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PLANT SYSTEM UTILIZATION FOR ACCIDENT MITIGATION

REPORT OF A TECHNICAL COMMITTEE MEETING ORGANIZED BY THE INTERNATIONAL ATOMIC ENERGY AGENCY AND HELD IN GARCHING, GERMANY, 26–30 NOVEMBER 1990

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WORKING MATERIAL

TECHNICAL COMMITTEE MEETING

ON

PLANT SYSTEM UTILIZATION

FOR ACCIDENT MITIGATION

Garching, Germany,

26-30 November 1990

Chairman: H. Willemsen, The Netherlands

Scientific Secretary: H. Mauersberger, IAEA

International Atomic Energy Agency

Vienna, Austria 1991

Acknowledgement

By the invitation of the Government of the Federal Republic of Germany the Technical Committee on Plant System Utilization for Accident Mitigation was held at the Gesellschaft für Reaktorsicherheit in Garching, 26-30 November 1990.

The International Atomic Energy Agency wishes to express its gratitide and appreciation for the invitation, and for all the support, co-operation and hospitality received.

Content

SUMMARY

During the week of 26-30 November 1990, the International Atomic Energy Agency, in co-operation with the German Gesellschaft für Reaktorsicherheit mbH (GRS) organized at the GRS Forschungsgelände in Garching, Germany, a Technical Committee Meeting on Plant System Utilization for Accident Mitigation. The objectives of this meeting which is part of a series of meetings to develop a user's manual on accident management in nuclear power plants, were to review and assess anticipated system utilization for the mitigation of the consequences of severe accidents and the implementation of corresponding guidelines to the operating and support staff.

Strategies to prevent and mitigate severe accidents will be conditioned by the availability of first line and support systems in the plant to perform the necessary functions. Therefore, as part of the preparation for severe accident management it is necessary to identify all plant systems (including non-safety related systems) that could be used, perhaps in a non-conventional manner, to control the accident and mitigate its consequences. This should include the identification of back-up systems which could be used to perform the same functions. Another aspect concerns other equipment and systems that might have to be obtained form another part of the site or elsewhere. For example, it may be possible to use a non-standard water source to provide long term cooling to the reactor core, or special equipment may be needed to get firemen close to the scene of a fire and to protect them from high radiation levels or contamination. The availability of such systems or equipment needs to be considered at the planning stage, as well as the means of transport that may be necessary in the event of a rapidly developing accident. In some cases, it may be desirable to install additional equipment to make a strategy feasible. Examples are filtered vents and hydrogen igniters which have been introduced into nuclear power units in some Member States.

The purpose of this meeting was to review and assess the current status and future trends in the use of available and/or additional systems to prevent and mitigate severe accidents.

The meeting was opened by Mr. K. Wolfert, Head of the Thermalhydraulics Department of the GRS who stressed the importance of accident management measures and the necessity of international co-operation in the field.

The meeting was attended by 25 participants from 10 countries. The participants presented technical papers on the subject and provided comments for the preparation of a draft report on the use of plant systems for accident management. Based on a Polish/USSR proposal, a few members of the TC proposed a scheme to generalize systems utilization for different reactor types. This proposal will serve as basis for the completion of the draft report on the subject which should then be evaluated by independent consultants to produce the final daft.

National Presentations

A review of the approach to particular aspects of severe accidents in LWRs in the European Community including Sweden and Finland
J. Harrison, P. Bacher, Y. Dennielou
(France)

General principles taken into account in France for the use of systems in the operating documents for accidental situations (including severe accidents) on PWR plants
S. Guieu, R. Soldermann
(France)

Accident management (AM) strategies for LWR in the Federal Republic of Germany
E. Kersting
(Germany)

Emergency actions (accident management actions) in German BWR's
H. Ohlmeyer
(Germany)

Secondary and primary bleed & feed, an AM-strategy to prevent and mitigate severe accidents
B. Pütter, et al.
(Germany)

Containment venting for BWR and PWR
J. Rohde, M. Tiltmann
(Germany)

Hydrogen mitigation measures
J. Rohde
(Germany)

Accident mangement possibilities for VVER-440
G. Lajtha
(Hungary)

Utilization of control rod drive (CRD) system
for long term core cooling
A. Huerta Bahena
(Mexico)

Design objectives on filtered venting and hydrogen
management for the Borssele nuclear power station
J. E. Speelman et al.
(The Netherlands)

Preparedeness of the nuclear centre Swierk and its
nuclear facilities for emergencies
J. Koziel
(Poland)

Accident mitigation measures for VVER-440 reactors
in the case of chosen severe accidents
A. Strupczewski et al.
(Poland)

Effect of passive and active elements of the safety
system on the moment of core destruction in several
out-of-design accidents in VVER-1000 reactors
A. M. Shumskij et al.
(USSR)

New technical solutions elaborated in VVER reactor NPS designs on preventing severe accidents and their control
V. Y. Shenderovich
(USSR)

Accident mitigation on NPP with VVER-1000
P. D. Slavjagin et al.
(USSR)

Utilization of the systems of power plants with RBMK reactors to mitigate accident consequences
V. P. Vasilevskij et al.
(USSR)

I A E A

TCM = SYSTEME UTILIZATION FOR ACCIDENT MITIGATION

GARCHING - RFA

26 - 30 NOVEMBRE 1990

NUCLESUR - UNIPEDE

A REVIEW OF THE APPROACH
TO PARTICULAR ASPECTS OF SEVERE ACCIDENTS IN LWRs
in the EUROPEAN COMMUNITY
including SWEDEN and FINLAND

* * *

Authors

John HARRISON	NUCLEAR ELECTRIC	UK
Pierre BACHER	EDF	France

Presented by *Mr Yves DENNIELOU* (EDF) France

NUCLESUR

A REVIEW OF THE APPROACH TO PARTICULAR ASPECTS OF SEVERE ACCIDENTS IN LWRs

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John HARRISON	NUCLEAR ELECTRIC	UK
Pierre BACHER	EDF	France

I- GENERAL SAFETY CONSIDERATIONS

Nuclear safety is defined in the IAEA's nuclear safety standards (NUSS) as "achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and the environment from undue radiation hazards".

Three basic functions must be ensured if those objectives are to be attained :

- the chain reaction must be controlled,
- the reactor core must be cooled,
- the radioactive fission products must be kept confined inside the power plant.

Power plant design and operation is based on the **fundamental principle** that the greater the consequences for the public of a hypothetical accident, the lower must be the probability of it actually occurring. That principle is satisfied by means of a **dual** approach :

- prevention or, if total prevention is not possible, reduction of the probability of an accident,
- mitigation of the consequences of an accident, should the preventive measures fail.

Three complementary methods have been developed, all of which comply with those general principles : the **deterministic** and **probabilistic** approaches (which are in general closely linked) and the systematic use of feedback from experience in order to identify and correct weak points, thus reducing risks.

The practical implementation of these methods necessarily involves drawing up ground rules that set limits for the precautions to be taken (earthquake intensity that the plant must withstand, probability below which it is not considered necessary to take any particular precaution ; feedback from experience that would be taken into condideration only for future plants, since it would not result in any significant improvement in the safety of existing plants. etc). Such grounds rules, which reflect the ALARP (As Low As Reasonably Practicable) concept, are laid down by the safety authorities in each individual country. They vary slightly between different Members States, but are remarkably homogeneous on the whole, while the few differences that do exist can be ascribed to local factors.

Aware of the complexity of the physical phenomena taking place inside a nuclear reactor, the research bodies concerned have always been conducting major theoretical and experimental studies with the dual aim of more precisely identifying the margins introduced in accident prevention and gaining a better understanding of the physical, and more particularly radiological, consequences of potential accidents.

Since operators play an essential role in plant safety, it is universally acknowledged that they must both be very highly trained and have a genuine safety "culture" and that the resources placed at their disposal must be capable, up to a point, of ensuring the safety of the plant even in the event of human error or inadequate fault diagnosis.

Work by the UNIPEDE Safety Working Group (NUCLESUR) and the Working Group n° 1 on Reactor Safety has made it possible to stress the consensus between the Member States and the European Commission on the meaning and value of the harmonization of safety within the Community.

A large measure of agreement has thus been shown to exist on the most important points, namely general safety objectives and principles and the above mentioned methods of obtaining those objectives and complying with those principles.

The membership of WG1 has also allowed wide-ranging exchanges of views to take place concerning the - sometimes different - technical means used in the different countries and the domestic organization of each country enabling the **internal consistency** of the means chosen to be ensured. Discussions of those means have been extremely fruitful in the past and will not doubt continue to be so in future.

Lastly, the activities of NUCLESUR and WG1 have clearly reflected the increasing atttention devoted to severe accidents, more particularly in the wake of the Three Mile Island accident and the Chernobyl accident. They have revealed that a great deal of work is continuing in the different countries and that exchanges of information and discussions are still needed in order to harmonize approaches to mitigation measures and plant operating practice.

The main effort should now focus on severe accidents management. All existing nuclear power plants were built according to designs incorporating **considerable margins** over and above the design basis accidents ("umbrella" assumptions and margins in design codes). Such margins make it possible to mitigate the consequences of severe accidents. This paper presents a state of the art report by NUCLESUR on accident management in PWR plants.

II- REVIEW OF THE APPROACH TO PARTICULAR ASPECTS OF SEVERE ACCIDENTS IN LWRs

2.1 - Introduction

This review follows the more detailed and wide ranging review by NUCLESUR of the approach to both design basis and severe accidents in LWRs which was completed in 1988. Since this earlier report showed a lack of uniformity in relation to severe accident aspects a decision was made to reconsider this area. A NUCLESUR study group on severe accident management was set up to perform this task. The objectives of the review report produced by this group were to consider in more detail the general approach to severe accidents and by considering how specific issues are addressed within this framework, to determine any underlying similarities and differences in approach.

The countries included in this review are Belgium, Finland, France, the Federal Republic of Germany, Spain, Sweden and the United Kingdom. Some details of the position in the USA are also noted since historically this country has had a strong influence on the practices in Europe. A summary of the position in the Netherlands is included in the tables only.

The approach to severe accidents and severe accident management can be seen to vary between countries depending on the existence of regulatory requirements, public opinion and the source of the technology which designed, built, and may still be supporting, the generating reactors.

The historical approach to severe accidents has been to design reactor plant such that severe accidents can be assumed not to occur. That is they were considered to be incredible and little or no design provisions were included to address such conditions. Off site emergency measures, if any, were left to the care of regulatory authorities.

The emphasis on severe accidents over the past decade has increased with the occurrence of the accidents at Three Mile Island and Chernobyl. Although the Nuclear Community consider these to be at opposite ends of the spectrum in relation to both the safe design and operation of nuclear power plant, they have both had large if different influences and effects on the industry and on the public at large.

Analysis in the severe accident field has been twofold. On the one hand, assessing the many severe accident phenomenological aspects has provided the tools to predict the capability of existing designs to cope with such conditions, to enable the inherent safety margins in the designs to be taken into account or to assess the effect of possible design or operational changes postulated to mitigate the effects of severe accidents. On the other hand, predictions of the probability of occurrence of such conditions have been made to provide further information to decide on the importance of severe accident phenomena.

These aspects are usually considered together in a balanced way via Probabilistic Safety Assessments (PSA). In any case they provide the tools to enable decisions on the need for further research or design or operational changes to be made against existing designs and regulatory requirements in each country.

This summary paper concentrates on severe accident aspects which are important in ensuring that the containment does not fail in the long term so that benefit can be taken of the margins inherent in the containment design. The items considered are hydrogen control in the containment, containment venting and the assessment of the safety of containment. Generally, loadings which occur in the short term, or are of short duration such as direct containment heating, are not included, since design measures or (and) operating procedures are very generally implemented to prevent such short term containment failures.

2.2 - The Approach to Specific Severe Accident Issues

The general approach to severe accidents can be seen to depend on regulatory requirements or self imposed safety criteria as summarised in Table 1.

PSA

The role of PSA varies between countries and has been used as a starting point for severe accident considerations in many countries (Belgium, RFA, UK) or is being used to complement the severe accident programme (France, Sweden, Finland) or will now be required or included as part of licensee renewal reports (Spain). (Table 1).

The total scope of severe accident programmes has expanded over the last few years but these are generally well underway in most countries and are now considered to be complete in Sweden. The level 1 PSA studies for both Finnish plants are completed, except for external events. The planned level 2 and 3 studies will continue for some years. Belgium and Spain expect their programmes to be completed in 1991 whilst work in the UK should be finalised with the Sizewell B POSR in 1992. In the FRA, the overall PSA based on Biblis B as reference NPP has been finalized in June 1989. On this basis, an individual PSA (mainly level 1) has to be performed to each NPP within the next ten years. In the USA the Individual Plant Examination (IPE) programme, assessing the vulnerabilities of all operating plant to severe accidents, should also be completed in 1992. The French 900 MW and 1300 MW level 1-PSA are now complete. The evaluation of the lesson of these PSA for future plants is presently underway and should be completed this year.

Hydrogen Control

Systems for hydrogen control are generally included for design basis accidents in all countries. (Table 2). BWRs are generally inerted which prevents problems with hydrogen control during severe accidents. The Spanish BWR design with the Mark III containment type has electric igniters fed by batteries to cope with hydrogen concentrations above 4 %.

The severe accident mitigation programme (containment venting, etc) is completed for the Finnish BWRs. The mitigating measures for the PWRs are not foreseen to be completed before 1991.

In general PWRs with large dry containments do not have and do not require specific measures for hydrogen control under severe accident conditions. Some design specific problems exist with German PWRs which have spherical steel containments and no containment spray systems. Suitable igniter and recombiner systems are therefore being assessed prior to installation. Belgium has investigated possible igniter systems for severe accident purposes but no decision or requirement to install them has been made. Other countries are re-examining their existing designs but have not identified the requirements for further measures to be taken.

Containment Venting

In both Sweden and France the decisions to install filtered vented containment systems were made to fulfil regulatory requirements. The marked difference between these systems relate to these specific requirements.

In Sweden a sophisticated and efficient multi venturi scrubber system has been installed for some BWRs and all PWRs, where, as the Barseback BWR units are furnished with a gravel filter. The decontamination factors for these BWR and PWR designs are between 100 and 1000. In France a simple system has been chosen since the requirement was to reduce releases due to late containment failure by a factor of ten to fit in with existing emergency plans. A full scale test has shown the decontamination factors are significanly higher than the requirement.

Germany's decision to implement FVC systems arose as a direct result of the Chernobyl accident. Hydrogen is a potential threat to their multi venturi scrubber venting systems and is currently being considered.

Other countries which are considering the need for such systems are addressing the requirements as part of their integrated severe accident programmes. Plant specific PSAs are generally used to determine such requirements as well as the need for any other severe accident measures. (Table 2).

It is also worth noting that this is also the case in the USA where decisions on FVC will not be made until the IPE programme is complete.

Assessment of Containment Safety

Containment safety and integrity under severe accident conditions is recognised to be of major importance to all countries. Most countries have assessed their own containment designs using less conservative methods than required for design purposes. This has shown significant margins to failure which gives assurance of the beneficial effect of the containment under severe accident conditions. Margins 2 - 5 times the specification have been predicted. (Table 2).

Instrumentation and Procedures

The requirements for instrumentation and procedures to specifically cater for severe accident conditions arise from the overall severe accident strategy and programme in each country. Sweden, Finland for the BWR units and France have implemented their necessary requirements. Germany is currently implementing severe accident procedures and the necessary instrumentation.

Belgium, Spain and the United Kingdom expect their existing instrumentation to be adequate for this purposes. (See Table 2).

III- CONCLUSIONS

The role of severe accidents in each of the countries reviewed in this report can be seen to have taken on a greater emphasis in the last ten years. The adequacy of the many different design features have been or are being assessed to determine their ability to cope with the demands set by severe accidents for which they were not originally designed.

Some design changes have been implemented to ensure the adequacy of these designs to meet regulatory requirements which now exist in each specific country, and to provide specific mitigation for some severe accident conditions.

The role of the operator and accident management in both preventing and mitigating severe accidents are other important areas where increased emphasis has been placed in recent years. For example this has identified the need to determine the actual plant conditions during fault conditions such as the margins to failure of the fuel rods, primary system and containment during fault conditions.

The significant role of the containment under severe accident conditions is recognised in every country. For extremely severe accidents beyond the design of the containment, mitigation measures such as containment venting can be seen to add to this role as they increase the prospects of maintening the integrity of the containment.

The status of PSA has been seen to increase in all countries in line with the growing awareness of severe accident considerations. Completion of the many plant specific PSAs that are currently underway, including the IPE Work in the USA, should provide a useful benchmark against which to assess the effect of design specific differences, operating procedures and regulatory requirements in each country.

Completion of the current severe accident programmes in each of the countries reviewed are expected by 1992. This does not imply that all work in these areas will cease. It is seen to be the starting point for ensuring the continuing safe operation of existing plant taking into account the possibility of appropriate measures for preventing unacceptable consequences of such situations.

For example, there are other aspects of severe accidents and in particular severe accident management outside the scope of this review which are currently being considered. These include measures such as water addition to degraded or degrading cores in or ex-vessel (to which existing methods are being applied) and the need for further analysis tools and refinements to current designs or methods of operation being determined. These are further examples of severe accident aspects where requirements are seen to be country and design specific.

Last but not least, it would appear important to arrive at a consensus within the Community on how to approach and present to the public the twin aspects of the prevention of severe accidents and the mitigation of their consequences. To simplify matters to the extreme, two trends are emerging : one is to incorporate sufficient margins to prove that there is no need to be concerned with severe accidents whose probability is very low (less than 10^{-6}/year) or their consequences, no matter what human errors should be made, while the other approach is to ensure that the consequences of any such accidents would be strictly limited.

REFERENCE :

- **Convergence of approaches to safety in the Member States of the European Communities**
  P. BACHER and J.P. PELE - NUCSAFE Conference AVIGNON 1988

- **Nuclesur Report 1989 "A Review of the Approach to Particular Aspects of Severe Accidents in LWRs**

Table 1 General Severe Accident Requirements

Country	Number and types of plant designs	Regulatory requirements	License Renewal/ Safety Reviews	Status of PSA	General Status of Severe Accident Programmes
Belgium	7 PWR, large dry double containment type.	Follow US guidelines.	10 year license review/renewal some input from PSA.	Doel 1, 4, level 1 PSA to be complete in 1990. (Internal/external events). Tihange 2 to begin next year. For other units to be completed for the 10 year review.	PSA/severe accident mitigation work to be complete mid 1991.
Finland	2 Soviet design of PWR, large ice condenser type containment. 2 BWR, pressure suppression type containment	YVL Guidelines: 1.0 General Design Guidelines. 2.2 Accident Analysis (Including Severe Accident Aspects). 2.8 PSA (In relation to the use of probabilistic methods).	Ongoing safety reviews. End 1991 for severe accident mitigation measures.	Level 1 PSA. Complete for both types of unit. (Internal events) Being used to identify where extra preventive measures are beneficial to increasing safety	Complete for BWRs (including design changes and emergency operating procedures) 1991 for completion of PWR aspects.
France	57 PWR: 1 900/910 MWe, large dry single containment type. 1300/1300 MWe, large dry double wall containment.	Overall safety goal of 10[illegible] for 'unacceptable' consequences.	10 year safety reviews are now being implemented.	PSAs for the 900 and 1300 MW designs are due for completion by the end of 1989.	

Appendix 1

Membership of the Working Group "Approach to Particular Aspects of Severe Accidents in LWRs"

[illegible].R. Harrison	CEGB, London
E. Boswick	CEGB, London
M. Delalande	EdF, France
R. Espefalt	Vattenfall, Sweden
M. Lienard	Tractebel, Belgium
J. de Santiago	Central Nuclear Valdecaballeros, Spain
H. Tuomisto	Imatran Voima OY, Finland

Table 1 - General Severe Accident Requirements (cont'd)

Country	Number and types of plant designs	Regulatory requirements	License Renewal/ Safety Reviews	[illegible] PSA	[illegible] status of Severe Accident Programmes
Germany	26 nuclear plants 19 PWR, large dry spherical steel containment (no containment sprays) 7 BWR, pressure suppression type containment	No formal regulatory requirements.	based on the overall PSA an individual PSA to each plant has to be performed within the next 10 years	The overall PSA (Deutsche RisikostudieB) which the reference NPP BiblisB has been fina-lized in June 1989	Severe accident progr[illegible] an accident management measures will be disc[illegible] with the authorities. The implementation is underway.
Spain	9 nuclear plants. 7 PWR, large dry containment type 2 BWR, pressure suppression type containment	Follow US regulations. Spanish Regulatory Body (CSN) requirements are included in the 'Integrated PSA Programme' which requires plant specific PSAs	Ongoing safety reviews. PSAs required for 10 year reviews	S. T. Garona, BWR, level 1 PSA complete C. N. Almaraz) PWR C. N. Asco) C. N. Cofrentes, BWR level 1 PSA nearly complete	Severe accident vulnerabilities and accident management being considered Completion of R&D programmes is expected in [illegible]
Sweden	[illegible] PWR, large dry containment type. 9 BWR, pressure suppression type containment	1980 - Barsebeck site. 1986 - All other sites.	Yearly requirement to report develop-ments in severe accidents to regulatory body Ten year safety reviews to cover operating experience. First ten year review to cover level 1 PSA for internal events	Level 1 PSAs essentially complete for all plants. External events/ level 2 work complete for some and underway for others within the longer term programme	All severe accident design and operating aspects were completed in 1988 as required for regulatory requirements

Table 1 - General Severe Accident Requirements (cont'd)

Country	Number and types of plant designs	Regulatory requirements	License Renewal/ Safety Reviews	Status of PSA	General Status of Severe Accident Programme
United Kingdom	1 PWR under construction, large dry containment type	No formal regulatory requirements. CEGB Design Safety Criteria include targets of 10^{-6} for uncontrolled release to ensure fundamental criteria of 10^{-6} risk to the individual is met. ALARP requirements.		Level 1 PSA (internal events) complete at [illegible] stage (WCAP [illegible] 1987). Level 3 full scope PSA (internal, external, operator error, hazards and incredible faults) to be completed for POSR [illegible]	PSA and severe accident management programme underway. Ongoing R&D to support all severe accident work. POSR - [illegible]
Netherlands	1 PWR, large dry double containment type. 1 BWR, natural circulation with pressure suppression type containment.	Overall safety goal of 10^{-6} for large releases. Specific requirements regarding beyond design basis accidents.	Integral safety review including PSA energy ten years (level 1 and reduced level 2).	Level 1 and a limited level 2 to be completed by 1991.	PSA/severe accident management modifications and programs to be completed 1991/199[illegible]

Table 2 - Specific Severe Accident Aspects (cont'd)

Country	Hydrogen	Filtered Vented Containment	Containment Safety	Instrumentation and Procedures for Severe Accidents
United Kingdom	No design provision for severe accidents. Analysis being repeated for POSR will address all sources of hydrogen but it is not expected to be a problem. Ongoing research being monitored in case such requirements are identified.	FVC not installed and not expected to be required. Decision to be underwritten prior to operation.	Calculations using design codes give a safety factor of 2 x design pressure. Model test gave a factor of 2½ x design pressure. More realistic assessment of survivability under severe accident loadings is underway.	No specific instrumentation for severe accidents. Survivability of existing instrumentation is being considered. Station operating procedures, including response to severe accidents, are being written.
Nether-lands	BWR is inerted. In both BWR and PWR recombiners can be used. Systems for early ignition for the PWR are studied.	FVC installation is expected in 1992.		Studies on the need for specific instrumentation are underway. Severe accident procedures will be drafted in line with the whole severe accident programme which has to be completed before 1994

Table 2 - Specific Severe Accident Aspects

Country	Hydrogen	Filtered Vented Containment	Containment Safety	Instrumentation and Procedures for Severe Accidents
Germany	69 series BWR are being inerted. 72 series BWRs. - system based on inerting only the wet well is developed. PWRs - automatic systems for early ignition and containment recombination being developed.	Venting systems obligatory. Further work required as hydrogen is a potential threat to the installed system.		Instrumentation and procedures for FVC and RCS depressurisation being implemented.
Spain	1 BWR is inerted (Mark I). 1 BWR has electric igniters fed by batteries to cope with concentrations above 4% (Mark III). PWRs - no specific design provision for severe accident aspects.	PWRs - decisions are awaiting results of IPEs. BWRs - awaiting regulatory bodies response to design and operation study.	Studies for BWRs have been performed which indicate gross failure capability of 3.5-5 x design pressure. For most PWRs calculations are currently being performed.	No specific instrumentation for severe accidents. Existing instrumentation is expected to be adequate for this purpose.
Sweden	BWRs are inerted. PWRs - analysis to date shows no additional provision necessary for severe accidents. Ongoing research being monitored in case such requirements are identified	FVC installed for all units.	Gross failure capability has been established but pressure will be controlled to just above design capacity.	All requirements for instrumentation and emergency procedures for severe accidents are in place.

Table 2 - Specific Severe Accident Aspects

Country	Hydrogen	Filtered Vented Containment	Containment Safety	Instrumentation and Procedures for Severe Accidents
Belgium	No design provision for severe accident aspects. No decisions made on the need or requirements specified.	FVC not installed, following US requirement. Possible system identified, decision on installation awaiting results of level 1 PSAs.	Best estimate calculations for gross failure under pressure and temperature loadings indicate a safety factor of 2.25 x design spec.	Instrumentation and procedures being considered in the overall severe accident programme
Finland	BWRs are inerted PWRs have igniters for severe accidents, adequacy of these is be investigated	BWRs have FVC. PWRs are planned to have external containment sprays.	Less conservative calculations show an ultimate pressure loading of 2 x design for gross failure.	Instrumentation added for PWR [illegible] containment pressure and level measurements. Instrumentation design for PWRs is being considered in support of design changes for severe accidents.
France	No design provision for severe accident aspects. Further calculations underway. The need for further countermeasures has yet to be identified. State of the art report being finalised.	FVC installed. Ongoing research on outstanding problems with the current system.	Collapse pressure of: 3 times design for the 900 MW designs (lined) >2 for the 1300/1400 MW design (double walled)	U and H procedures in place and necessary supporting instrumentation/design changes (eg US for FVC)

INTERNATIONAL ATOMIC ENERGY AGENCY

Technical Committee Meeting on Plant System Utilization for Accident Mitigation

Garching, Federal Republic of Germany, 26-30 th november 1990

Référence : J7-TC-744

GENERAL PRINCIPLES TAKEN INTO ACCOUNT IN FRANCE FOR THE USE OF SYSTEMS IN THE OPERATING DOCUMENTS FOR ACCIDENTAL SITUATIONS (INCLUDING SEVERE ACCIDENTS) ON PWR PLANTS

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1. SCOPE OF THE OPERATING DOCUMENTS

The safety philosophy developed in France for PWR's has lead to the division of operating conditions into three main fields as defined below :

Design basis operating conditions :

These concern all normal unit operating states and transients, incidental or accidental which it is agreed to take into account during design and dimensioning. These operating conditions have been divided into four categories according to the size of their estimated frequency of occurrence. Within each category, a limited number of conventional operating conditions has been chosen, whose consequences are envelope conditions when compared with those of other operating conditions which could have been considered.

Complementary operating conditions :

In particular, the field covers events linked :

- either to total loss in the short term of the safety systems frequently called into play in design basis operating conditions of the 1st and 2nd categories,

- or to the total loss in the mid and long term of the redundant safety features coming into play in the operating conditions of the 3rd and 4th categories, particularly the loss of coolant accident.

The field completes that covered by the design basis operating conditions ; the union of the two being called the operating conditions considered as being plausible.

Operating conditions not considered as being plausible

The studies of operating conditions considered as being plausible taken together make it possible to guarantee a satisfactory and homogenous level of safety. Ta'·ng into account conditions not considered as plausible aims, in keeping with the concept of defence in depth, at protecting populations in the case of severe and hypothetical accidents i.e : with inevitable core meltdown.

From an operating point of view, all the operating occurrences defined above are managed by operating documents which are :

- the operating procedures : these exist for the whole of the operating field defined above,

- a special document, called the "severe accident action guide", which aims to define, in advance, the specific actions to be taken by the operators in a post-accidental situation with core meltdown ;

Following this, this document presents, for each of the operating documents called upon in these different situations :

- the main principles chosen for using the systems particularly for severe accidents as defined above,

- the type of system it has called upon.

2. USING THE SYSTEMS

2.1. GENERAL APPROACH

The choice of systems to be used for an accidental situation, including severe accidents results from the following general approach :

- inventory of means available to carry out the desired function,
- classifying these different means, starting with the operating conditions of the systems closest to the nominal one. To illustrate this, if we consider the function "refilling the auxiliary feedwater tank", the following hierachy of events can be drawn up :
  . normal refilling with demineralized and dearated water,
  . special refilling using demineralized water,
  . ultimate refilling with raw water.
- define the criteria for use, and thereby determine the limits of these various resources ; several variants (called substitutions) are proposed to the operator within the means available to achieve the objective of the procedure under consideration. The means under consideration might be non safety-classified in accordance with the information given in paragraph 3.

This final stage is extremely important, since it makes it possible, among other things, to ensure that the accident will not be made more serious due to operation beyond the limits of mechanical resistance of the equipment.

2.2. SEVERE ACCIDENT ACTION GUIDE

The aim of the "severe accident action guide" is to provide the operator in advance with a definition of specific actions to be undertaken to ensure the best possible containment of radioactive products for as long a time as possible so as to :

- avoid or keep to a minimum the release of such products outside the reactor containment, either atmospherically or hydrogeologically,
- gain time for setting up the emergency plans, particularly for the population,
- bring the unit back to a more easily controlable situation.

Applying the action guide involves a change in strategy with respect to the management of the situation : we move away from the objective of core safeguard to one of containment function safeguard.

At the present time, the actions in the guide involve :

- refilling the tank for the safety injection system and the containment spray system,
- operation of the safety injection system (including the accumulator tanks),
- operation of the containment spray system,
- steam generator (available and unavailable) operation,
- the primary pumps,

- the pressurizer relief lines,

- specific procedures such as decompression of the containment using the sand bed filter.

In practical terms, this guide consists of a grid of "systems/actions" defining the operations to be undertaken on each system, making it possible to minimize the consequences of the accident.

The actions involved result from choices based on release criteria (containment safeguard strategy). They are based on knowledge available at this time, and are therefore likely to change.

3. CHECKING THE APPROPRIATENESS OF THE SYSTEMS USED

3.1. DESIGN BASIS OPERATING CONDITIONS

For this type of operating conditions, the operating documents must allow for the use of the necessary systems to bring back the power units to a safe state. These are all safety-classified systems, which satisfy in addition the following complementary requirements.

- single failure criterion,

- geographical component separation,

- environment conditions resistance,

- seismic resistance.

In addition, it should be noted that the analysis of the operating conditions of the systems used was made using conservative hypotheses, both for the accident scenarios and for the evolution of the resistance of the barriers and the installation in general.

3.2. COMPLEMENTARY OPERATING CONDITIONS

For this type of operating conditions, non safety-classified systems may be called upon.

Analysis of these situations takes into account "realistic" hypotheses, i.e : which include lower conservation margins than those used for design basis operating conditions.

In particular :

- the criterion of single failure is not applied, i.e : a further deficiency is not added to the initial failure but the consequences of this initial default and of the operating procedure are taken into account.

- unless this is the sequence under study, the loss of offsite power is not taken into account,

- the control devices operate normally,

- the instant at which the systems used are supposed to start up is determined realistically.

3.3. OPERATING CONDITIONS NOT CONSIDERED AS PLAUSIBLE

The systems used specifically in this type of operating condition are not classified as safety-related.

The analysis of the situations, particularly as far as the effects on the environment of a deteriorated core are concerned, are made with "realistic hypotheses". This is notably the case concerning the evaluation of the transfer of radioactive substances through the containment, depending on the various failure modes and their diffusion into the environment.

4. CONCLUSION

For PWR's in France, operating documents make it possible to cover all the operating conditions, including severe accidents, resulting from the developed safety philosophy (see appendix).

APPENDIX

UNIT SITUATION				Examples of operating documents
Operating conditions considered as plausible	Design basis operating conditions	Category 1 Category 2 Category 3	Normal operating Moderately frequent incidents Very infrequent accidents	"S" Procedures "I" Procedures
		Category 4	Serious hypothetical design basis accidents such as : - LOCA - Feedwater pipe rupture - Main Steam pipe rupture - SG tube rupture	"A" Procedures
	Complementary operating conditions	Short term loss of frequently required redundant safety systems : - total heat sink loss - total loss of SG feedwater - total loss of electrical power ("blackout") Mid and long term loss of safeguard systems coming into play following LOCA		"H" Procedures
Operating conditions not considered as plausible	Severe accident with inevitable core meltdown			"U" Procedures "Severe accident guide action"

Gesellschaft für Reaktorsicherheit (GRS) mbH

Accident Management (AM) Strategies
for LWR
in the Federal Republic of Germany

IAEA - Technical Committee
Meeting on
Plant System Utilization
for Accident Mitigation
Garching, FRG
26. - 30. Nov. 1990

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Content Seite

1. Concept of Accident Management in FRG

- Introduction

Nuclear power plants are provided with comprehensive and reliable safety systems which prevent damage to the reactor core and ensure the safe containment of radioactive substances in the case of incidents.

Beyond this area of incidents, which must be thoroughly guarded against when designing the plants, there remains an area of event sequences which does not have to be taken into account in the design of the plant,e.g. because their occurrence is accepted as being extremely unlikely. Such sequences of events always occur if the safety functions needed to cope with the event fail for a long period of time. Thus a core meltdown accident is only possible if it is assumed, e.g. that during an incident the core cooling fails completely for a long period of time.

Among the possible causes of such sequences of events are:

- unconsidered multiple failures of systems and components,
- delayed or no detection of a deviation from the design sequences of events,
- human error, such as an incorrect diagnosis or wrong intervention before or during the event,
- combinations of the above.

- Extension of the defence-in-depth concept

In all fields of human activities the protection against possible dangers is based on two major concepts. The first deals with preventing damage by taking appropriate precautionary measures, the second tries to limit the extent of possible damages. The greater the extent of possible damages becomes, the more important become precautionary measures which, by eliminating endangering mechanisms already in the forefield, provide for a greater distance to the manifestation of the danger.

Already in the early days of civilian use of nuclear energy it was realized that preventive measures in this respect have a great significance. This led to the development of an extensive safety concept based on the principle of graduated prevention mentioned above. The danger potential from radioactive materials is contained within many consecutive physical barriers. These barriers themselves are protected by a graduated system of precautionary safety measures, the defence-in-depth concept (see Figure 1). In line with the principle of applying protective measures as far ahead as possible in the preventive area, the underlying principle is to prevent defects and disturbances from the beginning before larger damages can develop.

- Level 1

At the first level, high quality standards and high safety margins are applied to all plant components with the goal of preventing failures from occurring at all.

- Level 2

At the second level, disturbances are intercepted before they can develop into incidents. Here, the limitation system and the reactor protection system which monitors all essential process variables of the plant are of central importance. Whenever specific limits are exeeded protective actions such as power reduction or shutdown of the reactor are initiated.

- Level 3

Third in line is the equipment of the safety systems (engineered safety features) which protects the radioactivity barriers. Here the design goal is that - in case of an incident - at least two radioactivity barriers should remain intact, thus preventing a dangerous release of radioactivity to the environment.

- Level 4

The safety design (levels 1 to 3) establishes an extensive set of protective measures. However, by law of nature even the most reliable safety system cannot prevent that a system failure or a combination of failures occurs which is not covered by the design. It is for instance possible that during a major incident the safety equipment itself fails. The analyses of beyond-design accidents, in the course of which a failure or the safety equipment is assumed, show that in most cases several hours remain before serious damage occurs to the reactor core. This leaves time for the prevention of damage using remaining safety systems as well as operational or even external systems.

The resulting concepts and measures for coping with or limiting the consequences of events, which were not explicitly taken into account when designing the plant, are investigated and discussed internationally using the term "Accident Management" (AM).

In the area of accident management a distinction is made between 'prevention' and 'mitigation'.

(1) Prevention includes measures to avoid damage to the reactor core. Owing to the relatively slow development from an initiating event to major core degradation (usually hours) there is in principle the possibility for the plant personnel to identify and diagnose the status of the plant and to initiate safety related measures, e.g. reactivating safety or operational or additional systems. These measures are considered to have the highest priority in the Federal Republic of Germany. (TMI may be an example of the borderline between the prevention and mitigation areas.)

(2) Mitigation includes measures to control and minimize the consequences of core melt sequences. If measures to reactivate sufficient core cooling and decay heat removal fail, core melt will progressively start. Even in this case measures to control and minimize the consequences should be initiated, The final goal is to maintain the integrity of the primary system or at least of the containment to avoid an uncontrolled and large release of fission products into the environment.

- Principles for implementation of AM-Measures

Accident management procedures extend the symptom-oriented accident control concept beyond the area of design-basis-accidents (see fig.2). A safety-function oriented approach is followed to cover as many conceivable failure combinations as possible with a reasonably small number of procedures. As in the design range, preventive measures are given priority. Their objective is to activate a heat removal from the reactor pressure vessel in order to prevent core meltdown or at least to arrest the process in time.

The accident management procedures will be contained in a dedicated accident management or emergency manual, separate from the operating manual.

The following principles are applied in the implementation of Accident Management Measures:

- The accident management measures may not impair plant operation under normal or upset conditions nor may they unacceptably interfere with design-basis accident control.

- Analyses are performed with realistic boundary conditions.

- Accident management measures take credit of all existing systems and equipments.

- The usual design criteria for safety systems, such as the single failure criterion, are not applied.

- Accident management measures may only be initiated after a sufficiently long time delay essential to proper decision making.

- It must be possible to interrupt and resume the accident management measures at any time.

- Any necessary equipment for initiating accident management measures must be arranged in such a way that operator errors or inadvertent initiation during normal operation are avoided.

- On account of the time available and the low frequency of the postulated events accident management measures will be initiated manually.

2. Preventive Accident Management Strategies for PWR

In case of PWRs, the concept of energy removal in all transient events and small leaks in the primary loop is based on heat removal via the secondary side. The secondary-side system functions for coping with such events are therefore of great importance from the point of view of safety. In case there is a failure of the secondary-side systems, alternative possibilities for residual heat removal should be available.

Many secondary and primary-side accident management measures are being discussed for this purpose. However, secondary-side measures are considered most suitable for all cases of failure of the secondary-side heat removal. Only if the necessary manual measures fail or if the secondary-side measures are not effective should there be a transfer to primary-side emergency measures (e.g., primary-side bleed and feed). This strategy makes sense because West German PWR plants exhibit only very minor primary-side leakages in the case of such events (e.g., station blackout) due to the special design of the sealing of the primary-coolant pump.

Examples have been selected from the many measures discussed and listed in the following Tables 1 and 2.

The results of German Risk Study, Phase B indicate that about 98 % of the frequency of uncontrolled events (plant damage states) would lead to core melt under high pressure. Therefore special emphasis was laid on measures to depressurize the primary system.

- Secondary Bleed and Feed

The secondary bleed and feed-measure in principle is characterized by a fast depressurization (bleed) of the dry steam generators below the respective pressure of the feedwater tank and a subsequent water injection (feed) at low pressure.

In Fig. 3 the feedwater- and main steam system of the reference plant for the German risk study is shown. The bleed action will be initiated by opening of main steam relief or safety valves. The depressurization is only effective if certain simulations in the reactor protection system (defeating interlocks, overriding protective trips) will be performed.

For the feed action there are two possible ways. At first the stored water in the feedwater tank can be used. It's volume of 610 m^3 is filled with 320 Mg saturated water. The pressure is 1.0 MPa in the reference plant. In some plants the pressure is lower. For better feed performance it can be increased by loading of the feedwater tank through an auxiliary steam line up to the setpoint of the tank's safety valves.

After opening of control- and shut off-valves the feedwater tank acts as an accumulator due to the pressure stabilizing effect of flashing water. Water can flow through the feedwater lines via the feedwater pumps and the HP-preheaters into the steam generators, according to the pressure difference between feedwater tank and steam generator.

The second possible way to inject water is the connection of low head pumps (e.g. mobile pumps) to the emergency feed lines. The necessary water can be taken from the demineralized water storage tank, cooling tower ponds, drinking water supplies, tank trucks or directly from the river.

For the above mentioned measures time (0.5 to 1 h) is needed to prepare and perform the necessary manual actions.

The preparation of the AM-actions will start if the water level on the secondary side reaches a very low value (e.g. 2 m).

- Primary Bleed and Feed

Primary bleed and feed measures are operator actions which consist of opening of pressurizer valves (bleed) to release the residual heat into the containment and feeding of the primary side with safety injection pumps. These pumps start the injection of borated water below 11 MPa.

To reach a long term stable state after depletion of the borated water storage tanks further actions are necessary. In new plants, operation of the low pressure injection system (LPIS) as booster pumps make a high pressure injection of sump water possible. In plants without this backup capability, a depressurization of the primary side below 1.0 MPa is necessary. This can be achieved by an enhancement of thermal mixing to remove hot spots in steam generators (operation of main coolant pumps necessary) or by an opening of additional pressurizer valves. At low pressure the LPIS can remove the decay heat in the recirculation mode and simultaneously inject coolant from the sump to make up for the loss of coolant. If the pressurizer valves can be closed after refilling of the primary side then the decay heat can be removed by the LPIS in the recirculation mode already from 3 MPa provided the coolant temperature is below 180 °C.

Assuming the safety injection pumps to fail, opening of all pressurizer valves will achieve fast depressurization below 2.6 MPa and the inventory of the accumulators may be used for feeding.

According to the strategy of early secondary and late primary bleed and feed measures an appropriate instrumentation for the initiation of the primary measures is needed. The time point of the derived initiation signal should be late enough to enable the performance of secondary measures but early enough to prevent core melting taking into account a sufficient response time for the manual initiation. In all PWR's in the FRG a level probe in the upper plenum of the reactor pressure vessel is or will be installed which measures the collapsed level (fig.5). The decrease of this level to the lower edge of the coolant loops is signaled to the operators. This criterion will be used to initiate primary bleed. Should this signal fail, primary bleed should be performed if the core outlet temperature is greater than 400°C.

- Assessment of Bleed and Feed Strategy

Beside the previously discussed transients "Total loss of feedwater" and "Station Blackout" there are some other sequences leading to a similar core damage frequency. Fig. 6 shows the frequency distribution which has been determined in the Phase B of the German risk study for the reference plant when no accident management-measures are considered. Uncontrolled pressurizer leaks, station blackout, loss of feedwater, loss of main heat sink, steamline break and U-tube rupture are sequences for further investigation of accident management measures, regarding probabilistic aspects. All these sequences and uncontrolled small breaks at the coolant loops having a break area below approximately 50 cm^2 will lead to core melting at high primary pressure.

In general the same strategy as previously discussed can be applied. If additional failures at the secondary side cause a decrease of the feedwater inventory, secondary bleed and feed is initiated to reestablish a sufficient secondary heat sink. In case of breaks and leaks at the primary side a primary injection for compensation of the loss of primary coolant is necessary to prevent core melting. Otherwise the secondary feed and bleed alone is sufficient. If the secondary measures fail, i.e. the collapsed level in the reactor pressure vessel continues to decrease, a switching to primary bleed and feed is necessary to enable safety injection pumps to inject borated water.

In fig. 6 estimated delay times for the initiation of the accident management measures for the various sequences are shown. In case of the more likely events the delay times are greater than 1 h and hence comparable to the previously discussed delay times. In the case of the uncontrolled rupture of one U-tube the delay time is about 1 h. In the case of two ruptured U-tubes the delay time is only about 30 min. But this event is about one order of magnitude lower in its frequency.

The analyses of bleed and feed measures demonstrate the effectiveness of this measure to prevent core melt respectively to prevent core melt under high pressure. A simplified event tree (see fig. 7) depicts the possible sequences.

- Prerequisites for the Implementation of Bleed and Feed

To enable the bleed and feed measures to be taken on the secondary and primary sides, changes are carried out at the plant depending on their design in particular:

- changes in the reactor protection system,
- changes which permit a depressurization and automatic feeding of the steam generators from the feedwater tank,
- the installation of additional connections for mobile pumps on the pressure side of the emergency feedwater pumps,
- the installation of a water level indicator (see Fig. 5) in the upper plenum of the reactor pressure vessel,
- the design of the pressurizer valves and the associated control valves to enable the relief of water/steam mixtures (two-phase mixtures). For this purposes, the pressurizer relief valves and the safety valves are provided with an additional control line which can be opened by motorized pilot valves. The motorized valves in turn will be provided with a power supply secured by batteries.

3. Preventive Accident Management Strategies for BWR

The numerous possibilities for injection into the reactor pressure vessel or the containment vessel of a BWR plant are shown in in Fig.8. One can see that even in the case of the failure of all safety systems, use can still be made of the available operating or fire-fighting systems. It would thus be possible, up to very high pressures, to feed into the reactor pressure vessel with the control rod cooling water system and the pump seal water system. In the low-pressure range, more injection systems are available. Below 10 bar, the feedwater tank could act effectively as an accumulator. In addition, there is the possiblity of injecting with the 2 fuel pool cooling pumps into the reactor pressure vessel or containment. As an emergency measure, it would be possible to inject with the 2 demineralized water fire-fighting pumps, with the 3 fire-fighting pumps or mobile pumps, and also with the drinking water network (capacity of the pumps approx. 60 kg/s). The use of fire-fighting system means, that the necessary hose connections must have been installed. All transient events as well as most loss-of-coolant incidents can be coped with by means of these emergency measures. More detailed analyses are still being carried out. In the case of many events, residual heat is removed via the suppression chamber. Should the suppression chamber cooling system fail for a long period of time, there is the possibility of removing residual heat by releasing steam via the depressurization system provided for the containment vessel. This measure will simultaneously prevent a failure of the containment vessel due to excess pressure.

The preventive Measures discussed for BWR are listed in the following table 3.

4. Mitigative Accident Management Strategies

In case of a total failure of preventive accident management, mitigating emergency measures would have to be taken. These would focus on the restoration of the coolability of the damaged core and on the reinforcement of the containment funtion. In the following some examples are given.

- Avoidance of High Pressure Core Melt

The primary bleed will be initiated if the water level in the reactor pressure vessel falls below the hot leg inlet nozzle. Should the high pressure injection fail, core melt would occur at low pressure and the containment integrity could be assured.

- Control of hydrogen in the Containment

In the Federal Republic of Germany, BWRs and PWRs are in operation, for which the hydrogen problem and related accident management procedures are different.

Inerting ist implemented already in a number of plants. The

BWRs of the 69 type have a small containment volume and inerting is required. Continuous inerting during operation is provided. In the BWRs of the 72 type, inerting of only the wet well is being studied (KRB). Also the use of ignitors and/or catalytic foils in the drywell and wetwell is considered.

In PWRs, a great amount of hydrogen can be formed by the steam-metal reactions during the in-vessel core melt phase an by dry melt-concrete interaction later on. In order to keep the hydrogen concentration low, early ignition and combustion of hydrogen will not jeopardize the integrity of the containment.

In some scenarios a longtime hydrogen accummulation due to partly steam inerted conditions could lead to containment failure. In case of 100 % of zirconium oxidation, nearly 1400 kg of H2 would be formed. A late deflagration of such a high quantity could lead to containment failure. In addition hydrogen inhomogeneous distribution can result in higher hydrogen concentrations locally, susceptible for detonations. Controlled early ignition by autonomous ignitors will be introduced. The recombination by catalytic devices, e.g. foils, has been investigated, and the evaluation is going on. Different designs of ignitors (spark and catalytic ignitors) and catalytic devices are studied. Catalytic devices e.g. foils have the advantage of being passive systems.

Reliable hydrogen monitoring under severe accident conditions is another problem which has to be solved in PWRs.

- Procedures to cool molten corium in the Containment

There is no unanimous strategy for all plants wether and how to cool molten corium in the containment. There are several possibilities to inject water into the containment.

A spray system is in general not available in German PWRs, in BWRs such system exist. In the BWRs, also a water pool in the lower drywell can be provided. The water may come e.g. from the pressure supression pool or from the fuel storage pool.

In the PWRs the cavity below the reactor pressure vessel is dry and will remain so for several hours in most plants. The basement is approximately 6 m thick, penetration of which would take several days.

The best strategy in BWRs is to fragmentate and cool the corium. Providing sufficient fragmentation, cooling could be achieved. Fragmentation depends highly on the reactor pressure vessel failure mode. Coolability of the corium and steam explosion is considered as an area of further research.

- Control of pressure in the Containment

In the Federal Republic of Germany, a venting system is obligatory since short after Chernobyl by RSK decision (December 1986 for PWRs, June 1987 for BWRs). In all BWRs and in some PWRs venturi-scrubbers are used with integrated steel fibre mesh filters. In other PWRs steel fiber mesh filters and iodine filters with molecular screen will be used. The employed filters meet the requirements of efficiency postulated by RSK (Reactor Safety Commission).

The basic design of the venting system using a scrubber is shown in Figs. 9 and 10.

The filtered venting system consists of a wet srubber with venturi nozzles followed by a combined droplet seperator and stainless-steel fiber deep-bed filter housed in a pressure vessel.

To activate the system, the isolation valves are opened by remote control from the control room. The venturi scrubber is operated at pressures close to the prevailing containment pressure levels due to the provision of a throttling orifice in the filter discharge line. The venting flow entering the scrubber is injected into a pool of water via a large number of submerged, short venturi nozzles.

As the vent gas passes through the throat of the venturi nozzle, the incoming gas flow develops a suction which causes scrubbing water to be entrained with it and on account of the large difference between the velocity of the scrubbing water particles and that of the incoming vent flow, a large proportion of the aerosols are removed.

At the same time, the particles of the entrained scrubbing water provide large mass transfer surfaces inside the throat of the nozzle, which permit effective sorption of iodine. Optimum retention of iodine in the pool of water inside the scrubber is attained by conditioning the water with caustic soda and other additives. In view of the mechanisms occuring inside the venturi, most of the iodine and aerosole particles will in fact be separated inside the throats of these nozzles.

The pool of water surrounding the nozzles acts as the primary droplet separation section and also serves as a secondary stage for retention of aerosols iodine.

The gas exiting from the pool of water still contains small amounts of hard-to-retain aerosols as well as scrubbing water droplets. In order to ensure high retention efficiencies even over prolonged periods of time - for example 24 hours - a high efficiency droplet separator and micro-aerosol filter is provided as a second retention stage.

The first stage of this unit serves to remove most of the water droplets. The second stage, a metal-fiber fine filter, is designed to retain the aerosols of particle sizes so small that they are difficult to retain under conditions at and especially above atmospheric pressure. A throttling orifice installed downstream of the scrubber unit provides for critical expansion of the cleaned gas, which is subsequently released to the environment through a separate stack. The entire venturi scrubber unit provides a retention efficiency for aerosols of 99.99 % or more. This retention capability also applies to mirco-aerosols of less than 0.5 μm so that, for example, variations in the particle size distribution of the aerosols cannot diminish the removal efficiency. The retention efficiency for elemental iodine under all operating conditions including overpressure conditions is above 99 %.

By selection of this process, superheating of the vent gas flow upstream of the scrubber when steam has been entrained from the containment is largely avoided so that the pool is operated under saturated water conditions and only an insignificant amount of water is lost from the pool, obviating the need for scrubbing water make-up.

Furthermore, the venturi scrubber pool is designed such that evaporation of the pool water caused by the decay heat generated by the aerosols in the water will likewise not lead to an unacceptable drop in the pool water level.

5. Accident Management Procedures

AM-procedures will be described in an emergency operating manual together with other informations potentially useful in emergency situations. This manual, conceived as a "living" manual independent from the existing operating manual is actually being introduced. The transition from the operating manual, to the emergency operating manual is described in the following:

If abnormal occurences or additional disturbances arise which cannot be allocated to any of the defined accident sequences, the operating personnel intervenes in the accident sequence following the procedures outlined in the chapter which addresses accident management based on <u>plant status and safety objectives</u>. This chapter is referred to at the very latest as soon as it becomes apparent that safety objectives are endangered. Each safety objective is assigned a set of process-related parameters with predefined limits with which fulfillment of the safety objective is ascertained.

The prescribed operating procedure is based on specific plant conditions and on the parameters typical of the particular safety objective concerned, regardless of wether certain events take place or not.

If it is not possible to fulfill the safety objectives by means described in the operating manual (safety-objective-oriented procedures), the procedures for accident management described in the emergency operating manual must be followed (see fig. 11). These show options available for making use of plant safety margins in order to prevent sequences leading to an uncontrollable plant condition. Usually special releases are required from a plant emergency operations' team in order to utilize these remaining safety margins since intervention into the intended mode of operation of the plant safety systems is necessary. If the emergency operations team is not yet available to make such decisions at the time when these decisions have to be taken, then the required release will begranted by the shift supervisor.

In order that, in the case of beyond-design events the accident management measures can be correctly selected, swiftly implemented and become optimally effective a series of preparatory investigations have been carried out taking realistic accident conditions into account. In particular, when the emergency operating manual was being prepared, the effectiveness of the system that would be used for such accident management measures had to be carefully analyzed under consideration of realistic conditions. The system capabilities calculated during the original plant design phase on the basis of extremely conservative assumptions were of no use for such purposes.

The emergency operating manual is organized according to the individual safety objectives mentioned earlier.

If more than one type of action would be possible for one of these safety objectives, a decision logic is available for selecting the optimum measure. This logic also contains the most important data regarding the effectiveness of the measures and the number of personnel required for their implementation. In the chapters describing the individual emergency response measures, first the objective of the measure and then the prerequisites for its effectiveness are described in detail. Following this the number of personnel required for implementation, broken down according to individual groups of activities, and the effectiveness of the measures (e.g. available coolant inventory, injection rates, etc.) are stated.

The operator actions themselves are first described in prose form and then complied in detailed checklists. For each group of activities, additional copies are provided for removal from the manual and taken to the place of action. These contain the checklists for the measures to be taken as well as vital supplementary information such as details from building drawings and photographs of plant compartments and components in or at which operator actions will be necessary.

6. Conclusion and Outlook

Nuclear power plants are designed in accordance with a defence-in-depth safety concept which each of its levels has the objective of preventing damage by appropriate precautionary measures and, at least, of mitigating the effects of damage that cannot be prevented.

The especially emphasised accident management measures (Level 4) further reduce the risk by enhancing the precautionary measures at levels 1 to 3. They are directed toward protecting against extremely improbable beyond-design accidents. Accident management measures encompass preventive as well as mitigating technical measures with major emphasis on preventive measures. They have to be oriented toward a few, but vital, safety objectives.

Accident management is an efficient tool for the further reduction of the probability of core damage and fission product releases and thus for the reduction of risk. The investigations carried out so far for bleed and feed measures show that a reduction by a factor of 100 should be feasible for specific event classes if the procedures are well prepared, specified and practiced in training.

Primary bleed and feed can not only prevent core melt, this action can also mitigate its consequences.If,following the reduction of pressure, the long-term cooling were to be unsuccessful, the core meltdown would at least be delayed, and a meltdown at high pressure with its potential to damage the containment would be avoided.

In order to prevent a long-term overpressure failure of the containment filtered venting of the containment would be executed. Controlled hydrogen combustion for instance by local battery-powered ignitors or by catalytic foils is being investigated as a mean to avoid the hazards of hydrogen burns in PWR's.The corresponding measure for the protection of BWR containments is the inerting of the containment.

Possible AM-procedures are described in an emergency operating manual together with other informations potentially useful in emergency situations. This manual, conceived as a "living" manual independent from the existing operating manual is actually being introduced.

The shift personnel should be supported by improved technical equipment. Advances in the man-machine interface may help confirm and accelerate decision-making and contribute to the prevention of extreme stress situations. It is essential in this context that the oversupply of information occurring in such extreme cases is reduced. In addition, the flow of information should be structured in such a way that the state of the plant can be reliably and quickly judged by the operating personnel.

Here, computer-assisted diagnostic tools and emergency manuals, devices for reduction of the amount of incoming information, expert systems and other tools of modern information processing may make a significant contribution.

In the FRG, those improvements and the further extensions of accident management will be an essential activity in the field of reactor safety. PSA will certainly provide important contributions to that development and support the selection and specification of further measures.

Table 1: Secondary-side Measures for PWR Plants

Aim: ensuring of core cooling in the case of transient events or small leaks with failure of the secondary-side emergency injection:

- Fast secondary-side depressurization via atmospheric dump valves or opening of main steam valves
- Steam generator injection from the feedwater tank
- Steam generator injection with
  - demineralized water recirculation pumps
  - fire-fighting system
  - mobile pumps

- Use of water supplies
  - demineralized water tank
  - cooling tower ponds
  - tank trucks
  - river

- Use of the connecting pipes between the emergency feed-water trains for the purpose of increasing flexibility during steam generator injection

- Heat removal via feedwater tank into the environment

- Heat removal via the auxiliary steam system

- Secondary-side depressurization via blowdown pipe or discharge via emergency feed turbine (relevant for older plants) for the purpose of preventing formation of stratified demineralized water in the primary circuit (relevant in the case of a steam generator tube leakage to prevent recriticality)

Table 2: Primary-side measures for PWR plants

- High-pressure sump operation with low-pressure pumps as booster pumps
- Use of the emergency residual heat removal pumps in the case of the failure of heat removal pumps
- Use of borated water supplies in the flooding tanks of unavailable high-pressure trains; transfer possible by use of the containment spray pump
- Coolant injection with boration pumps, mobile pumps
- Depressurization and heat removal via pressurizer discharge system and injection with safety injection pumps (bleed and feed)
- Additional use of primary-coolant pumps even when supply conditions are insufficient

Table 3: Preventive Measures for BWR Plants

- Safeguarding the power supply by use of hydro power stations

- Connections of the auxiliary drives of the steam-driven injection system to the battery-supplied alternating voltage: it is thus possible to operate the injection system in the case of the failure of the emergency power supply

- Prevention of the initiation of forced pressure suppression at a suppression chamber temperature above 55 °C. In the case of failure of residual heat removal from the suppression chamber, the level in the reactor could then still be maintained with the steam-turbine driven injection system

- Use of the contents of the feedwater tank (250 m^3; 10 bar) accumulator operation possible

- Actuation of the suction valve for the flooding system from the containment sump by the reactor protection system to further increase the redundancy of the recirculation systems

- Depressurization of the reactor pressure vessel via the auxiliary steam pipe to the condenser

- Depressurization of the containment for the purpose of residual heat removal from the pressure suppression system

- Use of the make-up (high-pressure) feed system at low pressure Maintaining the level in the reactor pressure vessel with the control rod cooling water system (control valves must be opened)

- Use of the seal water system for reactor pressure vessel injection (isolation to be removed)

- Use of the reactor pressure vessel suction pipe to the residual heat removal loop for the purpose of shutdown cooling in the case of a low level in the reactor pressure vessel

- Maintainance of the reactor pressure vessel level with the demineralized water fire-fighting system and fire-fighting pumps
- Reactor pressure vessel injection with demineralized water pumps

- Reactor pressure vessel injection with the fuel pool cooling system

- Refilling of fuel pool and demineralized water tanks via the drinking water supply or fire-fighting system

- Reactor pressure vessel injection with an independent emergency system

- Residual heat removal in the case of loss-of-coolant incidents using the containment sump recirculation system

- Residual heat removal in bypass to the suppression chamber via the drywell spray system using the containment sump recirculation system

- Special operation with controlled lowered level in the pressure vessel and utilization of the void effect (negative reactivity coefficient) for the purpose of reducing power in the case of anticipated transients without scram

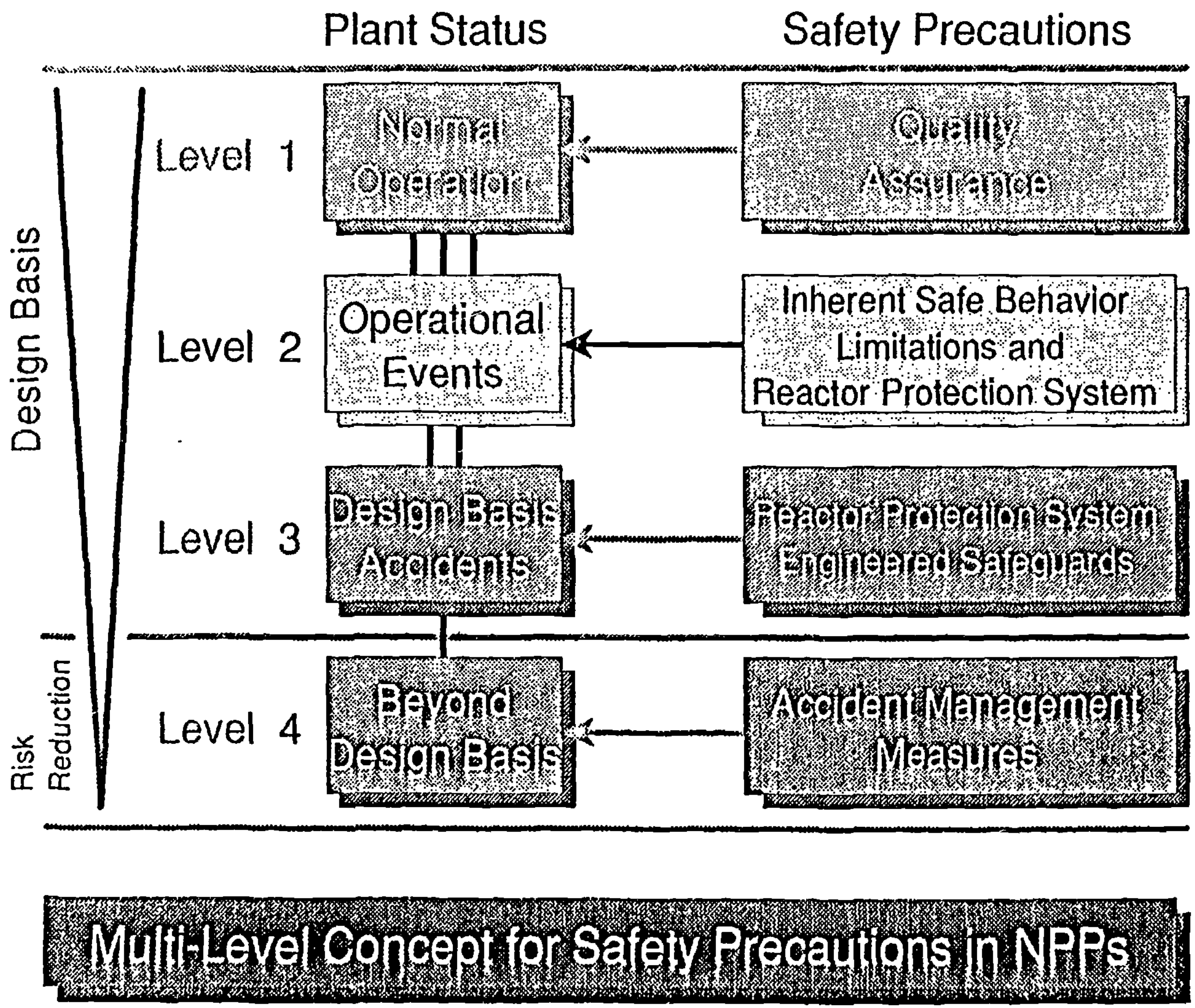

Plant Status
Safety Precautions
Design Basis
Risk Reduction
Level 1
Normal Operation
Quality Assurance
Level 2
Operational Events
Inherent Safe Behavior
Limitations and
Reactor Protection System
Level 3
Design Basis Accidents
Reactor Protection System
Engineered Safeguards
Level 4
Beyond Design Basis
Accident Management Measures
Multi-Level Concept for Safety Precautions in NPPs

Fig. 1

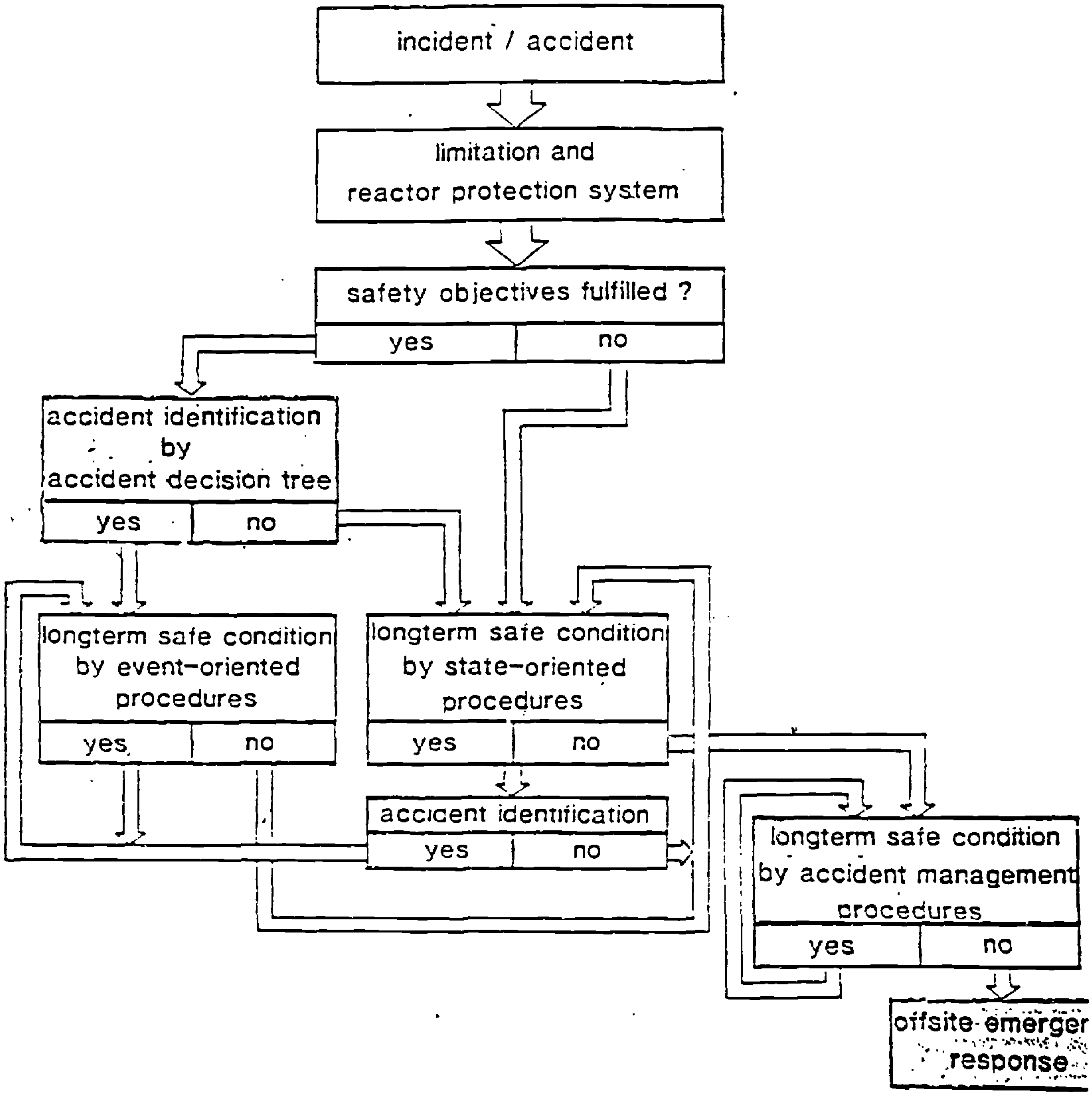
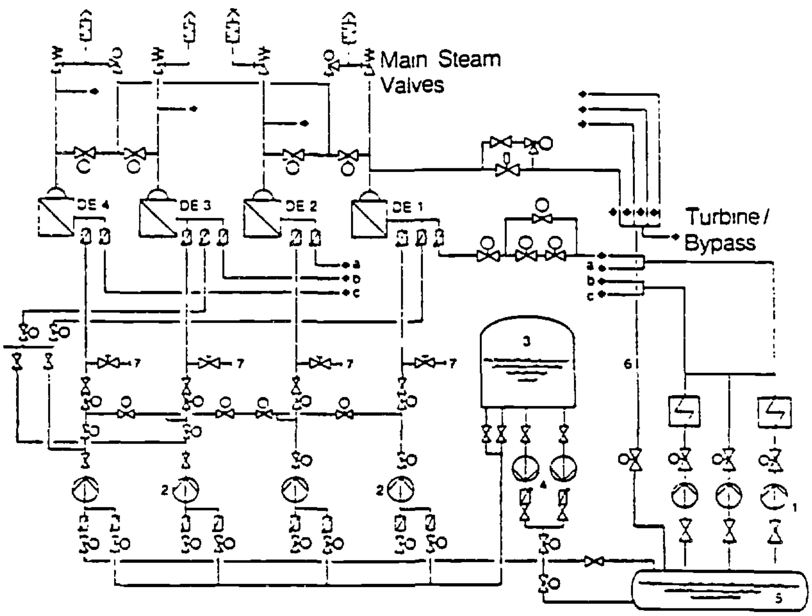


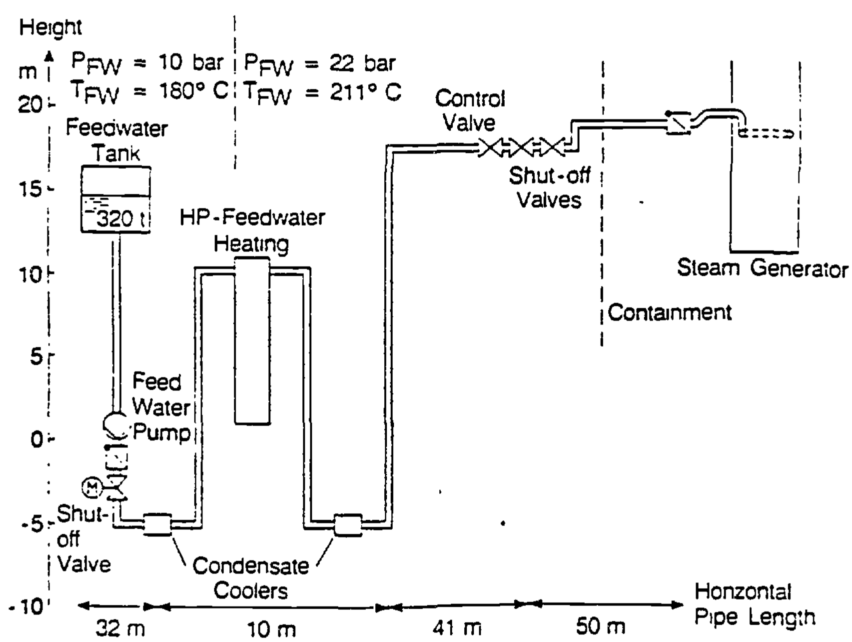
FIG. 2

EMERGENCY PROCEDURE GUIDELINE
Event-oriented and State-oriented Procedures

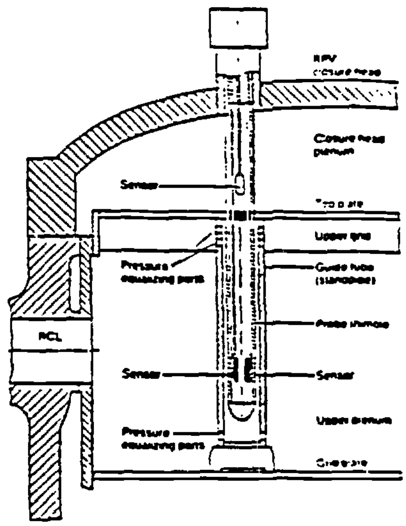


1 Feedwater Pumps
2 Emergency Feedwater Pumps
3 Demineralized Water Storage Tank
4 Feed Pumps
5 Feedwater Tank
6 Auxiliary Steam Line
7 Connection for Mobile Pumps

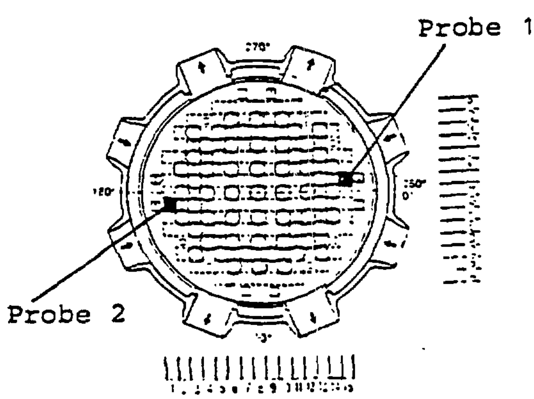
Feedwater- and Main Steam-System / PWR   Fig. 3



Feedwater-System   Fig. 4

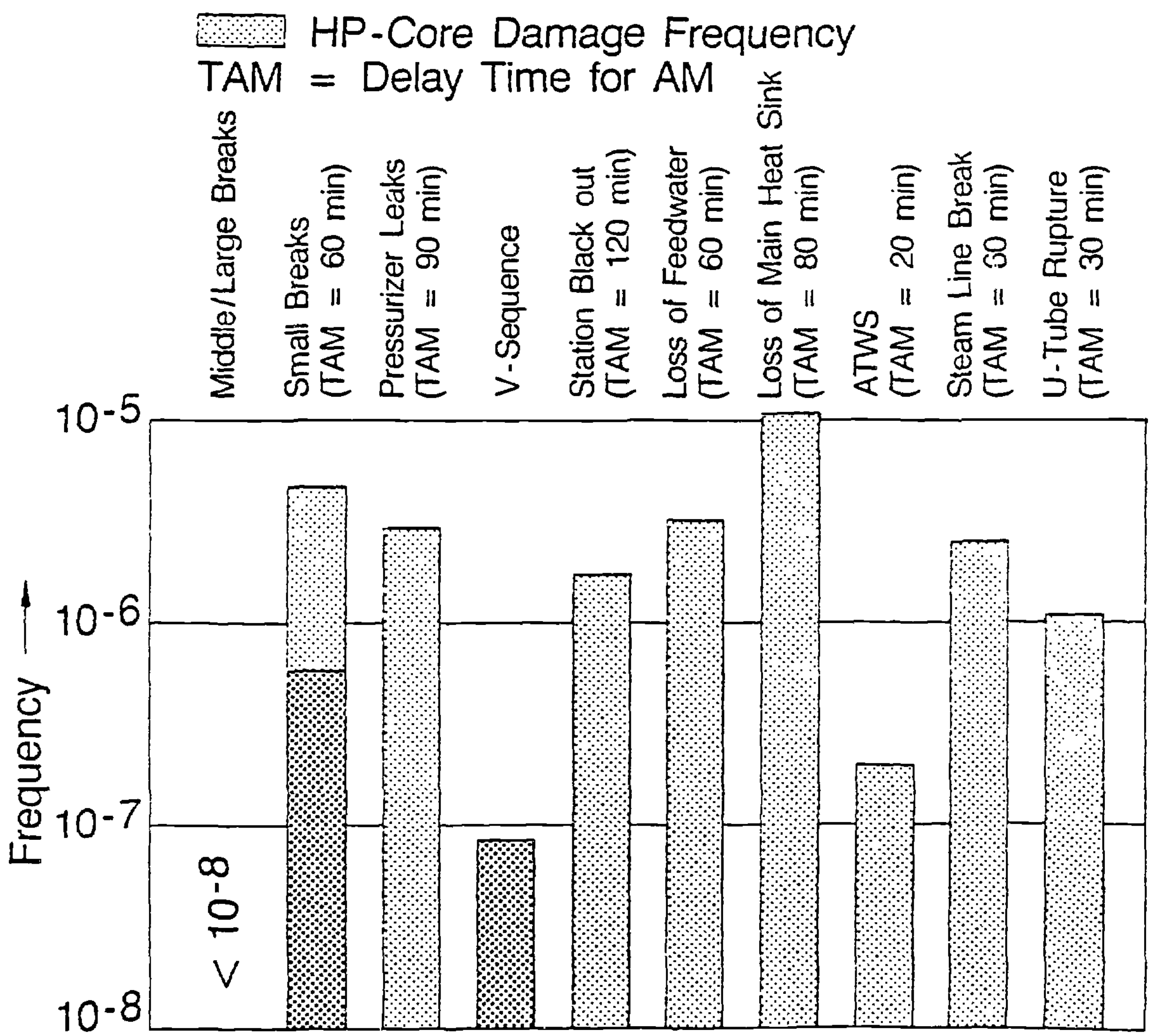


Schematic of guide tube, probe thimble
and sensor arrangements



Location of level probes in upper plenum

Fig. 5



3235K

Core Damage Frequency
without AM-Measures (DRS/B)

Fig. 6

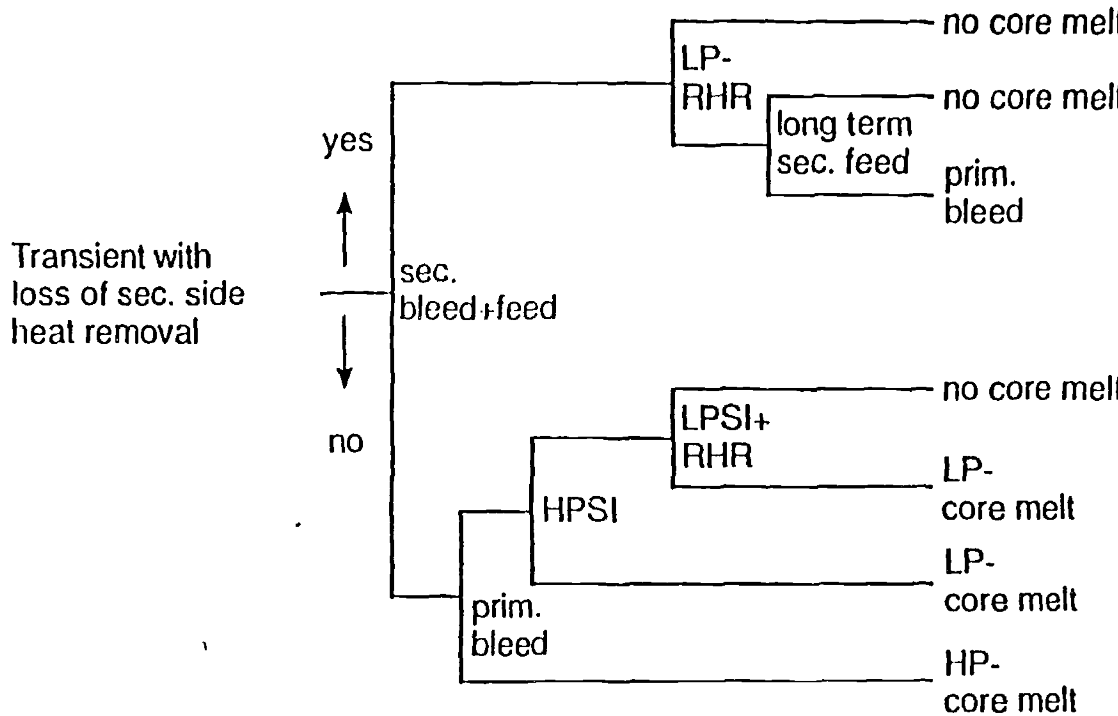


Fig. 7

Event tree for transients with loss of sec. side heat removal (plant damage states without loss of coolant) including AM

