitable algorithms the estimation failure gives an information about the reason of the disturbance.

On the other hand the use of accident models produce a mathematical redundancy, which gives additional informations about the process state.

2.3.1 Reconstruction of state by the Luenberger observer

The structure of the model-based method with the help of the Luenberger observer is shown in figure 1.

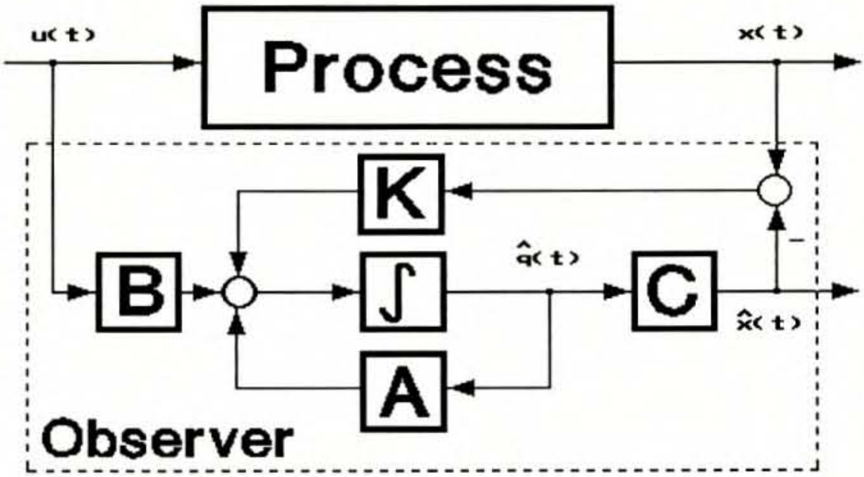


Figure 1: Structure of process and observer

The observer is described by the following equations:

$$\hat{\mathbf{q}}(k+1) = \mathbf{A} * \hat{\mathbf{q}}(k) + \mathbf{B} * \mathbf{u}(k) + \mathbf{K} * [\mathbf{x}(k) - \hat{\mathbf{x}}(k)]$$

$$\hat{\mathbf{x}}(k) = \mathbf{C} * \hat{\mathbf{q}}(k)$$

with

$\mathbf{x}(k)$ - vector of output variables of the process
$\hat{\mathbf{q}}(k)$ - vector of reconstructed state variables
$\hat{\mathbf{x}}(k)$ - vector of reconstructed output variables
$\mathbf{K}$ - gain.

2.3.2 Estimation of state by the Kalman-Filter

According to the task to observe stochastically disturbed processes, the algorithm of the Filter obtains characteristics with stochastical properties.
The algorithm of Filter shows that the gain was calculated to each sampling cycle depending on the stochastical disturbance.

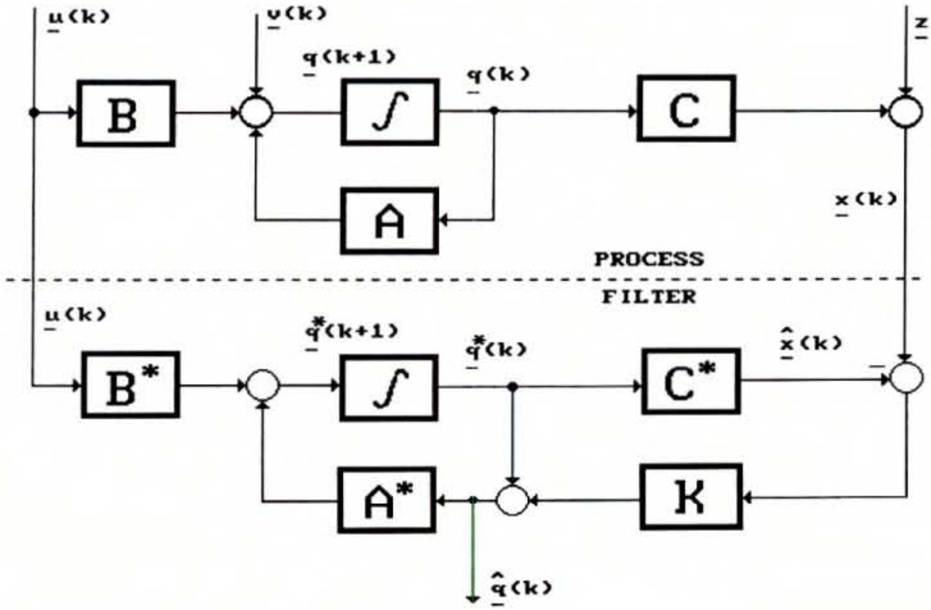


Figure 2: Structure of the Kalman Filter

The algorithm is subdivided into two steps:
- measurement update
- time update.

Measurement Update

$$\hat{\mathbf{q}}(k+1) = \mathbf{q}^*(k+1) + \mathbf{K}(k+1) * [\mathbf{x}(k+1) - \mathbf{C}^* * \mathbf{q}^*(k+1)]$$

Time Update

$$\mathbf{q}^*(k+1) = \mathbf{A}^* * \hat{\mathbf{q}}(k) + \mathbf{B}^* * \mathbf{u}(k)$$

Important values are:

- the Kalman Gain

$$\mathbf{K}(k+1) = \mathbf{Q}(k+1) * \mathbf{C}^{*\mathbf{T}} * [\hat{\mathbf{Z}}(k+1) + \mathbf{C}^{*} * \mathbf{Q}(k+1) * \mathbf{C}^{*\mathbf{T}}]^{-1}$$

- the covariance of the extrapolation error

$$\mathbf{Q}(k+1) = \mathbf{A}^{*} * \tilde{\mathbf{P}}(k) * \mathbf{A}^{*\mathbf{T}} + \hat{\mathbf{V}}(k)$$

- the covariance of the state error

$$\tilde{\mathbf{P}}(k) = \mathbf{Q}(k) - \mathbf{K}(k) * \mathbf{C}^{*} * \mathbf{Q}(k).$$

Figure 3 represents a graphic description of the algorithm.

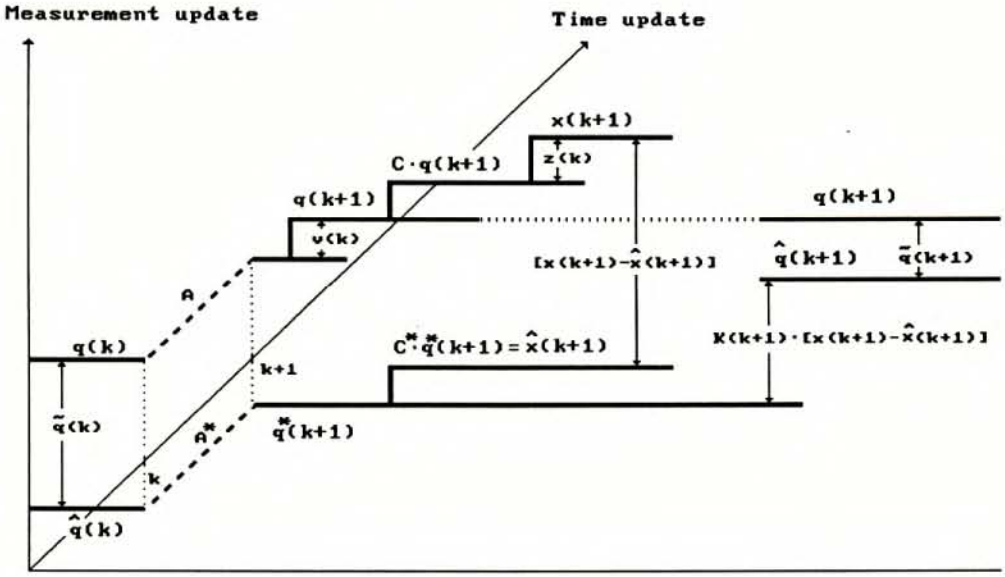


Figure 3: Graphic description of the filter algorithm

x - vector of output variables of process
$\mathbf{q}$ - vector of state variables of process
$\mathbf{q}^{*}$ - vector of extrapolated state variables
$\hat{\mathbf{q}}$ - vector of estimated state variables
$\tilde{\mathbf{q}}$ - state error
v - disturbance of process
z - disturbance of measuring system

$\tilde{\mathbf{q}}(k) = \mathbf{q}(k) - \hat{\mathbf{q}}(k)$ - state error

$\mathbf{q}^*(k+1) = \mathbf{A}^* * \hat{\mathbf{q}}(k)$ - extrapolated state

$\hat{\mathbf{x}}(k+1) = \mathbf{C}^* * \mathbf{q}^*(k+1)$ - estimated output variable

$[\mathbf{x}(k+1) - \hat{\mathbf{x}}(k+1)]$ - deviation between measured and estimated value of the output variable.

Based on the actual state $\mathbf{q}(k)$ the real process reaches the state $\mathbf{q}(k+1)$ dependent on the transient behaviour, which is characterized by the matrix $\mathbf{A}$, and under the perturbing effect $\mathbf{v}(k)$.

The measuring system, characterized by the matrix $\mathbf{C}$, represents the process state in form of the output variable $\mathbf{x}(k+1)$, which is falsified by the disturbance $\mathbf{z}(k)$.

Based on the a-priori-information $\hat{\mathbf{q}}(k)$ the filter reach the extrapolated state $\mathbf{q}^*(k+1)$ depending on the transient behaviour, characterized by the matrix $\mathbf{A}^*$. It produces the estimated output variable $\hat{\mathbf{x}}(k+1)$. The filter estimates the new state $\hat{\mathbf{q}}(k+1)$ by feeding back the error of estimation $[\mathbf{x}(k+1) - \hat{\mathbf{x}}(k+1)]$ and its amplification with the Kalman gain $\mathbf{K}$.
The difference between the new estimated state and the real process is the state error $\tilde{\mathbf{q}}(k+1)$.

2.4 Application of model-based measurement methods

The model-based measurement methods are used in the following general fields (figure 4):

1. Diagnosis of the complete system state (include not directly measurable variables)
2. Realization of state controller
3. Fault detection.

The concrete application takes place to investigate the complete system state of a pressure vessel in consquence of negative pressure gradients, which are a result of disturbances.

The task is the reconstruction of important state variables like steam quality and mixture level, to use this information for the improvement of control.

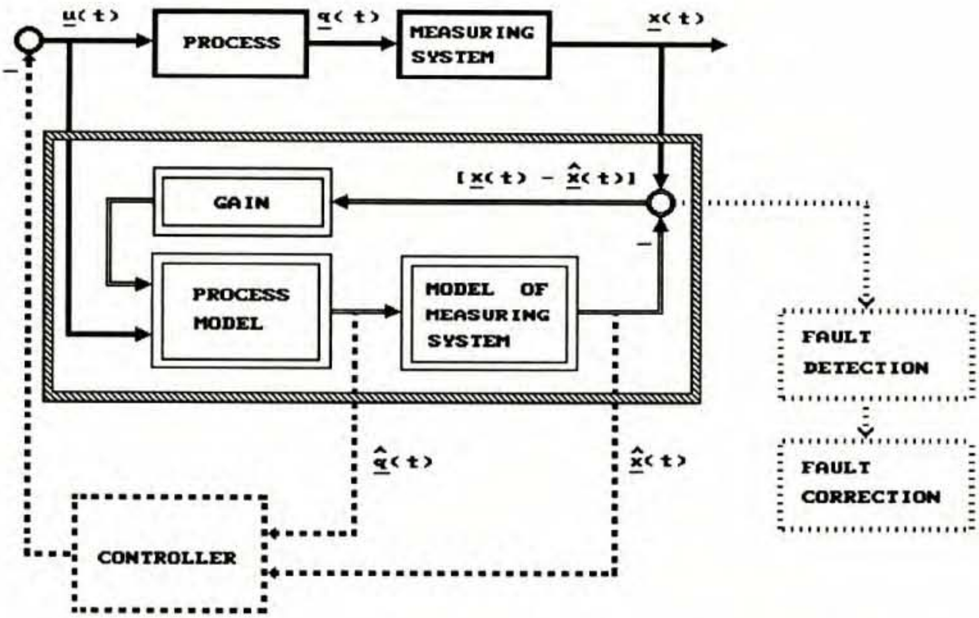


Figure 4: Model- based measurement methods and their applications

3. Simulation with ATHLET - Code

The Athlet- Code will be used for the test of model- based measurement methods. The thermohydraulic code was developed by the GRS. The four basic moduls are:
- Thermofluiddynamics
- Heat Transfer and Heat Conduction
- Neutron Kinetics
- General Control Simulation (GCSM).

By simulating of pressure vessels it is possible to obtain all measurable and not measurable values. A verification of models for pressure vessels will be made by means of these data, which are the basis of model- based measurement methods. These can be very useful for accident management and are the basis for a better process control.
Blow down experiments on the pressurizer facility (figure 12) of the TH Zittau, which will be still described, were post calculated by the ATHLET- code. Calculations were made with a reduced data set of a WWER 440 - 230 too. The aim of these calculations were the development, verification and investigation of the behaviours of a modul for a level measurement system in the general applied form of a two chamber comparison vessels. While the knowledge of the level of steam generators plays an important role during disturbances. It will also be used for the generation of

comparison datas for other simulation programs like models of pressure vessels.
A data set for the pressurizer facility was made, containing the geometrical data, the partition in seperate objects and control volumes and also instructions for the simulation. Figure 5 shows the nodalization scheme.
Blow down experiments were post calculated, because most disturbances cause negative pressure gradients. The pressure vessel models can be verified by these generated data. During these experiments steam blows to a blende into a blow down vessel. The control was made by a magnetic valve.
The electic heaters were turned off. The time of the blow down for the here selected experiment was nearly 70 seconds. The pressure gradient was about -0.018 MPa / s. Primary disturbances (Failure of 2 main coolant pumps) will cause lower and secondary disturbances (Break of the main steam line) will cause higher pressure gradients in most case.
The calculated and measured pressure (figure 6) were nearly identical. After the blow down the calculated pressure is a little to high (0.04 MPa).
The calculated collaps level between the connections of the measurement systems is similar to the level, measured by an one chamber vessel (figure 7).

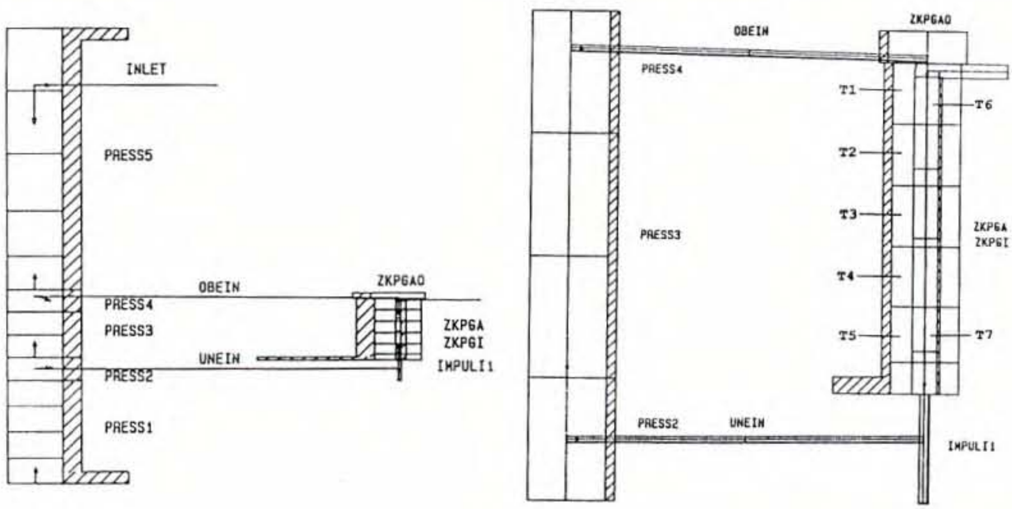


Figure 5: Nodalization scheme of the pressurizer facility for the ATHLET- Code

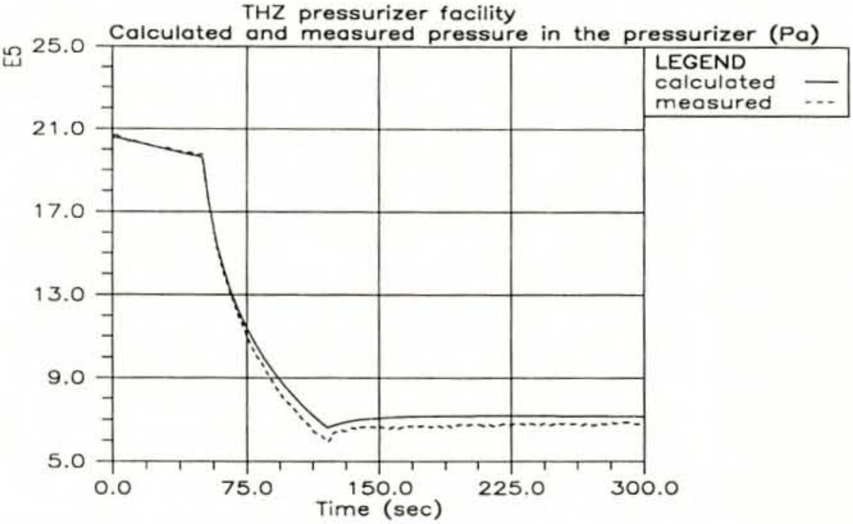


Figure 6: Calculated and measured pressure

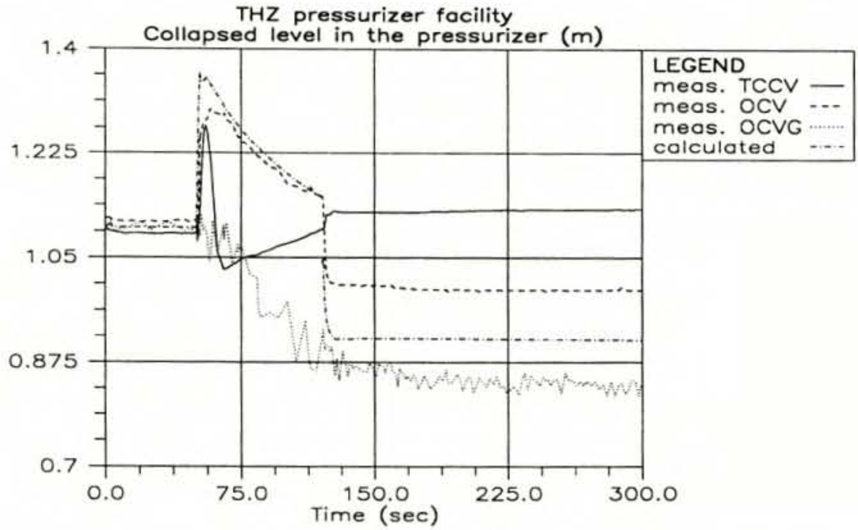


Figure 7: Collapsed level in the pressurizer
- meas. TCCV : measured level by two chamber comparision vessel
- meas. OCV : measured level by one chamber vessel
- meas. OCVG : measured level by one chamber vessel above the complete heigth of pressurizer

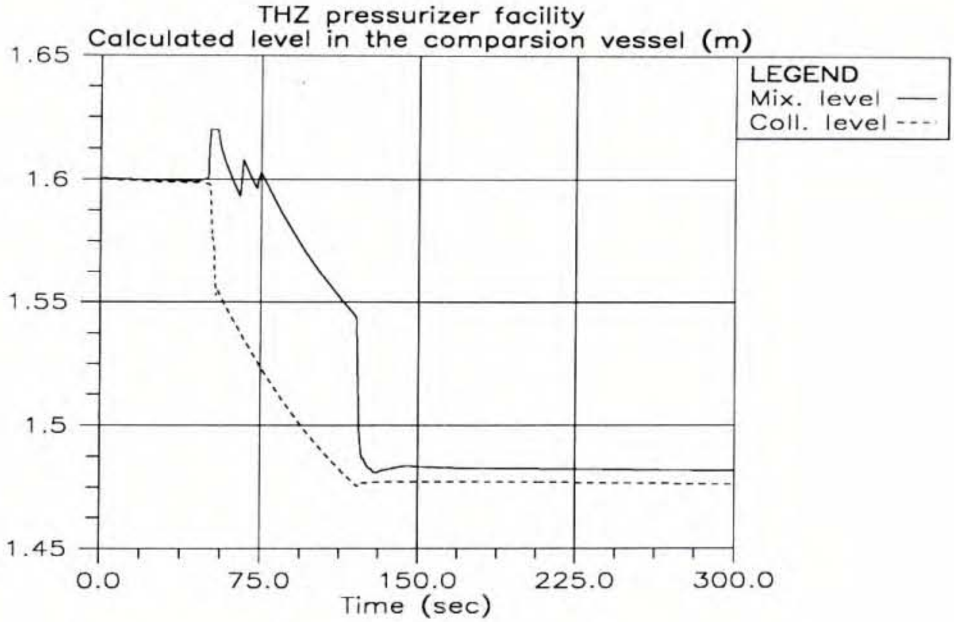


Figure 8: Calculated level in the comparison chamber

The maximal difference is 4 cm. The pressurizer level decreases under the lower connection of the one chamber measurement device (lower limit of this measurement system). Therefore it can only measure the height of the lower connection (1 m). The two chamber comparison vessel measures equal values like the one chamber vessel at the beginning of the blow down.
The mixture level in the pressurizer rises above the upper connection of the two chamber comparison vessel, whereby the measured level decreases suddenly.
After the blow down the two chamber comparison vessel measures a false level of 1.13 m. The reason of this is, that the level in the pressurizer is under the lower connection of the two chamber comparison vessel (lower limit of measurement system) and that the level in the comparison vessel was decreasing about 13 cm in consequence of loss of water.
In figure 8 the calculated level in the comparison vessel is shown. The zero point is not the bottom of the two chamber comparison vessel, but the bottom of the pressurizer, which is the geodetic zero point for the ATHLET simulation. During the blow down the mixture and collaps level are different, because of boiling in the comparison chamber. After the blow down the measured and calculated level in the comparison vessel are nearly identical.

Additionally the loss of water in the comparison vessel was measured by probes for void fraction.
Following this work the model of two chamber comparison vessel was tested at a model of steamgenerator within a complete circuit model of nuclear power plant. A reduced basic data set for the WWER 440/230 unit 1 in Greifswald was the basis of this investigation. The data set contains the primary circuit with 1 double and 1 quadruple loops. The steam generators, main steam line and steam collector were simulated on the secondary side. The disturbance "Failure of 2 main coolant pumps" was simulated, because during this disturbance the negative pressure gradient is relatively great and experimental data were available too. The pressure in the steam generator decreased from 4.5 to 4.0 MPa. The transient was post calculated, with special respect to the pressure of the secondary side and the behaviour of the model of two chamber comparison vessel especially. Different variants were calculated. The pressure in the steam generator decreased from 4.5 to 3.5 and 3.0 MPa, too for investigating the behaviour of model of two chamber comparison vessel by different pressure gradients.

The results of these simulations have shown, that the measured and calculated values rather coincide. The conclusion is, that not measurable, calculated values are almost identical with real process values, too. By means of this calculated parameters the verfication of the models for model-based measurement is possible.

4. Structure of correction and diagnosis algorithms

On the basis of analysis of existing faults and special effects by considering the requirements in a model-based measurement, described in introduction, a structure was developed, shown in Fig.9.

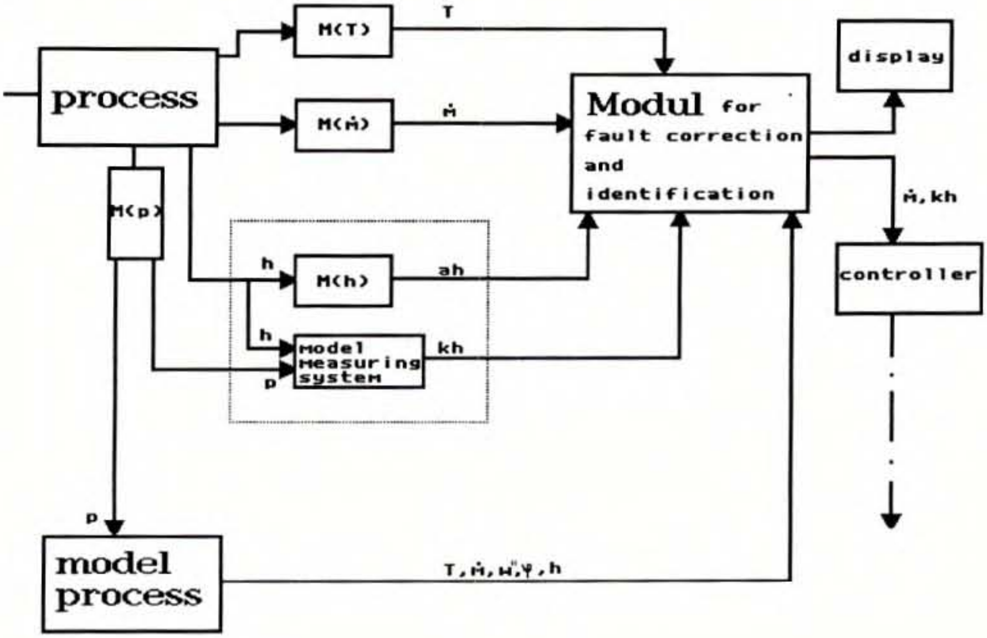


Figure 9: Total structure on fault identification and fault correction
M - measuring point, p - pressure, w"- steam bubble velocity, φ - void content, ah - shown water level, kh - corrected water level, T - temperature, h - water level

Thereby a thermohydraulic model will be coupled not only to the process but also to the level measuring system. The equalized pressure and the pressuring gradient respectively are the input parameters for the thermohydraulic model

The flux flatting used the method of the exponential middle-value-formation. This method guaranteed a very good flux flatting of values at real-time-conditions. The decison of the time constant is depended on the spezification.

The process modul gives the volumetric steam quality (void fraction) between the connections of level measurement, which is necessary for the mixture level calculation.

The modul "measuring system" describes the process in the comparison vessel of the measuring system. The mass transfer during negative transients is calculated and the condensation rate, which also counteracted the mass loss. The structure of this algorithm is shown in Figure 10.

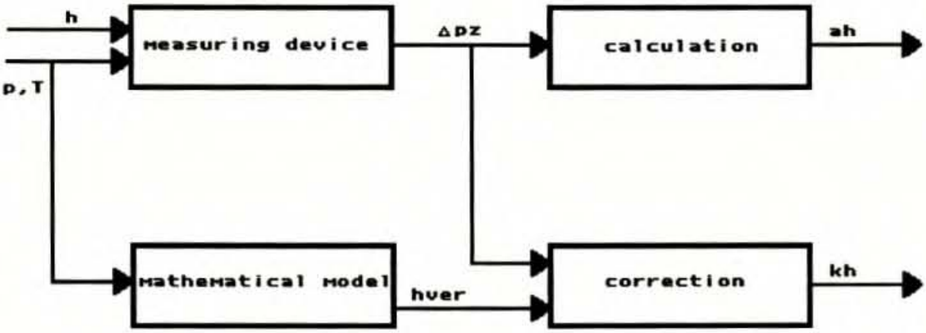


Fig.10: Stucture model-based-measuring system for level measurement
p - pressure difference; h - level,
T - temperature; ah - shown water level;
kh - corrected water level

The modul fault correction and fault identification forms the core of the system (Fig.11). Here the recognition and correction of level measuring values is carried out. The diagnosis of the measurements on the basis of a comparison between measuring and calculated state variables respectively and experiencing values aiming at the plausibility check is carried out, too. Methods of fuzzy-set-theory are used for the plausibility check .

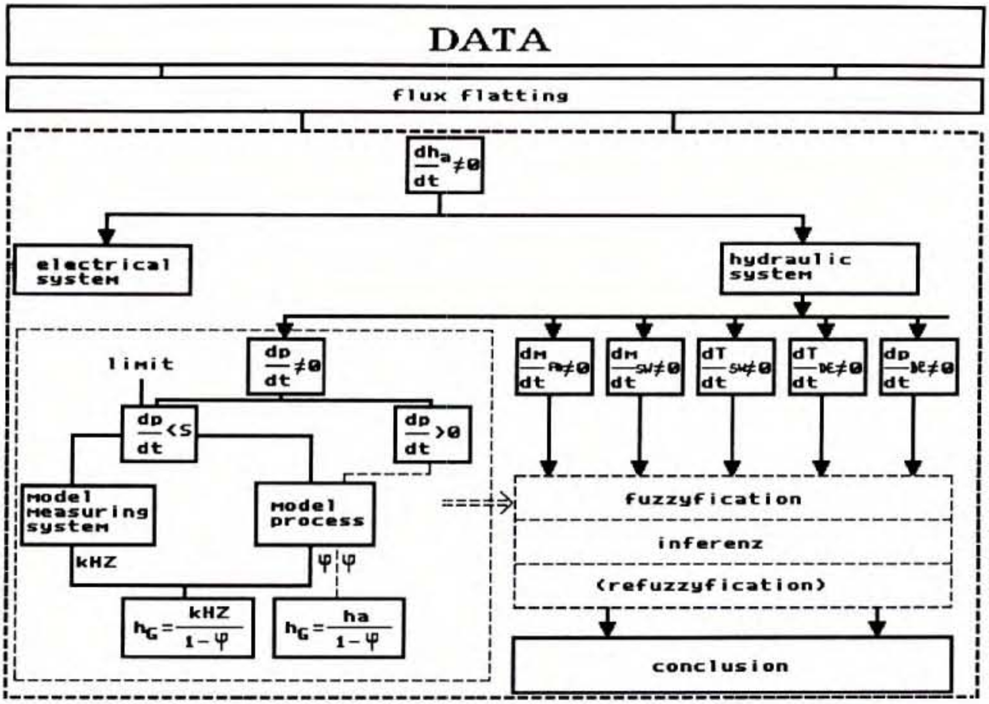
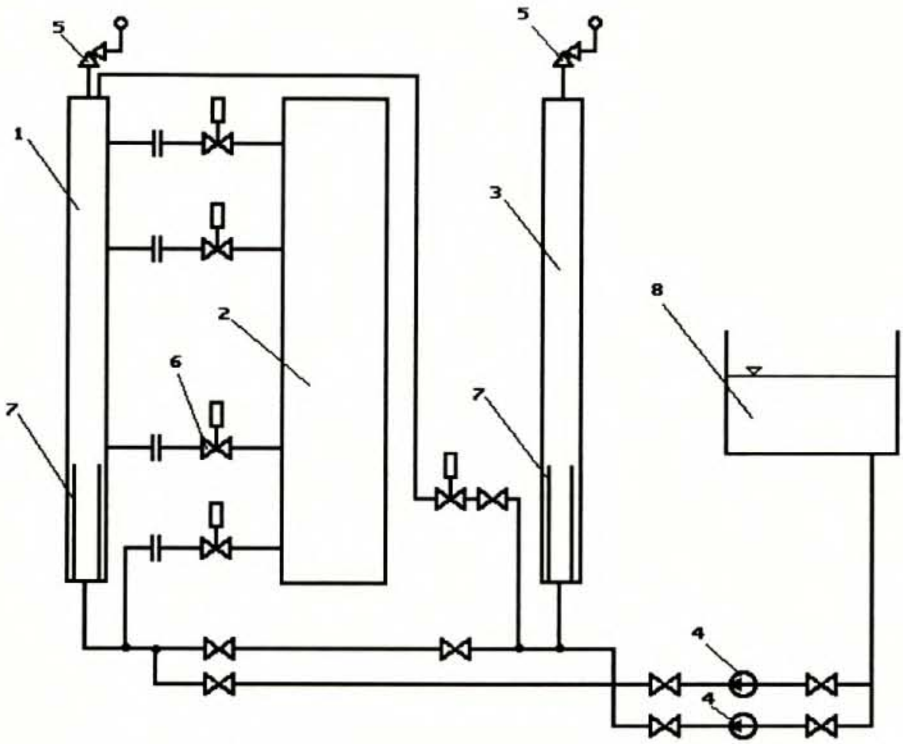


Figure 11: Modul fault correction and fault identification

There by the gain of knowledge is a combination of measuring values, calculated variables and linguistic assertion. Besides considering the momentary state for the state control the described algorithms consider temporal transients, too. So the system is very stable.