3392K

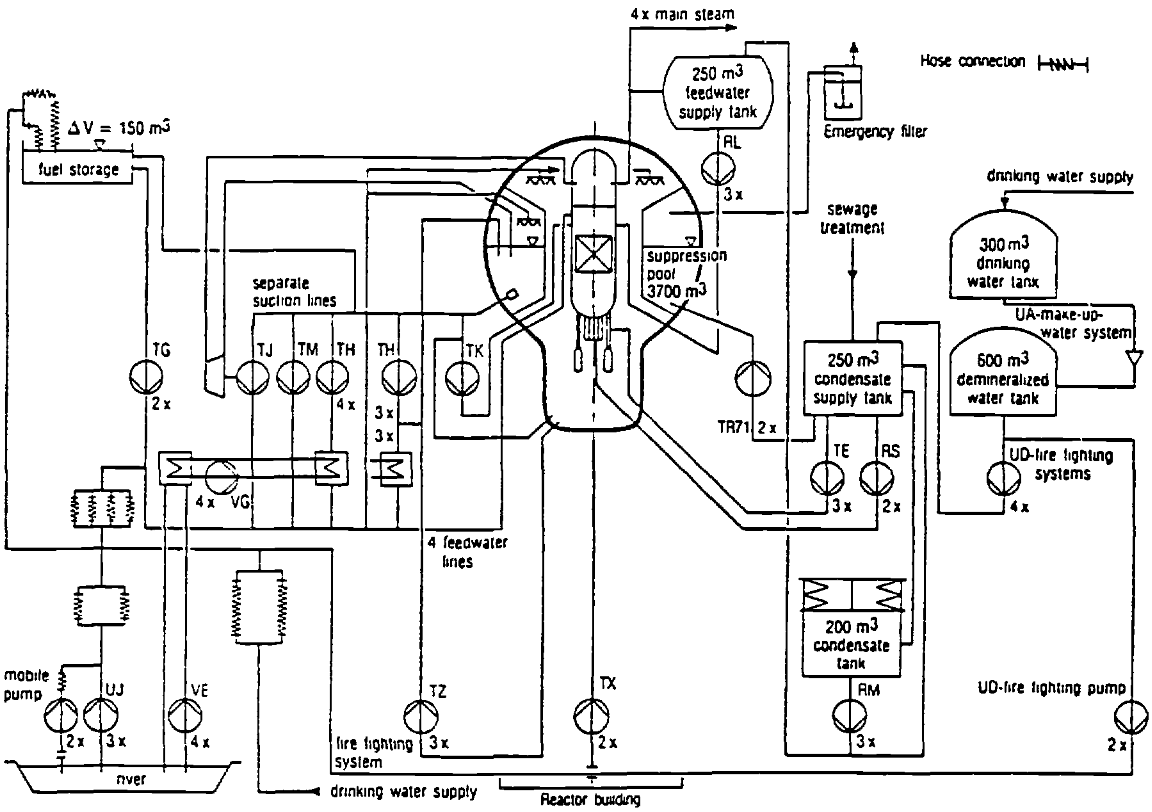


Fig. 8

Injection Possibilities (Krümmel)

2939K

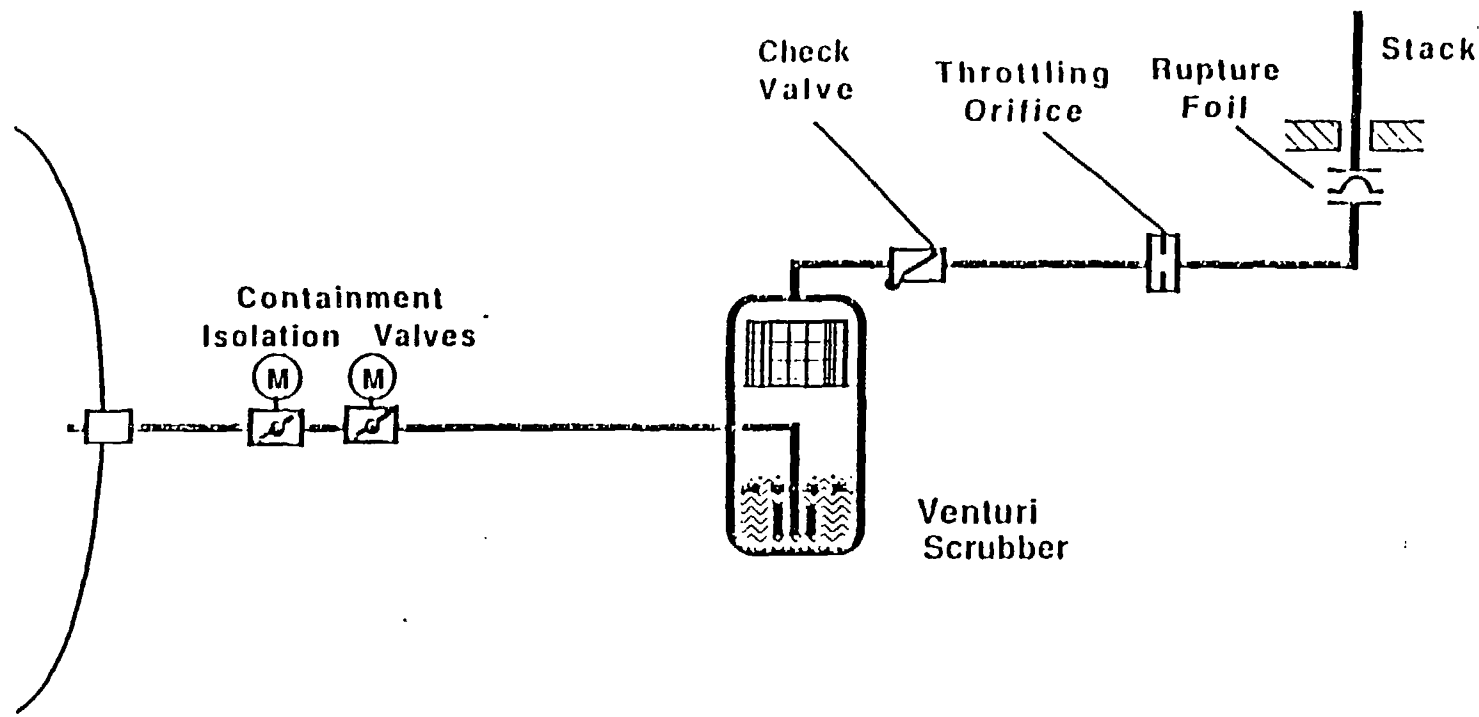
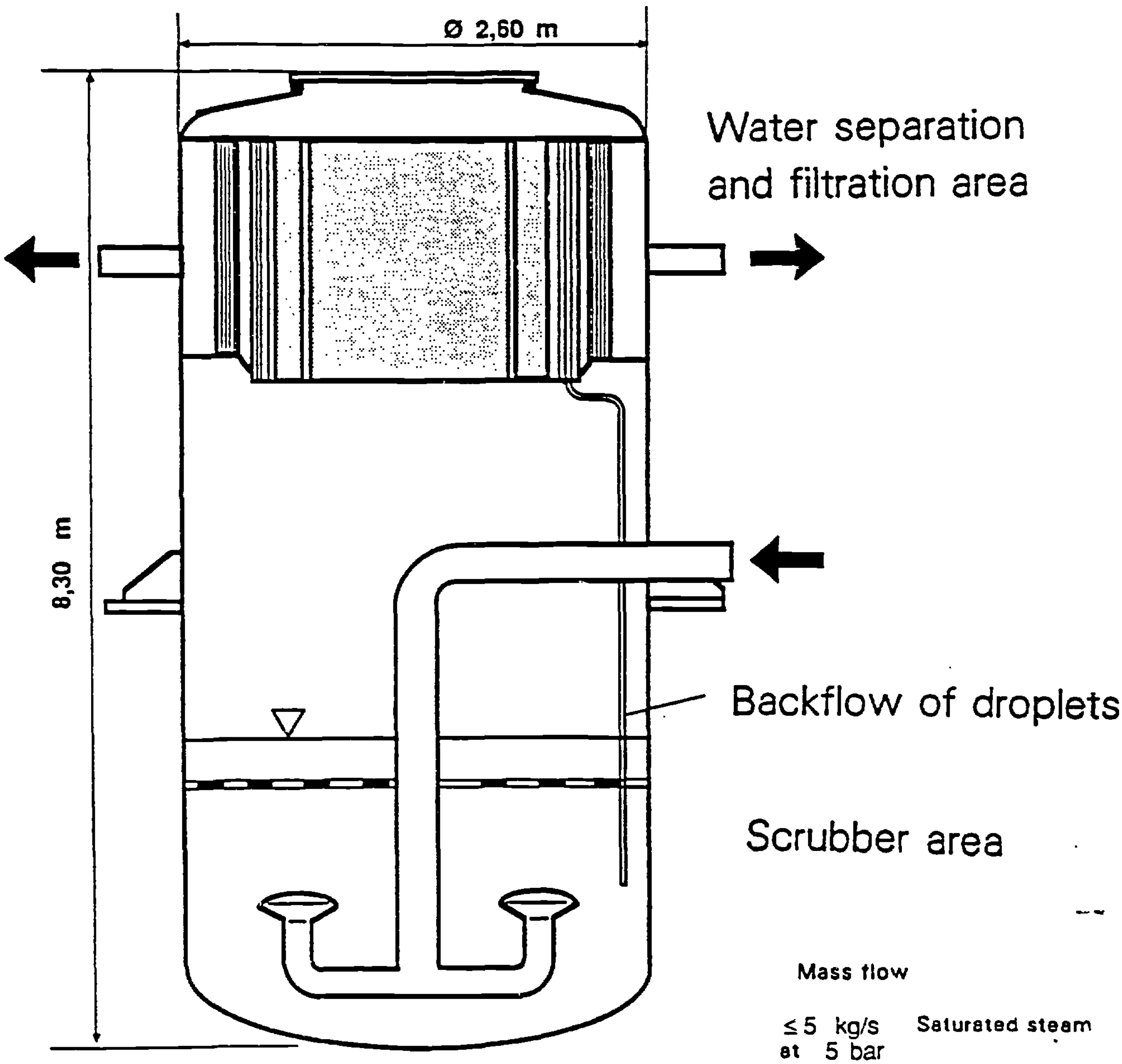


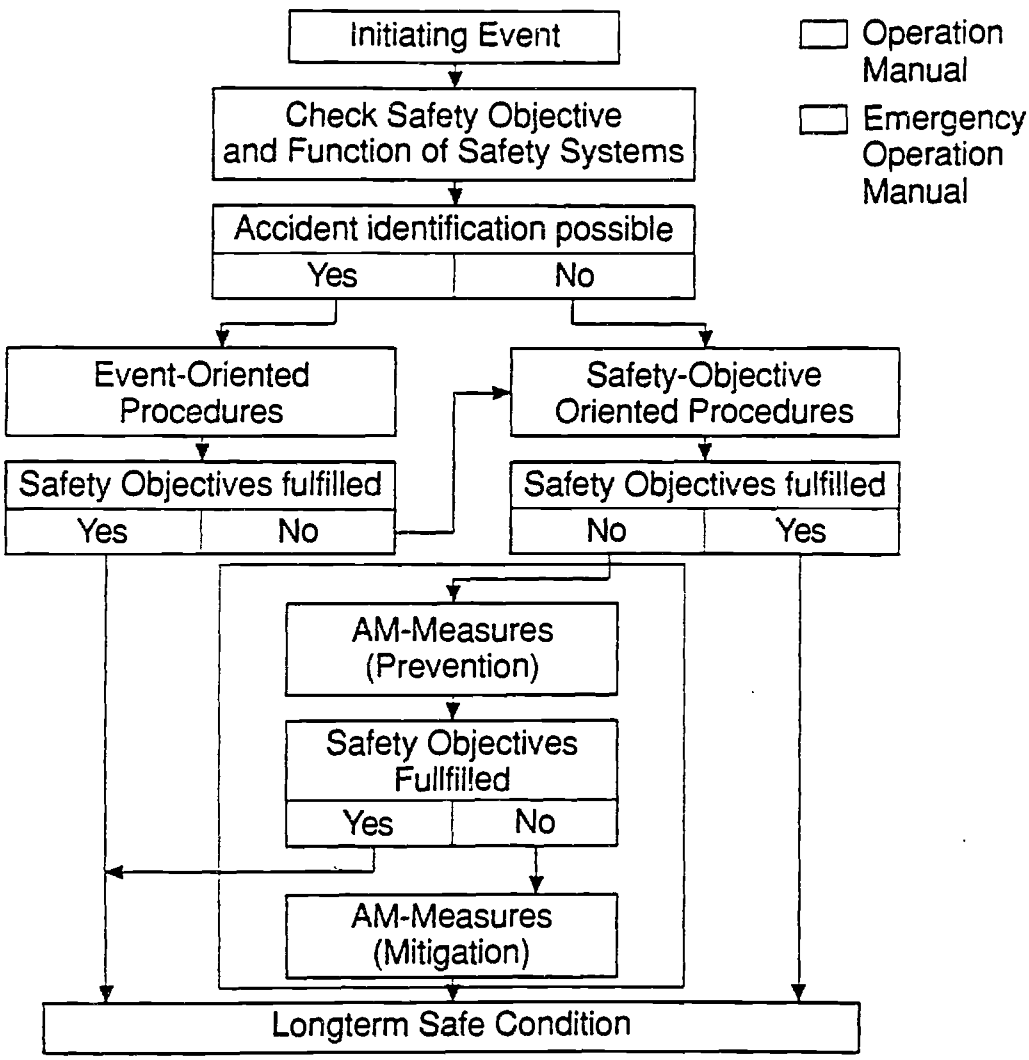
Fig. 9
Containment Venting Flow Diagram



Containment Venting

Venturi - scrubber

Fig. 10



3416K

Accident Decision Tree

Fig. 11

Emergency actions

(accident management actions)

in German BWR's

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Paper prepared for IAEA training course "Accident Management", Nov. 1990, Munich - FRG

1. Introduction

The systems in a German BWR can be used very flexibly. This advantage has been utilized for the prevention and also for the mitigation of severe accidents. According to the IAEA "critical safety parameters", a lot of emergency measures are specified in the "emergency manual". The following gives an illustration of these actions.

2. Emergency measures

2.1 Scope of the emergency measures

According to the defined "critical safety parameter" the measures can be divided into

- Containment venting (containment integrity)
- RPV injections (coolant inventory)
- Containment injections (coolant inventory)
- Activity filtration
- Power supply

2.2 Containment venting

2.2.1 Filtered containment venting

2.2.1.1 Background

The last barrier for the prevention of a high activity release in German boiling water reactors is a containment with an integrated condensation pool. This pool condenses the escaping steam during a design based accident. It also provides an alternative heat sink if the main heat sink is not available over a period of several hours. This leads to the well known compact containment design.

There are two types of containments, both designed by Siemens/Kraftwerk Union:

- Type 69: Würgassen (KKW), Brunsbüttel (KKB), Isar 1 (KKI-1), Philippsburg 1 (KKP-1) and Krümmel (KKK); and

- Type 72: Gundremmingen (KRB) plants B and C.

Figure 1 shows a cross section of each containment type. The main difference here is that the condensation chamber of the design 72 is placed directly on the ground plate of the reactor building. This lowers the design requirements on the containment and reactor building regarding an earthquake or an airplane crash.

Multi-layered redundant safety facilities ensure that the containment will remain intact, even when technical failure of components or malfunction of systems should occur.

For improvement of the acceptability of nuclear power, the German utilities proposed the installation of a filtered containment venting system. Later on the German Commission on Reactor Safety (RSK) recommended this and the German licensing authorities decided that measures should be taken to mitigate the impact of hypothetical events. Primarily the intention was focused on maintaining the integrity of the containment by a pressure relief via a special vent line.

The German utilities asked the reactor vendor Siemens/Kraftwerk Union to study possible event-sequences which could lead to a core melt. The result of these considerations are three scenarios of severe accidents.

Case A: Station blackout and loss of the turbine driven high-pressure injection system TJ.

In this case a hydrogen detonation is prevented, due to the inerted atmosphere in the containment. The portion of remaining oxygen must be less than 4 %. For the drywell of KRB the final decision has not yet been made. A nitrogen atmosphere might have advantages compared with ignition devices because of its much larger volume, compared with the 69-drywell. In any case the wetwell will be inerted.

Case B: Loss-of-coolant accident combined with a failure of all make up systems 10 min after the Loca has occured (failure of the second switch-over from pool cooling mode to RPV-flooding mode).

This case is characterized by a slow evaporization of the water inventory of the reactor pressure vessel and hydrogen production which increases the pressure within the containment to 4 bar.

Case C: Failure of all pool cooling systems after the loss of the main heat sink.

In case C the core will remain intact, but the permanently produced steam increases the pressure within the containment and endangers its integrity.

In the cases B and C failure of the containment would occur within approx. 4 to 12 hours and would lead to an uncontrolled release of activity. The filtered venting system should avoid high radiological consequences.

The mentioned 4 to 12 hours are the result of a rather conservative estimate because the heat capacity of the containment shell and its internals have not been considered.

2.2.1.2 Design criteria

The construction, design and function of the filtered venting system should comply with the following criteria:

A Use of existing structures for activity-retention

B No effect on existing reactor safety system

C Independence from normal power supply

D Design on the basis of conventional rules

E Simple design and construction with high reliability

F Maximum possible retention capability for aerosols

G No consideration of external events, such as earthquake or airplane crash

2.2.1.3 Design and function

The following explanations are related to figure 2.

As previously mentioned the German boiling water reactors, in contrast to the pressurized water reactors have an integrated condensation pool. The existence of a water volume between 2200 m^3 and 3700 m^3 - depending upon the plant power - with it's ability to condense steam and to remove aerosols, has made it possible to vent from the air space of the wetwell (see point A in design criteria).

The isolation of the condensation chamber and thus the containment is assured by two isolation valves in series. These valves are controlled by the reactor protection system. In this way the existing safety concept for the management of design-based accidents is not effected (see point B). The valves are battery powered and independent of the emergency diesel generators (see point C).

Immediately after the isolation valves the containment venting system begins. The design is based on German conventional rules.

A piping system connects the containment to the vent stack via a filter vessel, a check valve, a throttle and a rupture disk.

The rupture disk hermetically seals the system from the atmosphere and prevents the penetration of oxygen into the system. It will burst if the system pressure increases to 1,5 bar. This occurs immediately after the isolation valves have been opened by remote control from the control room. Thus the installation of an additional isolation valve could be avoided.

After the closure of the vent line the gases within the system will cool down and suck air into the system. This could lead to a hydrogen-oxygen-reaction. The installed check valve avoids this problem.

As figure 3 shows, the filter consists of a wet scrubber and a post aerosol filter. Both components are placed inside a pressure vessel, 4 m in diameter and 8 m in height. The scrubber is positioned in the lower part of the vessel. It is composed of a number of venturi nozzles positioned in a star pattern and submerged in a pool of demineralized water dotted with sodium hydroxide (ca. 0,5 %) and sodium thiosulfate (ca. 0,1 %). The gas, mixed with aerosols, flows through a central main pipe and the venturi nozzles into the scrubber. In the venturi nozzles the gas is accelerated and injected into the wash solution. The high differential speed between the scrubber fluid and the gas causes an interaction, providing a high washout for aerosols and elementary iodine. Subsequently a droplet separation is also installed. The performed tests have shown that organic iodine will also be retainted to 80 %.

The gas then flows to the post-aerosol filter elements in the upper part of the vessel. This filter consists of packages of stainless steel fiber fleeces. Even the finest aerosol particles will be deposited here.

The presence of radioactive aerosols does not lead to relevant scrubber fluid evaporation because the dominant portion of their decay heat is transferred through the filter vessel wall. Thus there is no need for a refill of scrubber fluid within days. Nevertheless a small pipe with a manually operated valve is provided. Shielding material protects the staff against radiation.

At least in two plants the filter vessel is positioned on the floor of the fuel storage pool. In order to facilitate the access to this area after venting, the highly activated water in the scrubber could be fed into the wetwell. After that the vessel would be flooded again with clean water up to the top.

Figure 4 shows the steam and gas mass flow through the system versus the pressure in the wetwell. Case C - loss of residual heat removal systems - is the case which determines the size of the vent line. The system is able to relieve steam produced by decay heat of 1 % of the thermal power of the reactor.

Figure 5 shows the pressure in the wetwell versus time. After about 24 hours the pressure will reach about 6,3 bar abs., if the venting valves are opened at 4,5 bar abs. The corresponding pressure for case B, shown in figure 6, increases to a value of 5,2 bar abs., about 4 hours after the initiation of the venting and allows to exhaust a mass flow rate of 7,3 kg/s. In both cases the pressure level does not exceed the test pressure of the containment (5,5 bar abs.) and is far from the failure pressure of the containment (9 to 11 bar plant specific).

Figure 7 gives some hydraulic data (pressure and velocity) at different points of the system for the two cases mentioned above.

The relevant data of the BWR containment venting system are compiled in table 1. Figure 8 shows the filter of the NPS Krümmel, placed on the floor of the storage pool; and figure 9 the two isolation valves of the venting system close to the containment.

2.2.1.4 Effectiveness

The explained pressure relief system ensures that a failure of the containment in a severe accident design can be prevented.

Beyond that the described filter is able to reduce significantly the radiation exposure in the environment. As Figure 10 indicates, the radiation exposure from gammasubmersion and inhalation is expected to be only about 5 % compared with the exposure caused by an uncontrolled activity release via the turbine building.

2.2.2 Containment venting via steam line

As a redundant measure to the filtered containment venting, a containment venting via the 5. steam line is possible. The flow path is shown in figure 11. The steam or the mixture of steam introgen flows into the feedwater storage tank and then into the condenser. After the opening of the rupture disks the activity will be released into the environment via the turbine building. This measure can prevent an uncontrolled failure of the containment. If the core is covered with water, only steam with low activity concentration will be released.

2.3 RPV-injections

2.3.1 Injection by steam driven high pressure injection system TJ

This injection is of special interest in a "station blackout" scenario. The power supply of this system is independent from all AC-busses. The capacity of the installed batteries is sufficient to operate the system for more than 3 hours (schematic system diagram see fig. 12). This time is long enough for a repair of the main grid (supply with house load) or the switchover to an independent hydroplant (KKK) or gas turbine (KKB).

Due to the unavailable pool cooling system in a "station blackout", some overriding actions must be performed to ensure the operation of the system for a longer time. After approximately 1 hour the initiation of the automatic depressurization (ADS) by the main relief valves must be avoided. Therefore the set points of the high temperature in the pool ($\geq$ 55°C in KKK and $\geq$ 60°C in KKB) must be overrided. These actions must be performed in the switchgear building close to the main control room. Necessary are only 2 simulation adapters which are placed in a special locker in a room directly beside the main control room. The second necessary action is the overriding of the ADS which is initiated if the pressure in the containment is > 1,25 bar. This setpoint is the second criterion to the low coolant level in the

RPV in a "LOCA". An additional action to reduce the mass flow of an ADS is the reduction of the RPV-pressure to 20 bar. If these actions are performed successfully a sufficient coolant level in the RPV can be ensured for more than 3 hours. If it is obvious that the repair of the grid or the switchover to the diversified power supply is not possible, a manual depressurization is necessary to ensure that an injection by the feedwater storage tank or a mobile pump is possible. During the blowdown the TJ-system is able to keep the coolant level more than 5 m above the top of the active core. The evaporation of this volume due to the residual heat takes a time of more than 2 hours. As described, the minimum delay between the shutdown of the plant in a station blackout and the start of the core heatup is $\geq$ 4 hours if the RPV-injection by the TJ-system can be performed sucessfully.

2.3.2 Injection by feedwater storage tank

This injection is also of special interest in a "station blackout" scenario. The schematic injection way is demonstrated in figure 12. Due to the loss of AC-power the motor valves in the feedwater system will fail in the open position. Only one check valve is equipped with solenoid pilot valves, but these valves close only if the coolant-level in the RPV is extremely high. This condition is only possible after the injection of about 80 % of the volume in the feedwater-storage tank. If the protection system has closed the check-valves (coolant level extremly high), it is possible to reopen them by manual measures from the control room or by manual measures permformed in the turbine building. The necessary measures in each plant condition are described in detail in an emergency procedure.

2.3.3 Injection by fire-protection pumps (UD) from the demineralized water storage tank

This measure is possible in order to prevent core damage if more than 1 hour is available for the preparation and the performance and if the RPV is in a low pressure state. The injection path is demonstrated in figure 13. The only necessary measure, needing about 15 to 30 mins, is the connection of 4 fire-hoses between the systems UD (demineralized water system) and the spent fuel pool cooling system TG. The distance between the two systems is less than 50 m. All necessary devices for the establishment of this connection are collected in a special box close to both systems. All necessary manual measures are specified in detail in an emergency procedure. Also described is the way the personnel has to choose the course for the fires hoses (see figure 14). The injection path is from the demineralized water storage tank via the fire protection system UD, spent fuel pool cooling system, make-up system and the feedwater system into the RPV by use of the fire protection pumps. The possible massflow is more than 50 kg/s, which is enough to ensure a sufficient coolant level in the RPV approximately 30 min after a scram (a bottom leak excluded). The volume of the demineralized water tank is sufficient to keep the core cooled for more than 8 hours. After that time the refill of the tank by the clear water pumps UA is necessary. This procedure is a well trained measure described in the normal operation manual. The refill of the clear water tank is possible from the district drinking water supply system of the local city. The UD-pumps are powered by the emergency diesel generators, which means that an operation is possible in case of a loss of offsite power supply but not in case of a station blackout. The district drinking water system UK has also its own emergency power supply, totally independent of the power plant.

2.3.4 Injection by district water supply system UK

One possible injection of drinking water into the RPV is described in chapter 2.2.4. Another direct injection is possible via the UD-system directly into the spent fuel pool cooling system (see fig. 14). For this measures two connections with 4 fire hoses must be established on the one hand between the drinking water system UK and fire protection system UD and spent fuel pool cooling system TG and fire protection system UD on the other hand. The maximum possible flowrate is about 66 kg/s. The maximum pressure is about 8 bar. Due to the independent power supply of the drinking water system, no power supply from the plant is necessary. All valves in the flow path can be operated manually.

2.3.5 Injection by mobile pumps or fire brigade pumps

At the KKK site two diesel driven mobile pumps are available. They are installed in special containers with electrical heating systems during the cold seasons. These pumps are possible redundancies if one or two fire protection pumps are unavailable due to maintenance. The containers are located close to the turbine building. To carry out this injection two connections with four fire hoses are necessary (see fig. 13 and 14). The maximum distance between pump and fire protection system is less than 50 m. The location of the systems are shown in figure 14. The maximum flowrate is about 66 kg/s at 12 bar. The necessary time to prepare and perform this measure is less than 2 hours.

The same task can be performed by use of a mobile pump of the local fire brigade.

2.4 Containment injection

An injection of water into the containment can be carried out in different ways. The easiest way is the use of the fire protection system located in the containment cavity. The flow path is shown in picture 15. The measures which must be performed is the overriding of the interlockings of the isolation valves and tne start of the fire protection pumps.

2.5 Filtration of main control room air

This measure is not a BWR-specific emergency action. If a venting of the containment is necessary the air of the main control room must be filtered to avoid a high radiation dosis for the plant personnel. The difference between the PWR- and BWR-concept is the fixed installation of the filter in a BWR, due to fact the venting of the containment is necessary about 4 hours after the accident occurs.

2.6 Alternative emergency power supply

Most of the German nuclear power plants can activate an alternative emergency power supply in a station blackout. Two examples are described.

2.6.1 KKB-plant

This BWR-plant has a diversified emergency power supply system with five emergency diesel generators. Only one of these diesels is necessary for the shut down of the plant. Two diesels are smaller and are manufactered by another vendor. The control system and the generator-voltage are different. The two diesel are located in a separate building. The frequency of a station blackout is estimated to be $1 \cdot 10^{-7}\ ^1/_a$ due to the high redundancy and diversity. A further reduction of this frequency is possible by the use of two gas turbines and the turbine driven high pressure injection system TJ. The TJ-system ensures a sufficient coolant level in the RPV and avoids an automatic depressurization. When the station blackout is identified by the plant pesonnel, they can start the two gas turbines from the main control room. With these turbines the supply of both emergency power supply systems or the house load busses is possible in approximately 10 min. The design of the entire electrical power supply system is shown in figure 17. This possible emergency measure reduces the frequency of an uncontrolled station blackout to a level which is negligible ($< 10^{-8}\ ^{1}/_{a}$).

2.6.2 KKK-plant

This BWR-plant has five emergency diesel generators which are important for safety. They supply five systems for core flooding and residual heat removal. One additional diesel has no safety importance. The turbine driven high pressure injection system is also powered by batteries which allow a system operation independent of all AC-busses. When the station blackout is identified, the switchover to a hydroplant is possible. The manual measures must be performed in the switchgear building. The required time is less than 30 mins. The hydro plant has 3 turbines of 35 MW electrical power. The design of the electrical power supply systems is shown in figure 16. This alternative power supply reduces the frequency of an uncoped station blackout significantly ($< 1 \cdot 10^{-7}\ ^1/_a$).

Design Pressure	11 bar abs. [2]
Venting Pressure	4,5 bar abs. [3]
Design Temperature	190 °C [4]
Piping Nominal Diameter	250 mm
Maximum Flow Rate (Steam)	13,6 kg/s [5]
Material	Stainless Steel

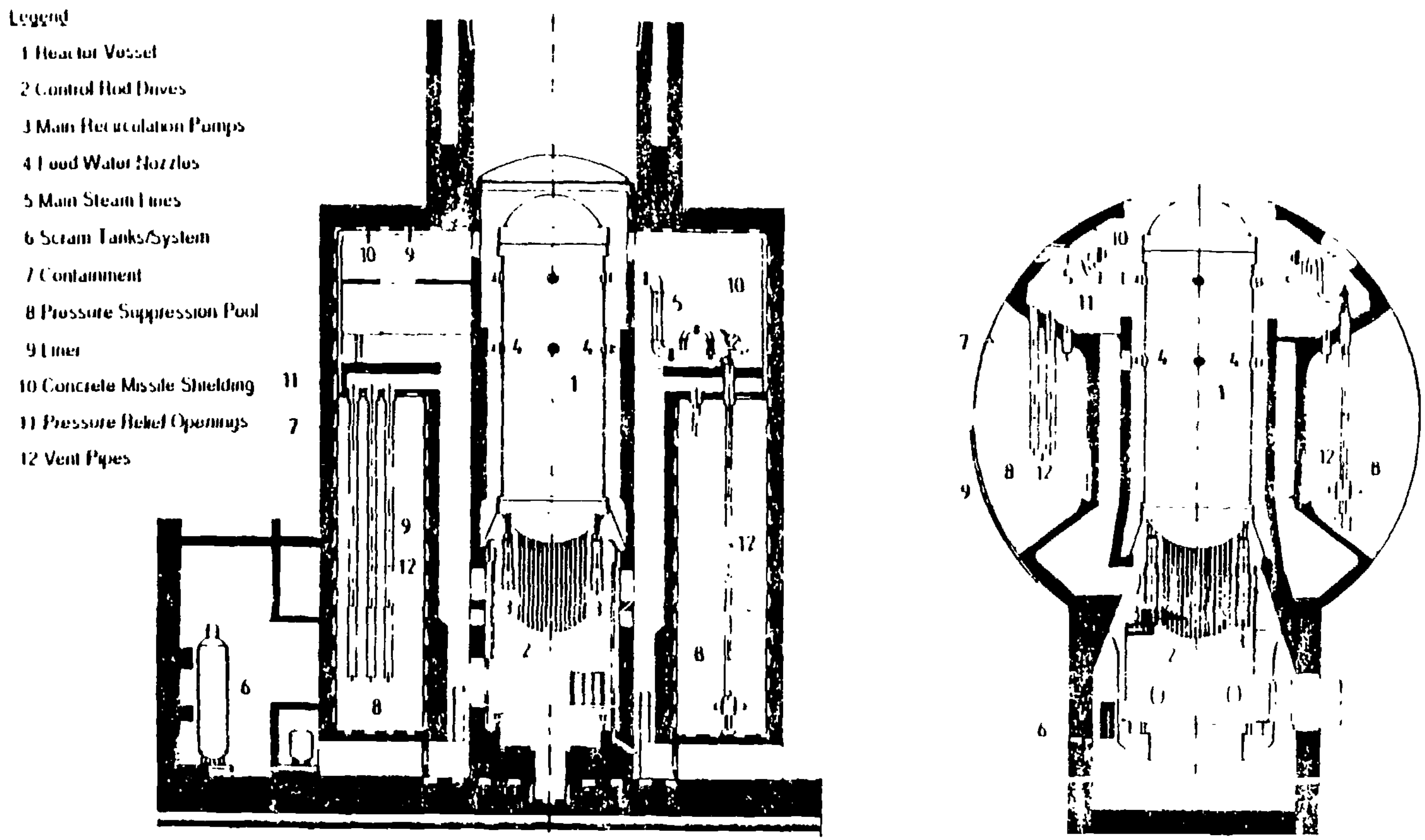
Filter-Data:

Design	Multi Venturi Scrubbers with Aerosol-Filter (Stainless Steel Wiremesh)
Dimensions	12 m in Height 4 m in Diameter
Wash Solution	Deionized Water with Na(OH) (0,5 weight-%) and $Na_2S_2O_3$ (0,1 weight-%)
Volume of Wash Solution	27 m^3
Decontamination Faktor (expected)	
for Aerosols	10 000
for Elementary Iodine	100

1) The data from other BWR of the design 69 deviates to some extent depending on the plant's capacity
2) Correspondence with the recommendation of the German Reactor Safety Commission from March 5, 1987
3) Design pressure of the containment
4) Corresponding saturation pressure is 12,5 bar abs
5) This corresponds to a volume flow rate of ca. 3,3 m^3/s at 8 bar abs.

HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM

Table I: Relevant Data for NPS Krümmel [1]



HEW 1988 | **BWR FILTERED CONTAINMENT VENTING SYSTEM**

Figure 1: Cross-Section of BWR 72 Containment (left) and BWR 69 Containment (right)

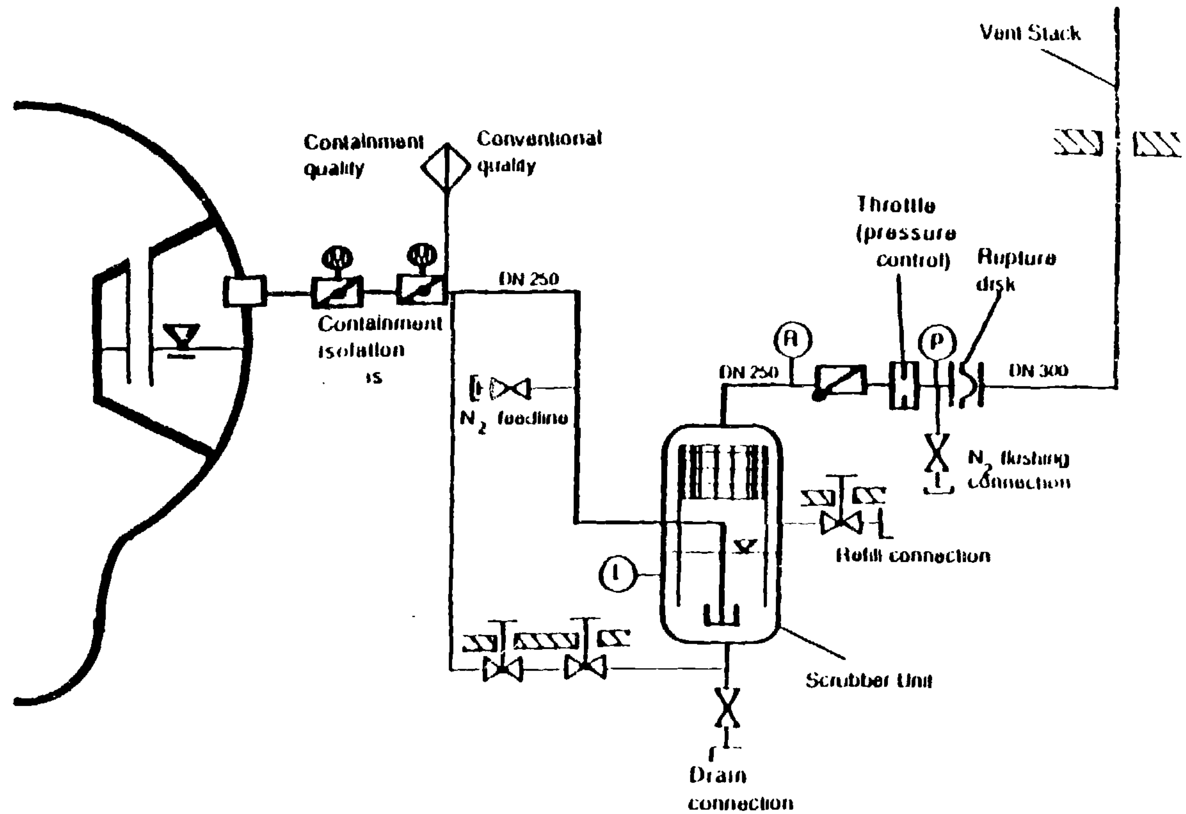


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Figure 2: Sketch of Flow Diagramm

Original: KWU

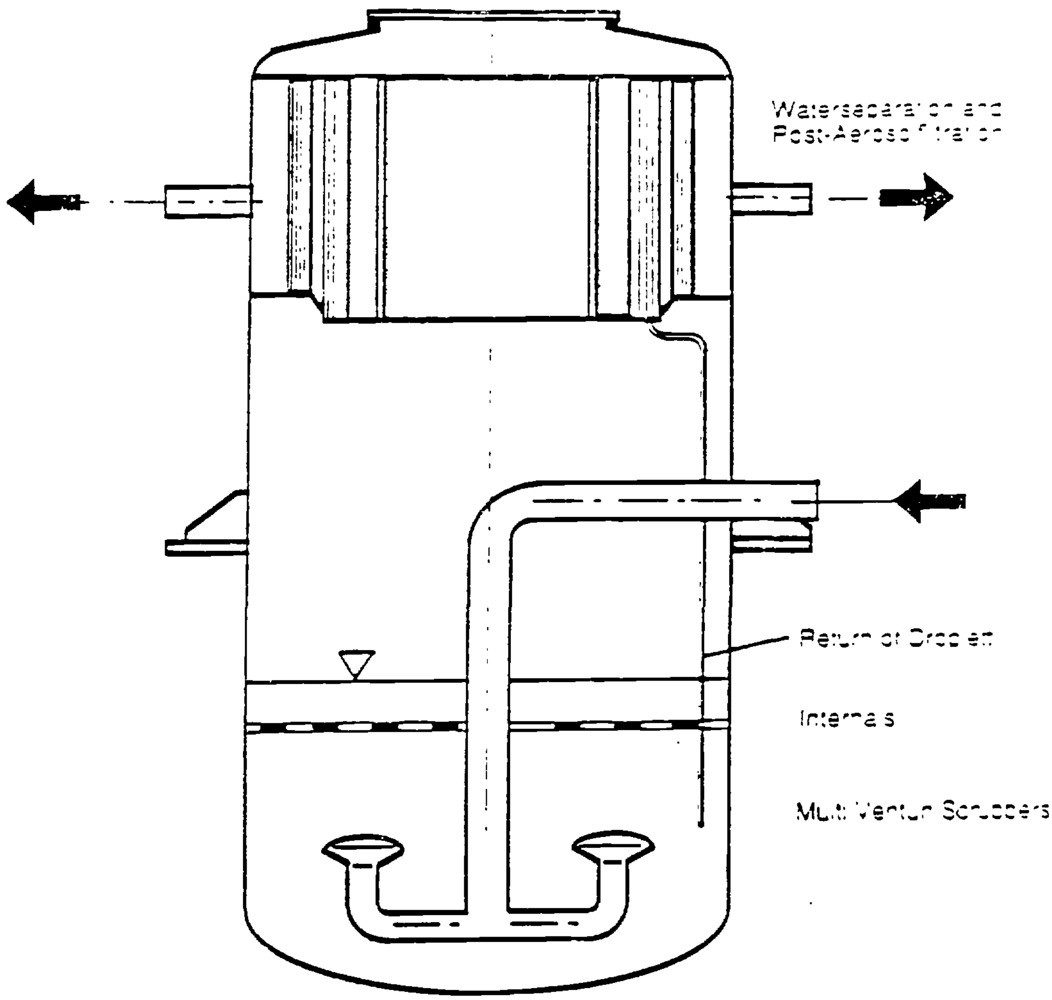

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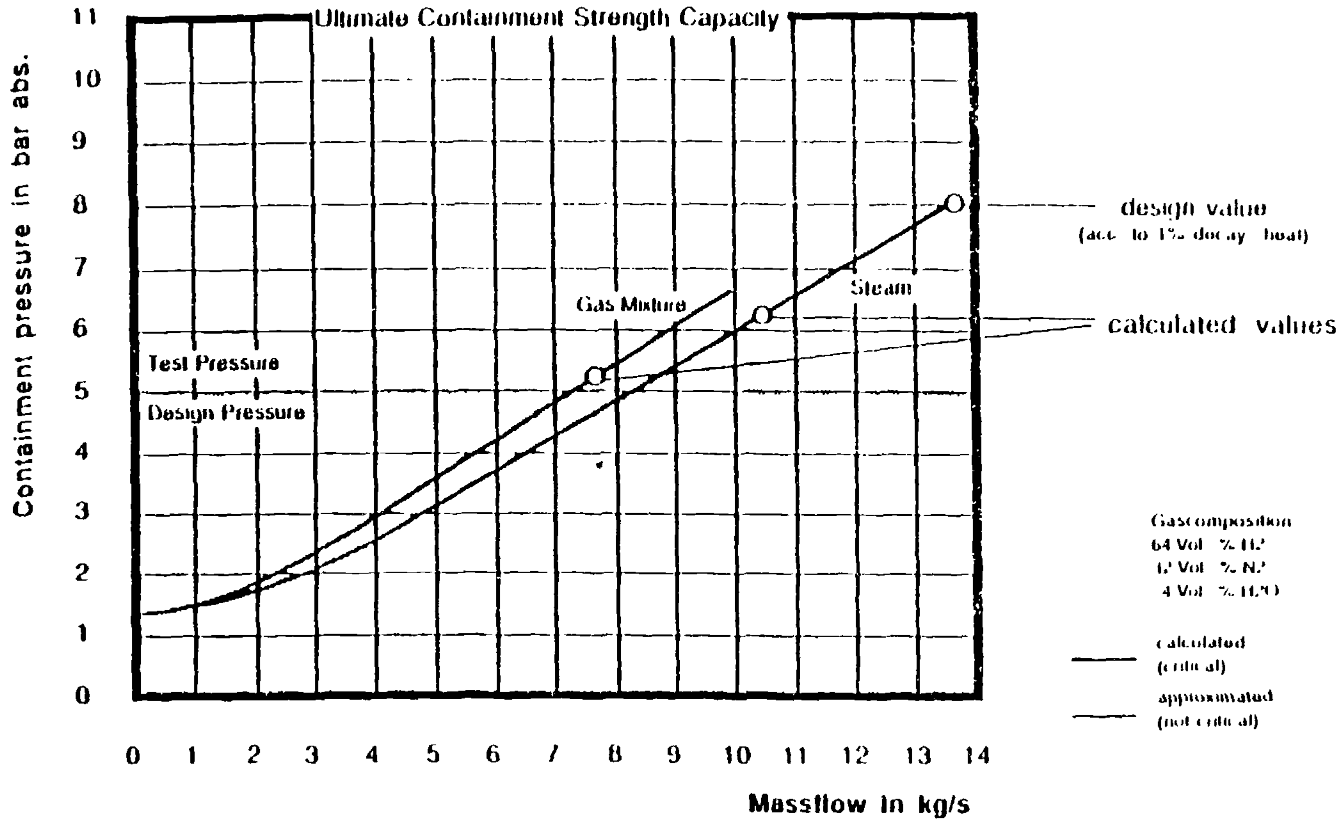
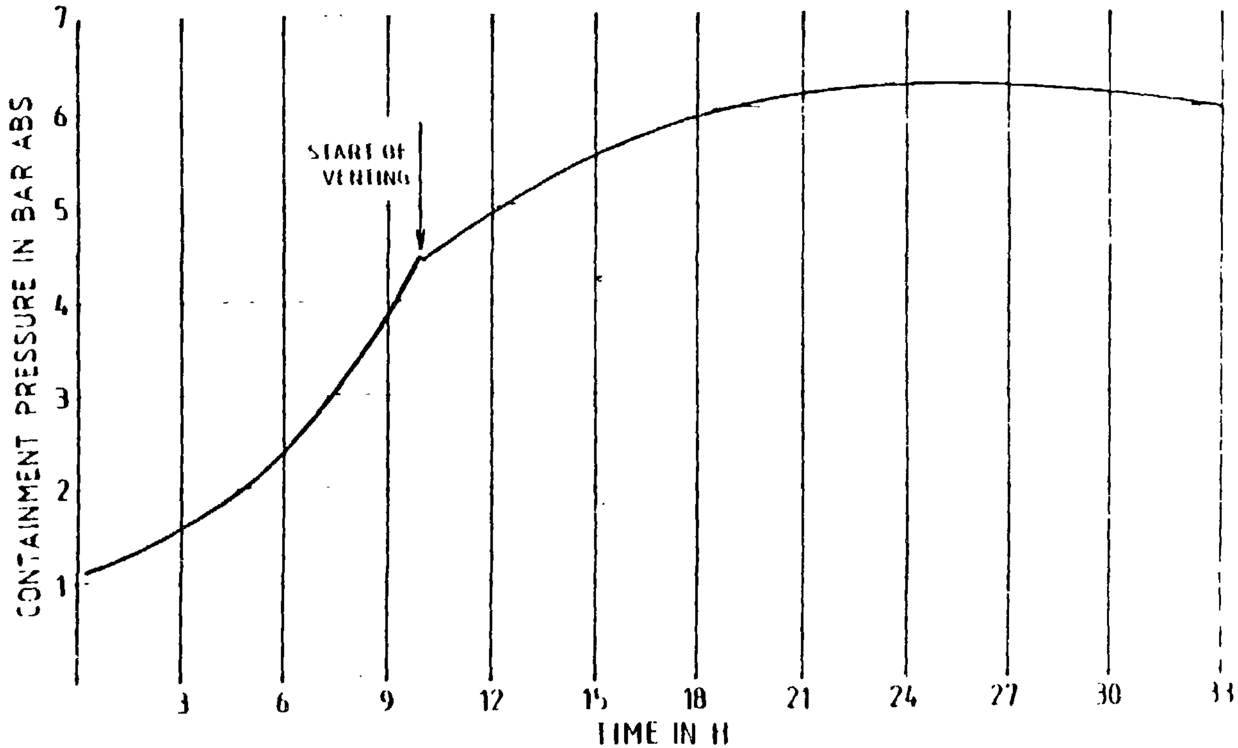
Figure 3: Venturi Scrubber Unit	Original KWU



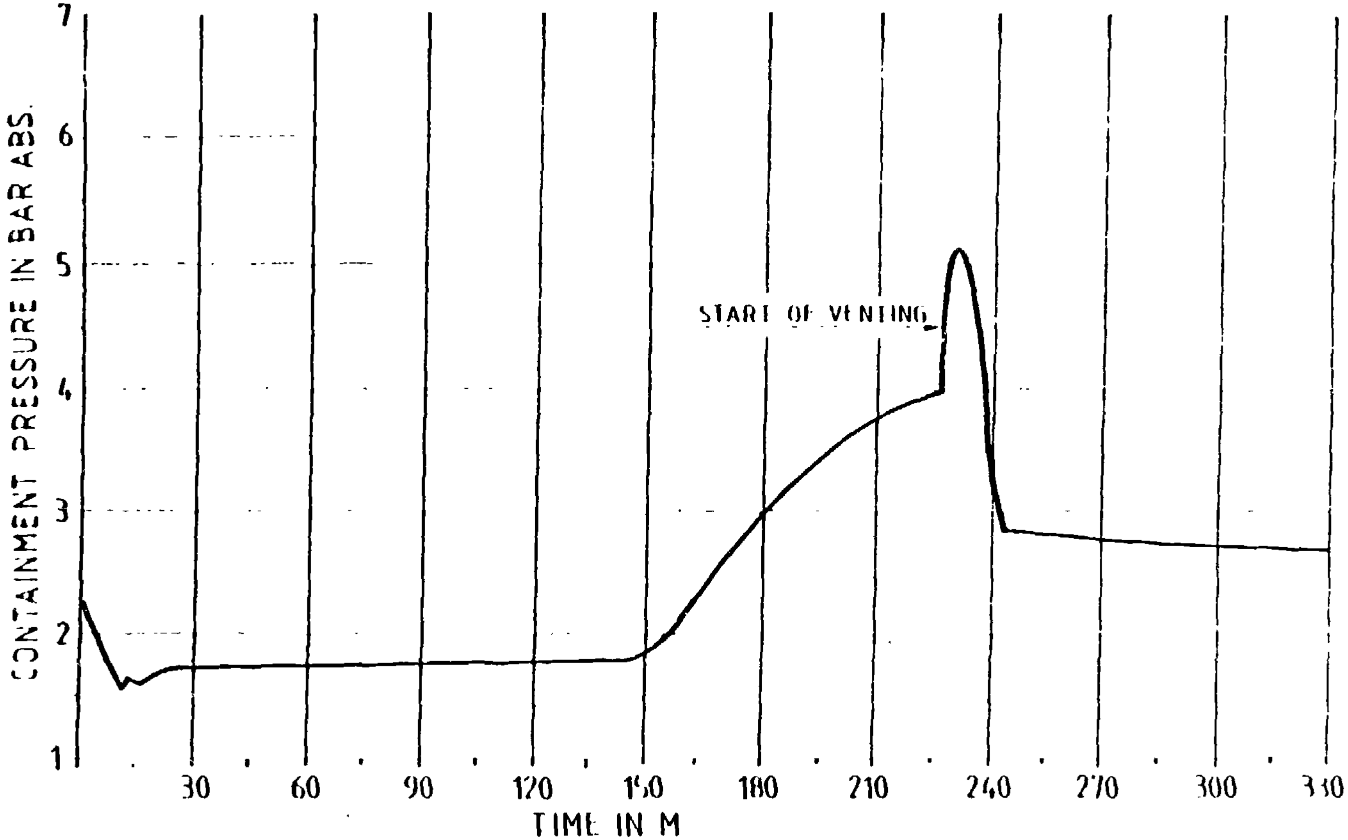
HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM

Figure 4: Massflow versus Containment Pressure NPS Krümmel	Original: KWU



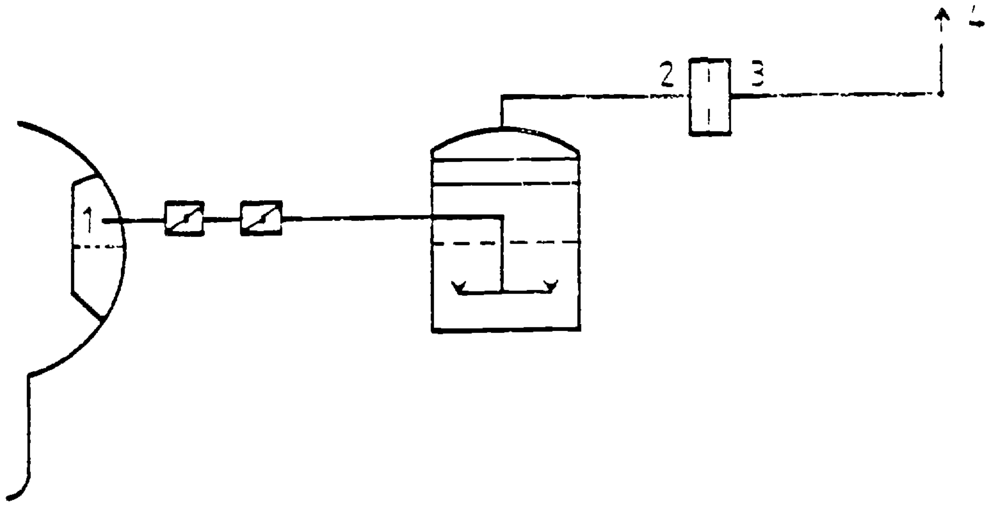
HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM

Figure 5: Containment Pressure versus Time, Case C, NPS Krümmel	**Original: KWU**



HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM

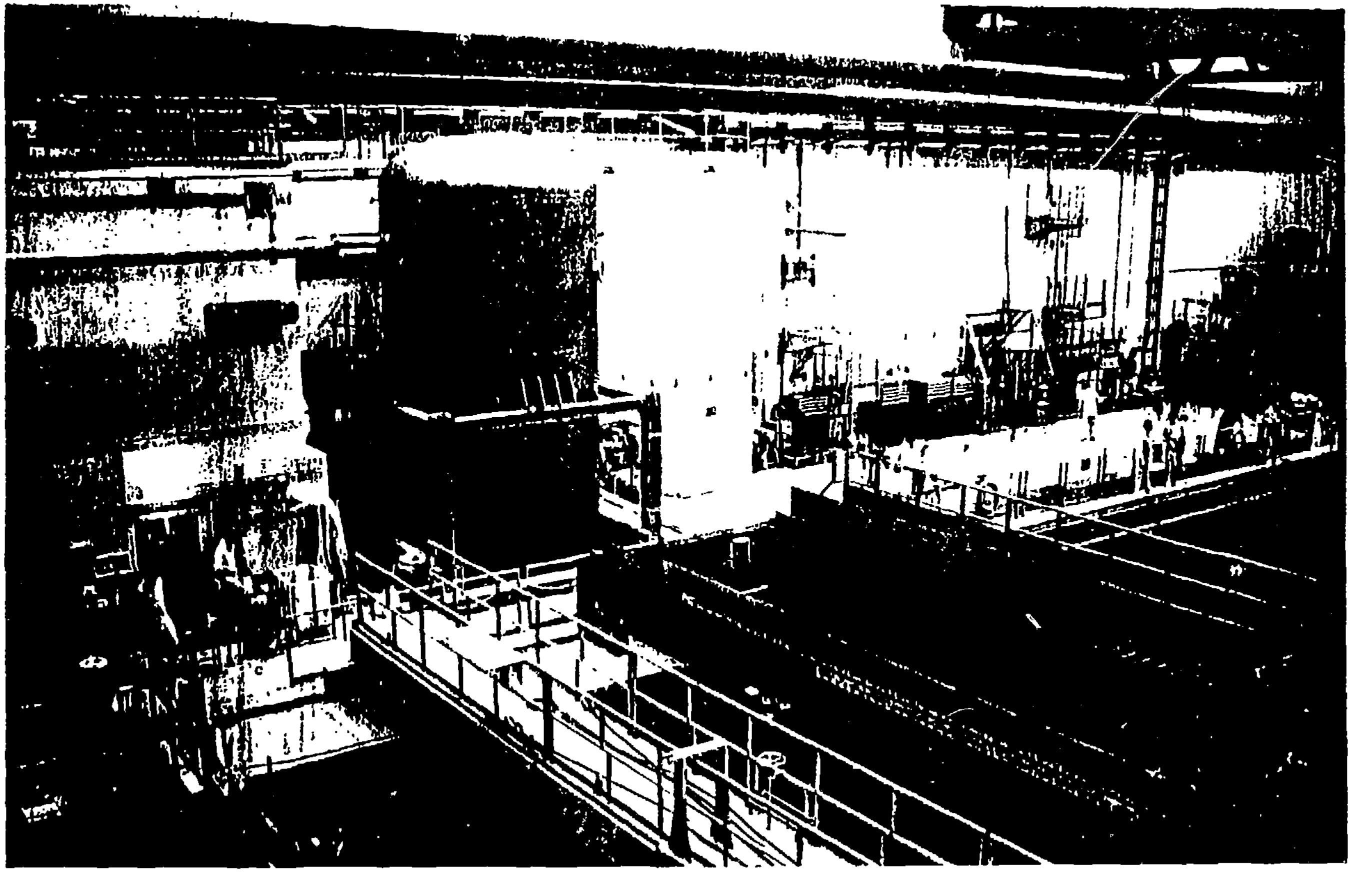
Figure 6: Containment Pressure versus Time, Case B, NPS Krümmel	Original: KWU



	Case B		Case C	
	bar	m/sec	bar	m/sec
1 Wetwell	8,0	67	4,5	--
2 Before throttle	6,0	86	3,1	112
3 After throttle	3,4	150	1,6	218
4 Ventstack outlet	1,4	342	1,3	269

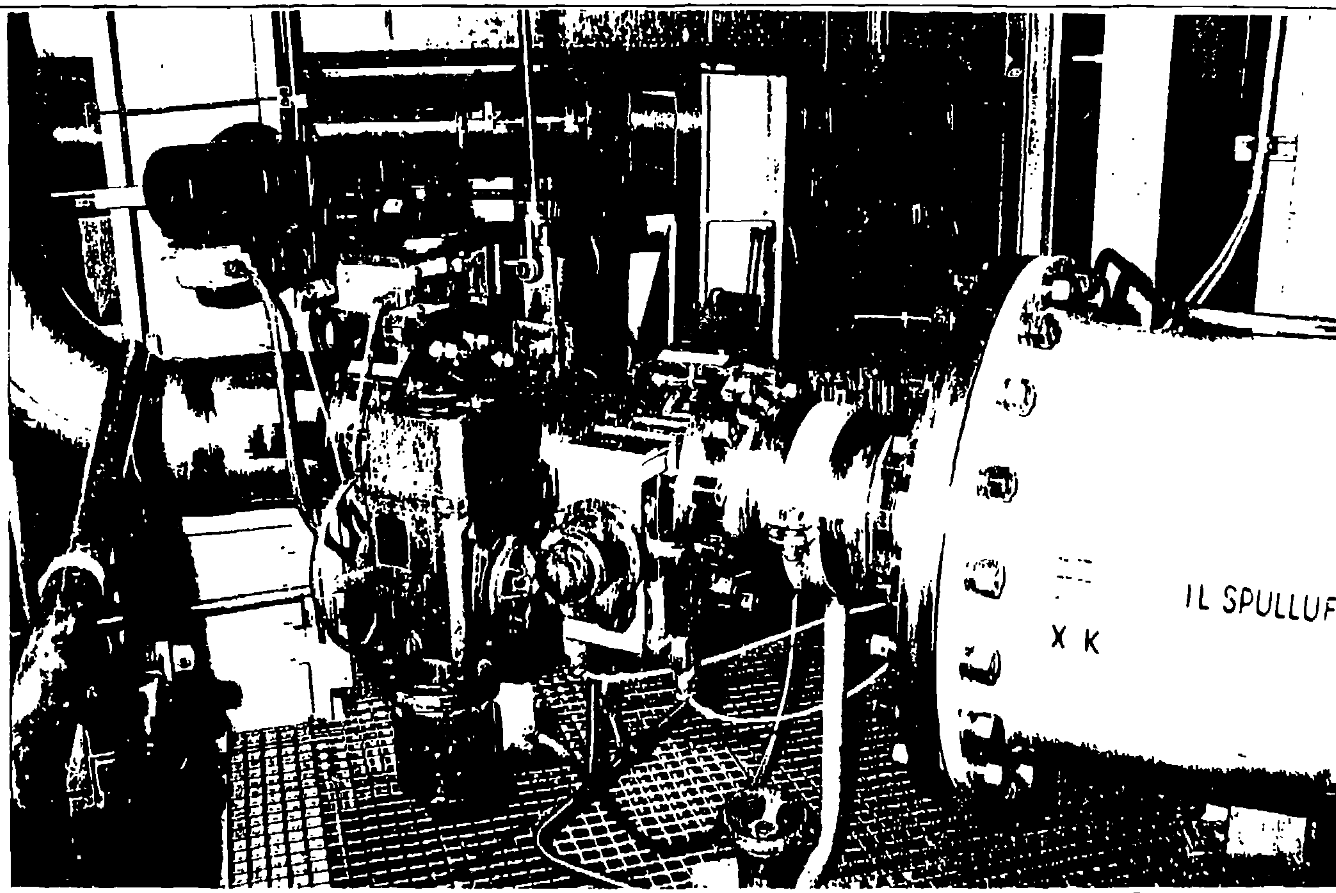
HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM

Figure 7 : Data of Pressure and Velocity for Various Components of NPS Krümmel



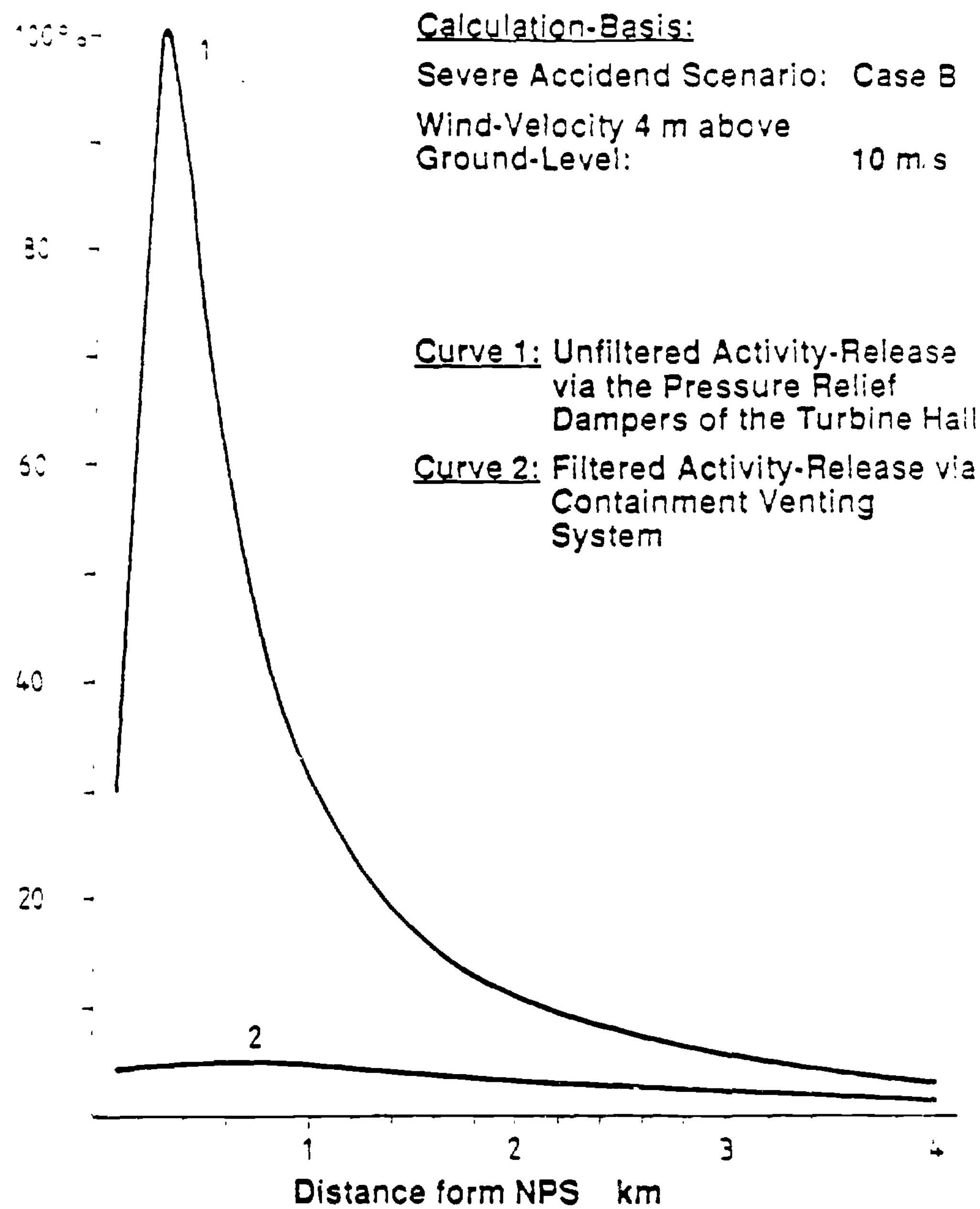
HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM NPS KRÜMMEL

Figure 8: View of the Filter Vessel on the Floor of the Fuel Storage Pool



HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM NPS KRÜMMEL

Figure 9: View of the Containment Isolation Valves



HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM

Figure 10: Relative Radiation Dose from Gammasubmersion and Inhalation versus Distance from NPS Krümmel

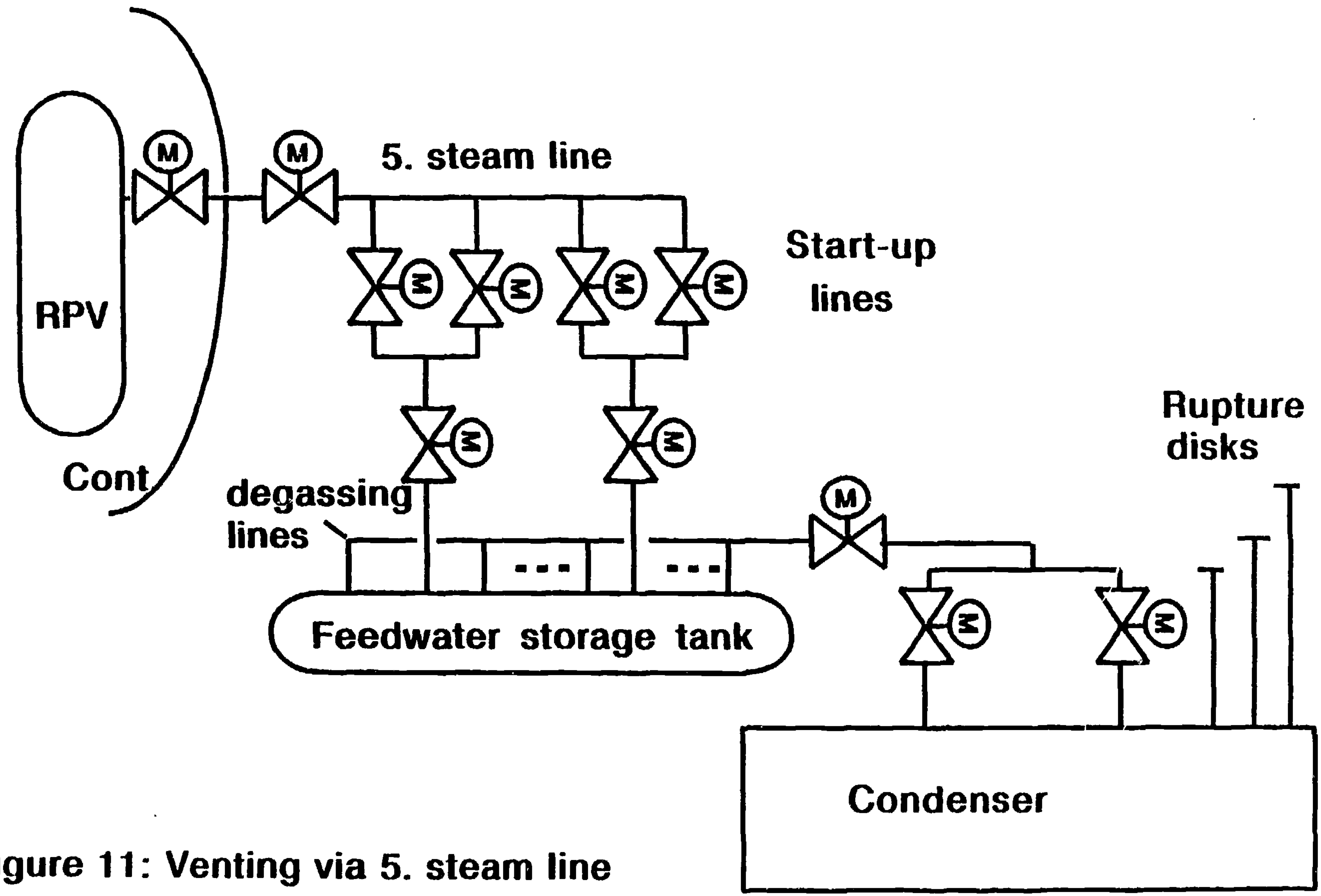


Figure 11: Venting via 5. steam line

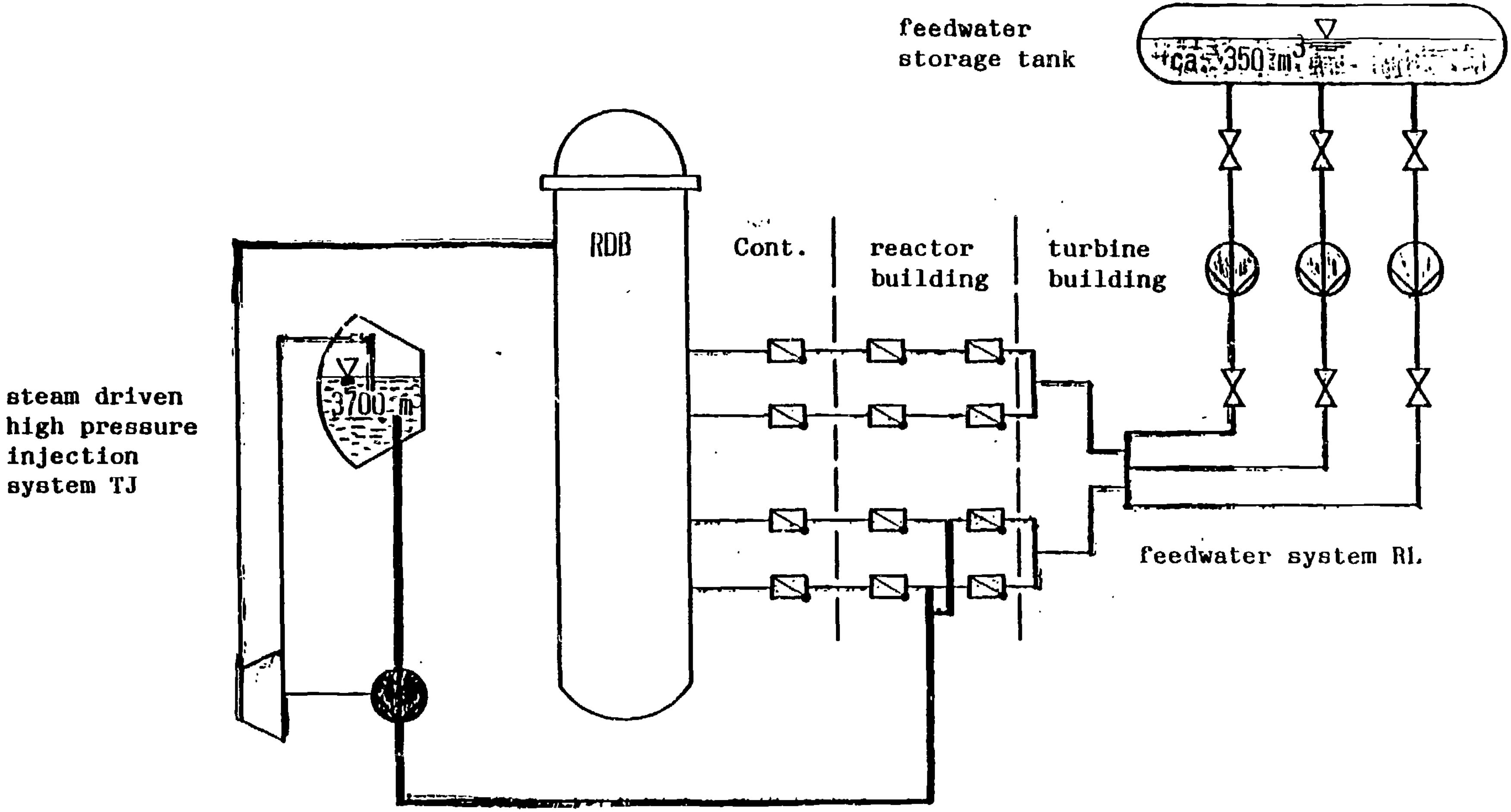


Figure 12: RPV-injection by TJ-system and feedwater storage tank

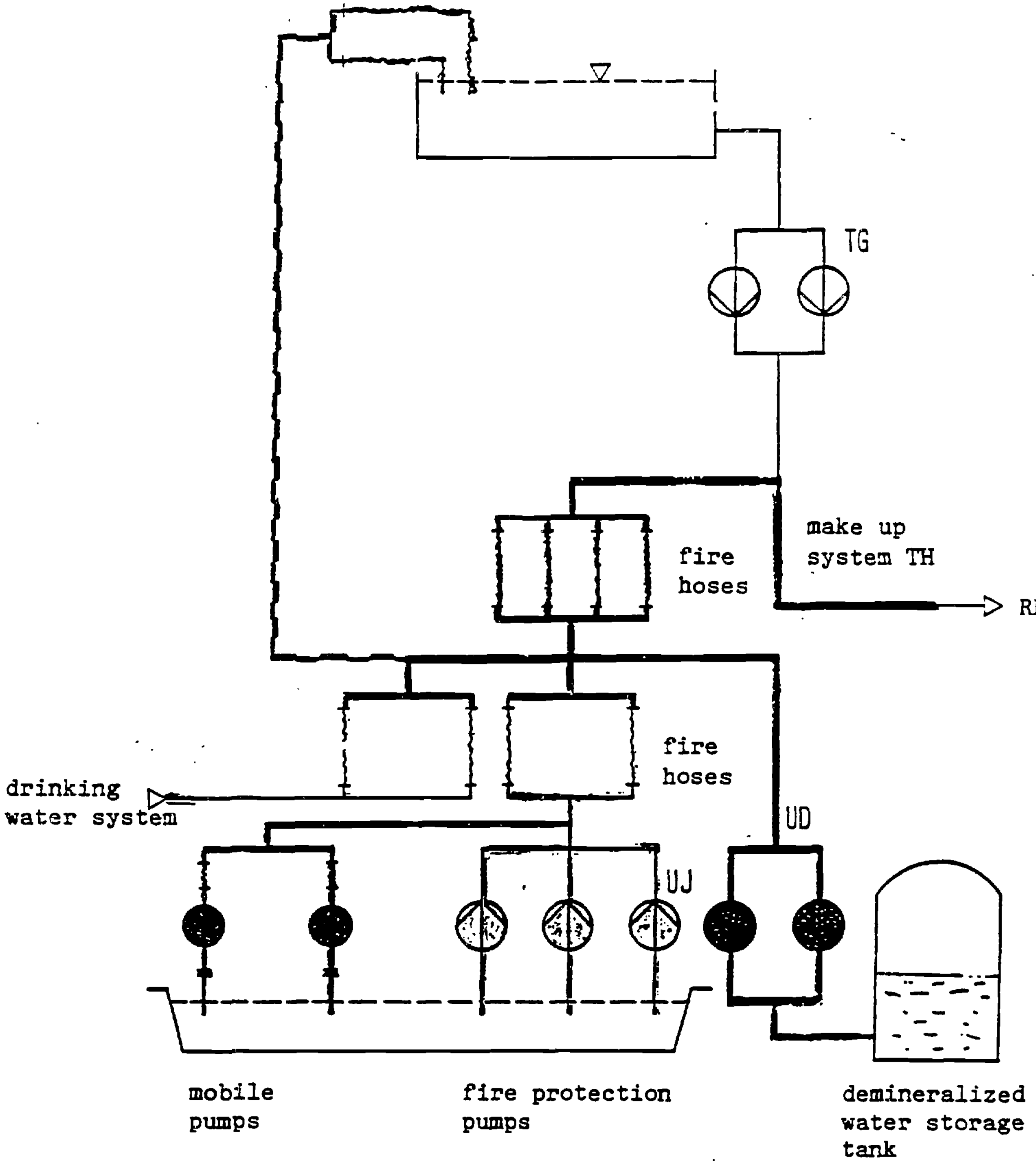


Figure 13: RPV-injections by mobile pumps, fire protection pumps and drinking water system