5. Experimental works

For experimental checking of the dynamic behaviour of the pressure vessel and the level measuring methods an experimental plant was installed at the Technical University of Zittau. The scheme of the experimental arrangement is presented in Figure 12. This pilot plant is designed for a pressure of 4 MPa.



1 pressure vessel model
2 steam collector (open)
3 vessel for simulation of transients
4 pumps
5 safty valve
6 throttle valve
7 heating
8 storage tank

Fig 12: Simplified diagramm of experimental plants pressurizer facility

For simulating real operation transients the following manipulations are possible

- to feed water with varying temperatures into the lower vessel part (water-filled room)
- to sprinkle water with varying temperatures into the upper vessel part (steam filled room)
- heating water with variable electrical power
- blowdown experiments with variable transients

- to blow- off steam and drain water (leakage simulation).

The experimental plant is connected to two computers for control and data aquisition, display and storage.
For demonstrating the generation of experimental data the comparison vessel instrumentation for hydrostatic level measurement is shown in Figure 13.

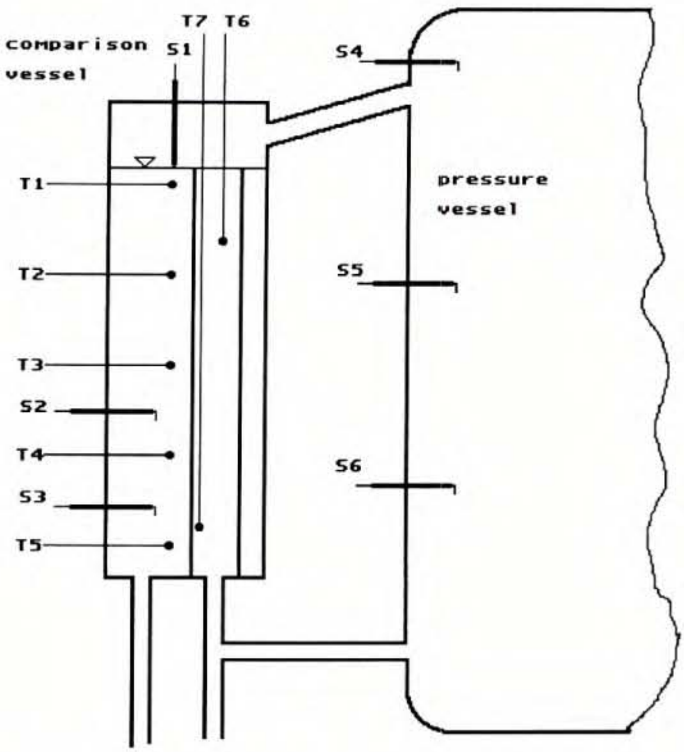


Fig 13: Instrumentation at the two chamber comparison vessel for verification

A very difficult problem is the determination of loss of steam and water from the comparison vessel into the pressure vessel by blowdown accidents. The measuring arrangement has to guarantee the verification of mathematical models. The measuring points of temperature T1 - T7 are used for the determination of the temperature distribution in the hydraulic system. The measuring points S1-S5 are probes for the measurement of the steam content, steam bubble velocity and the boundary surface detection, too. The steam content has to be known in the pressure vessel and in the comparison vessel for model calculations and the interpretation of the experimental data, too.

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Conductivity Probes for Two-Phase Flow Pattern Determination During Emergency Core Cooling (ECC) Injection Experiments at the COCO Facility (PHDR)

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1. INTRODUCTION

The scientific programme of nuclear safety investigations at the experimental power plant HDR in Karlstein near Frankfurt, a former prototype reactor which was shut down in the early seventies, includes thermal hydraulic emergency core coolant injection experiments too. They were carried out at the COCO facility located within the containment of the HDR. COCO stands for COntact COndensation, that means, the study of the condensation phenomena in a mixture of saturated steam with sub-cooled water was one of the main goals of the tests /1/.

In the case of the hot leg injection experiments, the COCO facility models the part of the main circulation loop of a KONVOI type NPP which connects the reactor outlet with the inlet of the steam generator (Fig. 1). The reactor vessel and also the steam generator are represented by two cylindrical separators on both ends of the COCO tube.

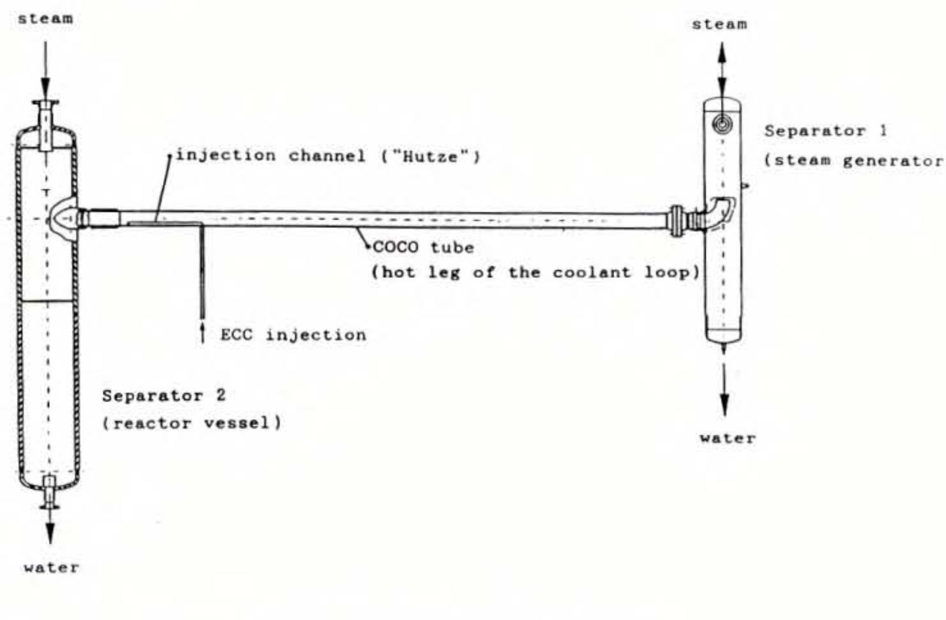


Fig. 1 General view at the COCO facility

Near separator 2 modelling the reactor vessel, the emergency coolant injection is located. The coolant is directed to the reactor vessel against the steam flow from the core by a cylindrical half-shell (Hutze) welded to the bottom of the main circulation tube.

In spite of the high number of different measurement sensors, the number of those ones which deliver immediate information about the structure of the two-phase flow is comparatively low. For this reason the use of needle shaped conductivity probes was obvious.

This kind of instrumentation had been developed at the Central Institute for Nuclear Research in Rossendorf especially for the utilisation on high pressure loops, as for instance LOCA test facilities. They allow the detection of the state of the fluid (liquid or vapour) at the tip of the probe with a relatively high time resolution in the range of milliseconds. During an off-line evaluation of the probe signal several average parameters of the two-phase mixture can be determined. But it is also possible to obtain information about the structure and the velocity of the two-phase flow.

As the probes had shown their suitability during the earlier cold leg injection tests at the COCO facility, the decision was made to use them again at the hot leg tests in autumn 1991.

2. THE MEASURING SYSTEM

2.1 The needle shaped conductivity probes

Fig. 2 presents a view of the probe used at the hot leg injection experiments. The sensitive element is a small ceramic tube with an electrically conducting tip which is in contact with the fluid. The supply with a small voltage (2 V (AC), 4 kHz) causes a current from the tip via the liquid toward the wall of the tube or vessel which is being interrupted when vapour (bubbles, plugs) covers the probe. In this way, water and steam can be distinguished from each other.

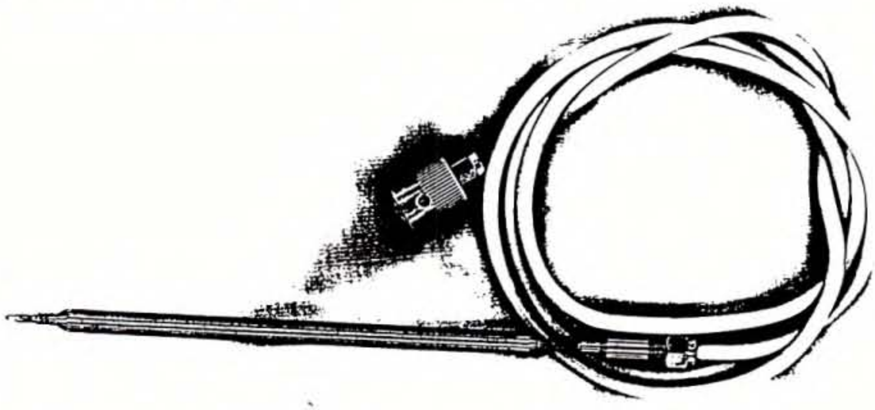


Fig. 2 View of the needle shaped conductivity probe

The ceramic tube is soldered into a steel bearing tube (Fig. 3). As the probes were developed for the utilisation under high pressure and temperature (130 bars, 300 ^{o}), but moderate velocities, the diameter of the ceramic was only 1 mm at the first tests during the cold leg injections. Although the absolute pressure at these experiments was comparatively low (4, 25 and 70 bars), the probes were not able to stand the high mechanical loads occurred by plug flow and void collapses in the COCO tube during the experiments. Most of them were destroyed after a few minutes plug flow.

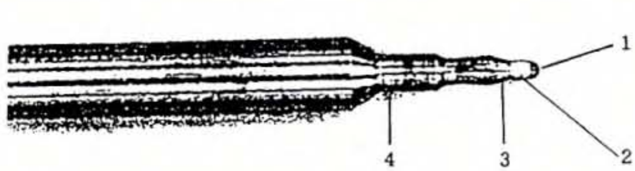


Fig. 3 The sensitive tip of the probe
1 - conducting contact, 2 - insulation tube (Al_2O_3 ceramic),
3 - brazing, 4 - bearing tube (stainless steel)

In the result of these pre-tests the probes were improved to increase their mechanical stability. First of all the diameter of the ceramic was increased up to 1.6 mm. In this way a satisfactory stability under the hard conditions within the COCO facility could be reached. Unfortunately, the higher diameter leads to a worse time resolution. Small bubbles touching the probe are not able to cover the large tip of the probe completely and can not be registered as individual events. In this case the probe current takes an average value between the values of steam and water which depends on the void fraction. This effect has to be taken into account watching the results of the measurements and has to be examined in the future.

1.2 Data acquisition

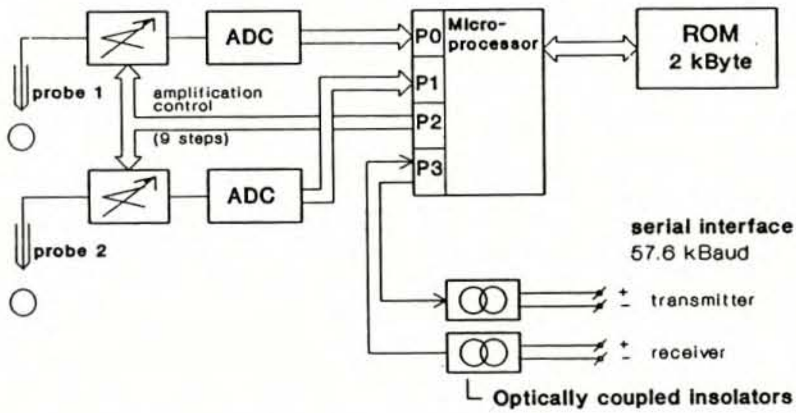


Fig. 4 Preprocessing module of the data acquisition system

For the utilisation of the probes at thermal hydraulic test facilities /2/ with directly heated electrical fuel rod simulators (the main goal before COCO) a special modular data acquisition system was developed, suppressing the high electrical disturbance levels occurred by the power supply of the rod simulators. The system is based on electronic modules for digital data preprocessing (Fig. 4).

Each module is equipped with a micro processor and can treat the signals of two probes. The signals are digitalized with the help of two ADC. Digital controlled amplifiers allow the optimal use of the range of the ADC. The preprocessed data are transmitted to a central data acquisition computer (PC) through a serial interface (RS232c). The data rate is 57.6 kBaud. With the help of optically coupled insulators the modules and the central PC are completely potential disconnected from each other. In this way, the disturbance level is kept low. The modules have to be mounted in the near of the probes to keep the connection short, therefore they dispose of robust water sprite safe aluminium casings.

The measuring system can be extended up to 16 probes by connecting 8 modules with one PC. The interface forms a double ring (Fig. 5). The PC has to be connected only with one of the modules by two screened cables not depending on the number of channels. This makes it easy to connect the modules over a long distance to the PC located outside the test facility (in the case of COCO outside the containment of the experimental NPP).

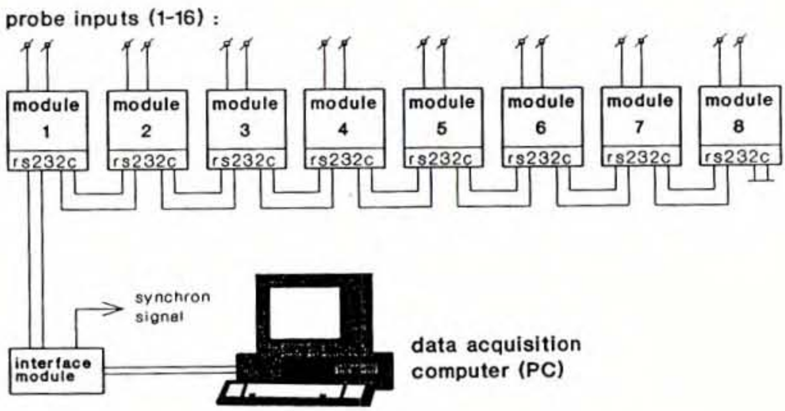


Fig. 5 Data acquisition system (network of modules)

The software of the measuring system includes the microprocessor code of the modules and programmes for the on-line data acquisition and the off-line data evaluation. The present version of the software allows two different working modes. The first is the average mode. The modules perform a data compressing by computing a set of characteristic values of the probe signal, which is transmitted to the PC in a polling cycle of at least 1 sec. These data sets allow to determine the state of the fluid and the probe and to calculate the void fraction and the phase changing frequency in every polling cycle.

Due to the expected transient behaviour of the flow in the case of the COCO experiments a second, additional working mode was developed to record the time signal with a resolution in the range of microseconds. The final time resolution depends on the number of the activated modules. In the case of one module with two probes the polling frequency is 1.92 kHz, 8 probes, for instance, can be scanned with 480 Hz. The modules can be switched on and off from the PC. The maximum duration of one measurement is 60 seconds.

As in time signal mode the measurement is to be started by pressing a button on the PC keyboard an additional binary output has been organised, which provides the standard PCM tapes of the COCO facility with a syncronization signal. The signal is being switched active during the recording of the time signals. In this way it is possible to compare the results of the probe measurements with the signals of the other sensors of the COCO facility (thermocouples, gamma-densitometres etc.) recorded on PCM tape.

2. REALISATION OF THE HOT LEG ECC INJECTION TESTS

2.1 Positions of the probes

During the hot leg injection tests at the COCO facility 8 probes were in use. According to this number of probes 4 modules had to be mounted near the COCO tube within the containment of the experimental NPP. The probes themselves were mounted into orifices with an inner diameter of 6 mm. Fig. 6 shows the positions of the probes. The probes number 1 and 2 were located right in front of the separator 2, which represents the reactor outlet.

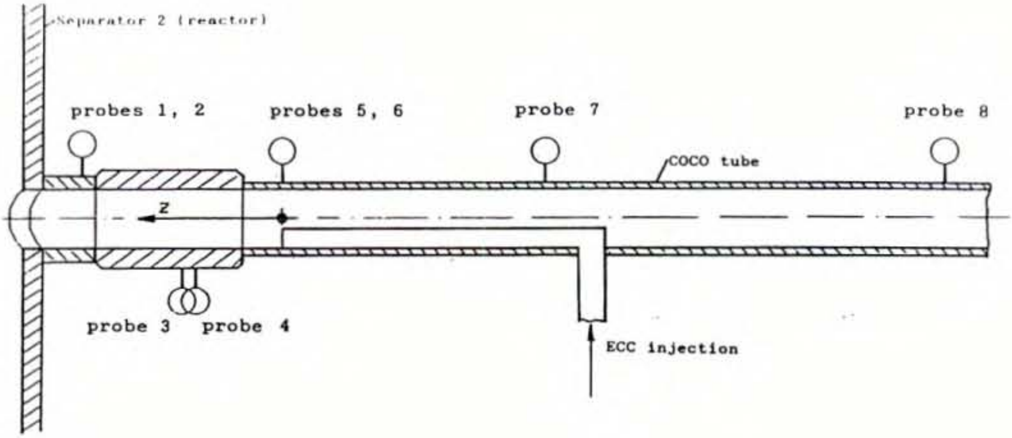


Fig. 6 Location of the probes at the COCO tube

Coordinates:

probe 1	z= 0.575 m	d=30 mm	a=336.5 deg
probe 2	z= 0.575 m	d=10 mm	a= 23.5 deg
probe 3	z= 0.303 m	d=37 mm	a=180.0 deg
probe 4	z= 0.263 m	d=37 mm	a=180.0 deg
probe 5	z= 0 m	d=30 mm	a= 23.5 deg
probe 6	z= 0 m	d=10 mm	a=180.0 deg
probe 7	z=-0.800 m	d=10 mm	a= 0.0 deg
probe 8	z=-2.000 m	d=10 mm	a= 0.0 deg

z - axial position, d - depth of mounting,
a - angle (180 deg = probe mounted from below)

The probes 3 and 4 can be utilised to measure the fluid velocity by means of cross correlation as they are mounted in a distance of only 40 mm from each other. The probes 5 and 6 were located directly upon the end of the half-shell shaped injection tube ("Hutze") to indicate the water flow reverse at high steam speed. The probes 7 and 8 were scheduled to observe plugs moving along the tube toward the separator 1 (steam generator).

2.2 Availability of the probes

In spite of the robust design of the probes a certain number of them was destroyed by the high mechanical loads within the COCO tube, especially during the 4 bars experiments. Due to the fact that the facility was standing under pressure for some days before the start of the 4 bars test series, 2 of the 8 probes were already out of order at the beginning. Series of hard void collapses led to the fast break down of the probes at 4 bars, while at 25 bars the situation was not so drastic (Fig. 7). Even at 14 of 64 test points all the 8 probes were available and the number of probes was dropping quite more slowly than during the 4 bars tests.

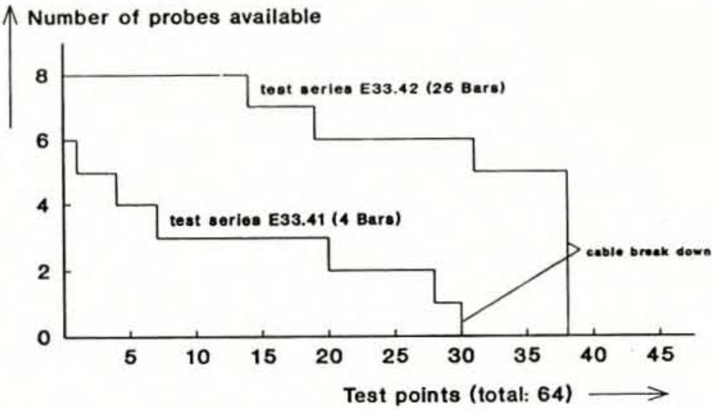


Fig. 7 Statistics of the probes available

The obviously higher stability of the probes at the higher pressure points out the dominating role of the void collapses in the destruction process. At 25 bars the density of the steam is 6 times higher then at 4 bars. That means that at approximately the same condensation rate comparable voids collapse slower at the higher pressure producing less mechanical loads for the probes.

Unfortunately besides the break down of the probes two times there were cable failures because of the high temperature within the containment, so that the whole measuring system went out of order. For this reason it was not possible to measure at approximately 50 % of the test points at 4 and 25 bars and during the whole 70 bars series despite of a certain number of good probes at this time.

3. FIRST RESULTS

3.1 Interpretation of the time signals

A broad evaluation of the extensive experimental data has not taken place yet. At the present time, only first qualitative results can be given. Especially the comparison with the data of other measurements is a task for further work. In spite of these circumstances, the discussion of the probe signals without quantitative evaluation can give valuable information about the flow pattern within the COCO tube.

To show the changes in the flow pattern caused by various steam velocities let's pick out a series of test points with constant ECC injection but varying steam flow (test points from E33.4271 to E33.4278 at 25 bars). During this series all important probes were available.

The water flow was constant at 14.958 kg/sec and the steam flow was decreased from 4.203 kg/sec at the point E33.4271 down to 1.121 kg/sec at E33.4278.

The probes 1 and 2 (and 5, 6 respectively) have always recorded nearly the same time signals according to their identical axial positions. Small differences occurred only because of their different mounting depth. The probes 3 and 4 have given almost similar results because of their small axial distance of 40 mm, but the delay between the signals of these probes can be used to determine the velocity of the fluid.

Dropping away these details for the first time the signals of the probes 1, 4, 6, 7 and 8 shown in Fig. 8-10 are characteristic for the three main flow regimes within the COCO tube.

At small steam mass flow (Fig. 8, test point E33.4278) the water flows out of the "Hutze" toward the separator 2, i.e. the reactor, forming a counter current flow with the steam coming out of the reactor and flowing toward the steam generator. The flow is stratified and within the boundary layer between water and steam a two-phase mixing zone is developing, indicated by the probe 4. All the probes mounted from above indicate only steam flow with very few droplets.

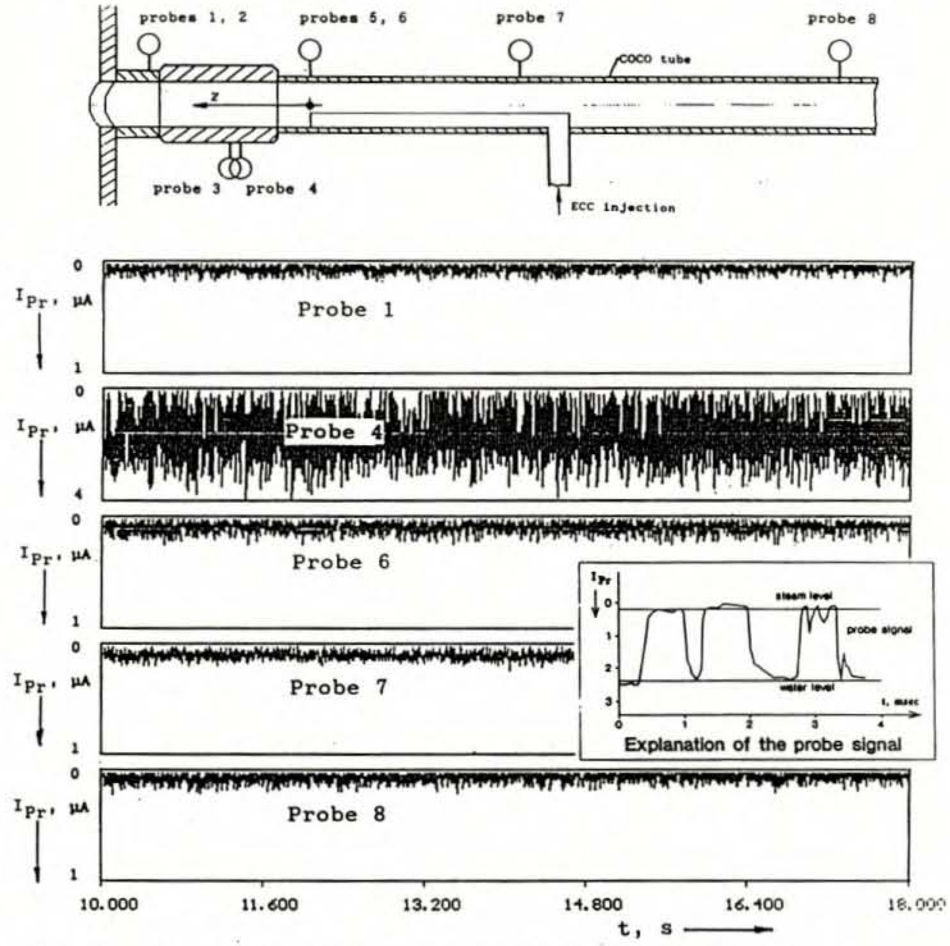
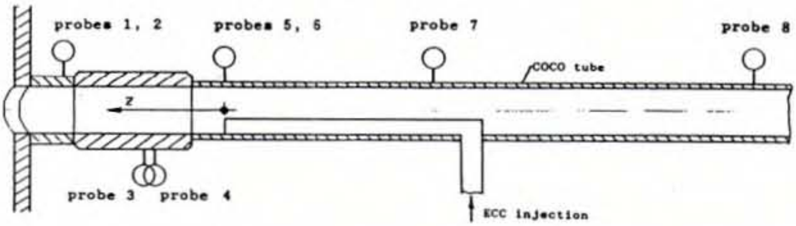


Fig. 8 Stratified flow toward the separator 2 (reactor)

Test point : E33.4278
Steam flow rate = 1.121 kg/sec
ECC injection flow rate = 14.985 kg/sec

At high levels of steam flow (Fig. 9, test point E33.4271), a perfect flow reverse can be observed. The water is directed toward the steam generator by the strong steam flow, so that the probes 1 and 4 (2 and 3 respectively) stay dry. There is a surplus of steam, so it can not be condensed completely. The flow pattern is now a droplet flow toward separator 1, which is indicated by the probes 6, 7 and 8.



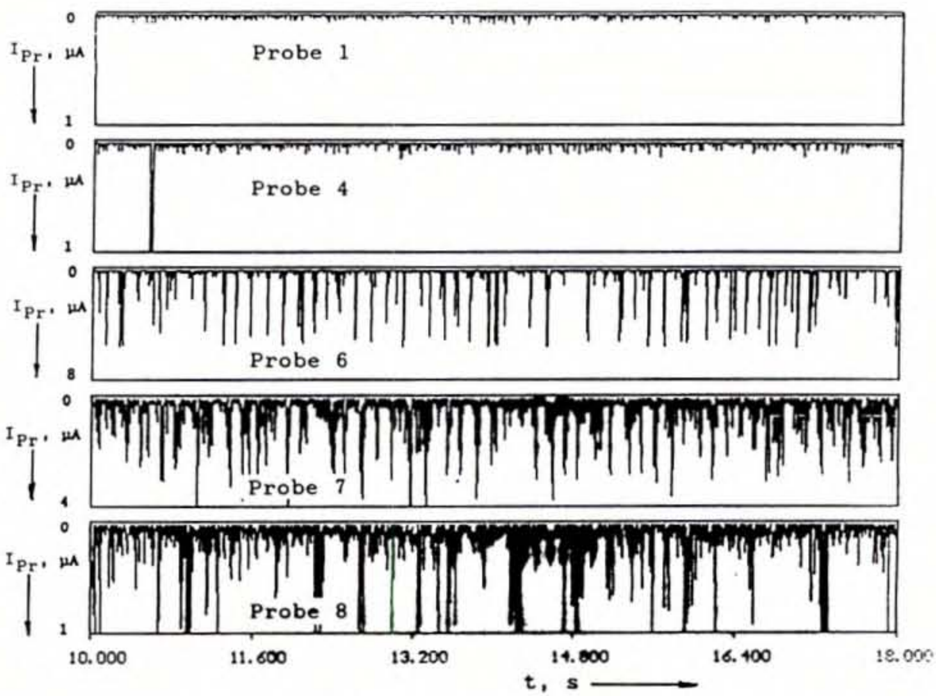


Fig. 9 Perfect flow reverse toward separator 1 (steam generator)

Test point : E33.4271
Steam flow rate = 4.203 kg/sec
ECC injection flow rate = 14.985 kg/sec

Between the two extreme cases there is a transition state at average steam mass flows characterised by particular flow reverse with an intermitting of counter current flow and flow reverse.

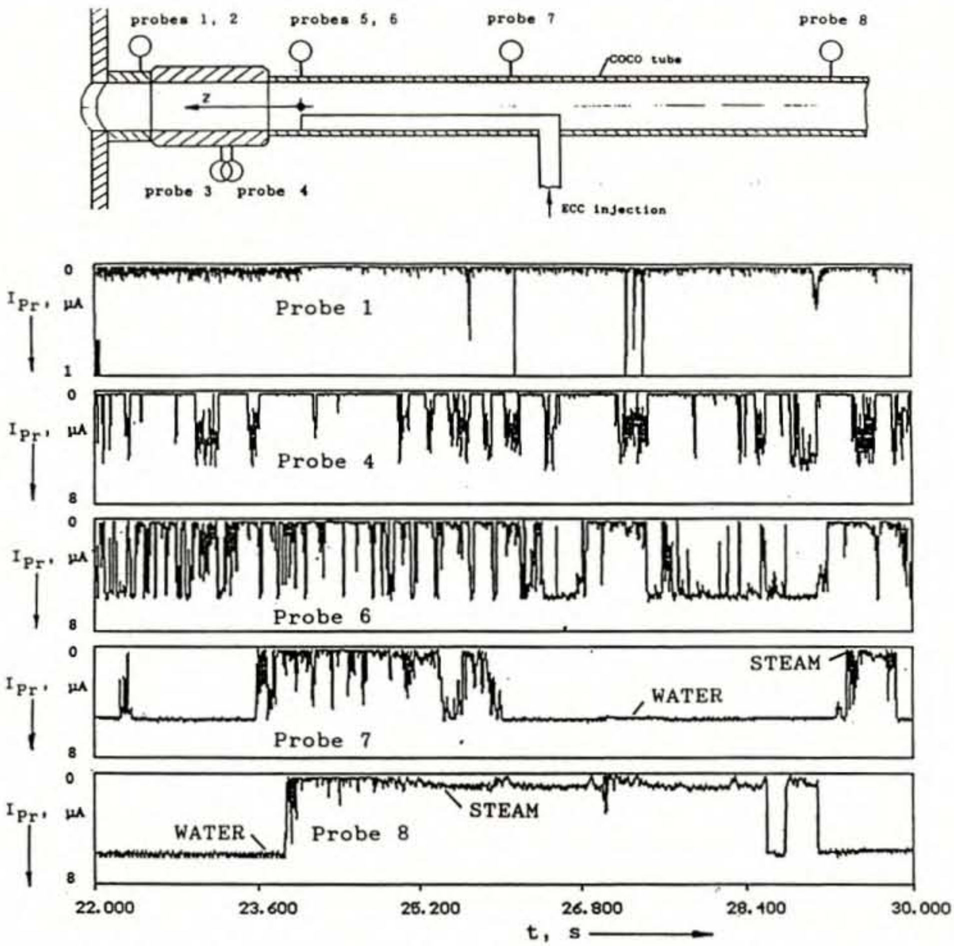


Fig. 10 Particular flow reverse with plugs and void collapses

Test point : E33.4274
Steam flow rate = 2.587 kg/sec
ECC injection flow rate = 14.985 kg/sec

(Fig. 10, test point E33.4274). The probe 4 shows alternating phases of steam and two phase flow. One part of the injected water is still flowing into the reactor, but another part is directed to the steam generator. At these values of flow ratio between steam and water injection, the steam can be mainly condensed and large plugs are moving toward the separator 1 (steam generator), indicated by the probes 7 and 8. Taking into account that these probes are mounted at a depth of only 10 mm from above, it is obvious that the plugs fill the tube completely.