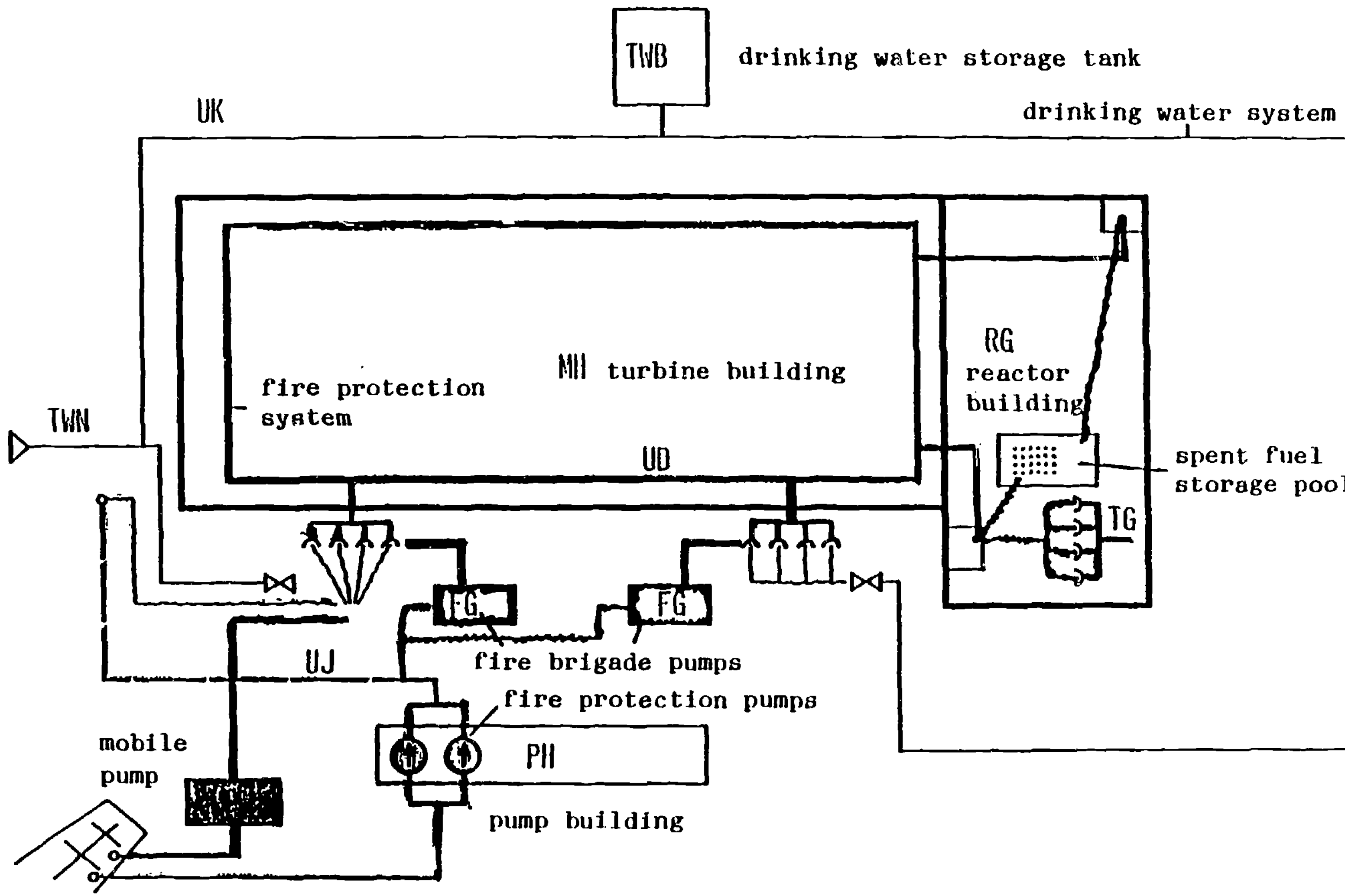


Figure 14: <u>Location of different systems used for accident management actions</u>

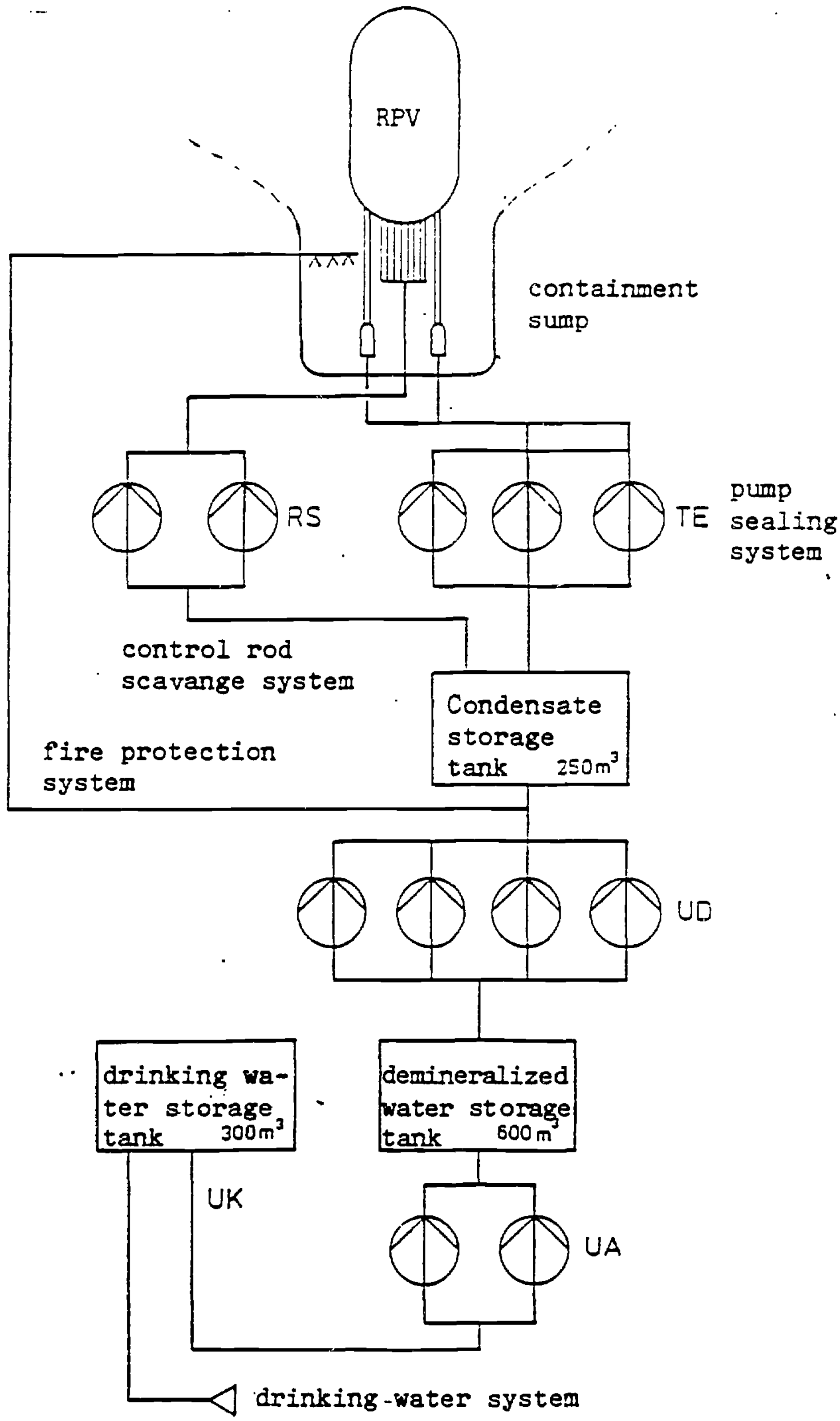


Figure 15: <u>RPV-injection by systems RS and TE, containment injection by system UD</u>

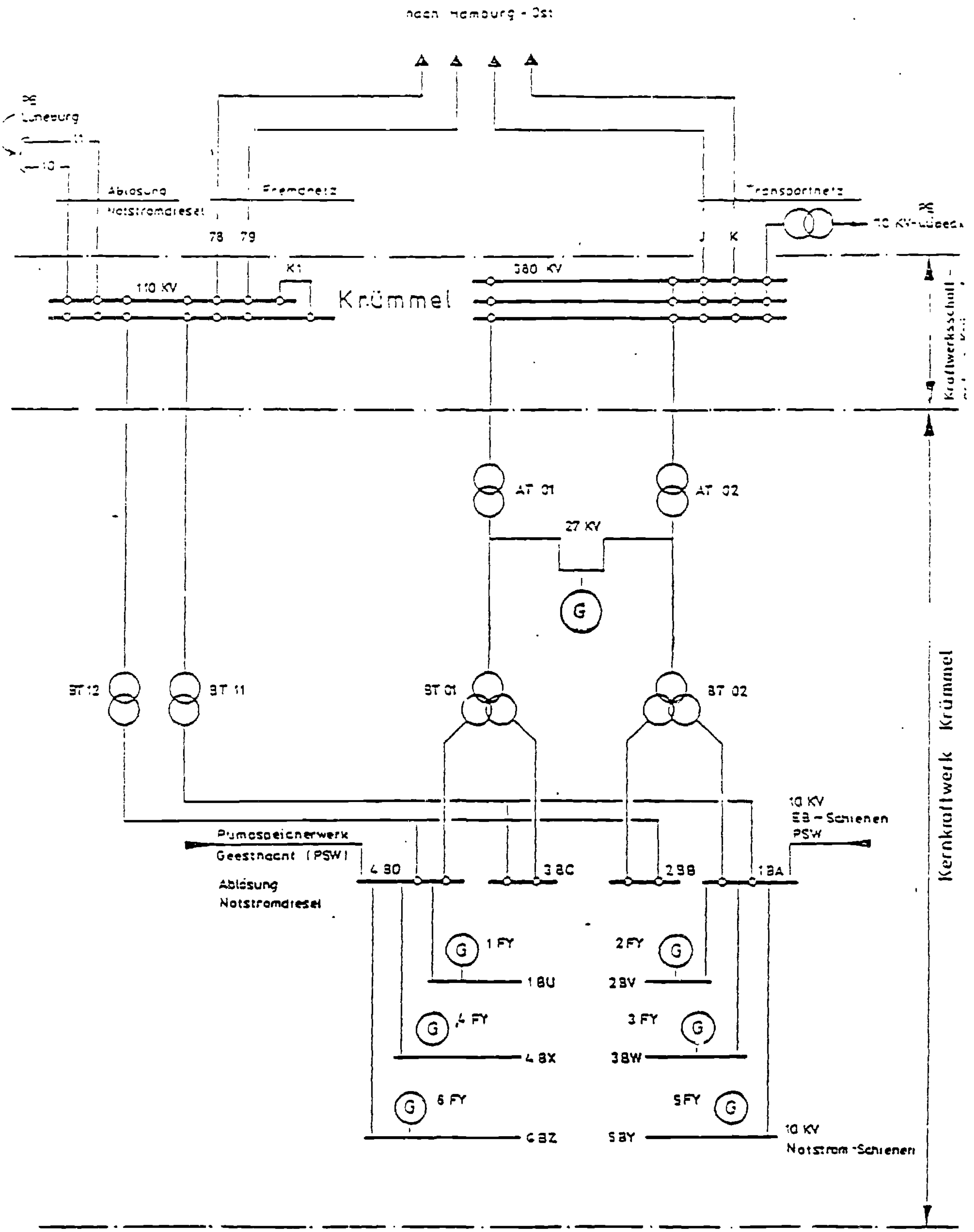


Figure 16	Grid connections, house load busses and emergency power busses plant KKK

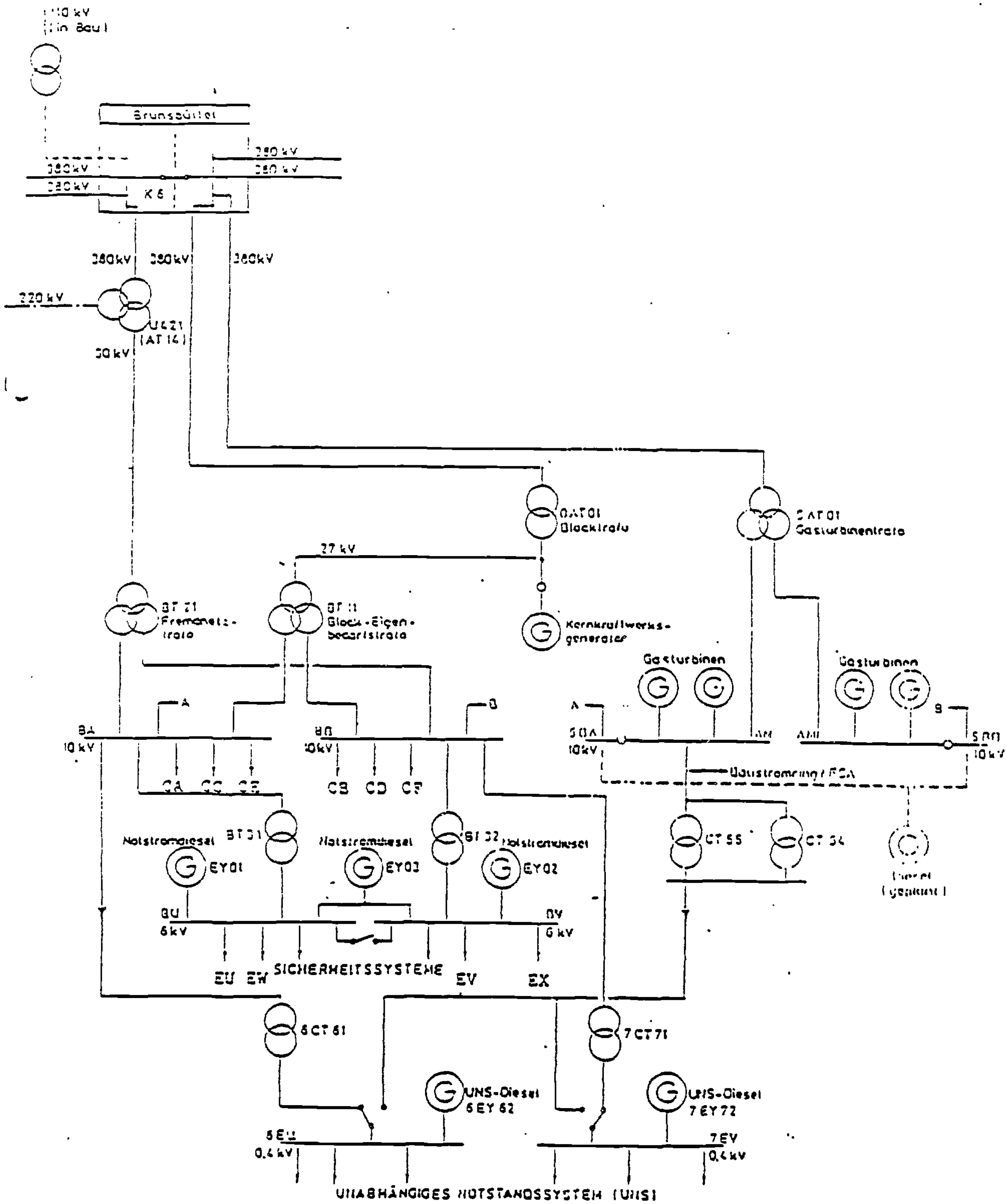
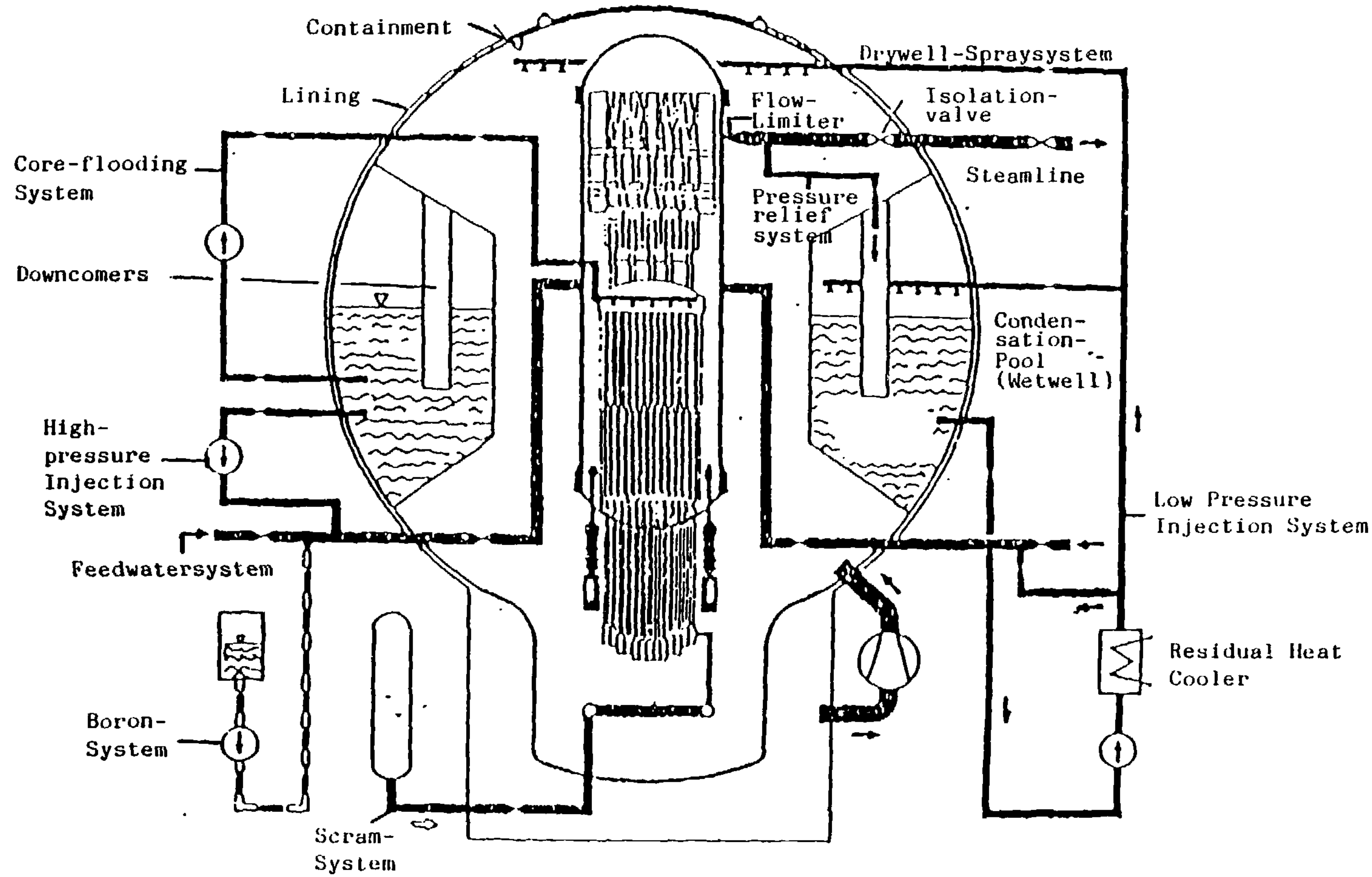


Figure 17	Grid connections, house load busses and emergency power busses plant KKB

KKK

Safety-Systems BWR 69



Safety Systems BWR 69

IAEA: Technical Committee Meeting on
Plant System Utilization for Accident Mitigation

Secondary and Primary Bleed & Feed, an AM-strategy to prevent and mitigate Severe Accidents

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ABSTRACT

Core melting at high primary pressure can lead to an early containment failure and uncontrolled large release of radioactivity. To prevent these events several accident management measures have been developed. In this paper secondary and primary bleed and feed measures will be discussed.

The effectiveness of these measures has been investigated by calculations with a best estimate code. The results of these investigations are presented. Secondary bleed and feed using feedwater line/-tank inventory or mobile pumps and primary bleed and feed with high pressure safety injection pumps are effective to prevent core melt. The efficiency of primary feed with accumulators is limited to delay core melt. A detailed description of the system behavior, including initiation criteria, delay times and feasibility is given.

1. INTRODUCTION

Nuclear power plants in the Federal Republic of Germany are equipped with comprehensive and reliable safety systems which prevent core damage.

However, beyond this area of design basis accidents remains an area with possible sequences against which the plant has not been designed explicitly, because these sequences have been assessed to be very unlikely. The causes for these sequences might be

- multiple failures of systems and components
- delayed or missing detection of disturbances
- human error.

Concepts and measures to prevent a core melt or mitigate its consequences which are not explicitly considered in the design are internationally known as accident management (AM). Accident management includes all measures which are initiated in a plant to identify as early as possible deviations from design basis sequences, to diagnose, control and terminate them with minimum damage.

The dominant part of accident sequences which potentially can lead to core melt is due to uncontrolled transient events and small leaks in the primary circuit [1]. Concerning transient events, the core melt frequency will be determined by the failure of the heat removal via the steam generators, i.e. the loss of the feedwater supply by the main feedwater pumps as well as by the auxiliary feedwater and emergency feedwater pumps, respectively. Without accident management measures core melting and subsequently failure of the reactor pressure vessel will occur when the primary pressure is high. So the integrity of the containment is endangered.

A strategy to influence high pressure core melt scenarios and to reduce the probability of early destruction of the containment has been developed, based on various qualitative and quantitative investigations [2, 3, 4].

In case of transients the loss of primary coolant is limited between 10 Kg/h and 120 Kg/h after main coolant pump coast. A special construction of the reactor coolant pump seals in West German PWR makes this possible even during a station blackout transient. Therefore the primary side coolant inventory may be sufficient for hours without additional primary feeding, as long as the decay heat removal by secondary-side accident management measures are successful. If the secondary-side AM-measures fail, a switching to primary-side AM-measures is possible to avoid core melting. In case of insufficient primary feed core melting will occur under low pressure conditions. So the above mentioned possible early failure of the containment can be avoided and the probability of high pressure core melt scenarios can be reduced. High pressure scenarios remain when no actions have been taken or the opening of pressurizer valves is impossible.

This paper confines to the discussion of secondary and primary bleed and feed accident management measures to prevent core melt. A further discussion, including accident management measures for mitigating the consequences of a core melt, can be found in [5].

2. SECONDARY BLEED AND FEED

This AM-measure is in principle characterized by a fast depressurization (bleed) of the dry steam generators and a subsequent water injection (feed) at low pressure.

In Fig. 1 the feedwater- and main steam system of the reference plant for the German risk study is shown. The bleed action could be initiated by opening of main steam valves, i.e. relief valves or safety valves. For the feed action there are two possible ways.
After the use of residual warm water in the feedwater line at secondary-side pressures below 2.2 MPa there are two possibilities of secondary feed.

At first the stored water in the feedwater tank can be used. Its volume of 610 m^3 is filled with 320 Mg saturated water. The pressure is 1.0 MPa in the reference plant. In some plants the pressure is lower. For better feed performance it can be increased by loading the feedwater tank through an auxiliary steam line up to the set-point of the tank's safety valves during secondary bleed.

After opening of diverse control- and shut off-valves the feedwater line and tank act as effective accumulators due to the pressure stabilizing effects of its saturation conditions. Water can flow through the feedwater lines via the feedwater pumps and the high pressure preheaters into the steam generators, according to the pressure difference between feedwater tank and steam generator.

The second possible way is the connection of low head pumps (e.g. mobile pumps) to the emergency feed lines. The necessary water can be taken from the demineralized water storage tank, cooling tower ponds, drinking water supplies, tank trucks or directly from the river.

The above mentioned measures need time for the manual operations and some system requirements, e.g. minimum relief capacity of the steam valves in the low pressure region, to be effective. The discussion of two transients will exemplarily demonstrate the effectiveness and feasibility of the secondary bleed and feed measures.

2.1 Total Loss of Feedwater and the use of the Feedwater Tank (in the reference plant of the DRS-B)

The total loss of feedwater transient had been chosen for further investigation because of its dominant contribution to the plant damage state and its fast degradation of primary coolant inventory with an early beginning of core damage in case of any AM-measures. Calculation have been performed for the reference plant in the German risk study with 1300 MW electrical output. Best estimate boundary conditions were chosen for these calculations using the advanced code DRUFAN [6], which is now implemented into the code ATHLET to enable best estimate calculations with it [8]. A detailed model is necessary to get a realistic behaviour of the feedwater system during the secondary bleed and feed measure. In fig. 2 the geodetic proportions of the feedwater system can be seen. The feedwater temperature is 180 °C with a saturation pressure of 1.0 MPa between feedwater tank and HP-preheater. Between the preheater and the steam generators the temperature is 211 °C at a saturation pressure of 2.2 MPa. Check valves control the back flow in the system.

During the secondary bleed and feed measure a pressure profile develops along the feedwater lines. In nearly stagnating flow a phase separation is possible with steam up flow and water down flow, especially in the vertical pipe sections. Therefore the effectiveness could be decreased by large amounts of steam in the injected feedwater. This is considered by special drift correlations in the code.

The important results of the calculation are presented in fig. 3 and 4. The beginning of the transient and time of the total loss of feedwater is 0.0 s. 40 s later the secondary water level decrease initiates the scram of the reactor and the turbine by the reactor protection system. After closure of the steam valves the pressure on the secondary and primary side increases but will be limited by the automatic operation of the secondary relief valves. The pressure on the secondary side will be kept constant at a level of 7.5 MPa. Due to the late reactor scram the remaining secondary side water inventory is only about 52 % of the initial inventory. In this case the reactor coolant pumps will still be in operation and will provide good core cooling. However, the hydraulic energy of these pumps (24 MW) heats the primary fluid and this leads to a faster decrease of the secondary side inventory. Within 20 min after scram the secondary side of the steam generator is completely dry and the energy transfer to the secondary side ceases.

The primary side will heat up and the pressure will be limited by the pressure control with the pressurizer spray system. At approx. 2100 s the pressurizer is full of water, the pressurizer spray nozzle gets flooded and the pressurizer relief valve opens for pressure limitation. Then mainly water is discharged. A further increase of the valve mass flow occurs at 3000 s, when flashing begins in the upper plenum causing a large fluid flow into the pressurizer. Then also the second relief valve opens. During the period of water discharge the loads on the valves and support structures will reach its peak value. For this calculation it has been assumed that the valves were designed for such water flow conditions and that the ECC-criterion at a low RPY-waterlevel fails.

The core uncovery starts at 3400 s, according to the large discharge flow. After initiation of the secondary bleed at 60 min the pressure on the secondary side decreases in a short time to the saturation pressure of the preheated feedwater of 2.2 MPa. Then a first flashing of preheated feedwater and steam begins. The additional steam causes a delay of the further secondary pressure decrease, according to the limited steam valve relief capacity. Despite the flashing of steam the primary pressure is lowered. The pressurizer valves close and the primary coolant loss ends. In this case it is assumed, that the operator only opens the control- and shut-off-valves in the feedwater line, if the pressure before the valves is lower than 1.0 MPa. Therefore the control-valves are opened at 3800 s. The temperature of the feedwater between the control valves and the HP-preheaters is still 211 °C and the pressure is 2.2 MPa. The resultant high pressure difference causes an instantaneous high fluid flow into the steam generator and a pressure peak due to evaporation. The primary pressure is lowered under 11.0 MPa at 4000 s. The combination (ECC-criteria) of high containment pressure and low primary pressure then initiates the automatic shut-off of the reactor coolant pumps and ends the addition of hydraulic energy to the primary coolant. Sufficient core cooling was provided by the running reactor coolant pumps, even when the primary side inventory was partially depleted. After shut-off of the reactor coolant pumps phase separation starts and the water gathers in the downcomer- and core region. There after the core is covered with fluid and is cooled in a "reflux condenser mode".

The delay of the decrease of the secondary pressure by the flashing of feedwater and boil off of the feedwater line section between the control valves and the feedwater pump shut-off valves continues. At 4500 s the secondary pressure falls below 1.0 MPa. At that time the shut-off valves are opened and a constant emptying of the feedwater tank inventory into the steam generators begins, causing a nearly constant depressurization of the primary side. The start of the injection of borated water by the HP-safety injection pumps at 11.0 MPa and by the accumulators at 2.6 MPa is also not considered in this calculation, because of sufficient core cooling even without these systems. At 7000 s the primary pressure is below 1.0 MPa so that the low pressure (LP) injection pumps can refill the primary side with borated water and then cool the primary side in a residual heat removal mode.

The remaining feedwater inventory is 17 % of the initial inventory when the operating range of the LP injection systems is reached. In comparison to this case with a late initiation of secondary bleed and feed an early initiation is more desirable to limit or suppress coolant loss into the containment. According to the then higher primary side inventory, which has to be cooled down to the temperature and pressure to switch over to the residual heat removal system, the necessary water mass to be evaporated in the steam generators is higher. But even in these cases the feedwater tank inventory is sufficient to cool down the primary side to the operating range of LP-safety injection systems.

If reactor control systems will not be activated at a low water level in the RPY or pressurizer the beginning of core uncovery is 400 s earlier. So a sufficient core cooling without safety injection is only possible, if the secondary bleed and feed measures are initiated appr. 10 minutes earlier or the reactor coolant pumps are shut off by the operator at least 30 minutes after the beginning of the transient. The transient level behaviour in fig. 4 shows clearly the important influence of the addition of hydraulic energy by the running reactor coolant pumps. If the secondary bleed and feed measure is initiated after 60 minutes instead of 50 minutes (reactor control system deactivated) safety injection pumps are needed to refill the core region for sufficient core cooling. The safety injection is automatically activated by the reactor protection system.

The calculations show the effectiveness of the secondary bleed and feed measure by using the residual feedwater to control the transient when the bleed begins with a maximum time delay of appr. 1h. To demonstrate the feasibility of this AM-measure an estimation of the necessary operator actions and time has been done [7]. The steam generators boil off during the first 20 minutes of the transient. The operators will be informed about that beyond design situation by the secondary side water level instrumentation. The first criterion for AM-measures is reached 10 min after the begin of the transient at a secondary water level of < 2.00 m. The reactor protection system has already blocked or will block the necessary steam- and feedwater valves. These blockage signals have to be bridged. Key-switches will be installed in the control room for the resets in order to gain time and to higher the reliability of the AM-measures. Also the feedwater control- and shut-off-valves can be opened by the operators in the control room. The opening of these valves is possible because check valves in the feedwater system prevent undesired pressurization of feedwater system sections. The pressure- and flow peaks as can be seen in the calculations (fig.3), would then be avoidable. The opening of the secondary steam valves to initiate the steam generator bleed is also possible in the control room. The time needed for these actions is about 30 min. The available time between the initiation of the secondary bleed and feed and the begin of core uncovery is > 50 min, if the reactor coolant pumps are shut off as part of the accident management measure. As first manual action it is planned to shut off these pumps right after initiation of the first AM-criteria. If doing so, the available time is prolonged to 60 min. So, as long as electric power is available, the use of the feedwater tank seems to be feasible even in relatively fast transients.

2.2 Station Blackout Transient and the Use of Mobile Pumps

The station black-out transient has been selected for further investigation because of the limited possibilities for AM-measures due to the failure of the electric power supplies. The effectiveness and feasibility of the use of mobile pumps as the alternative to the use of the feedwater tank should also be demonstrated.

The transient behaviour of the pressure, water-level and the discharge mass flow is shown in fig. 5. The best estimate boundary conditions are the same as in the calculation of the total loss of feedwater transient.

The loss of the AC-electric power supply causes an early trip of the reactor coolant pumps and subsequently the scram of the reactor compared to the total loss of feedwater transient. Therefore, the dry out time of the steam generators due to the loss of emergency feedwater by the failure of the emergency diesels is much longer, i.e. approx. 1h.

At that time the primary pressure increase is limited by opening of pressurizer relief valves and the loss of primary coolant begins. At 1,7 h the primary fluid is saturated. This causes a rapid increase of the valve discharge mass flow, which consists mainly of water, and a rapid decrease of the collapsed level in the reactor pressure vessel. At 2.0 h the core uncovery starts. At this time a secondary bleed measure by opening of secondary steam valves is initiated, similar to the bleed action in the total loss of feedwater transient. After depressurization of the steam generators low pressure head mobile pumps with their own power supply begin an injection of approx. 30 kg/s cold water into the steam generators. The primary pressure is lowered, the pressurizer valves close and the primary coolant loss ends exept the low leakages through the reactor coolant pump seals. The core is sufficiently covered with two-phase-mixture and is cooled in a "reflux condenser"-mode.

Connections for mobile pumps are foreseen. The use of mobile pumps is easier in a station blackout situations than the use of the feedwater tank inventory.

The fast depressurization and feed with cold water may cause thermal shock and mechanical stresses in the steam generators. In order to minimize thermal chocks secondary feed shall start with warm water out of the feedwater-line followed by using mobile pumps. The discussed calculations are momentarily used to quantify these stresses with appropriate finite elemente codes.

3. PRIMARY BLEED AND FEED

Primary bleed and feed measures are operator actions which consist of opening of pressurizer valves (bleed) to release the residual heat into the containment and feeding of the primary side with safety injection pumps. These pumps start the injection of borated water below 11 MPa.

To reach a long term stable state after depletion of the borated water storage tanks further actions are necessary. In new plants operation of the low pressure injection system (LPIS) as booster pumps makes a high pressure injection of sump water possible. In plants without this backup capability, a depressurization of the primary side below 1.0 MPa is necessary. This can be achieved by an enhancement of thermal mixing to remove hot spots in steam generators (operation of main coolant pumps necessary) or by an opening of additional pressurizer valves. At low pressure the LPIS can remove the decay heat in the recirculation mode and simultaneously inject coolant from the sump to make up for the loss of coolant. If the pressurizer valves can be closed after refilling of the primary side then the decay heat can be removed by the LPIS in the recirculation mode already from 3 MPa provided the coolant temperature is below 180 °C.

Assuming the safety injection pumps to fail, core melt probably can't be prevented though opening of all pressurizer valves will achieve fast depressurization below 2.6 MPa and the inventory of the accumulators is used for feeding.

According to the strategy of early secondary and late primary bleed and feed measures an appropriate instrumentation for the initiation of the primary measures is needed. The time point of the derived initiation signal should be late enough to enable the performance of secondary measures but early enough to prevent core melting taking into account a sufficient response time for the manual initiation. In all PWR in the FRG a level probes in the upper plenum of the reactor pressure vessel are or will be installed which measures the collapsed level (fig.7). The decrease of the water level below the lower edge of the coolant loops is the second AM-criterion. There after the operators shall perform primary AM-measures. A third and last AM-criterion will be reached at a fluid-temperature > 400 °C at the core exit. The third AM-criterion has been introduced as a backup to the RPV-water-levelcriterion. Some calculations have been done with the best estimate code DRUFAN [6] to determine the effectiveness of primary bleed and feed with regard to different time points of initiation, using safety injection pumps or accumulators [7].

3.1 Total Loss of Feedwater with primary bleed and feed with safety injection pumps

For the case of total loss of feedwater and 2 safety injection pumps available, the primary pressure and collapsed level history before and after the initiation of the depressurization with 2 of 4 pressurizer valves (cross sectional area 60 cm^2) is shown in fig.6. The depressurization has been performed, when the collapsed level in the upper plenum reaches the lower edge of the coolant loops 60 min after the loss of feedwater. The level probe in the upper plenum gives then a signal not only to the operators for initiation of primary depressurization, but also in combination with high containment pressure a shut off signal to the reactor coolant pumps. The pump coast down leads to separation of steam and water in the reactor pressure vessel. So the water in the upper plenum flows into the core region, where the collapsed level momentarily rises. This effect cannot hinder the depletion of the core region and the core temperatures begin to rise. About 10 minutes after the opening of the pressurizer valves the injection of the safety injection pumps is possible, when the primary pressure falls below 11 MPa. The core is flooded within about 10 min. The rise of core temperatures stops. The peak temperature level in the cladding is 600 °C.

A second calculation has been done to determine the maximum response time for the operators. The assumptions are the same but the opening of the pressurizer valves is delayed for 15 min. The temperature rise in the core is higher but it does not endanger the core integrity (peak cladding temperature ca. 1000 °C). So one can say that core melting in the case of the fast running transient "total loss of feedwater" could be prevented, if 2 of 4 safety injection pumps are available and 2 of 4 pressurizer valves are opened after the decrease of the collapsed level below the lower edge of the coolant loops. A response time of the operator action up to 15 min is acceptable. It is foreseen to enable the opening of the pressurizer valves from the control room to shorten the necessary time for manual actions. Now after determination of the maximum delay time for primary measures one can see that there is time enough between the first AM-criterion (10 min after begin of the transient) and the signal for primary measures (60 min after begin of the transient) of 50 min in order to perform the secondary measures without the need for primary measures. But the time is to short to avoid also the contamination of the containment by secondary measures.

3.2 Total Loss of Feedwater with primary bleed and feed with the Accumulators only

For the case of total loss of feedwater and an assumed unavailability of all safety injection pumps, the primary pressure history before and after the initiation of the depressurization is shown in fig. 8. The depressurization by opening of one pressurizer valve with a cross sectional area of 40 cm^2 has been performed just before core uncovery at 50 min. 25 min later the primary pressure sinks below 11 MPa. By recognizing the unavailability of all safety injection pumps the operators open additional pressurizer valves to accelerate the depressurization below the pressure in the accumulators of 2.6 MPa. Below this pressure the accumulators begin to inject cold borated water (temperature 30 °C) into the hot and cold legs. If one additional pressurizer valve with a cross sectional area of 40 cm^2 is opened, the refilling of the lower plenum and the core region is to slow to prevent the steady heat up of the core up to temperatures where Zirconium-water reactions begin (fig. 9, 10). Therefore a second calculation has been performed with the assumption of opening all pressurizer valves to enlarge the pressure release capacity. Now the lower plenum and the core are refilled (fig. 9). The core is quenched and the cladding temperatures drop to saturation temperature (fig. 10). The evaporation of primary fluid and of injected cold water after heat up by cooling of structures, compensation of decay heat and depressurization exceeds the condensation capability of the colu injected water and the steam release through the pressurizer valves. The resulting primary pressure rise stops the accumulator injection for about 35 min. The cladding temperatures rise up to 1200 °C.

The results of the calculations with an best estimate code indicate difficulties in using the accumulator inventory alone to delay core damage for about 4 h, as calculations with simpler, homogeneous codes like MARCH have shown [5]. The realistic delay times could be shorter. A detailed discussion of the phenomenological aspects and model influences connected with the calculation of the effectiveness of accumulator injection with a best estimate code is given in [8].

4. PROJECTION OF ANALYTICAL RESULTS ON OTHER TRANSIENTS

Besides the previously discussed transients "Total loss of feedwater" and "Station Blackout" there are some other sequences leading to a similar core damage frequency [7]. Fig. 11 shows the frequency distribution which has been determined in the Phase B of the German risk study for the reference plant when no accident management-measures are considered. Uncontrolled pressurizer leaks, station blackout, loss of feedwater, loss of main heat sink, steamline break and U-tube rupture are sequences for further investigation of accident management measures, regarding probabilistic aspects. All these sequences and uncontrolled small breaks at the coolant loops having a break area below approximately 50 cm^2 will lead to core melting at high primary pressure.

In general the same strategy as previously discussed can be applied. If additional failures at the secondary side cause a decrease of the feedwater inventory, secondary bleed and feed is initiated to reestablish a sufficient secondary heat sink. In case of breaks and leaks at the primary side primary injection for compensation of the loss of primary coolant is necessary to prevent core melting. Otherwise the secondary feed and bleed alone is sufficient. If the secondary measures fail, i.e. the collapsed level in the reactor pressure vessel continues to decrease, primary bleed and feed is necessary to enable safety injection pumps to inject borated water.

In fig. 11 estimated delay times for the initiation of the accident management measures for the various sequences are shown. In case of the more likely events the delay times are greater than 1 h and hence comparable to the previously discussed delay times. In the case of the uncontrolled rupture of one U-tube the delay time is about 1 h. In the case of two ruptured U-tubes the delay time is only about 30 min. But this event is about one order of magnitude lower in its frequency.

5. SECONDARY AND PRIMARY FEED AND BLEED IN CONVOY-PLANTS

In comparison to the reference plant of the DRS-B, Biblis B, the accident sequence in case of the total loss of feedwater supply with electrical power available is mainly influenced by the reactor- and turbine power-limiting-system (RELEB).

The reactor- and turbine power ist adjusted on the feedwater flow. Due to a decreasing feedwater flow the reactor- and turbine power are reduced and the void at steamgenerator (SG) secondary side risers drops. When the reactor- and turbine shut down is activated at low secondary side water levels the water mass at SG secondary sides hasn't changed very much from the initial inventory.

At reactor- and turbine shut down the SG secondary water inventory in Convoy plants is about twice as high as in Biblis B. Therefore simalar plant conditions are reached in Convoy plants twice the time later.

The first criterion for AM-measures at SG secondary side water levels < 4.00 m (!) is reached 15 minutes after the main feedwater supply has failed (s. fig. 12). Before the plant personel should start in preparing feed and bleed measures the main coolant pumps shall be switched off. The SG secondary sides are empty 30 minutes later. The pressurizer relief valve opens the first time another 10 minutes later (s. fig. 13). It can be assumed, that the secondary feed and bleed can be prepared, initiated and effective within the 40 minutes between the actuation of the first AM-criterion and the first opening of the pressurizer relief valve. Therefore not only core melt but also a contamination of the containment will be prevented.

If secondary side feed and bleed is delayed, the second AM-criterion at a RPV-water level below the lower edge of the main coolant line is reached 113 minutes after the failure in main feedwater supply. Only 12 minutes later the fluid temperature at the core exit exceeds 400 °C, if no AM-measures have become effective in advance. In case of a station blackout the available times are simular in Convoy-plants and the Biblis B plant.

A measure to gain more time for e.g. delayed secondary side AM-measures or the repair of failed components is the feeding of the primary side with the make-up and the additional borating system, which is in discussion now (s. fig. 14).

As a result of the Convoy-possibilities the reference plant of the DRS-B will get also a reactor- and turbine-power-limiting system as well as an additional borating system with an injection rate of 22 kg/s.

6. SUMMARY

The phase B of the German risk study clearly has shown the general need to depressurize the reactor coolant system in case of uncontrolled events to minimize the frequency of core melt scenarios at high pressure. In case of total loss of feedwater supply an early initiation of secondary bleed and feed is preferred to avoid loss of primary coolant into the containment. The initiation of primary bleed and feed should be performed as late as possible.

The secondary water level instrumentation, the primary level probe in the upper plenum and the fluid-temperature measurement at the core exit are used for AM-citeria. So time criteria in the emergency procedures will be avoided.

The analysis of secondary bleed and feed measures demonstrates their capability of preventing core melt even in relatively fast transients, e.g. the total loss of feedwater transient. The feasibility of this measure has been shown by the us of the feedwater tank inventory. In case of the station blackout transient there is an additional time of 1 h for operator actions, but the electric power supply is restricted. But the use of residual water in the feedwater line as a passive feeding system and the use of mobile pumps with their own power supply is effective to prevent core melt and will overcome the main restrictions by the loss of the electric power supply.

The analysis of primary bleed and feed measures demonstrates the effectiveness of this measure to prevent core melt in the case of a total loss of feedwater, even if the system availability is restricted to 2 of 4 pressurizer valves and 2 of 4 safety injection pumps and the initiation is delayed up to the begin of core uncovery. Calculations with a best estimate code (DRUFAN) indicate that accumulator injection only is not as much effective to delay core damage as simpler codes e.g. MARCH have shown assuming the injection of the whole accumulator inventory.

The projection of the analytical results on other risk relevant sequences shows that the overall risk by core melt scenarios at high pressure may be reduced for a broad spectrum of uncontrolled events.

An estimation of delay times for bleed and feed measures in case of the most probable core melt sequences give delay times of 1 h or longer. So the available time budget is sufficient to perform the actions with high probability.

Recently performed ATHLET-simulations of a total loss of feedwater supply with electrical power available in Convoy-plants show that the minimum time for AM-measures is about 2 h. This gain in time is caused by a reactor- and turbine-power-limiting-system. The reference plant of the DRS-B will therefore introduce the same limiting system. A further time gain can be achieved by primary feed with the make-ups-and the additional borating system.

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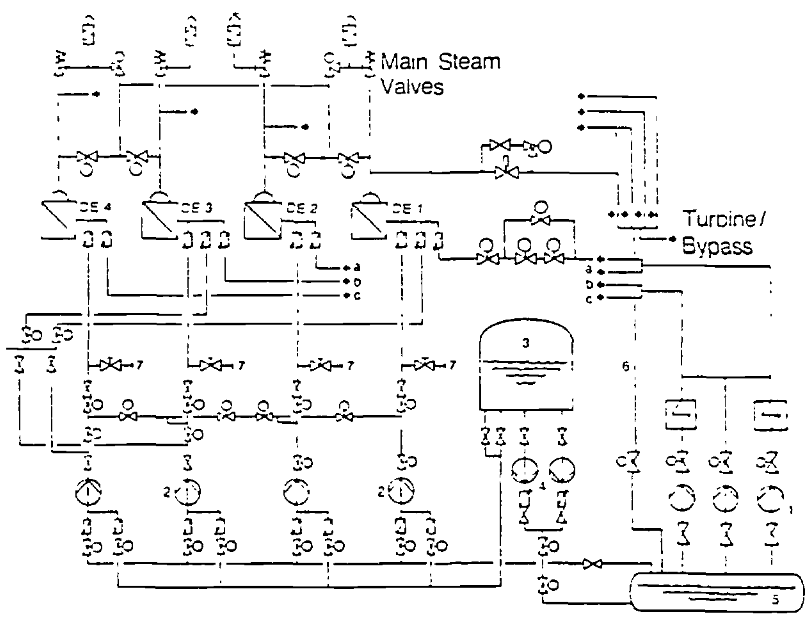
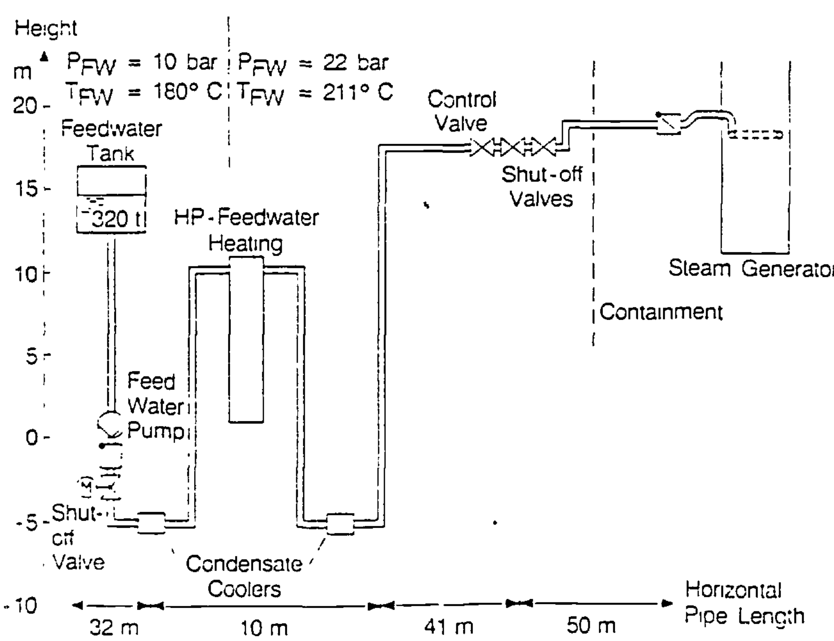
[7] Deutsche Risikostudie Kernkraftwerke Phase B, Gesellschaft für Reaktorsicherheit (GRS) mbH, Verlag TÜV Rheinland 1990

[8] W. Pointner, V. Teschendorf, K. Trambauer, Experience with the ATHLET-Code and its impact on further development, Proc. CSNI-Specialist Meeting on intentional coolant system depressurization, GRS-Garching, June 1989



1 Feedwater Pumps
2 Emergency Feedwater Pumps
3 Demineralized Water Storage Tank
4 Feed Pumps
5 Feedwater Tank
6 Auxiliary Steam Line
7 Connection for Mobile Pumps

Feedwater- and Main Steam-System / PWR Fig. 1



Feedwater-System Fig. 2

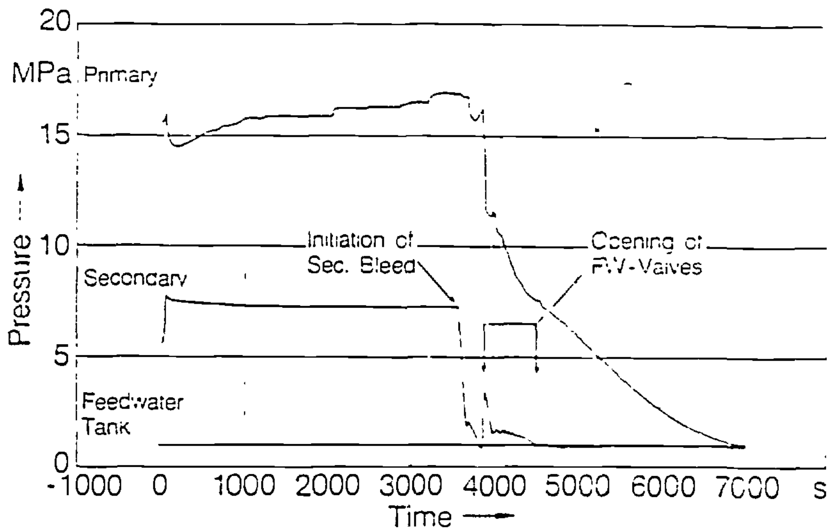


Fig. 3

Primary and Secondary Pressure
Total Loss of Feedwater with Secondary
Bleed & Feed

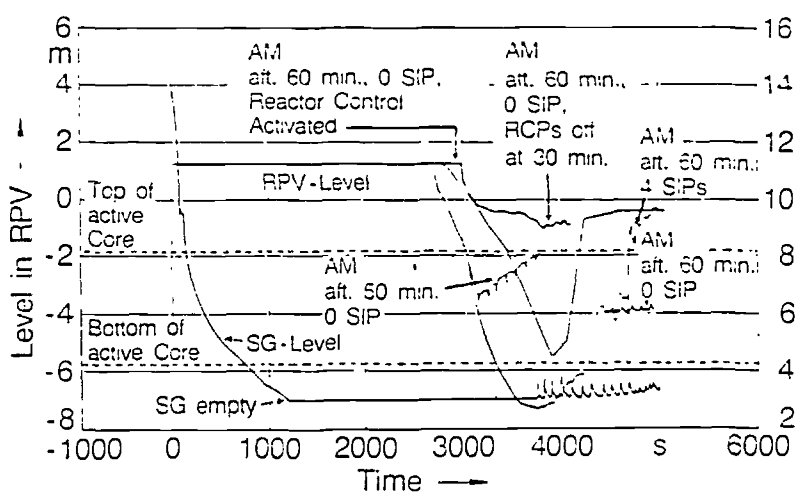


Fig. 4

Collapsed Level in RPV and Steam Generator
Total Loss of Feedwater with Secondary
Bleed & Feed

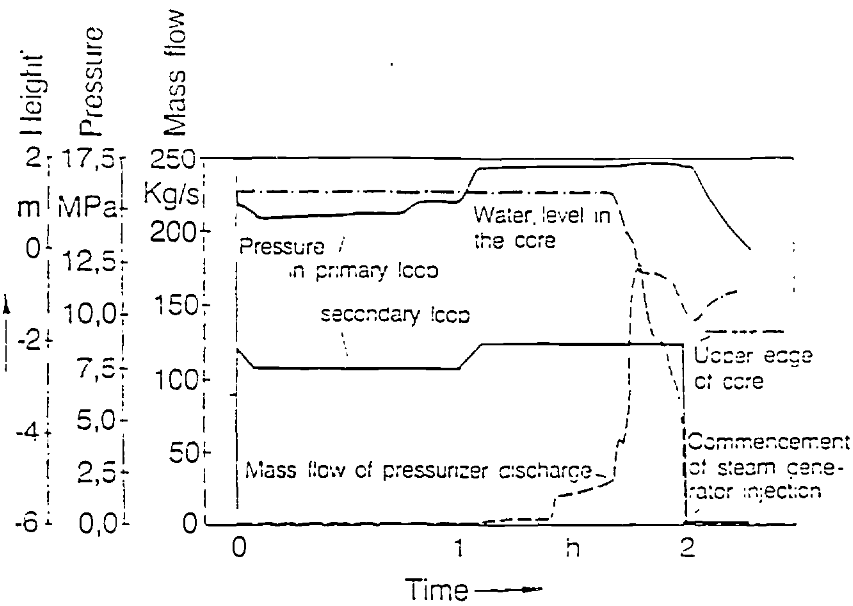


Fig. 5

Emergency Injection to the 4 Steam Generators with Mobile Pumps after 2 Hours in the Case of Station Blackout

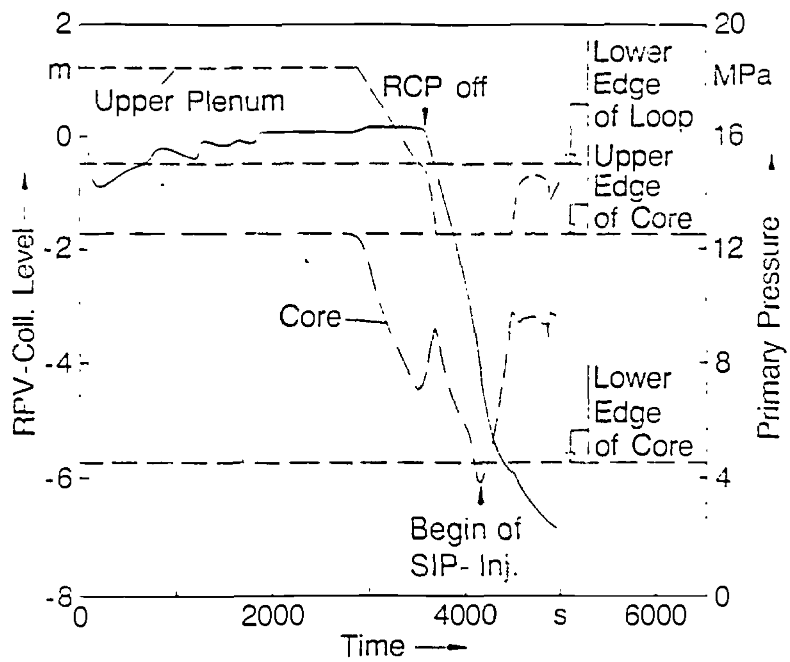
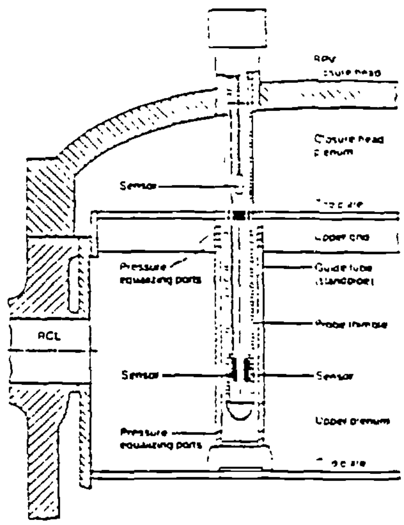
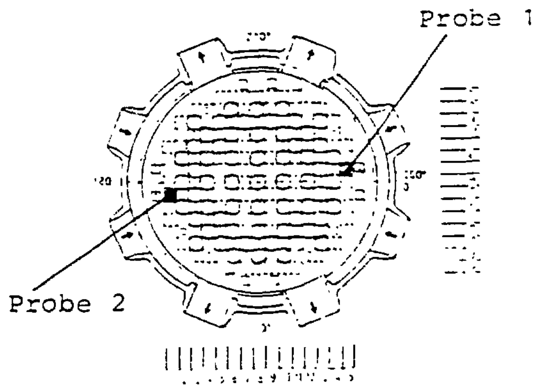


Fig. 6

RPV-Coll. Level and Primary Pressure Total Loss of Feedwater with Primary Bleed and Feed (60 cm^2/2 SIP)



Schematic of guide tube, probe thimble
and sensor arrangements



Location of level probes in upper plenum

Fig. 7

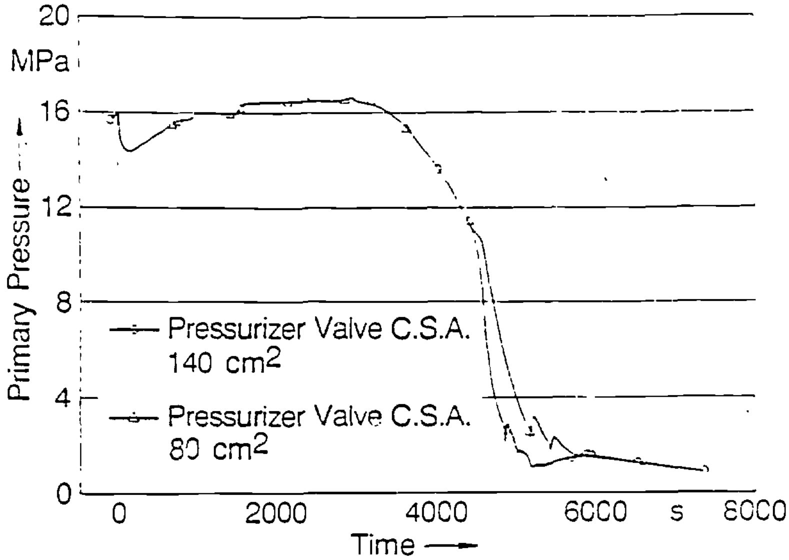


Fig. 8

Primary Pressure
Total Loss of Feedwater with Primary Bleed and Feed (4 Accumulators)

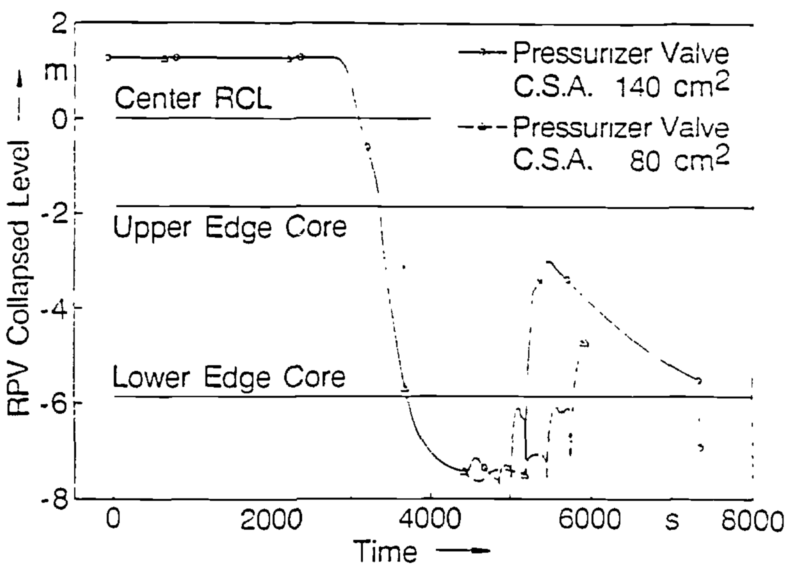


Fig. 9

RPV Collapsed Level. Total Loss of Feedwater with Primary Bleed and Feed (4 Accumulators)

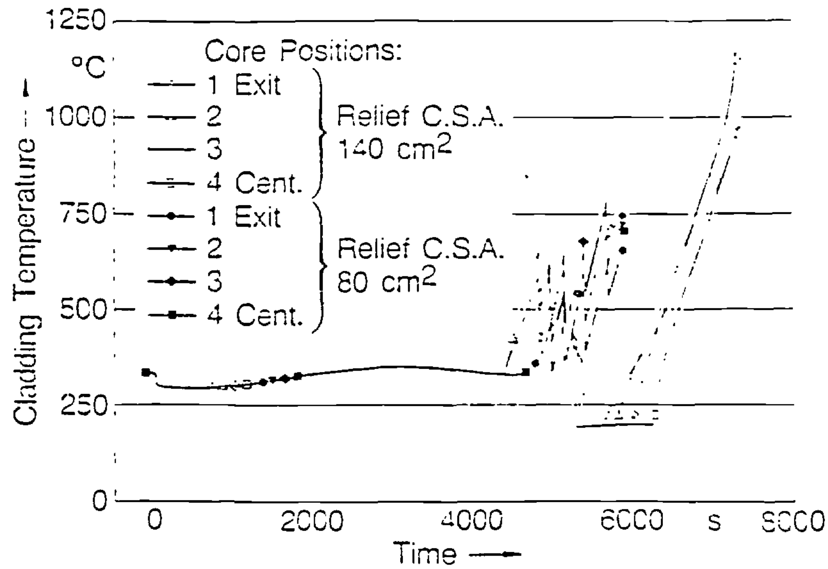
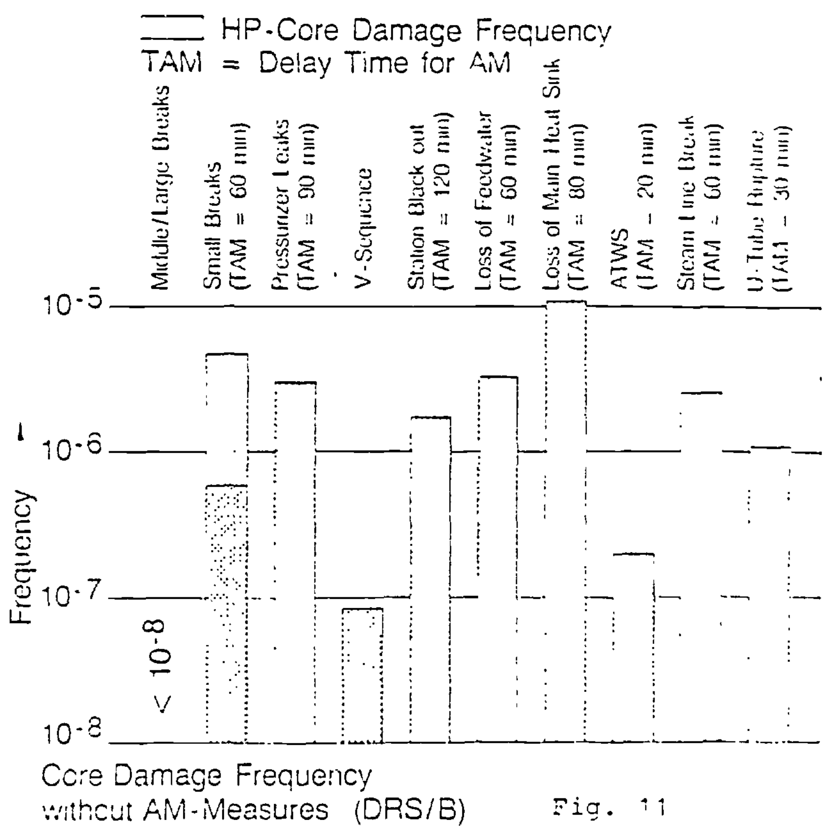


Fig. 10

Cladding Temperature
Total Loss of Feedwater with Primary Bleed and Feed (4 Accumulators)



Core Damage Frequency without AM-Measures (DRS/B)

Fig. 11

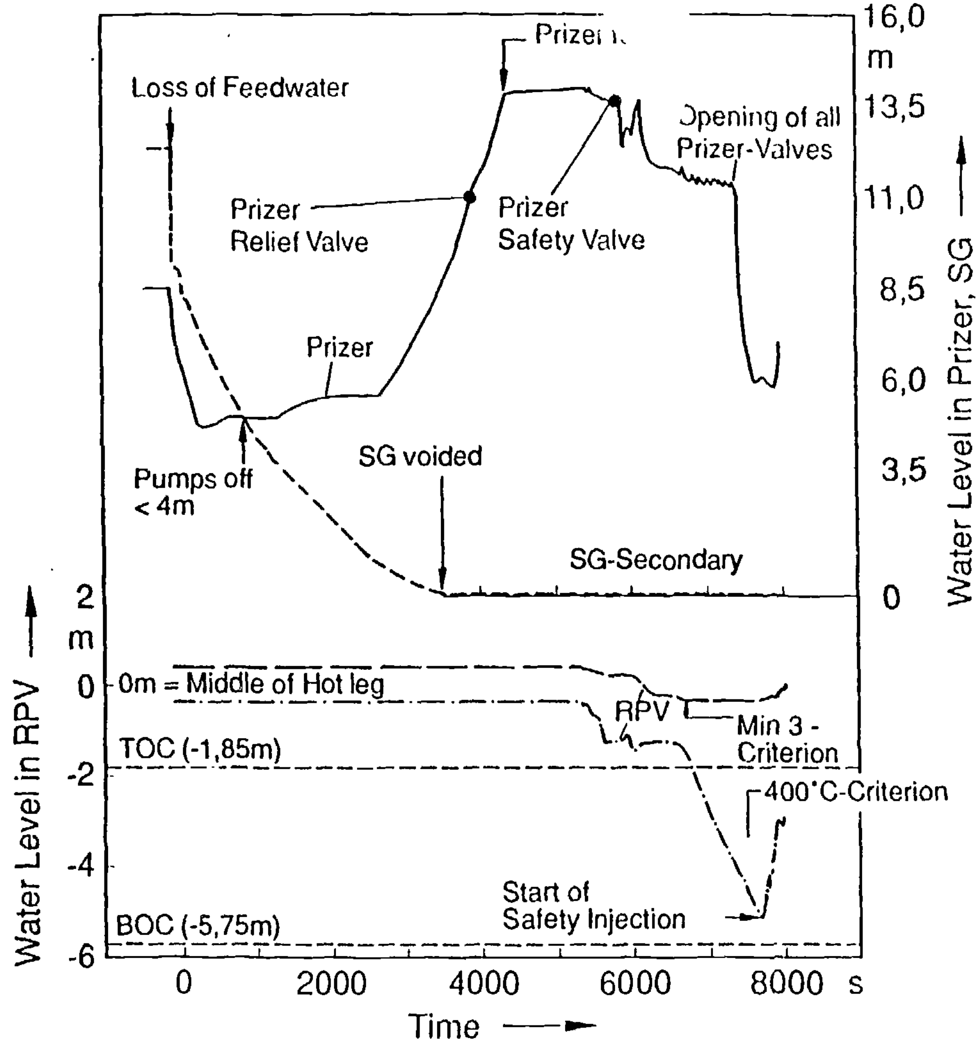


Fig. 12: Coll. Water Level in RPV, Pressurizer, SG
Loss of Heat Sink with Loss of Feedwater (Konvoi)

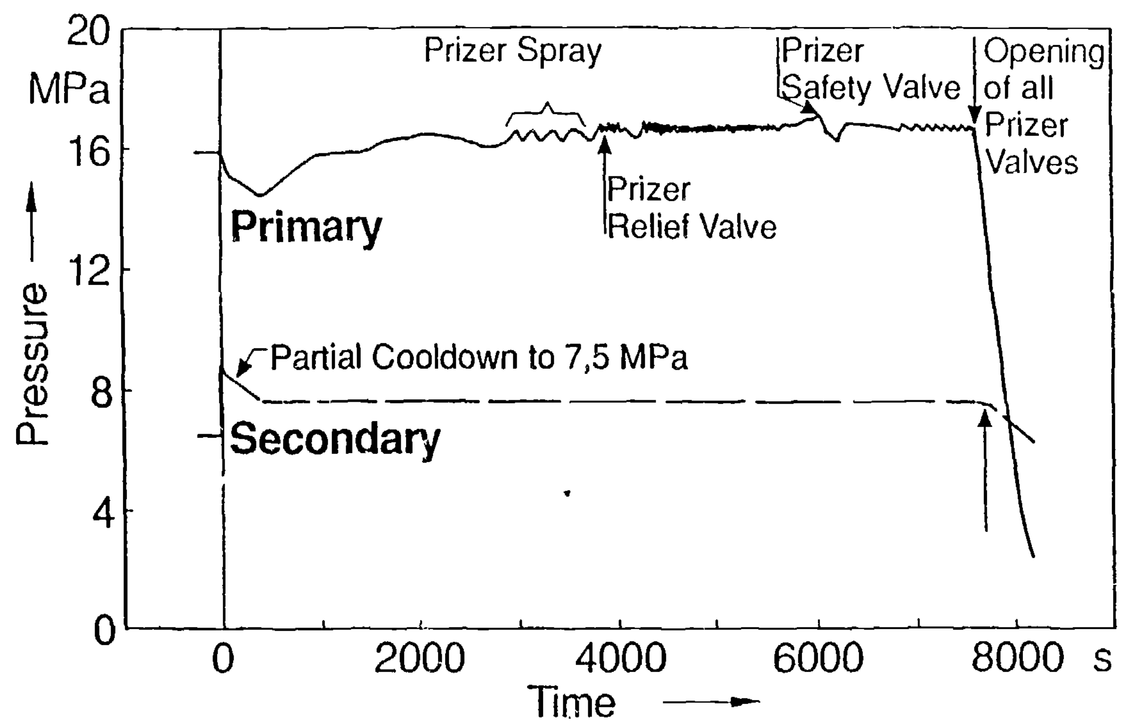


Fig. 13: Pressure in Primary and Secondary Circuit
Loss of Heat Sink with Loss of Feedwater (Konvoi)

- 136 -

KfK 93201 c2

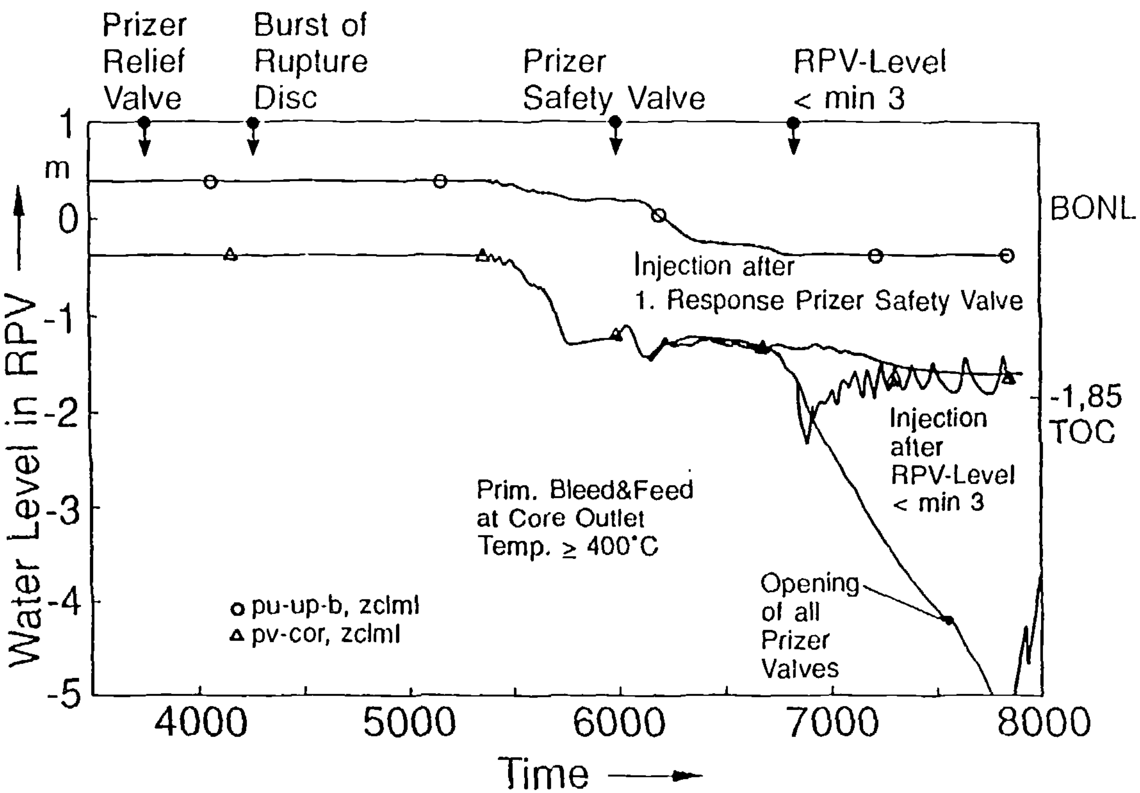


Fig. 14: Loss of Heat Sink with Loss of Feedwater (Konvoi)
Injection with Vol.-Control and Extra
Borating System (21 Kg/s)

Kee 90201-03

International Atomic Energy Agency

Technical Committee Meeting on :

Plant System Utilization for Accident Mitigation

Containment Venting for BWR and PWR

J. Rohde, M. Tiltmann - GRS mbH

26. - 30. November 1990, Garching/Germany

roh003

PLOT : 13.19.45 FRI 8 SEP. 1989 JOB=TILSEN: , GRS MUC-CGN/Germany DISSPLA 10.0

REQUIREMENTS FOR CONTAINMENT - VENTING

PRESSURE WATER REAKTOR (PWR)

a) Layout and Operating

- To open at about the testpressure of the containment.

    * Pressure limitation by venting without water injection into the containment.

    * Pressure reduction to half of the test pressure within two days with water injection into containment.

- Layout of the valves for redropping also at the test pressure of the containment.

- Layout of the valves for stepwise opening and dropping.

- Activation of water injection possibilities into the containment for compensation of the vented water mass at point of time for venting.

b) Loads to be considered.

- Up to the outer - respectively the second - isolating valve: failure pressure of the containment; alternatively: double design pressure.

- For the connected system:

  * Pressure, temperature and composition of the discharging mixture according to the accident conditions at fully opened valve.

  * Layout margin for pipes and supporting systems for taking into account dynamic loads, alternatively: safety factor 2 at operating loads.

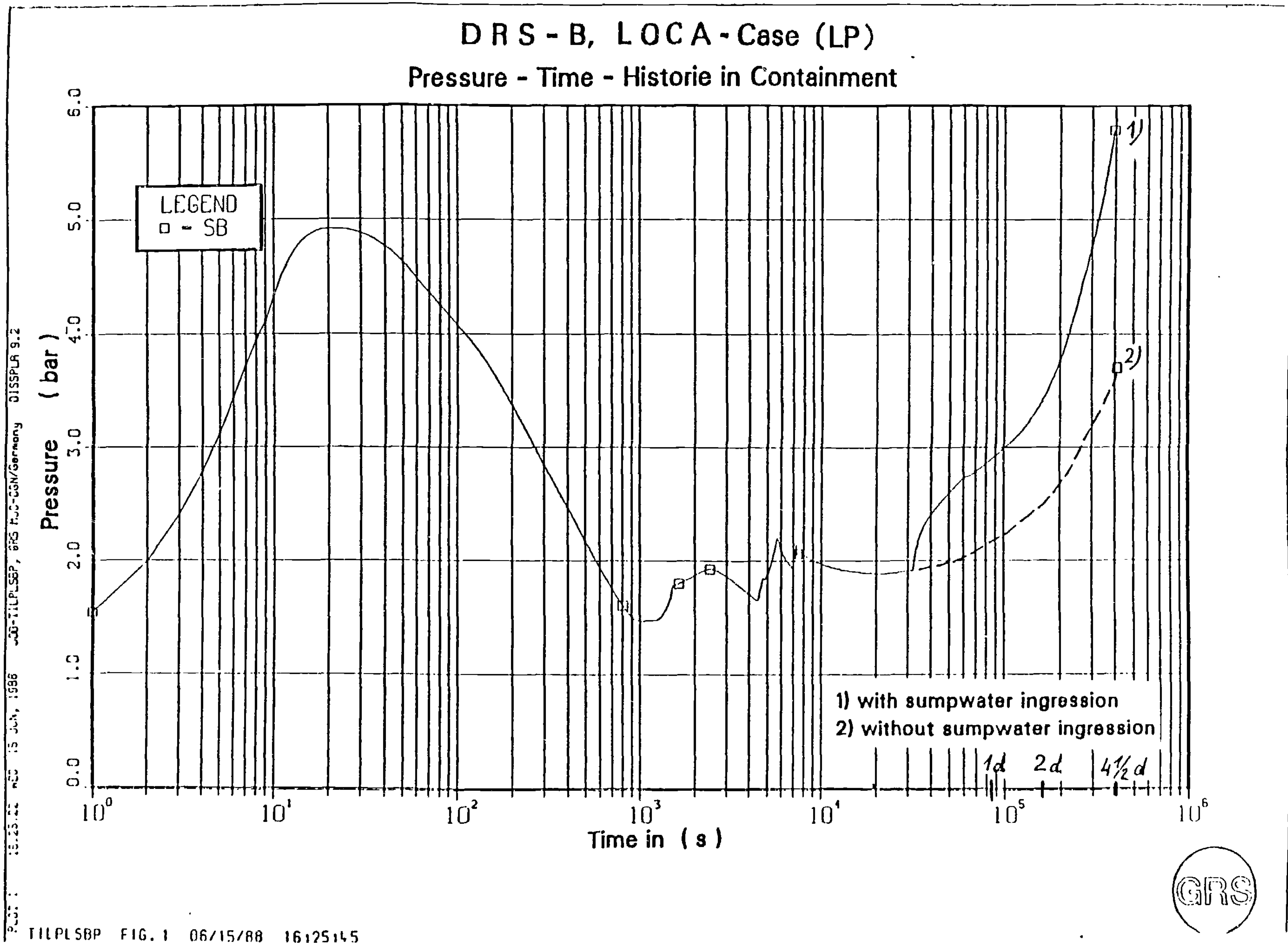
c) Structural data

- Fixed construction of the systems behind isolating valves.

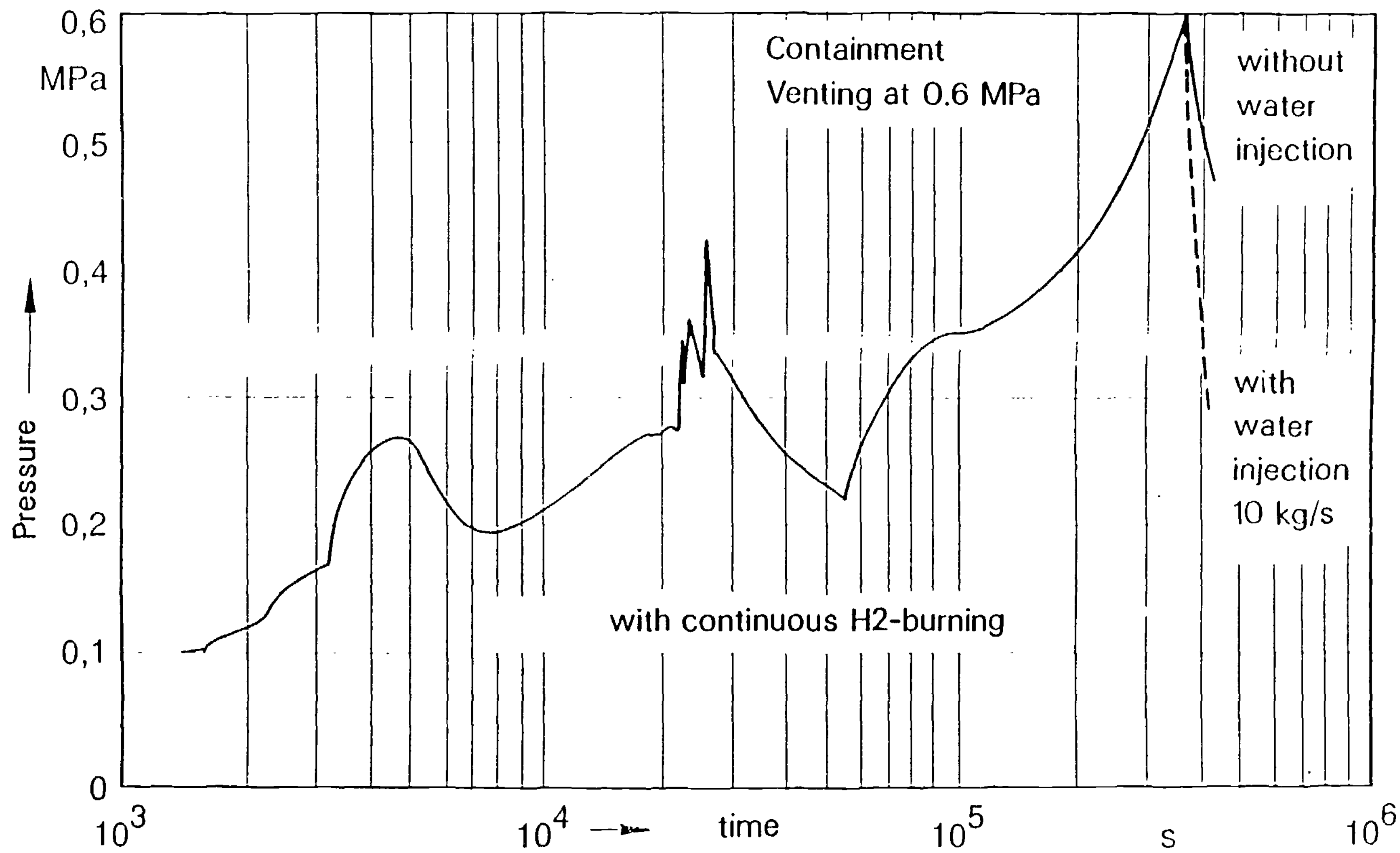
- Remote operation and additionally power supply possibility.