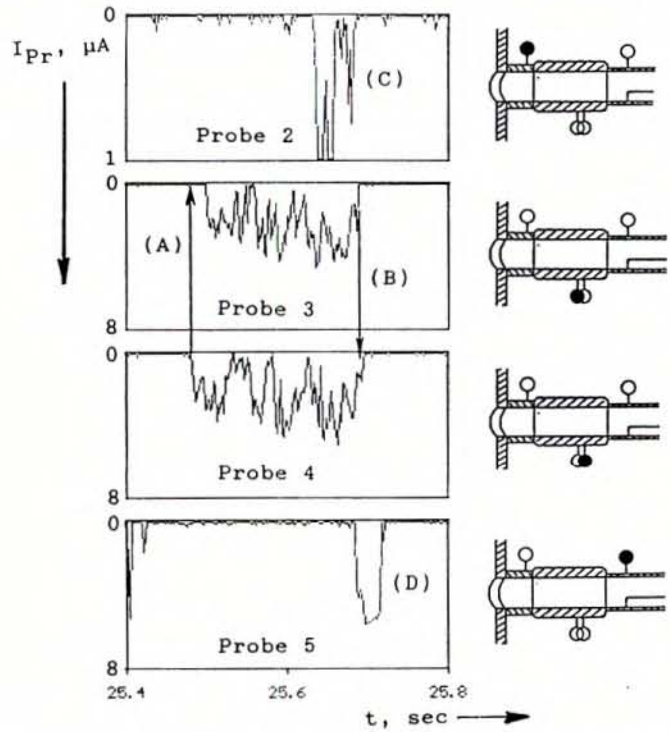


Fig. 11 Unsteady flow pattern in the case of particular flow reverse (highest time resolution)

Test point : E33.4274
Steam flow rate = 2.587 kg/sec
ECC injection flow rate = 14.985 kg/sec

Fig. 11 shows a typical process of this unsteady flow regime. The transition from flow reverse to counter current flow starts with the moistening of probe 4 (A) a few milliseconds earlier than probe 3. That means, the front of water moves from the injection point toward the reactor. The tear off of the flow (B) happens in the inverse order. The probe 3 indicates steam at first. The swapping back of the water is also indicated by the probe 2 (and 1) just before the probes 3 and 4 start to see steam (C). The wave of water also reaches the probe 5 above the injection point (D).

3.2 Velocity measurement by cross correlation

The signals of the probes 3 and 4 can be used to determine the velocity of the fluid. Fig. 12 shows the cross correlation functions calculated from the time signals in the case of three different ECC injection mass flow rates responding to three test points with stratified flow toward the separator 2 (reactor). The steam flow was kept constant.

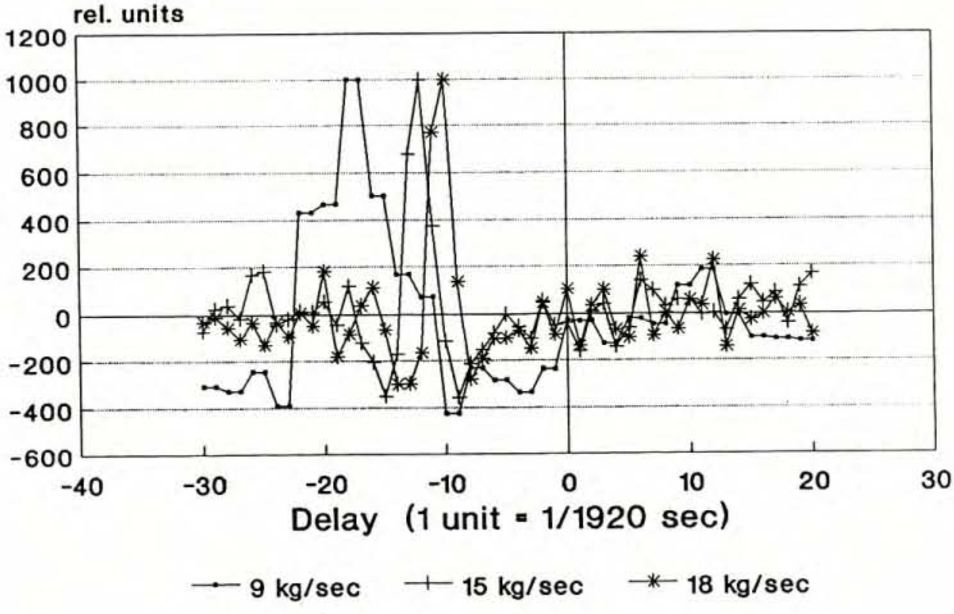


Fig. 12 Cross correlation functions, calculated between the signals of probe 3 and 4

It is obvious that the cross correlation functions have a narrow peak at a negative delay, that means the flow is directed from probe 4 toward probe 3. The values of the velocity, which respond to the cross correlation function are displayed at Fig. 13. They have been compared with the average velocities of the injected water within the injection channel ("Hutze") calculated from the mass flow and the cross section of the channel. There is a remarkable agreement between the measured and the calculated values, that means that the interaction between the two phases does not lead to a significant slow down of the water on the way from the injection point to the position of the probes (approximately 300 mm).

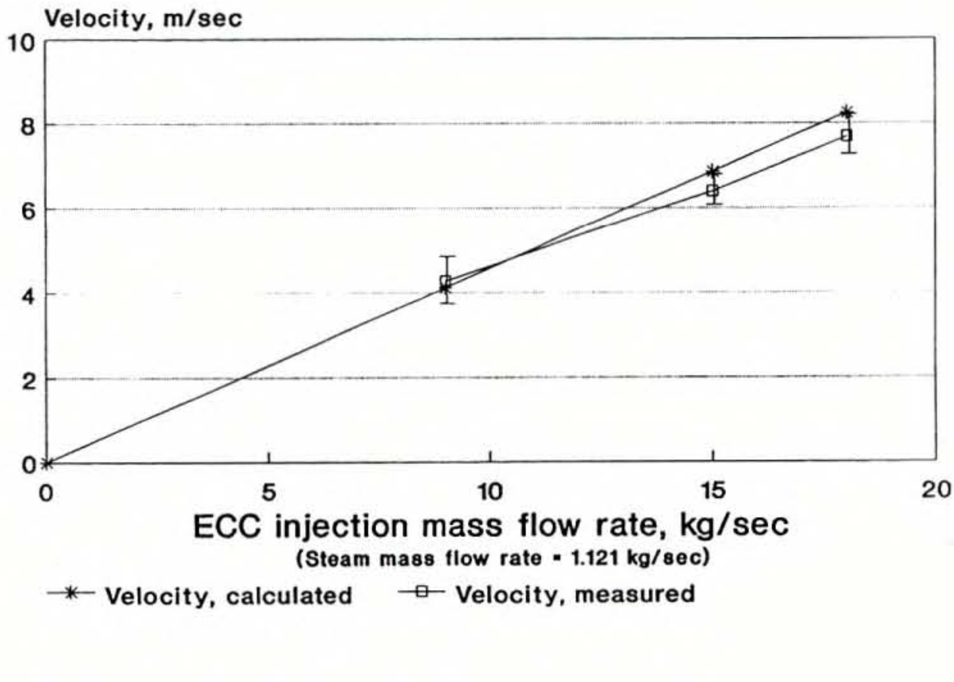


Fig. 13 Velocities measured by cross correlation

4. SUMMARY

The first qualitative evaluation of the data obtained by the needle shaped conductivity probes during the COCO experiments shows the great use of such probes for the determination of the two-phase flow pattern. The signals are very clear and the interpretation is easy. The derivation of void fractions and phase change frequencies by an off-line evaluation seems to be very promising and is the task of further work.

The needle probe is an interesting kind of a low-cost sensor for the instrumentation of reactor safety experiments. It is also possible that it can be used as an additional instrumentation on NPPs after a future improvement.

After the end of the Central Institute for Nuclear Research Rossendorf in December 1991 the work on two-phase flow instrumentation including the needle probe is being continued at the Institute for Safety Research which is a part of the new Research Centre Rossendorf.

5. LITERATURE

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SEVERE ACCIDENT MANAGEMENT INSTRUMENTATION IN THE FINNISH NPP'S

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ABSTRACT

New instrumentation has been installed at the TVO plant (ABB BWR with 710 MWe net output) in connection with the severe accident mitigation program implementation in 1989. The developmental stage of the severe accident management program of the Loviisa plant (VVER-440) has recently allowed the definition of instrumentation needs. This paper is aimed at discussing the principal approaches, how the plant-specific instrumentation needs have been derived from the safety functions of the severe accident management in each case. A distinction is made between the instrumentation that is of crucial importance for performing a correct management measure and the instrumentation needed for monitoring the success. New instrumentation is rather strictly limited to those ensuring the safety functions.

1. INTRODUCTION

The Finnish utilities follow the Finnish Regulatory Guides (YVL Guides) set by STUK (Finnish Centre for Radiation and Nuclear Safety) in developing their severe accident management (SAM) plans. Mitigative measures are required based on deterministic criteria. YVL Guides [1] require that the containment must remain intact and the environmental release of cesium must be less than 0.1% of the core inventory in case of a severe accident. A Decision of the Council of State [2] further specifies that releases of Cs-137 must be less than 100 TBq.

There are four nuclear power units in Finland: two ABB 735 MW BWR's, TVO I and II in Olkiluoto, which are owned and operated by Teollisuuden Voima Oy (TVO), and two VVER-440 PWR's, Loviisa 1 and 2, which are owned and operated by Imatran Voima Oy (IVO). All these units started power operation between 1977 and 1981.

In June 1986 STUK requested the utilities to prepare against severe accidents at all units. Even though it is of foremost importance to prevent a serious reactor damage or reduce its probability, irrespective of such measures STUK required a preparedness to face a potential severe accident and to implement the measures which are necessary to restrict its consequences. Thus, STUK required the utilities to start a project concerning this issue [1].

TVO completed the implementation of SAM program in 1989. Since Loviisa plants have plenty of very plant-specific features, it has taken considerably longer for IVO to develop a consistent SAM approach. To complete the implementation of the IVO program is now foreseen to take place in 1993-94.

This paper focuses on the severe accident management (SAM) instrumentation that has been or will be added in the Finnish NPP's. We begin (Chapter 2) with a short description of the plants and their SAM programs. The safety functions to be ensured with SAM application are then presented in Chapter 3, and the instrumentation necessary for ensuring the given safety functions in Chapter 4. Specific requirements and features of the SAM instrumentation will be discussed in Chapter 5.

2. PLANT-SPECIFIC FEATURES AND SAM PROGRAMMES

TVO I AND II

The design philosophy of minimizing the likelihood of the core damage accidents has been achieved by applying diversity and redundancy of (4x50%) in the design of the engineered safety features (ESF) and by certain structural solutions of the primary circuit (internal recirculation pumps, no large pipeline connections below the top of active fuel). The containment (see Fig.1) is normally inerted with nitrogen during power operation. This was deemed necessary already from the hydrogen considerations of design basis accidents.

The severe accident mitigation programme for the TVO units was started in August 1986, and plant modifications and procedural changes [3] were mainly completed by the end of 1989. The backfitting changes are schematically shown in Fig. 1. These systems include large-capacity containment overpressure protection, flooding of the lower drywell, shielding of penetrations in the lower drywell, containment filtered venting and water filling of the containment. Some new instrumentation was added to ensure successful implementation of severe accident management measures. This instrumentation and its design principles will be discussed in detail in the following chapters. The whole accident management concept is based on manual actions. These operator actions are guided by the new emergency operating procedures (EOPs) [4].

Level 1 probabilistic safety assessment (PSA) has been completed for the internal initiators (1989) and for most external events (1991) [5]. PSA level 2 scoping analysis is currently under preparation.

LOVIISA 1 AND 2

The Loviisa units have been furnished with ice condenser containments as illustrated in Fig. 2. The ESF system is a two train solution of its mechanical part with two parallel pumps in each train (2x2x100%). The ESF actuation system is a 4-train Simatic N dynamical control system. The reactor trip system (2 trains) logic has been implemented by relay technique.

The primary circuit incorporates six loops and six horizontal steam generators. The reactor pressure vessel has a large-volume lower plenum. The control rods are designed as followers of fuel assemblies. The essential implications of these features for severe accidents have been discussed in Ref. 6. Large water inventories of primary and secondary circuits define very long response times for transient-initiated severe accident scenarios (start of core melt in 7...9 hours).

The severe accident mitigation programme started as early as 1986. Because of the very specific plant design, the utility was forced to launch an intensive research programme before any decisions of backfitting measures could be taken. The most important measures taken until now are the external containment spray system (operable in 1991) and reinforcement of some specific containment structures to withstand elevated pressure and temperature conditions. The current status of severe accident assessments and SAM programme has been described in detail in Ref.[6]

Level 1 PSA has been completed for internal events in 1989 [7]. Results of the study have been integrated to our severe accident assessments. The PSA work continues with external initiators and shutdown conditions.

3. SAM SAFETY FUNCTIONS

TVO I AND II

The critical functions of accident management for the TVO plants were defined as ensuring the following goals [3]:

- Containment overpressure protection
- Prevention of early containment failure
- Limitation and control of releases
- Reaching a safe stable state.

Containment overpressure protection

A break in the primary circuit in combination with a failing pressure suppression function might cause a violent containment rupture, which could incapacitate the ESF systems and thus lead to a severe accident. To eliminate these accident sequences, the containment was equipped with an overpressurization protection system with rupture disks and shutoff valves for later closing.

Prevention of early containment failure

An early containment failure could be caused by an energetic phenomenon or by corium attack on the lower drywell penetrations. Of the energetic phenomena, hydrogen combustion can be ruled out as a failure mechanism, because the containment is normally inerted with nitrogen. To avoid direct containment heating (DCH), the primary circuit has to be depressurized before the reactor vessel melt-through. This can be done with the help of the ordinary relief and depressurization system.

To protect the penetrations in the lower drywell, the compartment has to be flooded with water before the reactor vessel melt-through. Flooding of the lower drywell using condensation pool water has originally been included in the LOCA management schemes, and hence provisions for it have been accounted for in the basic design. The penetrations in the lower drywell were also shielded against direct contact with the molten corium and against missile impacts.

Limitation and control of releases

If it would become necessary to release steam and gas from the containment to limit the pressure rise, it can be performed in a controlled manner with the containment filtered venting system. The Siemens/KWU filter employed ensures that integrated releases to environment are below the tight requirements set in Ref. [2].

To make sure that the releases can be limited and controlled, it had to be shown that the containment can preserve its long-term leak-tightnesss under severe accident conditions.

Reaching a safe stable state

This goal can be accomplished by filling the containment with water up to the normal core upper level. To this end, connections have been provided between the fire fighting water system and the containment spray system.

LOVIISA 1 AND 2

The top level critical functions of the Loviisa accident management scheme are as discussed in Ref. [6]

- Absence of energetic events
- Coolability and retention of molten core on the lower head of the reactor vessel
- Long-term containment cooling.

In addition to these, primary circuit depressurization starts the severe accident phase of emergency response.

Primary circuit depressurization

The decision has been made to install manually-operated depressurization capability by power-operating the safety valves. Depressurization capacity will be designed for accident prevention. Yet, it is to be initiated with first indications of superheated temperatures at core exit thermocouples and it sets in motion the severe accident phase of the emergency response.

Absence of energetic events

It is possible to demonstrate absence of energetic reactor vessel failure and absence of vessel melt-through. Thus, for Loviisa, the only real energetic concern is due to hydrogen combustion events. Because of the relatively low design pressure, this concern involves all large scale combustion events that are rapid enough to yield an essentially adiabatic behavior. Glow-plug igniters were installed in the Loviisa containments in 1982. Intensive research program is currently under way to study a reliable hydrogen management scheme. The studies concentrate on two functions: ensuring air recirculation flow paths to ensure well-mixed atmosphere (opening of ice condenser lower doors) and effective ignition/combustion.

Lower head coolability and melt retention

In-vessel retention of molten core on the lower head of the reactor vessel constitutes the cornerstone of the Loviisa accident management approach. Since the ice condensers melt in most accident scenarios, the reactor cavity is fulfilled with water and the reactor vessel is submerged. A typical decay heat power level is low (~9 MW). Thus local heat fluxes and vapour velocities around the vessel are rather moderate. As a accident management measure, however ,the lower head insulation and neutron shield blocks should be lowered during the accident.

Long-term containment cooling

In the absence of corium-concrete interactions, there is no production of non-condensible gases, except hydrogen which is to burn. Stabilization of containment pressurization can be achieved by steam condensation on the containment walls. The external spray system of the inner steel containment has been installed to induce the necessary rates of steam condensation inside.

4. SEVERE ACCIDENT MANAGEMENT INSTRUMENTATION

All the Finnish units have their own critical function monitoring system, which supports the operator in severe accident prevention. Also the standard instrumentation, including the DBA instrumentation, is available in many sequences and operators have access to these measurements in likely cases. On the other hand, in some nonexpected cases, such as a loss of control room, there is a very limited number of measurements available for accident monitoring. To ensure also in these cases the success of the severe accident safety functions, as discussed in previous Chapter, some new instrumentation has been and will be added to the plants.

When planning the needed instrumentation, the distinction is made between the instrumentation that is of crucial importance for performing a correct SAM measure and the instrumentation needed for monitoring the success of the measure. As well the new instrumentation is rather strictly limited to those helping to ensure the above safety functions. Tables I and II present the SAM instrumentation scope for TVO and Loviisa plants, respectively. Figure 3 illustrates schematically the SAM instrumentation of TVO plants.

TABLE I Added SAM instrumentation for TVO plants

SAFETY FUNCTION	INSTRUMENTATION AND CONTROL EQUIPMENT FOR SAM ACTION	INSTRUMENTATION FOR PARAMETER MONITORING
Containment overpressure protection	Automatic actuation (rupture disc)	Indication and alarm of disc rupture
Prevention of early containment failure	Containment pressure Reactor pressure Water level in lower drywell Drywell/wetwell pressure difference	Pressure, temperature and water level inside containment
Limitation and control of releases	Dose rate in drywell/wetwell Water level in containment Containment pressure	Activity and flowrate in filter stack Scrubber water level
Reaching a safe stable state	Containment water level and pressure	Containment water level and pressure

TABLE II SAM instrumentation and related hardware for Loviisa

SAFETY FUNCTION	INSTRUMENTATION AND CONTROL EQUIPMENT FOR SAM ACTION	INSTRUMENTATION FOR PARAMETER MONITORING
Primary circuit depressurization	Core exit temperatures Power-operating of safety valves	Primary circuit pressure
Absence of energetic events	Glow plug actuation Power-operating of ice condenser lower doors	Hydrogen concentration
Lower head coolability and melt retention	Water level in the sump and cavity Lowering device for neutron shield and thermal insulation	Sump temperature Water level in the sump and cavity
Long-term containment cooling	Containment pressure Controls of the external spray system	Containment pressure

5. SPECIFIC FEATURES AND REQUIREMENTS OF SAM INSTRUMENTATION

The instrumentation related to SAM function are classified to safety class 3 according to Finnish safety guide YVL 2.1 (Draft).

TVO I AND II

The following design bases were applied to the SAM instrumentation:

- The instrumentation shall remain capable of functioning during a 24 hour loss of all AC power. This is accomplished by a dedicated battery system.

- The primary instrumentation shall withstand the environmental conditions prevailing inside the containment during a severe accident so that the instrumentation remains capable of functioning. The design range for the containment pressure is -0.1 to +0.6 MPa, the maximum conceivable temperature is 300°C and the integrated dose may amount to 10^4 Sv.

- The measurements fulfill seismic requirements throughout the measuring chains.

- The measurements have been implemented in two parallel, redundant chains.

- Water levels, pressures and differential pressure over the diaphragm floor are measured with the help of nitrogen purging with sufficient, passive backup capacity for 24 hours of operation against maximum containment pressure.

- The impulse tubes inside the containment are protected against missile impacts by shields made of channel bar.

- Molten core material must not damage the measurements.

- Transmitters are all installed outside the containment in rooms where there are no process system components. Hence, accessibility is assured.

The measurements are indicated on a central panel in a special emergency monitoring centre, which is located in the emergency exit corridor of the reactor building, close to the front door.

The measurements can also be presented on the CRTs of the plant computer.

LOVIISA 1 AND 2

The I&C functions, related to the above defined SAM safety functions, and their implementation principles have been presented in Table III. Some of the devices have been installed already. A system for deliberate hydrogen ignition was installed in 1982. The igniter system consists of 72 glow plugs, which are powered from the emergency diesel generators. The containment external spray system is operable since 1991.

TABLE III SAM I&C functions at Loviisa

I&C function for SAM	Implementation principle	Number	Note
Primary pressure	Pressure transducer outside containment	2	e, m
Core exit temperature	Thermoelement	4	e, m
Containment pressure	Pressure transducer outside containment	4	e, m
Hydrogen concentration	Catalytic device	6	e
Sump temperature	Tesistance thermometer	2	e
Water level in sump	DP-transducer outside containment	2	n(e)
Water level in reactor cavity	DP-transducer outside containment	2	n
Power-operating pressurizer safety valves	Manual remote control	2	n
Lowering lower head neutron shield	Manual remote control	2	n
Hydrogen igniter actuation	Manual remote control	2	e, m
Single drive controls of the external spray system	Relay logic in a local safety control panel	6	n(e)

Notes:
e existing system
m existing system will be modified
n new system to be implemented
n(e) new system already implemented

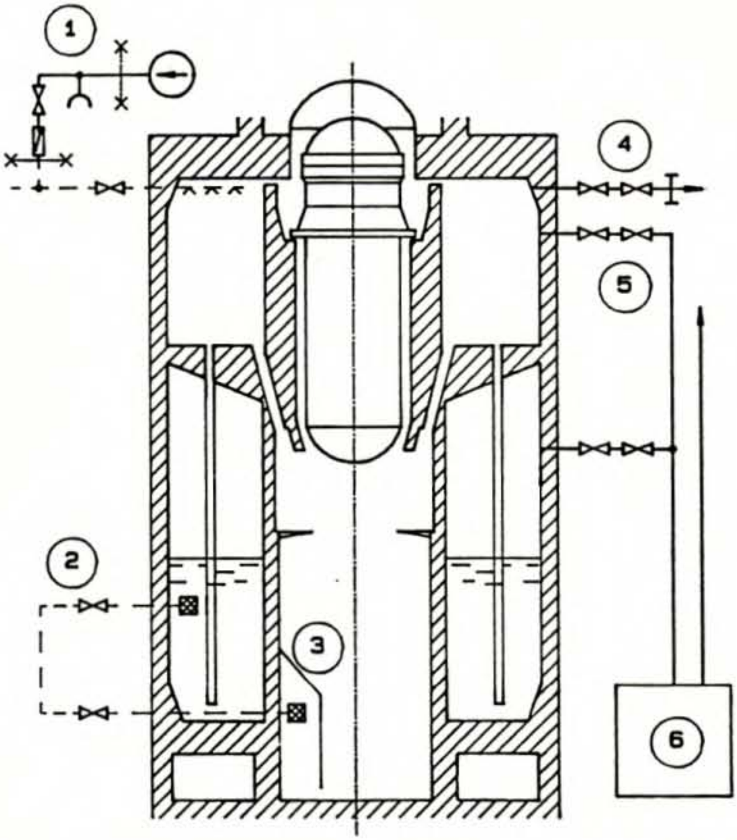
Some of the new systems are still under development. Therefore, Table III shows our principal approach and changes to these plans are still possible.

The design objective of SAM i&C is to be independent of the DBA instrumentation. In first place, it means independence of the control building of the unit and of electonics located there. The electronic part of the SAM instrumentation is designed to be located in the control building of the neighboring unit. Pressure transducers will be located outside the containment in the lower part of the reactor building.

All SAM instrumentation have at least two trains. Also in some cases, functional and/or component diversity can be employed. The aim is to have clear technical solutions employing reliable and passive (to the possible extent) components. Special attention has to be paid on equipment located in the rooms where environmental conditions of a severe accident can appear.

All SAM I&C equipment, and DBA I&C equipment having a support function for SAM, will have a backup power supply. The power supplies of equipment will be backed by a diesel system dedicated to SAM.

The environmental qualification requirements for SAM equipment will be specified for Loviisa conditions. Even though this work has not yet been completed, it is assumed that capability can be proven in most cases by analytical methods. Hence, extensive new testing is not deemed necessary.



1	Water filling of the containment
2	Flooding of the lower drywell
3	Shielding of penetrations in the lower drywell
4	Containment overpressurization protection
5	Containment venting
6	Filter

Figure 1 Severe accident mitigation systems at TVO I and II

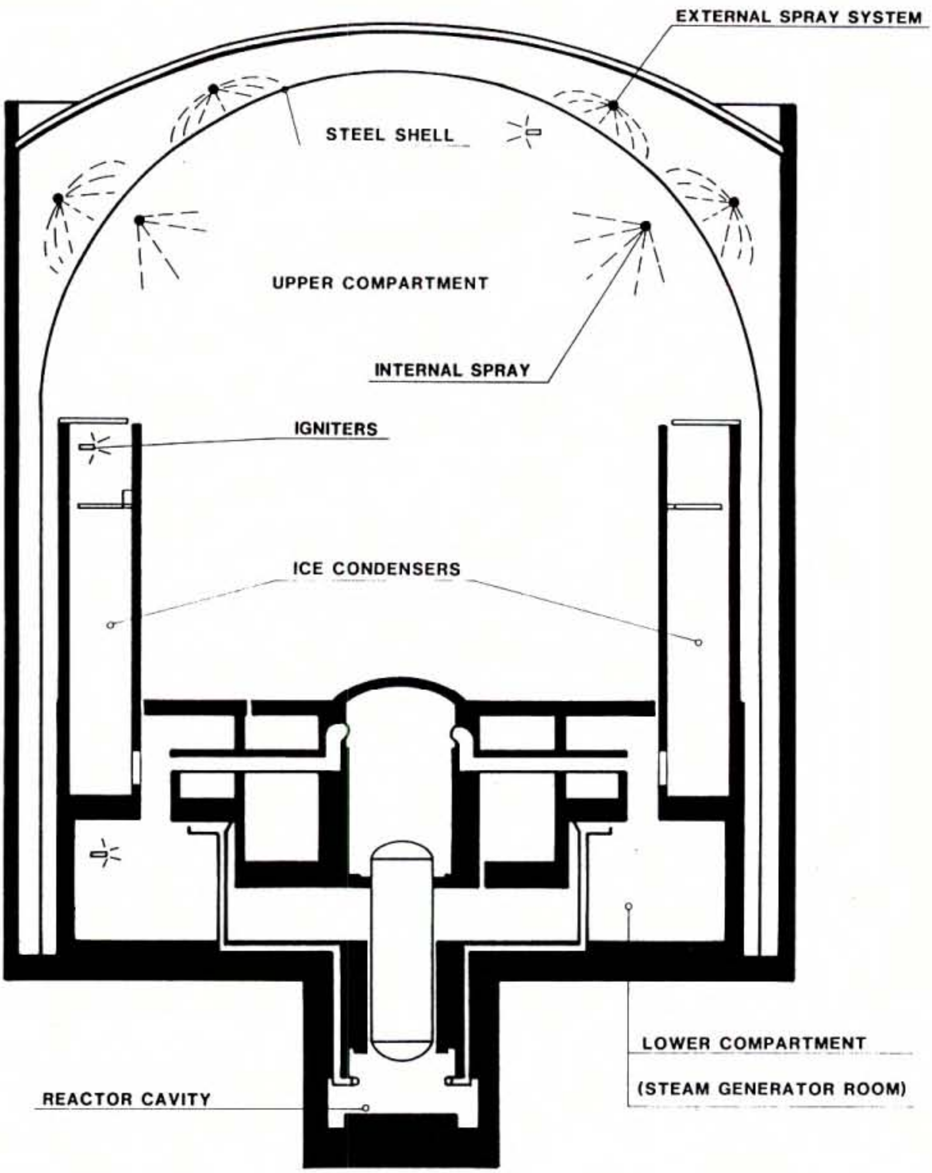
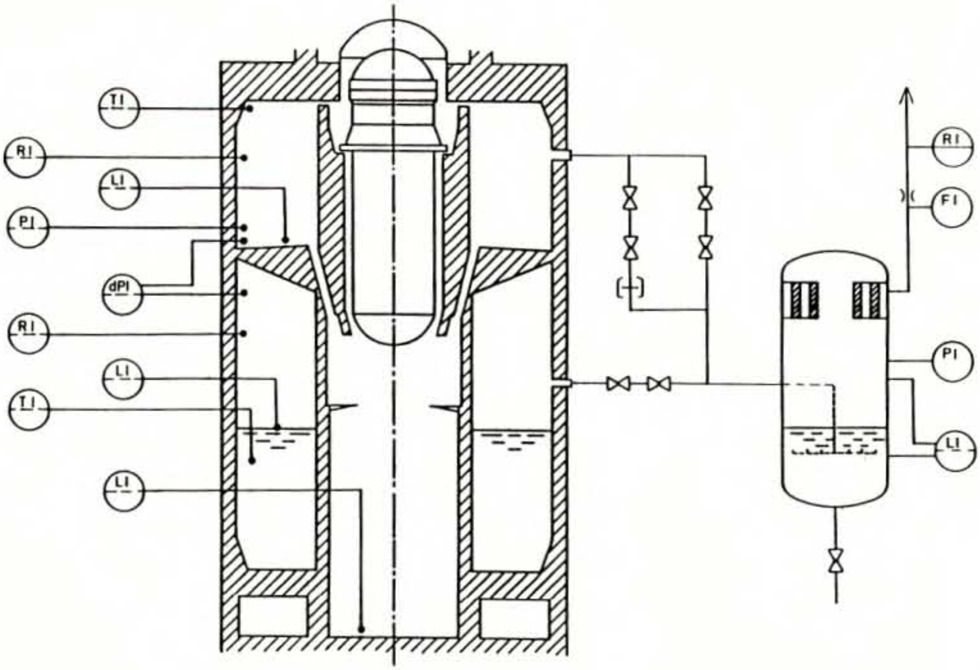


Figure 2 Containment systems of Loviisa 1 and 2



TI	temperature
LI	water level
PI	pressure
dPI	differential pressure
RI	dose rate
FI	flow

Figure 3 SAM instrumentation of TVO I and II

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SUMMARY of SESSION III

This Session discussed the unconventional use and further development of plant instrumentation, which can contribute to increase the diversity and flexibility of accident data available.

The first paper is presented by **Dr. Richard Oehlberg** (EPRI, USA) and **Mr. Revis James** (ERIN Engineering and Research, USA). It highlighted that other information sources can help to augment and confirm data available from dedicated accident instrumentation such as Reg. Guide 1.97 Instrumentation: inferences of plant status are possible from measurements and measurement trends obtained from instruments not expected to function, observations of system or component operability/inoperability, and observations of locally harsh environmental conditions. Detailed plant-specific examples are given, e.g. regarding the reactor pressure and level indication in BWRs, or the reactor cavity temperature indication on Westinghouse-type PWRs which the authors speculate may yield information related to vessel and core temperature. The authors advocate that others look at their information sources in a creative way.