- Removal of the condensate from the vent pipe.

- Provision of a filter system.



D R S - B, L O C A - Case (LP)
Pressure - Time - Historie in Containment
LEGEND
□ - SB
Pressure (bar)
Time in (s)
1) with sumpwater ingression
2) without sumpwater ingression
1d
2d
4½ d
GRS
TILPLSBP FIG. 1 06/15/88 16:25:45



Pressure course within containment

CONCEPTIONS OF CONTAINMENT VENTING

- Sliding pressure with throttling orifice.

Dry filtration

* aerosol filter inside containment
iodine filter outside containment (two units).

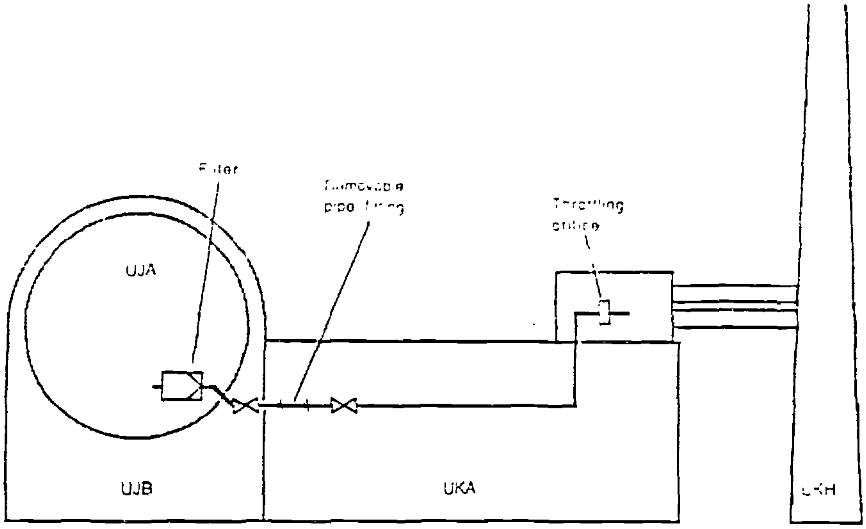
Wet filtration.

* aerosol filter and iodine filter outside containment
(one unit = venturi scrubber).

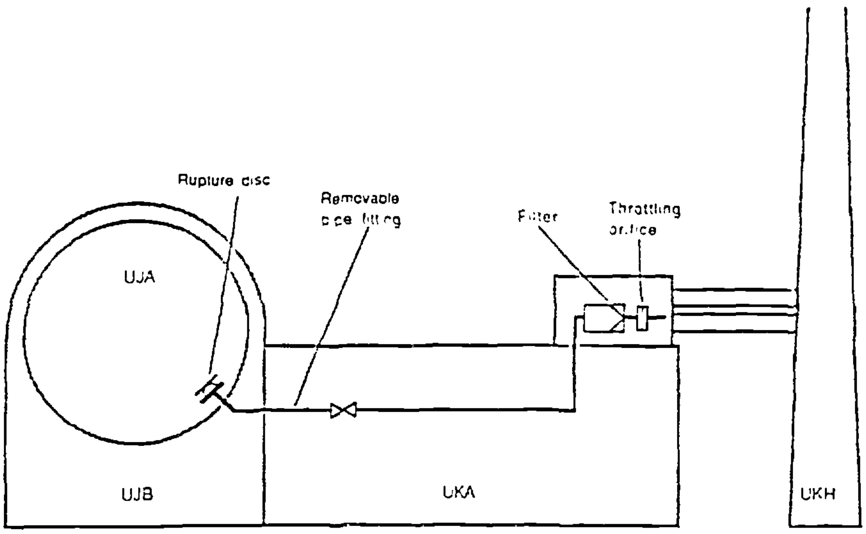
- Atmospheric filter with throttling orifice or control valve.

Dry filtration

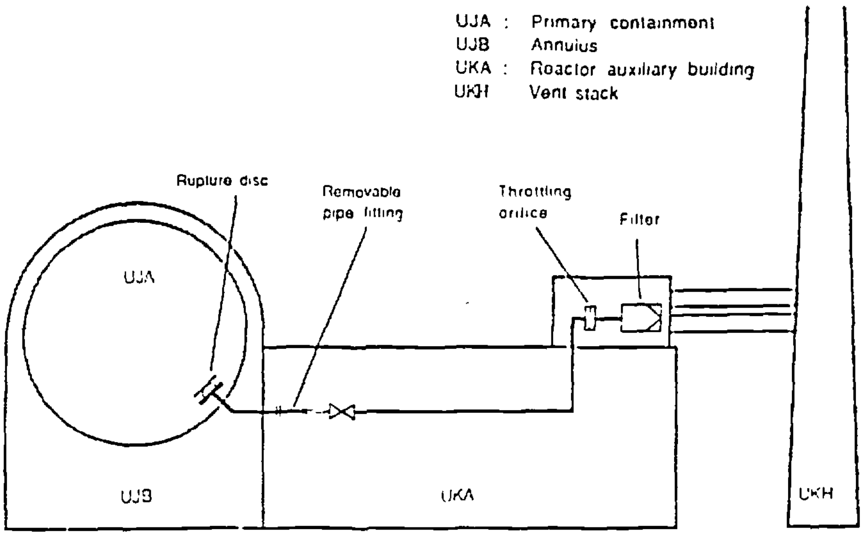
* aerosol filter and iodine filter outside containment
(Zeolite filter).



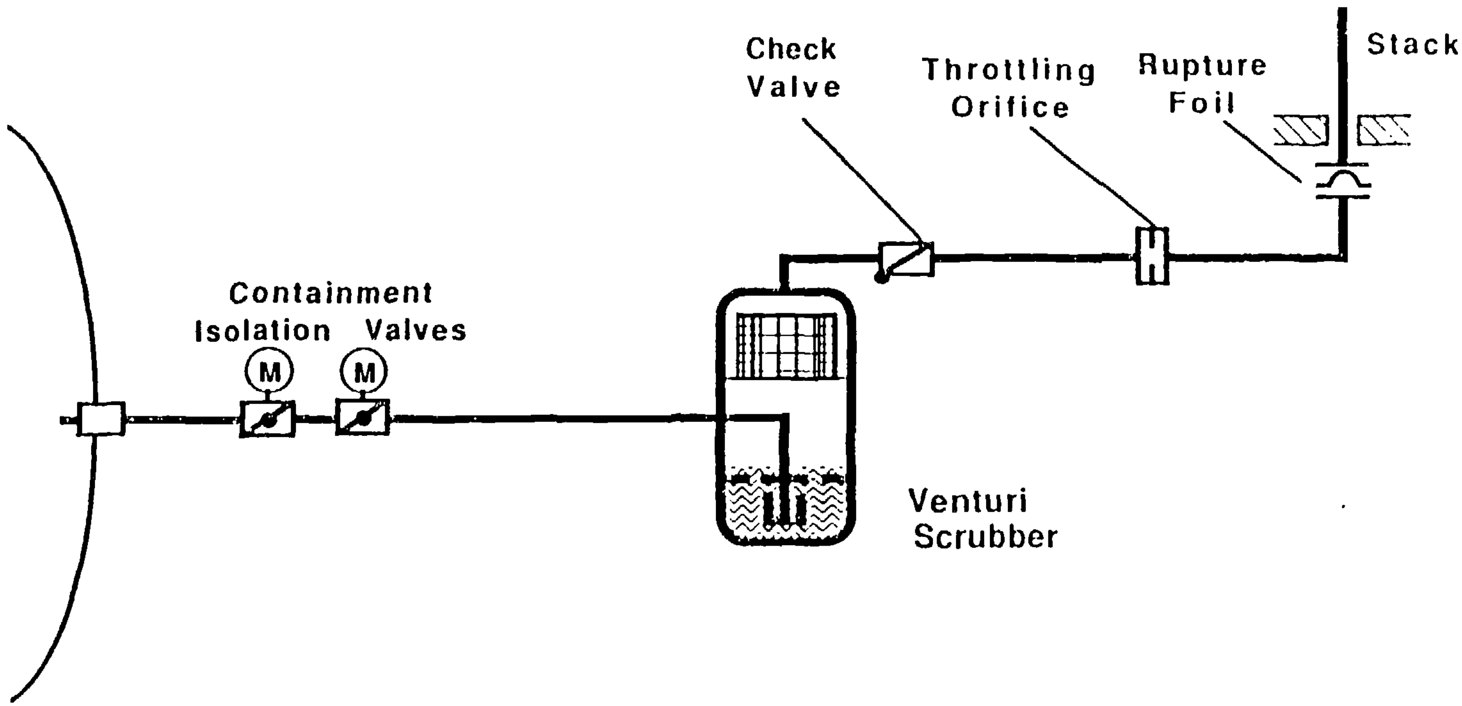
PWR 1300 MW CONTAINMENT VENTING
FLOW DIAGRAM AND ARRANGEMENT . SLIDING PRESSURE FILTER INSIDE CONTAINMENT



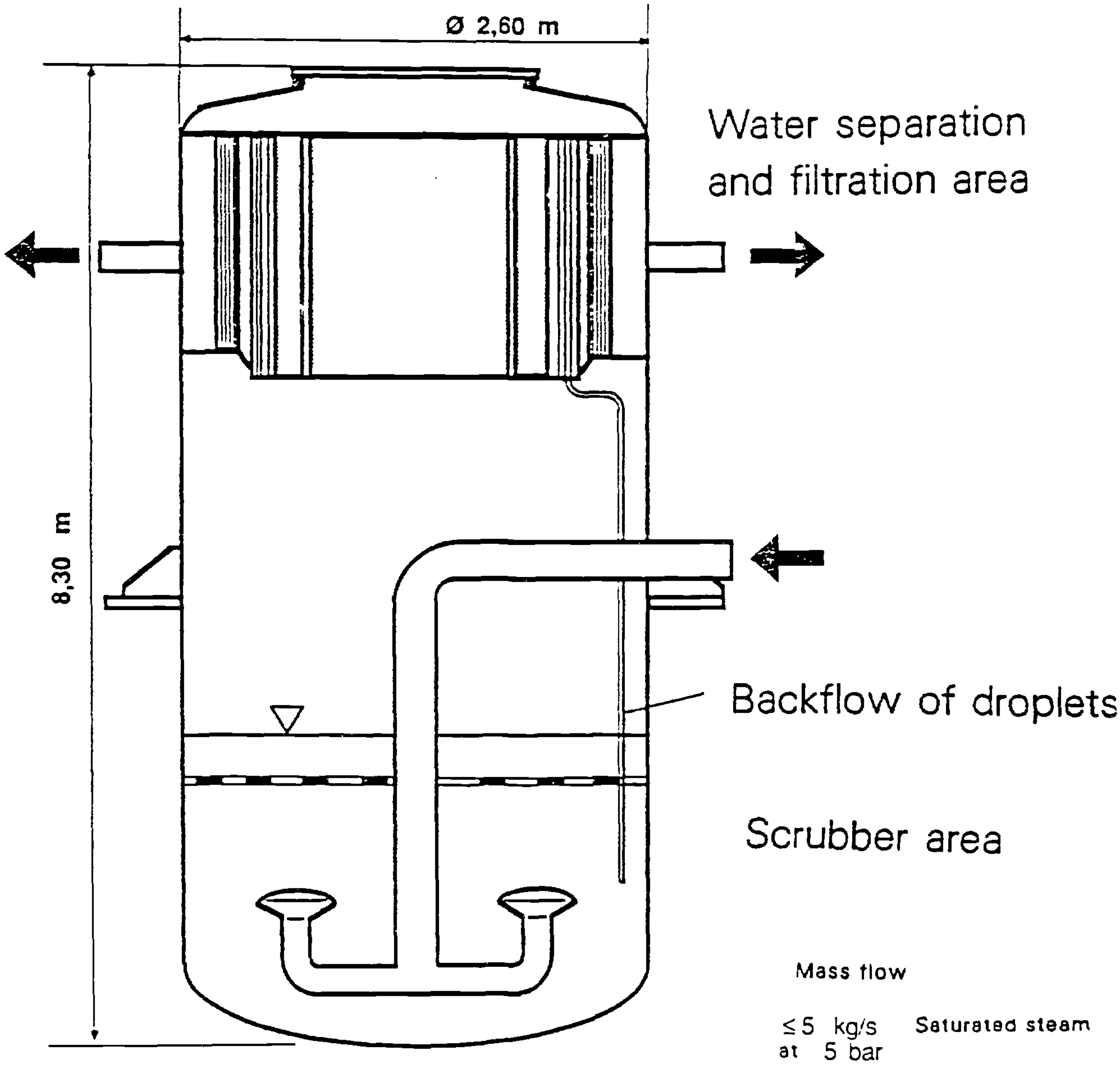
PWR 1300 MW CONTAINMENT VENTING
FLOW DIAGRAM AND ARRANGEMENT : SLIDING PRESSURE FILTER OUTSIDE CONTAINMENT



PWR 1300 MW CONTAINMENT VENTING
FLOW DIAGRAM AND ARRANGEMENT . ATMOSPHERIC FILTER (with throttling orifice)

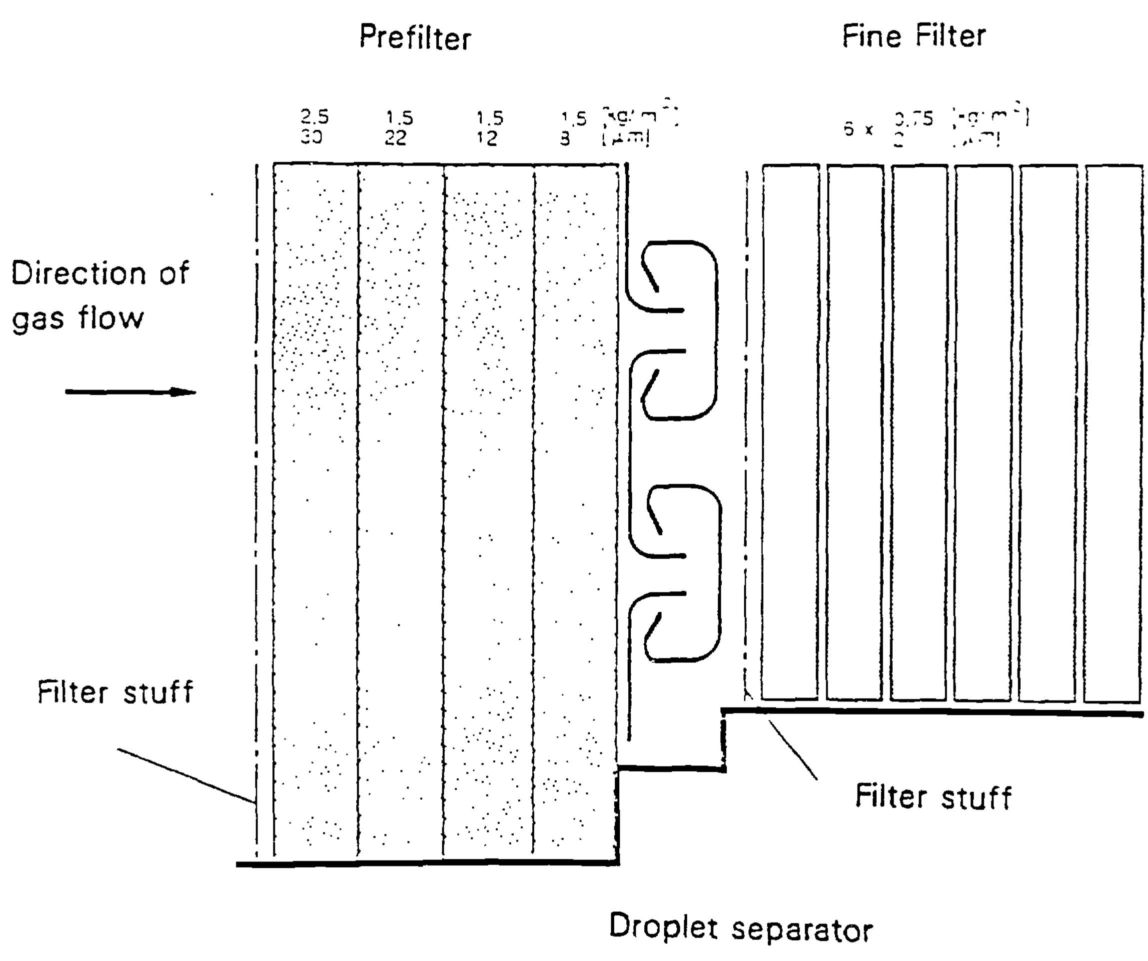


Containment Venting
Flow Diagram



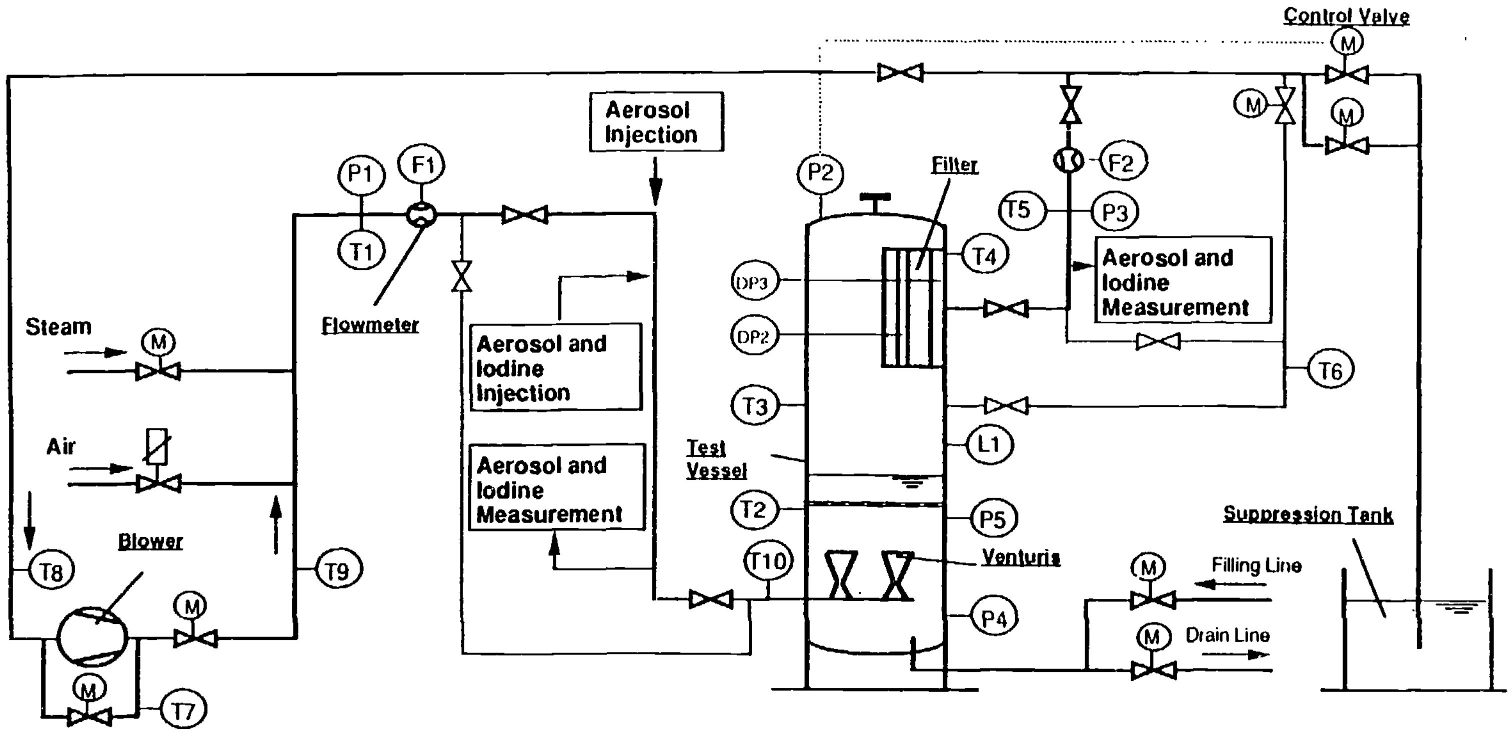
Containment Venting

Venturi - scrubber



Section of a stainless - steel
deepbed fiber filter

SIEMENS



Containment Venting
JAVA Test Facility in Karlstein
Fig. 3 a

UB KWU
S 331
13-02-B

Plant		MWe	Planning and Design	Venting System with Sliding Pressure Filter	Venting System with Atmospheric Filter	Filter Type	Completion Piping	Completion Filter
Obrigheim,	KWO	357	YES	x, outside cont.		in Discussion	exists	1989
Stade,	KKS	662	YES		x	in Discussion	1989/90	1989/90
Biblis "A",	KWB-A	1204	YES	x, inside cont.		in Discussion	1989/90	1989/90
Biblis "B",	KWB-B	1300	YES	x, inside cont.		in Discussion	1989/90	1989/90
Unterweser,	KKU	1300	YES		x	in Discussion	1989/90	1989/90
Neckarwestheim,	GKN 1	855	YES	x, outside cont.		in Discussion	1989	1989
Grafenrheinfeld,	KKG/BAG	1300	YES	x, outside cont.		in Discussion	1989	1989
Grohnde,	KWG	1365	YES	x, outside cont.		in Discussion	1989/90	1989/90
Philippsburg,	KKP 2	1362	YES	x, outside cont.		in Discussion	1988	1989
Brokdorf,	KBR	1365	Completed		x	MF	1987	1987
Neckarwestheim,	GKN 2	1314	Completed		x	MF	1987	1987
Isar Block 2,	KKI 2	1370	Completed		x	MF	1987	1987
Emsland,	KKE	1314	Completed		x	MF	1987	1987
Muhlheim-Kärlich		1300	YES	x		in Discussion	1989/90	1989/90

MF = Metal Fibre Filter for Aerosols

Table

Implementation Program of Containment Venting Systems for German PWRs

REQUIREMENTS FOR CONTAINMENT - VENTING

BOILING WATER REACTOR (BWR)

a) Layout and Operating

- To open between design pressure and test pressure of the containment.
- Energy release by the wetwell.
- Layout of the valves for redropping also at the test pressure of the containment.
- Layout of the valves for stepwise opening and dropping.
- Possibility of water injection into the Venturi - scrubber.
- Determination of the vented mass flow.
- Determination of the radioactivity of the vented flow.

GRS

b) Loads to be considered.

- Up to the outer - respectively the second - isolating valve: failure pressure of the containment; alternatively: double design pressure.

- For the connected system:

  * Pressure, temperature and composition of the discharging mixture according to the accident conditions at fully opened valve.

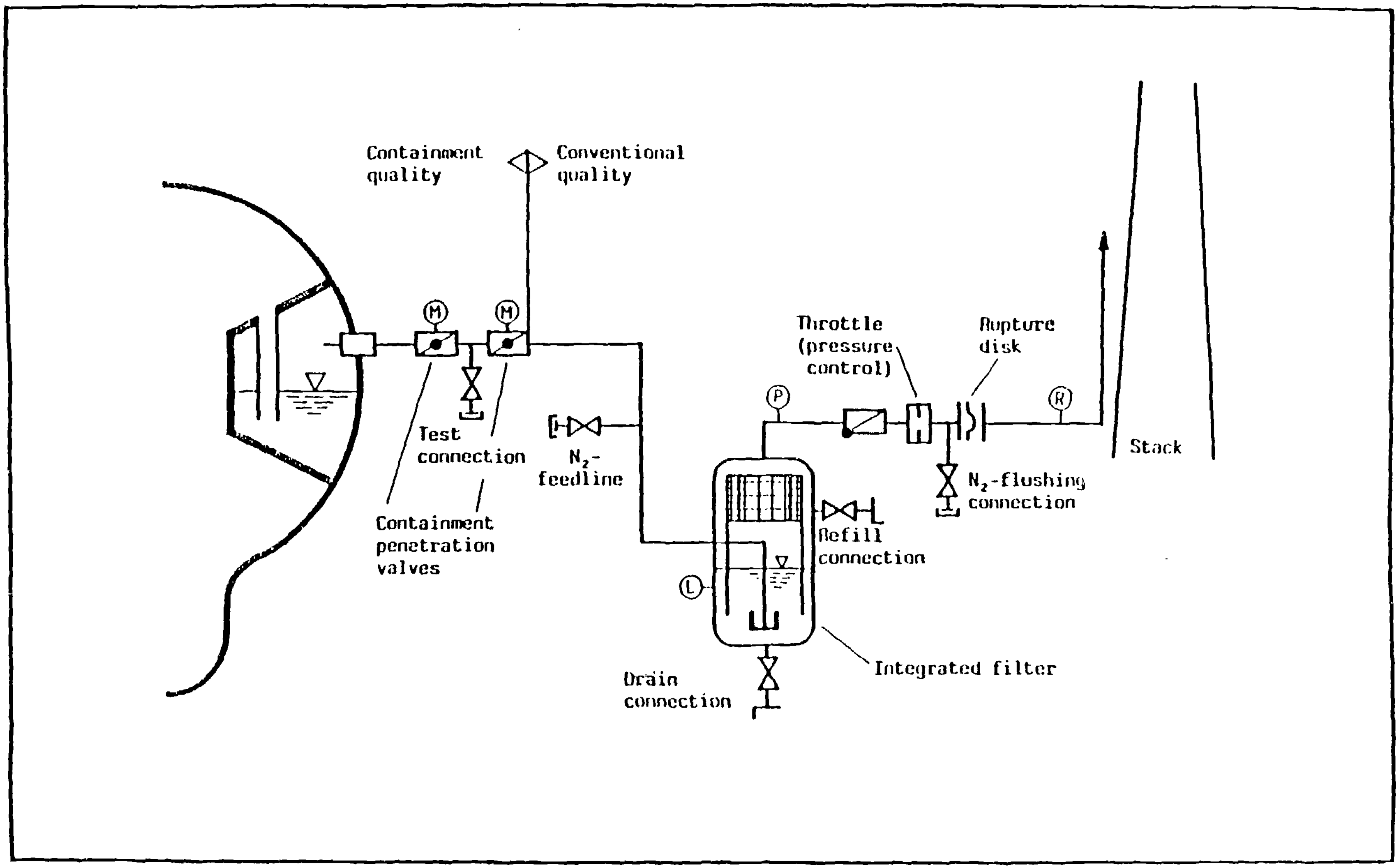
  * Layout margin for pipes and supporting systems for taking into account dynamic loads, alternatively: safety factor 2 at operating loads or 10 bar.

c) Structural data

- Fixed construction of the systems behind isolating valves.

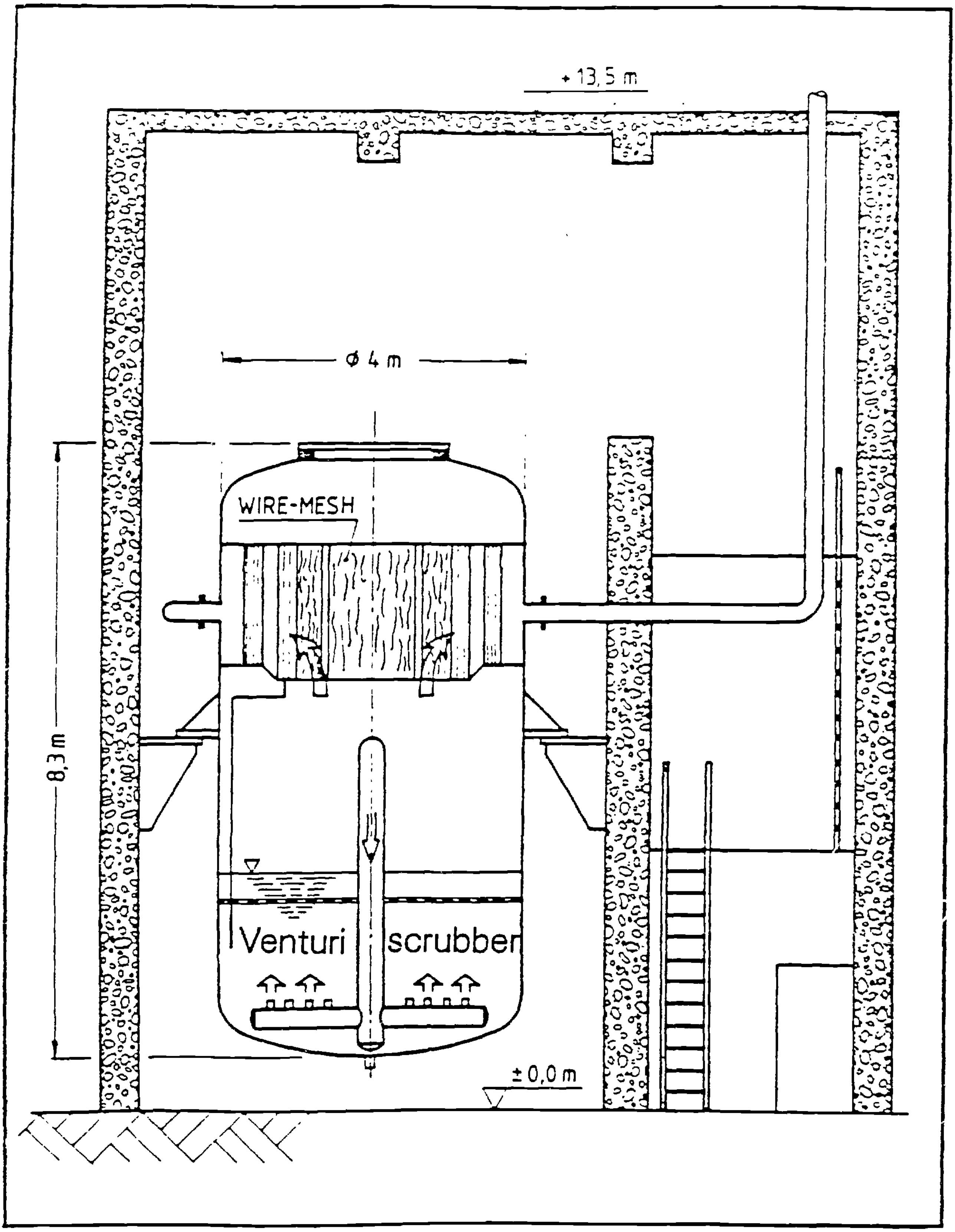
- Remote operation and additionally power supply possibility (battery).

- Fixed instalation of filter system (preferably: Venturi scrubber with stainless steel fiber filters).

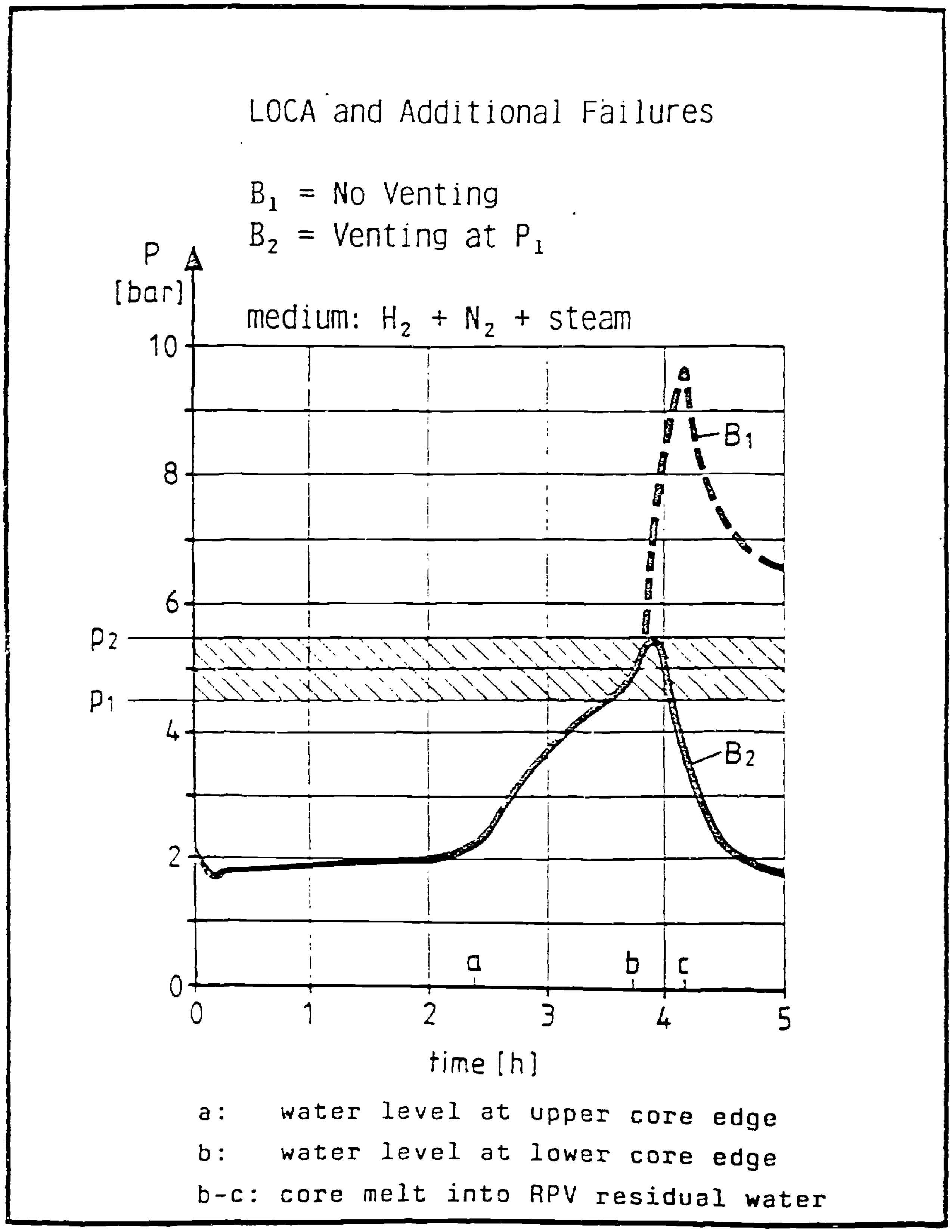


Pressure Relief and filtered Venting System

Containment Vessel of BWR 69



Siemens/KWU - Filter for BWR 69 (KKK, KKB)



Severe Accidents of BWR 69 1300 MW_e

Progress of Pressure within Containment Vessel

SIEMENS

German Requirements

	PWR		BWR
Test aerosol		SnO_2 MMD ~ 0,5 µm	
Filter loading capacity	≥ 60 kg		≥ 30 kg
Filter efficiency			
Aerosols Iodine (elem.)		≥ 99,9 % ≥ 90 %	
Passive removal of decay heat from filter	> 7 kW		≥ 200 kW
Flow rate		3 - 14 kg/ s	

Spezial requirements from other countries

	PWR	BWR
Finland TVO		
Decay heat from the filter		480 kW
UdSSR		
Aerosol size distribution	0,5 µm	

International Atomic Energy Agency

Technical Committee Meeting on :

Plant System Utilization for Accident Mitigation

Hydrogen Mitigation Measures

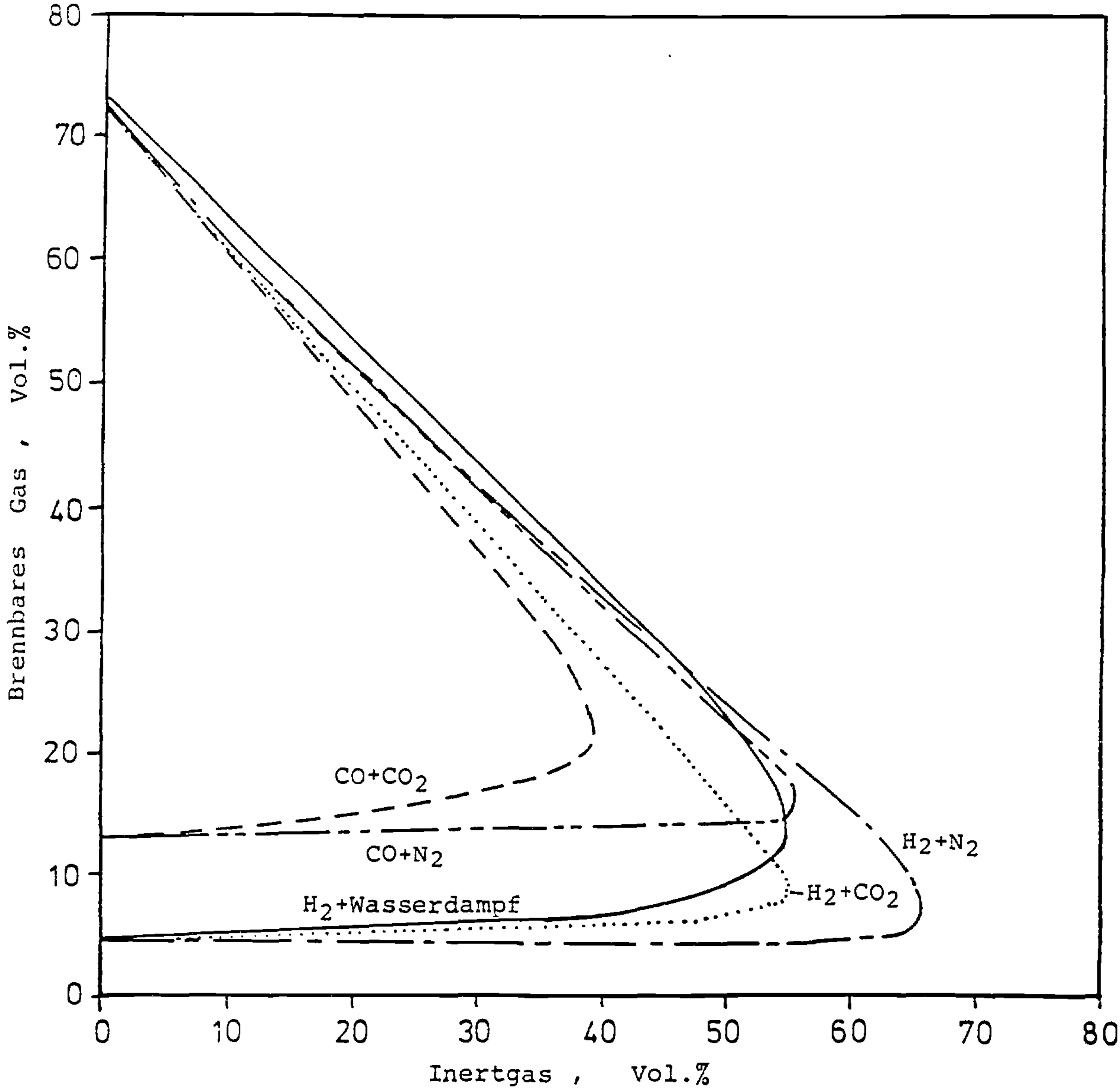
J. Rohde, GRS mbH

26. - 30. November 1990, Garching/Germany

roh002

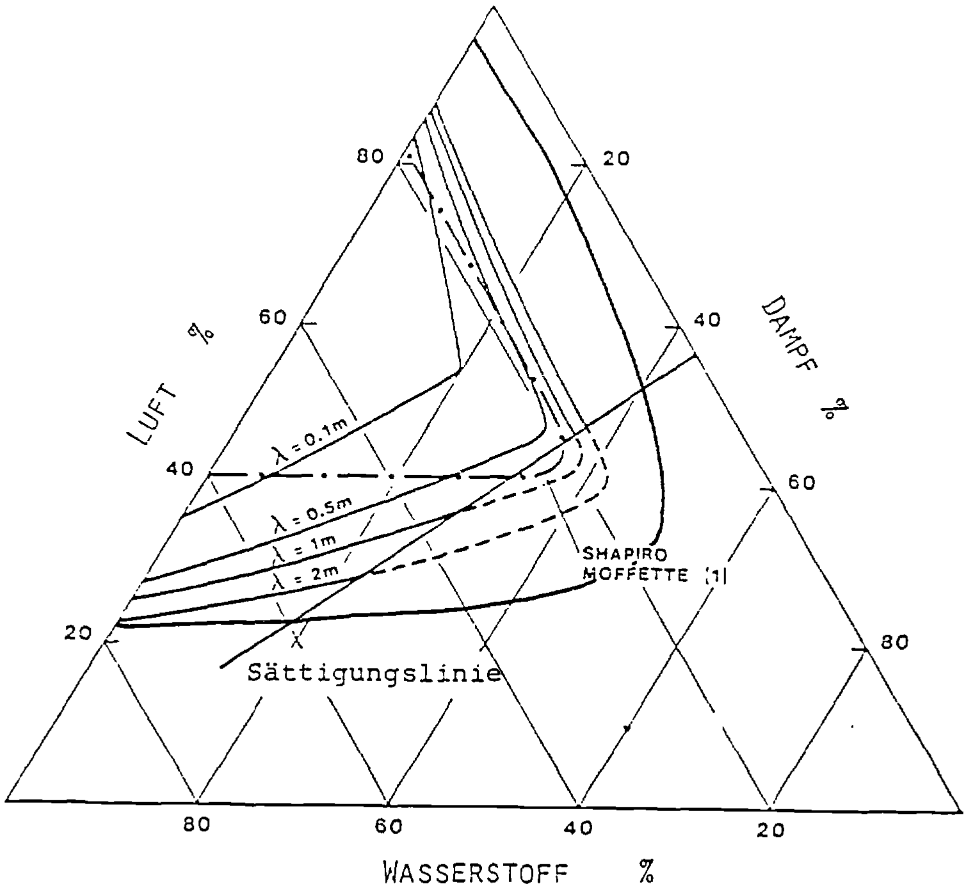
Content:

- Introduction
- Safety Goal of Mitigation Measures
- Classification of Concepts
- Status of Discussions in FRG
  - Inertisation
  - Ignitors
  - Catalytic Devices
- Final Remarks and further Procedure



BRENNBARKEITSGRENZEN VERSCHIEDENER GASGEMISCHE.
BEZUG: 1BAR 295 K

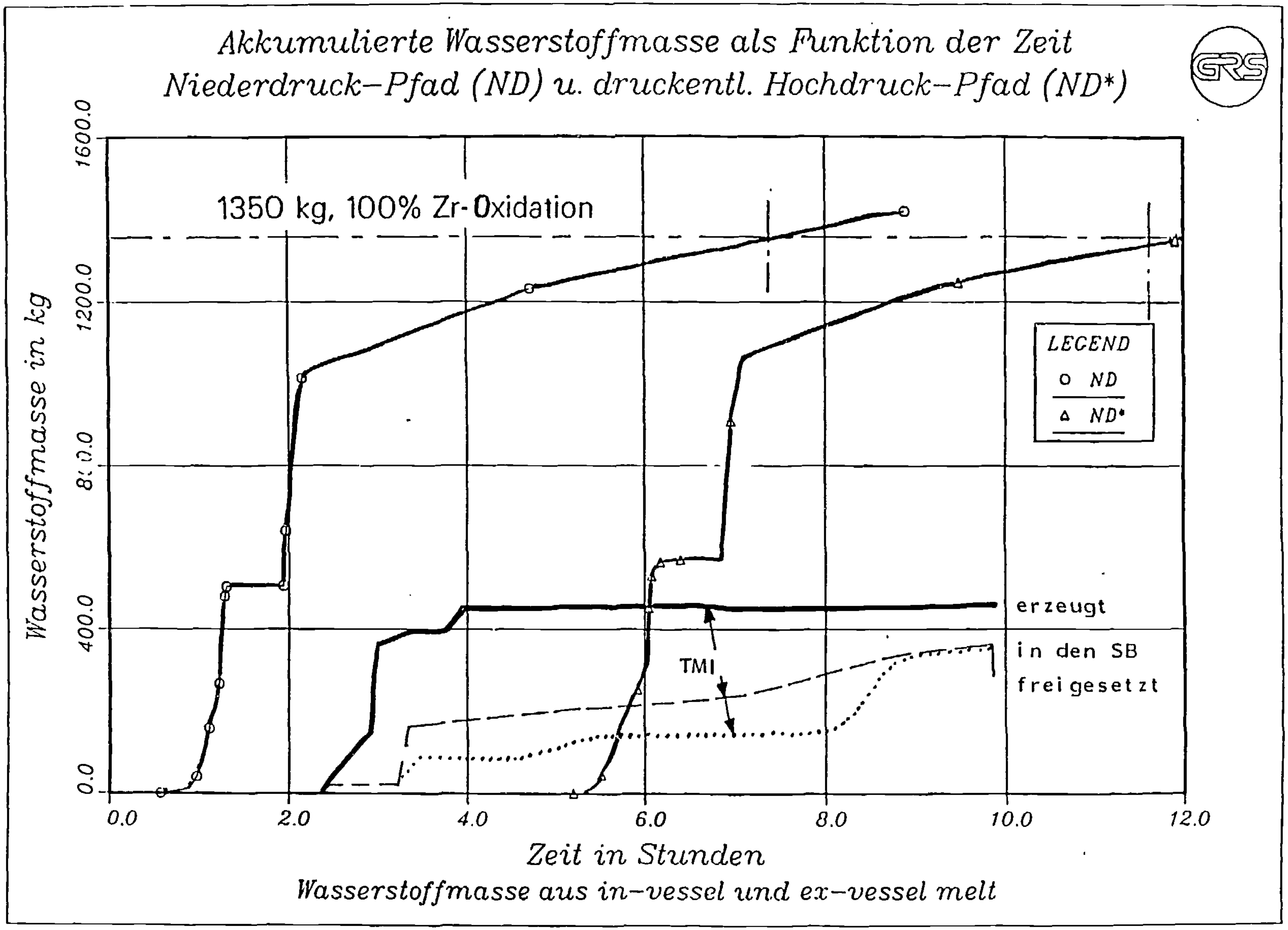
Limits for the Ignition of different Gasmixtures



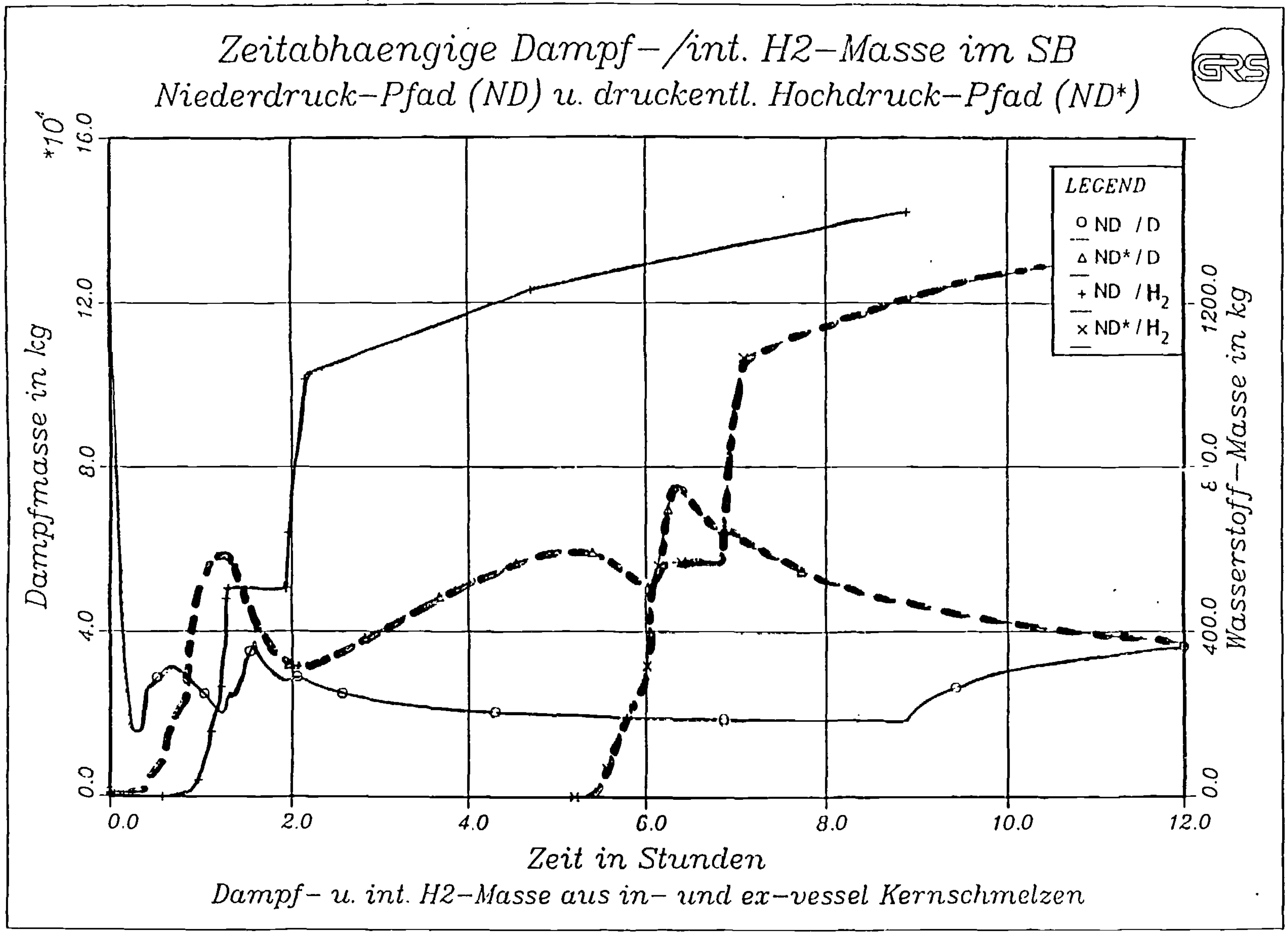
(For T = 100 °C , Air Density = 41.6 Mol/m^3)

(S.R. Tieszen, M.P. Sherman, W.B.Benedick, M. Berman
Detonability of H_2-Air-Diluent Mixtures
NUREG/CR-4905, SAND 85-1263 (1987))

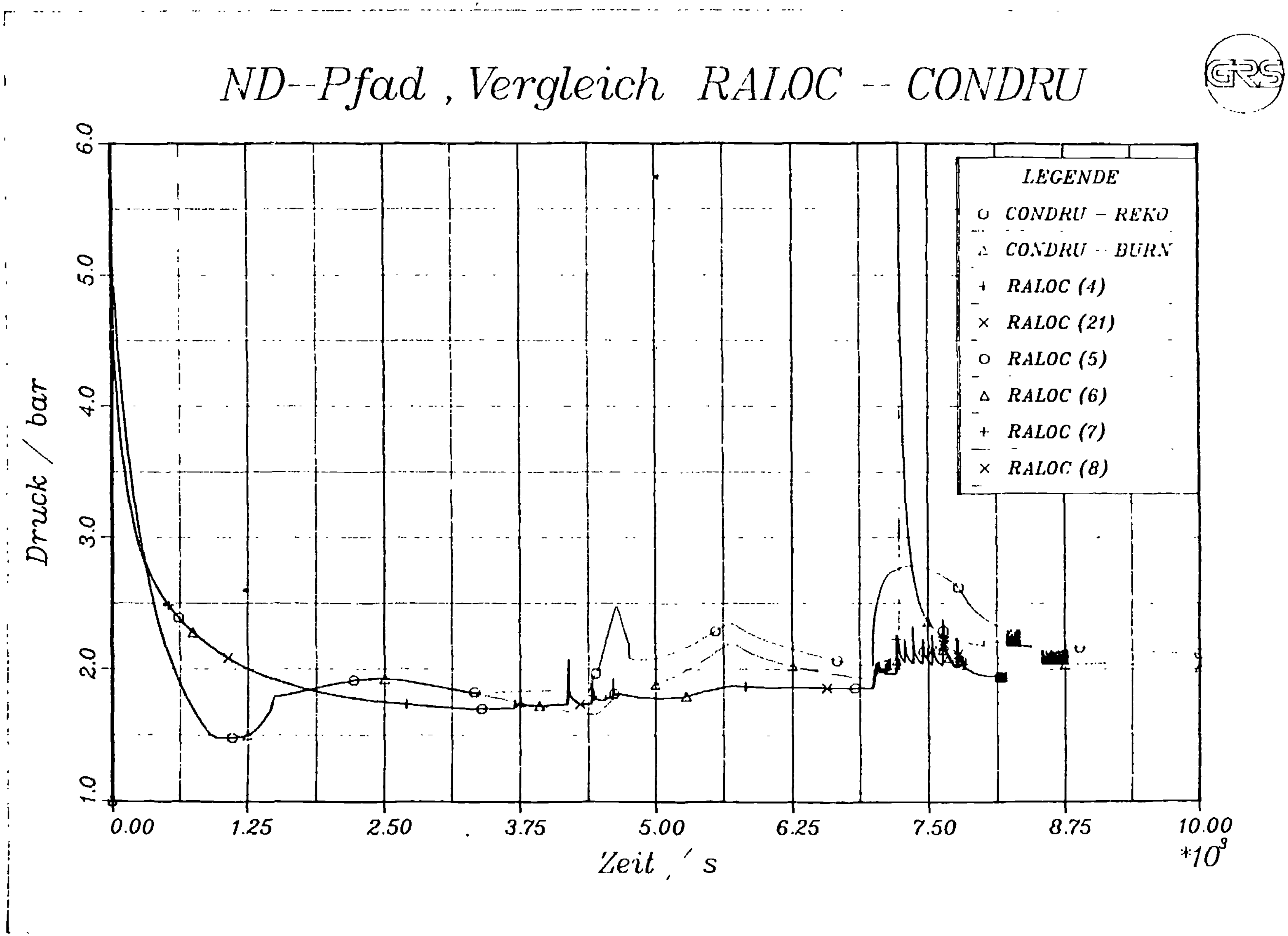
Lines of constant Detonation-Cell-Width
in H_2-Air-Steam Mixtures



Akkumulierte Wasserstoffmasse als Funktion der Zeit
Niederdruck-Pfad (ND) u. druckentl. Hochdruck-Pfad (ND*)
1350 kg, 100% Zr-Oxidation
Wasserstoffmasse in kg
1600.0
1200.0
800.0
400.0
0.0
0.0
2.0
4.0
6.0
8.0
10.0
12.0
Zeit in Stunden
LEGEND
ND
ND*
erzeugt
in den SB
freigesetzt
TMI
Wasserstoffmasse aus in-vessel und ex-vessel melt

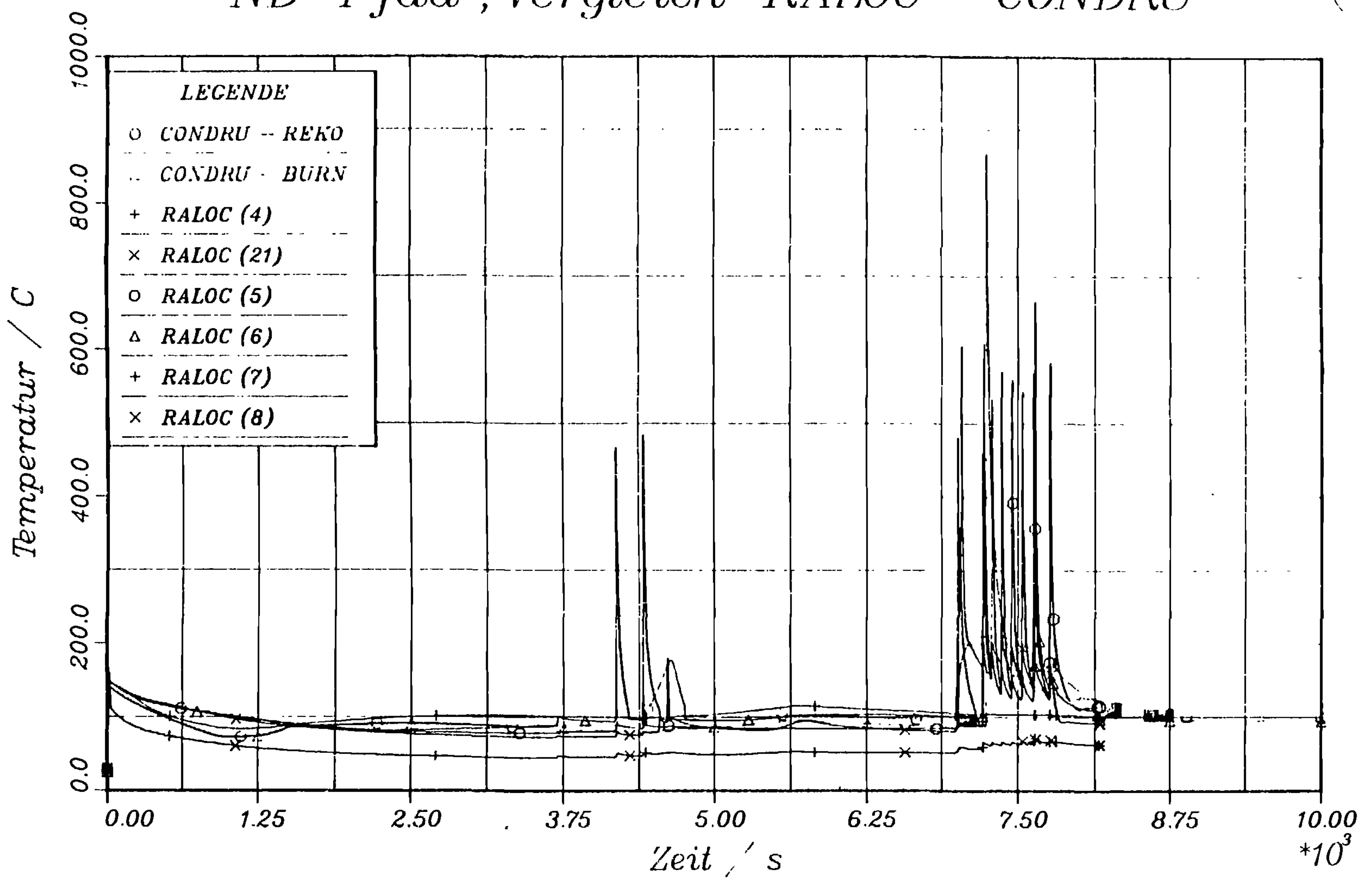


Zeitabhaengige Dampf-/int. H2-Masse im SB
Niederdruck-Pfad (ND) u. druckentl. Hochdruck-Pfad (ND*)
GRS
LEGEND
ND / D
ND* / D
ND / H2
ND* / H2
Dampfmasse in kg
*10^4
0.0
4.0
8.0
12.0
16.0
Wasserstoff-Masse in kg
0.0
400.0
800.0
1200.0
0.0
2.0
4.0
6.0
8.0
10.0
12.0
Zeit in Stunden
Dampf- u. int. H2-Masse aus in- und ex-vessel Kernschmelzen



28 JUN 1988 5:19 PM BY HGT
PRODUCED BY MD-PLOT REL 5.1.0

ND--Pfad , Vergleich RALOC -- CONDRU



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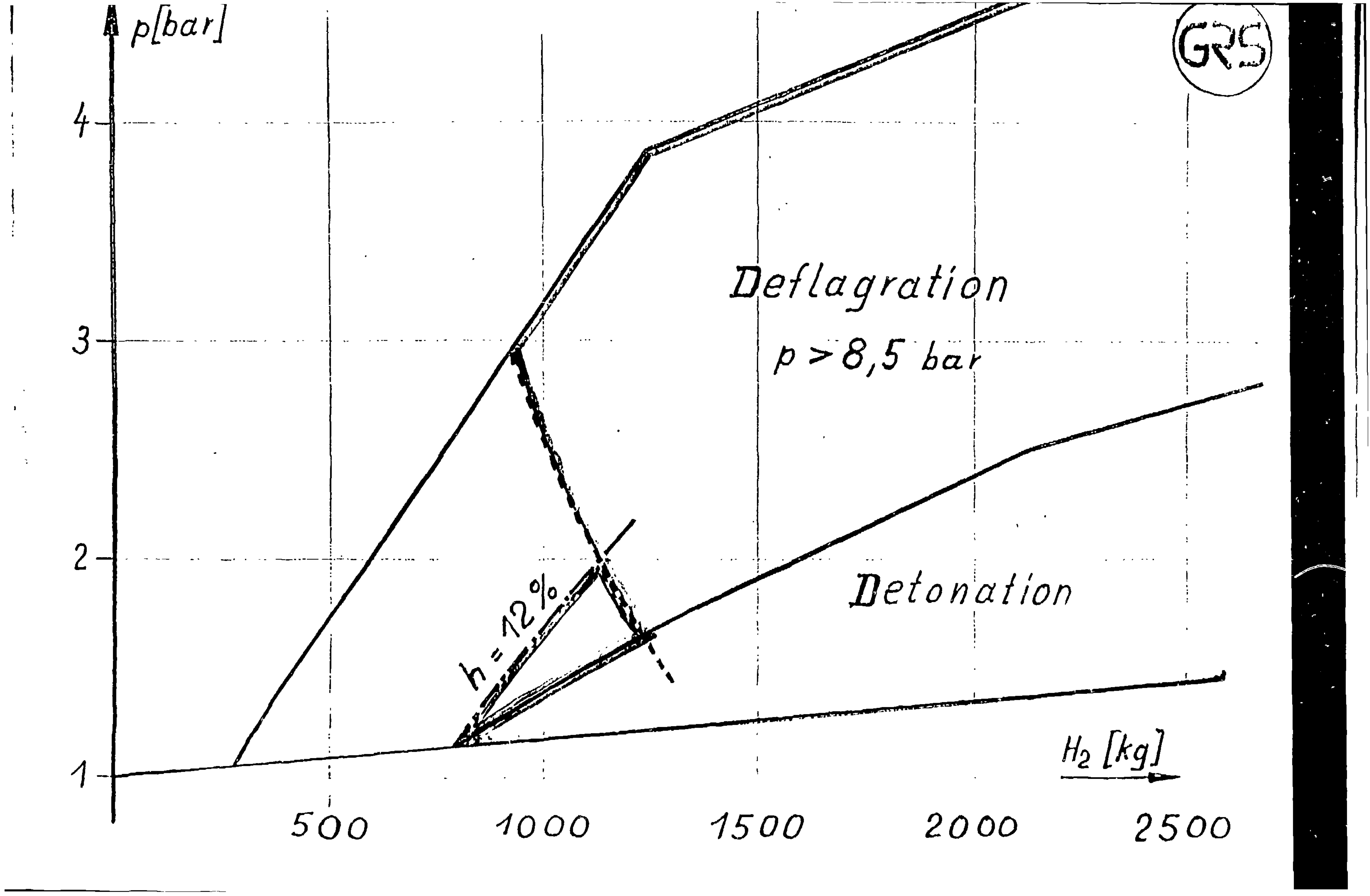

p[bar]
GRS
4
3
2
1
Deflagration
p > 8,5 bar
Detonation
h = 12%
H2 [kg]
500
1000
1500
2000
2500

Safety Goal of Mitigation Measures

"To prevent early and late Containment Failure by Hydrogen-Burns"

1. Priority: Exclude a global Detonation and for a Deflagration with the Potential to reach Failure-Pressure of Containment

2. Priority: Prevention of local Detonations which could lead to Missile-Generation

3. Priority: Prevent local Hydrogen Concentration of greater than 10 Vol-%

4. Priority: Mitigate the consequences of local, multiple Burning, leading to high Temperatures. (failure of local equipment)

Classification of Concepts

- Limitation of Hydrogen Production and Release
  - In-Vessel Zirconium Oxidation
  - Ex-Vessel Coolability of Core Material
  - Limitation or Termination of Core-Concrete-Interaction
- Limitation of Hydrogen Concentration
  - By deliberate Ignition of burnable Gasmixtures
  - catalytic induced Reactions
  - Venting of the atmosphere
- Prevention of a Hydrogen Burn
  - Inertisation with carbondioxid or nitrogen

IGNITERS

- GLOW PLUG IGNITERS

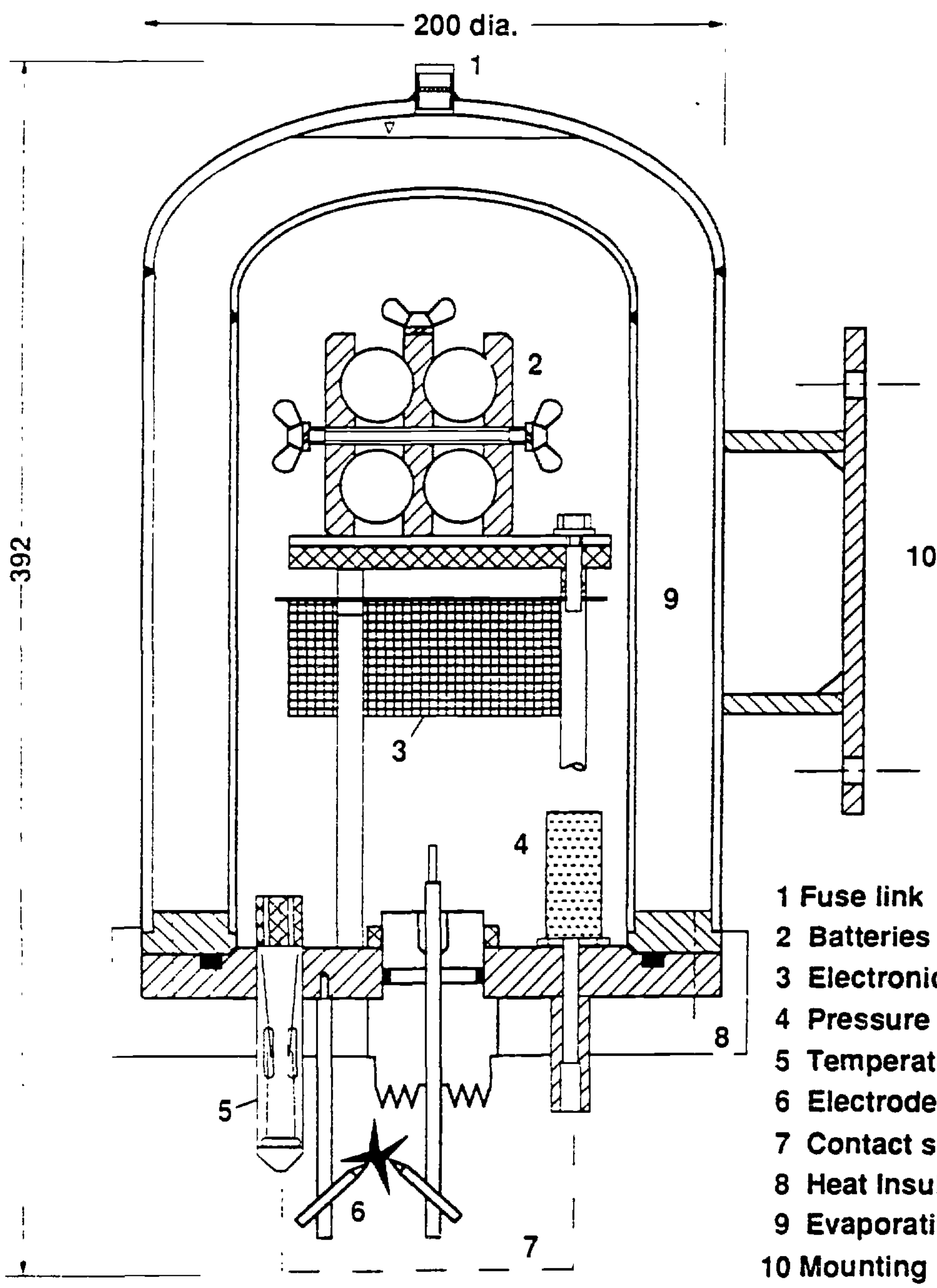
  HIGH NUMBER AND LENGTH OF CABLES,
  ADDITIONAL PENETRATIONS INTO CONTAINMENT
  HIGH ENERGY CONSUMPTION

  THEREFORE IN FRG THE DEVELOPMENT WAS CONCENTRATED ON BATTERY POWERED IGNITERS AND IGNITERS, WORKING ON A CATALYTIC BASE

- SPARK IGNITERS WITH BATTERIES

- CATALYTIC HYDROGEN IGNITERS

SIEMENS



1 Fuse link
2 Batteries
3 Electronics
4 Pressure Switch
5 Temperature Switch
6 Electrodes
7 Contact safety device
8 Heat Insulation
9 Evaporative cooling
10 Mounting

Spark Igniter with Batteries WZB 89

UB KWU
R 243
17.05.90
18-18-A

SIEMENS

• Actuating temperature	70 °C to 160 °C (adjustible)
• Actuating pressure	0, 2 to 6, 9 bar (adjustible)
• Spark frequency	≥ 7 per minute
• Ignition energy	> 10 mJ
• Ignition voltage	> 20 kV
• Radiation resistance	> 72 kGy at dose rates between 500 - 10. 000 Gy/ h
• Design pressure	6 bar
• Design temperature	160 °C (Blow Down, 1 h) 125 °C (Long Duration, 5 d) 350 °C (Multiple combustion, 0, 5 h)
• Dischargetime of batteries	≥ 5 days
• Size/ Weight	ø 200 mm; 390 mm high/ approx. 15 kg
• Lower H_2 - ignition concentrations	
in dry atmosphere	
quiescent	4, 5 Vol. - % H_2
moved	4,5 Vol. - % H_2
in saturated steam/ air - mixtures	
quiescent	4, 5 .... 7,5 Vol. - % H_2 *
moved	4,5 .... 6 Vol. - % H_2 *

* rise with increasing steam coutent

Spark Igniter
Main Technical Data

UB KWU
R243
He/10. 90
18-01-A

SIEMENS

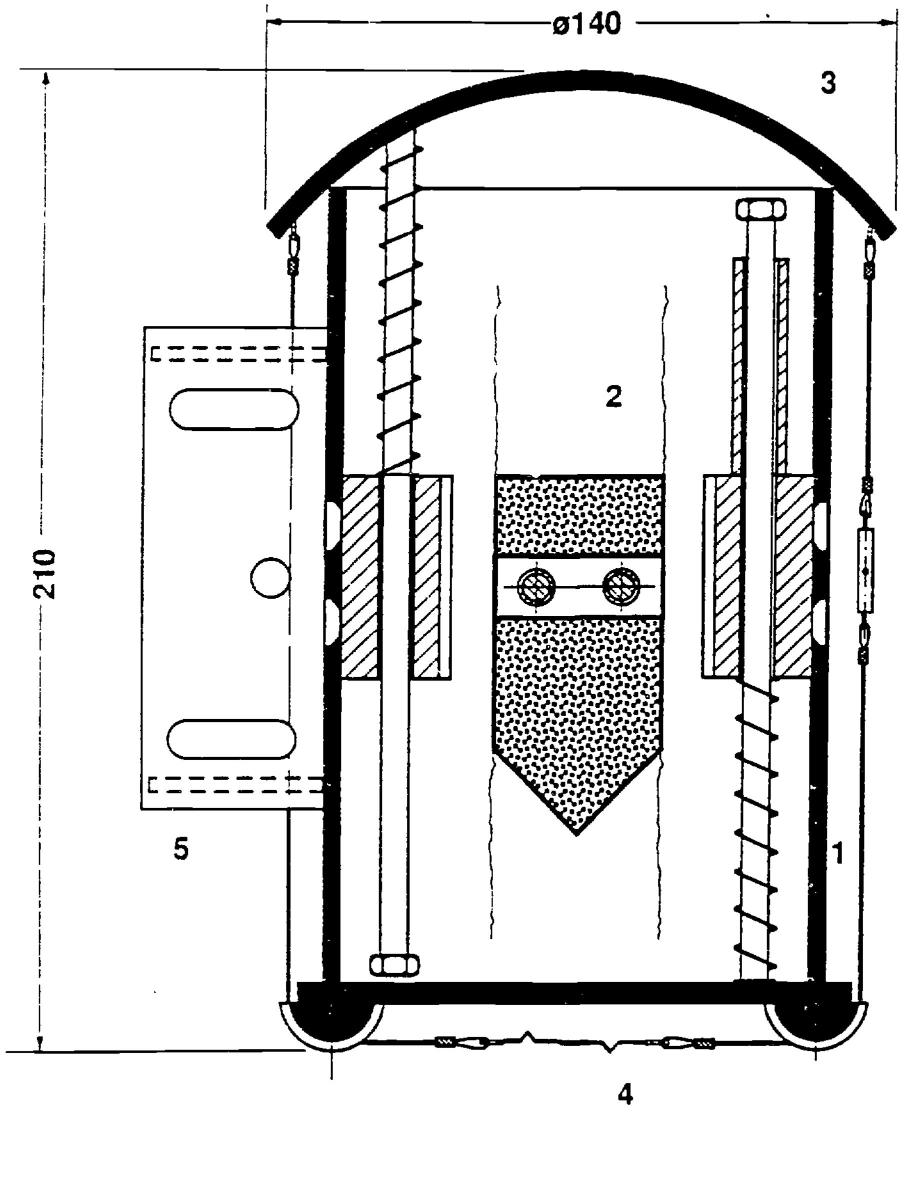
Test step	Conditions
Thermal Aging (Normal Operation)	1 bar; 125°C; 360 h
Radiation (Normal Operation)	50 kGy; 500 Gy/ h; 100 h
Vibration (Normal Operation)	acc. to IEEE 382; IEC 68-2-6
Radiation (DBA and Severe Accident)	10 kGy; 10000 Gy/h; 1 h 7 kGy ; 1000 Gy/h; 10 h 5 kGy; 500 Gy/h; 1 h
Thermodynamic Load (Design Basis Accident)	1 .. 6 bar; 40 .. 160°C; 2.5 h
Multiple Combustion (Severe Accident Conditions)	2.5 bar; 125 °C; 1.5 h 2.5 bar; 350 °C; 0.5 h
Long-Duration Function (Severe Accident Conditions)	2.3 bar; 125°C; ≥ 5 days

Spark Igniter

Qualification Programm and Conditions

UB KWU
R 243
17 05 90
18-17 A

SIEMENS



1 Housing

2 Ignition Element

3 Cover

4 Fuse link

5 Mounting

Catalytic Igniter

UB KWU
R 243
17.05.90
18-19-A

SIEMENS

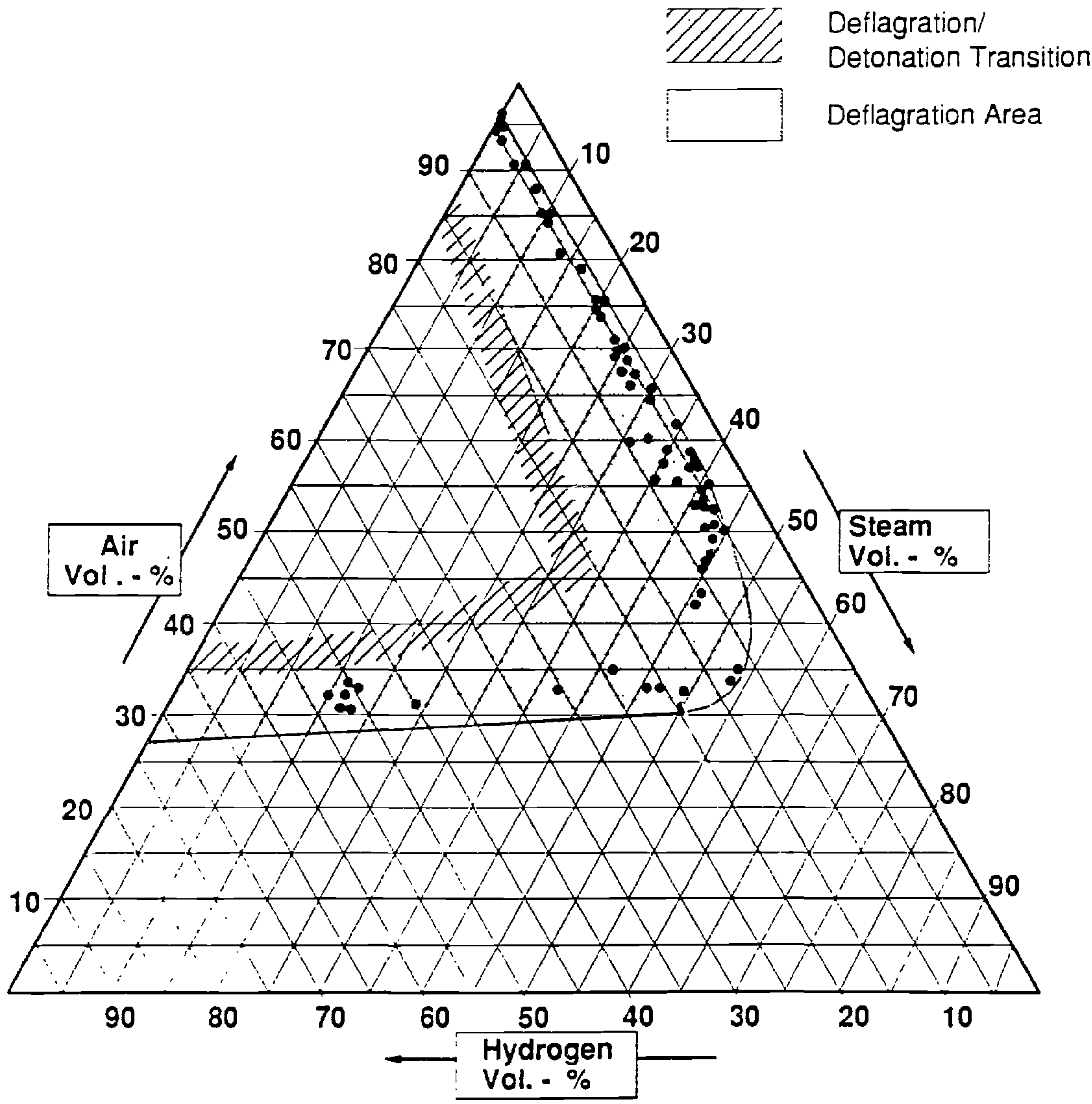
• Type	WZK 88
• Functional principle	Gas - ignition on catalyst surface heated by exothermic reaction
• Catalyst	Platinum
• Catalyst Basis	high temperature resistant metallic foil with active intermediate layer (washcoat)
• Actuating	automaticly during temperature increase
• Size of Housing	ø 140 mm; 210 mm high
• Weight	approx. 2 kg
• Lower H_2 - ignition concentrations	
in dry atmosphere	
quiescent	4, 5 Vol. - % H_2
moved	> 6 Vol. - % H_2
in saturated steam/ air - mixtures	
quiescent	4, 5 .... 8 Vol. - % H_2 *
moved	> 6 .... 8 Vol. - % H_2 *