The second paper is presented by **Mr. Jörg Pauls** (TH Zittau, Germany). It describes a new water level measuring method in vessels, based on the utilization of a combination of exterior gamma detectors, which measure the internal activity of N^{16}, generated by a (n,p) reaction on the O^{16} in the primary water. Two applications are given, one on the water level indication in a steam generator with U-tubes, the other on the water level indication in BWRs, with the corresponding calculation methods. Such a method, which appears to be valid during power operation or shortly after reactor shut-down, will be further qualified on a test reactor before being proposed on power reactors.

The third paper, presented by **Mr. Bernd Eckardt** (Siemens/UB KWU, Germany), discussed methods developed in Germany for measurements of hydrogen concentration and airborne nuclide concentration in the containment atmosphere, which are due to provide additional information on the accident history, plant status and effects of countermeasures to reduce hydrogen concentration, and also to help assessing the potential hazard in the vicinity of the plant. In-site hydrogen measurements as well as measurement through extractive sampling are discussed, the former system, using sensors in the containment, providing continuous and simultaneous information displayed in the control room without radiation exposure. Regarding containment atmosphere activity monitoring, the difficulties of extraction pipe systems are described and some preference is given to in-containment sampling and to pool sampling system.

The fourth paper, presented by **Dr. Horst Michael Prasser** (Z.K. Rossendorf, Germany), describes the use of needle-shaped conductivity probes for two-phase flow pattern determination during simulated emergency core cooling hot leg injection experiments at the COCO facility in the HDR containment. The first results appear promising and the use of such probes as additional instrumentation can be envisaged in the future on power reactors, e.g. for the control of water level, once some improvements have been achieved, in particular regarding the stability of the probe.

The fifth paper, presented by **Harri Tuomisto** (IVO, Finland), focuses on plant-specific severe accident management instrumentation that has been or will be added to the Finnish NPPs TVO I and II, Loviisa I and II, as a consequence of the severe accident management policy adopted, and of the resulting safety functions to be ensured.

A distinction is made between the instrumentation that is of crucial importance for performing a correct management measure and the instrumentation needed for monitoring the success. New instrumentation is strictly limited to those ensuring the safety functions.

The sixth paper, presented by **Professor Kurt Becker** (RIT, Sweden), describes the performance studies of a new core cooling monitor for BWRs. Such a detector has been successfully tested at various elevations, including in the lower plenum, in the Barsebäck nuclear power plant under normal operating conditions, and also in various environments in a 160 bar loop (with sudden uncoveries) and in the laboratory (up to 1265 °C). It can be operated in two modes: the core cooling mode and the temperature mode, where it actually acts as a thermometer. It currently appears ready for implementation in BWR installations.

SESSION IV

Operational Aids and Artificial Intelligence

Chairman: L. FELKEL

PERFORMANCE STUDIES OF A NEW CORE COOLING MONITOR IN A BOILING WATER REACTOR

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SUMMARY

Performance studies of a new type of core cooling monitors have been carried out in the Barsebäck Nuclear Power Station during the operation periods 1988-10-04 to 1989-07-05, 1989-08-03 to 1990-09-05 and 1990-09-28 to 1991-07-04. The results showed that the monitors, which were placed inside the reactor core, are very sensitive to variations of the reactor operating conditions, and that 34 months of irradiation did not influence the signals from the monitors.

Experiments were also carried out in a 160 bar loop, where sudden uncoveries of the monitors were achieved by decreasing the liquid level of the coolant surrounding the monitors. The experiments included the pressures of 5, 20, 50, 70 and 155 bar, and the responses to uncovery were in the ranges between 11 and 82 mV/sec or a total step change of 2 V at typical BWR conditions. This is of the order of two decades higher than the responses from monitors based on thermocouple readings.

The monitors can be operated in two modes, the core cooling mode and the temperature mode. In the former mode the electrical current is 3-4 A, and in the latter mode, where the monitor actually serves as a thermometer, the current is in the order of 50-100 mA.

In the laboratory the monitors have been studied for temperatures up to 1265 °C, which is very useful in case of a severe reactor accident. Thus, during such events the temperatures in the reactor core could be followed up to this level and the monitors could also be used to activate certain safety equipment.

The function as well as the design of the instrument is verified in laboratory experiments, computer calculations and reactor tests and is now ready for implementation in the BWR instrumentation.

1.0 INTRODUCTION

The need for reliable core cooling monitors became evident during the Three Mile Island accident, and it has been proposed that perhaps the destruction of the reactor core could have been avoided if the reactor had been equipped with monitors, which had given direct information about the cooling conditions inside the reactor core. In several countries the Nuclear Power Inspectorates have prescribed that core cooling monitors (CCM) should be installed in LWR's.

Such instruments must be able to indicate rapidly and clearly when the cooling of the reactor core is lost, which would occur if the liquid level in the reactor vessel falls below the level, where the reactor core starts to uncover. Especially, it is of great importance that the response of the monitor is sufficiently large in order to avoid any misinterpretations, which could lead to undesirable operator actions.

In most boiling water reactors the liquid level is today determined by means of differential pressure drop measurements in the downcomer. The instrument readings from this kind of measurements may, however, be difficult to interpret during the transient conditions, which may exist during an accident situation. Hence, computers must be used for the interpretation of the signals, preventing a direct and reliable observation of the coolant inventory in the reactor vessel. It would therefore be desirable to install additional instrumentation, which provides information about the cooling conditions inside the reactor core, and several proposals for such instrumentation are available in the literature. The most important of these systems have been discussed in a report by Anderson (1).

The concept proposed by Neuschaefer (2) and chosen by Combustion Engineering was based on a heated junction thermocouple (HJTC) probe. The probe contains a dual thermocouple, with one heated junction and one unheated junction. Thus, the output from the probe is directly related to the cooling capacity of the surrounding fluid. This principle suffers from the disadvantage that the output signal is in the order of a few millivolts only. For that reason the signal could easily be disturbed by other phenomena occurring during a reactor transient. A number of such probes are placed inside a splash shield, providing for coolability measurements at different levels inside the reactor core.

Scandpower (3) has developed an instrument called Radcal Gamma Thermometer, RGT. This instrument is in fact based on the same principle as the HJTC probe developed by Combustion Engineering. Inside the instrument there is an electrical heater, and the heat transfer conditions on the surface of the probe are monitored by means of thermocouples. However, the details of the RGT probe differ significantly from the CE design, but in both cases the response of the probes is in the order of mV only. Other proposals for this kind of instrumentation are found in references 4, 5, 6 and 7.

Becker (8,9) proposed and studied experimentally in a 160 bar loop the performance of a CCM, which consisted of an electrical cylindrical heater, and which had the same diameter as the fuel rods. When the monitor was uncovered the heat transfer coefficient decreased drastically, causing a rapid increase of the monitor temperature. The responses in terms of the voltage over the heater were in the order of 30-50 mV/sec and totally up to 3-4 volts. Figure 1 shows an example of the response obtained for a pressure of 70 bar and a constant current through the heater of 3.6 A. In comparison with the previously mentioned and available systems, the response of the monitor in case of a loss of coolant is therefore at least 150 times higher. The monitor can also be operated in a second mode called the temperature mode, where it is used as a thermometer, detecting the temperature inside the reactor core. This is, indeed, of great importance in case the reactor core has been uncovered and starts to heat up. Then the the signal from the detector can be used to initiate the operation of certain safety equipment and thus, it may play an important part in the accident management.

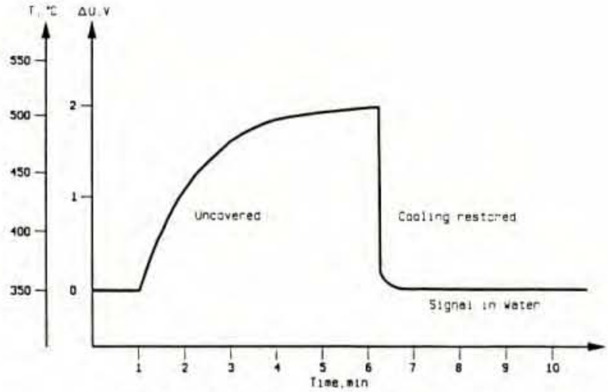

FIG. 1 RESPONSE TO UNCOVERY AND REWETTING OF DETECTOR

Thus, the proposed monitor provides great advantages in comparison with other available systems. In particular the following items should be considered:

1. For a loss of coolant the response of the monitor would be at least 150 times larger compared with the signals attainable from monitors, which are based on the heated thermocouple junction method. Because of the large response the risk for misinterpretations of the signals can be regarded as negligible.

2. The instrument can during a severe accident be used for measuring the temperature inside the reactor core up to 1265 °C

3. The instrument, which is constructed from very few parts, is simple and robust and thus reliable in operation. Especially the probe has been designed to be contained within the instrumentation guide tubes, already existing in the reactor.

4. If a probe is arranged within the reactor core the instrument may serve as a core cooling monitor, a liquid level indicator or as a thermometer. If, however, the probe is placed above, below or beside the reactor core the instrument functions as a liquid level indicator or as a thermometer.

5. The probe does not need any shielding, since the output signal is so strong, that the influences from water droplets hitting the probe are negligible. Heated junction thermocouple probes are often shielded against such disturbances, especially against the splashing, which occurs just above the surface of a two phase steam water mixture.

Because of the promising performance, which was revealed by the loop studies, the Swedish Nuclear Power industry decided to study the behaviour of the proposed system in a BWR. A BWR was chosen because the monitors could easily, as shown in figure 2, be mounted by means of guide tubes of the same type, which have already been developed for neutron flux measurements. It should be emphasized that in all the Swedish BWR's empty positions for guide tube penetrations through the reactor vessel boundary are available for additional in core instrumentation. Inside one guide tube 4 detectors may easily be placed at different vertical positions and in this manner detect the water level in the reactor vessel. It was considered to be of special importance to investigate the influence of long time neutron irradiation upon the response of the instrument.

The measurements were carried out in the Barsebäck I nuclear power plant, and the purpose of this paper is to present some of the experimental results, which were obtained in the 160 bar loop and in the Barsebäck I reactor. A more comprehensive review of the results is found in a project report by Becker et al (10). Although the present reactor studies have only been performed in a BWR, we would like to point out that the proposed monitor just as well can be used for liquid level and coolability measurements inside the pressure vessel of a PWR.

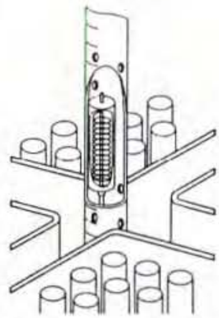

FIG 2. DETECTOR IN GUIDE TUBE BETWEEN FUEL BOXES

2.0 DESCRIPTION OF THE NEW MONITOR

The monitor consists of an electrical heating coil placed inside a 12.0 mm outer diameter Inconel cylinder. The diameter of 12 mm was originally chosen in order to simulate a fuel rod for a BWR. When the probe is submerged in water the cooling is rather efficient and the temperature of the probe will differ only slightly from the water temperature. However, if the liquid is lost and the probe is surrounded by steam, the temperature of the probe increases rapidly and will finally approach a much higher temperature level. This temperature can be chosen in advance by selecting a suitable power input to the heater. The temperature of the probe is determined on the basis of the electrical resistance of the heater coil, which is obtained by measuring the voltage over and the current through the resistor. This makes it possible to exclude the thermocouples, which were important parts of the previously mentioned HJTC and RGT probes.

The instrument is designed to operate in the following two modes; the core cooling mode and the temperature mode.

1. The core cooling mode, where an electrical current of 3-4 A is applied, is the normal operational mode. In the application in Barsebäck a DC current of 3.6 A was used, yielding a resistance wire temperature of approximately 380 °C at full reactor power.

2. In the temperature mode a current of 100 mA is supplied and the instrument is now operating as a resistance thermometer. This mode should be used during a severe reactor accident after the operation in the core cooling mode has established that the reactor core is uncovered and in a state of heatup.

The dual mode of operation of the monitor requires 2 separate power supply units as indicated in figure 3, which shows the electrical circuit for one detector.

The monitor, which is shown in figure 4, consists of a 64 mm long cylindrical heating element of 12 mm diameter. The heated section of the cylinder is ~40 mm long. The heater coil is a ~2.35 m long Thermo-Kanthal N Wire of 0.5 mm diameter. Kanthal N consists of approximately 97% Ni, 2,5% Si and small amounts of Mn and Co. The electrical resistance of the wire is 1.12 Ohm/m at 20 °C. The wire was wound around a ceramic cylinder, which was placed inside the Inconel tube. All the welding was performed with electron beam welding in vacuum. The space between the wire and the canning was filled in vacuum with MgO powder, packed with an efficiency greater than 98% of solid material. MgO was selected because it has the best high-temperature insulation capability.

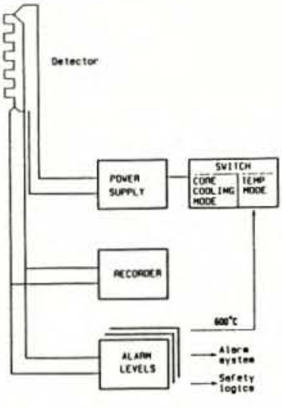

FIG. 3 INSTRUMENTATION

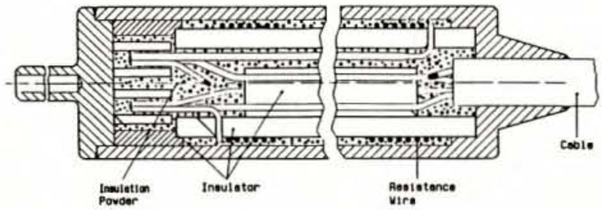

FIG. 4 DETECTOR

The change of the electrical resistivity with temperature for the Kanthal wire is shown in figure 5. The diagram, which is used for the determination of the wire temperature, was obtained by calibration in an electrical furnace.

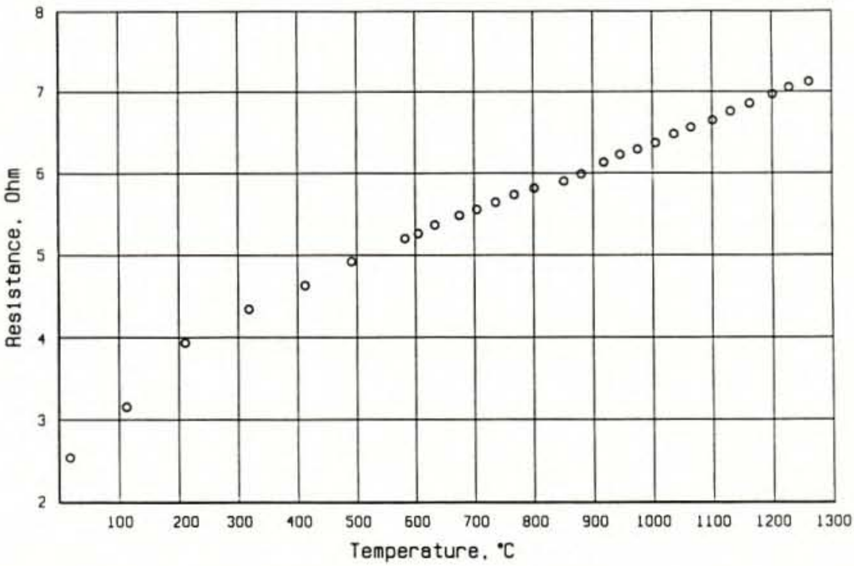

FIG. 5 RESISTANCE OF KANTHAL N WIRE

For the present study 11 monitors were manufactured, 8 for the Barsebäck experiments, No 11-18, and 3 for the laboratory tests, No 19, 20 and 21. Because Cu melts at 1083 °C and because it would be desirable during a severe reactor accident to measure the temperature in the reactor core up to at least 1260 °C, which is recommended in the US Regulatory Guide 1.97 (11), three of the detectors were equipped with Pt leads and three with Ni leads in accordance with the table below.

Assembly	*Detector*	*Cable length* *mm*	*Cable Material*
1	11	6203	Cu
	12	13305	Pt
	13	14133	Ni
	14	14487	Cu
2	15	6203	Cu
	16	13305	Pt
	17	14133	Ni
	18	14487	Cu
Laboratory Tests	19	2000	Cu
	20	2000	Pt
	21	2000	Ni

The four 0.5 mm leads, two for power supply and two for voltage measurements, were contained inside an Inconel 600 tube with an outside diameter of 3.6 mm. MgO was used as electrical insulation between the leads and the Inconel cable. The choice of Pt and in particular Ni has a disadvantage because the electrical resistances of Ni and Pt are high in comparison with Cu. This causes relatively high power losses in the rather long cables, and more expensive power supply equipment is needed. Four lead cables were used in order to eliminate the influence of the lead temperatures on the detector signals. This is, indeed, important during a large LOCA, where the temperature decrease in the reactor vessel for BWR's and PWR's may be up to ~130 °C respectively ~170 °C. Considering that the cables are up to 14 meter long, their ohmic resistance decrease would certainly disturb the interpretation of the detector signals during this type of accidents.

The Barsebäck detectors with their cables were mounted inside a 16.9 mm inner diameter guide tubes of the same type as for PRM measurements. The end of the cables were sealed and the four leads were connected to four exterior Kapton insulated copper wires. The detectors and the connectors were manufactured at the Paul Scherrer Institute in Switzerland.

The electrical resistances of all the heater coils and the MgO insulations as functions of temperature were measured at the Paul Scherrer Institute in the range between 24 °C and 599 °C and for the laboratory detectors in our laboratory in the temperature range between 20 °C and 600 °C. After the completion of the investigation additional measurements of the resistances were obtained up to 1265 °C for detector 19 and 21. The electrical circuit of the former detector broke down at 1170 °C, which was not surprising considering that the melting point of Cu is 1083 °C. The latter detector, however, functioned normally after this severe treatment. The three sets of data gave of course identical values and figure 5 showed the ohmic resistance versus temperature for the Kanthal wire in detector 21. This figure is used for temperature measurements when the detector is operated in the temperature mode. With regard to the other resistance measurements the reader is referred to the project report by Becker et al (10).

3.0 EXPERIMENTAL LOOP STUDIES

A great deal of measurements were carried out in a 160 bar loop at the Royal Institute of Technology. The research program for the 160 bar loop included the following items:

1. Liquid level or loss of coolability measurements.
2. Blowdown tests.
3. Influence of guide tubes.
4. Temperature cycling tests.

In addition, the detectors were studied in an electrical furnace at temperatures up to 1265 °C.

3.1 APPARATUS AND EXPERIMENTAL PROCEDURES

The apparatus, which was made of stainless steel, was designed for an operating pressure of 160 bar. This allows to study the performance of the detectors for BWR as well as for PWR conditions. Figure 6 shows the apparatus, which is an U-tube, where one of the legs is the test vessel and the other leg is a pressurizer. The connection between the two vessels is a 10 mm inner diameter tube supplied with a valve. The pressure in the pressurizer was controlled by means of nitrogen from a bottle and a relief valve to the ambient. The pressure in the test vessel was adjusted by a 500 mm long electrically heated 8-rod bundle, which was mounted on the bottom flange and an air cooled condenser. The test vessel consisted of a lower 615 mm long cylinder with an inner diameter of 70 mm and a 785 mm long upper section of 112 mm inside diameter. The two cylinders were connected by means of a 155 mm long cone. The upper cylinder was connected to the head of the vessel by means of a 155 mm long cone and a 115 mm long flange. Thus, the total height of the vessel became 1825 mm.

In the present study three detectors (19, 20 and 21) were mounted with their lower edge 1 cm above the upper end of the rod bundle. The performance of the detectors were identical, as the only difference between them was the material of the 4-lead cables, and this does not influence the signals.

The power to the monitors were supplied from three direct current supply units, which delivered constant current at any selected value between 0 and 20 A. The maximum voltage to be obtained from the unit was 100 volts. When the heatup of a monitor started after uncovery, causing the electrical resistance of the heater coil to increase, the voltage over the monitor increased asymptotically towards the equilibrium value for natural convection steam cooling. The transient voltages over the monitor were recorded by means of a 4 channel W+W recorder. One of the channels was used for recording the static pressure in the test vessel. The liquid level inside the test vessel was measured by means of a Barton cell. Measurements were also made of the liquid and vapor temperatures employing thermocouples, and of the pressure in the vapor phase using a pressure transducer.

At the onset of an experiment the liquid surface in the test vessel was adjusted to a level of a few cm above the upper edge of the monitors. Due to the expansion of the steam during the uncovery of the detectors the initial pressure in the test vessel was chosen somewhat above the nominal pressure. For example, at 70 bar the initial pressure was chosen 71 bar in the test vessel and 65-67 bar in the pressurizer. The monitors were uncovered approximately 2-5 seconds after opening the valve and then the valve was closed. The liquid level in the test vessel after closure was 1-2 cm below the monitors. As the cooling of the monitors deteriorated drastically their temperature increased rapidly yielding a strong response. During this phase of the experiment the pressure of the pressurizer was increased to a level above the pressure in the test vessel. This was achieved by supplying gas from the high pressure nitrogen bottle. When steady state had been obtained or when the center temperatures of the monitors approached 600 °C, the valve between the pressurizer and the test vessel was again opened, causing the water to return to the test vessel and quench the monitors, which cooled off rather rapidly to the initial conditions.

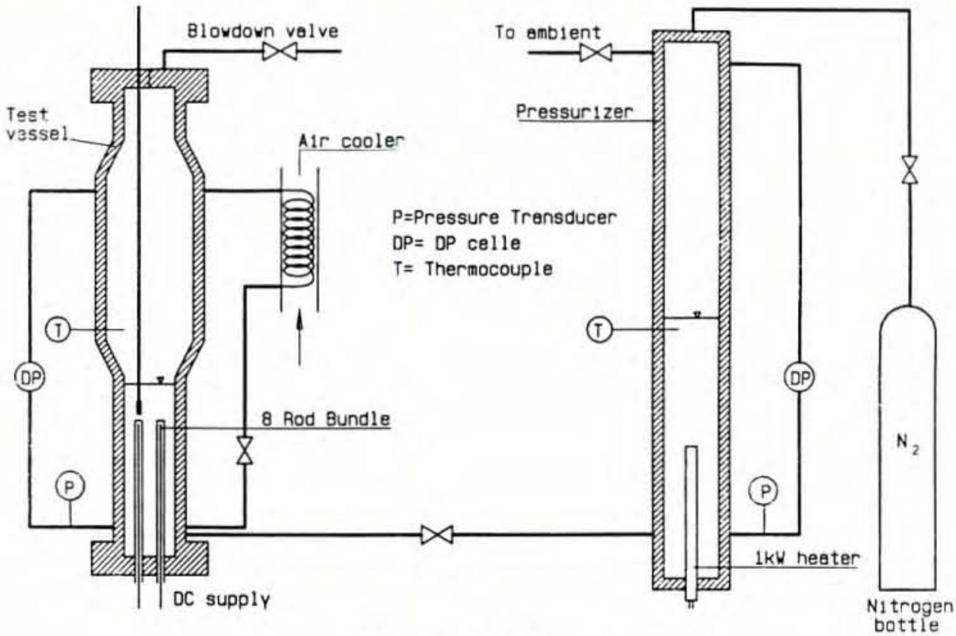

FIG. 6 APPARATUS

3.2 LIQUID LEVEL AND COOLABILITY MEASUREMENTS

In order to cover the conditions, which may be encountered during accidents in BWR's as well as PWR's, liquid level or loss of coolant experiments were carried out at the nominal pressures of 5, 10, 70 and 155 bar.

At each pressure the uncovery of the monitors was carried out at the electrical currents of 3.2, 3.6 and 4.0 A, which correspond to power inputs in the range between 40 and 75 W or average surface heat fluxes between 2.1 and 4.0 W/cm^2. The results for monitor 19 at 70 bar are shown in figure 7. Considering the current of 3.6 A, which is employed in the monitors being tested in the Barsebäck I boiling water reactor, the average response for the first 30 seconds after uncovery is 25.5 mV/sec. This is, indeed, a very strong signal, and in order to obtain a similar response with thermocouples a temperature transient of 640 °C/sec would be required. The total response was 1.85 V, which is 150 times larger than the voltage, which would have been obtained from detectors based on thermocouples. Misinterpretations of the signals is therefore very unlikely.

The large response is not only a result of the drastic decrease of the heat transfer coefficient after uncovery, but also of the increase of the power, $Q = RI^2$, during the transient when the ohmic resistance, R, increases and the current, I, is kept constant. Thus,

$$\Delta V \sim \Delta q''/\Delta\alpha$$

where q" is the surface heat flux of the monitor and α is the heat transfer coefficient.

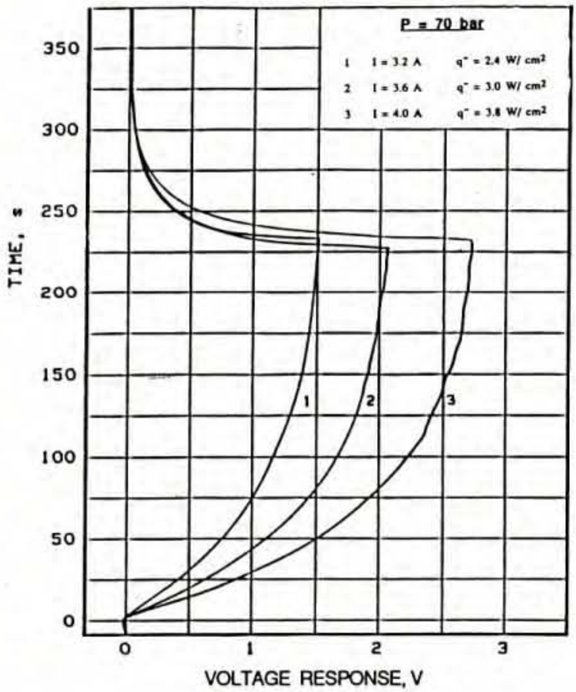

FIG. 7 INFLUENCE OF HEAT FLUX ON DETECTOR RESPONSE

3.3 BLOWDOWN MEASUREMENTS

Blowdown measurements were performed at the initial pressures of 70 and 155 bar. During these tests the uncovery of the monitors was achieved by opening the valve on the 4 mm inner diameter duct to the ambient, which caused a loss of coolant and depressurization of the test vessel. At both initial pressures the currents of 3.2, 3.6 and 4.0 A were employed. All of the three monitors were tested simultaneously and identical results were obtained. The response of monitor 19 at 70 bar and 4.0 A is shown in figure 8 as a plot of voltage versus time.

At the time t = 0 the valve was opened and the pressure in the vessel starts to fall from the initial value of 70 bar. Due to the decrease of the pressure the water temperature as well as the monitor temperature decreases, yielding a negative response. At point A the liquid level has fallen below the monitor and due to the decrease of the cooling capacity of the surrounding fluid, the temperature of the monitor increases rapidly as well as the voltage of the resistance wire. Despite the large temperature decrease from 284 °C to ~150 °C during the depressurization, the voltage over the instrument becomes larger than the initial steady state voltage less than 1 minute after uncovery. Then the voltage continues to increase and reaches 2 volts after another minute. Thus, reliable and unambiguous information can be provided for the reactor operating personnel. With regard to reactor applications the time between uncovery of the monitor and positive voltage output is negligible in comparison with the time needed to heat the reactor core to an undesirable level. At point B in figure 8 the LOCA or blowdown is terminated by closing the valve to the ambient and opening the valve to the pressurizer, which allows for injection of water into the test vessel. Then the monitors are quenched and rapidly cooled off.

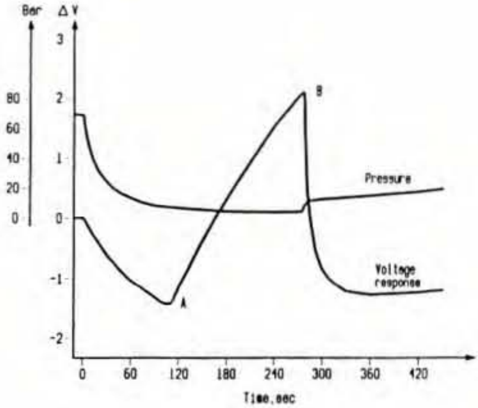

FIG. 8 BLOWDOWN MEASUREMENTS

3.4 INFLUENCE OF GUIDE TUBE

For BWR applications the monitors are mounted in 16.9 mm inner diameter guide tubes. In order to ensure that the readings are representative for the conditions in the by-pass channel, four 5 mm diameter holes, 90° apart, were drilled 1 cm above the upper edge as well as 1 cm below the lower edge of the detectors.

The influence of the guide tube was studied experimentally in the 160 bar loop by placing two monitors at the same level in the test vessel; one inside a guide tube and one outside the guide tube.

Loss of coolability measurements were carried out at the pressures of 5.4, 70 and 140 bar and the currents of 3.2, 4.0 and 4.8 A, thus investigating a total of 9 conditions. In all cases almost identical results were obtained for the two monitors, indicating that the influence of the guide tube on the transient response is negligible. In particular, it was important to observe that the heatup of the two monitors started at the same instant, showing that the guide tube do not delay the dryout and that the liquid level inside the guide follows the outside level during the whole transient.

During reactor operation at 70 bar one detector in the core cooling mode uses an electrical power ~50 W. Because of γ-ray absorption an additional ~40 W is generated in the detector. Further, because heat is also generated by γ-ray absorption in the cables and because three monitors may be present in one guide tube in the reactor core region it was suggested that perhaps the guide tubes could be filled with steam, causing the monitors falsely to indicate a dry reactor core. In order to resolve this problem two series of tests were performed, mounting one monitor inside the guide tube and one monitor outside the guide tube. Two test series were carried out, one for central mounting and one for eccentric mounting in the guide tube. The experiments were carried out in saturated water at 70 bar. The power of the monitor was gradually increased from 35 to 265 W. Almost identical results were obtained for the shielded and the unshielded monitors, demonstrating that placing a monitor inside a guide tube is not a source of error.

It should also be recognized that the loop experiments were conservative because:

1. The tests were carried out at saturated water, while the water in the by-pass channel in a BWR is ~10 °C subcooled.
2. Powers up to 265 W was used, while the monitor power in a BWR is less than 100 W including the γ-heat.

3. The flow area of the two 3 mm diameter discharge holes in the test is small in comparison with the four 5 mm diameter holes used in the BWR application.

3.5 TEMPERATURE CYCLING OF MONITORS

It is important that the detectors can operate during transient conditions without being damaged. In order to prove their reliability in this respect the following temperature cycling program was carried out:

1. Before the detectors were delivered from the Paul Scherrer Institute, two prototypes were moved into a fluidized bed of ZrO_2, which was kept at a temperature of 650 °C. After heatup of the detectors to this temperature, the detectors were withdrawn from the bed and cooled to room temperature. This procedure was repeated 600 times.

2. In the 160 bar loop the detectors 19, 20 and 21 were uncovered and rewetted 150 times at 5 bar pressure. The temperature of the detectors varied during each cycle between 235 °C and 580 °C. The transient performance of the three detectors were identical.

No sign of any damage was observed after this rather severe treatment, indicating an excellent reliability of the instruments.

3.6 HEATUP TEST IN ELECTRICAL FURNACE

As previously mentioned monitors 19 (Cu leads) and 21 (Ni leads) were studied in a furnace up to a temperature of 1265 °C. The heatup was slow and lasted for 10 hours. Then the temperature was kept at this level for three hours before the furnace was shut off. The temperature of 1265 °C was chosen, because the US Regulatory Guide 1.97 (11) prescribes that certain instrumentation must function at this temperature in case of a severe accident.

The monitor 19 with Cu leads broke down at 1170 °C, which is far above the melting point of Cu, which is 1083 °C. It is interesting to notice that the MgO insulation kept the molten copper together and, thus, ensured the integrity of the electrical circuit up to 1170 °C. The monitor 21 with Ni leads, however, survived this rather severe treatment without any visible signs of damage. Further, uncovery tests, which were carried out in the 160 bar loop after the heatup tests, did not indicate any change of performance of monitor 21.

On the basis of the heatup tests and the temperature cycling tests we conclude that the monitors would operate satisfactory during reactor transients as well as during severe reactor accidents.

4.0 NUCLEAR REACTOR STUDIES

The major purpose of the BWR studies was to observe if the ohmic resistance of the monitors and hence the signals would be stable over a long period of neutron irradiation. The Barsebäck nuclear power station was chosen for the studies. Two assemblies A13 and C11, each containing four monitors were installed in the Barsebäck I reactor in September 1988. The reactor started to operate 1988-10-04 and the performance of the monitors was observed during the whole operational season, which ended with a planned shutdown 1989-07-05. During this shutdown the assembly C11 was removed because the Inconel sheath of the cable supplying power to the upper monitor 18 was damaged by wear between the cable and a support pin. For the assembly A13, however, measurements were also taken throughout a second operating period starting 1989-08-03 and ending 1990-09-15 as well as during a third period starting 1990-09-28 and ending 1991-07-04. Thus, this assembly received a total of 1032 days of full power irradiation. During these periods two additional cables were damaged by wear between cables and support pins. The damage of the cables is further discussed in paragraph 4.2.

The two assemblies were mounted in the corner between 4 fuel elements as previously shown in figure 2. The location of the two assemblies is given in figure 9. Three of the monitors in each assembly were placed at levels 0.5 m 1.0 m and 1.9 m below the upper edge of the reactor core. The fourth monitor of the assemblies was located at the bottom of the reactor vessel. The purpose of this location is based on the plans to use the monitors as guidance in the management of severe reactor accidents involving core meltdowns and where a granulate bed, consisting of core debris, has been formed on the bottom of the reactor vessel. When these monitors indicate dryout melt-through of the reactor vessel can be expected within a few minutes.

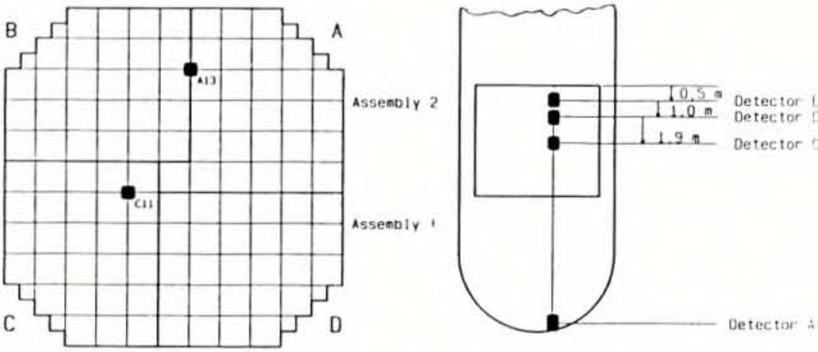

FIG. 9 DETECTOR POSITIONS FOR BARSEBÄCK BWR STUDIES

The monitors were placed inside a standard instrumentation guide tube with an inner diameter of 16.9 mm, as shown in figure 10. At the positions of approximately 50 mm above and below the monitors four 5 mm diameter holes, 90° apart, were drilled through the guide tube wall. The purpose of these holes were to ensure an efficient communication between the water inside the guide tube and the water in the by-pass channel. In order to ensure the correct axial positions of the detectors in case of an accident with core heatup a pin through the guide tube was placed below each detector.

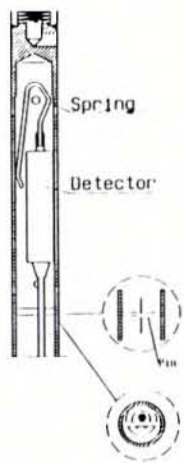

FIG. 10 ASSEMBLY WITH SUPPORT PIN

Each monitor has two current supply units, one for the high current level, which is used for measurements in the core cooling mode and one for the low current level, which is used for measurements in the temperature mode. The monitors with Pt and Ni leads obtained the power from four TCR-30056 Electronic Measurement units. The monitors with Cu leads obtained the power from four SB-60-5 Power Box laboratory units. The power supply units are automatically controlled at the selected current of 3.6 A in the core cooling mode and at 0.1 A in the temperature mode. The voltages over the monitors were continuously recorded by means of two BBC-SE-460, six channels, recorders. The instrumentation and the recorders were mounted in two panels, one for each of the two monitor assemblies, located in the rear of the control room.

Generally, the monitors were operated in the core cooling mode with an electrical power of ~55 W and a surface heat flux of ~3.0 W/cm^2. Since the heat fluxes of the monitors are high in comparison with the values, which would occur on the fuel rods after reactor shutdown, the monitors will in an accident situation react faster than the fuel rods, and, thus, be able to detect in an early stage any sign of overheating the reactor core. During normal operation at full reactor power, the γ-radiation induces an additional power of approximately 40 W in each of the monitors located inside the reactor core. In an accident situation with reactor shutdown, however, this power is rapidly reduced. The following measurements were carried out for each of the eight monitors at regular time intervals of 2-3 weeks.

1. Electrical resistance of the Kanthal wire in the core cooling mode.
2. Electrical resistance of the Kanthal wire in the temperature mode.
3. The electrical resistance of the Mg0 insulation.

The electrical power input to a monitor operated in the temperature mode is ~0.042 W corresponding to a heat flux of ~0.0022 W/cm^2. The low power ensures negligible heatup of the monitors, thus, making it possible to use the monitors to measure the temperature inside the reactor core after shutdown, when also the heatup due to γ-absorption is small.

4.1 EXPERIMENTAL RESULTS AND DISCUSSIONS

The recorder traces proved the high sensitivity of the monitors. Small changes of the reactor operating conditions were detected by the monitors. An interesting example of the recorder traces is given in figure 11, which shows the response of the monitors 11, 12, 13 and 14 to a reactor scram. The apparent differences between the signals from the monitors 12, 13 and 14 are due to different origins, which were selected in order to separate the curves. The output from monitor 13 was adjusted to give the correct Kanthal wire temperature with regard to the scale of the diagram. The signal from monitor 11, which was located at the bottom of the reactor vessel, where γ–radiation was negligible, indicates as expected the lowest value during steady operation of the reactor.

After reactor scram, the pressure decreased from 70 bar to ~17 bar. Simultaneously the temperatures of the monitors decreased due to the decrease of γ-absorption and the decrease of the saturation temperature of the water. It is interesting to notice that the response from monitor 11 is smaller than the signals from the other monitors. This behaviour is explained by the low γ-radiation, which occurs at the bottom of the reactor vessel. Also the increase of pressure, which follows the restart of the reactor at 15 hours into the transient, is detected by the monitors as the voltages over the monitors increase. At 23.5 hours the pressure has reached 70 bar. However, the reactor power is now 5 per cent of full power. This explains why the monitor signals are somewhat low in comparison with the initial values. Later on, when the reactor operates at full power, the signals are identical to the initial signals.

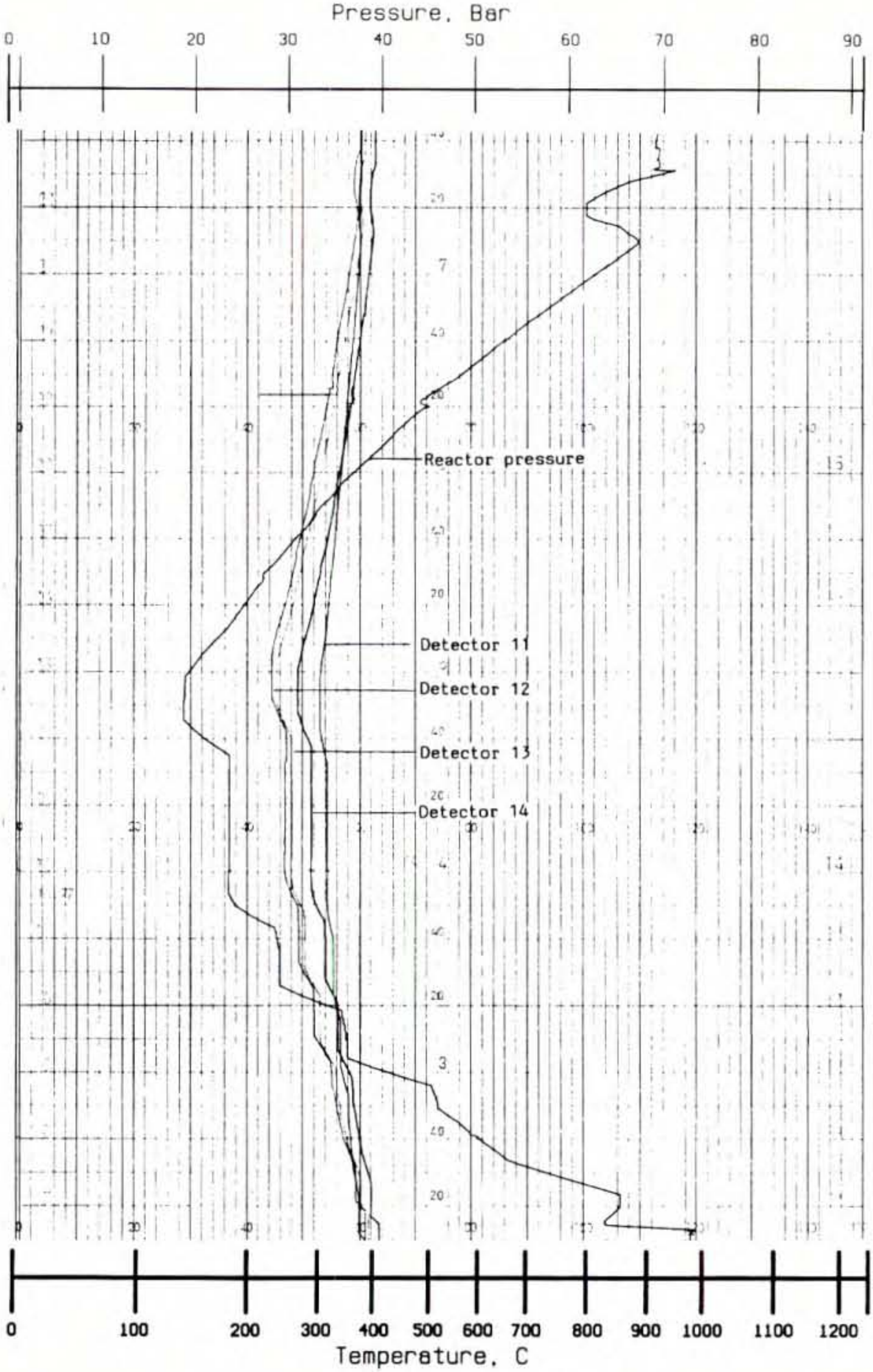

Fig.11 Response to Reactor Scram

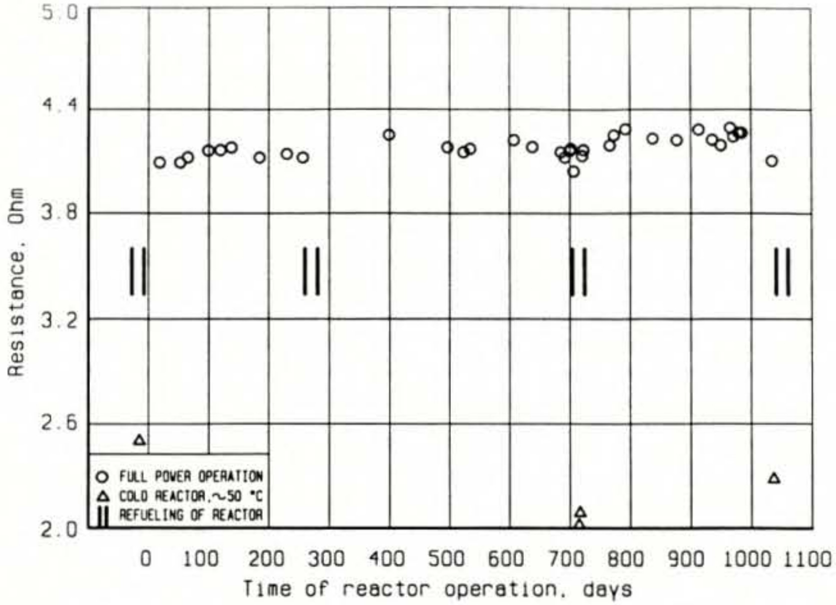

Fig. 12 Ohmic Resistance of Monitor 12. Temperature mode

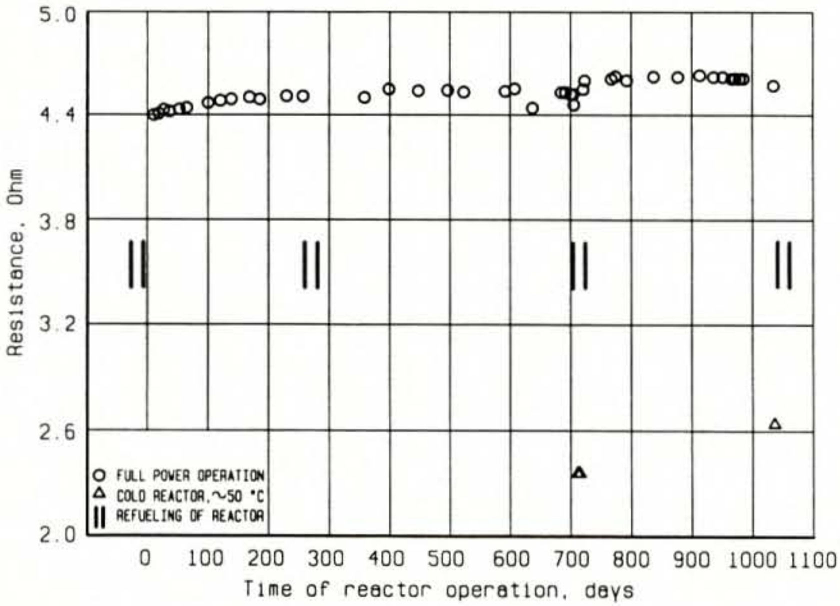

Fig. 13 Ohmic Resistance of Monitor 12. Core Cooling mode

From the current and the voltage measurements the electrical resistances of the Kanthal wires were calculated and plotted versus the time of irradiation. Examples of the results for operation in the core cooling mode as well as in the temperature mode are shown in figures 12 and 13. The origin for the time scale is chosen at the start of full power operation of the reactor. One observes that the performance of the monitors did not indicate any significant sign of ageing. The scatter of the data is probably caused by local time dependent variations of the power and γ-radiation. One observes that the ohmic resistances of the Kanthal wire in the core cooling mode is high in comparison with the values obtained in the temperature mode. This difference is explained by the heatup caused by the electrical power employed in the core cooling mode. The temperatures of the Kanthal wires can be determined from the calibration curve previously shown in figure 5. For monitor 12 they were on the average 285 °C respectively 380 °C in the temperature mode respectively the core cooling mode.

The ohmic resistances of the MgO insulation at room temperature were 10^{10}-10^{11} Ω. During reactor operation the ohmic resistances were measured at ~280 °C and the values were then 10^{6}-10^{7} Ω. No significant ageing of the MgO insulation was observed during the reactor studies.

Despite the observed stability of the ohmic resistances of the Kanthal wires we found it desirable to address this problem in more detail. The detectors 16, 17 and 18 of assembly C11, which had been irradiated for 270 days at full reactor power, were therefore investigated in the hot cell laboratory at Studsvik Nuclear. This investigation involved ohmic resistance measurements in a furnace. All three detectors yielded identical results in the whole temperature range within a scatter of ±1.0 per cent. Figure 14 shows the results for detector 16, and a comparison with the initial resistances. The effects of the irradiation is rather interesting. Below 300 °C the ohmic resistance has decreased. Between 300 °C and 600 °C one observes an increase and above 600 °C no effect of irradiation was observed. The increase of ohmic resistance between 300 and 600 °C is explained by lattice defects caused by the neutrons. The decrease below 300 °C, however, is not understood. The measurements taken during the cooling of the Kanthal wires show that heating to 600 °C causes the ohmic resistance to be restored to the original values in the whole temperature range. The deviations are within the accuracy of the measurements. Thus, the lattice defects as well as all other material changes influencing the ohmic resistance disappear by the heat treatment. A more detailed discussion of the measurements is found in reference (10).

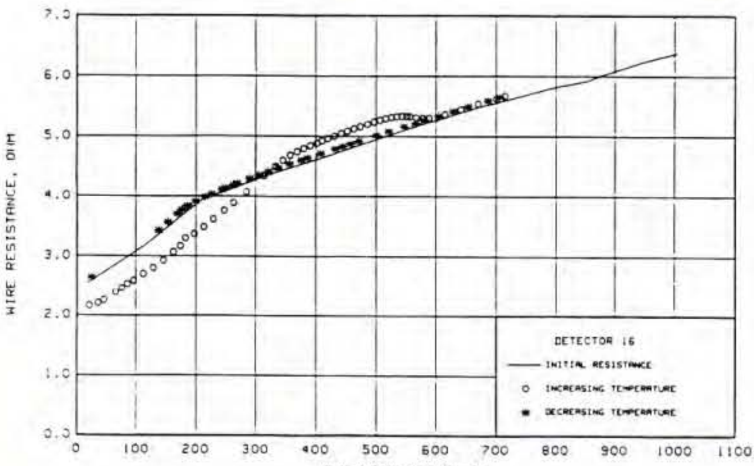

FIG. 14 EFFECT OF IRRADIATION ON KANTHAL WIRE RESISTANCE

It should be emphasized that the changes of ohmic resistance caused by irradiation have no significance for the operation of the detectors in the core cooling mode. However, if the detectors are operated in the temperature mode during an accident, an error of ±30 °C may be introduced with regard to temperature readings in the range between 300 °C and 600 °C. However, the consequences of this inaccuracy is rather small. Ohmic resistance measurements versus temperature were also obtained with the detector 12 during the reactor shutdown in July 1991. These measurements did not indicate any further effects in comparison with figure 14 or similar measurements taken during the reactor shut downs in May 1989 and in September 1990. Preparations for hot cell studies of detector 12 is now in progress.

4.2 DESIGN OF NEW ASSEMBLIES

During the present reactor studies it became evident that the monitor supporting pins must be removed in future applications. Except for the holes in the cables, which were caused by the pins, detailed examinations of the detectors did not reveal any other faults. The examinations included x-ray as well as neutron radiograph pictures of the detectors 15 and 18, and confirmed that the detectors were undamaged. We therefore believe that if proper mounting of the monitors is accomplished a life time of at least 5 years is possible. In this respect it should be mentioned that last year it was found by visual inspection that the wear between PRM guide tubes and fuel boxes were significantly higher in the Barsebäck 1 and Oskarshamn 2 reactors than in other Swedish BWR's, indicating a higher vibration level in the former reactors. This may explain the unexpected and early damage of the detector cables.

The revised assembly design is shown in figure 15, where also the design of the penetration of the cables through the pressure boundary is included. The assembly design satisfies a number of prescribed criteria and should ensure the desired lifetime of 5 years. The vibration durability of the new assembly was also analysed and it was concluded that the system would perform satisfactory even during an earthquake.

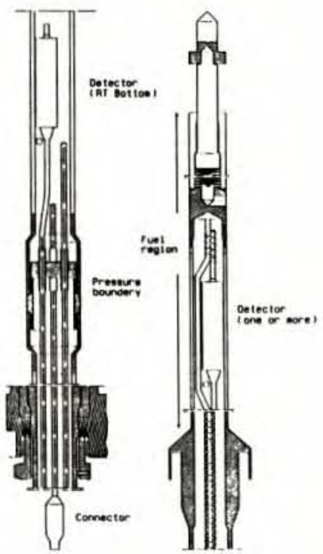

FIG. 15 NEW ASSEMBLY DESIGN

It has also been proposed to design a dual purpose assembly, which contains one core cooling monitor as well as neutron flux monitors.

4.3 ANALYSIS OF MONITOR BEHAVIOUR DURING POSTULATED ACCIDENTS

In the scheduled program for verification of the detector response in accident sequences, two simulations of accident progress and the monitor response was made with the ABB Atom GOBLIN program system. The calculations were carried out by Bredolt (12), and included the following two accident sequences:

1 Total loss of feed water, automatic depressurization of the reactor vessel at downcomer level 1.8 m below top of active fuel. Emergency Core Cooling System (ECCS) start was delayed until the hottest fuel assemblies cooling medium bulk temperature reached 900 °C.

2 Main steam line guillotine break immediately outside the vessel. ECCS delayed until the mean coolant temperature of the hottest fuel assemblies reached 900 °C.

The performed computer simulations of the behaviour in these accident sequences verifies that the monitor will give an early warning for loss of cooling of the fuel, and will also verify the restoration of the cooling. We therefore conclude that the monitor is a very useful tool in the understanding and the management of reactor accidents.

5.0 APPLICATIONS IN BWR

The proposed system consists of a robust detector, positioned inside the core and gives a large signal at a loss of coolant. It is therefore very useful for the personnel in the control room and also for increasing the reliability of automatic functions in the emergency core cooling system. Indication of detector temperature on a recorder or a Video Display Unit gives the operator direct information about the changes in the core cooling status during operation as well as during hot or cold shutdown.

5.1 INSTRUMENTATION

A general layout of the instrumentation was shown in figure 3. The detector is supplied with constant current from a power supply by means of two of the wires in a four- lead cable. The voltage over the detector is measured on the two other wires with an amplifier and the output is presented in centigrades on a recorder or a computers VDU. The detector voltage is also connected to trip units for alarm signals and automatic functions. The detector is supplied with 3.6 A in the core cooling mode and with 0.1 A in the temperature mode. For protecting the detector of being overheated one of the trip units will switch the monitor from core cooling mode to temperature mode if the temperature of the wire exceeds 6-700 °C.

5.2 REACTOR OPERATOR INFORMATION

In various situations the operator can get important information, for instance

1. At operation transients the core cooling monitor gives information about the core cooling situation independent of the reactor tank level instruments based on dP-measurements.
2. During transients the monitor can confirm the restoration of adequate core cooling.
3. At serious incidents the detector gives information about the core heat up.

4. In a severe reactor accident the monitor gives information about its progress by following the core temperature up to 1265 °C, and if a detector is placed at the reactor vessel bottom the heatup of the core debris can also be observed.

5. During reactor shutdown the monitors operating in the temperature mode give information about the cooling system status. In most reactors today the temperature is measured in the cooling system outside the reactor vessel, and one is therefore forced to rely on the functioning of that system.

5.3 SIGNALS TO AUTOMATIC FUNCTIONS

Automatic functions of certain safety systems can be improved by implementing signals for high rate of change of the detector temperature, for example:

1. The reliability of the logic for the opening of the blowdown valves can be improved by adding a two out of four detector logic.

2. The reliability of the logic for flooding a dry pedestal region below the reactor vessel can be improved by adding logics from the core cooling monitors.

6.0 CONCLUSIONS

1. The proposed monitor can operate in two modes; the core cooling mode and the temperature mode.

2. Laboratory studies have shown that the responses to uncovery are two decades higher than signals from monitors based on thermocouple readings.

3. No effects of splashing or other secondary phenomena were observed during the laboratory measurements.

4. The instrument can measure the temperature in the reactor core up to 1265 °C.

5. The guide tube, in which the monitors are mounted, is not interfering with the signals from the monitors.

6. Operation in the Barsebäck BWR has shown that the signals from the monitors have not changed significantly over an irradiation period of 34 months.

7. Provided satisfactory mounting in the guide tubes is accomplished, a life length of at least 5 years is possible.

8. Computer simulations with the GOBLIN code verified that the detector will give an early warning for loss of cooling and also verify restored cooling in the core.

9. Analysis of the vibration durability of the detector showed that the instrument will perform satisfactory during an earthquake.

10. The monitor is very useful for reactor core supervision and for the management of reactor accidents.

11. The function and design of the monitor is verified and ready for implementation in BWR instrumentations.

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ACCIDENT MANAGEMENT ADVISOR SYSTEM (AMAS): A Decision Aid for Interpreting Instrument Information and Managing Accident Conditions in Nuclear Power Plants

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Introduction

Accident management can be characterized as the optimized use of all available plant resources to stop or mitigate the progression of a nuclear power plant accident sequence which may otherwise result in reactor vessel and containment failure. It becomes important under conditions that have low probability of occurring. However, given that these conditions may lead to extremely severe financial consequences and public health effects, it is now recognized that it is important for the plant owners to develop realistic strategies and guidelines.

Recent studies have classified accident management strategies as:

- the use of alternative resources (i.e., air, water, power),
- the use of alternative equipment (i.e., pumps, water lines, generators),
- the use of alternative actions (i.e., manual depressurization and injection, "feed and bleed", etc.)

The matching of these alternative actions and resources to an actual plant condition represents a decision process affected by a high degree of uncertainty in several of its fundamental inputs. This uncertainty includes the expected accident progression phenomenology (e.g., the issue of high pressure core ejection from the vessel in a PWR plant with possible "direct containment heating"), as well as the expected availability and behavior of plant systems and of plant instrumentation.

To support the accident management decision process with computer-based decision aids, one needs to develop accident progression models that can be stored in a computer knowledge based and retrieved at will for comparison with actual plant conditions, so that these conditions can be recognized and dealt with accordingly. Recent Probabilistic Safety Assessments (PSAs) [1] show the progression of a severe accident through and beyond the core melt stages via multi-branch accident progression trees. Although these "accident tree models" were originally intended for accident probability assessment purposes, they do provide a basis of initial information for the development of models specifically tailored to real-time accident management.

While it is almost impossible to develop and utilize exact models of the evolution of all possible accident sequences for each given type of nuclear power plant and containment design (e.g., PWR and BWR designs at various power ratings, large dry containment or ice condenser types, etc.), it appears possible to develop a sound approach to monitor the progression of an accident with respect to the integrity and effectiveness of a set of principal safety functions. The key to doing this is the development of a knowledge base "housing structure", where uncertain knowledge regarding the predicted plant behavior and real-time, but also uncertain, information compiled from plant instrumentation readings can be compared and matched to produce the best possible identification of plant states and possible accident control actions.

Instrument Uncertainty and Accident Management

One of the principal problems to be faced during the progression of an accident, no matter whether or not computer-based decision aids are available, is the problem of how to deal with plant information and instrument readings that may not be available in the desired form -- i.e., only indirect readings for a parameter of interest may exist, with uncertainty on which physical models may be used to deduce its value from these indirect indications --, or may be coming from instruments whose accuracy and reliability in the face of the severe conditions produced by the accident may be far from what may be expected under normal operating conditions. Because under all circumstances plant instrumentation de facto remains the only "eyes and ears" that the operators have to assess the course of events inside the plant and especially the containment boundary, and because any decisions that the operators make will depend on how instrumentation readings are ultimately "filtered" and interpreted by them, this issue is of paramount importance.

In assessing the "information needs" for accident management during the progression of an accident, references [2] and [3] (which constitute parallel studies for a large dry containment PWR and for a "Mark I" containment BWR, respectively) attempt an identification of what instruments may be available at what time during an accident, in order for the plant operators to be capable of identifying the status of key plant systems. These references base their conclusions on the assumption, which may be overly conservative, that, if the accident produces conditions more severe than those for which an instrument and the associated "exposed" systems have been qualified, then no information at all will be available from that channel. In the approach that we describe in this paper for the development of an Accident Management Advisor System (AMAS), we will take a less conservative view, by treating this same issue in probabilistic, rather than deterministic terms, with respect to how AMAS will assess the availability and reliability of certain important instrument readings. This issue is discussed in greater detail later in the paper, since it relates to one of the key characteristics of the AMAS decision aid.

Reference [4] describes the operational management of a nuclear power plant process in terms of a set of hierarchically and sequentially ordered functions/activities, namely: problem or disturbance detection, problem or disturbance diagnosis, plant state identification, and corrective action identification. Accident management can be viewed and characterized in almost exactly the same terms, except that the very definition of the term accident implies an abnormal event that has progressed already to the stage of being very severe. Under accident conditions, therefore, the question of "detection" is practically moot. Also, because of the harsh environment that may result from an accident, the question of diagnosis becomes in large degree one of determining to what extent the plant instrumentation can be relied upon to provide the plant operators with information that can be correctly interpreted and understood, and which may thus lead to a correct identification of the type of accident and progression stage the plant is in, at any given time. Finally it should also be noted that the problem of corrective action identification, under accident conditions, becomes more appropriately one of optimal mitigation action identification, since, again, an accident is typically a transient that has progressed well beyond the point where a full reversal to normal conditions may be obtained by "corrective action." On the contrary, an "optimal" mitigation strategy, i.e.

one which seeks to minimize accident damage and radiological effect consequences, is the only meaningful objective that can (and should) be pursued.

During an accident, the plant operators and the other plant management authorities which support in "real-time" the operators' decisions and actions (e.g., the plant technical support center), will have to make crucial decisions under the stress caused by the awareness of the potential severe consequences of an unfolding accident and of the often severely limited response time available to respond to it. This will have to be done despite the sparse or even conflicting nature of the information provided by the plant instruments under accident conditions, while attempting to account effectively for the pros and cons of one strategy over the other and trying to identify the best course of actions to be taken. The design of AMAS as a tool to aid the diagnostic and strategy-identification functions of the operators and technical support center personnel, is meant to provide access to all the available knowledge on the probable evolution of classes of severe accidents, as well as the ability to analyze the plant conditions and identify accordingly a "best strategy" for managing an evolving accident in light of such prevailing conditions and of the uncertainty associated with these at any given time.

(Please note: for the sake of brevity and simplicity we prevailingly use in the rest of this document the term "plant operators" to mean any group of people with decisional responsibility regarding plant operation during an accident.)

AMAS Architecture Definition

The AMAS architecture reflects the idea of creating a "meta-association table" or structure, i.e., a system of associative knowledge pointers which serve the purpose of linking together diagnostic plant physical models with plant state representations and decision-support models and algorithms. The AMAS architecture is thus articulated on three hierarchically superimposed levels of models linked together by a system of "information pointers" or "knowledge pointers," but performing separate functions, as illustrated in Figure 1.

The first level, which is called the "Parameter State Identification Filter" (PSIF) has the function of using the information provided by the existing plant instruments to arrive at the best possible identification of the state of key plant parameters. Because (as we have earlier discussed at some length) the information supplied by the plant instrumentation may be highly uncertain, the identification of these states will also be uncertain to a degree (depending on the particular type of accident and on how far it may have progressed). Thus the methodology applied to implement the PSIF function will be based on the use of probabilistic evaluation as well as on models of the interaction between the plant parameters and processes with the instruments available to monitor them. A more detailed discussion of the PSIF models and functions is given in the next section of this paper. The PSIF is called a "filter", because it distills raw instrument readings into a "best possible estimation" of plant parameter and system state conditions.