* rise with increasing steam content

Catalytic Igniter
Main Technical Data

UB KWU
R243
He/10. 90
18-02-A

SIEMENS



Performed Ignitions with Catalytic and Spark Igniters

UB KWU
R 243
08.1990
13-29-A

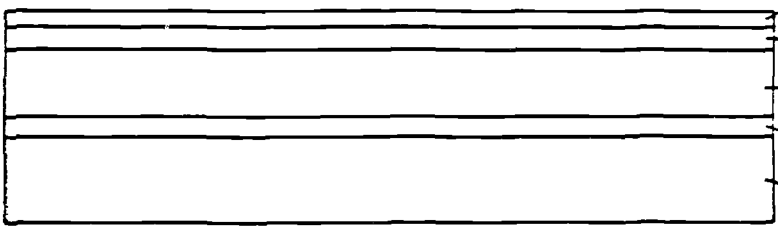
CATALYTIC DEVICE TO REMOVE HYDROGEN FROM CONTAINMENT ATMOSPHERE

ADVANTAGES:

- PASSIVE SYSTEM
- FUNCTIONABLE ALSO FOR VERY LOW HYDROGEN CONCENTRATIONS (PPM) AND UP TO 90% STEAM
- THE EFFECTIVENESS OF CATALYTIC REACTION INCREASES WITH TEMPERATURE
- NO RESTRICTIONS CONCERNING MASS OF HYDROGEN
- INCREASES CONVECTION IN ROOMS (MIXING OF GASES)
- CONTINUOUS ENERGY - INPUT TO CONTAINMENT ATMOSPHERE

DISADVANTAGES:

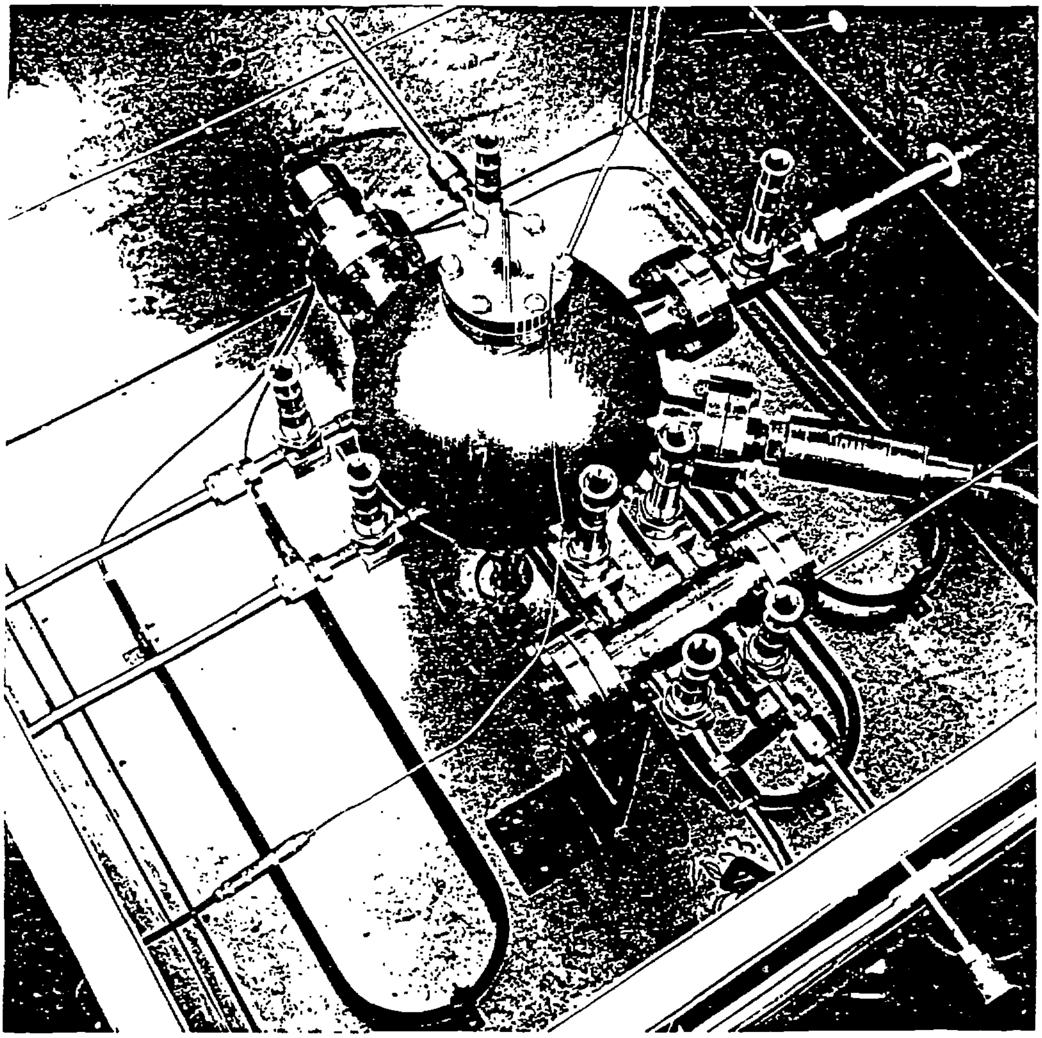
- NEEDS LARGE SURFACE - AREAS
- POSSIBILITY OF CATALYTIC POISSONING
- LAY-OUT AGAINST PRESSURE-TRANSIENTS



2036K

COMPOSITION OF THE SANDWICH FOIL.
VACUUM COATING CAN BE UNDERTAKEN ON
BOTH THE SIDES OF AL

GRS



2045K APPARATUS TO DETERMINE THE CATALYTIC REACTION OF THE FOIL

GRS

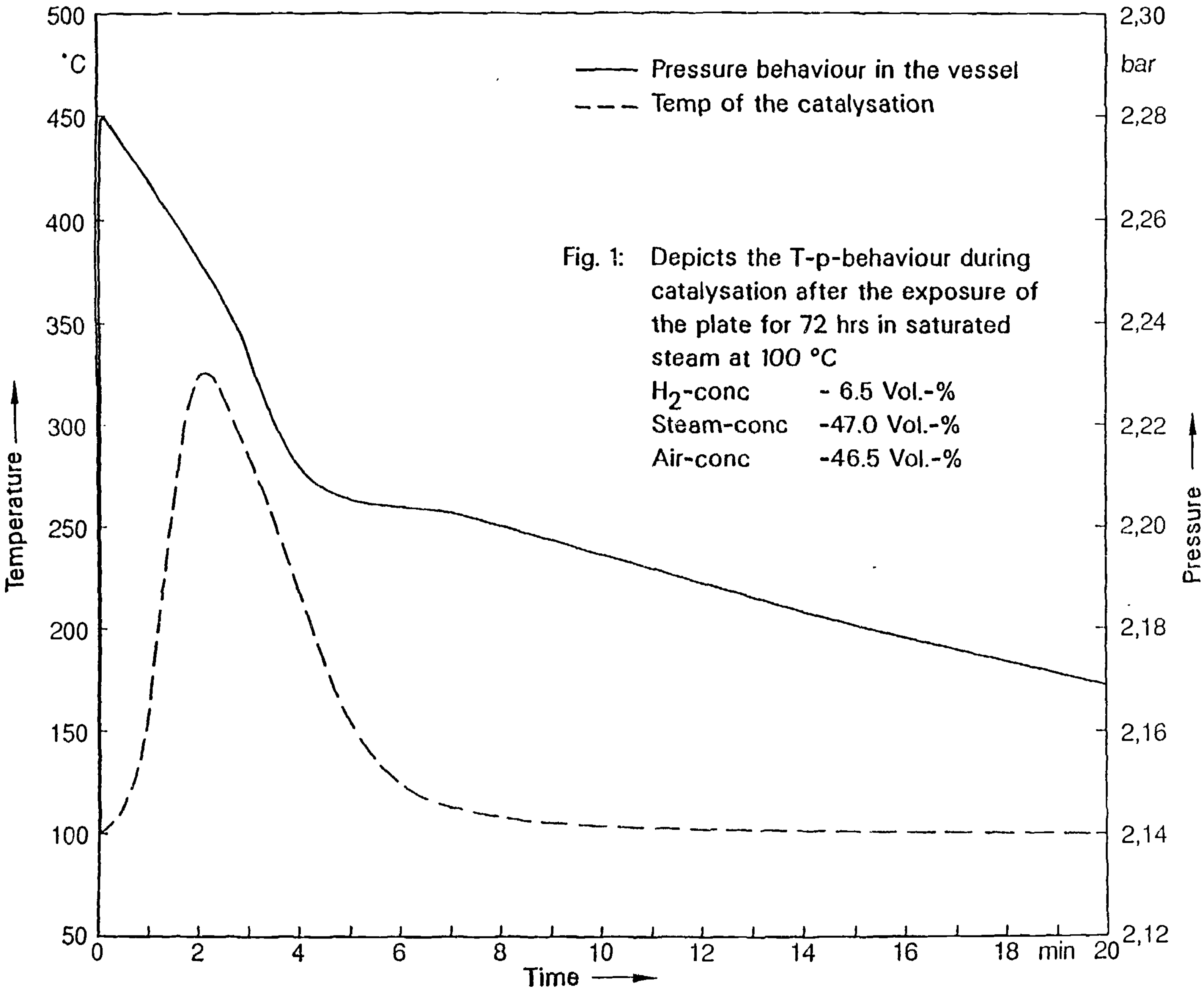


Fig. 1: Depicts the T-p-behaviour during catalysation after the exposure of the plate for 72 hrs in saturated steam at 100 °C

H_2-conc - 6.5 Vol.-%

Steam-conc -47.0 Vol.-%

Air-conc -46.5 Vol.-%

CHA\7290-1

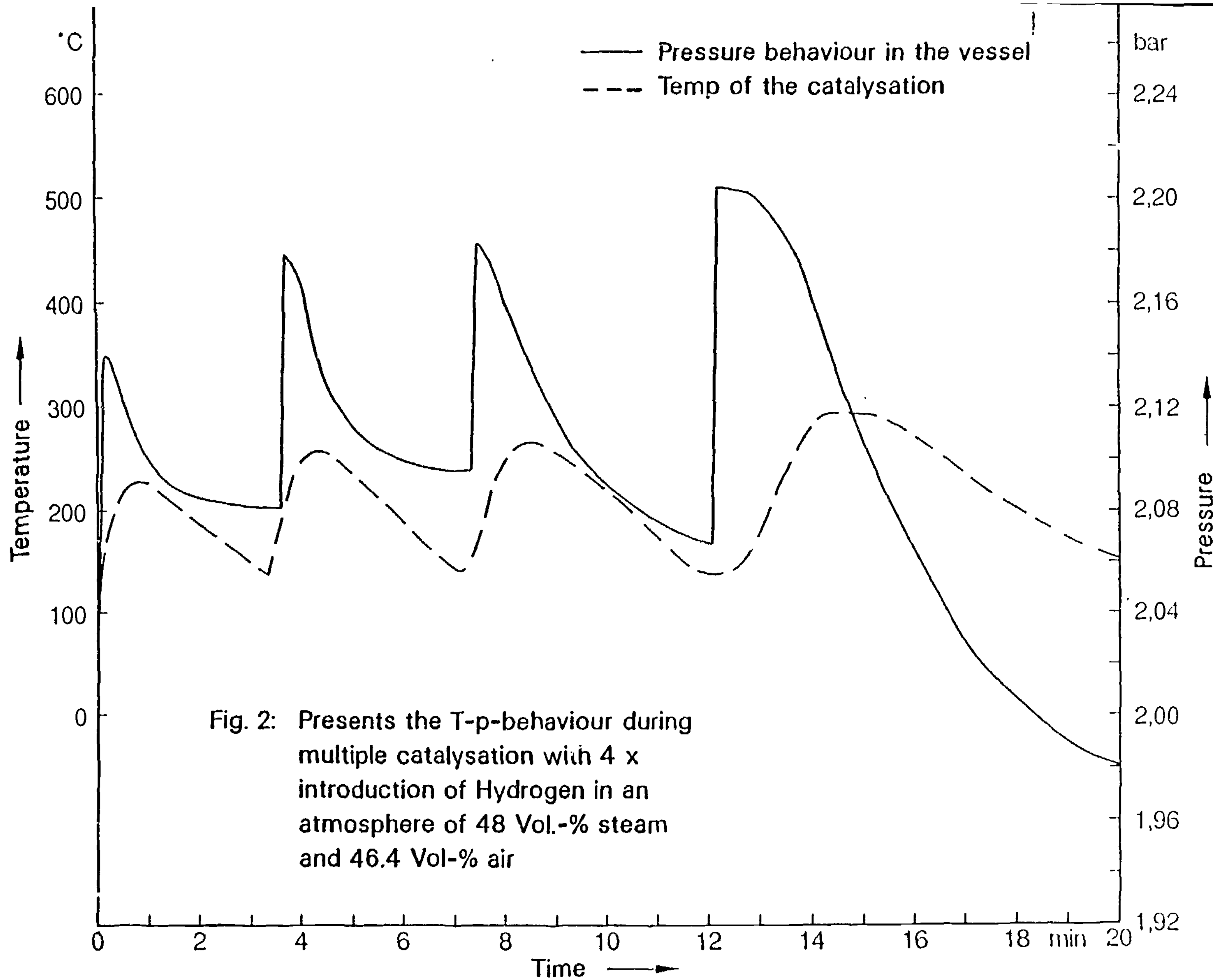


Fig. 2: Presents the T-p-behaviour during multiple catalysation with 4 x introduction of Hydrogen in an atmosphere of 48 Vol.-% steam and 46.4 Vol-% air

PERFORMED TESTS

° NUMBER OF EXPERIMENTS	> 300
° RANGE OF HYDROGEN-CONCENTRATIONS	1 - 25 VOL%
° STEAM-CONCENTRATION	UP TO 90 VOL%
° INITIAL ROOM TEMPERATURE	20 - 250 °C
° INITIAL PRESSURE	1,01 - 6 BAR
° VOLUME OF TEST APPARATUS	6; 265 LTR
° TESTS WITH CATALYTIC POISONS	CO; CO_2; J_2; SULFUR POWDER CS (OH) , NACL; BORIC ACID

PLANED TESTS

- TWO TESTS IN THE MODELL-CONTAINMENT (1:64) AT BATTELLE/FRANKFURT IN 1989 (FORCED AND FREE CONVECTION TESTS, T ≈ 100 °C, P ≈ 2 BAR)

- ADITIONAL TESTS ARE PLANED FOR THE HDR-TEST-FACILITY IN 1990/91 (LARGE SCALE!)

STATUS OF COUNTER MEASURES IN THE FRG:

BWR - PLANTS:

- ALL REACTORS OF THE CONTAINMENT TYPE BL 69 (5) HAVE TO BE PRE-INERTED. THE INSTALLATION IS NOT FINISHED UP TO NOW IN SOME PLANTS.

- FOR THE REACTORS WITH THE CONTAINMENT TYPE BL 72 (2), SIMILAR MARC III, A CONCEPT TO ONLY PRE-INERTE THE WETWELL IS STILL UNDER DISCUSSION.

PWR-PLANTS:

- THE GERMAN RISK STUDY, PHASE B STATED, THAT THE RISK FROM HYDROGEN BURNS COULD LEAD TO EARLY CONTAINMENT FAILURE WITH HIGH CONSEQUENCES TO THE ENVIRONMENT.

- DIFFERENT MEASURES WERE ANALYZED. UP TO NOW A DECISION FOR A SPEZIAL CONCEPT IS STILL OPEN

- THE GERMAN MINISTRY FOR ENVIRONMENT , NATURAL CONSERVATION AND NUCLEAR SAFETY (BMU) HAS ANNOUNCED, THAT THE HYDROGEN QUESTION MUST BE SOLVED IN THE NEXT TIME.

ACCIDENT MANAGEMENT POSSIBILITIES FOR VVER-440

G. Lajtha

Institute for Electric Power Research
Budapest

To be presented at the

IAEA Technical Comittee Meeting

on

Plant System Utilization

for Accident Mitigation

Garching, Federal Republic of Germany
26-30 November 1990

INTRODUCTION

The first Hungarian report on the topic of severe accident mitigation [1] was published after the TMI-2 accident in 1979. In that report the qualitative analysis of the TMI type accident for Paks NPP and the possibility of preventation of this accident was worked out. At that time no mathematical model was available to us for the calculation.

In 1986 the IAEA gave us the MARCH-2 code, and later the Source Term Code Package which consisted of the MARCH-3 [2] code. This code was good for calculations of severe accident events of western type nuclear power plants. The code's application for VVER-440 reactors needed programing modification. With the support of the IAEA and the Kurchatov Institute an international working group performed the code modifications in 1989 [3]. Thus a suitable tool for severe accident quantitative calculations is available to us on an IBM compatible personal computer.

Calculations were made by this programme and the experience of some severe accident events pointed to the possibility of operator interventions. The first steps were thus made to see how the operator can mitigate or prevent a severe accident.

In this paper I give a description of those systems of the Paks NPP which may be important to prevent accidents. Two examples will be shown. A transient with loss of on-site and off-site power and a small break LOCA connected with the operator intervention will be analyzed.

DESCRIPTION OF THE PAKS NPP

First of all a short description of Paks NPP is given for a better understanding.

Hungary's nuclear power plant at Paks /the Paks NPP/ is operated with four units of VVER-440 type reactors. The VVER-440 reactor is a PWR type one; each reactor has two steam turbines with a capacity of 220 MWe each. Pressure is 123 bar in the primary circuit and 45 bar in the secondary circuit. Concerning its design criteria the unit well corresponds to other PWRs throughout the world, however, it differs significantly in certain constructional features.

The most important features of one unit of VVER-440 plant (Fig. 1.):

- six loops (c.a. 225 m3 water) in the primary circuit (Fig. 2.) with valves.
- four hydroaccumulators
- three redundant high pressure and low pressure injection systems for emergency core cooling. They are completely seperated from the normal feedwater and make up system of the primary circuit.

- two independent secondary circuits for each reactor
- five feedwater pumps, so the malfunction of one pump cause only 20 percent decrease in capacity.
- there are two emergency feedwater pumps, furthermore four pumps are in reserve for two units.
- the reserved water in the secondary side is more than 3000 m3.

- the containment consits of hermetic compartments and bubble/condenser tower. There is about 1500 m3 water in the trays of the condeser. (Fig. 3.)
- three redundant spray systems for containment pressure reduction.

THERMO-HYDRAULIC ANALYSIS OF THE S2B EVENT AND POSSIBLE OPERATOR INTERVENTIONS [4]

According to the PSA analyses for the VVER-440 type nuclear power plant two events were choosen. The first event is a small break (dia. 40 mm) LOCA with station blackout. Table 1 consists of the summary of the most important events of the S2B sequence.

EVENT	TIME min
CORE UNCOVERY 1	44
HYDROACCUMULATORS START	48
HYDROACCUMULATORS EMPTY	95
CORE UNCOVERY 2	152
START F.P. RELEASE	177
START MELT	182
CORE SLUMP	192
START GRID1 HEATUP	200
GRID1 FAILED	221
START HEAD HEATUP	222
VESSEL DRYOUT	246
BOTTOM HEAD FAILED	427
CAVITY DRY OUT	427
START CORE-CONCRETE INT.	428
END OF CALCULATION	1625

Table 1. Events of VVER-440 S2B sequence

The primary system pressure transient (Fig. 4) may be the best illustration of the events during the sequence. In the first minutes of accident time the primary pressure decreases rapidly until t=3 min (point A on Fig. 4), when the saturation state is reached in the primary system and the pressure gradient is changed. At 43.8 min the mixture level sinks to the top of the core (C) and uncovery starts. The mixture level decreases steadily from the start of the accident, until the break elevation is reached at 31.6 min (B).

When the primary pressure equals p=6 MPa, hydroaccumulators start to inject water (D, t=49.7 min) and the mixture level is oscillating around the break elevation. Accumulators are depleted by 94.7 min (E); this starts a steeper depressurization rate. At point F, t=105.1 min, the subcooled primary coolant reaches the saturation state, boiling and steam generation begins and the primary pressure begins to rise again.

Core uncovery occurs for a second time at 152.2 min, followed by slowing pressurization. The partially covered core generates less steam, so pressure rise stops at 160 min (point G) and depressurization begins. The next characteristic point H at 221.5 min corresponds to vigorous vapour flashing and pressure rise after the bottom grid plate failure. Gradient change at point SGin is attributed to the steam generator influence. From t=50.4 min (SGout) the primary system saturation pressure has been lower than the secondary system saturation pressure. According to the MARCH model, the steam generator does not cool the system under these conditions. From t=233 min the primary pressure exceeds the secondary side value and heat exchange takes place again. Moderated by the cooling effect of the steam generator, pressure rise will be stopped at 242.5 min (pont J). Starting from this point, depressurization will go on until 246.3 min (point K), when the vessel dries out completely. From this point, primary pressure will be decreasing continuously until bottom failure is reached at 426.8 min.

IDENTIFICATION OF OPERATOR INTERVENTIONS AND THEIR TIMING

There are several kinds of operator interventions that could be identified:

A) delay core damage or vessel failure by cooling the primary system from the secondary side;
B) stop the accident sequence via restoring the emergency core cooling;
C) supply water in the reactor cavity

A) Delay of core damage or vessel failure.

It has been mentioned earlier that steam generators do not cool the primary system during a prolonged time interval, from 50.4 min until 233 min, because the primary pressure drops below pressure of the secondary coolant. Operators can bring down secondary pressure, and get the steam generators cool the primary system again. The time frame is shown in Fig. 5., representing energy loss to steam generatos without operator intervention. Secondary system discharge may be started from 50.4 min of accident time . In the time period 50.4-152.2 min (until the core uncovery) the action can delay core damage by about 1 hour. Fig. 6 shows the effect of the cooling of the steam generator. If the intervention is started in the time period from 152.2 min (uncovery) to 181.6 min (start melt), then at least a partial core damage is sustained. After 181.6 min the secondary side discharge cannot save the core, but it can help to maintain vessel integrity for some time. The deadline of the intervention is 146.3 min, when the vessel dries out. After this time, there is no use of restoring secondary side cooling.

B) ECC recovery

Emergency core cooling recovery - meaning first of all a power supply recovery in the S2B sequence - is a most plausible action to stop a severe accident sequence. There is a broad time frame to supply water to the reactor; however, the result of the action will be depending on the timing. This is shown in Fig. 7, which represents core temperatures with no operator interaction. The best opportunity to recover ECC would be in the time frame 0-152.2 min, i.e. until core uncovery. This could prevent any core damage. From 152.2 min to 181.6 min (start melt) severe core degradation is suffered. From 181.6 to 200.5 min (core collapse) the intervention cannot stop core melting; on the other hand it may cause steam flashing or explosion. Although the first battle has been lost, there is still a hope after core collapse that vessel integrity could be preserved by cooling the debris in the bottom head. The optimal time span extends to 246.3 min, i.e. until vessel dryout, but the option is pending until bottom head failure (426.8 min).

C) Reactor cavity flooding before core damage

The reactor vessel is installed in the cylidrical cavity called reactor shaft. The reactor shaft is divided into lower and upper regions by a bellows-type seal located immediately below the top of the vessel. We deal with the lower region only. Figure 8 shows the reactor shaft configuration. It has an inner radious about 3.5 m and an outer radious of 4.8 m. It is reinforced with multiple layers of steel. The primary system inlet and outlet pipes pass through the rectangular openings of the cavity. The inner surface of the shaft is lined with carbon steel. In addition, the shaft is lined with a thermal insulation layer. The surface of the foundation of the shaft is about 36 m2 and the water volume which can be poured into the cavity is 450 m3. Water can be introduced into the shaft through the door between the hermetic compartment (reactor shaft) and non-hermetic shielding compartments. For this reason the door will be modified, 4 pipe junctions will be formed on the door where fire hoses may be connected.

The water level in the shaft may be 14.5 m, which means that it is about 2 m above the top of the core. When the water level in the cavity is more than 12 m (about 360 m3) then the water can cool the core for 3 hours after the initatiting event. The water is evaporated (about 11 kg/s) by the core decay power, thereby preventing the accident. According to the calculations the heat transfer area of the vessel is enough after 3 hours of the shutdown to asssure the core cooling. This intervention has an effect on containment pressure. The pressure slowly increases (Fig. 9.) but it does not reach the design pressure 2.5 bar until 12 hours; then the decay power decreases so that the water evaporation rate drops below 5 kg/s.

THE TMLB PROCESS IN A VVER-440 TYPE NUCLEAR POWER PLANT

Table 2. gives an overview of the key stages of the process which is triggered by the loss of normal and emergency (diesel-generated) power (station blackout) following an initial transient.

EVENT	TIME min	PRIMARY PRESSURE MPa
STM GEN. DRY.	566	12.665
CORE UNCOVER	965	14.780
START F.P. RELEASE	993	14.959
START MELT	1001	14.956
START GRID1 HEATUP	1039	14.959
CORE SLUMP	1041	14.948
GRID1 FAILED	1081	14.942
START HEAD HEATUP	1082	14.942
BOTTOM HEAD FAILED	1116	14.958
CAVITY DRY OUT	2103	
START CORE-CONCRETE INT.	2261	
END OF CALCULATION	2860	

Table 2. Events of the VVER-440 TMLB sequence

As we saw in Table 2., the operator has about 16 hours to intervene into the process of the primary circuit. Therefore this time interval makes several kinds of operator interventions possible:

A. Prevent core damage by secondary side cooling

B. Stop the accident sequence by restoring the emergency core cooling system and decreasing the pressure in the primary circuit

C. Delay core damage by primary bleed using the water of the hydroaccumulators

A.) Prevent core damage by secondary side cooling

Restoring the secondary side feedwater injection to the steam generator - meaning first of all a power supply recovery - is a main action to stop an accident sequence. If one of the auxiliary emergency feedwater pump is operable and an oparator can open one of the manual valves on the main steam line (Fig. 1.) before the water level in the vessel starts to decrease sharply at t=850 min then the core damage is prevented.

The four auxiliary feedwater pumps' energy supply may come from both units of the block. The operators of the units have to work together to start one of these pumps to prevent the accident. The water vessels are situated out side of the reactor building. The reserved water is in three 1000 m3 vessel. Two of them contain 'clean' condensed water, enough for more than two days cooling.

Fig.10 shows the primary system pressure. The secondary side cooling starts at t=845 min. As soon as it begins the pressure falls sharply to c.a. 7 MPa. The pressure is stabilized at 6.5 MPa by the cooling effect of the steam generator. The reactor is placed in cold shutdown.

If the secondary side feedwater pumps are unoperable until t=850 min, after this time the secondary cooling by feedwater has no effect, because the water level is down below the hot leg in the primary circuit.

If the feedwater pumps start earlier than 525 min - before the steam generator dries out - the loss of water from the primary circuit is retarded. In this case the reactor is put in cold shutdown at 600 min. The primary circuit remains intact in this case.

B.) Restoring the emergency core cooling system (ECC) and decreasing the pressure in the primary circuit

The common feature of these events is that we presume the restoring of the ECC (high and low pressure injection systems). When the ECC is operable from the early phase of the event without reducing primary pressure the process is identical to loss of secondary heat sink.

The operation of the ECC system in itself is unable to prevent the core melt (Fig 11.). In this case the high pressure injection system (HPIS) is working from 20 min. The pumps of the HPIS stop when the primary pressure exceeds 14 MPa therefore the pumps are able to for about only 120 min, as later the primary system pressure is higher than 14 MPa. In this process core melt is delayed about 15 min by the ECC system's operation. The main characteristics of the process of loss of secondary heat sink corresponds to the TMLB.

Therefore the primary feed is effectual only with primary bleed to reduce the primary system pressure. A long time interval is at the operator's disposal for opening the safety relief valve with the aim of stopping the accident sequence and avoiding core damage. The operator can start the primary system pressure decreasing when the high pressure pump is operable. If the pumps start within 120 min of the initating event and the operator opens the relief valve by 125 min, the reactor will be placed in cold shutdown at about 750 min.

The deadline of this action to prevent core damage is t=960 min, before the core is uncovered. After this time period the necessary bleed to reduce the pressure under 14 MPa decreases the water level below the core and for a short time period the core is not cooled. As a reasult the core temperature becomes higher than 1200°C, and the rods get damaged. About .7 percent of the zirconium cladding reacts, but the accident sequence is stopped.

According to the MARCH code calculation the operator can stop the accident sequence by the HPI and LPI systems if these sytems are restored before the bottom head failed. The operator starts the primary bleed after core slump, therefore the pressure fall below 10 MPa. The HPIS injects water to the hot debris and steam is generated. This causes a small pressure spike. When the water level becomes higher than the level of the debris the pressure decreases to about 0.6 MPa. From t=1440 min the pressure stabilizes at 1.5 MPa and the temperature of the bottom head is stabilized at 140°C.

C.) Delay core damage by primary bleed

If there is no energy supply for the pumps, only the passive systems (hydroaccumulators) can cool the core. The significance of this intervention that the water of the hydroaccummulators is activated to delay the core damage. The operator can activate the hydroaccumulators by reducing the pressure in the primary circuit below 6 MPa.

During the time interval between the start of pressure increase (t=250 min) and the time when pressure reaches the opening value of the safety relief valve (t=572 min) the intervention is effective. Out of this interval the core melt comes sooner. As the curve in Fig. 12 shows this intervention is the most succesful when the operator opens the safety relief valve at t=560 min - before the steam generator dries out - and finishes the intervention at about 620 min. This bleed delayes the core melt process. by about 2 hours In Table 3 the events of TMLB and TMLB with primary bleed (in the best time) are compared.

EVENT	TMLB	TMLB+op. intervent.
	TIME min	TIME min
STM GEN. DRY.	566	569
CORE UNCOVER	965	1113
START F.P. RELEASE	993	1142
START MELT	1001	1151
START GRID1 HEATUP	1040	1190
CORE SLUMP	1040	1190
GRID1 FAILED	1081	1234
START HEAD HEATUP	1082	1235
BOTTOM HEAD FAILED	1116	1268

Table 3. Events of TMLB sequence with operator intervention (primary bleed at t=560 min) and without it

REFERENCES

1 L. Vöröss : Status of Implementation of the Lessons Learned from the Three Mile Island Accident IAEA TM Vienna, 18-22 Oct. 1982

2 Gieseke, J.A.; Cybulskis, P.; Jordan, H. et a. : Source Term Code Package A User's Guide (Mod 1). Battelle Columbus Division, 1986 NUREG/CR-4587.

3 The STCP Application for the VVER-440 Reactors. Interim Report (1987-1988) The Source Term Code Package Workshop. TCAC RER/9/004

4 L.G. Horvàth, R. Taubner, Z. Tèchy: Severe Accident Managment for a VVER-440 Nuclear Power Plant

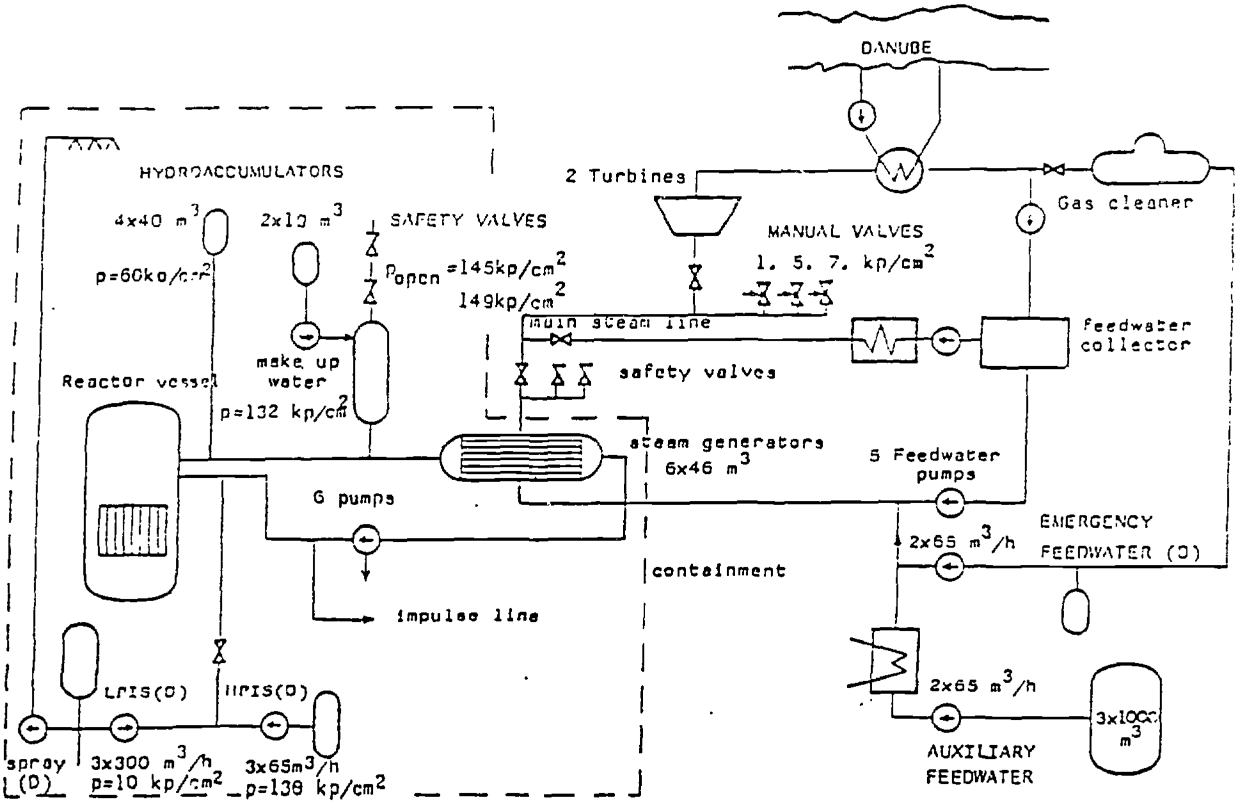


Fig. 1 Important features of one unit of Paks NPP

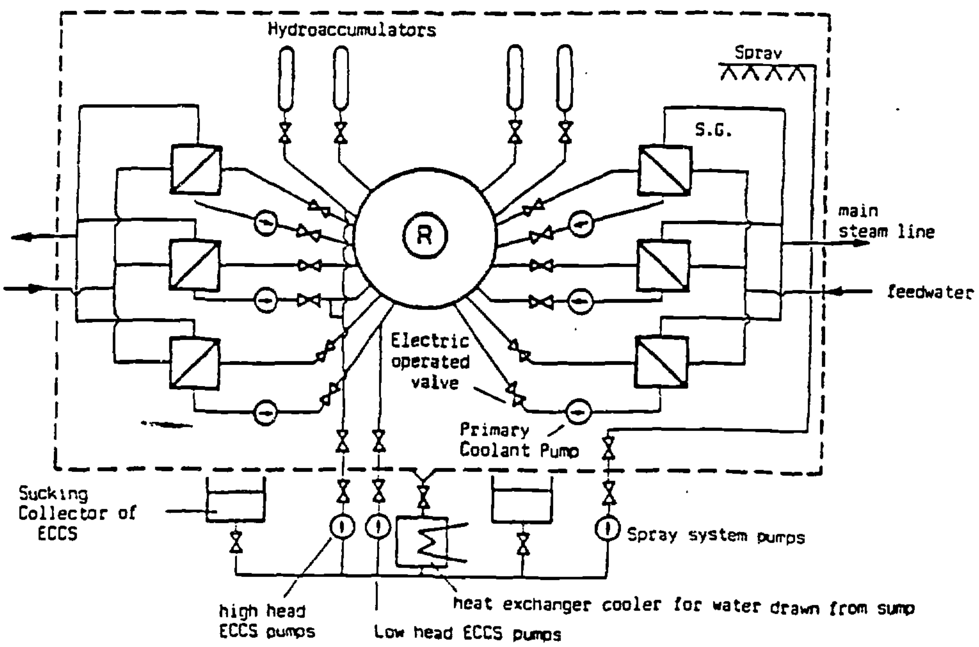
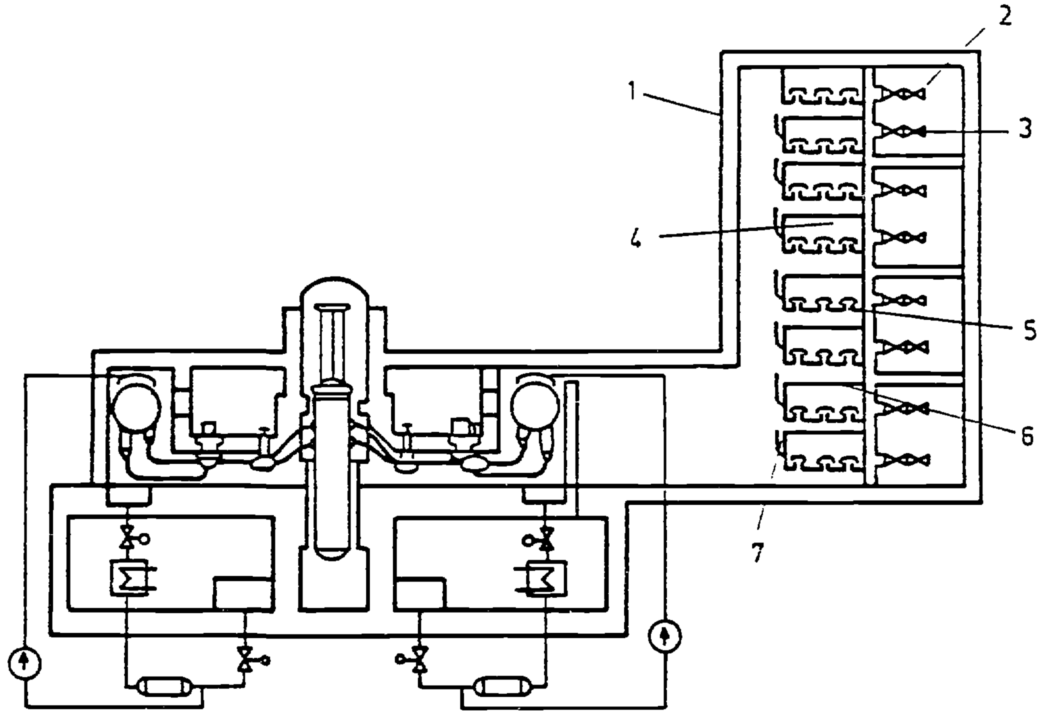


Fig. 2 Shematic diagram of primary circuit of Paks NPP



1 - reinforced concrete walls

2 - air trap volume

3 - check valve

4 - tray

5 - steam channels

6 - plenum region cover

7 - overflow discharge

Fig. 3

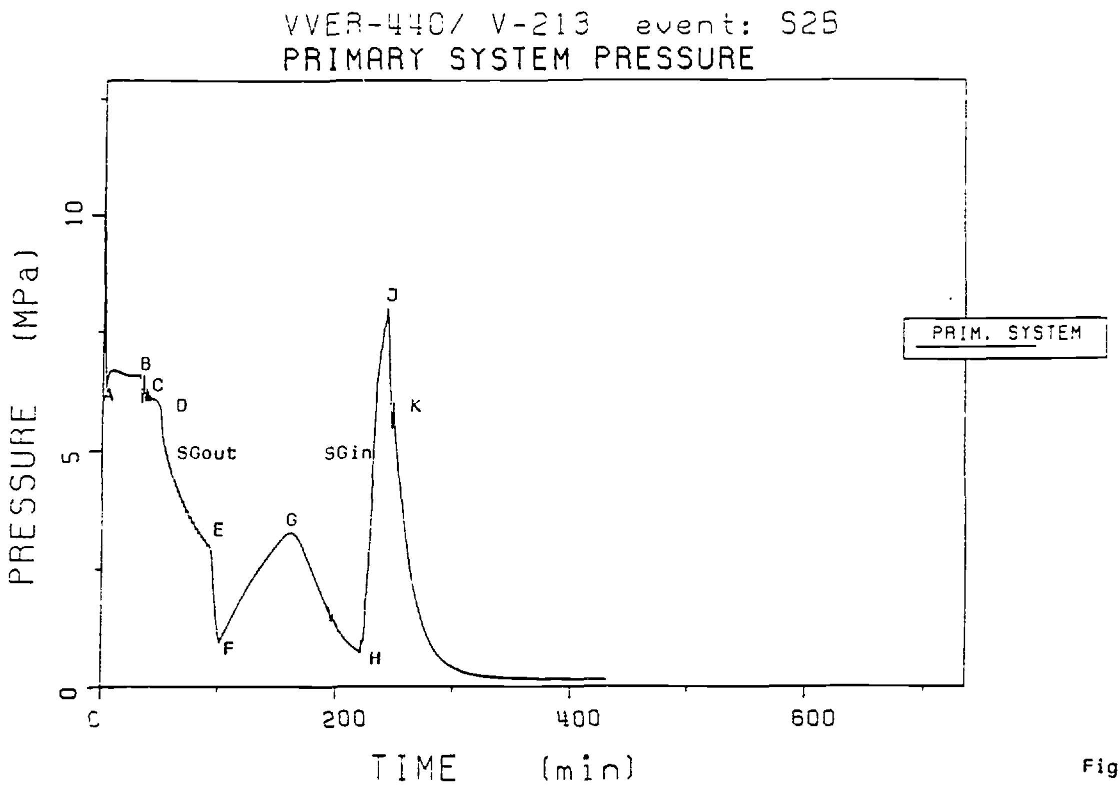

VVER-440/ V-213 event: S2B
PRIMARY SYSTEM PRESSURE
PRESSURE (MPa)
10
5
0
0
200
400
600
TIME (min)
B
C
D
SGout
E
F
G
SGin
H
J
K
PRIM. SYSTEM

Fig. 4

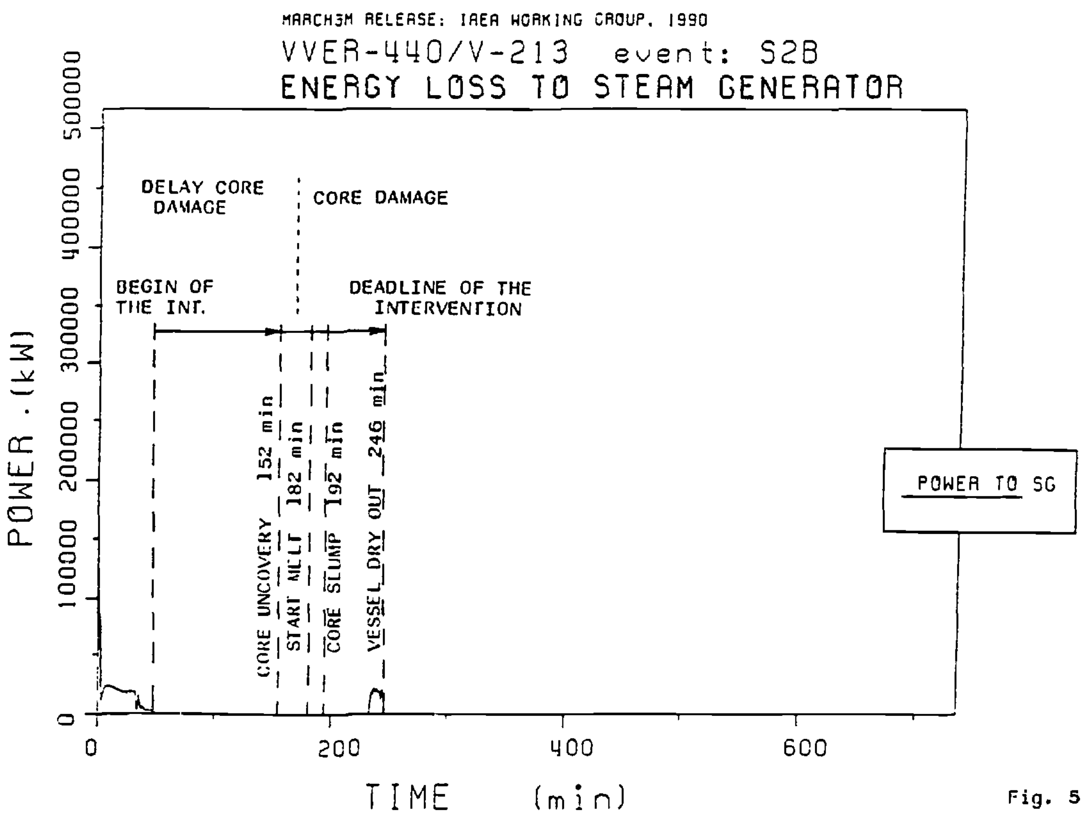

MARCH3M RELEASE: IAEA WORKING GROUP, 1990
VVER-440/V-213 event: S2B
ENERGY LOSS TO STEAM GENERATOR
POWER . (kW)
500000
400000
300000
200000
100000
0
0
200
400
600
TIME (min)
DELAY CORE DAMAGE
CORE DAMAGE
BEGIN OF THE INT.
DEADLINE OF THE INTERVENTION
CORE UNCOVERY 152 min
START MELT 182 min
CORE SLUMP 192 min
VESSEL DRY OUT 246 min
POWER TO SG
OPERATOR INTERVENTION: SECONDARY BLEED

Fig. 5

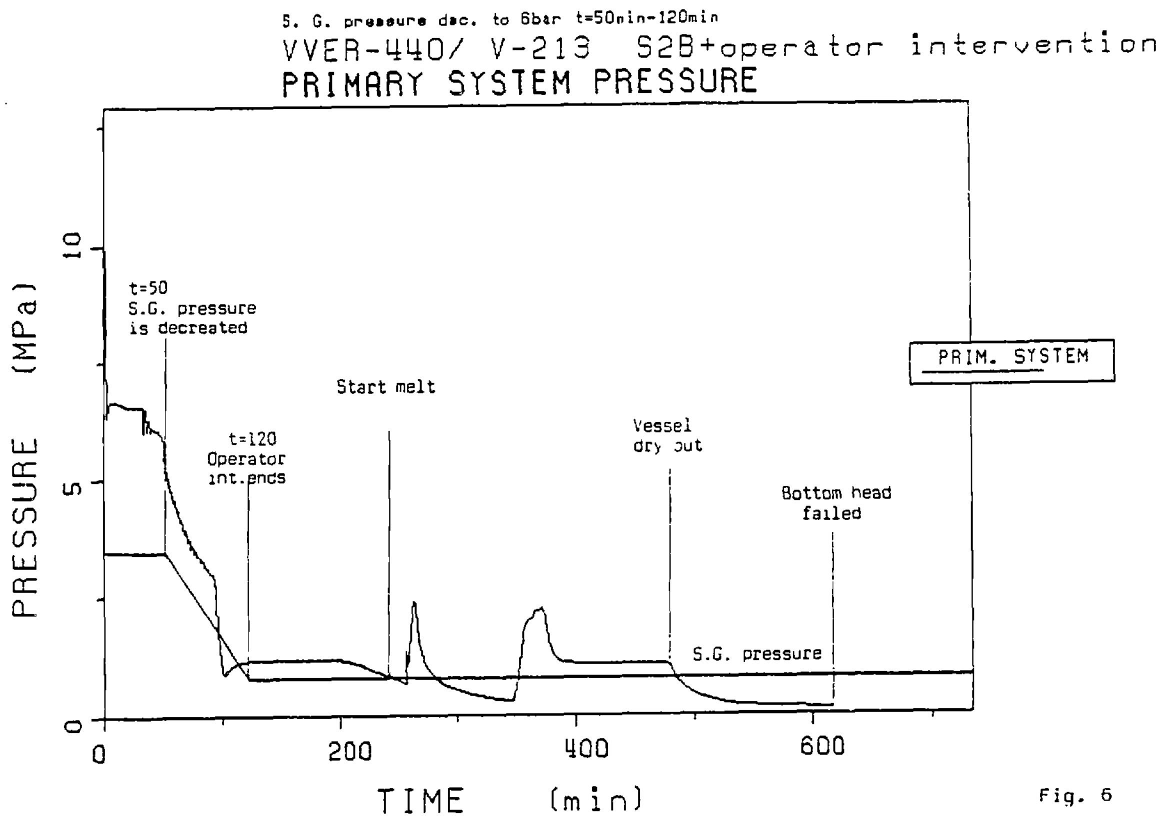

S. G. pressure dec. to 6bar t=50min-120min
VVER-440/ V-213 S2B+operator intervention
PRIMARY SYSTEM PRESSURE
PRESSURE (MPa)
0
5
10
t=50
S.G. pressure
is decreated
t=120
Operator
int.ends
Start melt
Vessel
dry out
Bottom head
failed
S.G. pressure
PRIM. SYSTEM
0
200
400
600
TIME (min)

Fig. 6

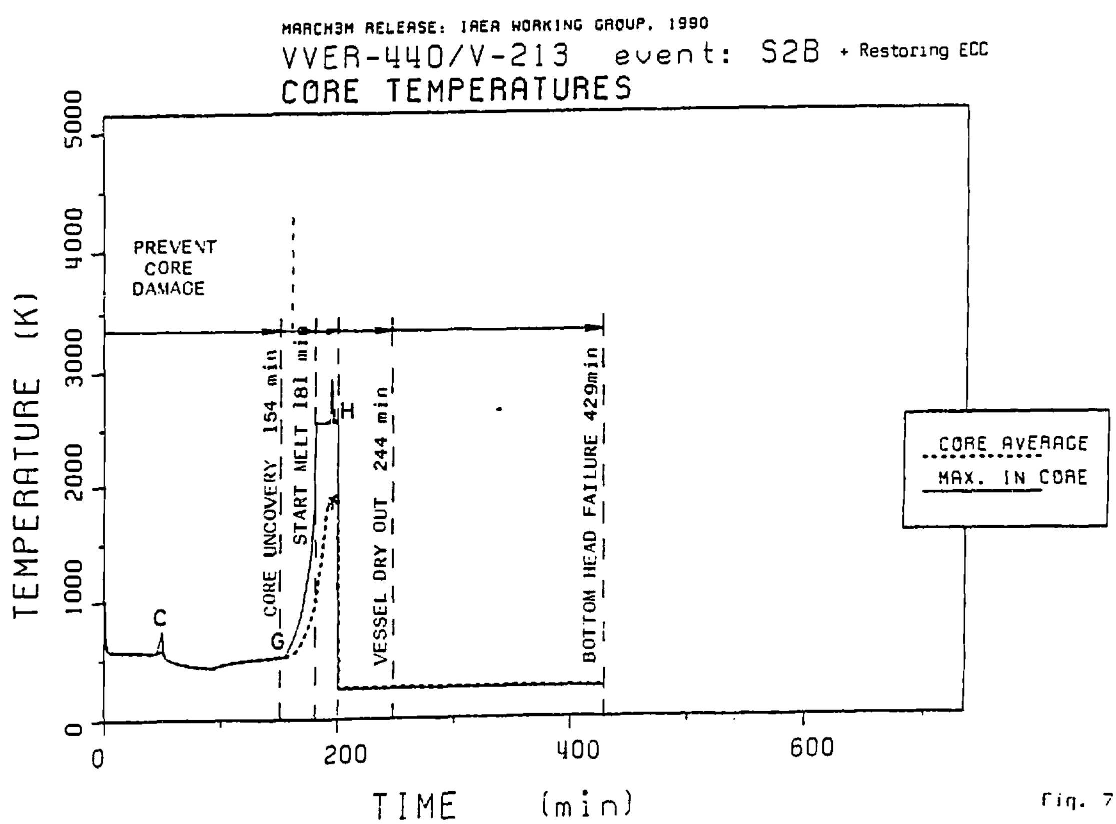

MARCH3M RELEASE: IAEA WORKING GROUP, 1990
VVER-440/V-213 event: S2B + Restoring ECC
CORE TEMPERATURES
TEMPERATURE (K)
0
1000
2000
3000
4000
5000
PREVENT
CORE
DAMAGE
C
G
H
CORE UNCOVERY 154 min
START MELT 181 min
VESSEL DRY OUT 244 min
BOTTOM HEAD FAILURE 429min
CORE AVERAGE
MAX. IN CORE
0
200
400
600
TIME (min)

Fig. 7

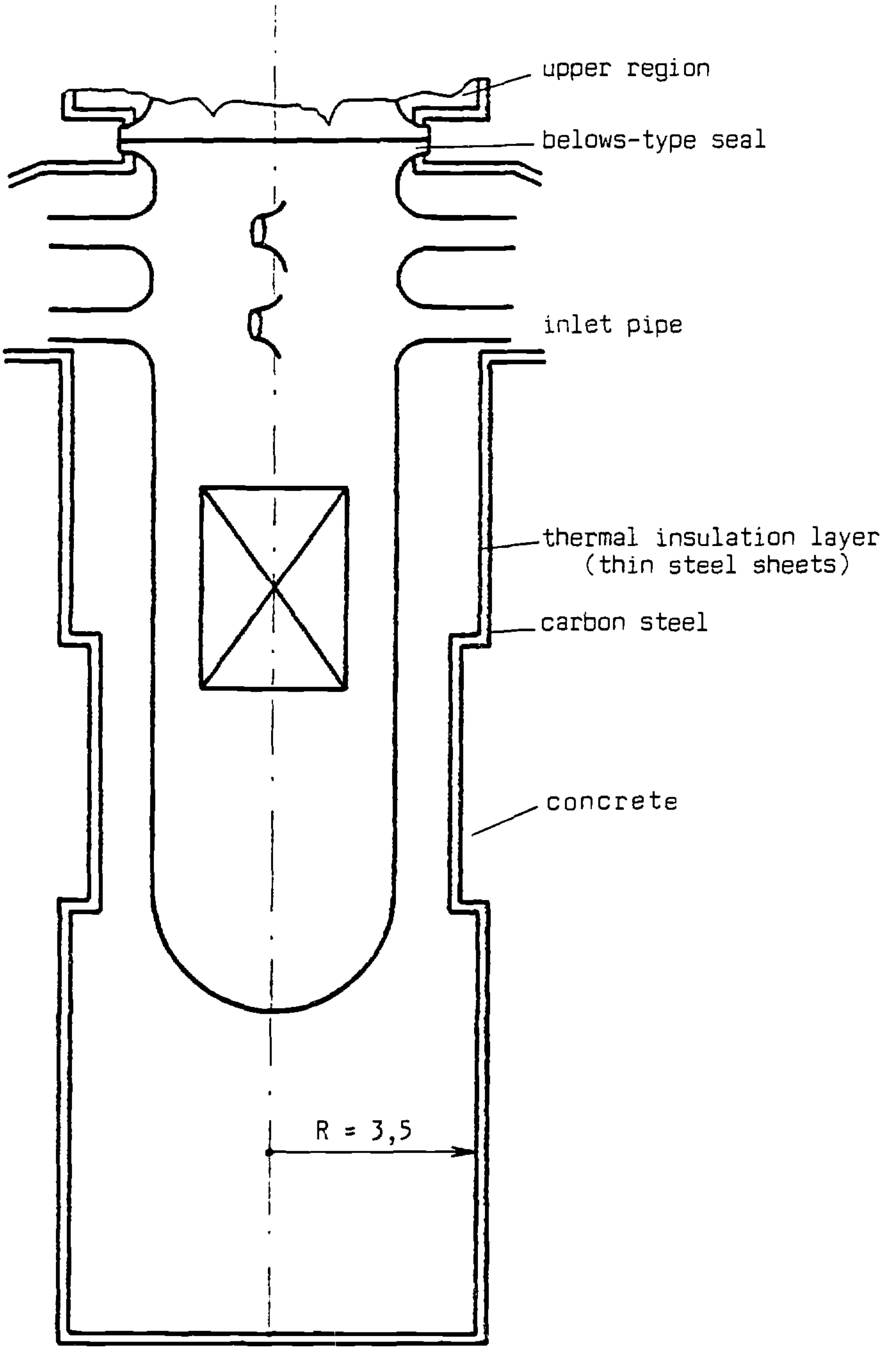


Fig. 8 Reactor shaft lower region

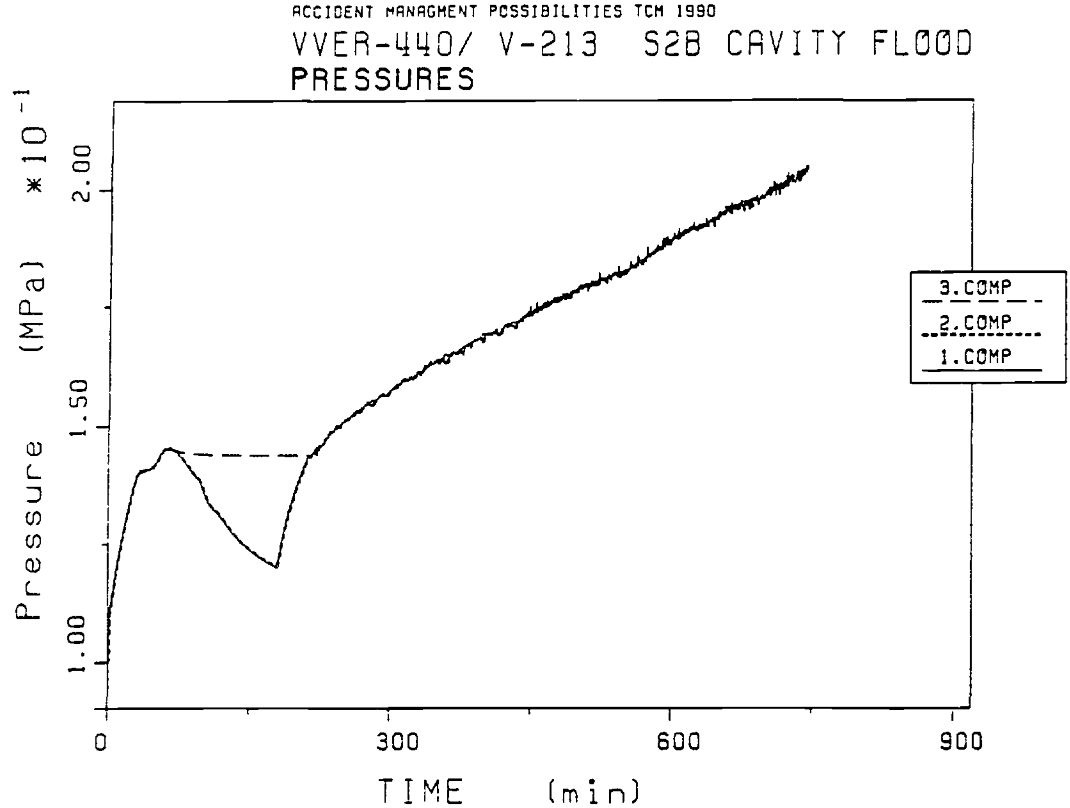

ACCIDENT MANAGMENT POSSIBILITIES TCM 1990
VVER-440/ V-213 S2B CAVITY FLOOD
PRESSURES
Pressure (MPa) *10⁻¹
1.00
1.50
2.00
0
300
600
900
TIME (min)
3.COMP
2.COMP
1.COMP

Fig. 9

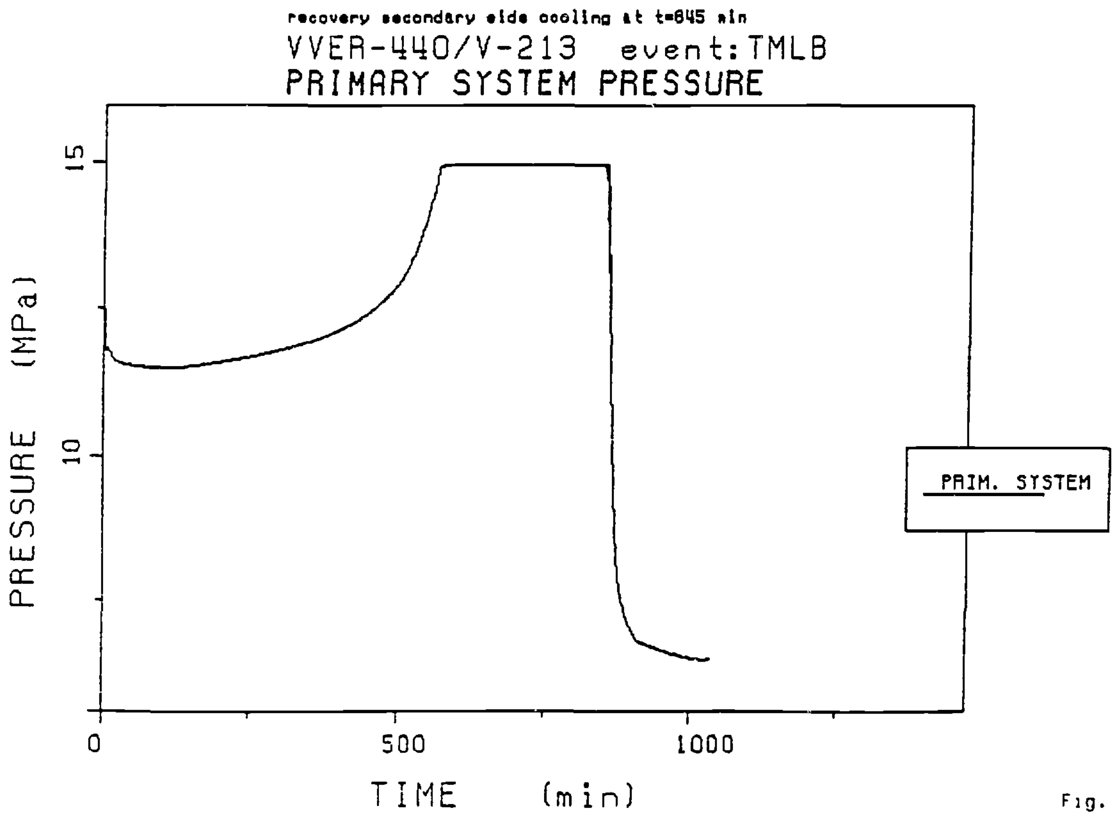

recovery secondary side cooling at t=845 min
VVER-440/V-213 event:TMLB
PRIMARY SYSTEM PRESSURE
PRESSURE (MPa)
10
15
0
500
1000
TIME (min)
PRIM. SYSTEM

Fig. 10

MARCH3M RELEASE: IAEA WORKING GROUP, 1990

VVER-440/V-213 event:Loss of sec.heat si

CORE TEMPERATURES

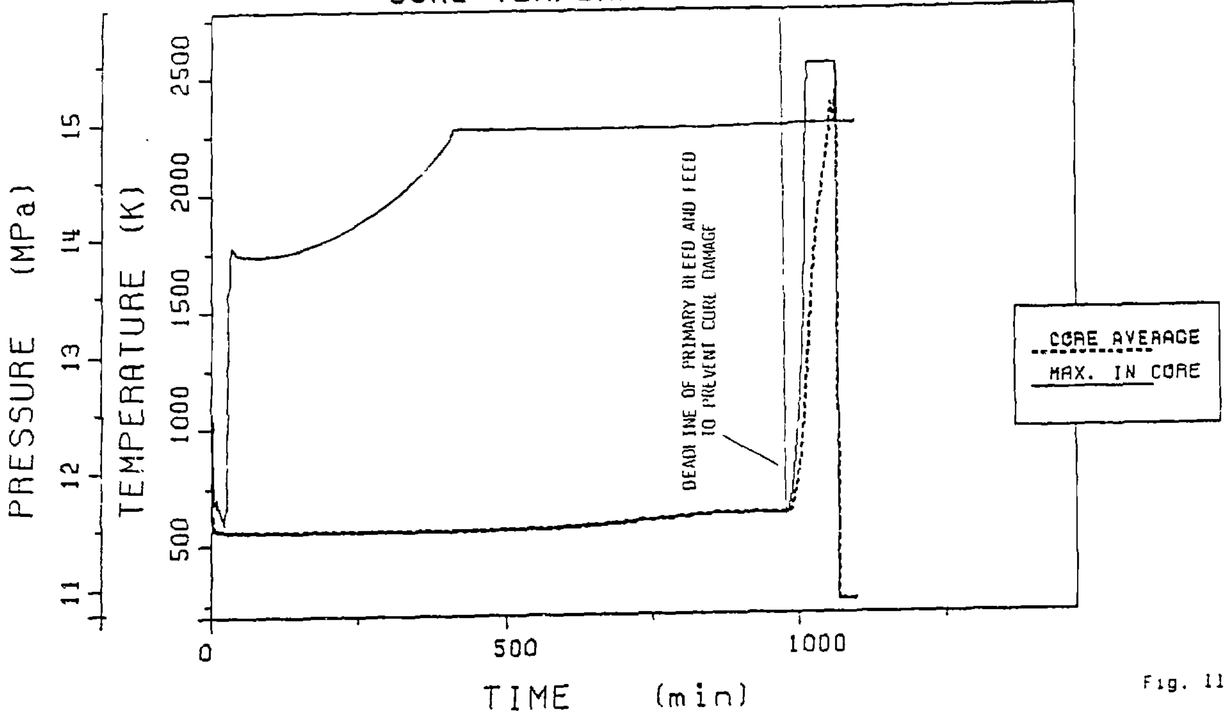


Fig. 11

Primary bleed from t=560 min to t=620 min

VVER-440/V-213 event:TMLB

CORE TEMPERATURES

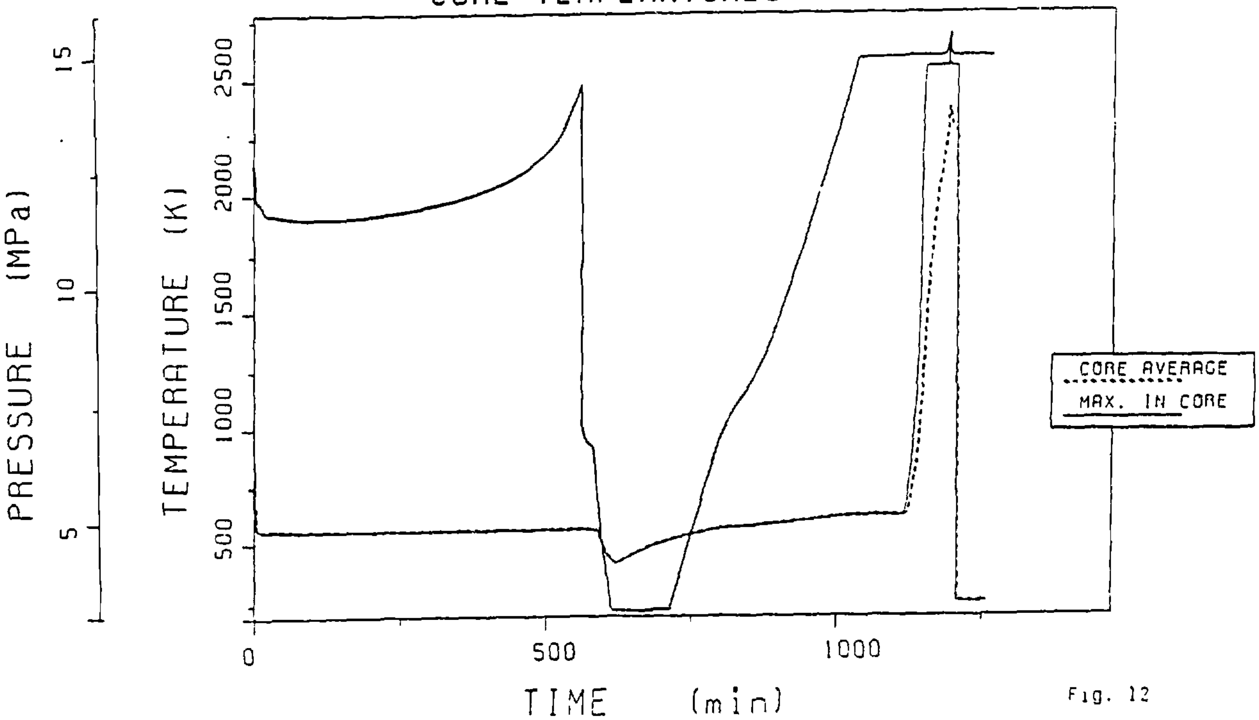


Fig. 12

TECHNICAL COMMITTEE MEETING ON PLANT SYSTEM UTILIZATION
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UTILIZATION OF CONTROL ROD DRIVE (CRD) SYSTEM FOR LONG TERM CORE COOLING

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ABSTRACT

In this paper we consider an application of Probabilistic Risk Assessment (PRA) to risk management. Foreseeable risk management strategies to prevent core damage are constrained by the availability of first line systems as well as support systems. The actual trend in the evaluation of risk management options can be performed in a number of ways. An example is the identification of back-up systems which could be used to perform the same safety functions.

In this work we deal with the evaluation of the feasibility, for BWR's, to use the Control Rod Drive system to mantain an adequate reactor core long term cooling in some accident sequences. This preliminary evaluation is carried out as a part of the Internal Events Analysis for Laguna Verde Nuclear Power Plant (LVNPP) that is currently under way by the Mexican Nuclear Regulatory Body. This analysis adresses the evaluation and incorporation of all the systems, included the safety related and the back-up non safety related systems, that are available for the operator in order to prevent core damage. As a part of this analysis the containment venting capability is also evaluated as a back-up of the containment heat removal function. This will prevent the primary containment overpressurization and loss of certain core cooling systems.

A selection of accident sequences in which the Control Rod Drive system could be used to mitigate the accident and prevent core damage are discussed. A personal computer transient analysis code is used to carry out thermohydraulic simulations in order to evaluate the Control Rod Drive system performance, the corresponding results are presented. Finally, some preliminary conclusions are drawn.

1.- INTRODUCTION.

In this work we consider the potential uses of Probabilistic Risk Assessment (PRA) in risk management programs. The methods and results produced in PRA studies provide a framework within which current risk management strategies can be evaluated, and future risk management programs can be developed and assessed.

Risk management can be divided into five separated, but related, phases [1]. Prevention of accident initiators, Prevention of core damage, Implementation of an effective emergency responses, Prevention of vessel breach and mitigation of radionuclide releases from the reactor coolant system, and Retention of fission products in the containment and other surrounding buildings.

We used a risk based methodology for identifying and evaluating risk management options for the above second phase, the prevention of core damage. We present the evaluation of the feasibility, for BWRs, to use the Control Rod Drive system to mantain and adequate reactor core long term cooling in some accidents and thereby reduce the probability and consequences of severe accidents. Based on past PRAs for BWRs and in the Internal Events Analysis for the Mexican Nuclear Power Plant [2], that is currently under way by the Mexican Nuclear Regulatory Body, we identify and discuss the important accident sequences, hardware failure and human errors within these sequences that impact most the total core damage frequency. The Internal Event analysis adresses the evaluation and incorporation of all the systems, included the safety related and the back-up non safety related, that are available for the operator in order to prevent core damage. The containment venting capability is also evaluated as a back-up of containment heat removal function.

A brief description of the system along with the identification and discussion of the accident sequences in which the Control Rod Drive system could be used as well as the methods used to perform this evaluation are presented in section 2. Finally, in section 3 we provide some results of thermohydraulic simulations as well as some preliminary conclusions.

2.- EVALUATION OF CURRENT RISK MANAGEMENT PRACTICES.

Severe reactor accidents involve extremely complex systems and phenomenological responses that are often nonintuitive. When developing and evaluating risk management strategies it is important to understand how a particular action may affect other portions of the accident progression [3]. The PRA methods provide an integrated analysis framework that can evaluate the potential ramifications of a specific action over a wide range of possible outcomes. All five phases of risk management described above can be included in a such integrated analysis, and different options can be compared using various risk measures, including health and economic risk. Even though a full scope PRA is not available for the plant being analyzed, some phases of risk management can be assessed with a PRA level 1 or PRA level 2. For example, with a PRA level 1, as the one that we are developing at the Mexican Nuclear Regulatory Body, is feasible to assess some phases of risk management like prevention of accident initiators and prevention of core damage.

As discussed before, a PRA framework is effective in evaluating the efficacy of current risk management practices at a nuclear power plant. These practices include hardware improvements already made and improvements in operating procedures. In principle, the methods used to evaluate current practices are straightforward. In order to determine the worth of risk management practices the total core damage frequency/risk can simply be calculated with and without a particular practice in place. However, in practice this can be a complex process, if removing an option changes the fundamental models in PRA. For phases 1 and 2 mentioned above, changes to basic event data or operator recovery actions are easy to evaluate, while changes to fault/event trees require more effort.

There are numerous improvements that have ocurred throughout the nuclear industry since Three Mile Island in the form of changing regulations and industry initiatives. Probably the most risk significant changes have ocurred in the areas of operating procedures and operating training. All the plants have, to some extent, incorporated symptom based emergency operating procedures in stead of event based procedures. Since events are defined in terms of symptoms, changes to symptoms are less important than the flexibility in operator responses now possible because an event no longer needs to be precisely defined. Besides, the operator have been trained more rigorously in the use of these procedures. These changes have been enhanced by the development of improved Safety Parameter Display Systems that help to reduce the confusion that was present during the TMI accident. Overall, these changes have resulted in PRA predictions of significantly lower human error probabilities. Additionaly, due to improved training and procedures for operators, PRA now considers additional operator recovery actions. The use of alternative injection systems and other actions are now explicitly included in many emergency procedures.

One of the back-up systems identified as an alternative injection system is the Control Rod Drive system which can be used as a long term core cooling system in sequences involving loss of containment heat removal or station blackout sequences.

Description of CRD System.

The purpose of CRD is to provide the hydraulic force to insert, withdraw and rapidly insert (SCRAM) control rods in response to signals from the Reactor Manual Control System and Reactor Protection System [4,5]. This is accomplished by pumping demineralized water from the Condensate system at the necessary pressure to the CRD mechanisms via the Hydraulic Control Units (HCUs). Valving in the HCU directs the high pressure fluid to reposition any control rod to a desired location. However, the CRD system is modeled as a back-up source of high pressure injection and its utilization is included in the emergency procedures for Laguna Verde Nuclear Power Plant. The CRD pumps take suction from the Condensate system through a manual isolation valve. An alternative supply from the Condensate Storage Tank (CST) is also directed to the combined suction header through a check valve. A simplified schematic of the CRD system is provided by figure 1. Flow enters the CRD hydraulic system at the pump suction filter. Flow from the suction filter enter the CRD pumps through a combined suction line. One of the 100% capacity, centrifugal, 10 stage CRD pumps is normally in operation to supply the required flow at the required pressure. When the system is started or after a SCRAM, the maximum nominal flow from the pump is 200 gpm at a nominal pump discharge pressure of 1160 psig. Each pump is provided with a minimum flow line which recirculates 20 gpm back to the CST. This minimum line prevents the pump flow from decreasing to where the pump would overheat and possibly be damaged.

Two discharge paths are provided for the CRD pumps. The first path is through the Hydraulic Control Units (HCUs) cooling header. Flow is controlled by one of two air operated control valves. When containment instrument air is lost, the control valves fail closed, thus blocking this path. The second path is through the HCU charging headers. This path is upstream of control valves and fails open on loss of air. However, with both CRD pumps running and the reactor at nominal pressure, the second discharge path restricts flow, by means of an orifice, to approximately 200 gpm. This flow rate is assumed insufficient for core cooling in the early stage of the accident and thus no credit is taken for this discharged path in the enhanced operation mode of the CRD system. However, when coolant makeup has been provided for a period of time and then lost, the CRD success criteria require only one CRD pump running and one of the two discharge paths available.

CRD pumps are powered from 4160 volts buses 1A and 1C. Upon total loss of offsite power the CRD pumps stop and will not automatically restart when the emergency diesel generators are loaded. A simplified CRD dependency diagram is provided by figure 2. The major dependencies are indicated by solid diamonds. The CRD pumps receive no automatic initiation signal. Instrument air is required for the operation of flow control valves. The CRD pumps are cooled by the Nuclear Closed Cooling Water system.

Accident Sequences.

PRAs for BWRs have indicated that accidents initiated by transients rather than loss-of-coolant accidents (LOCAs) dominated the total core damaged frequency (CDF) estimates. However, there appeared to be no consistent pattern of relative ranking of transient sequences among the PRAs reported. It is also important to observe that for a given accident sequence, contributors to differences in quantitative results between the PRAs included subjective modeling assumptions, plant differences as well as data differences.

In the Reactor Safety Study (RSS) [6], which used the Peach Bottom Plant, and the Interim Reliability Evaluation Program Study (IREP) [7], which used the Browns Ferry Plant, loss of containment heat removal sequences (TW) were found to be important contributors to core melt. The more recent Accident Sequences Evaluation Program (ASEP) [8] have reduced the CDF, attributable to these sequences, based on operating procedures that now include venting and alternative injection. For Browns Ferry IREP, accident sequences with failure of high pressure injection system were important contributors to CDF. Most of these contributors were because of high failure rate for the Automatic Depressurization System (ADS).