The second, and hierarchically intermediate, level of AMAS is the "Plant State Identification Module" (PSIM). This module receives the parameter and system state information elaborated by the PSIF and uses it in conjunction with a pre-ordained and organized collection of logic models which establish direct correspondences between certain combinations of parameter states and accident types and progression stages. Knowledge of the most likely plant parameter state vector at a given time allows the PSIM to point at the accident sequence which appears as the most likely to be actually unfolding, as well as to identify the apparent stage of progression of such a sequence. The combination of knowledge of the accident sequence which is unfolding and its progression stage represents knowledge of what in the probabilistic risk assessment (PRA) language is referred to as the "plant damage state" (PDS). The information produced by the PSIM is thus an identification, as time progresses, of the plant damage states traversed by the plant during the accident. The accident sequence progression models have often, in PRA studies such as reference [1], been cast in the form of APTs (accident progression trees). Although this type of models could also be used in AMAS , we have found more effective to use PSIM models in the form of "Bayesian Belief Networks" (BBNs). BBNs, which are discussed in further detail in the next section of this paper, are a very attractive option because of their natural compatibility with the other types of digraph models (e.g., logic flowgraph models and influence diagrams) which are used in the other two AMAS functional levels, and because they have been specifically developed to handle uncertain and probability-based information.

The third, and topmost, AMAS level is the "Management Action Decision Support Module" (MADSM), which uses the knowledge of most likely present plant conditions (e.g., damage states) to apply an accident consequence minimization scheme. In the current AMAS architecture, we have opted for the use of influence diagrams as the underlying decision models that capture the knowledge base of the overall consequence minimization strategies associated with the different possible plant damage conditions. Influence diagrams are very similar in nature to BBNs, the main difference consisting in the explicit representation in them of "decision nodes", i.e., variables the outcome state of which is directly under control of the plant manager/operator.

Information Processing through AMAS: from Diagnosis to Decision and Action

This section and the included subsections discuss the details of the knowledge base definitions that are used in the three AMAS levels as well as the details of how the information resident in these knowledge bases is combined with on line information obtained from the plant instrumentation and processed through AMAS to ultimately fulfill the AMAS primary function of assisting operators' accident management decisions.

Level One PSIF Models and Functions

As we have explained in the preceding section, the bottom level of AMAS consists of diagnostic models which have the ability to take into account and give representation to the uncertainty associated with instrument readings, when these readings are used to identify the most likely states of key plant process parameters (such as temperatures, flowrates, etc.). These models, to accomplish this, are developed according to the rules of the Logic Flowgraph Methodology (LFM) representation [5,6]. Under this representation, the diagnostic knowledge base consists, rather than of fault tree or similar binary state models, of LFM models of the plant physical and instrumentation processes which are related to, or affected by, potential accident sequences. The innovation of this approach lies in the treatment of the combination of instrument readings available at any given time during an accident as evidence with varying degrees of phenomenological and statistical consistency with certain plant damage states, and in the development of models of physical and probabilistic behavior of the plant instrumentation in the presence of conditions that are expected to develop within the RCS and containment in correspondence to these states.

The power of LFM is also in the fact that, unlike fault tree models which represent the deductively obtained representation of how one particular "top event" may occur, the LFM graph models give a synthetic representation, in one graph model, of all the cause-and-effect relationships which exist within a given process (or modeled portion thereof). This representation contains, in implicit form, essentially all the events and associated event-producing sequences that may be of interest for that process, and the LFM inference engine has the capability of automatically deriving and make explicit the representation or "explanation" of how any one of these top events may come to occur.

The deterministic version of the LFM representation and inference engine have been used in several diagnostic implementations and are widely documented in the open literature. For the AMAS application we have developed a Bayesian inference extension of the LFM concept which is described below in its most essential features and whose execution is illustrated by a simple AMAS execution example which is discussed later in this section.

Figure 2a shows the basic building block of LFM. In this figure the circle A represents a variable, possibly a physical parameter and B represents an instrument that measures variable A. A consists of a set of discretized states $\{a_i\}$ and B consists of the states $\{b_j\}$. The box TF represents a transfer function that maps the states $\{a_i\}$ to the states $\{b_j\}$, i.e.

$$B = TF(A)$$

or,

$$\{b_j\} = TF\{a_i\}.$$

If the transfer function is deterministic, i.e. there is no uncertainty, each state a_i maps into a unique state b_j. However, if there is uncertainty, the transfer function is probabilistic, i.e. given a state a_i, the mapping into $\{b_j\}$ is a probability mass function over $\{b_j\}$. In this case the transfer function, TF, is a matrix of probabilities, i.e.

$$TF \quad = \quad \begin{vmatrix} p(b_1/a_1)\ldots p(b_1/a_i)\ldots \\ \ldots \\ p(b_j/a_1)\ldots p(b_j/a_i)\ldots \\ \ldots \end{vmatrix}$$

Now, if A represents a physical parameter and B an instrument, we would like to determine the probability of any state ai of A being the true state, given that B is observed to be in state b_j. Then

$$p(a_i/b_j) = p(b_j/a_i)\ p(a_i)\ /\ p(b_j)$$

where $p(a_i)$ is given and

$$p(b_j) = \sum_i p(b_j/a_i)\ p(a_i).$$

In some cases the transfer function TF will depend on the value of another parameter. This is shown in Figure 2b, where C represents a physical parameter that consists of the states $\{c_k\}$. This would model the case where an instrument's performance depends on the value of a certain physical parameter, such as temperature. TF, then, is a set of matrices $\{TF_k\}$, and each matrix in the set is associated with each state c_k of C. Then, the probability that a_i is the true state of A given that B is in state b_j and C is in state c_k is given by

$$p(a_i/b_j) = p_k(b_j/a_i)\ p(a_i \cap c_k)\ /\ p(b_j)$$

where

$$p(b_j) = \sum_i p_k(b_j/a_i)\ p(a_i \cap c_k).$$

If A and C are independent, then

$$p(a_i \cap c_k) = p(a_i)\, p(c_k).$$

If A and C are not independent, but are influenced by the state(s) of another variable D which is upstream in the cause-and-effect chain, then the "probabilistic causality" relation expressed by the LFM model must be traced back to this common cause variable and the appropriate mathematics applied to express the probabilities of the ai states eventually only as functions of the states the "common root variable" D may be in.

In the AMAS formulation, the variable A would represent a physical parameter which is being measured by an instrument represented by B. C would represent another physical parameter which affects the performance of the instrument modeled by B. At each step, the AMAS Level One (PSIF) will gather all available relevant instrument readings, and from that information infer the most likely values of all physical parameters that are of interest for the plant damage state identification that needs to be carried out in Level Two. The output from Level One to Level Two will thus be a set of probability distributions over the likely values of the relevant parameters being inferred from the instrument readings.

Level Two PSIM Models and Functions

The basic function of the AMAS Level Two is the recognition of the type of accident sequence which is developing in the plant and the identification of the corresponding plant state. Plant states during an accident have been conventionally referred to, in PRA and accident management work, as "plant damage states" (PDSs). Because the identification of PDSs in real time is obviously a very important issue in the actual application of AMAS, we have dedicated a good amount of effort to obtaining a good understanding of how PDSs should be characterized and identified for a typical PWR plant with large dry containment.

Although the details of accident sequences that lead to core damage differ from sequence to sequence, many of the sequences behave similarly in the accident progression following the "uncovery of the top of active fuel" (UTAF), and certainly even more so in terms of the type and severity of the threat that they mount against the plant protective safety systems. It is, therefore, feasible to identify sets of plant-state-determining-parameters (PSDPs) that characterizes the states of the plant and to define potential plant-damage-state (PDS) groups based on these PSDPs. An example of PSDP is the parameter "vessel status" which describes whether the reactor vessel may be intact and unthreatened, or still intact but threatened, or compromised in its integrity. An example of PDS group is the class of situations in which the vessel is intact but threatened by an on-going core melt, while emergency systems are still available to flood the vessel internals with water.

Thus, a plant damage state can be defined in terms of the states of all plant safety functions and safety systems and the values of all key plant parameters which together constitute the PSDP combination sets. Plant safety functions can be in one of the following states: "normal", "threatened" and "impaired". Plant safety systems can be in one of the following states: "operating", "not operating/available", "unavailable/recoverable" and "unavailable/not recoverable". All of the possible plant damage states can be conveniently grouped so that classes of accidents resulting from PDSs in the same PDS group can be assumed to progress initially in a similar way, at least in terms of the conditions that determine the identification of the most appropriate accident management strategy and mitigation options.

The Level Two of AMAS will utilize information about the values of key physical parameters, as inferred by Level One, as well as information about the availability status of all plant safety systems, to identify the actual accident sequence path that the plant is following and to infer the status of the plant safety functions. The combination of the values of certain key physical parameters, and the status of all safety systems and safety functions constitutes, as we have discussed above, a PSDP set which characterizes the "plant damage state" (PDS). For actual applications, it is likely that the PDS group definitions may have to be extended in number and enriched with respect to the traditional definitions of PDSs found in probabilistic safety studies. The definition of PDSs and PSDPs will be part of the groundwork that is a prerequisite to the formulation of a standard procedure for the construction of AMAS Level Two models.

The interrelations between PSDPs and the correspondence of their state combination sets to specific PDSs is modeled in the AMAS Level Two (PSIM) by means of Bayesian Belief Networks (BBNs).

A Bayesian Belief Network [7] is a directed graph containing nodes, which represent variables, and directed arcs, which represent direct dependencies of one variable on another. Figure 3 shows a very simple belief network. The circles (nodes) represent variables and the arrows (arcs) represent direct dependencies between the variables. The variables are discretized into sets of states, i.e. $\{a_i\}$, $\{b_j\}$ and $\{c_k\}$, and their dependencies are modeled as probability matrices, i.e., for Figure 3, the probability that a_i is the true state of A given the states of B and C is given by

$$p(a_i) = p(a_i/b_j,c_k)\ p(c_k/b_j)\ p(b_j).$$

This can be generalized to a network of any size, i.e.

$$p(x_1) = p(x_1/x_2,\ldots,x_n)\ p(x_2/x_3,\ldots,x_n)\ldots\ p(x_n).$$

In AMAS the nodes represent variables such as the values of physical parameters, the status of safety systems and their associated components and the status of safety functions, and the arcs represent direct dependencies between these variables. The status of safety systems can be represented by the states: operating, not operating/available, unavailable/recoverable and unavailable/not recoverable. The status of safety system components can be represented by the following states: normal, failed/recoverable, and failed/not recoverable. The status of the safety functions could be represented by the following states: normal, threatened and impaired.

The AMAS Level Two BBN can infer the status of the plant and its safety systems from the probability values associated with the states of physical parameters which are direct or indirect indicators of the working conditions of certain components, as estimated and input to it by the Level Three LFM models. For example, the states of the parameter vectors "discharge pressure" and "discharge flow" are indicators of the status of an injection pump; less direct indicators would be the voltage drop across, and current absorption by, the pump motor. The Level Two can also infer the status of safety functions from the states of associated safety systems and the values of relevant physical parameters. As we have explained above, the status information on key parameters and safety functions allows AMAS to identify the most likely "PDS group" the plant is in at any specific time in the accident progression. At each evaluation step, AMAS infers such most likely PDS (group), given the possible values of physical parameters and the possible states of safety system components.

Level Three MADSM Models and Functions

The top level of AMAS must, given a plant damage state, determine the optimum set of operator actions in terms of preserving safety functions (which translates into minimizing radioactive release).

Since the primary aim of accident management is to preserve the integrity of safety functions, this information will be used by the top level of AMAS to select the accident management strategies that are most likely to accomplish that in a balanced and optimal fashion. This means that short term mitigation of the accident should not be achieved at the expense of long term damage containment goals, and that a balance of consequence minimization objectives should also be sought. In other words, direct plant damage should not be avoided at the cost of higher overall radiological exposures to the public, nor should insignificant gains in radiological consequences be sought at the cost of inducing massive additional plant equipment damage and disruption, which may in turn increase the risk of further radiological releases at some later time.

The top level will use information about physical parameters and the status of safety systems and safety functions and search for operator actions which will either prevent a transition to a less favorable plant state or cause a transition to a more favorable plant state. Adverse effects of operator actions will be considered, in that an action may cause the transition of one safety function to a more favorable state, but might also cause the transition of another safety function to a less favorable state. To accomplish this the use of influence diagrams is proposed.

Influence diagrams [7,8] are similar to Bayesian belief networks, but, in addition to chance nodes (equivalent to the nodes of a belief network) and directed arcs, influence diagrams contain decision nodes and a special kind of chance node called a value node. Decision nodes represent variables under a decision maker's control, in the case of AMAS, they would represent operator actions. The value node is a sink node (it has no successors, i.e. it has no arcs originating from it that point to other nodes), and in the case of AMAS it would represent the criterion against which the operator actions are assessed.

Figure 4 shows a simple influence diagram. The square is the decision node, and it consists of a set of options $\{d_i\}$. A and B are chance nodes and they consist of sets of possible states $\{a_j\}$ and $\{b_k\}$. V is the value node, and it consists of a deterministic function that maps the states $\{a_j\}$ and $\{b_k\}$ to a value V. In the diagram in Figure 4 only chance node B is directly influenced by the decision D, and the value node directly depends only on the chance nodes A and B. So the value, given a decision d_i, is

$$V(d_i) = V(a_j)\,p(a_j) + V(b_k)\,p(b_k/d_i).$$

In the top level of AMAS many decision nodes will be present, to represent all the candidate high level actions. In addition to the nodes that represent the plant damage states as determined by the middle level, there would also be chance nodes that represent uncertain phenomena that are important in terms of the possible progressions of the accident. Given a plant state and a set of actions, the influence diagram at the top level can be evaluated to determine all possible plant state transitions, as well as their associated probabilities of occurrence. These are compared through the value function represented by the value node to determine the optimal course of action.

Example of AMAS Execution

We discuss here an example which demonstrates the concept of the three level AMAS architecture. This example shows how the Parameter State Identification Filter of the bottom level of AMAS will employ the Logic Flowgraph Methodology to infer the states of key plant parameters from possibly uncertain instrument readings, and how this information is processed by the Bayesian belief network of the middle level to infer the current plant damage state. The example also shows how the influence diagram of the top level will evaluate candidate high level accident management actions, given the most likely plant damage state, in order to recommend to the plant operators a suitable course of recovery actions.

The test case used for the example is a hypothetical core damage situation induced by a station blackout (i.e., a "TMLB' " type of sequence). This test case has been used to validate the concept and the capabilities of the AMAS architecture in general and of the PSIF diagnostic algorithm more specifically, since this latter is an especially important element of the AMAS overall functionality. It should be understood that this example is for demonstration purposes only, and that some assumptions that have been made in the model presented here may not be entirely realistic when referred to a specific plant or another.

Figure 5 shows a logic flowgraph which is evaluated to infer the state of the "core location" plant parameter from the readings of the Source Range Nuclear Instrumentation (SNRI) and the Self Powered Neutron Detectors (SPNDs). It has been suggested [2] that these instruments can provide information as to the location in the vessel of the majority of the core material, but their effectiveness is conditioned by their survivability under accident conditions. These instruments are represented by the node labeled "SSRP" in Figure 5 and the "core location" parameter is represented by the node labeled "CL". "CT" represents the "core temperature" parameter, which conditions the transfer function between CR and SSRP.

In essence, the "SSRP" (SRNI & SPND Reading Pattern) variable is a composite core flux state vector which condenses in one synthetic representation the spatial reading pattern provided by the combination of the ex-vessel Source Range Nuclear Instrumentation (SRNI) and the in-core Self Powered Neutron Detectors (SPNDs).

It should be noted that, while the SRNI is likely to survive unscathed at least the first phases of a core-melt accident, it can be expected that a large fraction of the SPNDs will not. On the other hand, while in typical plants there are only two symmetrically placed spatial locations for the SRNI chambers (with no axial distribution discerning capabilities), a typical last-generation Combustion Engineering PWR plant (such as San Onofre 2 or 3) has a total of 280 SPNDs, distributed in 56 radial and 5 axial locations within the core. Thus we expect that under initial core-melt conditions, the geometric failure pattern of the SPNDs in itself will provide vital information as to whether a core-melt is occurring and as to the type of core relocation that this may be inducing. In order to translate and summarize this failure pattern into an identification of which SSRP vector states appear to be active at any given time, it may be necessary to construct a dedicated SPND signal pre-processor which intelligently analyzes not only the static "footprint" of SPND readings, but also the time evolution of how specific geometric areas covered by SPND detectors go from a state of "signal-provided" to one of "no-signal-provided." This combination of "geometry vs. time" preprocessor interpretation of SPND failures may provide the key to distinguish between local core relocation phenomena which induce the failure of specific SPND clusters, versus other accident-induced phenomena within the containment which may produce the disappearance of SPND signals by damaging, by a common-mode failure, the cabling that carries the signals out of the vessel and out of the containment. The example that follows does not try to speculate on the details of how the function of such a SPND signal preprocessor would be implemented, since this would entail a plant-specific analysis which is beyond the scope of the discussion in this paper. It is, however, assumed that such a preprocessor will indeed be capable of providing a "condensed reading" of the SSRP vector.

For simplicity, it is assumed here that the SSRP node has four possible states, which represent reading patterns of the combination of the SRNI and SPND instruments, and that the core location parameter has four states which correspond to the intact core state, the state in which the core is relocating, the state in which the core has slumped into the bottom head and the state where the core has exited the vessel. The matrices that represent the transfer function TF shown in Fig. 5 between the nodes CL and SSRP are shown in Figure 6. There are three transfer function matrices, and they are associated with the three states of the core temperature parameter, which represent average core temperature ranges, as shown in the figure. Thus, if CT is in state 2, there is a 0.25 probability that TF1 is the true transfer function, a probability of 0.5 that TF2 is the true transfer function and a probability of 0.25 that TF3 is the true transfer function, etc. For simplicity, the transfer function between CT and CL, which is not explicitly shown in Fig.5, is assumed to be almost deterministic, where the state 1 of CT (lowest temperature range) corresponds to state 1 of CL (intact), state 2 of CT (middle temperature range) corresponds to state 2 of CL (relocating) and state 3 of CT (highest temperature range) corresponds to state 3 of CL (slumped into bottom head) or state 4 (ex-vessel), with a probability of 0.5 for each. So, given a reading pattern of the SRNI and the

SPNDs, the probability distribution of the core location parameter can be found by evaluating the logic flowgraph described above. For example, if the SSRP node is observed to be in state 3, the evaluation of the logic flowgraph results in a probability of 0.06 that the core location parameter is in state 1, a probability of 0.5 that it is in state 2, a probability of 0.32 that it is in state 3 and a probability of 0.12 that it is in state 4. These results can be obtained via Bayesian probability matrix inversion (or, in more complex formulations, by means of heuristic solution algorithms). The simple situation discussed here is a good example of the nature of the information that the AMAS Level One (PSIF) can provide to the Level Two (PSIM).

Figure 7 shows a belief network representing the PSIM model that is interrogated by AMAS after the PSIF-generated information which we have discussed above is made available. This BBN models the dependence of the "core status" (represented by node CS) and "vessel status" (represented by node VS) PSDPs on the core location parameter. Also modeled in this network is whether or not the core is in a coolable configuration (node CC), the "vessel temperature" plant parameter (represented by node VT) and whether or not the vessel has been flooded (represented by node VF). Node CL stands for the "core location" parameter, as in Figure 6.

The core status characteristic is directly related to the core location parameter, in that the states of this characteristic are "intact", "relocating", "slumped" and "ex-vessel". The core coolability characteristic has two states, coolable and not coolable, and depends probabilistically on the core status characteristic, in that as the melt progresses there is a greater chance that the core will not be in a coolable configuration. The vessel temperature node has three states that represent peak temperature ranges, and it depends on nodes CL, CC and VF, in that if the core has slumped into the lower head and is in direct contact with the vessel wall, the vessel temperature will increase. However, if the vessel is flooded, the water will cool the vessel wall, especially if the core is in a coolable configuration. The vessel status node has three states: "acceptable", "threatened" and "failed". These correspond directly to the states of the vessel temperature node, i.e. if the vessel temperature node is in state 3, which would correspond to the melting temperature of the vessel or higher, the vessel status node would be in the "failed" state. Finally, the vessel flooded node has two states: "flooded" and "not flooded".

Thus, given a state for the core location parameter, the belief network described above can be evaluated according to the general BBN probabilistic rules briefly described above in this section, to find the probability distributions for the states of the vessel status and core status plant damage state characteristics. Assuming that the distribution for the core location parameter states is as determined by the PSIF in the discussion given above and that the vessel is not flooded, the probabilities for the core status states are the same as those for the core location parameter found above, i.e., 0.06 for "intact", 0.5 for "relocating", 0.32 for "slumped" and 0.12 for "ex-vessel." The probabilities for the vessel status states can then be calculated to be 0.31 for "acceptable", 0.57 for "threatened" and 0.12 for "failed". If we assume then that at this point in the TMLB' sequence a limited AC power availability from the diesel generators is recovered, the PDS which appears to be the one with the highest likelihood of being true is one belonging to the class or "group" characterized by: "reactor vessel intact but threatened" and "limited AC power available." The determination of the probabilities associated with each of the possible PDSs can at this point be made available to the AMAS Level Three (MADSM), which can then identify the PDS mitigation actions/strategy that are most likely to be effective.

As an example of the MADSM upper level of AMAS functionality, Figure 8 shows an influence diagram that models the decision problem of whether or not to inject water into the vessel, assuming that the systems required for vessel injection are available. The decision node "D" represents the decision of whether or not to recommend that the operators flood the vessel. The node CS represents the "core status" plant damage state characteristic, and the node VS represents the "vessel status" plant damage state characteristic, as in Figure 7. Node CC represents whether or not the core is in a coolable configuration and node FCI represents whether or not an in-vessel fuel-coolant interaction that results in the failure of the vessel occurs. Node ECF represents whether or not such an in-vessel fuel-coolant interaction results in an "alpha-mode" failure of the containment. The value node V represents a value function which weighs the remote possibility of early containment failure versus the possibility that vessel failure will likely be prevented.

Since the effectiveness of this action, as well as the possibility of the adverse effect of the occurrence of an in-vessel fuel-coolant interaction, depend on the stage of the melt progression, the decision of whether or not to flood the vessel depends on the "core status" plant damage state characteristic. The CL, CS, CC, VT, VS and VF nodes are the same as those described above for the belief network. The node FCI has two states: "vessel failure due to FCI" and "no vessel failure due to FCI" and depends on the stage of the core melt progression and whether or not the vessel has been flooded. Node ECF has two states: "alpha-mode containment failure" and "no alpha-mode containment failure", and of course depends on whether or not there was an in-vessel FCI that resulted in vessel failure.

The influence diagram just described can be evaluated to find the change in the probabilities of the vessel status states and the change in the probability of early containment failure if the vessel is flooded, versus if it is not. If a suitable value function is specified, these probabilities can be collapsed into a single utility value that will allow the upper level of AMAS to recommend to the operators whether or not they should flood the vessel. If the probability distribution for the states of the core status characteristic found above are assumed, evaluation of this influence diagram when the decision node is in the flood state results in a probability of 0.84 that the vessel status characteristic will be in the "acceptable" state after flooding, a probability of 0.04 that it will be in the "threatened" state after flooding, a probability of 0.12 that it is already failed, and a probability of 0.032 that the vessel will fail due to an in-vessel FCI. However, there is also a probability of 3.2E-4 that an alpha-mode failure of the containment will occur if the vessel is flooded.

Looking at these results it can be seen that flooding of the vessel reduces the probability of the vessel status characteristic being in the threatened state by about a factor of ten (from a probability of 0.57 to a probability of 0.04). However, it increases the probability of it being in the failed state by 0.03, due to the possibility of an in-vessel FCI, and increases the probability of an alpha-mode failure of the containment from 0.0 to 3.2E-4. These factors must be weighed by the value function represented by the value node in order to recommend the appropriate action (or non-action). If, as it would appears to be the case, there exists no special reason to view the alpha-mode containment failure more negatively than the type of containment failure that may follow with a non-negligible probability if the vessel fails by melt-through, the AMAS recommendation would under these conditions be of taking advantage of the partial AC recovery and flooding the vessel with injection water.

Another aspect of the flow of information within AMAS is the feedback of information from the top level and middle level to the bottom level. Given that the plant state is known at an acceptable level of confidence, prediction of parameter evolution will be fed back to the bottom level for sensor prior updating. To translate this concept into the terms of the example given here, it must be noted that the inversion of the Bayesian probability matrices needed for the determination of the probabilities associated with the "core relocation" (CR) parameter states requires knowledge of the probabilities associated with the states of the parameter "core temperature" (CT). If we assume, as is reflected in the LFM model shown for this example in Fig. 5, that CT is a "basic node", i.e., a node for which no further causally conditioning relationship has been identified and represented, the unconditional probabilities of its states must be updated, as time progresses, as a function of the determinations made in Level Two as to which damage state the plant is in, or headed towards. Thus, before UTAF, the "initial condition" values assigned to these probabilities will reflect the fact that the core temperature is most likely to be in a close-to-normal range (which means that low unconditional probability values will be assigned to the "core temperature" = < high > state). Later, if the accident starts to progress and instrument readings show a raise in core temperature, the prior probabilities themselves will be modified to reflect the overall knowledge that the plant is moving toward a damage state which comports high core temperatures. Still later, when the core temperature instrumentation readings may be no longer available because of the effects of the accident progression, the updating of these unconditional probabilities will have to be based on the determination of the plant damage state that the AMAS Level Two would have itself arrived at in the preceding assessment iteration.

The final function of AMAS is the verification of the positive effect of accident management actions. Given a selected optimum action, it will be natural for AMAS to compare the actual evolution of the plant state to the predicted evolution and verify whether the implemented actions are having the desired effect. This verification objective is easily met by the intrinsic nature of the influence diagram models used by the AMAS Level Three. These in fact have to show explicitly in a complete Level Three MADSM model, as target nodes downstream of the decision nodes which represent possible operators' actions, both the "desirable" and "undesirable" plant damage state outcomes that may result after these actions are or are not taken. Once an action has been taken, it is automatically known from the associated influence diagram which "desirable PDS outcome" AMAS was trying to obtain as a result of that action. In the next assessment iteration AMAS will then seek to verify that the actual PDS reached, or about to be reached, matches to a satisfactory extent the "desirable PDS" targeted in the preceding assessment iteration. This simple verification scheme offers the advantage of an easier implementation with respect to more complex trend analysis schemes (involving parameter derivative value tracking) that could be envisioned for the same purpose.

Conclusions

This paper illustrates the concept and the architecture of the Accident Management Advisor System, a decision aid which enables the use of combined instrument information to reduce uncertainty in decision making associated with nuclear plant accident conditions. The principal benefits offered by this concept are the definition of an approach to utilize instrument information under uncertain accident conditions in such a way as to allow the best possible assessment of plant status and the implementation of a formalized accident management decision-making strategy by means of a computer-based operator assistance tool. When fully developed, we expect AMAS to find application in both the commercial and government sections of the U.S. nuclear industry. We currently plan to have a working prototype of the system, ready to demonstrate its functionality for a representative commercial PWR plant, by the end of the next phase of our research, in which both model development and software development activities will have to be carried out.

Finally, the AMAS architecture and the models of its implementation (e.g., influence diagrams and logic-flowgraph models) could be utilized to develop operators' aids for other process industries, in which real-time diagnosis under uncertainty and response to emergencies may be required. For such an extension, it would be necessary to develop a knowledge base appropriate to the specific process industry under consideration and this would not be a trivial task; however, the potential benefits from such computerized aids would make such an effort worthwhile.

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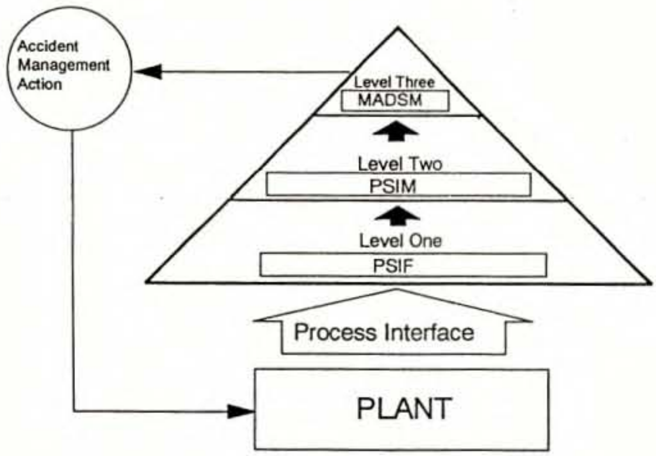


Figure 1: AMAS Architecture

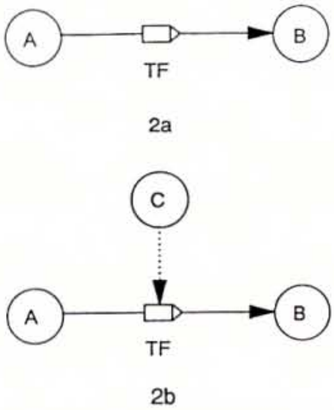


Figure 2: Basic LFM Building Blocks

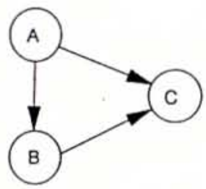


Figure 3: Example of a Simple Bayesian Belief Network

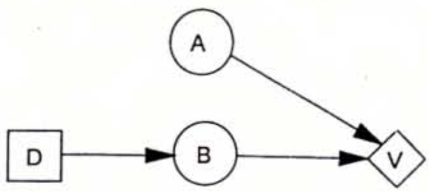


Figure 4: Example of a Simple Influence Diagram

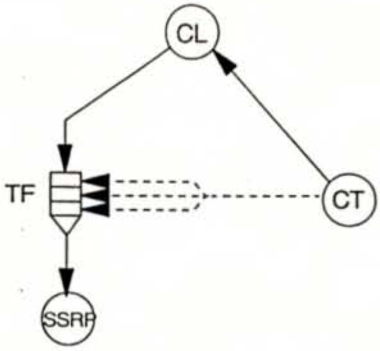
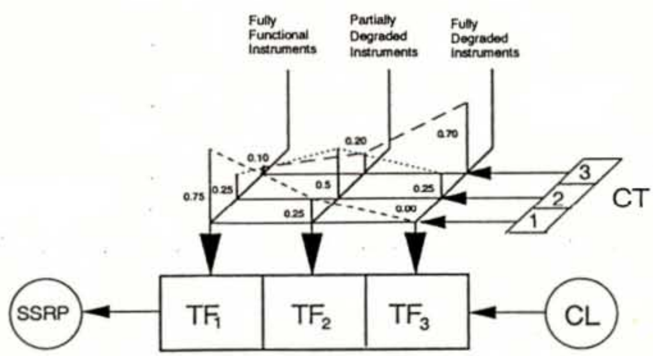


Figure 5: Example of LFM Model Used in AMAS Level One



		CL				CL				CL			
		1	2	3	4	1	2	3	4	1	2	3	4
SSRP	1	0.9	0.1	0.0	0.0	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	2	0.1	0.8	0.1	0.0	0.0	0.1	0.1	0.0	0.1	0.0	0.0	0.0
	3	0.0	0.1	0.8	0.1	0.1	0.8	0.5	0.2	0.1	0.1	0.1	0.1
	4	0.0	0.0	0.1	0.9	0.0	0.1	0.4	0.8	0.8	0.9	0.9	0.9

Figure 6: Example of LFM Probabilistic Transfer Function

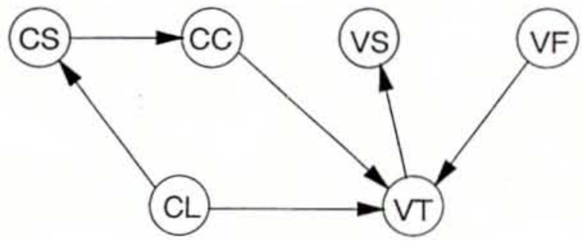


Figure 7: Example of BBN Used in AMAS Level Two

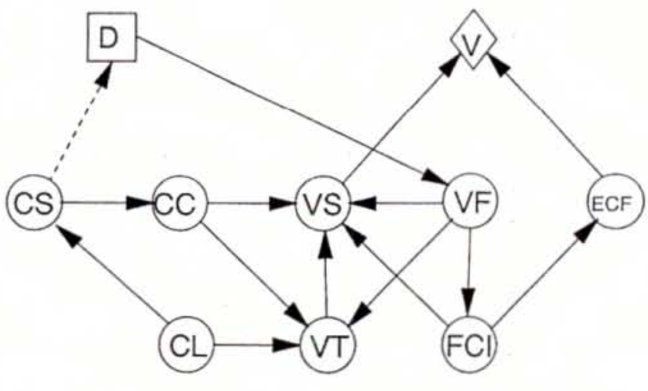


Figure 8: Example of Influence Diagram Used in AMAS Level Three

TRACTEBEL
ENERGY ENGINEERING

OECD SPECIALIST MEETING ON INSTRUMENTATION TO MANAGE SEVERE ACCIDENTS

KÖLN, 16th - 18th MARCH 1992

Session IV : OPERATIONAL AIDS AND ARTIFICIAL INTELLIGENCE

OPERATOR SUPPORT SYSTEMS USING ARTIFICIAL INTELLIGENCE : ACTUAL SITUATION AND FUTURE IN BELGIUM

Michel DE VLAMINCK (Belgatom / Tractebel)

Abstract

This paper describes the concept of the OPERATOR ADVISOR(=OPA) expert system developed at Tractebel to assist plant operators for optimal recovery following an accident.

OPA uses an object-oriented expert system to access an appropriate knowledge base for either knowledge acquisition or retrieving information for the operators.

The main advantages of such a system are :

- On-line access to a flexible and easy adaptable knowledge base;
- Post accident monitoring of all operator actions;
- Priority listing of all needed actions;
- The availability of relevant background information.

1. Introduction

While artificial intelligence techniques were still considered a laboratory curiosity in the eighties, they are now increasingly seen as a practical management tool in many countries.

Expert system technology, a particular application of artificial intelligence techniques, has matured enough to offer some interesting applications in the nuclear industry as can be observed by the growing number of publications in this area.

While such techniques are being developed for various purposes such as to power production enhancement, or to increase productivity, the operator advisor (OPA) development at Tractebel is aimed at reducing the safety challenges to the plant.

The need to bring expert systems in the field of operations of complex nuclear power plants results from two main observations :

The stringent safety measures taken in operating nuclear plants do not allow a full automatisation of all operations, such that the plant operator plays a crucial role to maintain the plant in a safe state or mitigate any plant transients or accidents.

The results of PRA studies clearly indicate that the operator is the weakest link in all the engineered safety features that are conceived to prevent the degradation of an abnormal event into a severe accident.

During the initial phase of an accident, when the protection and safety systems are conceived to operate automatically, the operator is nevertheless exposed to a large number of alarms and has to monitor a large number of gauges in order to identify the plant malfunction, since he has to take over after the initial automatic phase. All these actions can easily outrange the human capabilities.

Currently, the actions to be taken are available to the operator in the condensed form of written sequential procedures which reflect the outcome of a large number of analyses, experiments and simulator sessions, all of which the operator cannot grasp at the moment he follows the procedures. At some stage of a severe accident, no procedure at all can even be available.

The motive for developing OPA is to make this expertise available to the operator and to prioritise the actions needed for optimal recovery.

2. Merits of an expert system versus written procedures

Written procedures are highly sequential and adapted to the reading behavior of the operator : all procedure steps must be read in sequence, even those that are not yet applicable or already satisfied. Furthermore, the goal is often to make the operator monitor a number of concerns or functions in parallel.

This is translated in the procedures as a cyclic run-through of a number of steps or even pages (e.g. the critical safety function status trees), which is a rather tedious task.

Second, the operator gets the information in a step only once he reaches that step. This is often not adequate since concerns and cautions, which must be monitored in parallel with a whole sequence, may be read when not yet applicable or forgotten when required.

Thirdly, procedures are written on the assumption that the operator "does the right thing". This is not always realistic. Indeed, it may very well happen that at some step a valve is correctly closed by the operator, but is reopened later in the transient by the operator himself or by some automatic action for which no warning is issued in the procedures. The same is true when an equipment malfunction occurs during later stages of the transient for which no warning is issued.

Finally, the expert, in drafting the procedures, is limited in the depth and the scope of knowledge he wants to transmit by the capacity of the operator to handle the complexity and the amount of text during an accident.

It is clear that the written form of procedures imposes constraints on the way the expert can represent his knowledge as well as on the way the operator is able to use it, especially under stress conditions.

The application of modern information technology gives the opportunity to relieve these constraints considerably :

- A computer can monitor all instruments quasi instantaneously and compare measurements with fixed or even dynamic criteria;
- A computer can monitor a quasi unlimited number of functions in parallel, and advise the operator in an event driven way;

- The computer can apply on-line an enormous amount of very intricate expertise such as fine-tuned diagnostics, refined context dependent monitorings, etc...;
- The computer can respond instantaneously to changes in situations caused by e.g. unexpected equipment malfunctions, by applying its continuous parallel monitoring;
- Finally, the computer, if conceived in a fail-safe and redundant configuration, is never tired, and not subjected to nervous stress.

3. The operator advisor approach

The background underlying the post-accidental operation reveals much more than a sequential process. Indeed, it requires a dynamic task planning in a real time changing environment to perform tasks of a different nature : sequential or parallel, activated by success or failure of these tasks, executed under dynamic conditions or inhibited in other situations.

Within that scope, an operator support expert system called OPA (=OPerator Advisor) has been developed by Tractebel.

The real strength of expert system techniques resides in the separation between what is called the knowledge base and the surrounding shell expert system. The shell is roughly what is left of the expert system when all factual knowledge is removed, while the knowledge base is specific for the mission of the expert system (in this case to assist the operator in off normal plant conditions).

This separation, together with the existence of powerful symbolic programming environment, enables the systems engineer himself to extend or modify off-line the knowledge base as warranted by new evidence. He does this by means of a knowledge representation formalism which he can access via the knowledge acquisition interface without the need to master the programming language.

The expert system can then use this knowledge on-line to schedule and manage the tasks, as driven by the process parameters.In this mode of operation, the process parameters are monitored continuously via the input processor. These data are being interpreted by the inference mechanism and the resulting messages are displayed to the operators using the run time unit.

Two basic interfaces are thus available : a knowledge acquisition unit for the expert, and a run time unit for the end user (the operator or the shift technical advisor).

The operator advisor approach has resulted in an appropriate knowledge representation scheme and an efficient inference mechanism that permit OPA to act as a truly dynamic task scheduler rather than a conventional computerised procedure display.

This is illustrated in Figure 1 where the following run time interface features can be recognised :

- A compact, action-oriented prioritized punch list;
- A history of past events that are relevant to the applied procedure;
- Detailed information and explanation upon request.

It is foreseen that the system will operate in an event-driven way without keyboard dialogue required, to supply the following functionalities to the operators :

- Status assessment : accident diagnosis and status tree monitoring to supply information for both the event oriented as well as the state oriented accident procedures;
- Automatic response management;
- Dynamic response management to supply :
    - Context dependent advice on actions to be taken;
    - Event-driven concerns and cautions;
    - Continuous check of validity of diagnosis;
    - Evaluation of operator actions;
    - Display of background information upon request.

The capability to monitor the operator's actions is one of the features that differentiates OPA from existing computerised procedure displays. The expert system not only suggests the actions the operator should perform in an event-driven way, but can also monitor a selected number of actions that for the actual situation should be avoided.

4. Development and testing phases for OPA

In order to demonstrate the feasibility of the approach, Tractebel selected the steam generator tube rupture (SGTR) accident scenario of a 2 loop PWR plant, since such accident is not only very probable and happened already in Doel 2, but it requires intensive operator interventions to mitigate such accident.

The first task is to enter the related knowledge into the OPA knowledge base via the graphical knowledge acquisition interface which provides the system expert with an effective tool to describe the evolution of the accident treatment in a natural way. This is illustrated in Figure 2 where one of the multiple subtrees of the procedure can be recognised i.e. the actions required upon identification of the ruptured steam generator. This figure also shows the various attributes that define the safety injection reset function as supplied by the knowledge acquisition interface.

The next stage consists of the testing and updating of the knowledge base by running the expert system in tandem with an interactive SGTR accident simulation program, which supplies the process data to the expert system. By simulating a large variation of SGTR scenarios, one can check and improve the knowledge base to be able to cope with multiple equipment malfunctions or erroneous operator interventions.

In a third step, at the end of 1990, the expert system was connected to the Doel 1 Training Simulator which was driven into a number of SGTR scenarios. The outcome of the tests was mainly :

- The ability of the system to correctly diagnose the events and advise the operators on the appropriate actions to take;
- A deeper insight in the information management support that should be available to the operators in the control room during emergency situations;
- A comparison of the merits of an expert system versus a more traditional computerised procedure tracking system. (wich has been developed with that aim).

5 Future trends

The next steps which started at the begining of this year are the following :

- to make the system run on conventional workstations;
- to improve the man-machine interface;
- to broaden the knowledge-base;
- to study the robustness of the system against invalid input data;
- to study the behaviour of the system in real industrial environment (response time, knowledge-base management, ...);
- to study the connection of the system with the computerized supervison system of the plant.

On the other hand, the experience we are progressively gaining in our company in the field of severe accident modelisation should help us to assess the interest to make use of knowledge based systems to help deal with such situations.

Some motives for persuing that trail are the following :

- There is at present a lack of expertise available under the form of written procedures;
- There is a need to monitor a large number of functions or parameters in parallel and to include complex expertise in the recovery strategy;
- Operator stress could be reduced in situations where the operator reaction quality is difficult to assess.

Nevertheless, serious limitations still remain : on the one hand, the existence and the reliability of the instrumentation and on the other hand, the knowledge of the phenomena evolution and of the optimal recovery strategies.

6 Conclusion

This paper has described an expert system under development at Tractebel aimed at providing assistance to the operators in plant off-normal conditions.

The capability to monitor the operator actions and to advise him in an event-driven way are the specific features that distinguish OPA from other software.

While the basic structure of the expert system is fixed by the shell software environment, the knowledge base can easily be adapted in view of new operational evidence.

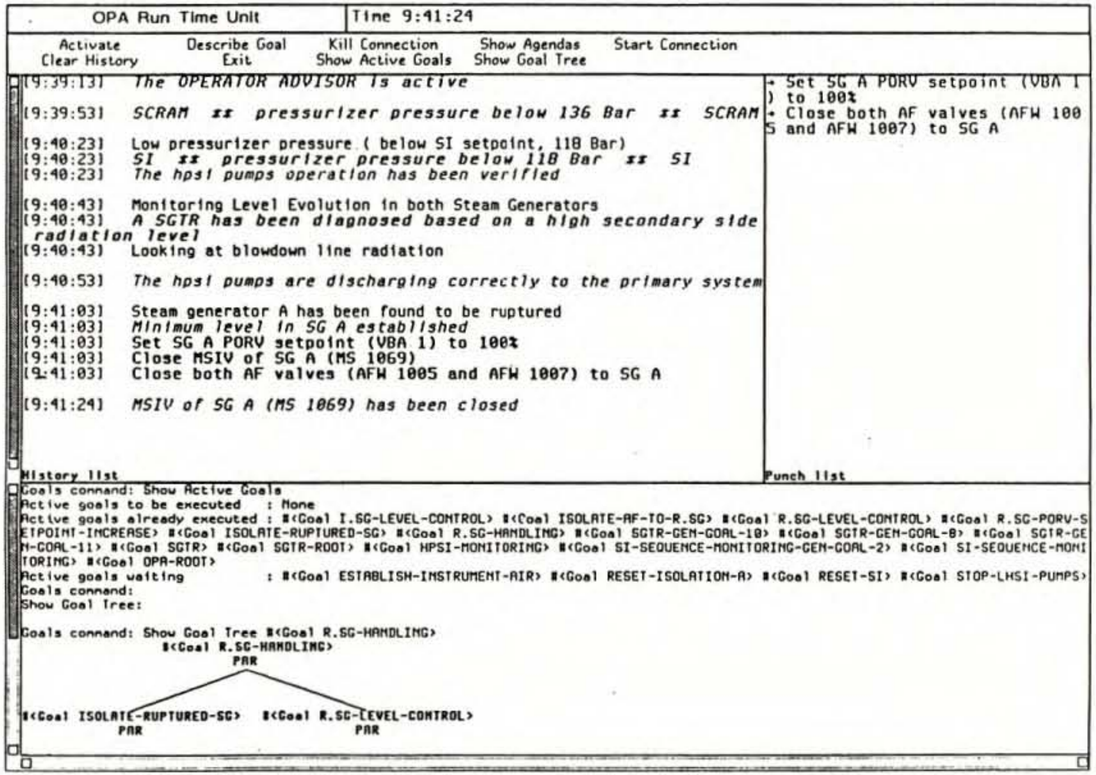


FIGURE 1

OPA Run Time Unit

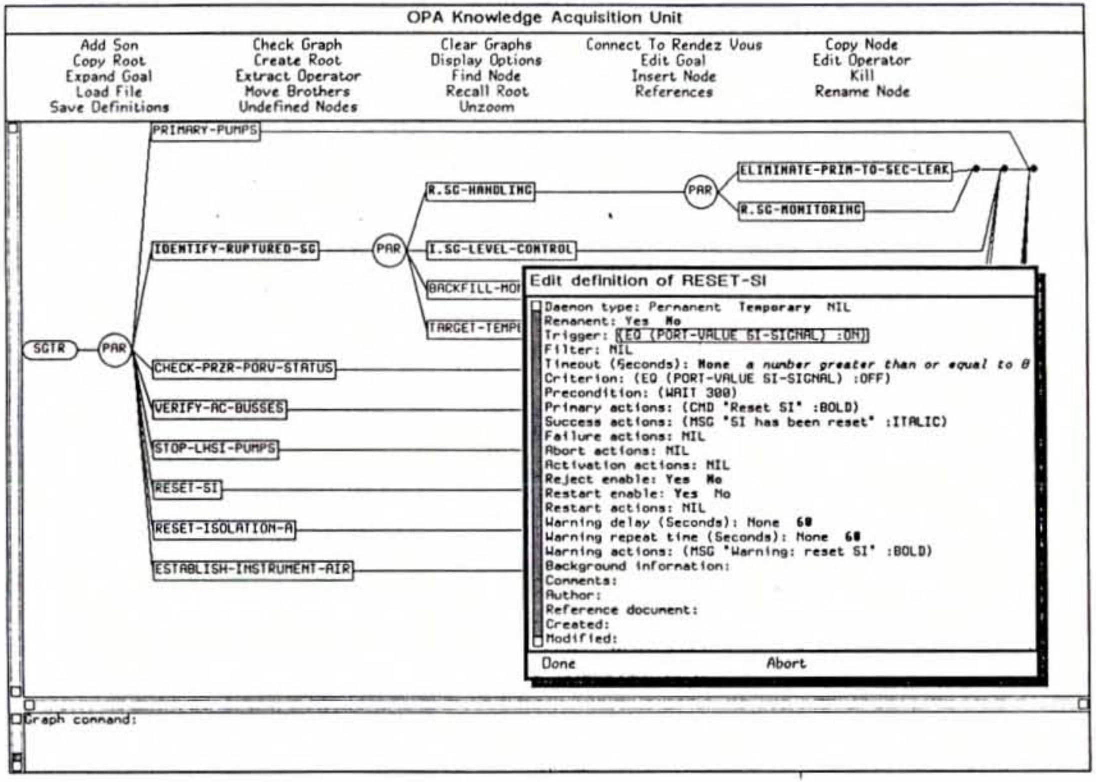


FIGURE 2

OPA Knowledge Acquisition Unit

Total Process Surveillance : (TOPS)

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1.0 Introduction

In order to operate a plant safely and economically, an operator requires a complete knowledge of the plant's operating state. Only a limited amount of information is generally available regarding the plant's current state, this being determined by a finite number of transducers measuring key process parameters. It is the responsibility of the operator to assimilate and interpret this, possibly conflicting, raw measurement data.

An operator accomplishes this data interpretation task by utilising his knowledge of the operation of the process and his experience of its behaviour under certain well defined conditions. Under normal operating conditions the operator may use only a fairly basic mental model of the process to facilitate his understanding. However under off-normal plant conditions and especially those associated with a severe accident, this mental model may not be sufficient to understand the behaviour of the process. This is highly likely to be the case where the process system has undergone a structural change as a result of a severe accident.

As with any management task, the prime components of dealing with a severe accident condition are *monitoring* and *control*. This implies the need to obtain information on the process state, assimilate this information, interpret and understand what it means in the context of the process, leading to a control decision and thus a control action. A key task here is the gathering and assimilation of all the available plant data.

The transducers installed on a plant represent a diverse range of information sources. Traditionally these are considered individually or in functional groups, such as fuel channel outlet temperatures. However each transducer measurement is almost invariably linked to other different transducer measurements via the physics of the process. This leads to the concept of *analytical redundancy* among a given set of diverse transducers. In an off-normal plant condition, the ability to exploit this analytical redundancy is of significant importance in order to obtain the maximum amount of information as to the state of the plant. A further logical step in this theory

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is the accommodation of failed or damaged transducers by deriving their measurement information from the available remaining transducers.

A system capable of providing a clear and improved picture of the plant's behaviour under a range of normal and off-normal operating conditions would thus be of significant economical and safety benefit.

2.0 TOPS Concept

A Total Process Surveillance system (TOPS) is currently under development by AEA Technology, Reactor Services, Instrumentation and Surveillance Techniques Department at our Risley Laboratories. TOPS has been conceived with this goal of a succinct plant status monitor in mind. There are two main constituents of the Total Process Surveillance System, Fig. 1.0.

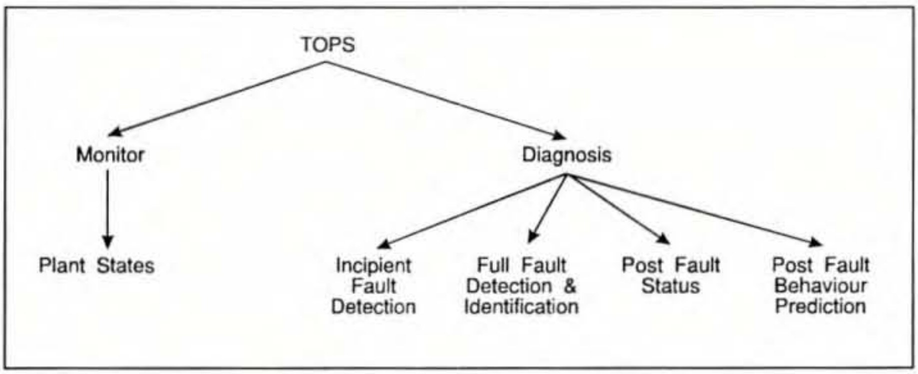

Fig. 1.0 TOPS Functional Structure

The first is a process monitor which uses a mathematical model of the process together with the available process control inputs and measurement outputs to provide additional information on the internal state of the process.

The second function of TOPS is to provide diagnostic facilities on the plant. The diagnosis function is essentially one of data reduction where a statement on the current state of the plant is made based on all the available information.

The mathematical model forms a key component of the TOPS algorithms as it is this which defines the static and dynamic relationships between the control inputs and the instrumentation signals, and also among the measurement signals themselves. This model based approach has been chosen as this represents the best method for obtaining additional information on the internal state of the process. It is the internal model variables which provide this additional information. Quite often these variables equate to physical or pseudo physical properties which cannot be measured directly due to unavailable technology or difficult transducer access.

3.0 Plant Monitor

The plant monitor is the heart of the TOPS system. It is designed using a concept from modern control theory known as an observer, Fig. 2.0.

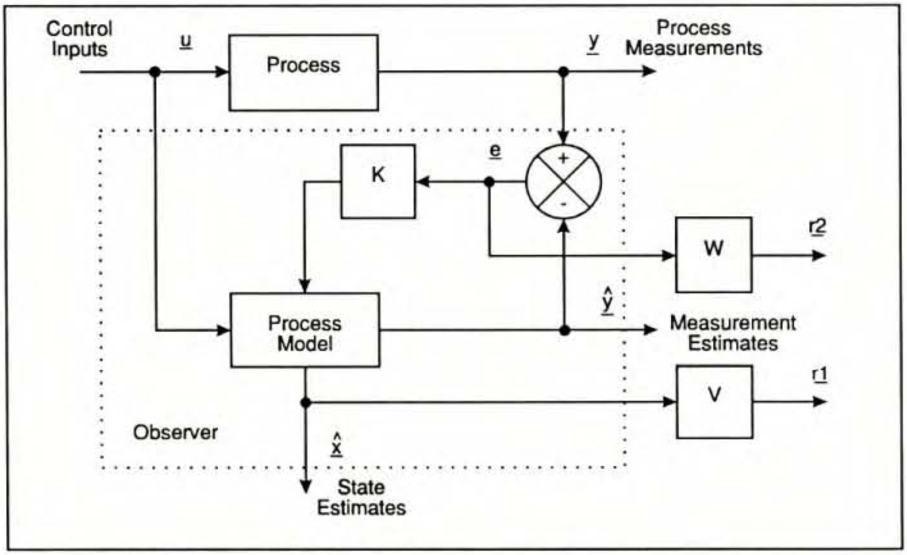

Fig. 2.0 Observer Structure

An observer uses a mathematical model of the process to combine the available process measurements in a structured way and as a consequence generates estimates of internal process states which are not, or cannot be, measured. This is achieved by transforming the plant model differential equations into state space form giving the following discrete time system description :

$$\mathbf{x}(k+1) = \boldsymbol{A}\mathbf{x}(k) + \boldsymbol{B}\boldsymbol{u}(k) \quad (1)$$

$$\boldsymbol{y}(k) = \mathbf{C}\mathbf{x}(k) + \boldsymbol{D}\boldsymbol{u}(k) \quad (2)$$

where

$\mathbf{x}(k)$ is the system state variable vector

$\boldsymbol{u}(k)$ is the vector of control input signals

$\boldsymbol{y}(k)$ is the vector of measurement signals

$\boldsymbol{A}$ is the system matrix
$\boldsymbol{B}$ is the control input distribution matrix
$\boldsymbol{C}$ is the state output matrix and
$\boldsymbol{D}$ a feedforward output matrix

Running the model in this form requires no complex integration algorithms between each time step, just simple algebraic multiplications and additions. Thus the statespace notation also facilitates a real time implementation, a feature which is a prerequisite for any usable diagnostic system.

The observer model has a feedback path whereby the model state estimate vector is modified according to the difference between the model's current output estimates and the plant measurement signals. The feedback gain matrix $\boldsymbol{K}$ is chosen to ensure that the model state estimates always converge towards the true plant state, and to impart some additional and very important properties on the observer. This ability to track the plant is especially important even under normal operating conditions when the process may move through several operating points. The observer structure is thus:

$$\hat{\mathbf{x}}(k+1) = (\boldsymbol{A} - \boldsymbol{KC})\hat{\mathbf{x}}(k) + \boldsymbol{Bu}(k) + \boldsymbol{Ky}(k) \tag{3}$$

$$\hat{\mathbf{y}}(k) = \mathbf{C}\hat{\mathbf{x}}(k) + \boldsymbol{Du}(k) \tag{4}$$

The physical transducer measurements can be considered as giving the symptoms of the plant's state. Through state estimation the observer effectively dissects the plant to give a surgeon's eye view of the internal plant states.

The choice of the observer gain $\boldsymbol{K}$ is extremely important. As with any mathematical description of a physical process, modelling errors inevitably exist. For the observer to provide accurate and useful additional plant state estimates it must be designed to be robust to these modelling errors. Techniques such as EigenStructure Assignment [1] and Unknown Input Observer Design [2] are available to compensate for these modelling errors.

The feedback matrix $\boldsymbol{K}$ is designed by firstly assessing the discrepancy between the plant and the model running in an open loop mode. This discrepancy is described as an unwanted input vector direction and $\boldsymbol{K}$ is thus chosen to decouple this unwanted input. A corollary of this design technique is that plant faults which manifest themselves in a similar input vector direction will also be decoupled. Nevertheless, this technique enables a useful robust observer to be designed with only a relatively simple plant model.

4.0 Fault Detection and Diagnosis

The plant monitor structure provides several extremely powerful facilities for providing fault detection and diagnosis.

The state estimates generated by the observer may be used directly by comparison with predefined normal behaviour using simple fixed or adaptive alarm thresholds.

A more powerful technique is to analyse the state estimates as a vector. Under normal operating conditions the magnitude and direction of the state vector have fixed signatures. Should a fault occur different magnitude and direction signatures will be obtained. This provides the fault detection. Diagnosis of the fault source or sources is achieved by analysing the fault signature and comparing with known fault signatures obtained by experience or simulation. A projection matrix ***V*** can be designed to pre-process the state estimate vector to produce a set of zero biased residual signals ***r1***. ***V*** is tuned to ignore the normal signatures of the state vector (***r1*** = 0) but generate non-zero residuals in the presence of faults.

The error signal produced by the difference between the plant measurements and observer output estimates also contains fault information. This can be processed in a similar way to the state vector by suitable design of the projection matrix ***W***. A second set of residual signals ***r2*** may be obtained with different properties and fault responses than ***r1***.

5.0 Information Assimilation

The residual signals obtained from the observer have already performed a significant amount of data reduction and can be presented to the operator directly. Depending upon the complexity of the process a further level of information interpretation may be required, perhaps including information from unmodeled sources such as covergas monitors, delayed neutron detectors or acoustic noise transducers. This leads us to consider the vertical structure of the TOPS system.

Any process plant, including a reactor, can generally be functionally and physically broken down into a number of small selfcontained but interconnected modules, e.g. primary circuit, secondary circuit, feedwater system etc. Treating each of these modules as separate systems for diagnosis makes the diagnosis of the whole process more manageable.

This logically leads to a hierarchical vertical structure, Fig. 3.0, where local diagnostic hypotheses are collated together at successively higher levels in order to formulate an overall process diagnostic hypothesis at the top level. The level of abstraction of the diagnostic information increases as the hierarchy is ascended. In this respect, alternative techniques from the TOPS model based algorithms, such as artificial intelligence which are better suited to dealing with this more abstract data, are worthy of consideration. Although not part of the existing TOPS development programme, we have proposed that expert systems may offer a potential solution for these higher level diagnosis functions.

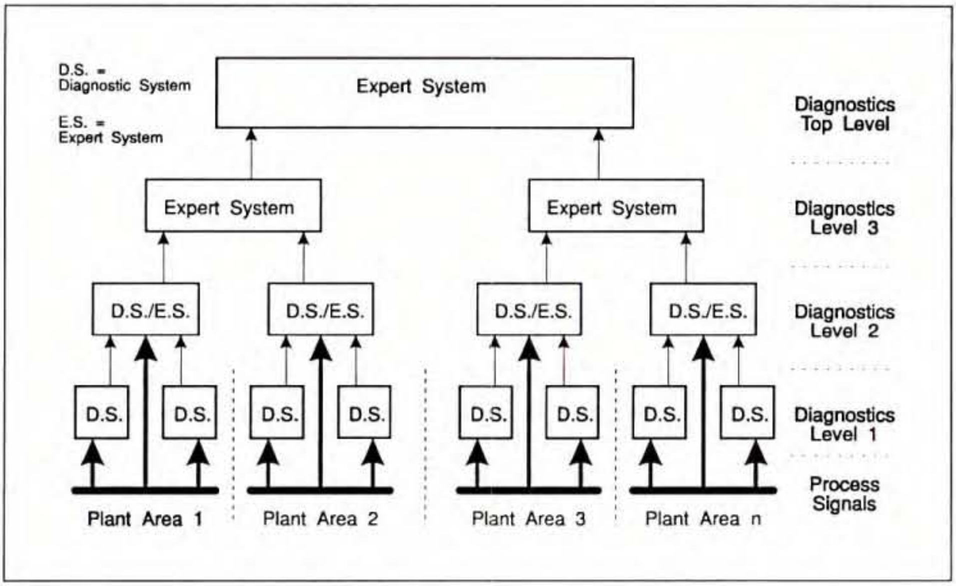

Fig. 3.0 TOPS Hierarchical Structure

This hierarchical vertical structure lends itself to adaptation of the diagnostic algorithms under severe plant failure conditions. If an event occurs which leads to the loss or damage of one or more transducers, this would be detected at the first diagnostic level. A local decision could be made at this level to reconfigure the TOPS observer using a different set of instrument signals, such that it still provided valid plant state estimates. The accommodation of plant component failures which render the TOPS observer model invalid is a more difficult issue. While adapting the observer model is conceptually possible, obtaining a practical robust adaptation algorithm may not be straightforward.

6.0 Application and Demonstration

The TOPS system is being developed and validated on the core of a fast reactor. The compactness and high energy density of a fast rector core restricts access and the normal operating conditions of a liquid metal coolant, high temperature and high neutron flux represent a particularly harsh environment for transducers. Typically, direct measurements of coolant inlet and outlet temperature, flow, and neutron flux are only available around the periphery of the core. To supplement the available information on the core behaviour, a range of novel instruments have been developed for fast reactors to provide ultrasonic temperature measurements, acoustic noise analysis, delayed neutron detection, and temperature and flux noise analysis.

Development of the TOPS algorithms to date have concentrated around using a simulation to mimic a fast reactor plant. However the promising results thus far obtained have encouraged us to evaluate their performance using pre-recorded plant data from the Prototype Fast Reactor (PFR) at Dounreay, Scotland.

We have also planned a demonstration of the real-time operation of the monitor capabilities of the TOPS system on PFR. This demonstration will validate the current simulation results and address the often unconsidered, yet important, engineering, safety and procedural requirements of attaching such a complex system to an actual full-scale plant. The first phase of the demonstration will emphasise the generation of robust state estimates with the diagnostic functions being added at a later date. Our initial objectives are to estimate, in real-time, the linear power rating and the linear power-to-melt margins of the fuel subassemblies.

7.0 Conclusions

A Total Process Surveillance system is under development which can provide, in real-time, additional process information from a limited number of raw measurement signals. This is achieved by using a robust model based observer to generate estimates of the process' internal states. The observer utilises the analytical redundancy among a diverse range of transducers and can thus accommodate off-normal conditions which lead to transducer loss or damage.

The modular hierarchical structure of the system enables the maximum amount of information to be assimilated from the available instrument signals no matter how diverse. This structure also constitutes a data reduction path thus reducing operator cognitive overload from a large number of varying, and possibly contradictory, raw plant signals.

The TOPS system provides a structured and complete means for plant data management under both normal and off-normal operating conditions and thus will significantly assist the operator during severe accident management.

8.0 References

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MARS (MAAP Accident Response System): On-line Software to Track and Predict (Faster than Real-Time) Plant Behavior Under Accident Conditions.