In accident sequences initiated by transients in which a subsequent loss of containment heat removal occurred, the safety systems that were providing coolant makeup to the reactor would fail as a result of harsh enviroment conditions generated at the primary containment. Simplified sequences, involving loss of containment heat removal, for Laguna Verde Nuclear Power Plant are presented in figure 3. These sequences are initiated by a transient that demands the operation of the Reactor Protection System to achieve subcriticality. After the reactor is shutdown the overpressure protection function should be accomplished through the Safety/Relief valves (SRV's) opening and reclosing. The next safety function that should be fulfil is the emergency core cooling. There are two safety systems at Laguna Verde that can provide high pressure injection. The High Pressure Core Spray system (HPCS) and the Reactor Core Isolation Cooling system (RCIC). The suction of these two systems is from the Condensate Storage Tank (CST) or the suppression pool. In the early stage of the accident, the suction is automatically or manually switched to the suppression pool in order to establish a closed cooling loop during the progression of the accident. The switching back of the suction to the CST is not included in the emergency procedures. The HPCS motor driven pump and RCIC turbine driven pump components could lose their integrity and fail if very high temperature water is circulated through the pumps. This situation is only possible if the pool temperature is increasing, and this condition is met when containment heat removal is lost, which occurs during station blackout sequences or sequences in which random failures of containment heat removal systems occurs. In this sequences, in which the high pressure systems fail and the low pressure systems would also fail by the same reason, the operator could realign the CRD suction but now from the CST in order to prolong the core cooling function and prevent core damage, thus reducing the probability and consequences of severe accidents.

In order to assess if the CRD system has the capability to mantain the reactor water level, a personal computer transient analysis code PCTRANB [9] was used to carry out a thermohydraulic simulation. The simulation started with the closure of Main Steam Isolation Valves. The reactor water level was mantained solely by RCIC system with the operator controlling reactor vessel water level between 500" and 540" as shown in figure 4. The scenario proceeded in this manner until the suppression pool heat capacity limit was reached at 45628 sec. (12.67 hrs). At that point the reactor vessel was manually depressurized to a range of 200 psig as indicated in figure 5. The RCIC system pump continued to inject until the suppression pool reached 250 ^{o}F at 103092 sec (28.63 hrs). The operator started a single CRD pump, and controlled the reactor vessel water level between 550" and 500". As shown in figure 4, the CRD pump was able to mantain the collapsed water level even in the case that the SRV's were forced to close by high containment atmosphere pressure.

3.- RESULTS AND CONCLUSIONS.

The purpose of risk management programs is to reduce the public health risk and provide additional capability for reducing the probability and consequences of severe accidents. This work presented a general approach for using PRA-type analysis to evaluate current risk management practices in order to prevent core damage. This advanced PRA technology allows the in depth, integrated treatment of all phases of a severe accident, although this was not attempted in this work.

Even though, the Internal Events Analysis for Laguna Verde Nuclear Power Plant is not concluded yet, we can anticipate the effectiveness of current risk practices. These risk management options involve the use of alternative injection systems like CRD, Firewater system, Nuclear Service Water/LPCI crosstie, as well as the posible venting of the primary containment in some accident sequences.

The results of the thermohydraulic simulation presented in figure 4 and 5 show that the CRD is capable to mantain the reactor water level and provide long term cooling if some other injection system has been operating successfully for 6 or more hours following an initiator. We conclude that the utilization of these kind of back-up systems as an alternative way of makeup coolant to the reactor could provide the operator with 10 or more hours in order to recover some containment heat removal system and thereby reduce the probability and consequences of severe accidents.

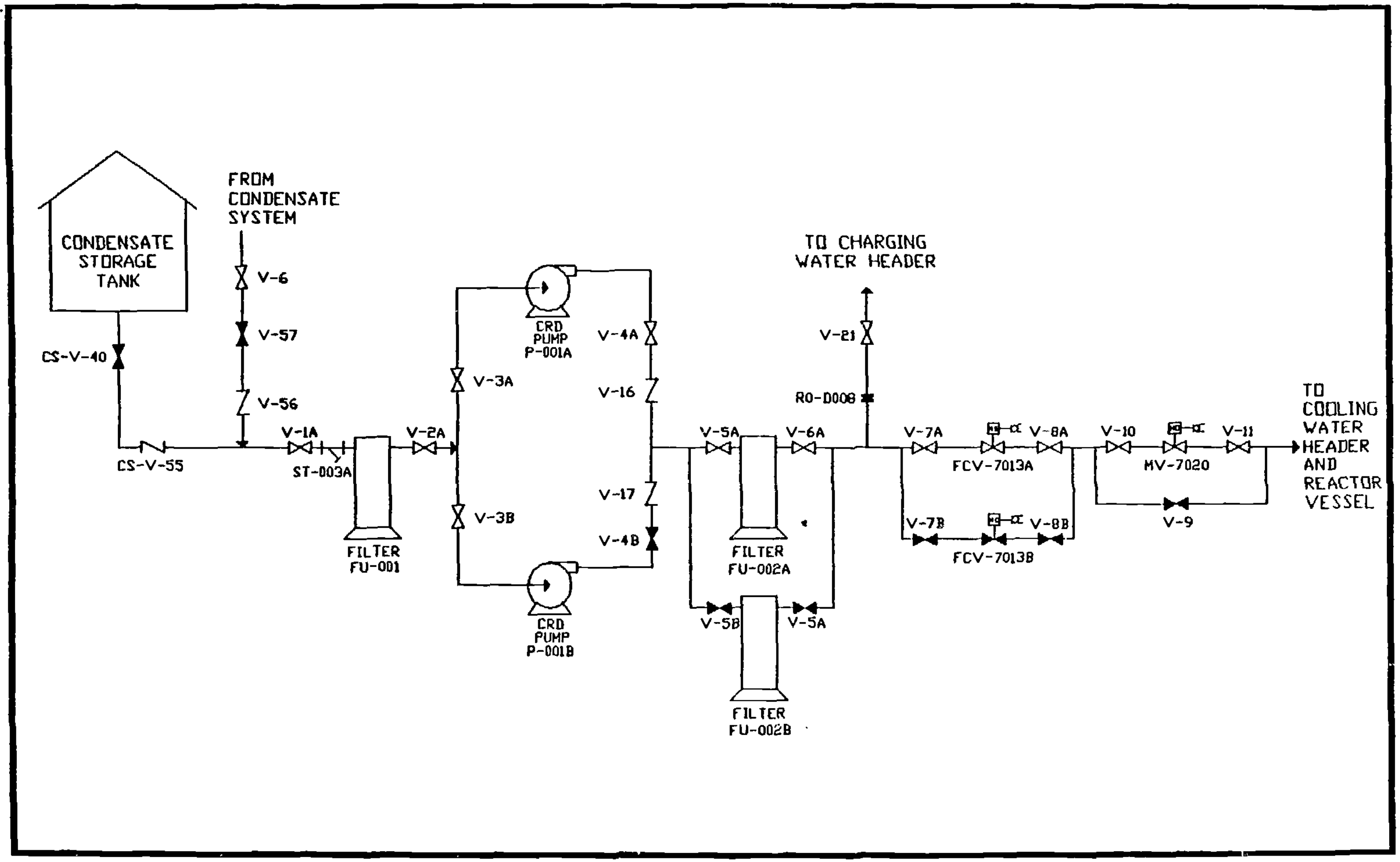


FIGURE 1.- CRD SYSTEM SCHEMATIC

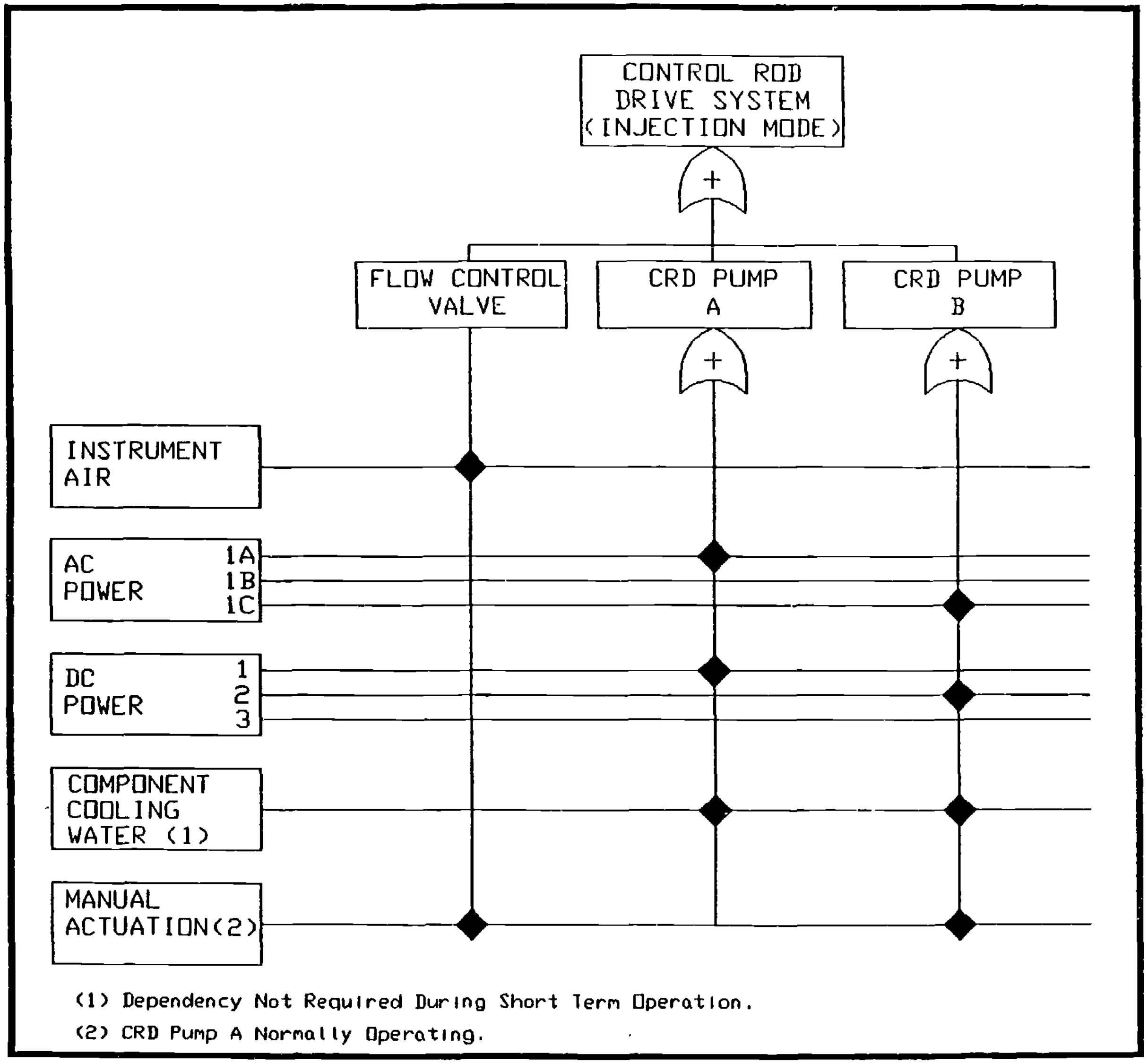


FIGURE 2.- CRD DEPENDENCY DIAGRAM

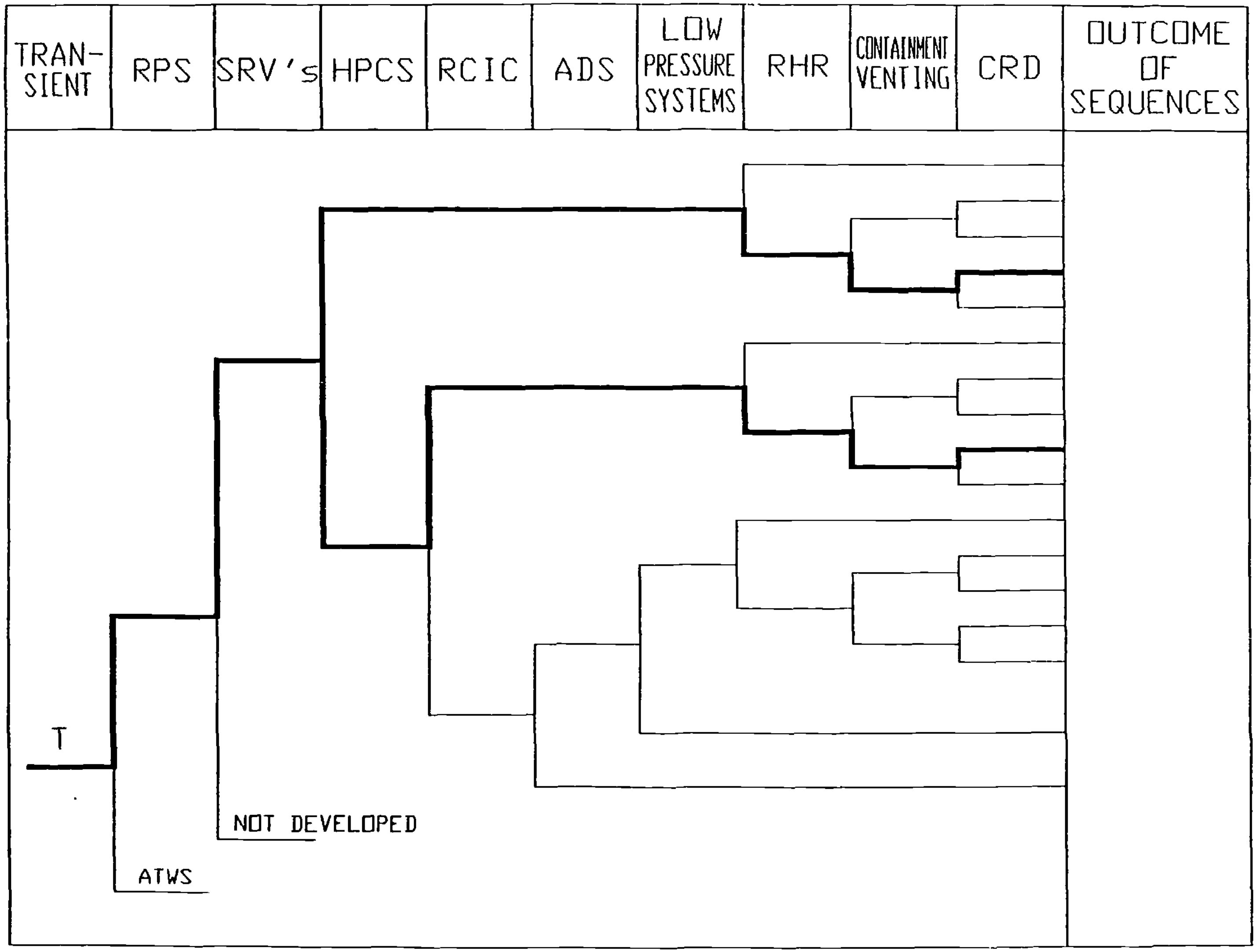
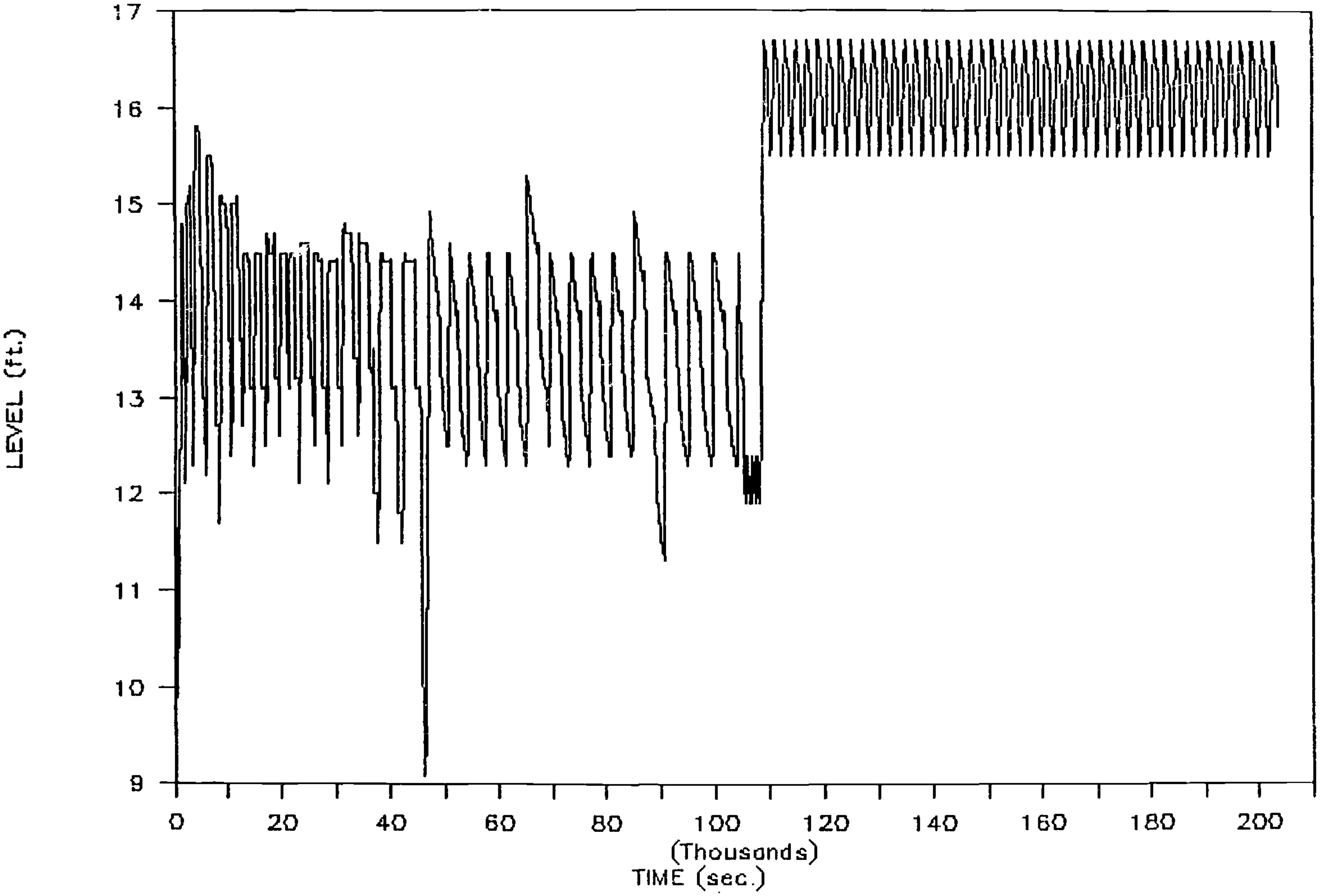
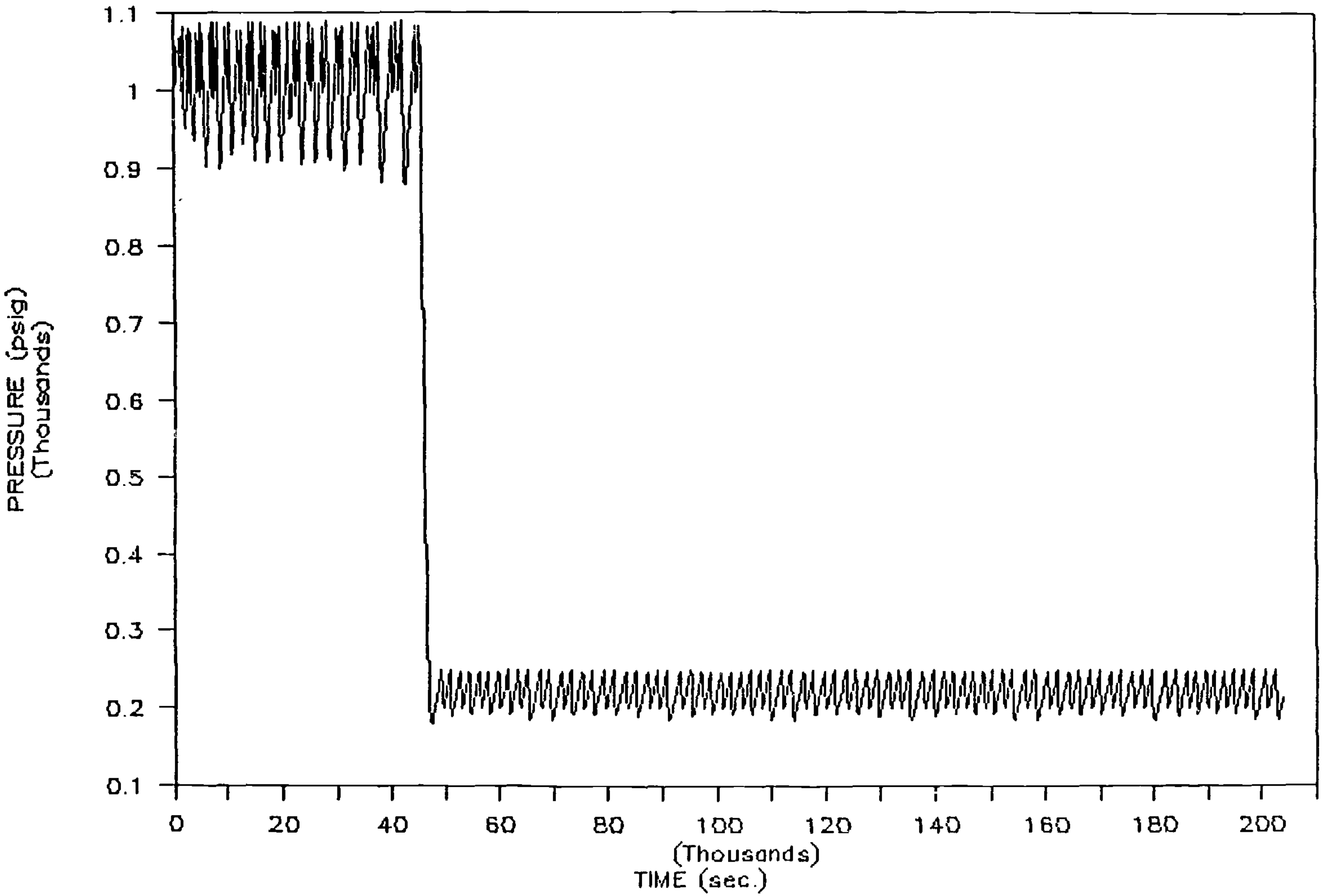


FIGURE 3.- LOSS OF CONTAINMENT HEAT REMOVAL SEQUENCES

LAGUNA VERDE TW SEQUENCE CALCULATION



LAGUNA VERDE TW SEQUENCE CALCULATION



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DESIGN OBJECTIVES ON FILTERED VENTING AND HYDROGEN MANAGEMENT
FOR THE BORSSELE NUCLEAR POWER-STATION

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ABSTRACT

The Chernobyl accident has led to a general review of nuclear energy in the Netherlands including the safety of the nuclear power stations.
The report "Beratungsstudie zu Accident Management Massnahmen für die KKW Borssele und Dodewaard" prepared by GRS has provided the basis of the Netherlands position on mitigation measures.
Two recommendations from the GRS report will be discussed in this paper: Filtered venting and hydrogen management for the Borssele nuclear power-station.
The Dutch Government requires assurance that, by means of both preventative and mitigating measures, the probability for a release should be small and that in case of a severe accident, discharges to the environment should be kept to a minimum. Such discharges should if possible be kept so low that:

a. no short term external emergency actions are required, and
b. no large scale contamination of land and surface waters occurs.

Hydrogen
Bearing in mind the results of the LOFT-FP and CORA experiments and the evaluation of TMI, the concern is that during core melt accidents the volume ratios of hydrogen, air and steam in localised areas of the Borssele Nuclear Power Station are such that explosions are possible, and counter measures should be taken. One of these is the deliberate burning of hydrogen, although it should be verified that deflagration loads are within acceptable limits and, in addition, that a deflagration never proceeds into a detonation.
Therefore, a number of other options are studied, including addition of CO_2 into the containment atmosphere, both pre- and post-accident.

Filtered venting is required to protect the containment against over-pressure.
From an evaluation of possible sequences it can not be excluded that such venting may be required when a considerable amount of airborne fission products is present in the containment, so that effective filters are necessary. In addition, possible leakages from the con-

tainment beyond technical specification limits may make it preferable to reduce the pressure in the containment already at an earlier stage. This may also be necessary if melt through of the foundation is at all possible. As the release of iodine is relevant for early external measures, filtering of elementary as well as organic iodine is desirable.

The issue of containment pressure relief is coupled with the hydrogen management, if controlled combustion or inertisation techniques are used.

The option of CO_2-addition will influence the performance of scrubber type filters, in particular for iodine. Venting during the blowdown phase of the accident is also considered.

1. INTRODUCTION

The Chernobyl accident has led to a Nuclear Energy review project [1]. One of the subjects of the project was the safety of the two Nuclear Power Plants (NPP) in operation in the Netherlands. For Borssele NPP (480 MWe, KWU-type) the German Gesellschaft für Reaktorsicherheit (GRS) carried out a study on the behaviour of the containment system in case of a serious accident [2]. Among others GRS advised mitigating measures for hydrogen management and filtered venting. By the end of 1988 the Dutch Regulatory body issued a letter [3] that most of the GRS-recommendations, in particular hydrogen management and filtered venting, should be implemented in the future.

In the first instance the utility itself is responsible for accident management. The Government has requested ECN and Professor Karwat from the Technical University München to advise how to judge the proposals of the Borssele staff. This advice has been reported in [4] and [5].

2. GOVERNMENTAL POSITION

In the letter in which the Dutch Nuclear Inspectorate announced the governmental positions, an overall requirement was made which stated that:
In the case of a severe accident, discharges to the environment shall be limited. If discharges to the environment are unavoidable, they shall be kept so low that:

a. no short term external emergency actions are required;
b. no large scale contamination of land and surface waters will occur.

For general accessible areas the formulated Dutch risk management policy [6] defines an acceptable level of individual risk as 10^{-6}/year for a single source and of 10^{-5}/year for all sources (man-made sources of radiation). With a risk factor of 2.5 % per sievert (death rate) this corresponds to doses of 0.04 mSv/year and 0.4 mSv/year, respectively.
However, these values correspond with "suitable for normal living" in

general accessible areas but are not applicable to post-accident conditions. For the time being, and in the framework of assessing the filter performance for the Borssele NPP a reference value of 0.4 mSv/year will be used for the definition of "suitable for general public access" in the context of large scale land contamination.

3. THE USE OF COMPUTER CODES

In order to develop accident management procedures for a severe accident it is important to analyze possible process sequences. For this purpose computer codes are useful tools.
The first objective for accident management is to prevent core damage. A number of computer codes such as RELAP, TRAC and CONTAIN in the US, DRUFAN and RALOC in Germany and CATHARE in France have been developed. These codes have been evaluated on a large number of experimental results. Frequently these assessments have been performed in an international co-operation, bilaterally or in the framework of CSNI and other international organisations. The present status is that a well trained analyst familiar with the limitations of the codes is capable of deriving reasonable results.

If an accident progresses into core melting, accident management should be focussed on preventing damage to the containment and limitation of releases if they can not be prevented. The processes taking place during this phase of the accident are very complex and difficult to model. As long as the core configuration is maintained the codes may give some guide as to how the accident proceeds, but if the core disintegrates the codes become inadequate. In addition, small variations in operator handling may have dramatic consequences for the ongoing accident sequence. Therefore one can not rely on the results of calculations for accident management.
Experimental research on core disintegration, Molten Corium Concrete Interaction (MCCI), hydrogen behaviour, direct heating, structural analyses etc. has been performed and important and sometimes unexpected results have been obtained. This research has led to some understanding of the processes taking place. The computer codes play an important role in the process of learning, but they are not adequate for the prediction of a process sequence.

Accident management under core melt conditions should be based on understanding. Sensitivity studies performed by computer codes play an important role in this process. In order to develop the process specifications for mitigating measures, an envelope should be determined which covers most of the possible conditions.

4. HYDROGEN MANAGEMENT

The German Risk study [7] and the GRS report [2] indicate that incorrect handling of hydrogen might endanger the containment. For hydrogen management it is important to know the total quantity, the mechanism of generation, and the release pattern in the containment. In addition the process of hydrogen burning, deflagration and detonation should be understood.

4.1. Generation of hydrogen

Accident management under conditions of core disintegration must be based on knowledge obtained by analyzing the results of experiments and postulations of events which could occur in a reactor and in the containment. This requieres support by sensitivity studies on reactor conditions. In the ECN report [4] several scenarios have been analyzed.
The source of hydrogen is the in-vessel oxidation of zirconium and construction steel and the ex-vessel production due to oxidation of the remaining zirconium and the iron bars in the concrete. It is important to have knowledge of the total amount of hydrogen produced and the rate of production. Highly localised concentrations should be avoided.

If the core melts under low pressure the limited amount of steam in the vessel limits the production of hydrogen. Under high pressure conditions the amount of hydrogen produced will be considerable, especially when a natural circulation flow develops in the primary system. In addition if corium, with a high metal concentration, drops in a pool of water, the hot core material will be quenched by the water and excessive hydrogen production will result.

Metal not oxidized in-vessel will be oxidized during the core concrete interaction. For Borssele it can be expected that if a considerable quantity of corium is dropped in the cavity, the interaction can not be stopped. Apart from the metal components in the corium the iron bars in the concrete will also contribute to the production of hydrogen.
In connection with accident management the timing of the hydrogen production is very important. This timing depends on the accident sequence. Minor changes in operator actions will have a considerable influence on these rates. It is very difficult to model the hydrogen production under conditions in which the core starts to degrade.
In current computer codes, some important processes are not modelled which raises doubts on the reliability of the calculations.

A severe sequence in which significant quantities of hydrogen are released is described in the ECN filter report [4]. The core melts under a high or moderate pressure (2-6 MPa) and a natural circulation develops in the primary system. During this period, a considerable amount of hydrogen is produced which is partially transported to the containment. The primary system will also be heated. If under such a condition a high pressure pump starts or a discharge of an accumulator occurs, a considerable amount of water will be quenched onto a very hot core. As the LOFT-FP-2 [8], the CORA-experiments [9] and the TMI-sequence indicate [10], this will result in severe oxidation in the core, and the pressure in the system will rise. Due to the increase in pressure the heated surge line or another section of the primary system may fail, which results in a cloud of a hydrogen enriched medium being blown into the containment.
It should be noted that accident management should be designed to avoid such a sequence, but because such a severe case can not be excluded, further consideration is necessary.

4.2. Hydrogen in the containment

For the Borssele NPP a 100% oxidation of zircalloy inventory gives a hydrogen production of about 440 kg. When completely mixed this amounts to 12 volume % under atmospheric conditions in the Borssele containment. In the high pressure scenario described above with ad-

ditional failure of the primary system, hydrogen is transported to the containment along two pathways, a low continuous rate via the safety relief valve and the cloud of hydrogen enriched medium via the leak path due to failure of the primary system. This means that in some compartments high concentrations of hydrogen can be expected.
In a report prepared by Professor Karwat on behalf of the Dutch Nuclear Inspectorate [5] the issue of deflagration leading to detonation is addressed. Deliberate burning of hydrogen might be considered, however it should be verified that deflagration loads are within acceptable limits and that deflagration will not lead to detonation. Thus a number of other options are studied, including addition of CO_2 into the containment atmosphere, both pre- and post-accident. In order to investigate the influence of the dilution of the atmosphere with CO_2 a number of experiments are planned by the Dutch organisation TNO.

5. FILTERED VENTING

In the GRS report [2] it is recommended to protect the containment against over-pressure. The government has therefore required the installation of a containment pressure relief system with filter.
The pressure relief station plays an important role in accident management procedures and is coupled with the hydrogen management strategy.
In the German risk study phase B [7] a scenario is described in which depressurisation becomes necessary after a few days. In this case a considerable amount of fission products will have been depleted by natural processes from the containment atmosphere.
As the Borssele reactor is of the KWU type it can be expected that the accident sequence as described in [7] will be similar for the Borssele NPP. However additional scenarios can be postulated in which the time span between the onset of core melting and the activation of the pressure relief system will be shorter.
These scenarios are:

a. For a considerable time the core can be cooled, but the decay heat is stored in the containment. At this time the pressure in the containment has reached a high value, core cooling stops, and the core starts melting.

b. The effect of pressure peaks might be underestimated in [7] so that there is a need for an earlier depressurisation.
c. Possible leakages of the containment beyond technical specification limits may make it preferable to reduce the pressure in the containment at an earlier stage. This may also be necessary if melt through of the foundation is anticipated.
d. If for the purpose of hydrogen management a strategy of post-accident inertisation is followed, then earlier pressure relief may become necessary. A strategy with venting during the blowdown phase of an accident combined with supply of CO_2 may be also chosen.

The following comments are pertinent to the behaviour of fission products:

a. Iodine behaviour
   Consideration should be given to elementary and organic iodine. The df for elementary iodine for a scrubber type filter depends on the amount of water in the sump, the pH of the sump, the amount of medium which passes through the filter, the amount of water in the scrubber and the pH of the scrubber water. In addition, the soluble iodine species may react in the scrubber and may form elementary iodine. Organic iodine will also be present.
   For this purpose a molecular sieve filter may be considered.
b. MCCI
   Because of the small reactor cavity in the Borssele NPP the amount of corium which falls in the cavity may form a layer of 60 cm with a heat load of approximately 1 MW/m^2. The construction of the cavity is different from most German reactors such that sump flooding becomes improbable. It can therefore be expected that the amount of fission products released during MCCI will be larger than in most other reactor applications.
c. Revaporisation
   An additional concern is the revaporisation from the primary system. Hot vapours will enter the containment and may form fine aerosols.

The option of early venting combined with post-accident supply of CO_2 may have the psychological difficulty that it is against the present philosophy of a leak-tight containment. There will be venting in cases where, in a later stage, the accident could be kept under control, so that venting is unnecessary. However these procedures may prevent larger releases at a later stage. If this strategy is chosen this will influence the use and therefore design specification for the filter. The consequences are:

a. Containment pressure relief will be activated during the blowdown phase of an accident, at which time very minor quantities of airborne fission products are present in the containment.
b. Because the inertisation strategy solves the hydrogen problem, the containment spray system can be activated to cool the containment atmosphere and to wash out aerosols. This procedure avoids depressurisation in the period during which airborne fission products are present in the containment.
c. If the spray system or other cooling systems can not be activated, a second depressurisation will become necessary. Because early venting releases a considerable amount of energy, it can be expected that the amount of aerosols supplied to the filter system will be limited. The supply of CO_2 to the containment will have an important influence on the iodine behaviour. In particular a considerable amount of organic iodine can be expected. A fraction of 1% or more can not be excluded. This means that if a molecular sieve filter is installed at the Borssele NPP a heat loading of approximately 20 kW for the molecular sieve filter can be expected.

A filter system which is intended to be used under accident conditions should be as flexible as possible. This means that the filter system should withstand the heat produced by a considerable amount of fission products. On the other hand the filter should be as passive as possible. The question of positioning the filter inside or outside the containment is still under dicussion. A large amount of fission products may require active control. These requirements are all conflicting. The task of the Borssele staff is to formulate proposals which will be assessed against the criteria described in this paper.

6. CONCLUSIONS

- During accident conditions it can not be excluded that the hydrogen concentration may reach unacceptable values. Therefore an accident management strategy should be developed.

- Several options are under study. One of these is the deliberate burning of hydrogen, although it should be verified that deflagration loads are within acceptable limits and, in addition, that a deflagration never proceeds into a detonation.

- Other options are also studied, including addition of CO_2 into the containment atmosphere, both pre- and post-accident.

- Filtered venting is required to protect the containment against over-pressure.

- The design specification of the filter depends on the strategy chosen for hydrogen management.

- The iodine behaviour may require an additional molecular sieve filter.

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PREPAREDNESS OF THE NUCLEAR CENTRE ŚWIERK AND ITS NUCLEAR FACILITIES FOR EMERGENCIES

by

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1. INTRODUCTION

The Nuclear Centre Świerk includes the Institute of Atomic Energy, the Institute of Nuclear Studies, Izotope Centre and also a number of enterprises associated with the institutions.

In the Nuclear Centre Świerk, there are more than three thousand persons employed.
Only the Institute of Atomic Energy owns two research, high flux reactors, named EWA and MARIA and a zero power reactor named AGATA, as well as two spent fuel storages.
It is five nuclear facilities that are under operation at present.

2. EMERGENCY STRUCTURE

It is evident that the Nuclear Centre Świerk is ready to undertake the indispensable measures to protect the persons on-site as-well as the public off-site, in case of radiological emergencies.

There is an organization for emergencies shown in Fig.1.
It covers several levels of activity:
- reactor operators with the shift manager and the facility emergency team,
- site emergency director with the emergency dispatcher staff,
- specialized, on-site emergency intervention teams,
- advisory group of experts,
- off-site notification and intervention organization for emergency, agreed with the local and national administrative authorities.

For each of the levels there are emergency plans, where the duties of individuals have been determined. There is a coordination plan to unify the activities on the levels. There is a special system of communication as well as a notification procedure.

Particular attention is paid to the notification of the Polish Regulatory Body Notification Station, having the liason with the International Atomic Energy Agency.

In the Centre Świerk there is an Emergency Control Centre where the Emergency Dispatcher turns of duty. He has at his disposal communictation and record means, operating with redundancy, as well as a transport.
Having the emergency equipment kits, the dispatcher can take prompt action to mitigate the consequences of a radioactive release. He can activate about thirty persons being on duty at the time out. He can also get immediatly, any indispensable aid.

Using calculation methods, a number of accidents and incidents, not to be excluded, on the nuclear facilities at Świerk, were taken into account and their consequences have been examinated.

Namely, we have taken under consideration five different cases, from a single fuel element failure up to a whole core meeting, for both the research, high flux reactors.

The consequences for in-site and for off-site were investigated, for three meteorological categories A, D and F. An exemplary graph is shown in Fig.2.

Taking into account the meteorological factors, the emergency planning has been carried out, with regard to the following intervention levels and protective measures:

- projected doses for evacuation

  $5\ 10^{-2}$ Sv (5 Rem) - for whole body exposure

  0.3 Sv (30 Rem) - for single organ or tissue

  0.5 Sv (50 Rem) - for skin

- projected doses for warning to stay in house and close doors and windows or to sheltering

  5.10^{-3} Sv (0.5 Rem) - for whole body exposure

  $5.\ 10^{-2}$ Sv (5 Rem) - for single organ or tissue

  $5.\ 10^{-2}$ Sv (5 Rem) - for skin

- projected dose for radioprotective prophylaxis, mainly intake of the stable iodine:

  $5.\ 10^{-2}$ Sv (5 Rem) - for adult thyroid

If the mentioned doses are projected, the protective measures should be taken into account, but not obligatorily, however. If the projected doses are 10 times higher than the mentioned above, the potective measures should be certainly undertaken.

In order to be sure that the emergency plans can be efectively implemented, when needed, drills and trainings the emergency respons personel and support teams, are conducted. Exercises based on a realistic simulation of a forseen emergency condition, on the reactors, are carried out once a year. The whole shortly presented emergency preparedness structure is sistematicly developed and modernized.

As the nuclear safety conditions of the nuclear facilities, the emergency operational readiness is the subject of regular inspection of the National Regulatory Body.

3. FACILITY SYSTEMS FOR ACCIDENT MITIGAION

Both the Polish research reactors EWA and MARIA /1/, /2/, /3/ are water moderated and cooled.

It is clear, that a deprivation of the core of water is the most serious accident that could occur.
To prevent the reactor core against overheating and melting, the passively operating, emergency core cooling systems /ECCS/ have been applied.

The ECCS's have been designed using the reactor operational systems.

The EWA reactor ECCS consists of sprinklers of the core and a water supply system, shown in Fig.3.
As the first water source, the four reserve tanks of demineralized water have been applied. There is about $35m^3$ of water which provides 3 to 4 hours to sprinkle.
After exhausting the reserve, the fire plug is turned-on and the water-pipe network supplies the ECCS, by the reserve tanks. If, at the emergency loss of coolant of the reactor, the water-pipe network is unserviceable, an emergency battery supplied pumping system is activated and the secondary coding water from the cooling tower pools (599 m^3) supplies the sprinklers for about 10 hours. If the last source of water is insufficient, the emergency recirculation pumping system can be put into service and the flowing-out water is turned back to the reactor tank. The minimal duration of the emergency water cooling of the reactor core is about 5 hours.
After that, the core can be cooled by the air flow forced by the ventilation system.

It is after the regular reactor operation period of 100 hours. The MARIA reactor ECCS, shown in Fig.4, consists of two special, self-opening valves, located on the suction pipe of the primary coding system, immersed under the reactor pool water.

The flowing-out, to the inderground tank, water is turned back to the reactor pool by means of an emergency, battery supplied, recirculation pumping system. There are two independent pumps.

The fuel subassemblies of the reactor, to avoid its overheating and meeting, must be kept in water for a very long time of about 20 months.

In case of the loss of coolant of the primary circuit, the operation of removing the channels, with water and containing the fuel subassamblies, to the spent fuel storage tank, is provided. The operation is permissible after a period of some hours.

It is clearly sean, that the systems contained in the reactors protect the nuclear fuel against a thermal destruction, then protect the environement against the fission product release.

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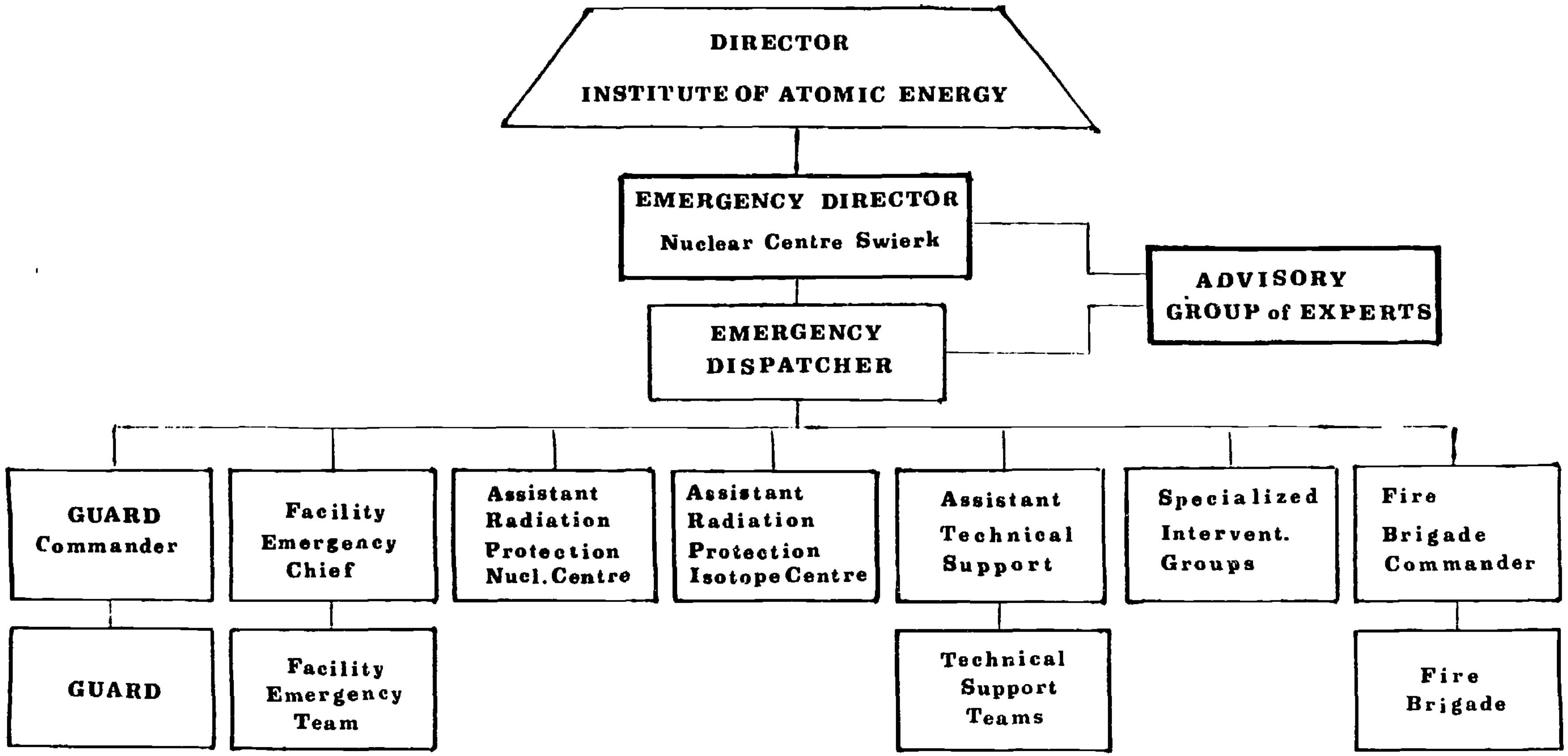

Fig.1. **NUCLEAR CENTRE SWIERK**
Structure for emergencies

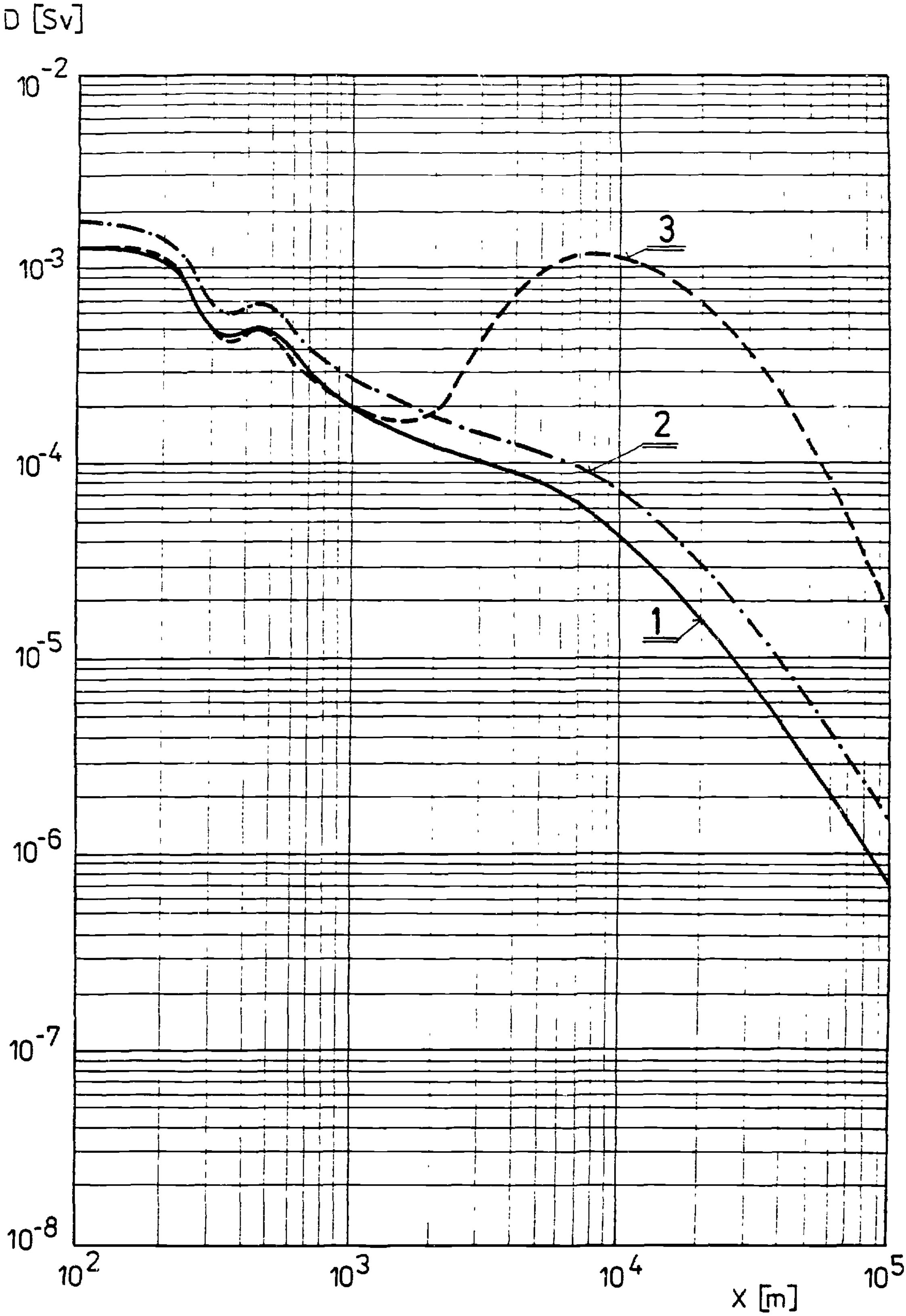

Fig.2. Effective dose equivalent in case of a single EOL fuel subassembly failure of MARIA reactor, at D meteo category.

1- whole body exposure from plume

2- skine exposure from plume

3- adult thyroid exposure

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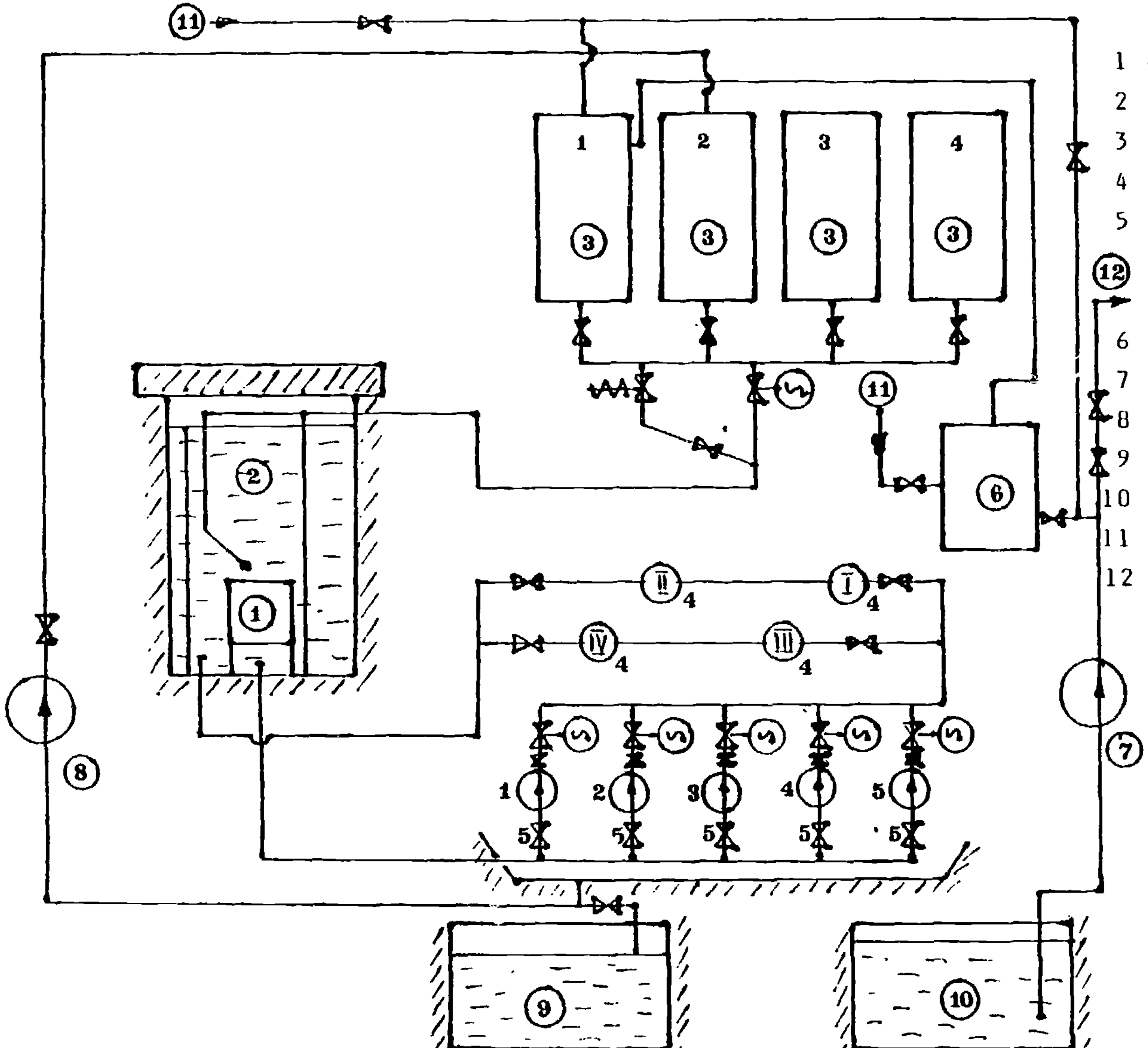
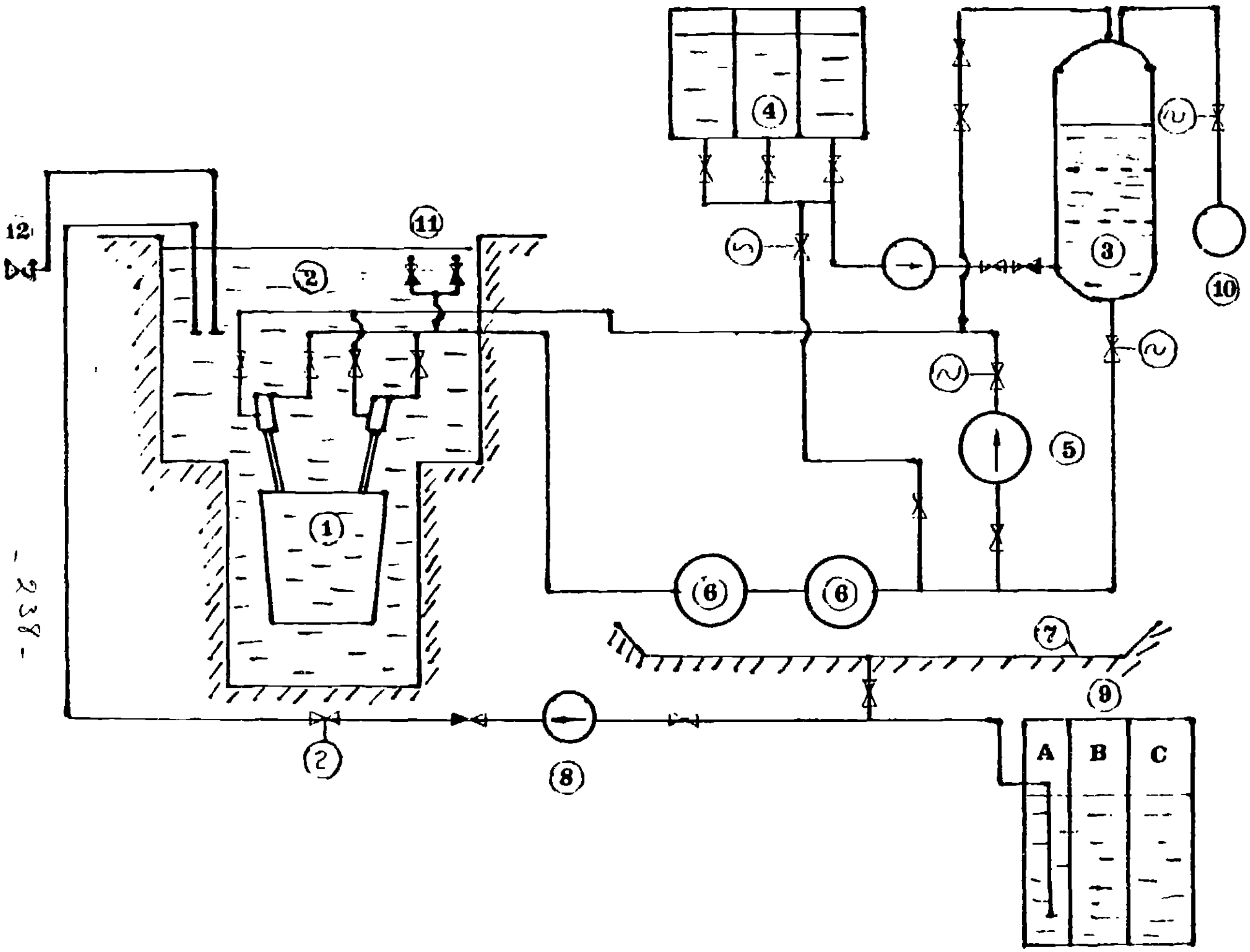


1 - core,
2 - reactor tank,
3 - reserve pure water tanks,
4 - heat exchangers,
5 - primary cooling pumps,

6 - auxiliary tank,
7 - emergency pump,
8 - emergency recirculation pump
9 - liquid wastes tank,
10 - secondary cooling pool,
11 - fire-plug water supply pipe,
12 - secondary cooling system.

Fig.3.EWA reactor emergency core cooling system



1 - reactor core
2 - reactor pool
3 - pressurization tank
4 - pure water reserve
5 - primary cooling pumps
6 - heat exchangers
7 - liquid waste draining system
8 - recirculation pump
9 - drain water tank
10 - gasholder
11 - self-opening valves
12 - water-pipe network

Fig.4. MARIA reactor emergency core cooling system

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ACCIDENT MITIGATION MEASURES FOR VVER-440 REACTORS IN THE CASE OF CHOSEN SEVERE ACCIDENTS

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Contents

1. Introduction

Reactors VVER-440 of V-213 type, operated and being built in several countries, are provided with all essential safety systems required to limit the consequences of various accidents including LOCA to the level acceptable under contemporary safety rules. Nevertheless, the public in most countries is not satisfied with this and reminds reactor specialists about severe accidents which did happen in the past, although they were considered to be beyond design basis accidents. Although the results of Chernobyl accident were so much different from the radiological hazards after TMI accident, in both cases the core was destroyed and the NPP was lost due to human errors. It is therfore our obligation to foresee possible beyond design basis accidents and to provide evaluations of their effects. And what is more to determine those measures, which can be adopted either in the design of the plant or during the accident as operators' actions and to make the reactor staff aware of their effects.

The recent studies of reactor safety show, that the correct actions of operators can substantially reduce the likelihood of core melt and fission product release. It is true, that the complexity of a modern nuclear power plant is such, that one can not expect the opertor to take any action immediately. Accordingly, the protection systems in our NPPs are operated automatically, without requiring any intervention of the operator for a considerable time, usually half an hour. And in the case of accidents remaining within DBA limits this automatic operation of safety systems is quite sufficient to bring the reactor into safe condition. However, should the failures multiply and occur in all safety systems, an intervention of the operator will be necessary. It is worth pointing out, that after a period of distrust to operators after TMI accident, we have returned to the belief, that under severe accident conditions it is the operator who may signifcantly help to minimize the danger.

This conviction is well grounded on excellent perfomance under high stress of many operators in various NPPs. But of course the operator should know beforhand what can happen in his plant and what would be the symptoms. If he is not given correct information, he may make mistakes, like was the case in TMI accident. Thus, the analysis of possible severe accidents in VVER-440 reactors will be valuable in two respects: first, ist will help the designers to implement new features into the plant, secondly - it will help the operator to avoid mistakes and to choose proper course in mitigating the effects of severe accidents.

In the text below we shall discuss the most important features distinguishing VVER-440 type V-213 reactors from PWRs built in Western countries and requiring special approach in accident mitigation.

2. Hydrogen threat in bubbler-condenser containment

The VVER-440 reactors of type V-213 are provided with a bubbler-condenser containment, with the exception of Loviisa NPP, in which ice condenser containment is used. The principle of bubbler-condenser operation is very attractive from the safety standpoint. It provides a large amount of cold water (about 1400 tons) within the containment, which can be used effectively as a heat sink in some accident sequences. In comparison with BWR water condenser pools, the arrangement utilized in VVERs has an additional advantage. It acts not only to retain fission products and condense steam passing through water shelves, but it has also a potential for self-actuating spraying of the space inside the bubbler condenser tower. This spraying action is fully passive, provoked only by pressure difference above and under the water shelf, the difference, which will appear in any accident sequence as soon as the rate of steam condensation on the walls of containment becomes higher than the rate of steam release from the reactor cooling system (RCS). This spraying reduces effectively overpressure in the containment, thus reducing releases of fission products to the environment.

Under most circumstances, the pressure reduction is so large, that the absolute pressure in the reactor compartments falls down below atmospheric. And no wonder, since a large part of air, initally filling up the compartments, is pressed out in the initial phase of the accident across watershelves and through check valves into so called "air traps", from which it can not return (Fig. 1). Upon steam condensation, the pressure must fall and can be kept below atmospheric for many hours.

The analyses of various failures in the bubbler condenser system have shown that single failures, like a check valve which fails to close or a door to the air trap left inadvertantly open do not singnificantly influence the overall response of the system. There is however a danger, which can destroy the elaborate structure of the bubbler condenser and even breach the containment outer walls. This danger is the deflagration of hydrogen.

The designers of VVER-440 reactors in Zarnowiec NPP in Poland similarly as in the previously built plants assumed, that after LOCA the ECCs will act as required and the amount of the cladding reacting with water steam will not exceed 1 % of overall Zr mass in the fuel. This assumption was in direct contradiction to another assumption, normally that the radiological hazards must remain below allowed limits even if 10 % of the core melts during the accident. Evidently, partial core melting would provoke extensive reaction of Zr with water steam and the amount of hydrogen produced in this way would greatly exceed that due to 1 % of Zr oxidation. Since we do not reject the possibility of fission product release due to partial core melt, we feel that the hydrogen production should be also calculated accordingly.

In order to learn the possible dangers to containment due to hydrogen burning under severe accident conditions, four sequences have been analyzed [1], namely:

- S_2B - the accident initiated by a rupture of a 25 mm I.D. pipe of make-up water system in a position close to the main RCS piping, with the simultaneous loss of AC electrical power, both off-site and on-site.

- S_2D - the accident initiated by a rupture of a 25 mm ID pipe of make-up water system in a position close to the main RCS piping, with the simultaneous failure of active ECCs (high head injection system and low pressure injection system).

- AB' - the accident initiated by a rupture of the cold leg of the RCS, with accompaning loss of AC power both off-site and on-site during 2 hours.

- TMLB - the accident due to long-time loss of AC power, both off-site and on-site.

It may be mentioned, that in the case of Zarnowiec NPP which is directly connected with a pumped storage water power plant the probability of total blackout lasting a long time is extremely small.

The calculations were performed with the source term code package (STCP) adapted to VVER-440 type V-213 reactors under an IAEA project. The process of core melting was described by means of model A, i.e. melting downwards. It was assumed that after 0.05 of the core is molten the corium starts to flow down to the support plate and after 75 % melting of the core the whole core slumps down on the support plate. The melting temperature was assumed equal to 2250^{o} C.

The rate of reaction of Zr with water steam was assumed to be limited by two processes, namely gas diffusion in the boundary layer to the cladding and oxygen diffusion in the solid body i.e. in the cladding. The reaction is assumed to occur in all layers in which there is some unoxidized Zr. Similarly, hydrogen production due to steel-steam reaction was taken into account. After corium slump to the lower plenum the heat exchange between corium and water is described by Lipinski equation assuming that corium does not constitute a large uniform mass but rater is divided into many spheres of the O.D. equal to the O.D. of fuel rods. The bottom cover of the reactor pressure vessel (RPV) is assumed to fail when the stresses exceed maximum allowable stresses equal to 5000 kg/cm^2.

The compartments inside the containment are divided into 3 areas:

V_1 - enclosing steam generators, pumps, reactor shaft, corridors, and the shaft in the bubbler condenser tower, altogether V_1 = 23740 m^3

V_2 - space above water shelves, V_2 = 8000 m^3

V_3 - volumes of air traps, V_3 = 16800 m^3

A leak of cross section area equal to 0.5 cm^2 is assumed to be in the reactor compartments (V_1). The containment ultimate strength is assumed to be 0.5 MPa, so that after exceeding this pressure the containment would be breached with the leakage area increased up to 1 m^2.

Between volumes 1 and 2 there is a water bubbler-condenser containing 1430 tons of boric acid solution. The thickness of water layer in the shelves is 0.5 m, so that steam flow from volume 1 to volume 2 is possible when pressure difference exceeds 0.05 bar.

The space above water shelves (volume 2) is moreover connected with the shaft (volume 1) through cut-off valves which at small pressures (below 0.148 MPa) open, preventing the outflow of water flow the shelves in the case of SBLOCA. At high pressures (above 0.165 MPa) the valves close, and the passive spraying of the shaft can occur.

It is assumed, that there are no hydrogen igniters, and self-ignition of hydrogen is determined according to Le Chatelier's equation, upward flame propagation starting at hydrogen molar fraction equal to 0.041 and self-ignition occuring at 0.08. Once initiated, the process of burning is assumed to go on until the fractions of flammable gases fall down below flammability limits. (see Fig. 2).

The results of the calculations can be illustrated with the curves for S_2B accident (see Fig. 3). After the small break LOCA the RCS pressure falls down to 6.55 MPa within 2.5 minutes. After 109 minutes the core gets uncovered and the fuel temperature increases up to 1090^{o} C. The reaction of Zr with water steam is observed in some points, but it is insignificant and the hydrogen release does not exceed 1.5 kg. After 134 min. the RCS pressure decreases below 6 MPa, i.e. the pressure in ECCs hydroaccumulators. Then the check valves open and passive ECCs delivers water to the core. The core becomes reflooded and cooled down, but within 260 minutes the ECCs hydroaccumulators become empty. Then the water level in the RPV falls down and gets below the upper core edge in 312 minutes after accident initiation. The temperature of the core increases and in 347 min. the hottest nodes in the core begin to melt.

The hydrogen released due to Zr-steam reaction flows out of the RCS and into the containment as shown in Fig. 4. During core melting 29,6 % of Zr in fuel cladding become oxidized. The overall amount of H_2 released out of the RCS till 372 minutes is 229 kg.

In 361 minutes the first molten nodes fall on the support plate, then in 372 min. when 75 % of the core is molten, the remainder of the core slumps down and in 393 min. melts the support plate. Then the corium together with the molten plate fall down into the lower plenum filled with water. Steam generation increases violently, the pressure increases and the water level decreases (Fig. 3). Then the water from lower plenum evaporates, the RPV bottom heats up and ruptures after 637 minutes. The molten corium falls down onto the bottom of reactor shaft, reacts with concrete and within 3000 minutes penetrates into the basement mat down to the depth of 160 cm and horizontally to 80 cm.

The changes of pressure inside the containment induce short-time sprays from the shelves of bubbler condenser, starting at 0.05 minutes and then occuring 6 times from 489 till 637 minutes. The concentration of hydrogen increases and after 777 minutes reaches 8 % in volume 2, i.e. above water shelves. In volume 1 this concentration is reached even earlier, but no hydrogen ignition occurs due to a large fraction of water steam, oscillating between 58 % and 65 %. However, there is very little steam in volume 2 (Fig. 5 and 6) and when the 8 % molar fraction of hydrogen is reached the self-ignition follows. Due to sudden pressure increase the water from the shelves is pressed out and the steam in the tower shaft becomes condensed on falling water drops, then due to pressure decrease the steam condenses in the whole volume 1. When the steam fraction decreases below 48 %, the hydrogen ignites in volume 1. After 2 seconds the shelves become empty and then the steam can penetrate without obstacles to volume 2 and 3.

After burning the whole hydrogen in volume 2, the pressure in this volume decreases. In volume 1 the process of burning stops when there is no more oxygen to keep up burning. Due to the pressure difference between volumes 1 and 2 a part of hydrogen which was not burned in volume 1 is pressed into volume 2. After 1081 minutes the hydrogen accumulated in volume 2 burns again and the pressure rises up to 0.5 MPa. Generally during two hydrogen burns about 384 kgs of H_2 are burned.

The release of hydrogen into volume 1 continues, but due to low oxygen content (high proportion of non-condensible gases released in corium reaction with concrete) no more burns occur. The curves of hydrogen accumulation during 48 hours after accident are shown in Fig. 7.

In the case of S_2D accident the spraying system is operational and keeps down the amount of steam in volume 1. This helps to reduce pressure but also facilitates hydrogen burning. The first burn occurs at 376 min., and rises pressure up to 0.21 MPa, the subsequent burns occur during corium reaction with concrete (4 times from 660 min. till 790 min.) raising the pressure to 0.254 MPa. At the end of analyzed period, i.e. after 48 hours, the pressure is brought down by spray system to 0.155 MPa. The releases of fission products to the environment are much smaller than in the case of S_2B accident and the pressure peaks are smaller.

In the case of AB' accident core melting occurs much earlier, beginning at 26.9 minutes, but the amount of oxidized Zr is similar as before, namely 24.7 % to the moment of core slump onto the support plate. The hydrogen molar fraction increases several times to 8 % and hydrogen burns occur, with maximum pressure reaching 0.21 MPa. Although the active spray system cannot work, passive spraying with water from bubbler condenser is sufficient to remove most steam from volume 1 and then the generation of steam is small since there is no water injection to the core. Therefore the steam fraction remains below 48 % and hydrogen can burn whenever its fraction exceeds 8 %.

In the fourth case, that of TMLB accident, drying out of the secondary side of steam generators occurs at 566 minutes and in 576 min. the pressure in the RCS reaches the threshold of safety valve opening. After 912 min. the level of water falls below the safety valve and the steam flow begins. The core gets uncovered in 965 min. and after 1003 min. the fuel melting begins. Before the core slumps down onto the support plate 24,8 % of Zr reacts with steam. The hydrogen is released into volumes 1 and 2, but due to high steam content in both these volumes no ignition follows. Even further events - RPV bottom breach and core - concrete reaction do not provoke hydrogen burning (Fig. 12). However, the operator must be aware of the potential danger. Once the sprays are activated, the steam will be condensed, the pressure maintained throughout the accident at a high level (Fig. 13) will fall with the corresponding increase of hydrogen molar fraction and the mixture can reach not only deflagration but detonation parameters.

The scenarios described above show, that installation of hydrogen igniters is necessary to avoid containment destruction in several types of severe accidents. The generators must be aware of potential dangers depending on the type of accident. In order to cover the whole spectrum of possible severe accidents, the igniters should be located in

- space above water shelves (deflagration in S_2B sequence)
- bubbler-condenser tower shaft (ignition after passive spray initiation)
- reactor compartments (burning conditions in the case of active spray operation)
- auxiliary rooms of small volume
- reactor shaft, where after RPV break-through hydrogen may burn.

If hydrogen igniters are available and work efficiently, then hydrogen burning will occur at high steam contents and low hydrogen fractions, so that in no case will the pressure in the containment raise to the maximum design value.

3 CONTAINMENT VENTING

Containment venting possibilities have been discussed for Zarnowiec NPP taking into account plant specific conditions, in particular the existence of a neighboring pumped water storage power plant, situated at the same lake and directly connected with the NPP. It has been found, that the bubbler-condenser containment provides effects similar to venting in many sequences, since the air is removed from reactor compartments and does not return, thus significantly reducing the threat of long-term overpressurization. A prolonged loss of ECCS and containment spray system is most probably due to total blackout, and this blackout can not be very long at Zarnowiec NPP.

The increases of pressure due to hydrogen burning can be limited to the values which do not threaten the integrity of containment by installing hydrogen igniters as discussed above. Thus, the venting system would be used only under very exceptional conditions. On the other head, releasing a part of containment atmosphere including ~~large part of containment atmosphere including~~ large noble gas inventory does not seem to be a solution attractive to the population. Moreover, if after pressing a part of air into the air traps additional portion of gases is lost through venting system, the underpressure in the containment arising after passive spray operation may provoke loss of leak-tightness of the inner steel lining, thus significantly increasing further leakage of fission products to the atmosphere.

In view of these considerations, the installation of venting system in Zarnowiec NPP has been found to involve more drawbacks than advantages.

4 FLOODING REACTOR SHAFT CAVITY

The idea of using water from the ice condenser to flood the reactor shaft cavity was first proposed for NPP Loviisa [2]. In a containment with the bubbler-condenser tower similar possibilities exist, since the amount of water is very large and this water is sprayed down under nearly all severe accident conditions before the molten corium falls into the lower plenum. An analysis of the geometry of the reactor shaft has shown however, that the shielding rings installed in the shaft around the RPV make natural convection of water difficult especially under two-phase flow conditions. The possibility of flooding the reactor shaft clearly deserves further attention, since if effective, it could remove the danger of RPV meltthrough. The shielding rings could be made movable, to be installed for reactor maintenance and removed for reactor operation, or their geometry could be changed to accommodate natural convection flows. Similarly, it is possible to solve the problem of water distribution after emptying the shelves of the bubbler condenser so that the reactor shaft cavity is flooded and yet sufficient water intake is assured for recirculation system.

Unfortunately, these problems have not been pursued further due to the decision of Polish government, stopping the construction of the NPP Zarnowiec and all design work connected directly to it. Still we hope to be able to continue to study this problem at the Institute of Atomic Energy so as to be able to show that very significant reduction of severe accident risks can be thus achieved.

5 CONSIDERATIONS CONNECTED WITH POSSIBLE LARGE BREAK OF RCS COLLECTOR IN THE STEAM GENERATION

In PWRs the range of accidents in steam generators included in design analyses is usually limited to a break of one or two S. G. tubes, and the break of the tube sheet wall dividing RCS from the secondary coolant is not considered.

In VVERs the accident involving the rupture of the hot collector of the RCS due to circumferential break has been given careful consideration and shown not to lead to radiological hazards exceeding those at Design Basis Accident.

The configuration studied is shown in Fig. 14. In the analysis it was assumed that

- all safety valves on the secondary side of steam generators are available
- one of two safety valves on the damaged SG fails in open position,
- dump values blowing steam to atmosphere are available
- dump valves blowing steam to technological condenser are considered unavailable in case A, and available in case B
- cut-off valves in RCS loops are modelled as available in case B
- other safety systems operate as designed.

The analysis was aimed at finding answers to the following questions:

- will the pressure increase on the secondary side remain within admissible limits?
- It is necessary to introduce an additional line to blow down the water from the damaged S. G. in order to avoid flooding safety valves with water?

Calculations were conducted with RELAP4/MOD6 code for 5 scenarios, the first corresponding to a 100 mm equivalent diameter break in an existing configuration, the other scenarios including various accident mitigation features.

For the basic case the pressure in the space between the hot RCS collector and the SG cover reaches 11.87 MPa after break, then falls down (Fig. 15). The peak pressure value does not exceed the strength of SG standpipe, although it comes dangerously close to it. The steam dump station to atmosphere is opened after 9.7 s at the pressure of 5.3 MPa, and safety valves open at 5.68 MPa after 19 s. The maximum pressure in steam line is 5.69 MPa, and after opening safety valves it begins to fall. The core is covered with water throughout the accident and the fuel is not damaged.

The secondary side of the SG becomes filled up with water after 78 s. If credit is taken for operation of RCS cut-off valves, then the damaged S. G. gets separated from the core within 110 s and water does not flood safety valves (Fig 16). Within time period from 135 s till 350 s the safety valve releases steam-water mixture, then the steam flow is restored.

If the credit for cut-off valves in the RCs is not taken, then the installation of an additional pipe of 100 mm ID is proposed to keep the water level below safety valves. The results are shown in Fig. 17 and are clearly satisfactory.

Another possibility was considered, namely installation of three membranes on the SG standpipe to prevent large pressure rise and to preserve the integrity of SG shell. This scheme is shown in Fig. 18. The resulting pressure increase does not exceed 5.2 MPa in the steam collector so that both dump valves to atmosphere and safety valves remain closed. In this case the maximum pressure above the ruptured RCS collector does not exceed 7.3 MPa.

The design decisions have not been yet taken. In any case however, the mitigation measures are possible to install and operator actions such as closing RCS cut-off valves or opening blow-down pipe (in case shown in Fig. 17) can help to limit the radiological hazards after the accident.

6 INTENTIONAL DEPRESSURIZATION OF THE RCS

The existing design of VVER-440 pressurizer and safety valves does not allow to depressurize intentionally the RCS. Due to very large water inventory in the secondary side of steam generators, the heat can be removed from the RCS for a long time through natural convection and secondary coolant evaporation (Fig. 19, 20). However, after partial uncovery of SG tubes the amount of heat transferred to secondary side decreases below residual power and the pressure in the RCS starts to increase (Fig. 20). Then the safety valves in the RCS open and the primary coolant is evaporated while high pressure is maintained in the RCS.

In order to avoid core melting under high pressure, which can lead to abrupt RPV melt-through and corium ejection into the reactor shaft with possible direct containment heating, an additional system of controlled RCS depressurization should be installed.

If the initiating event is not TMLB but e. g. uncontrolled pressurizer leaks, loss of feedwater, steam line break or SG tube rupture the mitigation measures should start with feed and bleed strategy. However, for a feed and bleed strategy it is also necessary to bring down the RCS pressure, so the installation of the depressurization system seems necessary.

Since the initiation of RCS depressurization should be delayed as long as possible, it is also necessary to install a water level meter in the RPV and to elaborate proper procedure for the operators, taking into account RCS pressure, water level and temperatures.