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ABSTRACT

Under actual severe accident conditions the plant operators, management, and those specific individuals responsible for managing the accident will be under considerable stress. The responsible individuals must determine what actions should be taken to correct the situation and how the plant will respond and with what consequences given the implementation of these actions. The timing associated with each of these functions (how long until core damage, etc.) will also be critical. In essence those individuals responsible for making decisions during an accident may be required to make important decisions when the plant is in a severely damaged state. However, with proper training and through the use of supplemental tools, those individuals will become more informed and confident to manage the accident.

The **M**AAP **A**ccident **R**esponse **S**ystem (**MARS**) was developed to provide those responsible for accident management with the much needed insights of the current and future status of their plant based on the current plant data and its trends. As an integral part of an accident response plan, the **MARS** software can be used to evolve and validate accident management strategies and to educate and train the accident management personnel. Furthermore, it can also serve as an accident management tool during such an actual event.

The **MARS** software uses the Modular Accident Analysis Program (MAAP) code as its basis to calculate the nuclear plant thermal-hydraulic and fission product response under accident conditions. The **MARS** software utilizes on-line data available from plant instrumentation (pressures, temperatures, water levels, system status, etc.) to initialize MAAP at any time during the accident (before

core uncovery, following core melt, etc.). Once initialized, **MARS** can then track the plant behavior and also allow for faster than real-time predictions to be performed. **MARS** performs several functions. These functions include:

1) Diagnosis of event and evolution of accident sequence,
2) Tracking plant behavior including impact of operator actions and changes in equipment status,
3) Correcting simulation such that it remains consistent with plant behavior as indicated by on-line plant data,
4) Performing root cause analysis,
5) Performing near term predictions based upon current plant conditions to assess impact of Emergency Operating Procedures (EOPs) or accident management strategies.

The following paper will provide an overview of the **MARS** capabilities and testing program.

1. Introduction

The **M**AAP **A**ccident **R**esponse **S**ystem (**MARS**) provides those individuals responsible for managing the nuclear plant under accident conditions, with a tool to obtain a considerable number of insights concerning the status of their plant. The **MARS** software uses on-line plant data to perform engineering calculations to track and predict both the thermal-hydraulic and radiological response within the plant during an actual or simulated accident. An illustration of a typical **MARS** installation is shown in Figure 1. In this configuration on-line nuclear plant data is transmitted to the accident management center where it is loaded into the **MARS** computer. Once the plant data is loaded into **MARS** (either automatically or manually), the **MARS** software can then be actuated to assess the current and future states of the plant.

In order to perform rapid calculations that assess the status of the nuclear plant under accident conditions a computer code which models all of the significant accident phenomena and is fast running is required. The **MARS** software uses the Modular Accident Analysis Program (MAAP) to rapidly assess the thermal-hydraulic and fission product (radiological) status of the nuclear plant under severe accident conditions. The MAAP code is widely used around the world for performing severe accident calculations to support Probabilistic Risk Assessments (PRA), plant studies and other related projects. The MAAP code is currently being executed on numerous computer platforms including personal computers (PCs), workstations and large mainframe computers. Even on the slowest of these current generation of computer systems (386 PCs) the MAAP code executes several times faster than realtime.

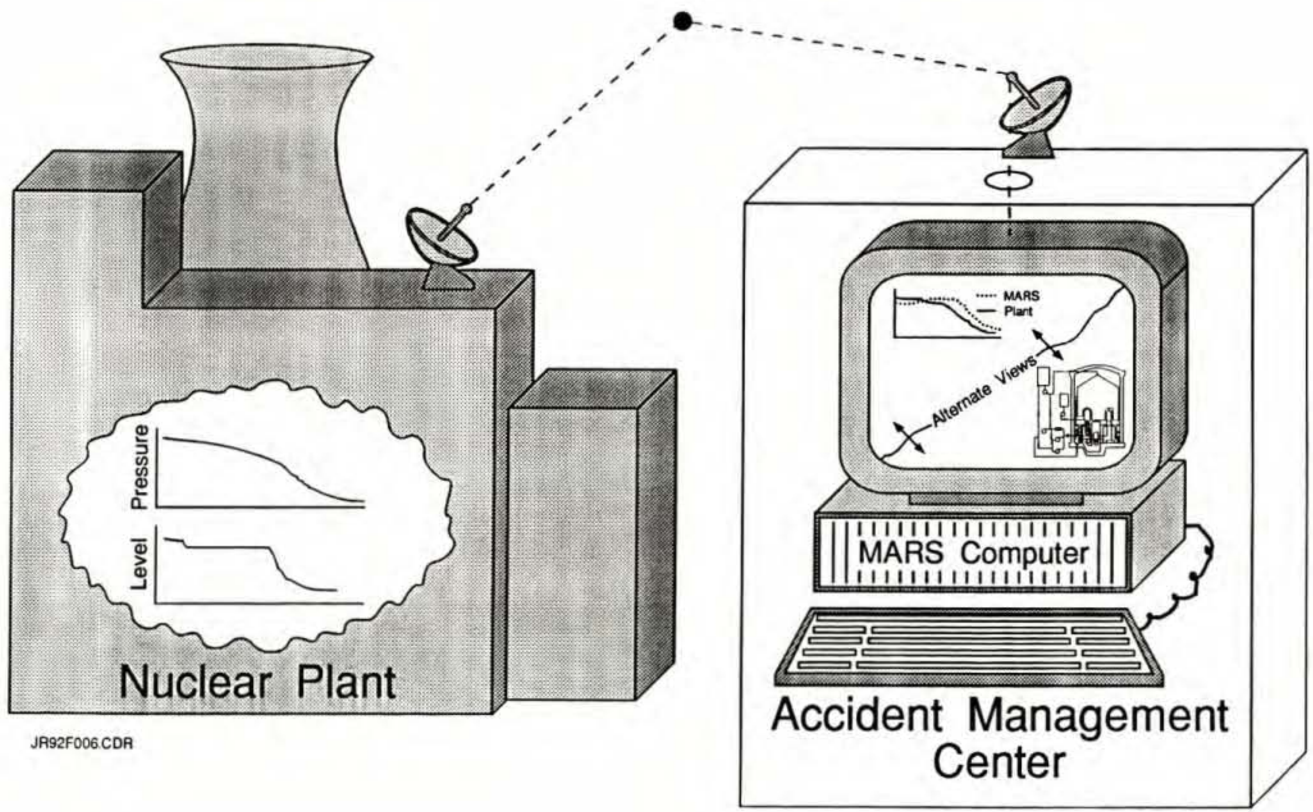

Figure-1 Typical MARS Installation

The **MARS** software, includes a significant amount of additional software beyond MAAP (as will be presented below). The additional software included in **MARS** does not have a significant impact on computer runtime, compared to the MAAP code. The faster the computer **MARS** is executed on, the quicker the turn around time.

2. **MARS** Capabilities

The **MARS** software has been developed with the understanding that those individuals involved in managing the accident will be under a considerable amount of stress. Thus, the features have been structured to be very flexible and permit easy operation. In addition, the **MARS** output provides the user with an easily understandable graphical representation of the status of the plant as well as being able to examine the intricate details of the plant conditions. An example **MARS** graphical plant display is provided in Figure 2 for a BWR Mark I type of plant and for a PWR Westinghouse Large Dry containment plant in Figure 3.

The **MARS** software is configured to operate in two modes: 1) Tracker and 2) Predictor. In the tracker mode, the **MARS** software is initialized based upon on-line plant data and then tracks and corrects the **MARS** simulation to follow the plant behavior. In the Predictive mode, once initialized **MARS** can perform faster than realtime simulations to determine the possible future status of the plant given an initial plant state.

In the **MARS** tracking mode, a dynamic assessment of the status of the plant is performed. In the **MARS** tracking mode the simulation proceeds only as fast as plant data is obtained (i.e., realtime). **MARS** can be initialized at any time during the accident progression. The potential times for such arbitrary initialization include:

1) Before Core uncovery,
2) Following core uncovery but before core damage,
3) After core damage but before vessel failure, and
4) After vessel failure.

Based upon any changes identified and/or internal **MARS** verification calculations, the **MARS** simulation is corrected to better represent the status of the plant. Comparisons of the plant data and **MARS** calculations are made to determine if agreement exists. If the comparison does not agree, the **MARS** simulation (for that plant data interval) is performed again making the necessary adjustments to converge upon a representative solution. If convergence cannot be obtained, re-initialization is performed. During the tracking mode, an assessment of the type of accident occurring at the plant and potential root causes is also performed. This information is used internal to the **MARS** simulation to track the accident and is also available external to the simulation to those individuals managing the accident.

In the **MARS** predictive mode, based upon the current status of

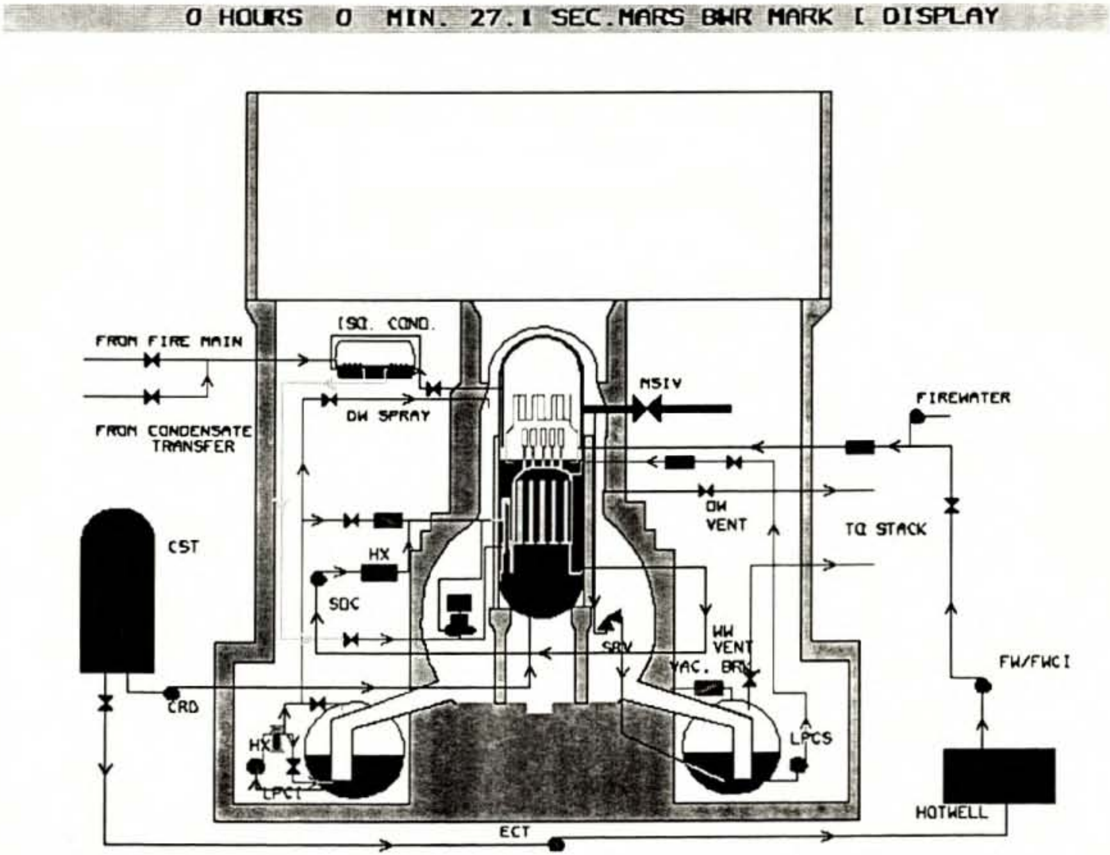

Figure-2 MARS BWR Mark I Graphical Plant Display

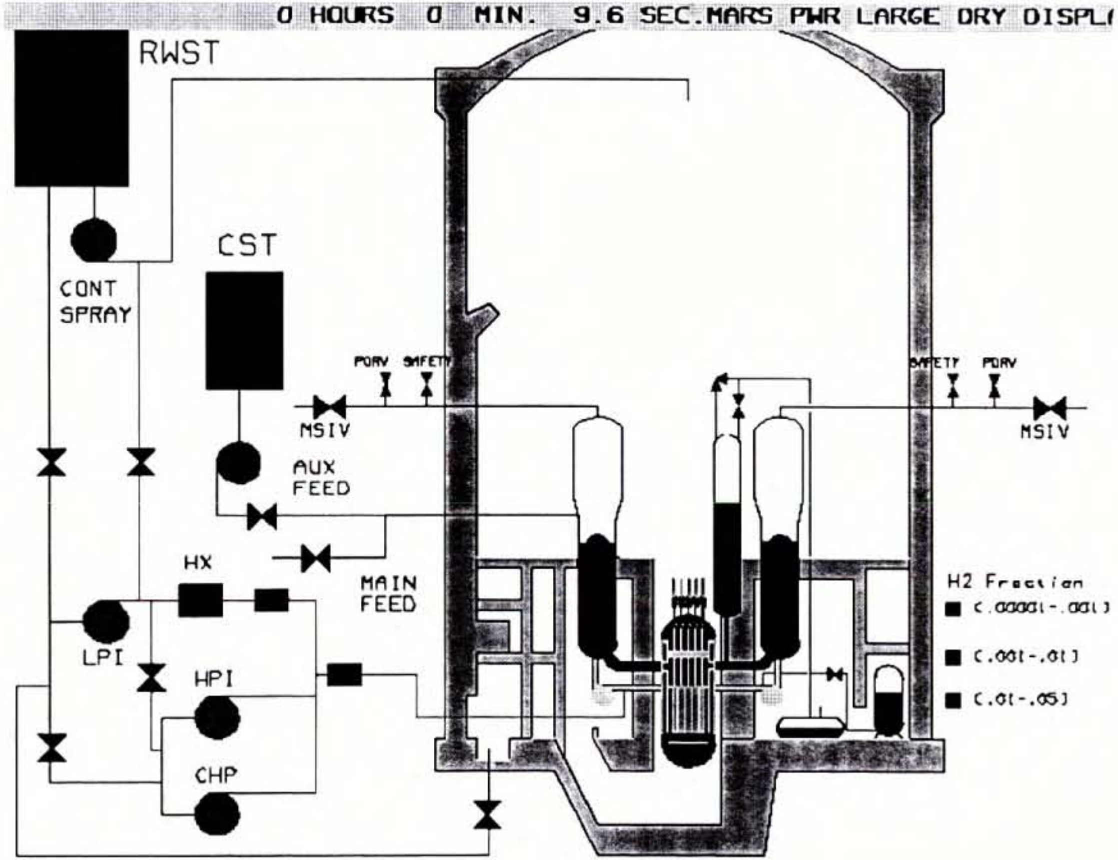

Figure 3 MARS PWR Westinghouse Large Dry
Containment Graphical Plant Display

the plant, faster than realtime predictions are performed to determine several possible future states of the plant. The future plant state predictions include an assessment of what will happen if no operator actions are taken, what will happen if the operators follow their emergency operating procedures, and how the plant would respond based upon any accident management guidelines. These types of simulations are very useful for helping decide appropriate and/or inappropriate actions during the course of the accident. In addition the potential consequences which may occur as a result of the accident can be determined and assessed well in advance of when they may actually occur.

The computer system configuration for **MARS** can vary widely depending upon the available hardware and the required system response. Several networked computers can be used for a fully functional **MARS** system. The networked computers could, as a minimum, consist of one computer for the Tracker and possibly three other computers for performing predictions. In addition, it is possible to run all four or a subset of the four applications, on the same computer in a multi-tasking (sharing the same computer CPU) environment. The power of the computer utilized to run **MARS** on and the resulting system performance will determine the feasibility of multi-tasking several **MARS** simulations (Tracker and Predictors).

3. **MARS** Testing

The **MARS** software has been tested in several plant data transfer and computer system configurations. Based upon the limited set of plant data required by **MARS** and each of the configurations, the **MARS** software has been able to successfully initialize and track the plant data. Once the Tracker is initialized, future predictions can be performed.

In order to test the **MARS** software, plant data during accident conditions is required. The majority of the plant data required by **MARS** has been generated via a real-time running computer simulation code called the MAAP Signal Generator (MSG). The MSG code uses the MAAP code as its basis for calculating the plant response but includes modules to simulate real-time plant data transfer. For instance, in one **MARS** installation, plant data is captured by the plant computer every 30 seconds but only transmitted outside of the plant (for instance to the Emergency Response Center) every 5 minutes. The MSG software was configured to simulate this plant data transfer method. In addition to using simulated plant data, **MARS** has been tested in a very limited scope by using actual plant data. The actual plant data utilization was limited to steady state analyses.

The MSG code uses the same code as **MARS** does (MAAP) for performing the plant accident response calculations. However, **MARS** only uses those variables generated by MSG which are readily available in the plant. The MSG software is used for the majority of the **MARS** testing since it is very fast running and models all of the important accident phenomena. In the future more testing is planned which will use actual plant data under accident conditions in addition to using other large computer codes to generate the simulated plant data.

The computer systems **MARS** has currently been installed on include 386 PCs and VAX workstations. In the 386 PC **MARS** configuration, the generation of the plant data (via the MSG) was performed on an IBM mainframe and then via network was transferred to the **MARS** PC. For this **MARS** installation, the use of the IBM mainframe for the MSG best represents the actual **MARS** installation since real plant data is currently received and maintained on the IBM mainframe. Thus, real plant data can be readily substituted for that generated by MSG. In the VAX workstation environment, several workstations were networked together, with each computer performing a separate function (MSG, **MARS** tracker, **MARS** predictor 1, etc.)

The **MARS** software has been tested for several accident scenarios including Station Blackout, Loss of All Injection and Small LOCAs. For each of these example scenarios, the **MARS** software can determine the probable cause of the accident. In addition, for the Small LOCA scenario, the **MARS** correction logic can successfully determine an approximate LOCA break size and location based upon the available plant data.

Included as Figure 4 is a plot of primary system pressure versus time for a BWR Loss of All Injection scenario. The figure has several curves plotted. The first curve (the solid line) represents the plant data (MSG generated) provided as input to the **MARS** tracker. The remaining curves, as noted in the plot legend, represent **MARS** initialization and start of tracking intervals at several different times during the accident progression. The key event times from the simulated plant data (MSG) are:

Core Uncovery occurs at:	1600 seconds,
Start of Core Melt starts at:	4000 seconds,
Vessel Failure Occurs at:	6700 second.

As illustrated by the plot, **MARS** can be successfully initialized and perform tracking during the various accident progression time intervals.

Included as Figure 5 is a plot of the primary system pressure response for a PWR Small LOCA scenario. The primary intent of this plot is to illustrate that **MARS** can be initialized and also determine an approximate LOCA size and location, based upon the limited set of plant data. The LOCA area and location were not input to **MARS** but rather were determined based upon the interpretation of the limited set of plant data.

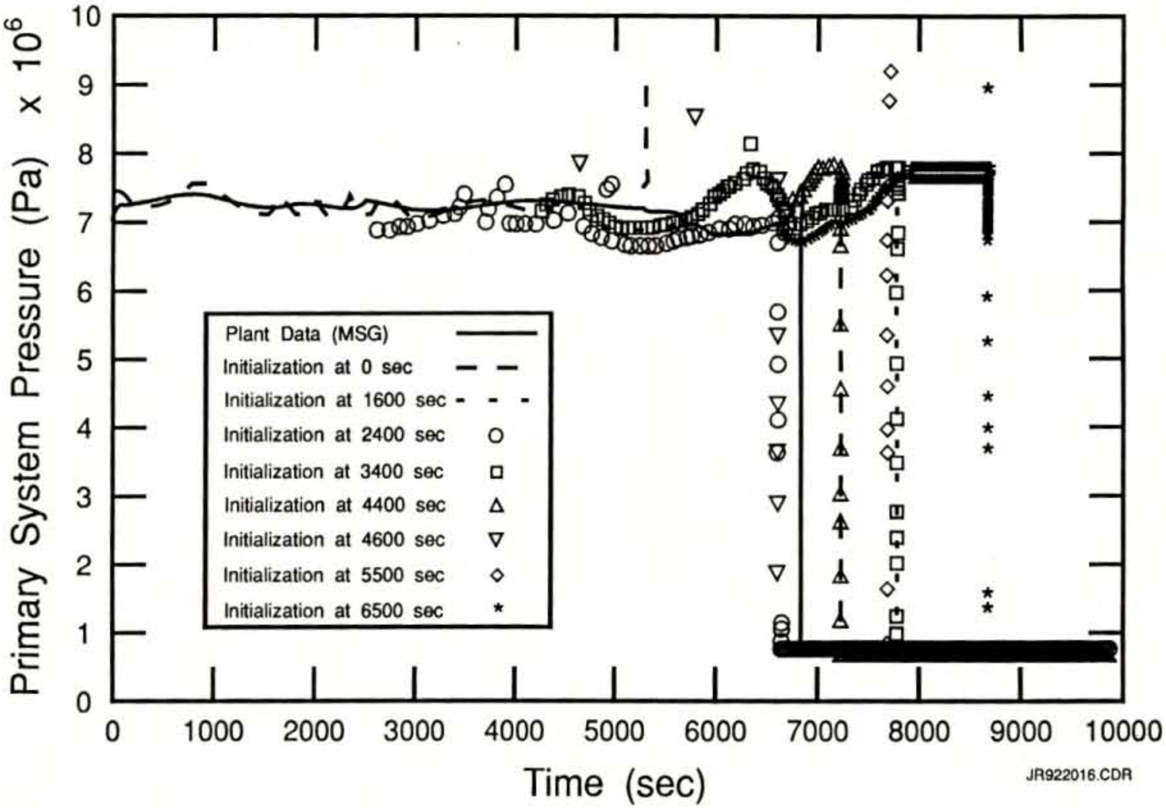

Figure-4 BWR MARS Results for a Loss of All Injection Scenario

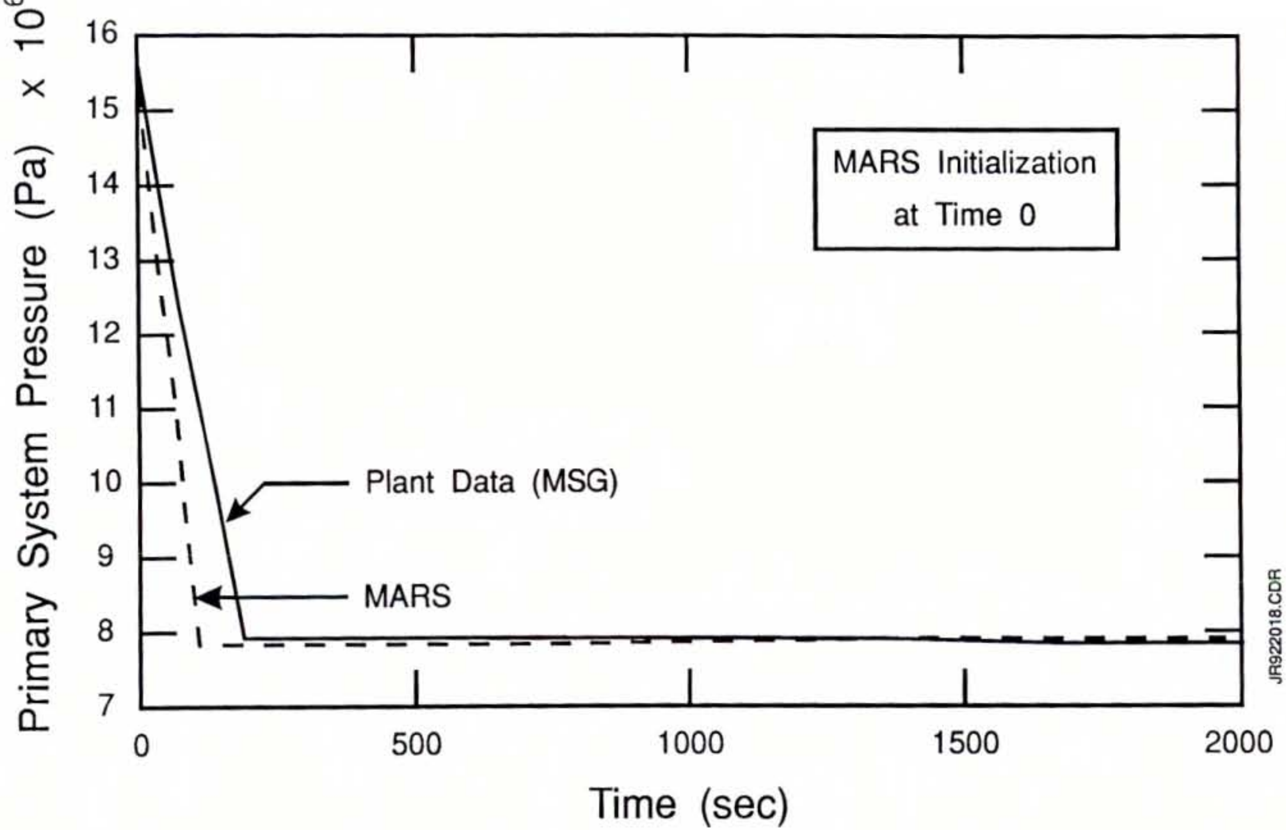

Figure-5 PWR MARS Results for a Small LOCA Scenario

4. Summary

The MARS software can provide those responsible for managing a nuclear plant accident with much needed insights to the current and future status of their plant based upon the current and evolving on-line plant data. The MARS software can be used to educate and train the accident management personnel and also serve as an accident management tool.

SUMMARY of SESSION IV

The session focused on the applicability of techniques in the domain of severe accidents. The subject treated can be split in three main categories:

- semantical*) concepts
- integrated applications
- operational tools.

These categories will be defined and discussed in the following.

The success of expert systems in other fields (like medicine, etc.) is largely due to the fact that large amounts of data have been gathered and represented as a homogeneous sets of "rules" that can be stored and retrieved in and by using modern computers. However, the intelligent selection, i.e. the systems analysis that has to be performed implicitly or explicitly, determines the structure of the resulting expert system. Thus, this structure varies from domain to domain, from one application to another; with regard to nuclear power: from one plant over to another.

Nevertheless, these structures are essentially semantical concepts, i.e. the attempt to describe implications in the problem domain.

Once such a semantical concept exists, it can be used to implement a special case (application). The way is to put the data for the application together and write a program to execute the concepts based on the data. However, as many changes, experiments, tests and further enhancements must be envisaged, so-called "shells" that represent the semantical concepts are programmed in which the data can be "fitted in". Such an approach is called an integrated application. One drawback, however, of today's integrated applications is that they require general concepts such as rules of the form "IF turbine is tripped THEN reactor will be tripped", which restricts the expressive power of such systems [you can only model (describe) what the concept yields!]. This implies that integrated applications can only be successfully used when the concepts already match the requirements completely.

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*) Semantics is the branch of linguistic research concerned with studying changes in meaning of words.

Many problems in many domains do not require very sophisticated inferences as the integrating element is the human being. Therefore conclusions are being drawn by the human and not by some concept. Here, the human must be supported by an array of tools (the sophistication of which may vary). Even though in the eyes of the "artificial intelligentsia" these tools are not fully accepted, they can be of great help especially in situations where the domain tends to break apart and can no longer be reviewed as a single entity with the conclusion that truly artificial intelligence strategies do not apply any more and become useless.

The presentation made by **Dr. Sergio Guarro** (Advanced System Concepts Associates, USA) clearly falls into the semantical concepts category. He describes the outline of AMAS (Accident Management Advisor System). The system is intended to have three levels:

- Parameter State Identification Filter,
- Plant State Identification Module and
- Management Action Decision Support Module.

Each of these levels use specific concepts. While level one strongly relies on the so-called Logic Flow Graph Methodology (LFM) to track key parameter interaction and consistency, level two refers to Accident Progression Trees (APT) and Bayesian Belief Networks (BBN). The top level associates plant conditions with accident consequence minimization schemes. The system is in the conceptual phase. A working prototype is to be developed in the second phase.

Also in this category was the paper presented by **Mr. Paul Millar** (AEA Technology, UK) about a system called TOPS (Total Plant Surveillance). The TOPS concept is also hierarchical and roughly divided in two sections at the second level, monitoring and diagnosis. Diagnosis provides on level three incipient fault detection, full fault detection and identification, post-fault status, and post-fault behavior prediction. The plant monitor uses an observer based on a mathematical model of the process. In essence, measured data are compared to computed (model-inferred) data. Diagnostics are performed on several plant levels and plant areas such that the computation-intensive parts are preferred on the lower level in a distributed environment (i.e. in parallel). In this way it is also claimed that real time operation of TOPS can be achieved. The system is currently under development.

The paper presented by **Professor Rainer Hampel** (TH Zittau, Germany), titled "Model-based Correction Algorithms", described model-based measuring methods, which.could be used for the reconstruction of non-directly measurable variables and therefore may contribute to the diagnosis of the complete system state, the realization of state controllers, or fault detection. The results of the methods and algorithms developed for the case of the hydrostatic level measurement on horizontal steam generators have been compared with experiment data on pilot plants and ATHLET calculations. Such methods are expected to be used in the future, after further qualification, on power reactors.

A more general paper on the use of Artificial Intelligence (AI) for operator support systems was presented by **Dr. Michel de Vlaminck** (Tractebel, Belgium). It described OPA (Operator Advisor), which is an expert system of the category "integrated applications". OPA's main advantages are:

- on-line access to a flexible and adaptable knowledge base,
- post-accident monitoring of operator actions,
- priority listing of all needed actions and
- the availability of background information.

The system heavily relies on rules in the classical sense. However, the system has capability to react to events and has elements to treat time-dependent actions. Presently, the system is implemented on a Symbolic LISP-machine, though chances are, the system will be available on regular workstations in the near future. The system is used in the reactor operator training center.

To satisfy the third category listed in the beginning **Mr. James Raines** (Fauske & Associates, USA) gave a presentation on MARS (MAAP Accident Response System). This system uses the Modular Accident Analysis Program (MAAP) code to calculate the nuclear plant thermal-hydraulics and fission product response under accident conditions. MARS uses on-line data from plant instrumentation to initialize MAAP any time during the accident.

MARS then is able to perform a variety of functions. Among them are:

- diagnosis of event and evolution of accident sequence,
- tracking plant behavior and operator actions as well as their impact, ensuring consistency of simulation,
- performing root cause analysis and near-term predictions.

MARS operates much faster than real-time. A typical set of computation times was given for a forty-hour scenario ranging from about one hour (on a 486 PC) to seven minutes on an optimized HP workstation.

A very lively discussion was going on at the end of the session, focussing on subjects like:

- "What is necessary for the operator to infer the plant status during a severe accident?"
- "Is no information better than incomplete or erroneous information?"
- "How can such artificially intelligent systems be validated?"
- "Is it not most important to avoid information overload?"

Whilst most of these issues were discussed rather controversially, the opinion was rather unanimous that if a severe accident happened at all it would not quite match any of the predicted scenarios.

Validation of AI systems is important. Bad information is worse than no information. With bad information, plant staff may try to do something based on it, and could make things worse. If the plant staff know that they have no good information they can make attempts to remedy that. Thus, AI systems should produce validated "good" information or indicate that no good deductions can be made. A participant suggested that the further the accident proceeds, the simpler should be the analytical attempts to analyze and understand it. This appears very consistent with the validation limitations for severe accidents.

Some participants were extremely worried about the use of computers in carrying out high-complexity inferences during a severe accident as the impression prevailed that this may eventually discard the operator. However, the general consensus was that the last decision-making will always be human responsibility; only the degree of support (and its necessary complexity) that such approaches may yield was divisive.

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