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Pressurization Design Studies for LOVIISA Reactors
Proceedings of Specialists Meeting on Intentional Coolent System Depressurization
Garching, 12. - 14. June 1989, CSNI-report 158

3. J. Szczurek:
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IAEA-report 11/1990

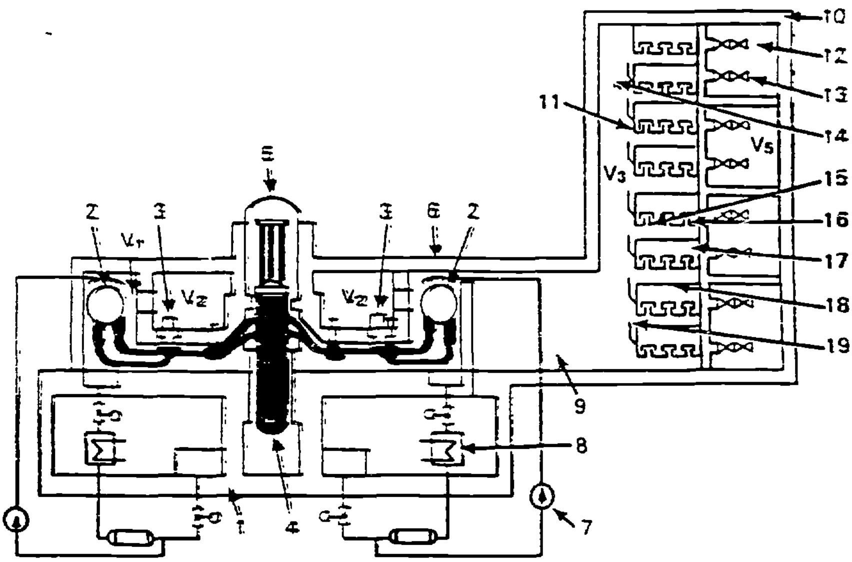


Fig. 1a **Illustration of V213 Bubbler/Condenser Tower**

1-reinforced concrete walls and floors; 2-steam generator; 3-primary coolant pump; 4-reactor vessel; 5-protective shielding cap; 6-sprays; 7-spray system pump; 8-heat exchanger cooler for water drawn from sump; 9-tunnel connecting steam generator compartment with bubbler-condenser tower; 10-reinforced concrete walls; 11- suppression pool system at each level; 12-air trap volume; 13-check valve; 14-shaft convecting steam and air to tray levels; 15-tray; 16-steam channels; 17-deflector cover tray; 18-plenum region cover; 19-overflow discharge; V_1-steam generator compartment; V_2-pump compartment; V_3-shaft inside bubbler condenser tower; V_5-air trap volume

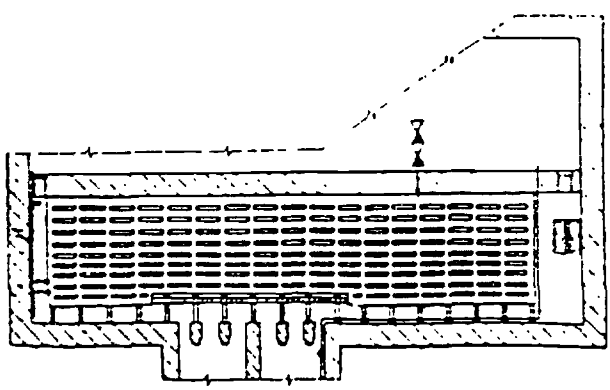
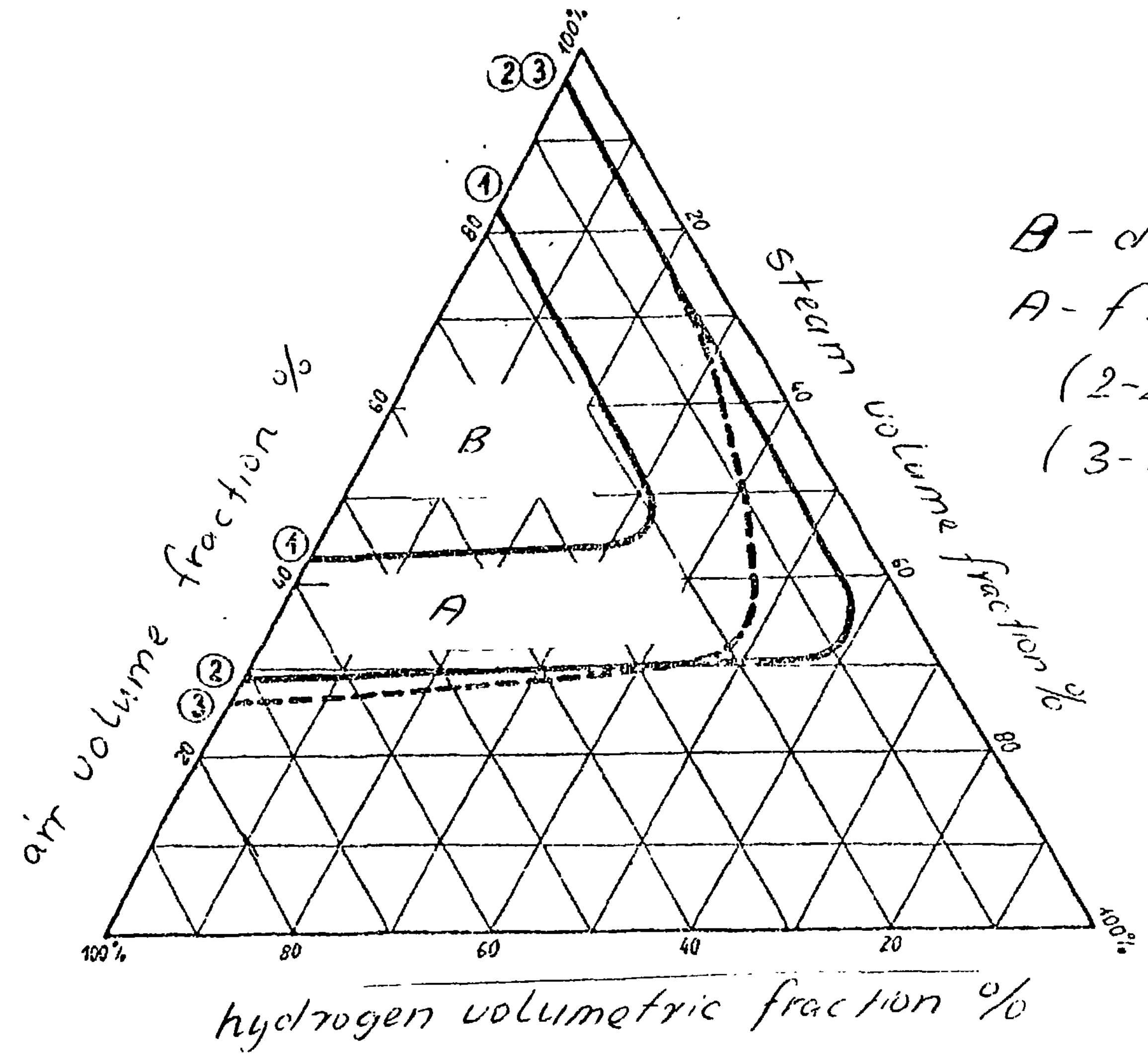


Fig. 1b Horizontal Cross Section Through the Bubbler/Condenser Tower of a V-213 Unit



B - detonation area (1-1)

A - flammability area,

(2-2) - at 20 °C, 0.1 MPa

(3-3) - at 150 °C, 0.1 MPa

Fig. 2. Flammability limits for H_2

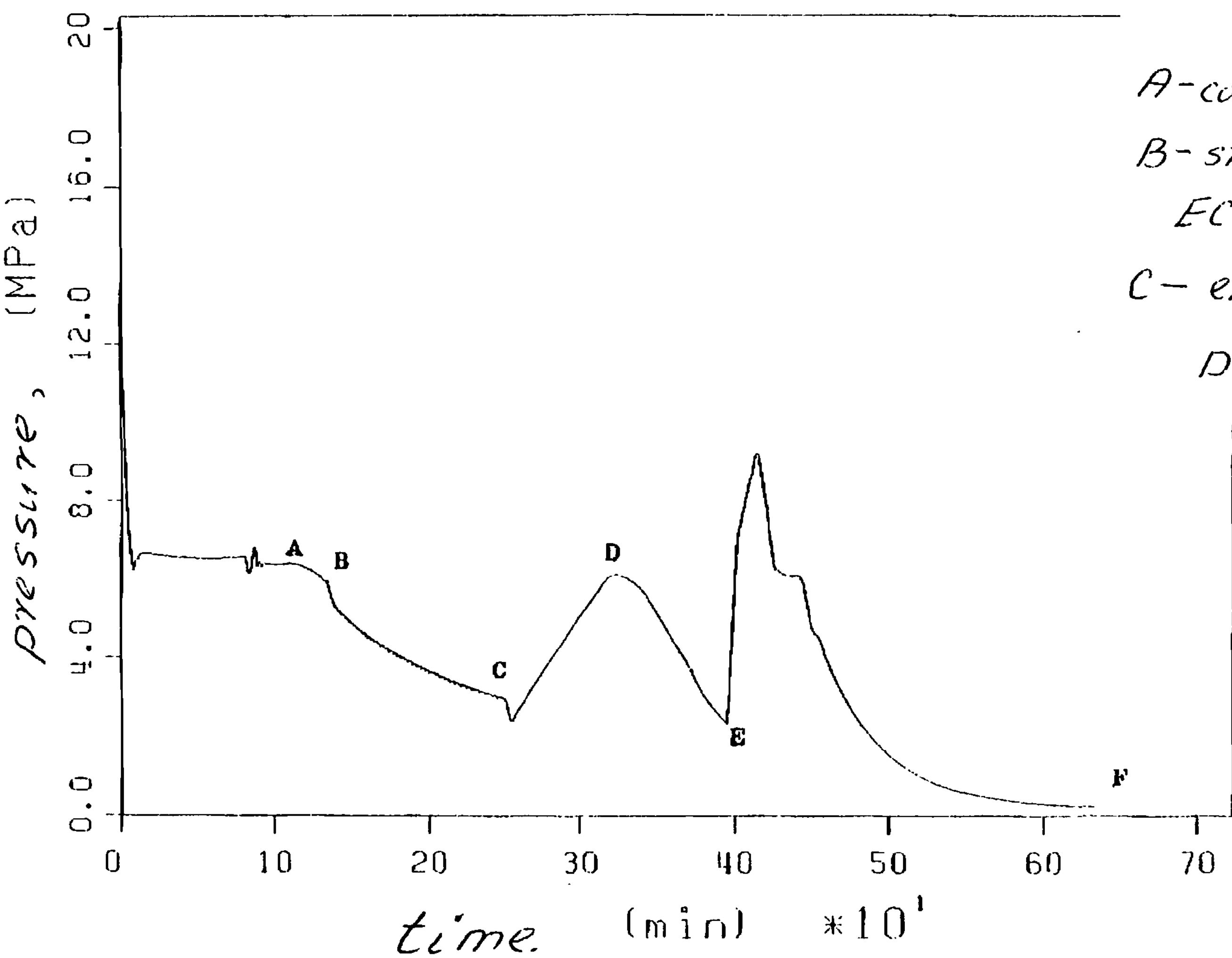


Fig. 3a RCS pressure during S_2B accident

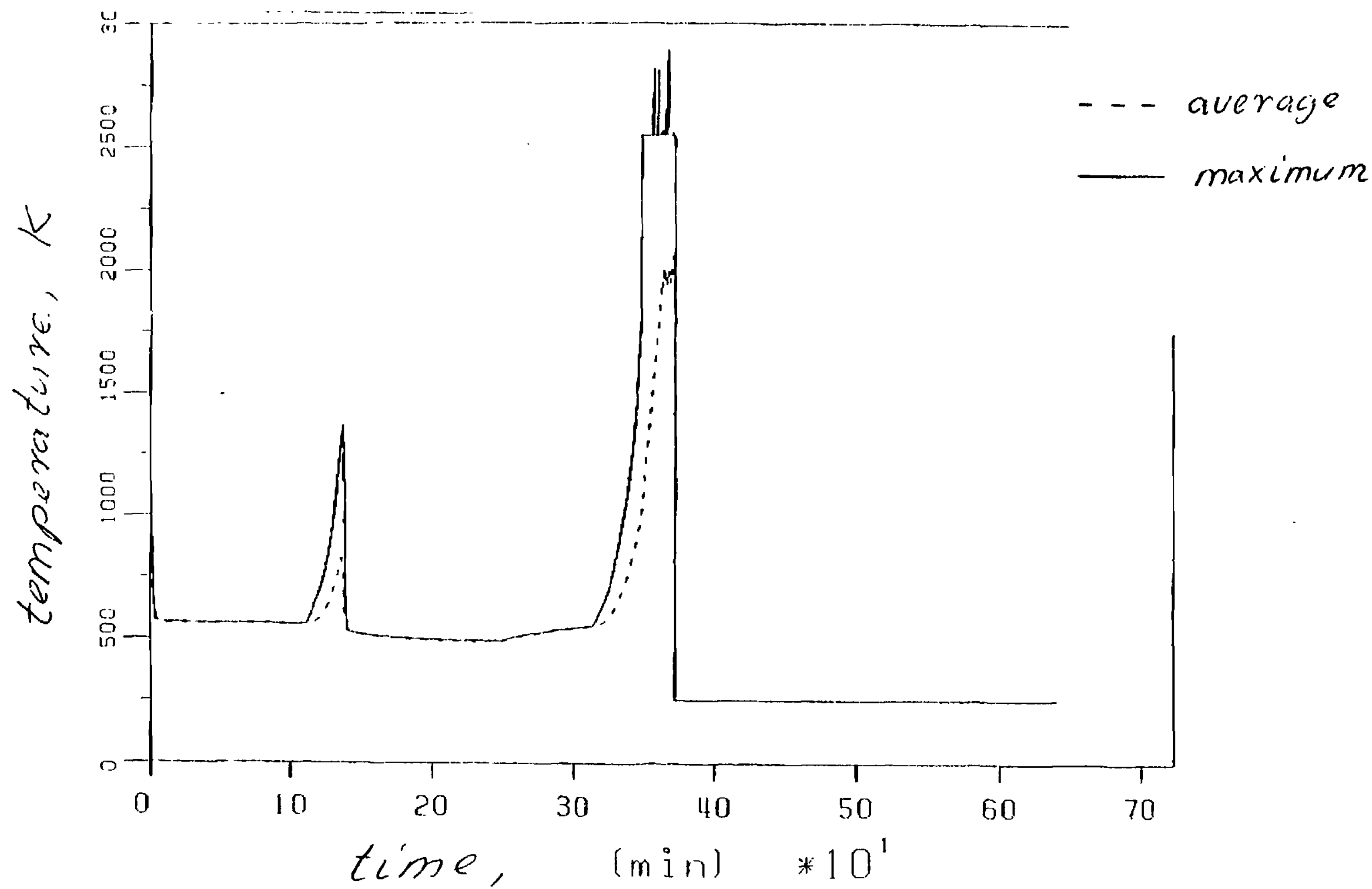


Fig. 3b. Core temperatures during S_2B accident

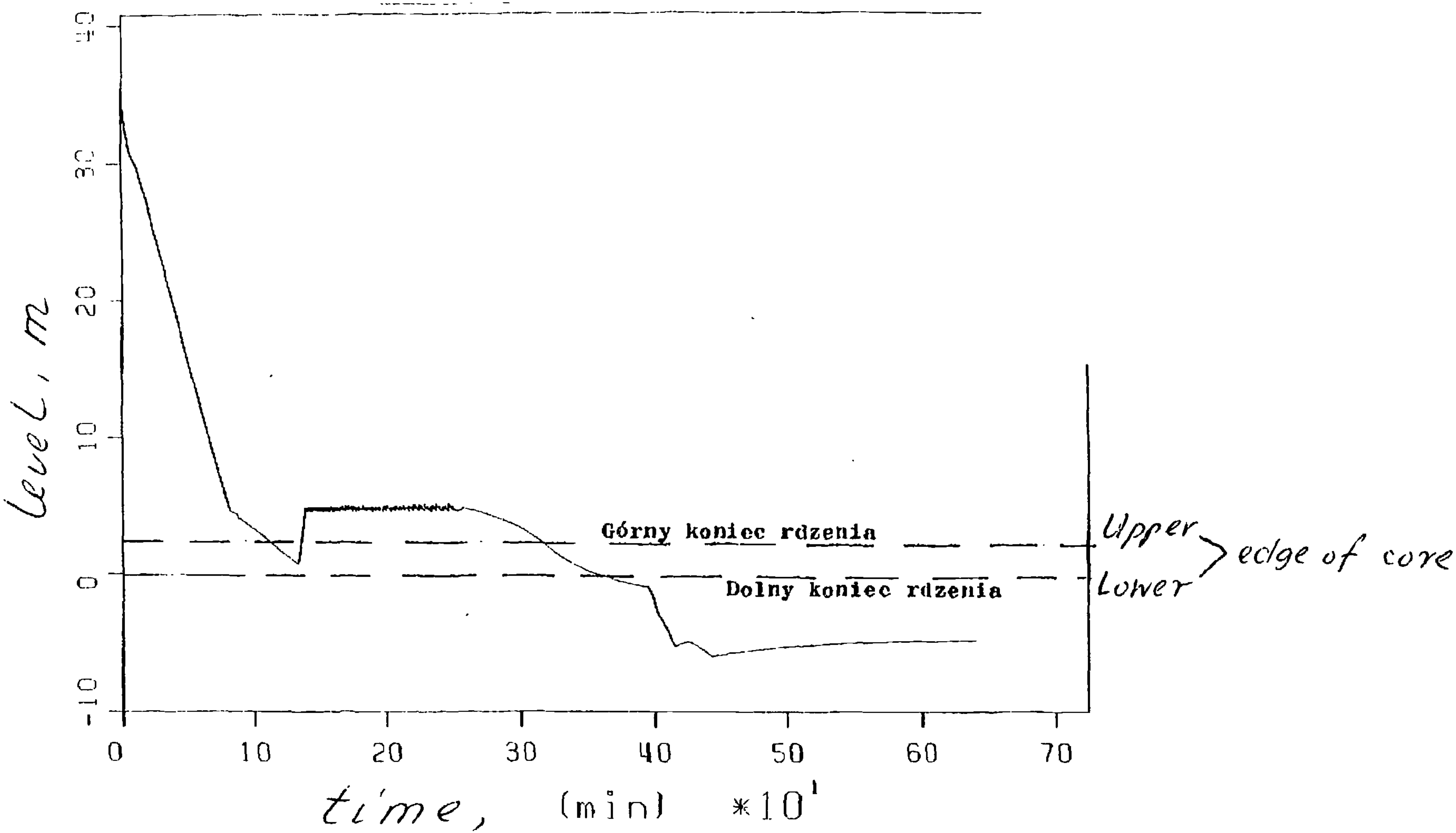


Fig 3c Water-steam mixture level during S_2B accident

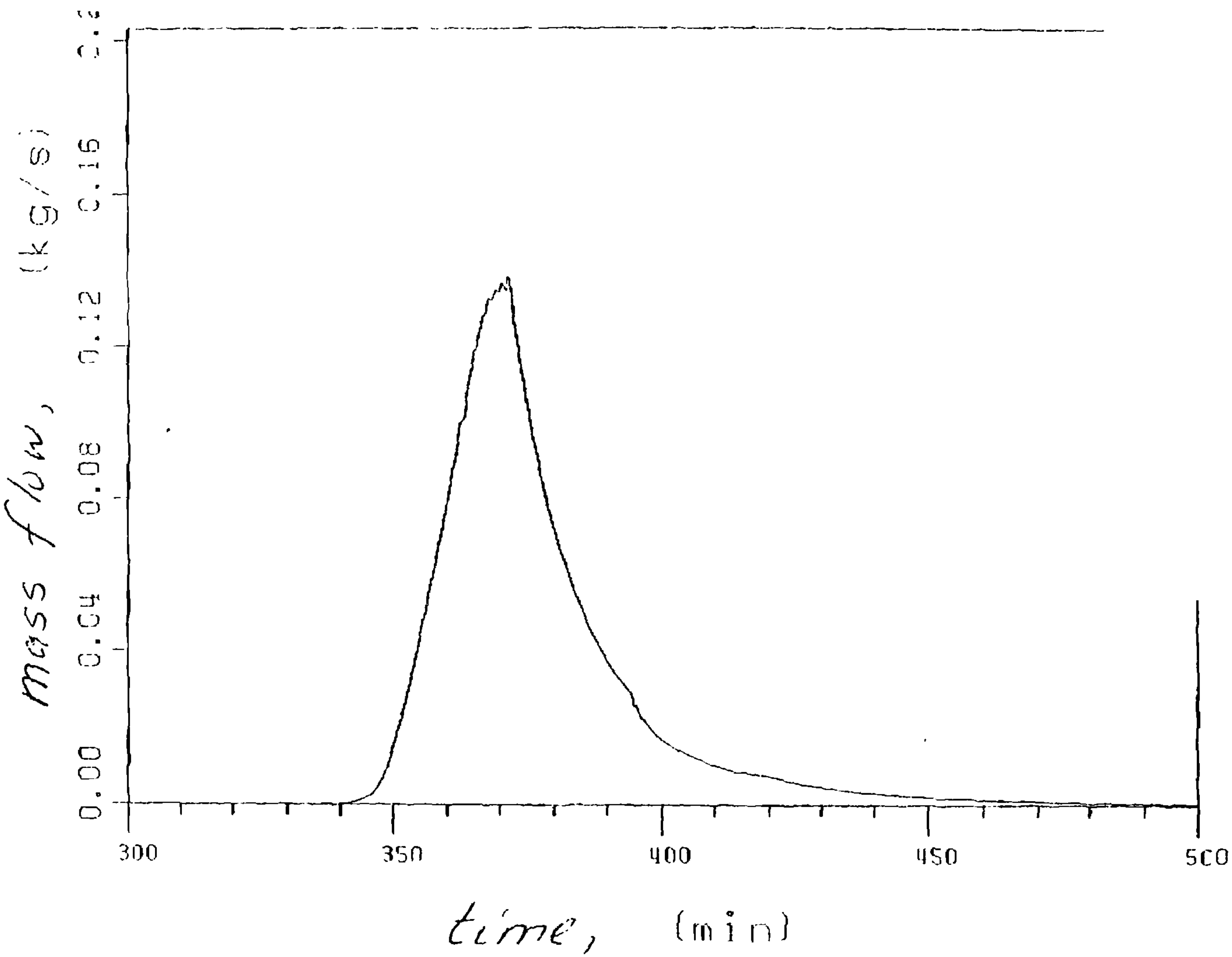


Fig. 4 Hydrogen flow out of the RCS during S_2B accident

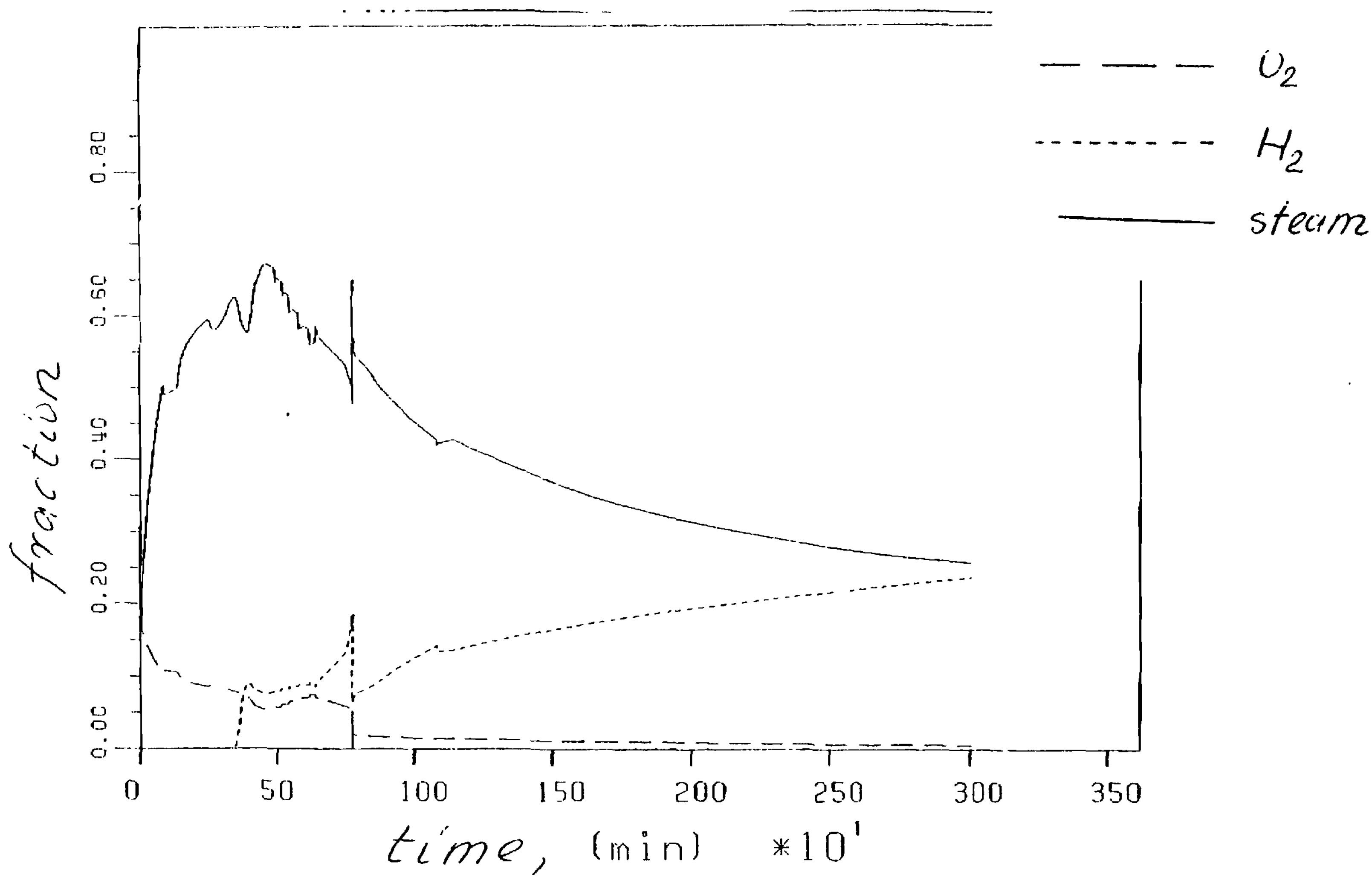


Fig. 5. Molar fractions of H_2, O_2 and steam in volume 1, during S_2B accident.

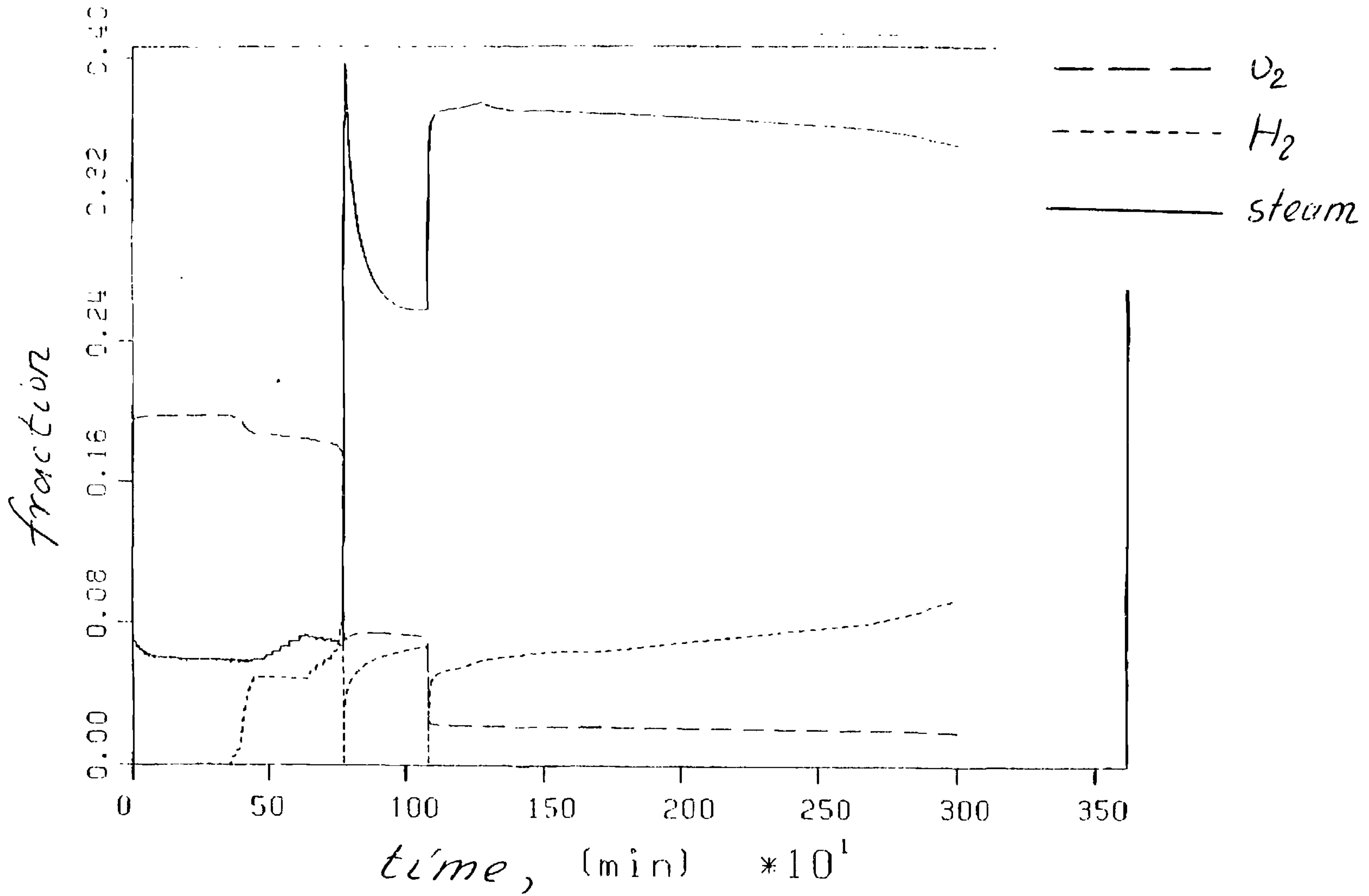


Fig. 6 Molar fractions of H_2, O_2 and steam in volume 2, S_2 B

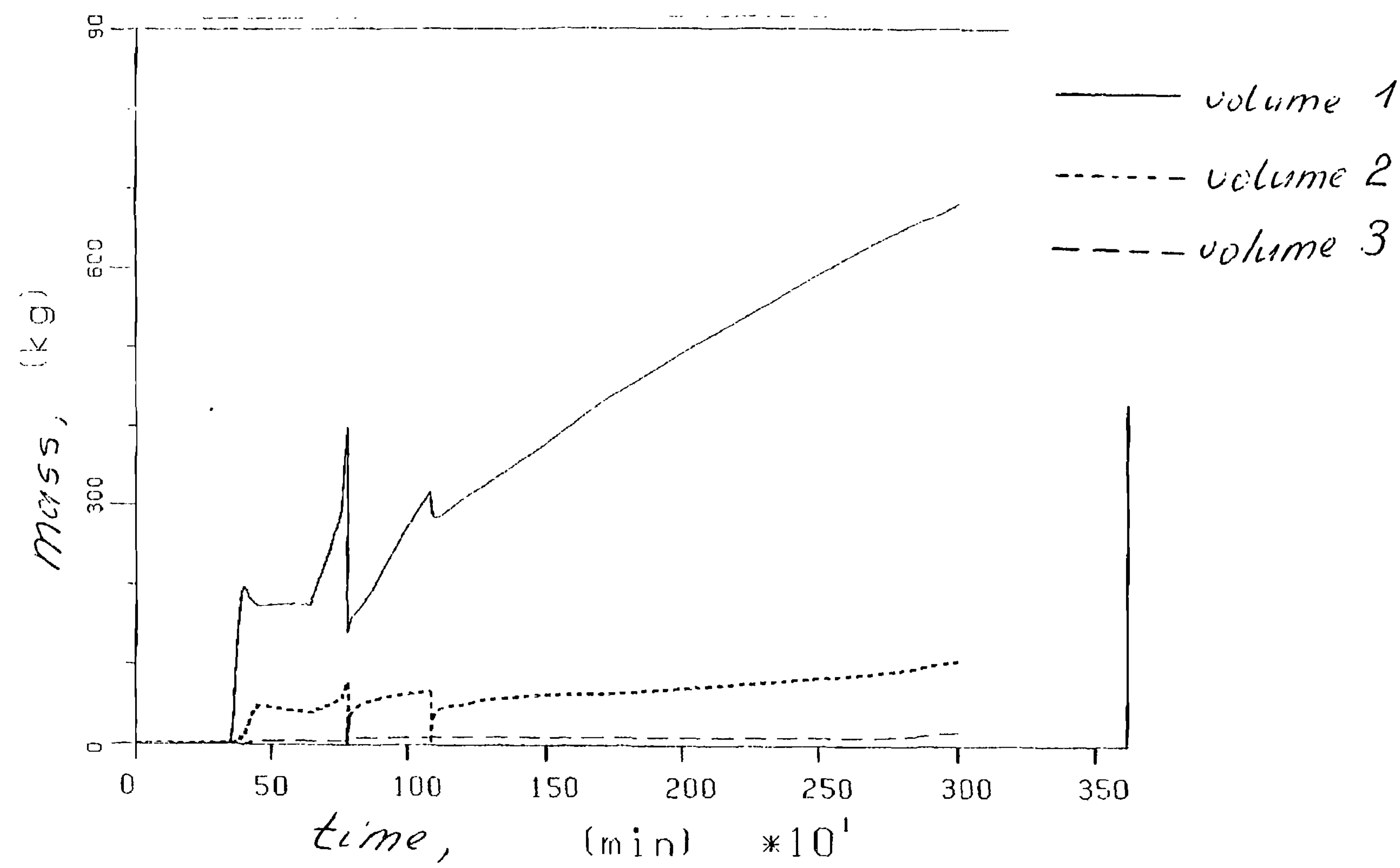
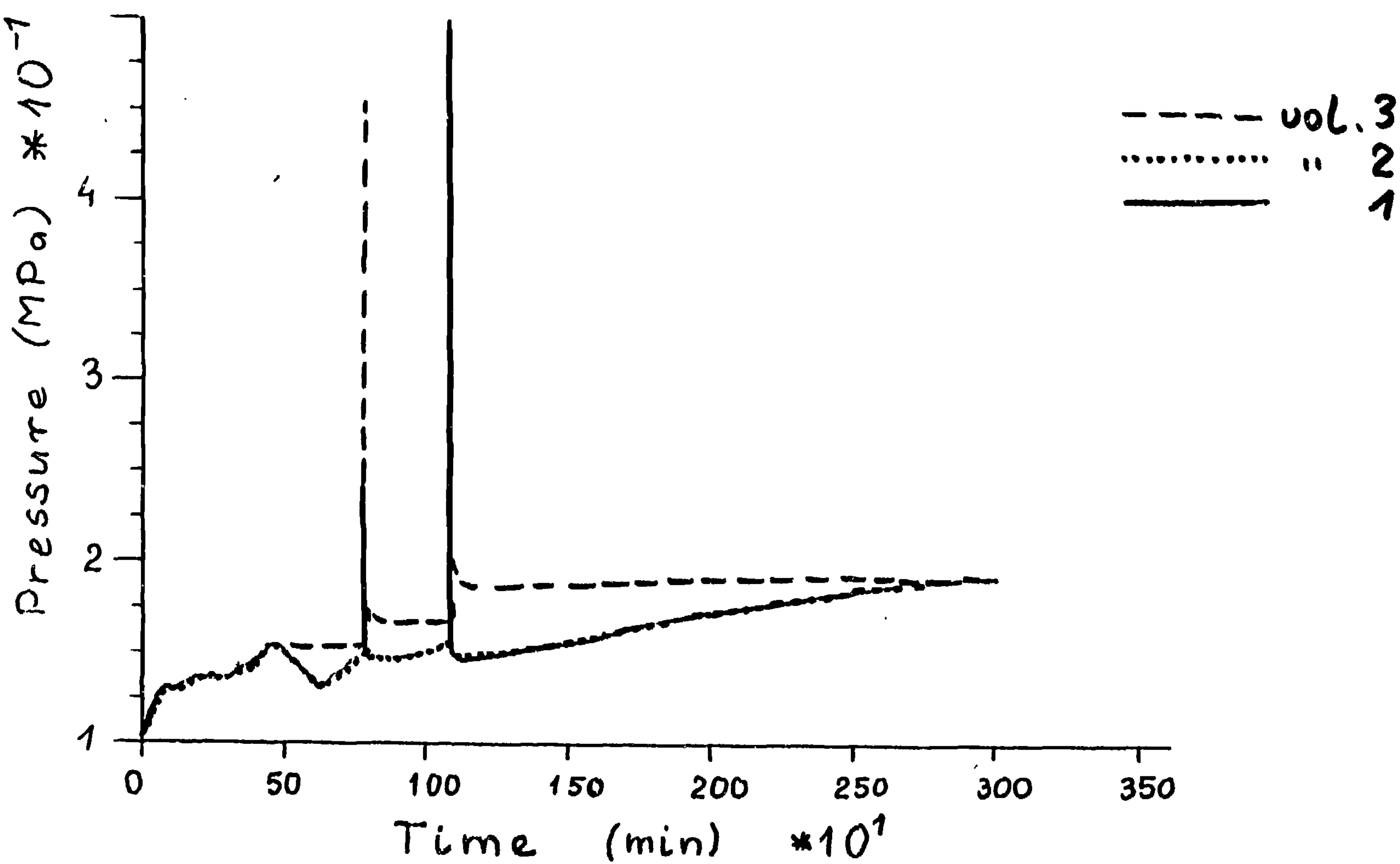


Fig. 7. Accumulation of H_2 after S_2B accident

Fig. 7b Pressures in containment during S_2B accident.



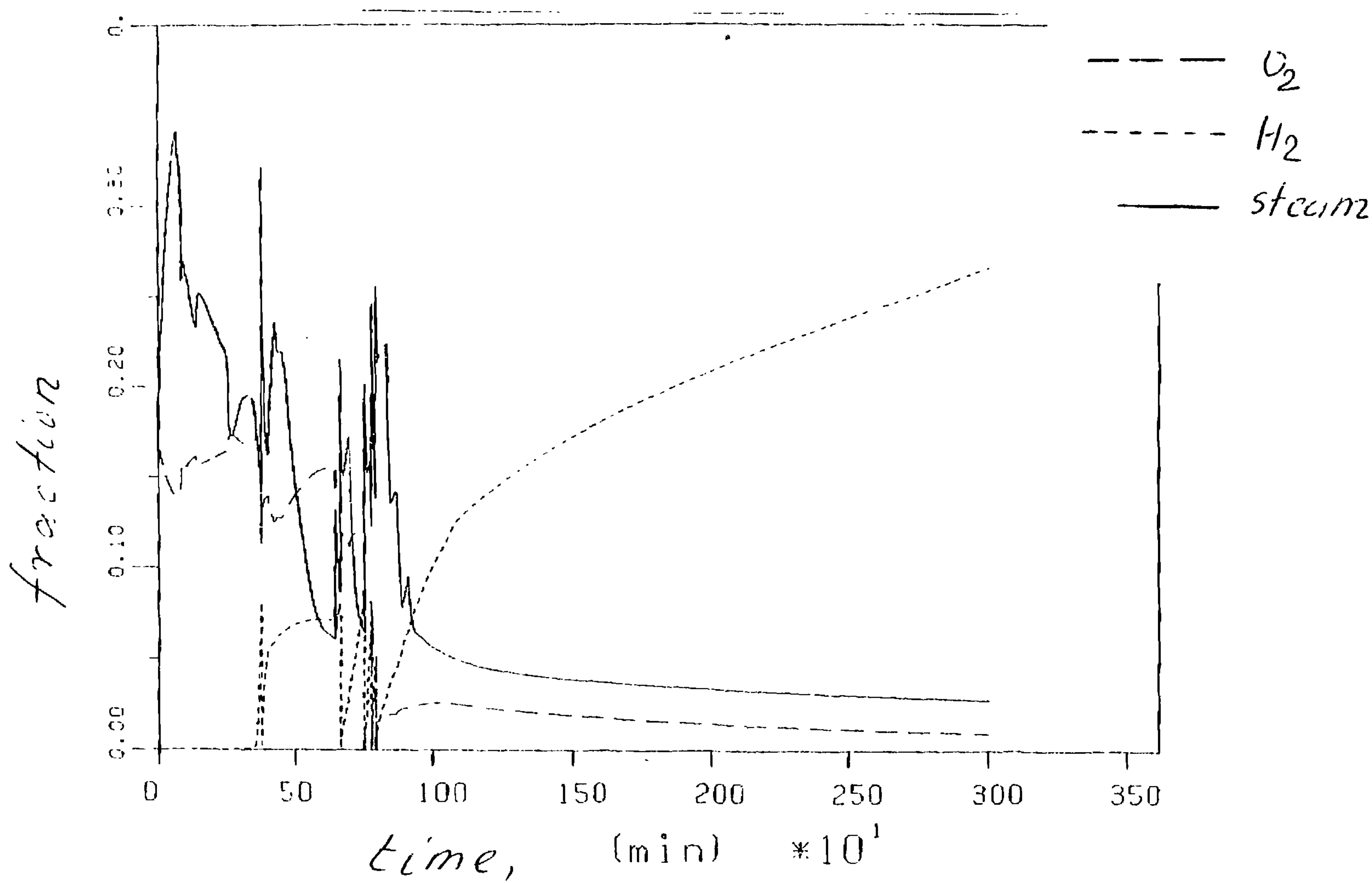


Fig. 8 Molar fractions of H_2, O_2, steam in volume 1, S_2D accident.

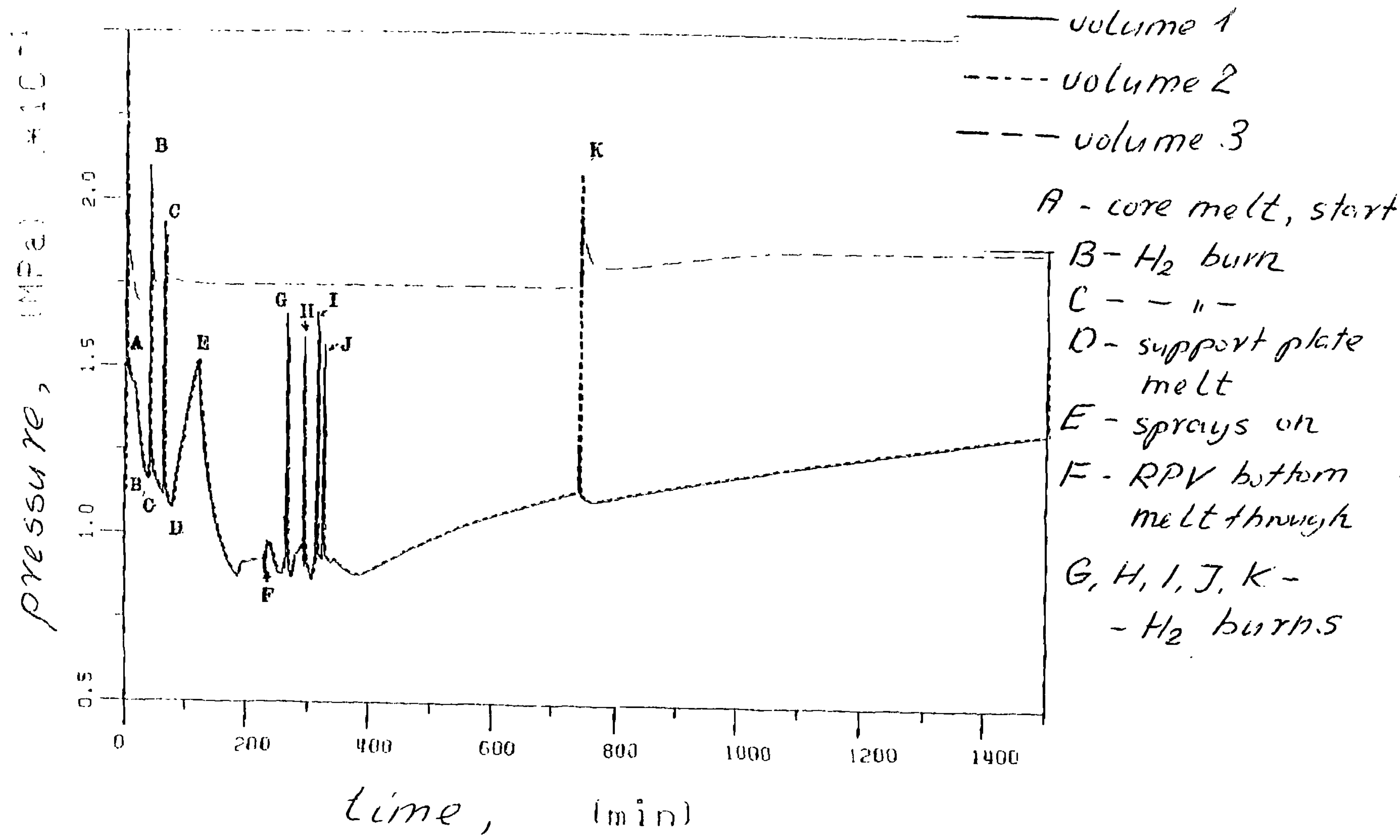


Fig. 9 Pressure in containment during AB' accident

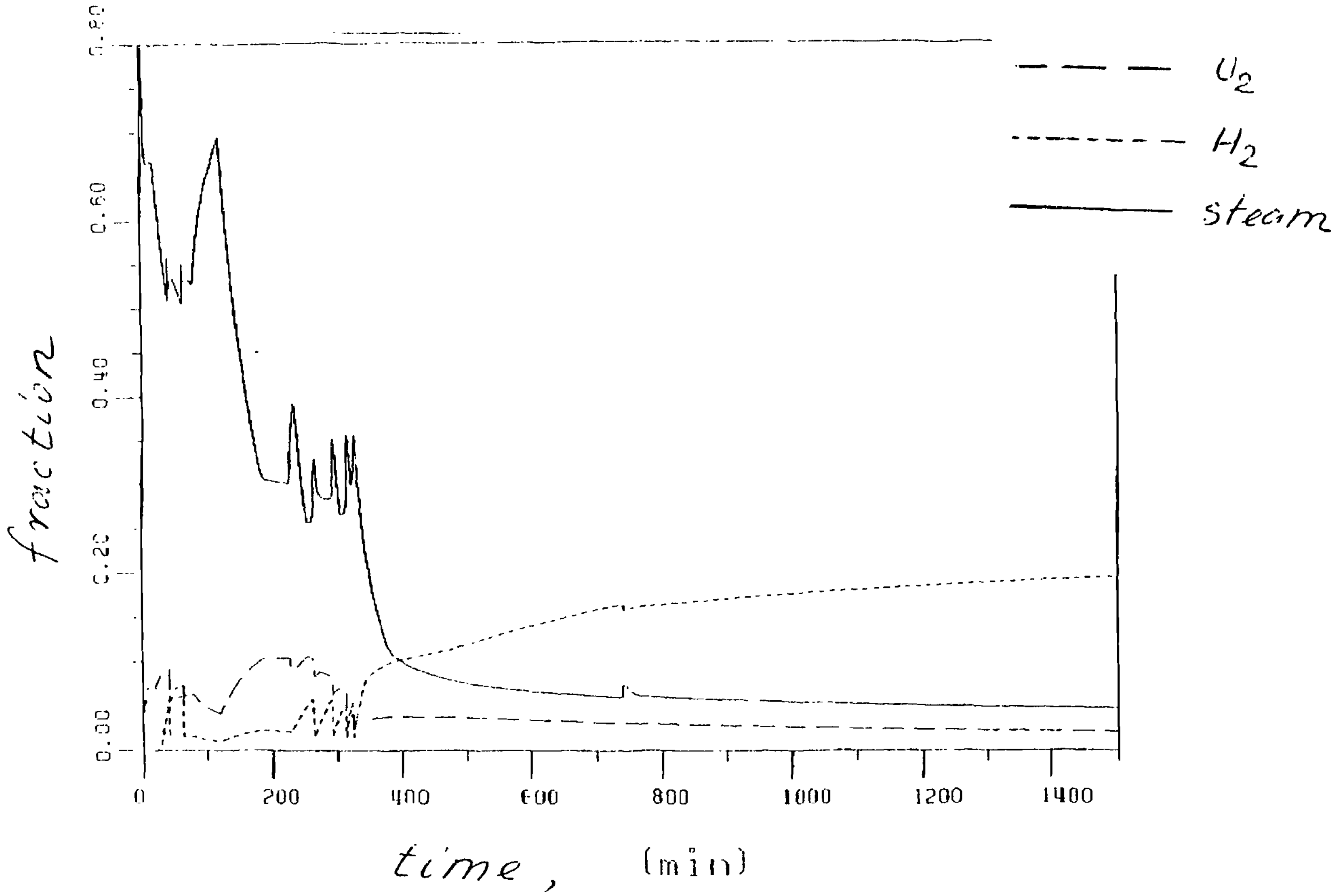


Fig. 10 Hydrogen molar fraction in volume 1 during AB' accident.

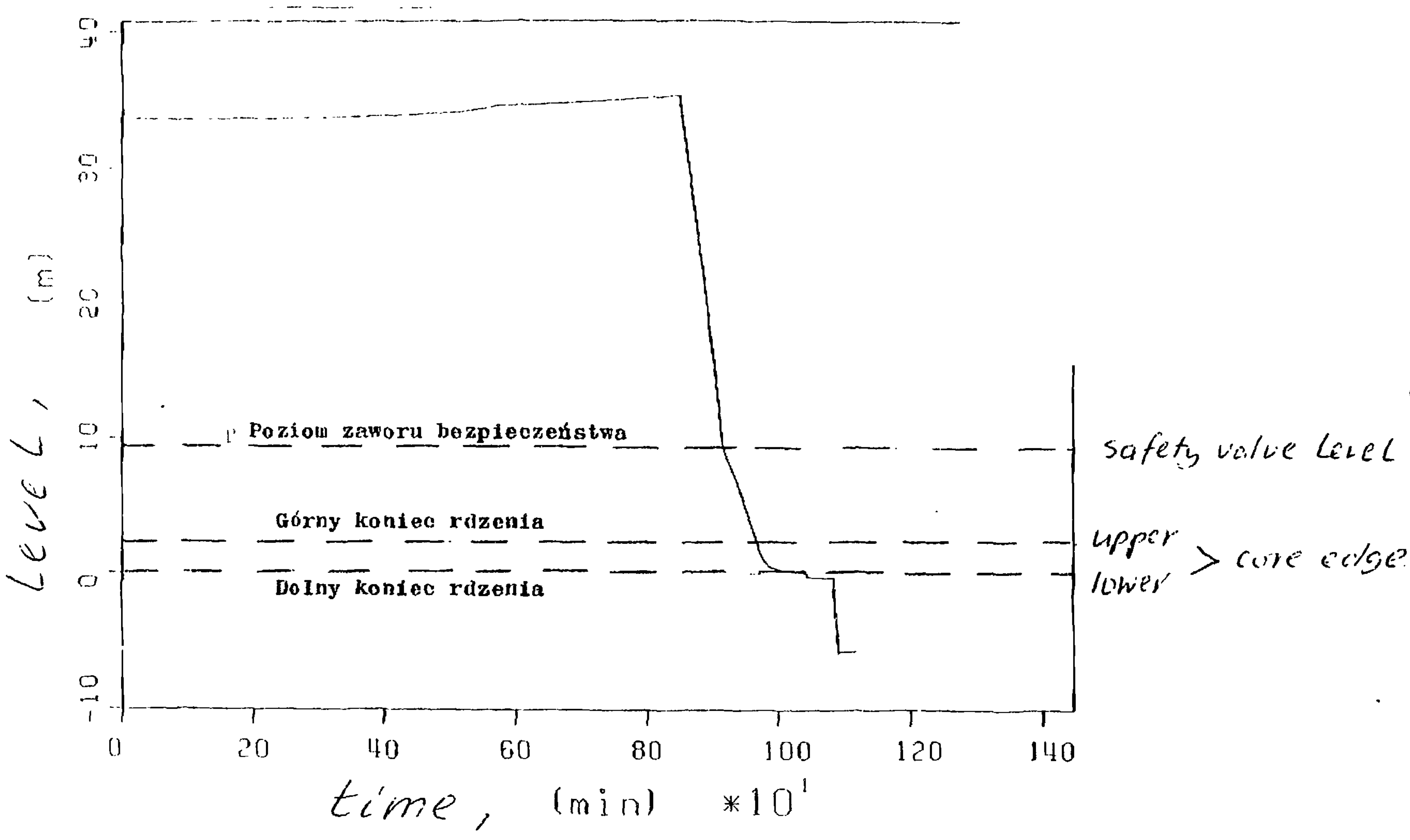


Fig. 11a Water level in RPV during TrLB accident

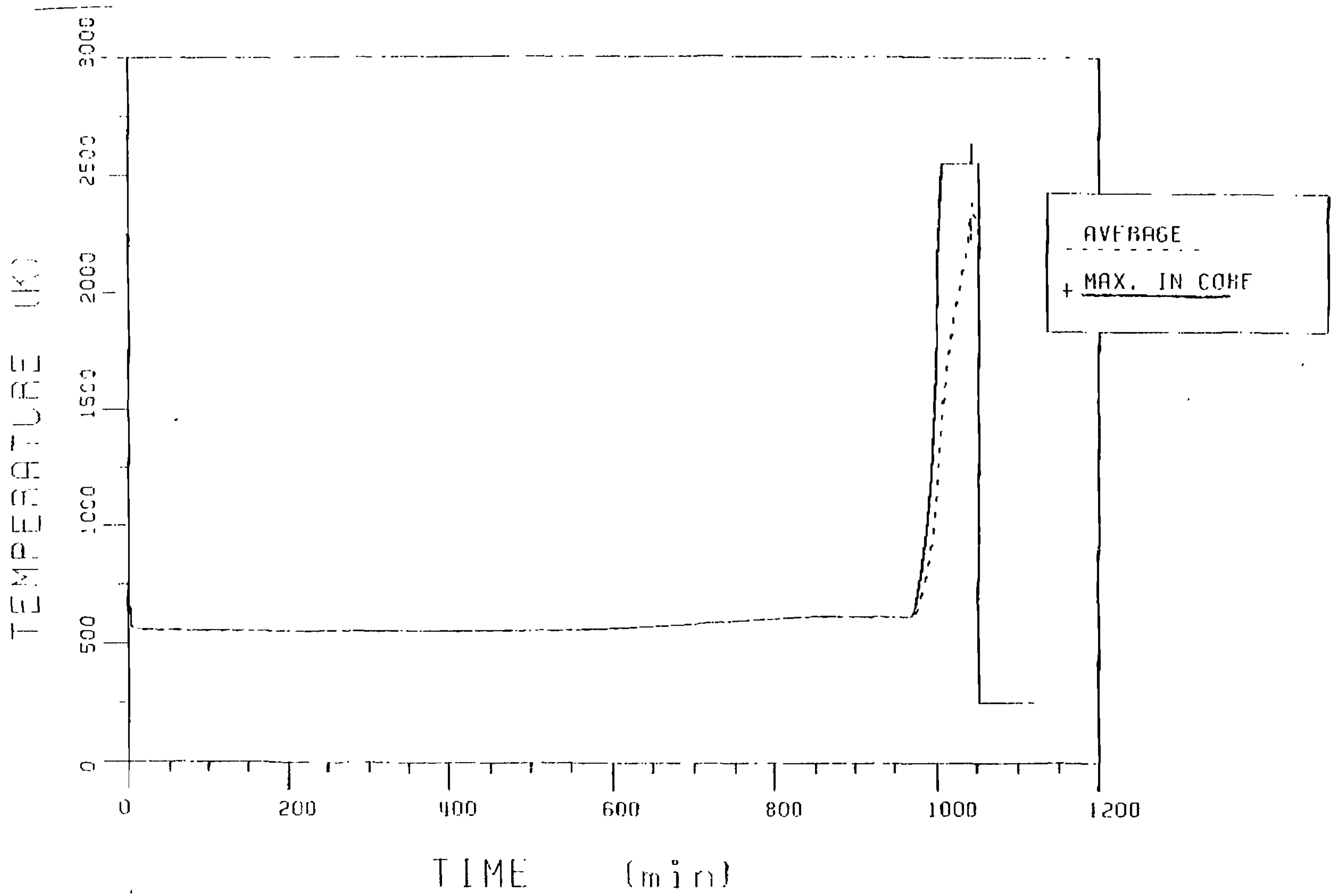


Fig. 11b Core temperature during TMLB accident

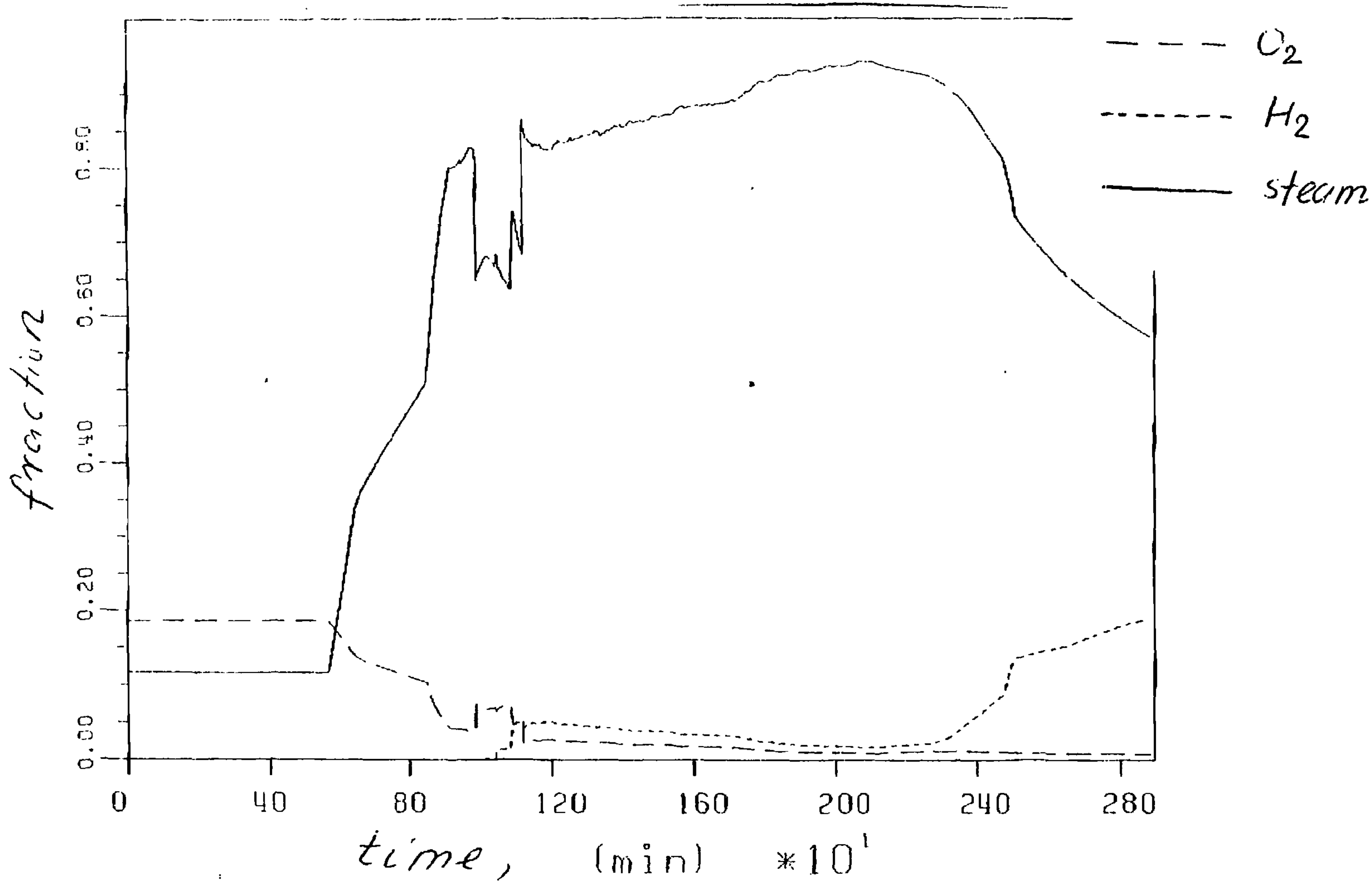


Fig. 12. Molar fractions of H_2, O_2 and steam in vol. 1, TMLB

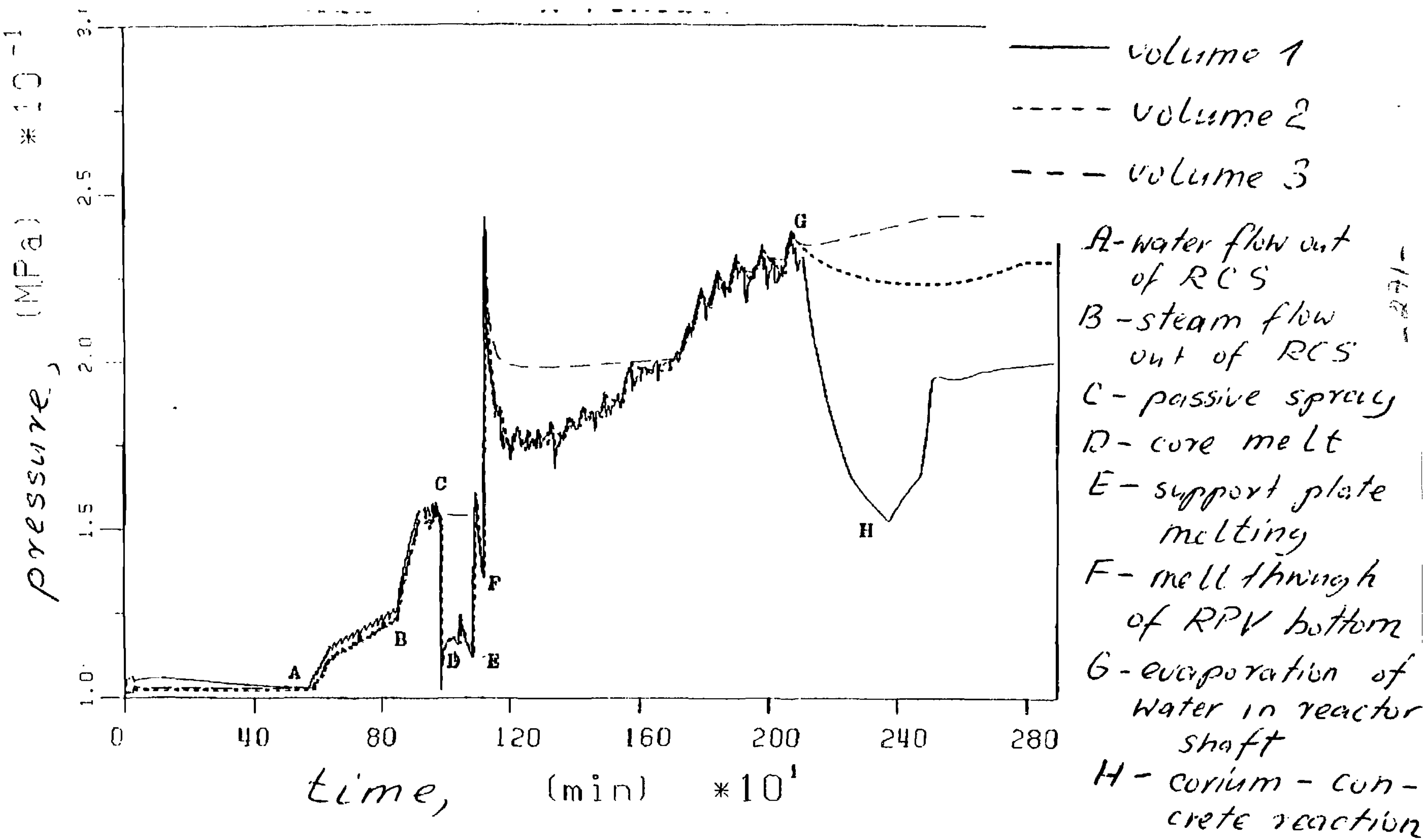


Fig. 13 Pressures in containment during TML B accident.

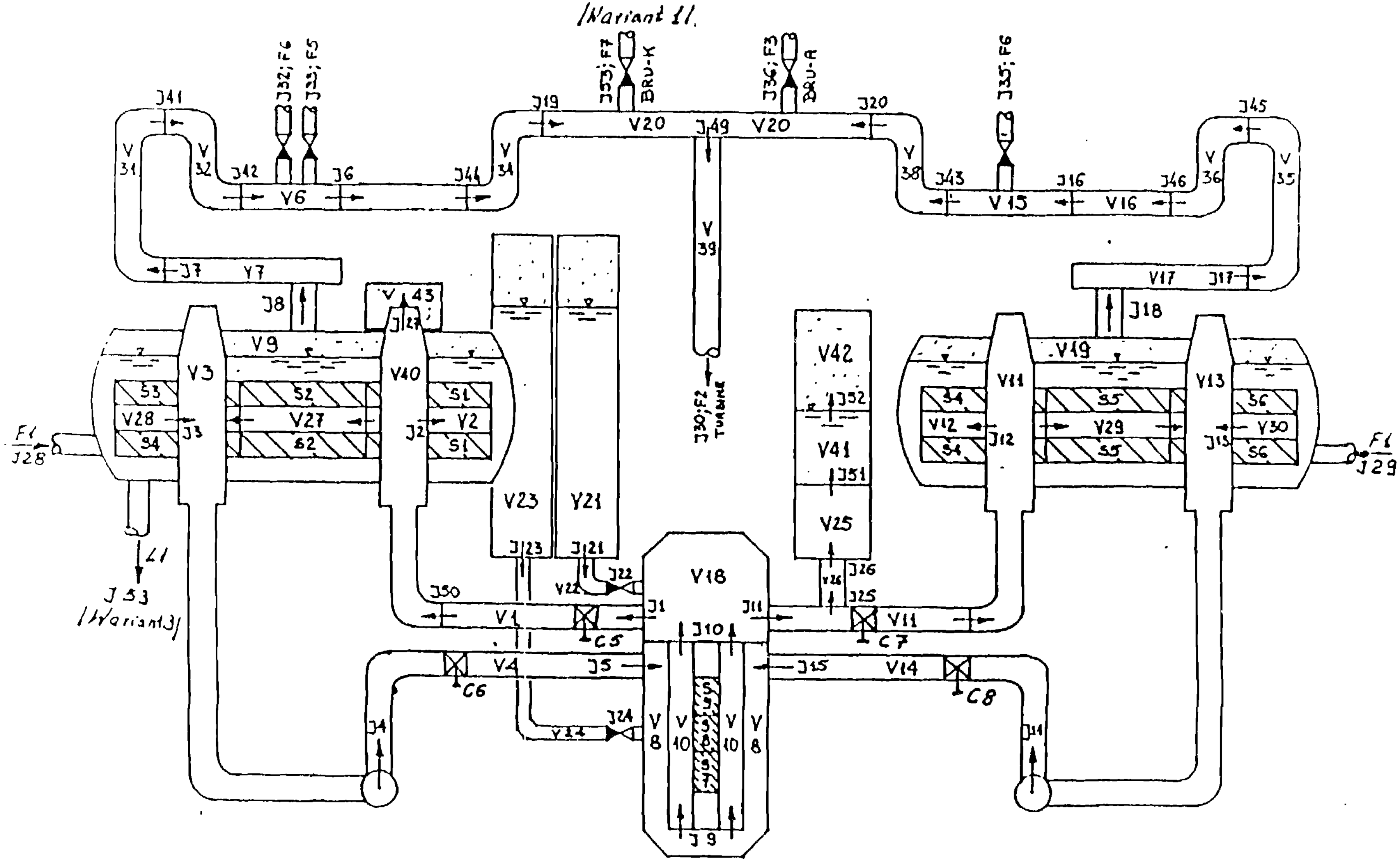
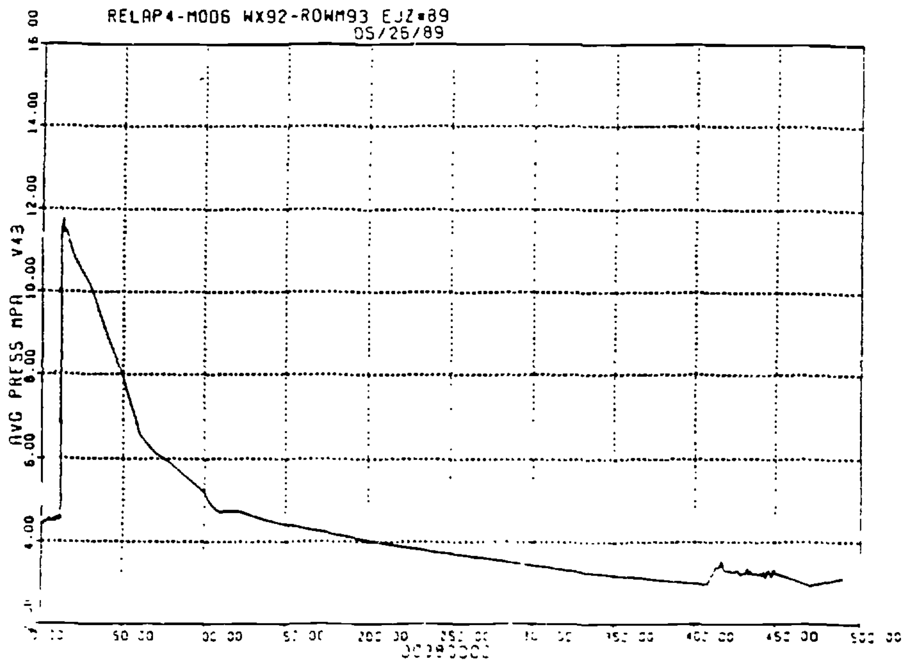
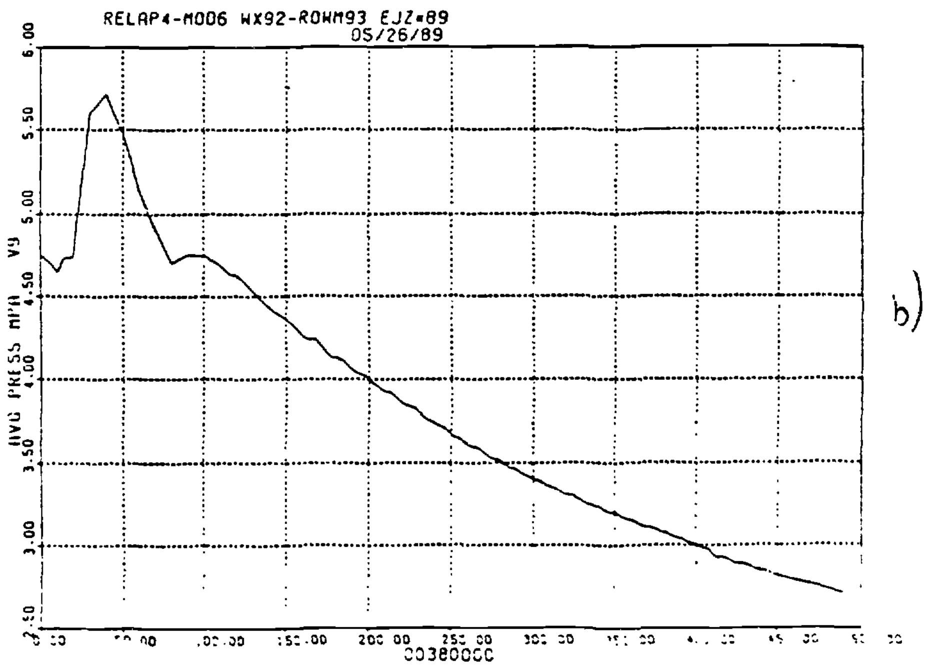


Fig 14 RCS and SG diagram adopted in analyses of RCS collector rupture in SG

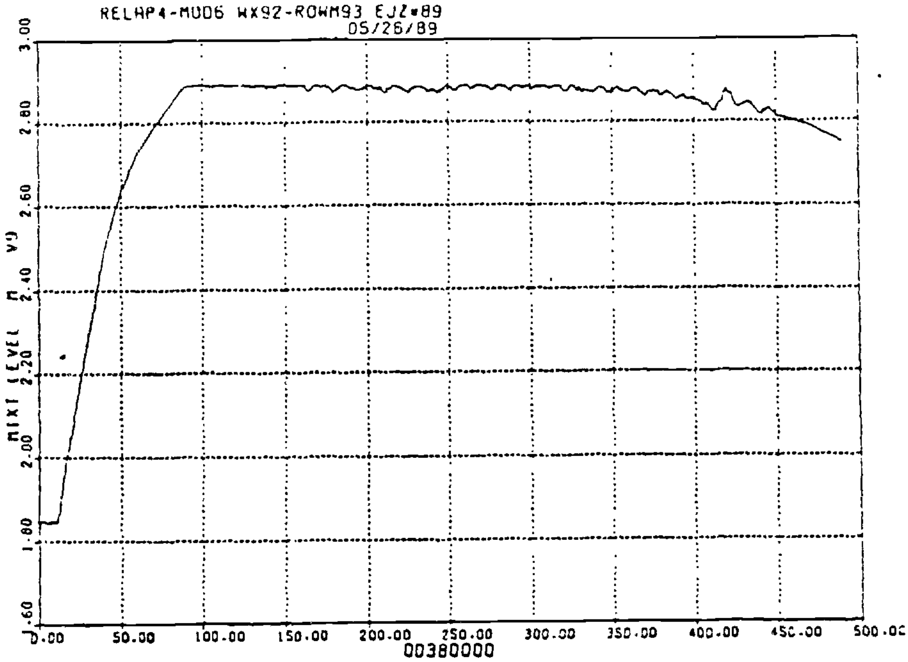


a)



b)

Fig. 15 Pressures in steam system after RCS hot collector break in SG, 100 mm equiv. diam.
a) in volume 43 (above RCS broken collector)
b) in volume 9 (in S.G.)



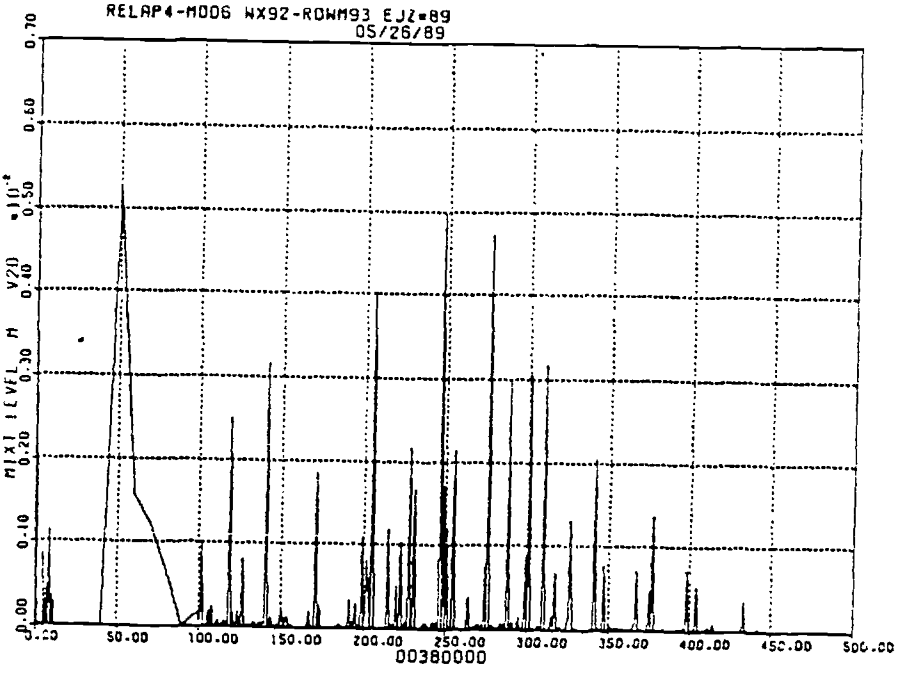
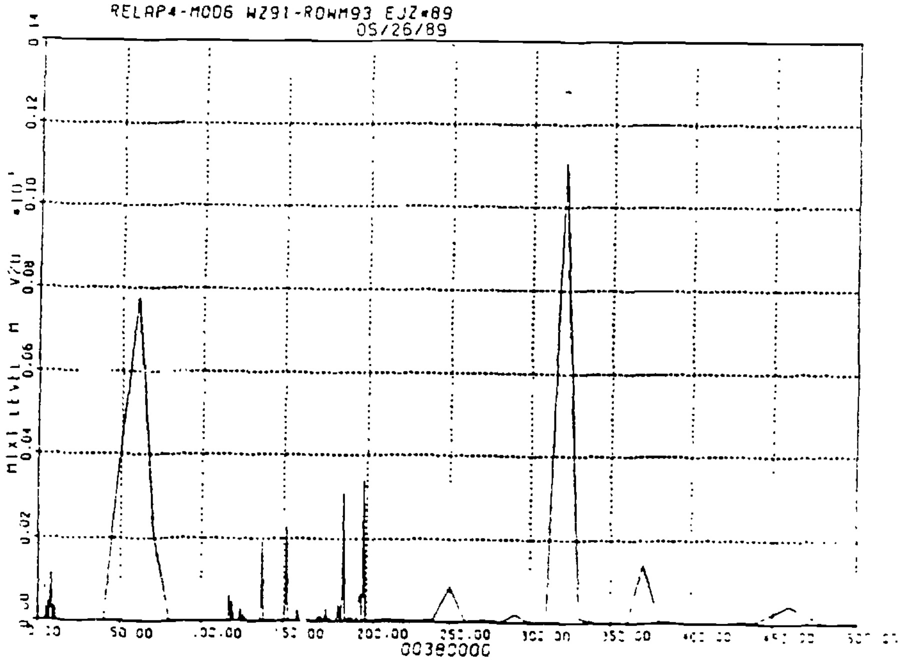
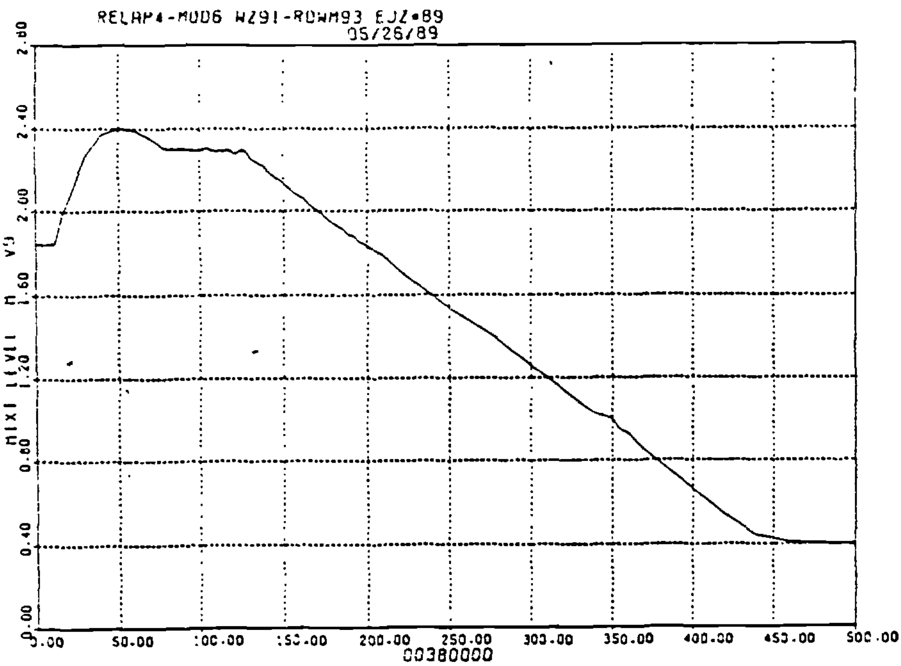


Fig. 16 Mixture levels after RCS hot collector break in S.G
a) in steam generator (V9)
b) at safety valves (V20)



a)



b)

Rys. 9. 9-W3

Fig 17. Water levels after RCS hot collector break in SG, case with blow-down pipe installed.
a) at safety valves (V20) b) in SG (V9)

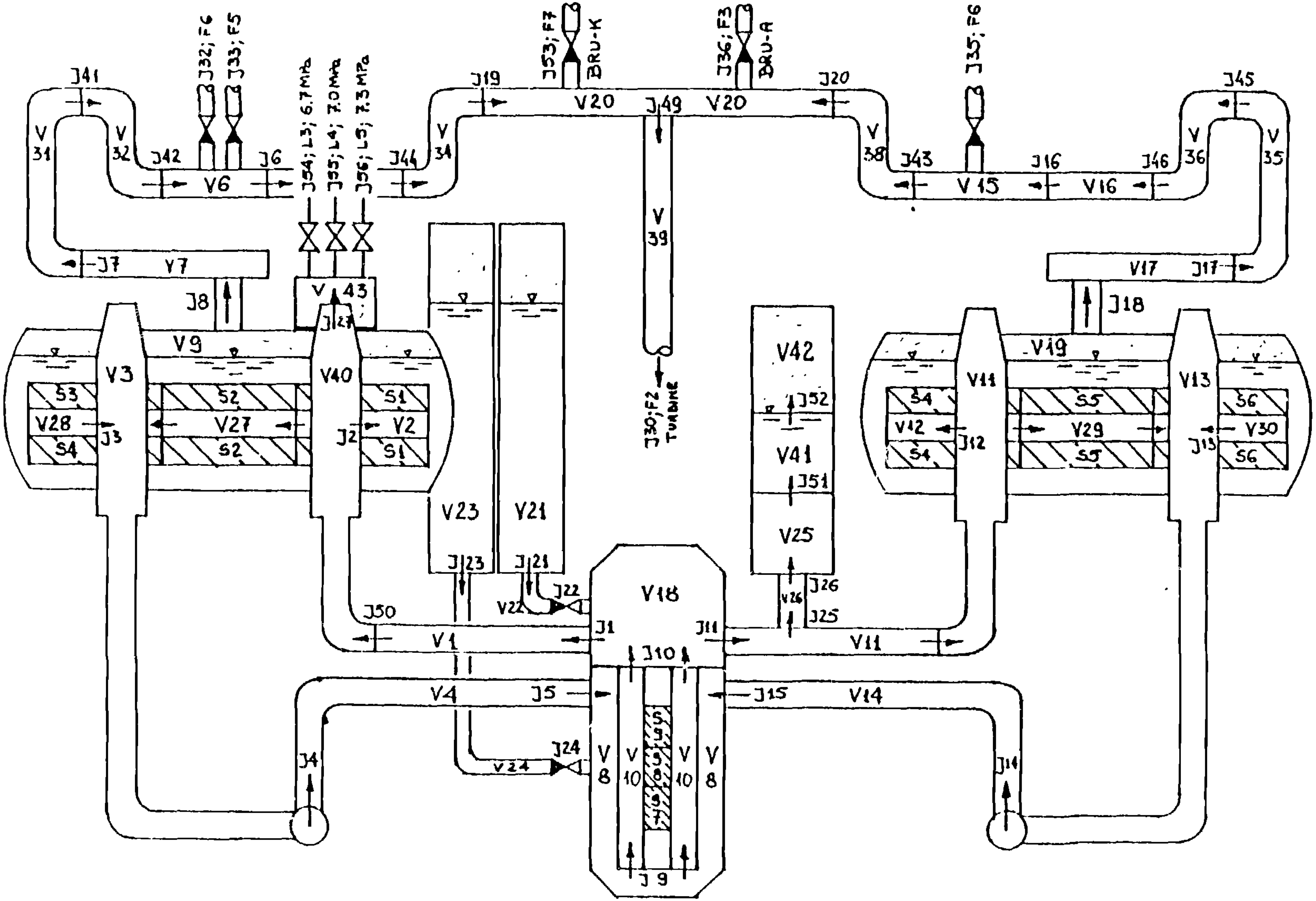


Fig. 18 RCS and S.G. diagram for RCS collector break analyses with three diaphragms on S.G standpipe added.

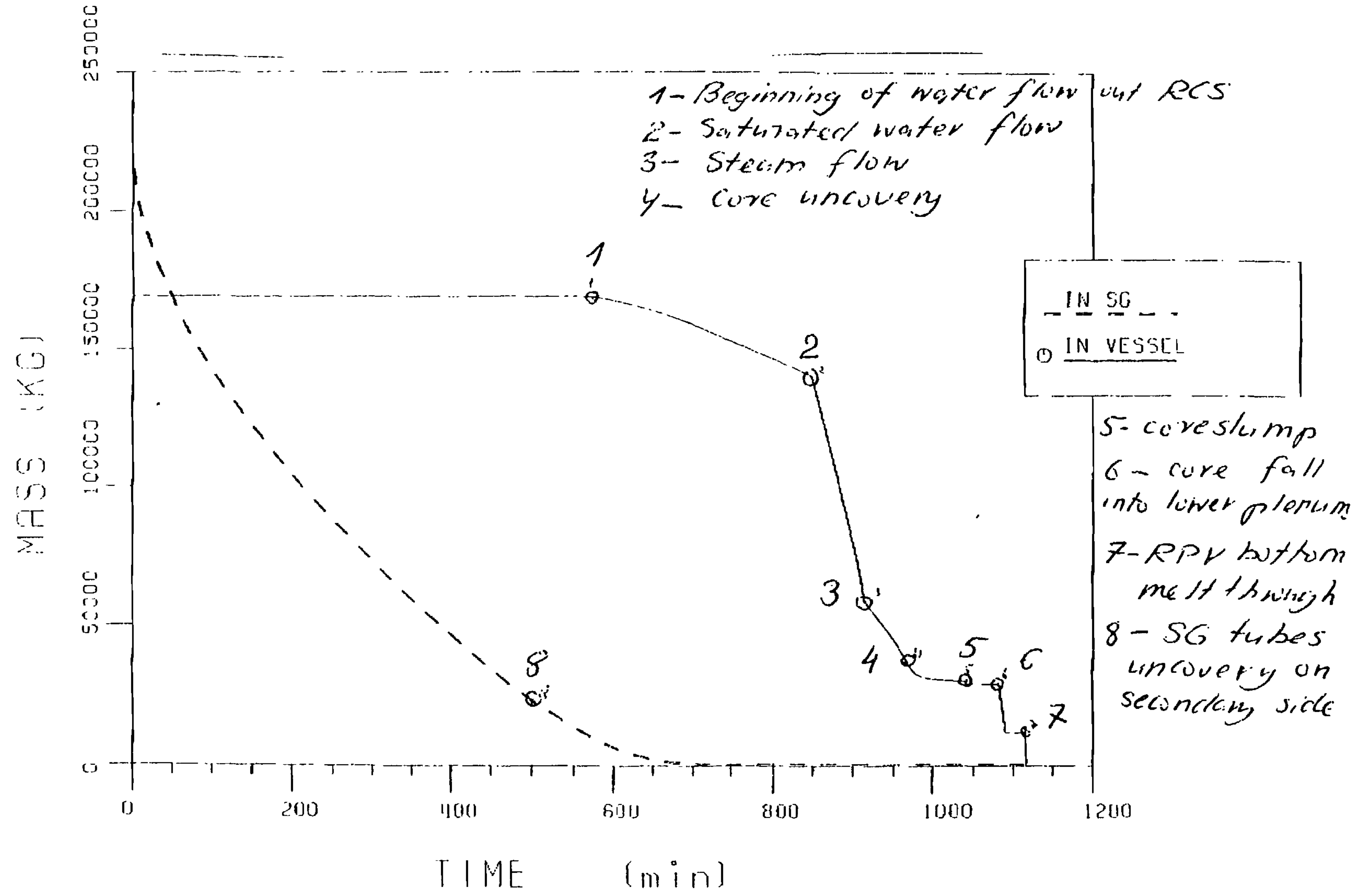


Fig. 19. Water mass in the RPV and S.G during TMLB

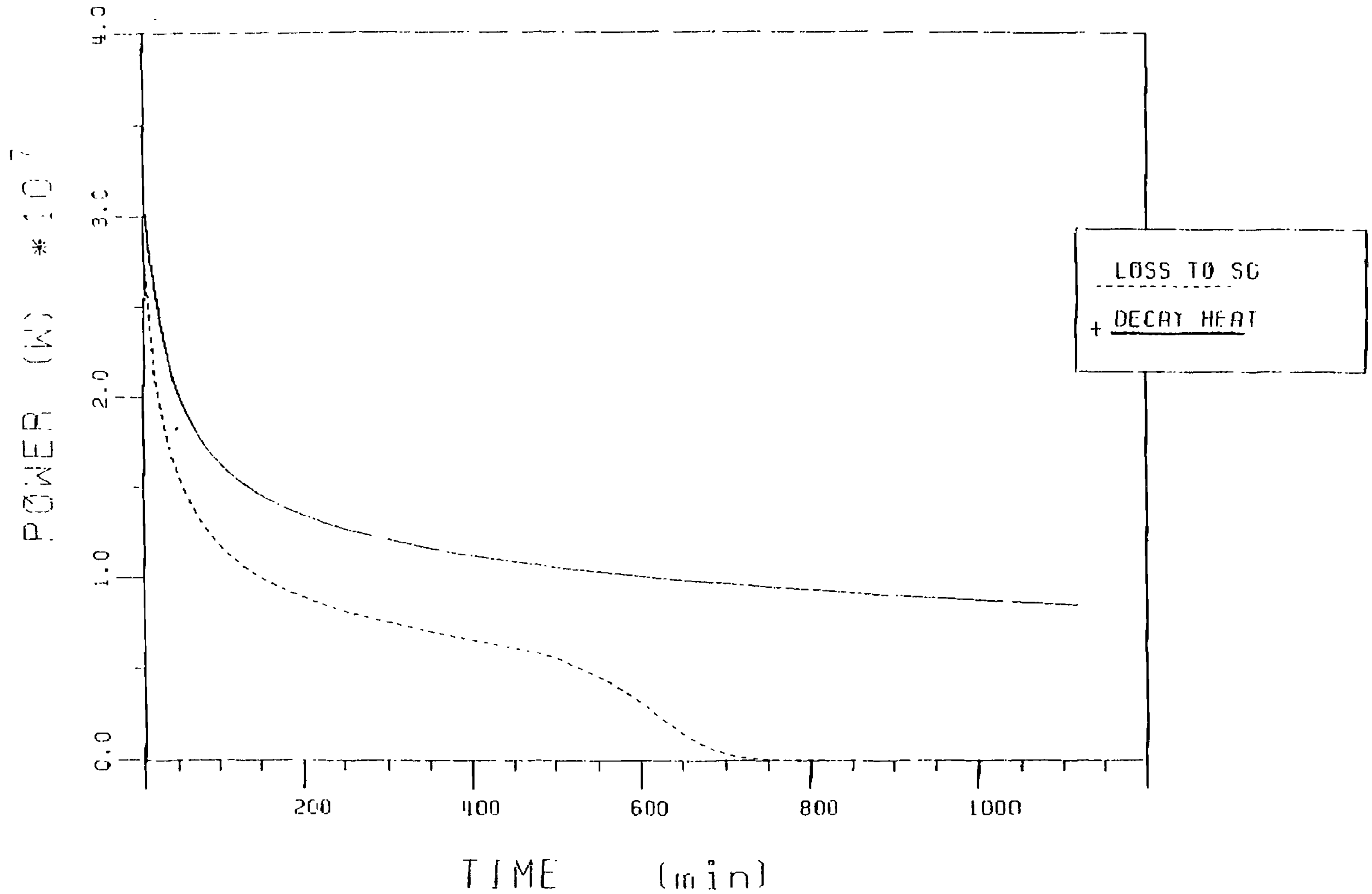


Fig 20a Energy balance in the RCS, during TMLB

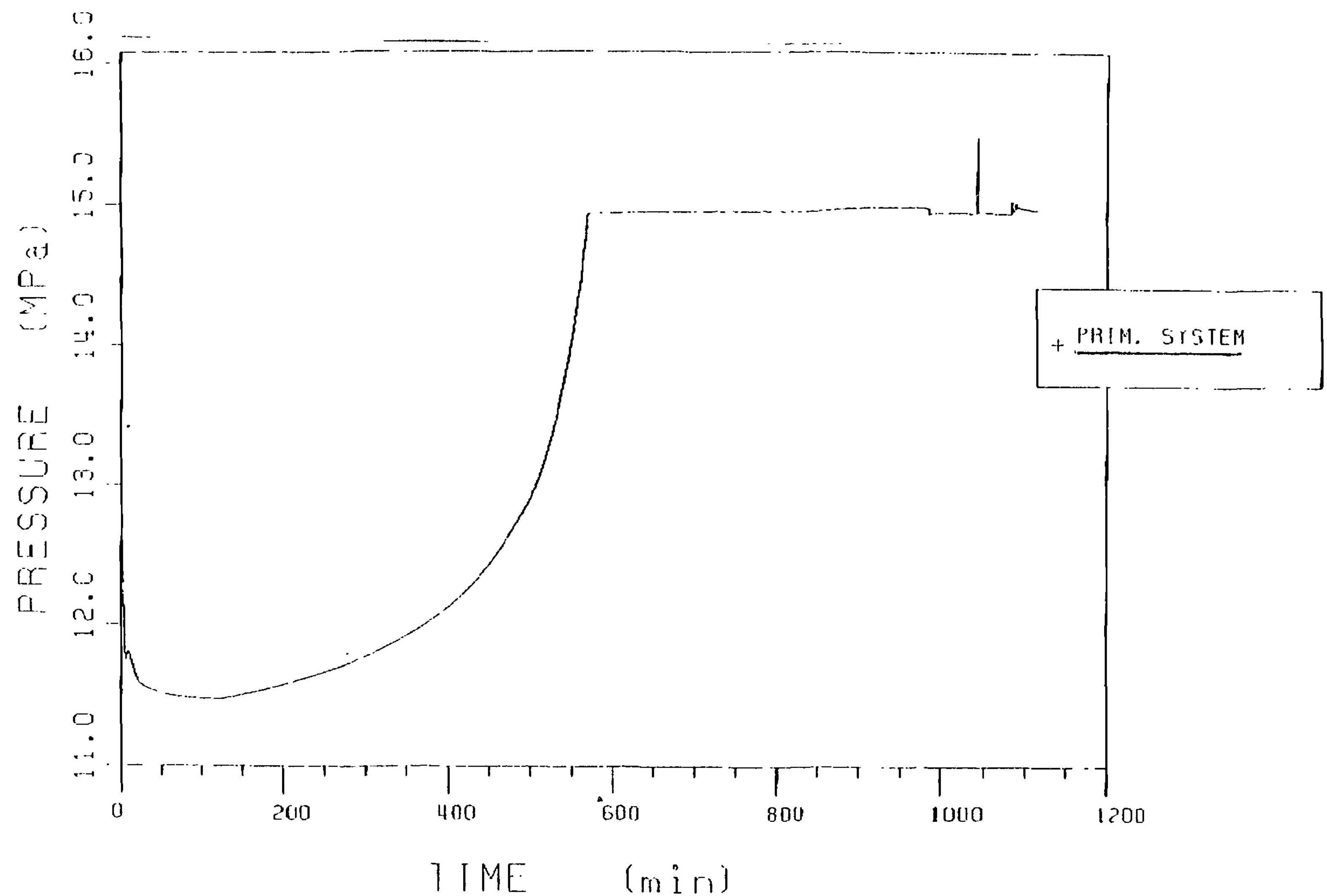


Fig 206 RCS pressure during TMLB accident

-15

GKAE
OKB "Gidropress"

EFFECT OF PASSIVE AND ACTIVE ELEMENTS OF THE SAFETY SYSTEM ON THE MOMENT OF CORE DESTRUCTION IN SEVERAL OUT-OF-DESIGN ACCIDENTS IN VVER-1000 REACTOR

T. A. Brantova, N. S. Fil', A. M. Shumskij, V. P. Spasskov

1. INTRODUCTION

Nowadays the problem of NPS safety assurance has become one of the most actual tasks as during operation of the existing NPS so in design of the new ones.

Therefore, preference is given to the inherent ultimate safety reactor plants that, in case of this or that accident situations, are brought into safe state without operator interference.

Safety is reached owing to application of the new conceptual decisions related to the safety systems and the reactor proper.

In analysis of accident situations it was determined that accident with the station blackout make significant contribution to the general frequency of core melting. The reason lies in the fact that existing safety systems are rather sensitive to events related to loss of power supply. In this connection one of the ways for considerable improvement of the safety system features is the maximum usage of the passive elements(devices) that need no power supply sources to fulfil their main functions.

In the course of accident localization and core prevention from the serious damages the operator may fase the necessity to accomodate the accident with account for the real situation. For example, during small-break accident there occurred the failure of normal and partially damaged power supply (failure of the ECCS high-head part). To put the ECCS low-head part into operation the operator shall decrease the primary pressure to setting for actuation of the hydroaccumulators and low-head pumps of emergency shutdown cooling. The question is whether there is enough time to perform this procedure, what may he use for this and what it may result in?

The present paper deals with the analysis results performed to develop recommendations for the accident accomodation during some of the VVER-1000 out-of-design accidents. Conditions of small leak from the primary circuit with failure of normal and emergency electric power supply were considered. Various versions of the accident scenarios have been compared:

- leak from the primary circuit (Dnom 25) without account for ECCS pumps;
- leak from the primary circuit (Dnom 25) without account for ECCS pumps with reactor plant shutdown cooling through the passive residual heat removal system;
- leak from the primary circuit (Dnom 25) without account for ECCS pumps with the pressurizer forced steam dumping through the safety valves (in 1800 s from the accident initiation).

2. DINAMIKA COMPUTER CODE

Calculations have been performed using DINAMIKA code that alongside with TECH-M code is the main code to analyze the transient and accident conditions of VVER-type reactors.

DINAMIKA model comprises description of operation of all the main components of the reactor plant: reactor, steam generator, pressurizer, main coolant pumps, pipelines, ECCS, system of control, protection and interlockings. Calculational diagram as used in the calculation is presented in Fig. 1. The core is presented by three channels (hot, middle and leak channels) with ten sections along the height, each loop of the main circulation circuit being "broken down" into 11 elements (3 - hot part, 8 - cold part). Loops of the main circulation circuit are arranged into three group: loop with the pressurizer was considered separately, other circulation loops are combined into two conventional loops of weight "1" and "2", respectively.

3. CALCULATION RESULTS

Calculation results are given in Figs. 2-7. Complete loss of normal and emergency electric power supply is the initial event for each of the version under consideration. The emergency protection actuates with 2.3 s delay, so the reactor power rapidly decreases to the residual heat level. Disconnection of the reactor coolant pumps and interruption of the turbine steam bleeding result in the primary pressure increase.

For the whole spectrum of VVER-1000 small accidents under consideration (equivalent leak diameter being from 0.025 m to 0.109 m) the conditions of leak with equivalent diameter 0.025 m are the governing ones because with such leak size the core uncovering occures at relatively high primary pressure excluding water supply into the primary pressure from the ECCS passive part.

The design of advanced VVER-1000 reactor (VVER-88 design) makes provision for the passive residual heat removal system that the in the case of station blackout, including loss of emergency power supply, ensures residual heat removal without the reactor core and the primary coolant pressure boundary to be failed. This system comprises the air heat exchanger placed outside the plant containment and connected with the steam and water lines of the steam generator secondary side, so steam from the steam generator is condenced in the heat exchanger and condensate returns to the steam generator water inventory.

When the station design is not provided with the above-mentioned system one may try to use steam dumping from the pressurizer steam inventory (through the safety valves or specially provided valves) for the controlled primary pressure decrease. Naturally, these valves are to be provided with reliable power supply.

As it is seen from Fig. 2 the operation of passive residual heat removal system ensures the primary pressure decrease to the level of hydroaccumulator actuation. Owing to this fact there appears a possibility to use 160 m3 water contained in four hydroaccumulators. Operation of this system provides also for keeping in the steam generators of such water inventory that is sufficient for long-term heat removal from the primary circuit (see Fig. 6) after recovery of the primary circuit emergency makeup.

When the primary pressure is dropped through the pressurizes control valves the great amount of coolant is discharged that decreases water inventory being available for heat removal during station blackout and deteriorates the core heat transfer(see Figs. 4-5).

4. CONCLUSION

Results of the analysis have shown that under the out-of-design accidents with station blackout the most preferable is application of the passive heat removal systems as it fasilitates keeping the primary and secondary water inventories sufficient to arrange the reliable and long-term cooling of the reactor core. As a result of this the moment of uncovering onset followed by the core melting is delayed for several hours that permits the operator to undertake the succesful actions on preventing the core from severe damages.

Commentary

Fig. 1 Calculational diagram

1- common steam header; 2- steam generator; 3- feedwater pipeline; 4- heat exchanging tube; 5- steam generator header; 6- main circulation pipeline "cold" leg; 7- main circulation pipeline "hot" leg; 8- reactor coolant pump; 9- injection pipeline; 10- ECCS tanks; 11- ECCS pump; 12- pressure chamber; 13- core; 14- reactor vessel; 15- collection chamber; 16- control and protection system members; 17- pressurizer; 18- heaters

Fig. 2 Pressurizer coolant pressure, MPa

- - leak Dnom 25 without ECCS pumps
x - leak Dnom 25 without ECCS pumps with shutdown cooling through the passive residual heat removal system
^ - leak Dnom 25 without ECCS pumps with actuation of 3 pressurizer safety valves

Fig. 3 Pressurizer water level, m

+ - leak Dnom 25 without ECCS pumps
x - leak Dnom 25 without ECCS pumps with shutdown cooling through the passive residual heat removal system
^ - leak Dnom 25 without ECCS pumps with actuation of 3 pressurizer safety valves

Fig. 4 Primary coolant mass, kg

\+ - leak Dnom 25 without ECCS pumps

x - leak Dnom 25 without ECCS pumps with shutdown cooling through the passive residual heat removal system

^ - leak Dnom 25 without ECCS pumps with actuation of 3 pressurizer safety valves

Fig. 5 Maximum cladding temperature (middle channel), C

\- - leak Dnom 25 without ECCS pumps

x - leak Dnom 25 without ECCS pumps with shutdown cooling through the passive residual heat removal system

^ - leak Dnom 25 without ECCS pumps with actuation of 3 pressurizer safety valves

Fig. 6 SG water level, m

\- - leak Dnom 25 without ECCS pumps

x - leak Dnom 25 without ECCS pumps with shutdown cooling through the passive residual heat removal system

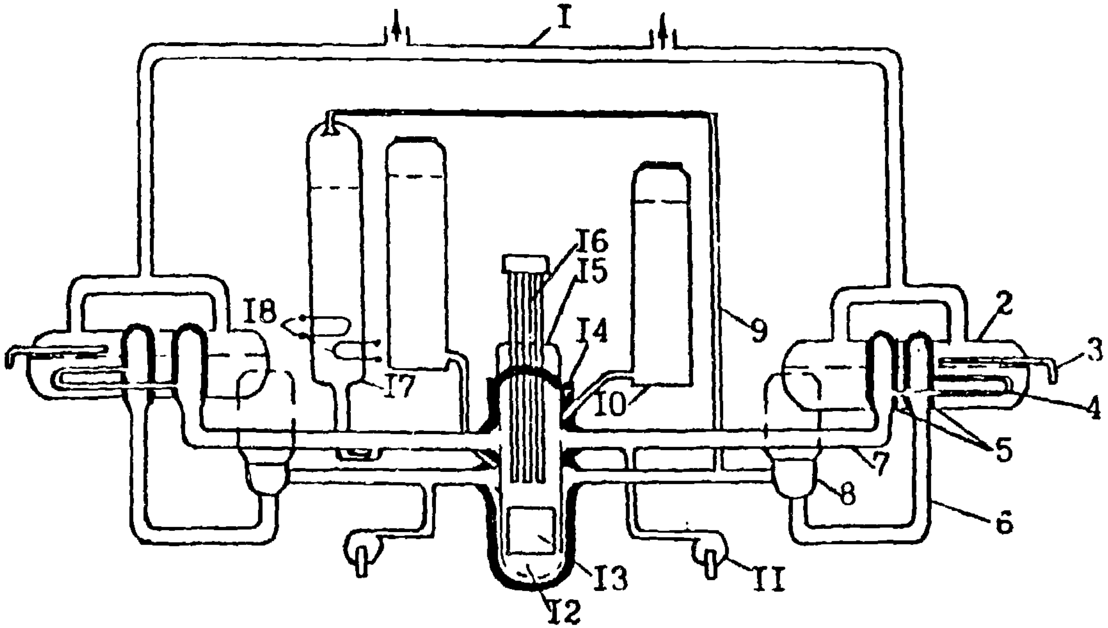
^ - leak Dnom 25 without ECCS pumps with actuation of 3 pressurizer safety valves

Fig. 7 SG coolant pressure, MPa

\- - leak Dnom 25 without ECCS pumps

x - leak Dnom 25 without ECCS pumps with shutdown cooling through the passive residual heat removal system

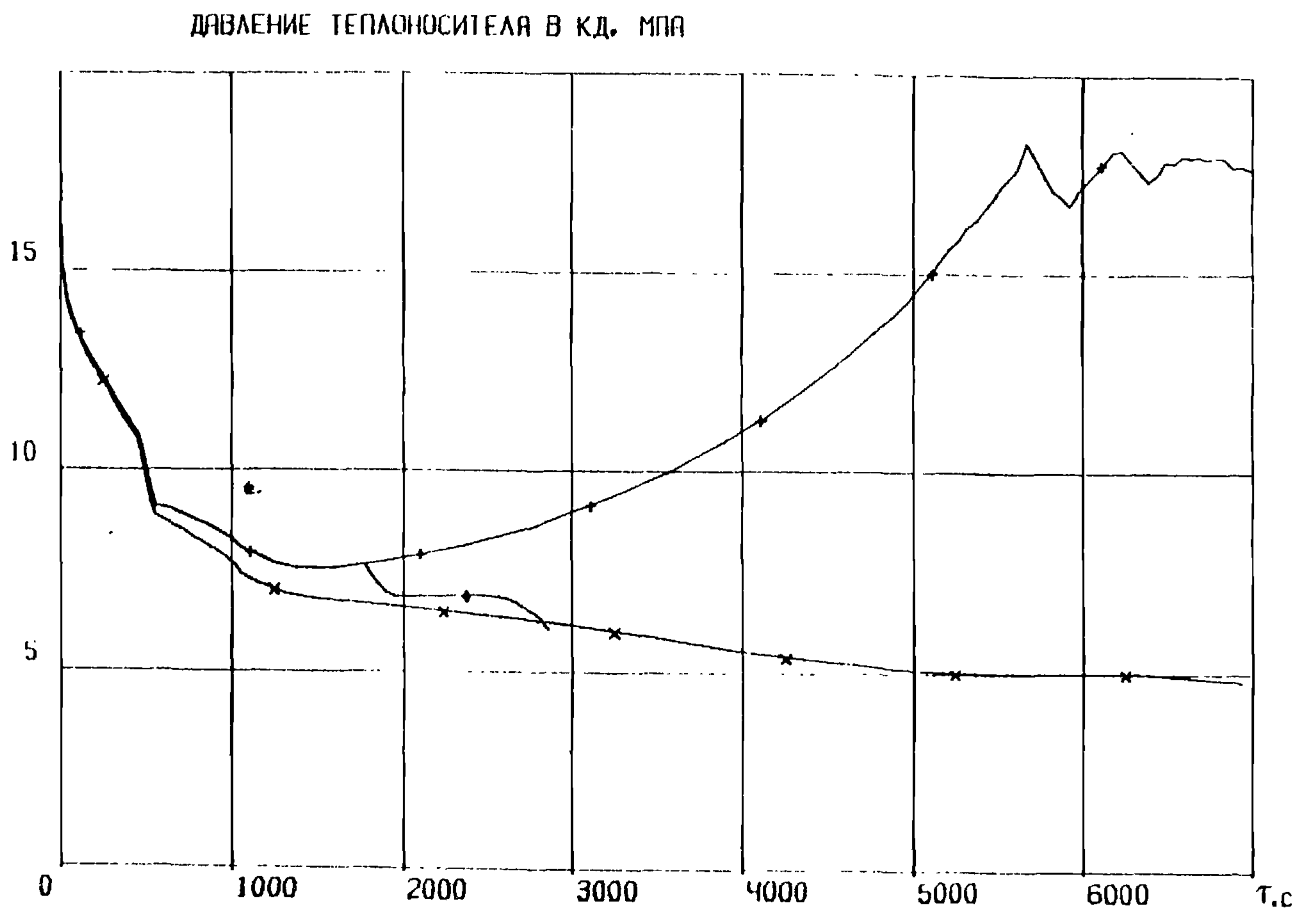
^ - leak Dnom 25 without ECCS pumps with actuation of 3 pressurizer safety valves



Расчётная схема:

I – общий паровой коллектор; 2 – парогенератор; 3 – трубопровод питательной воды; 4 – теплопередающая трубка; 5 – коллектор парогенератора; 6 – "холодная"ветвь ГЦТ; 7 – "горячая" ветвь ГЦТ; 8 – ГЦН; 9 – трубопровод впрыска; I0 – ёмкость САОЗ; II – насос САОЗ; I2 – напорная камера; I3 – активная зона; I4 – корпус реактора; I5 – сборная камера; I6 – органы системы управления и защиты; I7 – компенсатор давления; I8 – нагреватели

Рис. I.

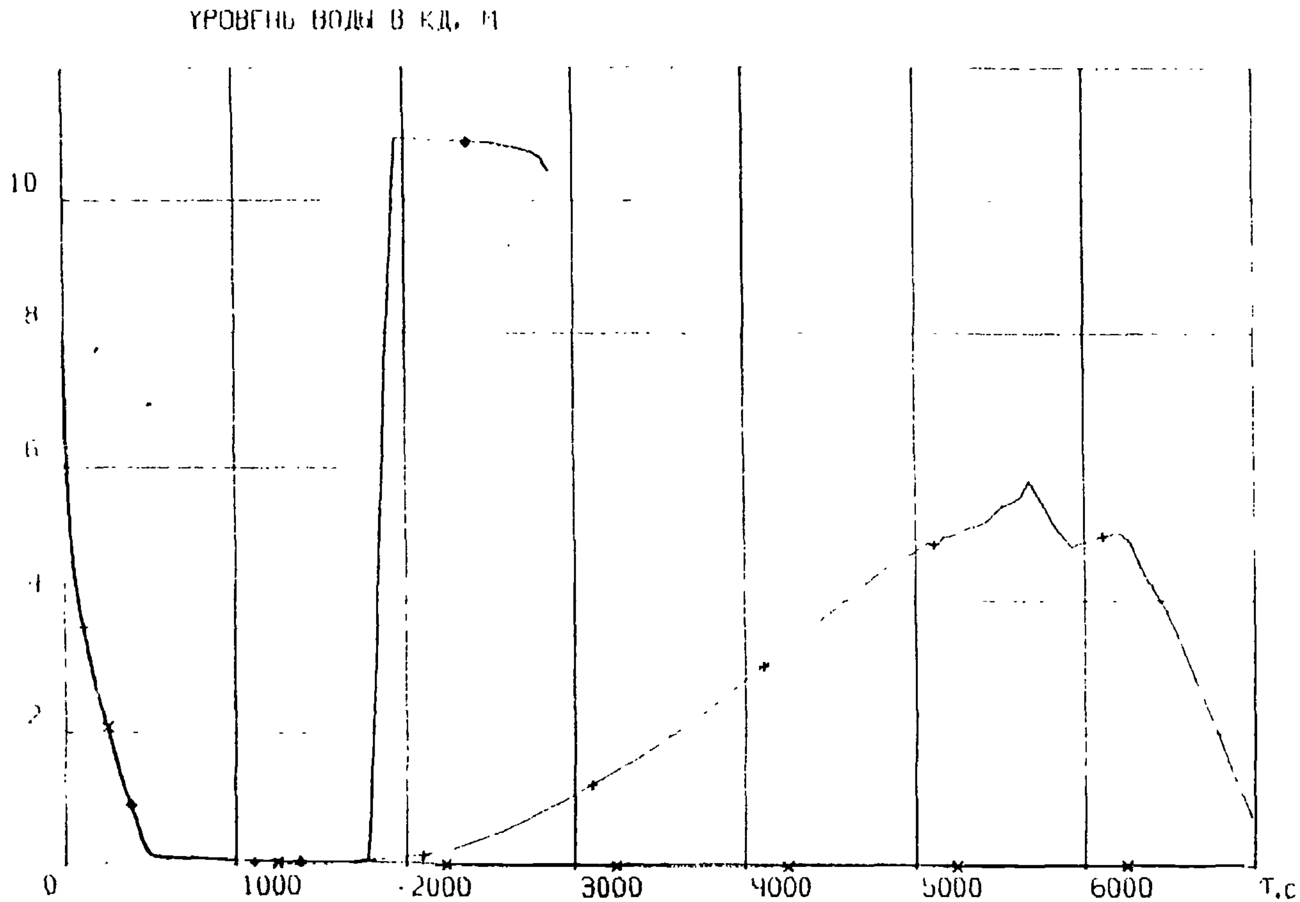


+ ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ

x - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ С РАСХОЛАЖИВАНИЕМ ЧЕРЕЗ ТЕПЛООБМ

♦ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ СО СРАБАТЫВАНИЕМ 3 ПК КД

Рис. 2

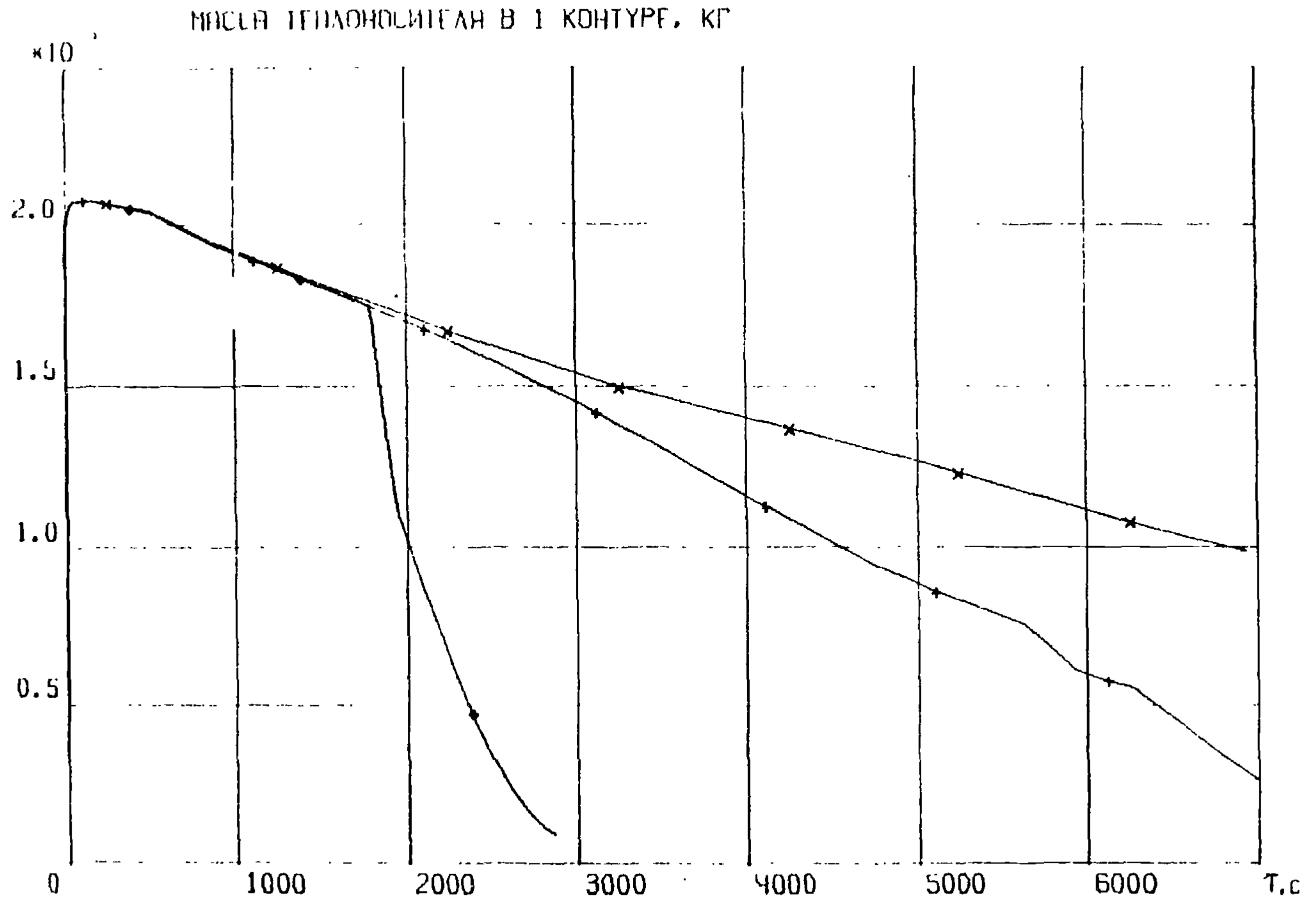


+ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ

× - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ С РАСХОЛАЖИВАНИЕМ ЧЕРЕЗ ТЕПЛООБМ

♦ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ СО СРАБАТЫВАНИЕМ 3 ПК КД

Рис. 3



\+ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ

x - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ С РАСХОЛАЖИВАНИЕМ ЧЕРЕЗ ТЕПЛООБМ

◆ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ СО СРАБАТЫВАНИЕМ 3 ПК КД

Рис. 4

МАКСИМАЛЬНАЯ ТЕМПЕРАТУРА ОБОЛОЧКИ СРЕДНЕГО ТВЭЛА,ГРАД

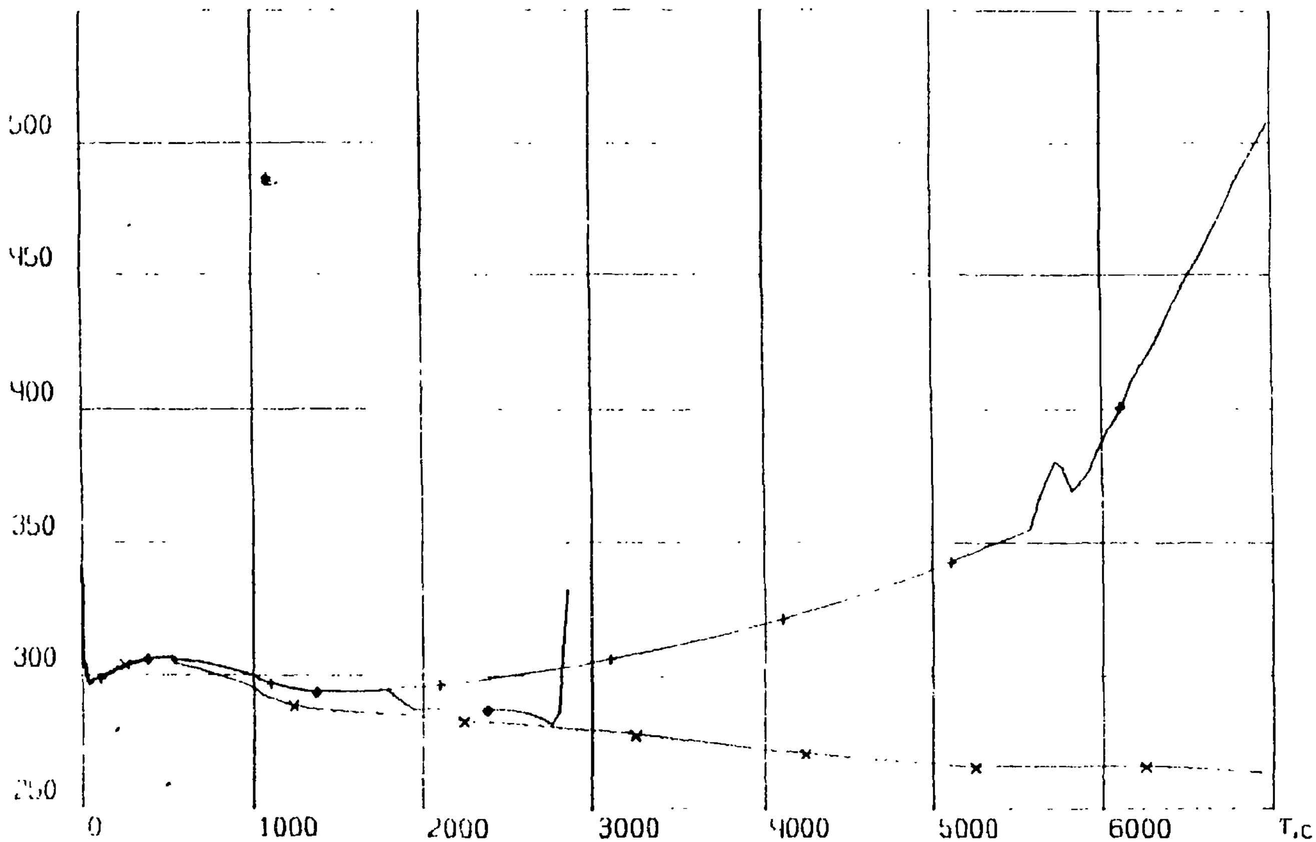


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Рис. 5

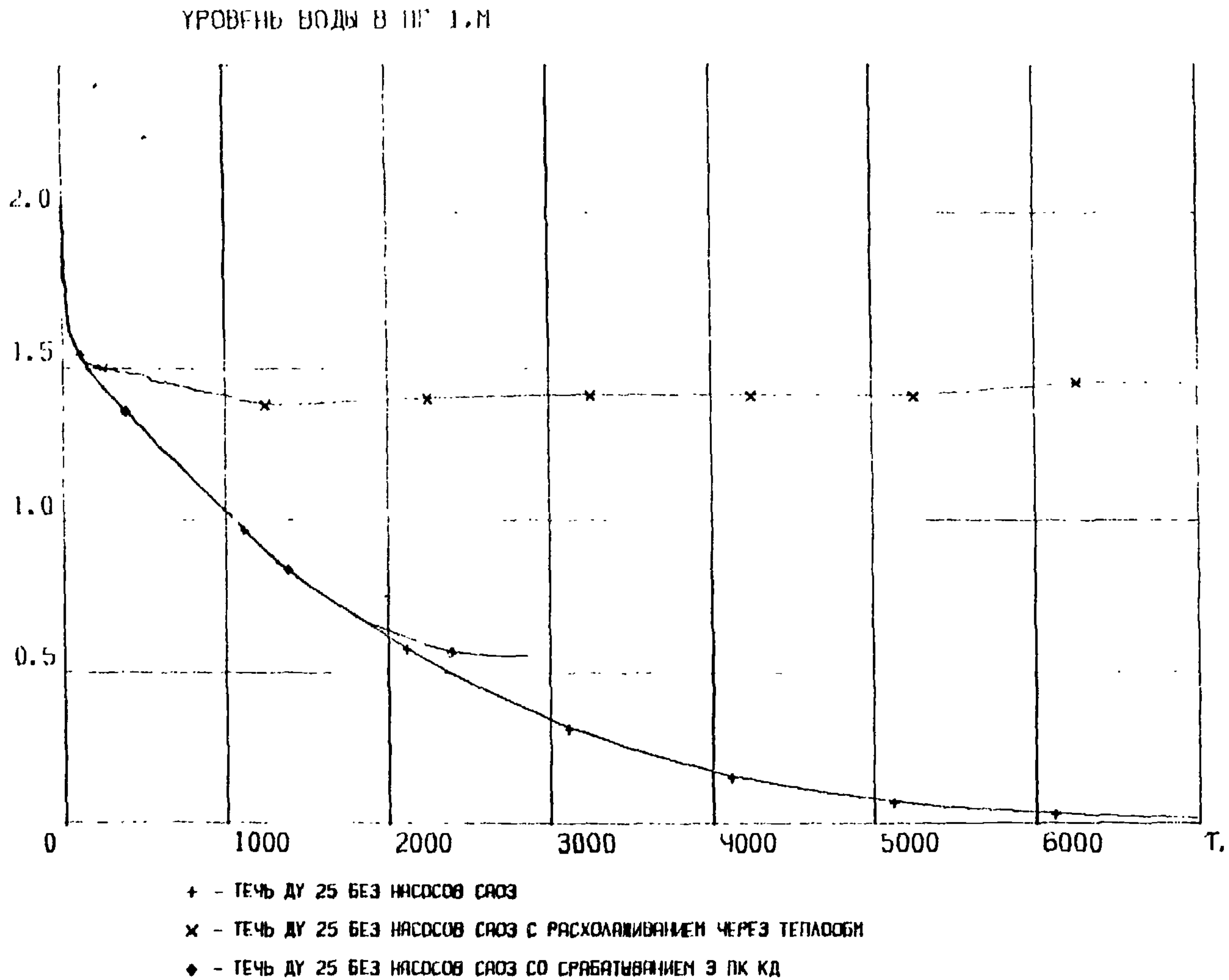
УРОВЕНЬ ВОДЫ В ПГ 1,М
2.0
1.5
1.0
0.5
0
1000
2000
3000
4000
5000
6000
Т,с
+ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ
× - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ С РАСХОЛАЖИВАНИЕМ ЧЕРЕЗ ТЕПЛООБМ
◆ - ТЕЧЬ ДУ 25 БЕЗ НАСОСОВ САОЗ СО СРАБАТЫВАНИЕМ 3 ПК КД

Рис. 6

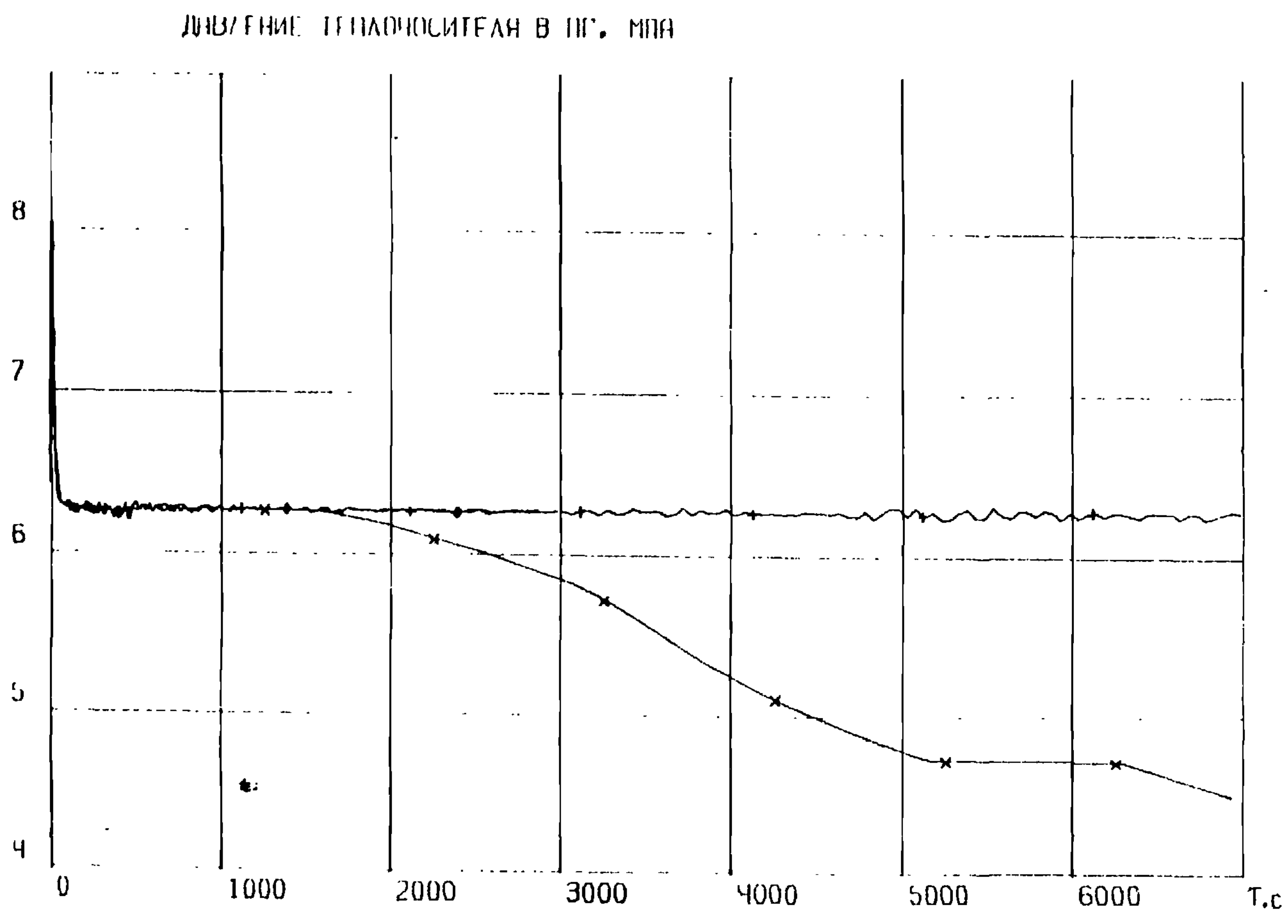

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Рис. 7

NEW TECHNICAL SOLUTIONS ELABORATED IN VVER REACTOR NPS DESIGNS ON PREVENTING THE SEVERE ACCIDENTS AND THEIR CONTROL

USSR „Atomenergoproekt" Kiev Shenderovich

The present situation having arisen with development of nuclear energetics throughout the world and particularly in the USSR requires the further steps directed for increasing of safety.

In spite of the fact that the activities of this direction have been pursued in the USSR during the whole period of development of nuclear energetics they particularly have been activized after the accident at Chernobyl NPS.

Prevention of "severe" accidents and overcoming of consequences of these accidents are the most important directions in increasing of safety.

At present the new normative document "General criteria on ensuring NPS safety" (OPB-88), which is the main normative-technical document of the top level and which regulate the problems of safety resulting from NPS specifity as the possible source of radioactive exposure to personnel, population and environment, is in valid in the USSR.

The following requirement is one of the main difference of the given document from the document which has been in force before: consideration of "heavy" (as per the normative document term - - "severe") accident; availability of the measures on the severe accident control in the design and establishing of the severe accident definite criteria.

In this case the "design accident" term means the accidents for which the design defines the initial events and the final states and stipulates the safety systems providing limitation of its consequences by the limits predetermined for such accidents based on a principle of a single failure or one personnel error being independent from the initial event.

"Severe accident" term means the accidents aroused by the initial events unaccounted for the design accidents or accompanyed by the additional safety system failures in comparison with the design accident beyond the single failure, by the personnel erroneous solutions tending to the core serious failure or its damage, consequences mitigation of which is achieved by means of accident control and/or by implementation of the measures on personnel and population protection.

In this case non-exceeding of probability of heavy damage or failure during the reactor core severe accident with value of IO^{-5} for one reactor per year is considered to be as a recommended criterian.

For selection of a population evacuation necessity exception criterion it is recommended to aspire to the fact that probability of limited emergency release will not exceed value of IO^{-7} for one reactor per year.

VVER-IOOO Units being under operation and under construction in conformity with the designs elaborated in the USSR before are characterized by rather high safety level and are comparable in this part with the similar Units of the other countries.

However, increasing responsibility for oecological consequences of nuclear energetics tends to necessity of the further elaborations in this direction.

In serial VVER-IOOO NPS designs elaborated at present the completed measures intended for increasing of safety with the problems given below are considered.

While studying of these measures elaboration of such technical solutions which may be implemented during the course of reconstruction of VVER-IOOO NPSs in operation and under construction is the most important task. This problem is highly actual if we take into consideration amount of operating Units and Units under construction.

Analysis of severe accidents for VVER-IOOO Unit permits to single out the groups of the following most typical accidents:

- black out of NPS with failure of all available a.c.emergency sources.

Diesel-generator plants being independent in each safety system are provided as such sources in NPS designs.

- failure of reactor control rod system (ATWS)

- insertion of positive reactance

- failure of heat removal along circuit II

- leakages from circuit I to circuit II

- spectrum of circuit I leakages with failure of high and low pressure emergency core cooling system.

The measures which may be subdivided into two groups are stipulated in the elaborated designs.

Group I. Measures on preventing of accidents.

I. Use of the improved reactor plant with perfected nuclear physical features and increasing of its safety at the expence of the following:

- decreasing of reactor core power peaking

- increasing of effectiveness of the reactivity effect mechanical system permitting to provide reactor core subcriticality under emergency conditions with cooling the reactor plant down to I00°C without boron injection (amount of the control rods has been increased from 6I up to I2I).

2. Use of the primary circulating pumps with minimum seal leakages permitting to provide the circuit tightness under the station black out during the period of time up to 24 hours.

3. Use of safety channel element availability monitoring system with giving the generalized signal on channel readiness to the main control room.

Group 2. Measures on severe accidents control and mitigation of their consequences.

I. Use of the passive residual heat removal system (PRHRS).

I.I. The passive residual heat removal system (draft I) is provided for residual removing of heat from the reactor plant over a long period of time under complete loss of power supply including emergency sources on retention of circuit I and II pressure-tight.

PRHRS is a closed system with the natural circulation. Heat removing to the environment is realized within the range of $\pm 50^{\circ}C$ temperature.

I.2. The system operates without power supply and provides circuit II steam cooling in volume required for residual heat removing at level of 2% from rated power.

I.3. Change-over of the system from the stand-by mode to the operation mode is carried out automatically during NPS black out and failure of reliable power supply sources.

I.4. Motion of coolant and air is carried out under conditions of natural circulation. For this purpose the heat exchangers erection and arrangement of the draught air lines after the heat exchangers are provided on the upper levels of the reactor bay (above the level of steam generator erection).

I.5. The system consists of four independent circuits.

The calculated substantiations being carried out show the effectiveness of the given system operation even at circuit I leakages. Availability of system permits to increase time of achieving the parameters of defected fuel elements. In this case the given time delay depends on leakage extent.

2. Passive core flooding system (PCFS).

2.I. The passive core flooding system is provided for preventing the fuel element failure during the course of accident with circuit I leakages, with failure of high and low pressure passive residual heat removal system or complete loss of a.c. power supply sources (including emergency sources).

2.2. The passive core flooding system shall ensure cooling medium supply following response of hydrotanks provided for compensation of circuit I coolant loss into the leakage thereby avoiding the reactor core uncovery.

2.3. At present the system features calculated substantiations are conducted based on conservative approach with taking into consideration the passive residual heat removal system (PRHRS) operation analysis under these conditions.

For the first priority designs the following are stipulated:

- erection of additional hydrotanks of $200m^3$ capacity for I,2 + I,5 MPa pressure, which operates on completion of water outflow from the "standard" hydrotanks;

- the further filling up of the reactor vessel is carried out from the cooling and reloading pond following water discharge from these hydrotanks.

For considered 24 hours accident $500m^3$ water storage is created at the expense of increasing the medium level in the cooling and reloading pond under operation and constant flooding of the reactor internals reloading shafts.

Such solution is realized at existing constructive solutions of handling equipment and doesn't require the principle change of the fuel reloading technology.

2.4. Passive successive connection of additional hydrotanks and the pond gravity tanks are stipulated. At present the process parameters are specified by means of calculations.

3. System of quick boron injection into circuit I (SQBI).

3.I. System of quick boron injection (draft 2) is provided for quick putting the reactor into subcritical state by means of 40-60g/kg concentrated boron acid solution supply under conditions of control rod system failure.

3.2. System of quick boron injection consists of 4 channels connected to circuit I loops "cold" legs. Each channel consists of

$8m^3$ hydrotank connected with the primary circulating pump inlet and outlet. Medium supply is provided at the expense of primary circulating pump "rundown".

3.3. The system operation is provided automatically on the engineered safeguards signal and the signal of exceeding the neutron power or at complete black out including the emergency sources.

4. System of pressure shedding and cleaning of releases.

4.I. The system is provided for preventing of excessive pressure increase inside containment above permissible which may appear during the severe accidents and tends to loss of containment functional properties. The containment used at present time at VVER-IOOO NPS has been calculated for 5,OMPa.

4.2. The system operation is provided to be automatical at increasing of pressure inside the containment up to 5,0 MPa without power supply from external sources.

Personnel control is possible during the accident taking place over a long period of time.

4.3. Operation without change of filtering medium, service and power supply during 2-3 days is provided.

4.4. Degree of catching the different forms of iodine and aerosols is not less than 99,9%.

Treatment of discharges is carried out in a single-case apparatus in an area with biological protection. Location is provided on the lower levels outside of containment. At the operating NPS the filter may be located in an annex to the reactor building.

5. Hydrogen suppression system.

5.I. The system is provided for preventing of formation of hydrogen explosive concentration and unorganized burning in the areas inside NPS containment under emergency conditions including severe accidents.

5.2. Hydrogen after-burning is carried out with using the passive catalytic hydrogen after-burners operating without power supply.

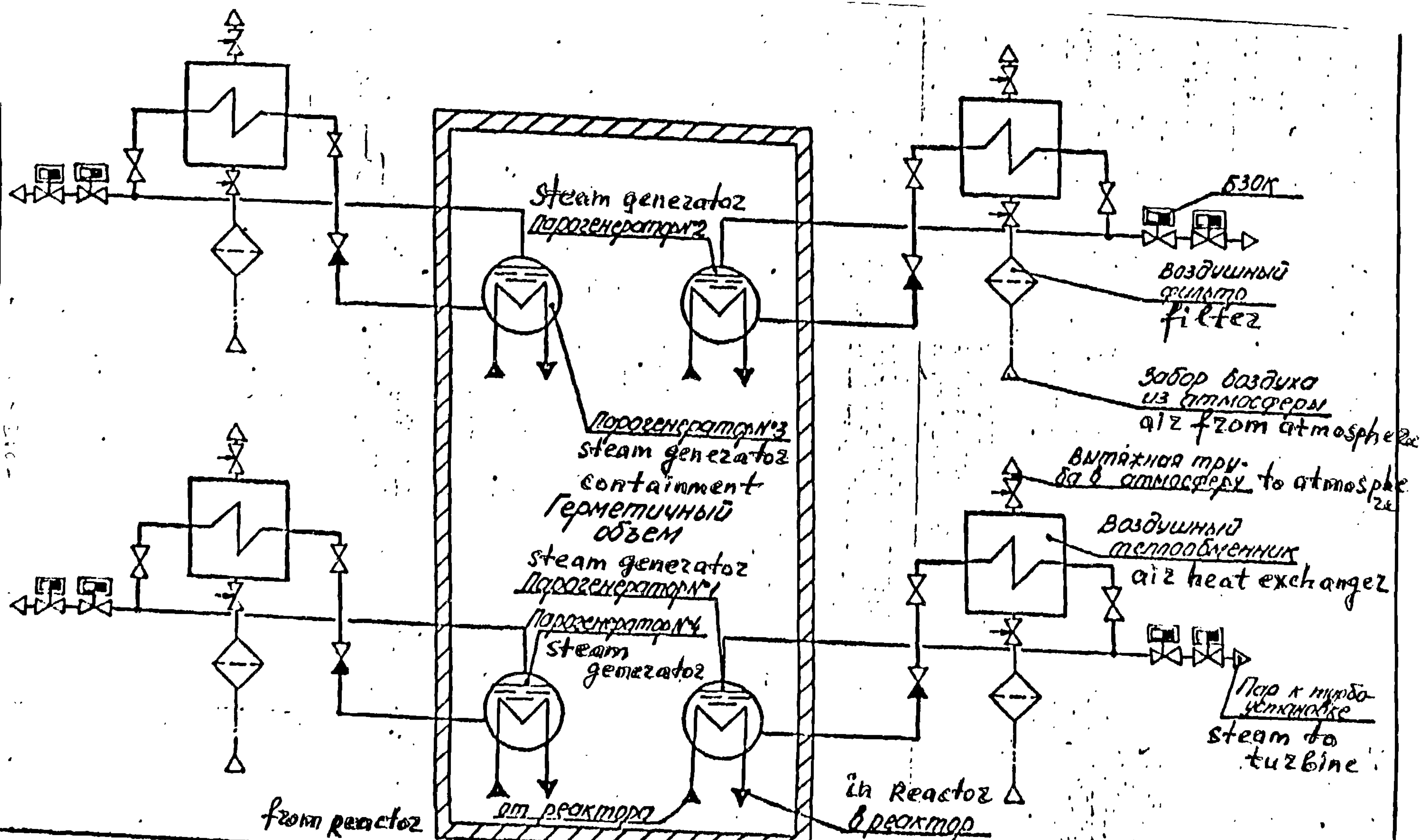
Structure of after-burners and their operational conditions are elaborated at the present time by the designers.

6. Arrangement of anti-emergency centre on the NPS site and NPS outside is stipulated for monitoring of emergency Unit state under severe conditions.

The mentioned centre provides the functions of control on the Unit state and conducting of technical and organizational measures for which purpose it is equipped in particular with off-line power supply source, information reprocessing system and communication means.

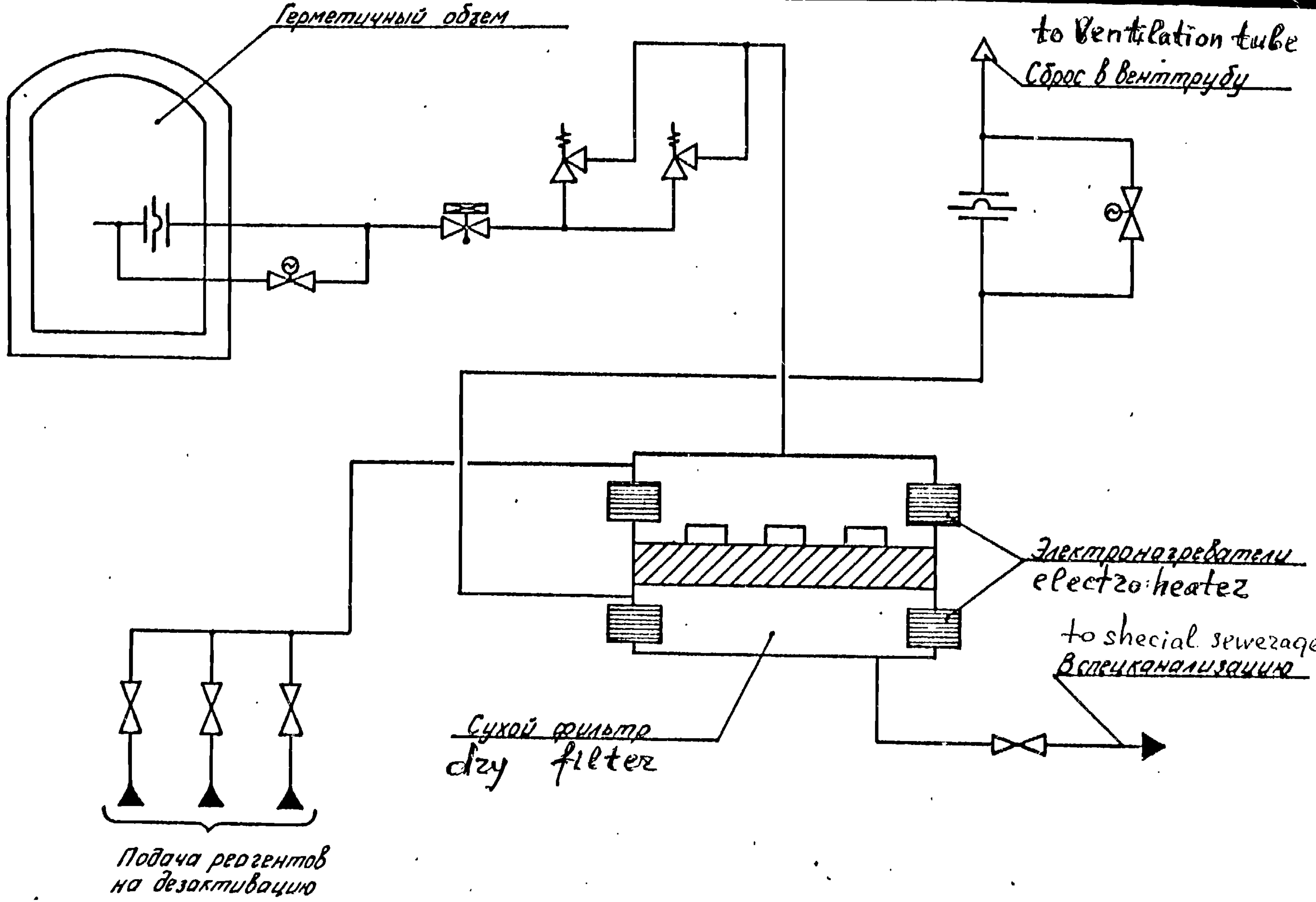
At present the scientific elaborations on the damaged reactor core cooling and catching system beyond the reactor vessel are conducted yet the specific technical solutions hasn't been defined. Therefore implementation of similar system is provided in future after technical substantiation.

In conclusion it's required to note that from the preliminary data the additional measures implementation permits to decrease probability of core failure and exceeding of limit releases in 50-60 times.



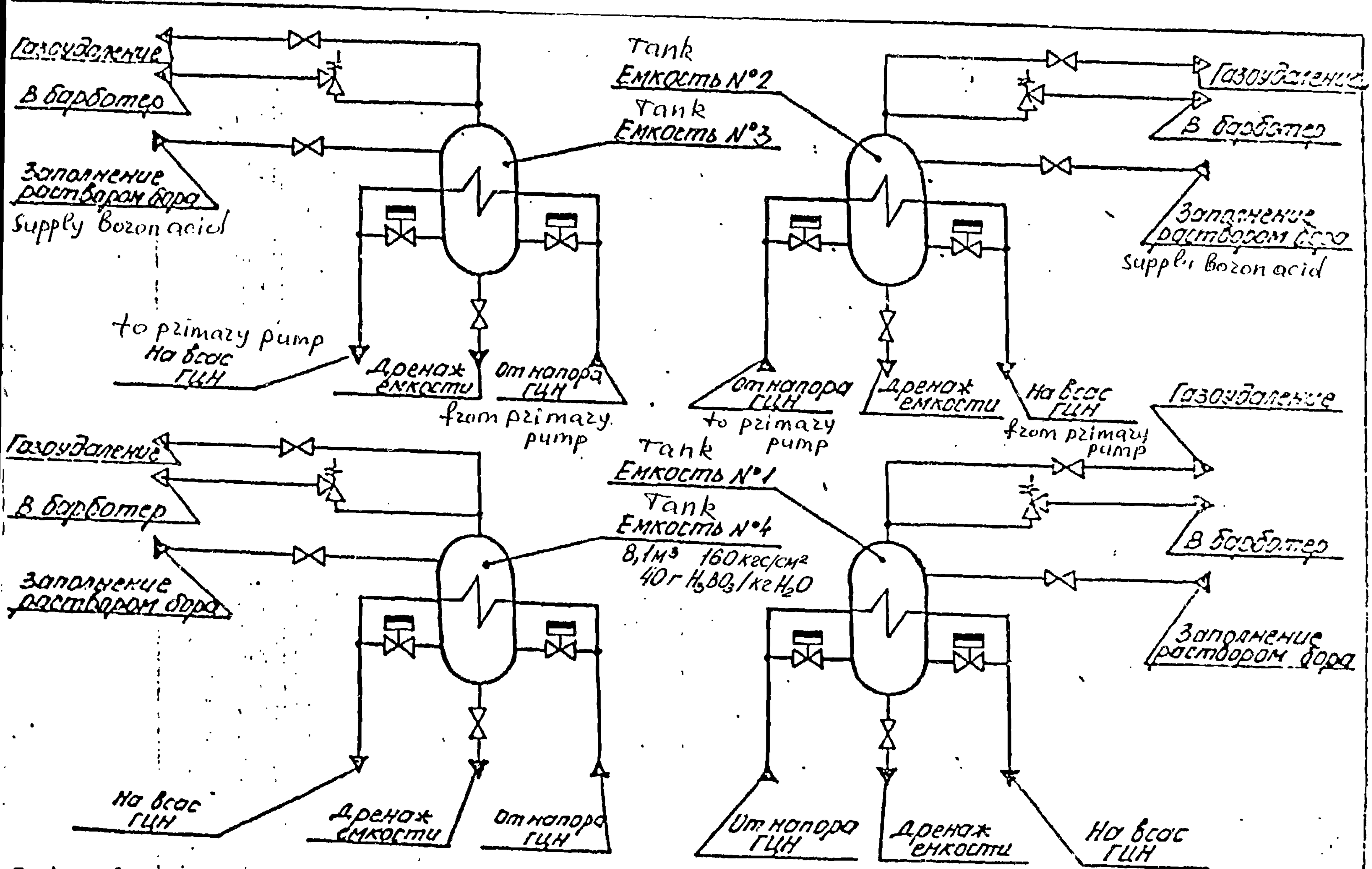
Эскиз №1

Система пассивного отвода тепла. Реакторное ...



Эскиз N 7

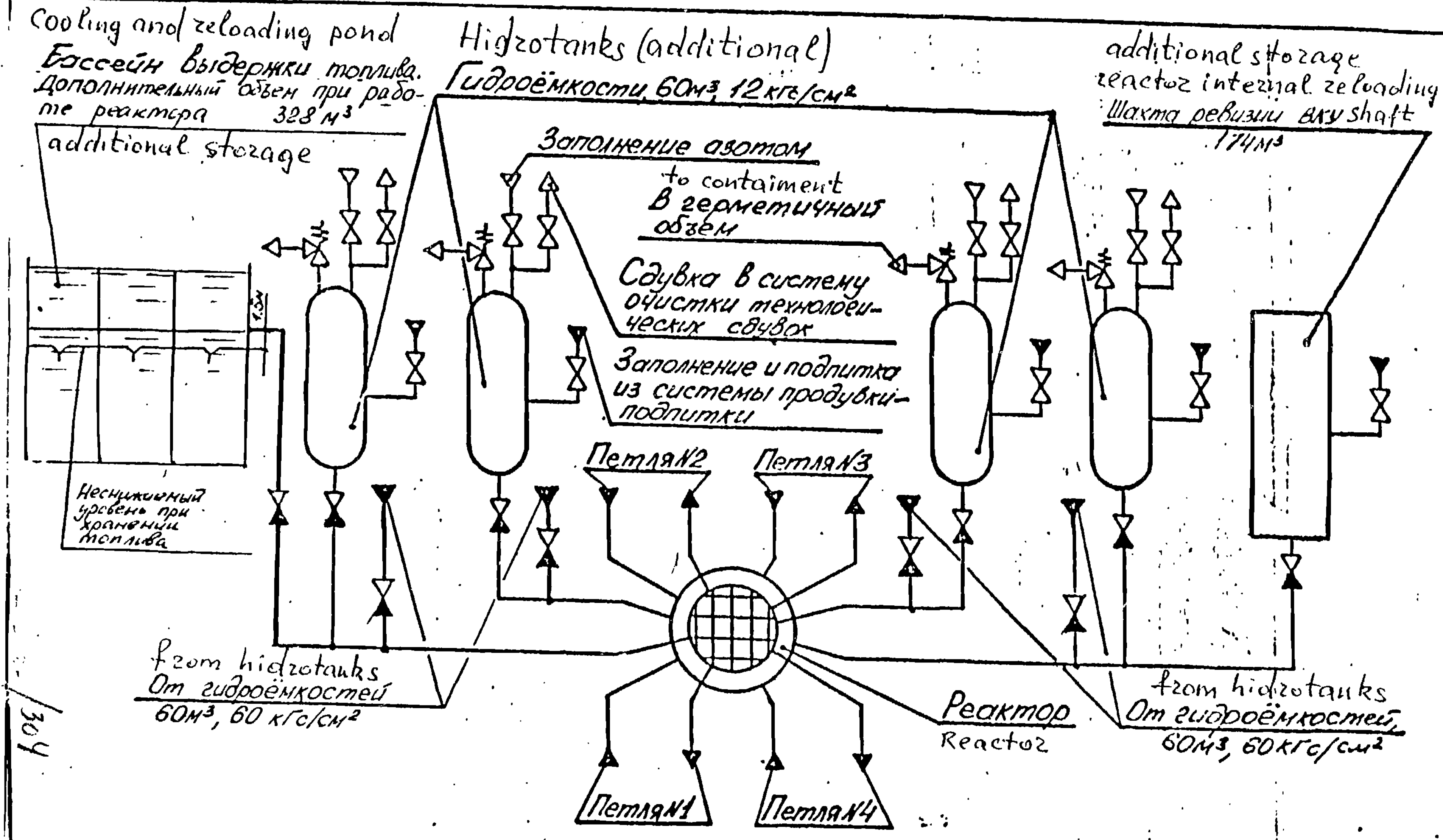
Система сброса парогазовой среды Sistem of pressure shedding and cleaning of releases



System of quick boron injection (SQBI)

Система быстрого ввода бора

Эскиз №3



Эскиз №3

Система пассивного охлаждения (залива) активной зоны реакторной установки.

Реакторное отделение Passive core flooding system (PCFS)

ACCIDENT MITIGATION ON NPP WITH VVER-1000

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for Accident Mitigation

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Abstract

Specific features of accident development at NPP with VVER-1000 and influence of operation of safety-important systems on radiation consequences of accidents are considered. It is shown that in some design-basis-accidents and beyond design-basis-accidents the design operation of the systems providing protection of the NPP barriers against destruction may result in considerable increase in radioactive product release to the environment. Possible actions for releable trapping of radioactive product in such situations are disscussed.

Introduction

The current approach to the nuclear power plants (NPP) safety problem suggest realisation of the "defence-in-depth" concept. This concept determines the general strategy of safety measures and means for the NPP and consist of some physical barriers intended to prevent activity propagation as well as of some protection levels ensuring the plant and the barriers integrity and radiological protection of the population and the environment in case the barriers would be destroyed.

It may be considered that NPP meets the safety requirements when under the normal operation conditions and in Design Basis Accidents (DBA) its radiation effect on the personnel, population and environment does not exeed the permissible doses of personnel and population exposure and the standarts on radioactivity releases and the concentration of radioactive products in the environment, as well as when it is capable to reduce this effect during the beyond design basis accidents.

The dose criteria for taking decisions on mesures for population protection at any accident stage and for various ways of dose formation are determined by the appropriate standarts documents. The permissible amounts of radioactive products released to the environment under normal and emergency operation conditions at NPP are also determined.

In July 1990 "General rules on NPP safety" (OPB-88) was implemented in the USSR. These are the main top normative document regulating the safety questions inherent to NPP as possible source of radiation effects on the personnel, population and environment. In accordance with this document the estimated probability of the altimate accident radioactivity release shall not exeed 10^{-7} 1/reactor.year (the permissible release for beyond DBA is equil to 30,000 Ci for iodine-131 and 3,000 Ci for ceasium-137). If the above requirement is not satisfied, it is necessary to provide additional technical measures in the plant design for management of the beyond DBA in order to mitigate its consequences.

In the USSR the probability safety analyses and assessment of the beyond DBA consequences for NPPs of different types including those with VVER-1000 are being carried out for determination of additional protection measures enabling the normative requirements to be satisfied. Below we shall discuss some important points releated in the process of provisional evaluation of the radiological effects of the accidents at NPP with VVER-1000 reactor.

1. Current status.

As of July 1990, the Soveit Union hed fifteen VVER-1000 power units, mainly of the V-320 model, in operation. The probability of a severe accident resulting in core melting does not exeed 10^{-4} 1/reactor.year and, hence [1], the safety level is acceptable for operating NPPs.

The reactor and the main primary and secondary systems are closed inside a containement intended for localisation of radioactivity releasing during the DBA with rupture of the primary pipes of any diameter up to 85 cm. The containment localisation properties are maintaind up to a pressure of 0.45 MPa (design pressure of containment failure is 0.8 MPa) and temperature of 150°C. The contaiment consist of several interconnected compartments where the reactor systems are accomodated. Its free volume is about 70,000 m³. The containment is provided with several plenum-exhaust ventilation systems for heat removal from some containment compartments (e.g., from the steam generator boxes) and for maintaining the concentration of radioactive products whithin the limits permissible for NPP personnel during the scheduled refueling operations. There are recirculation venting systems and the systems for air discharge through the vent stack; in some recirculation systems and on the line of discharge to the vent stack the aerosol and iodine filters with the decontamination efficiency of about 100 are installed.

The designe release of the medium from the containment to the environment is 0.3% of free volume per day at a pressure of 0.45 MPa in the containment. The contaiment localisation function is considered to be lost when the isolation valve on the 40 cm diam pipe (penetrating the containment shell) is unseated.

The containment also accomodates the main systems for protection of the NPP barriers during the accidents and for mitigation of the accident consequences (some components of these systems such as the spray pamps may be located outside the containment, see Fig.1 and Table 1). The systems include ECCS, Emergency Core Cooling System (hydroaccumulators, HP and LP systems), and the spray system. There are three independent channels for HP and LP ECCS, and a spray system supplied with boron from the tanks for emergency storage of boron solution: after the tanks are emptied the pumps are changed over to supply from the sump.

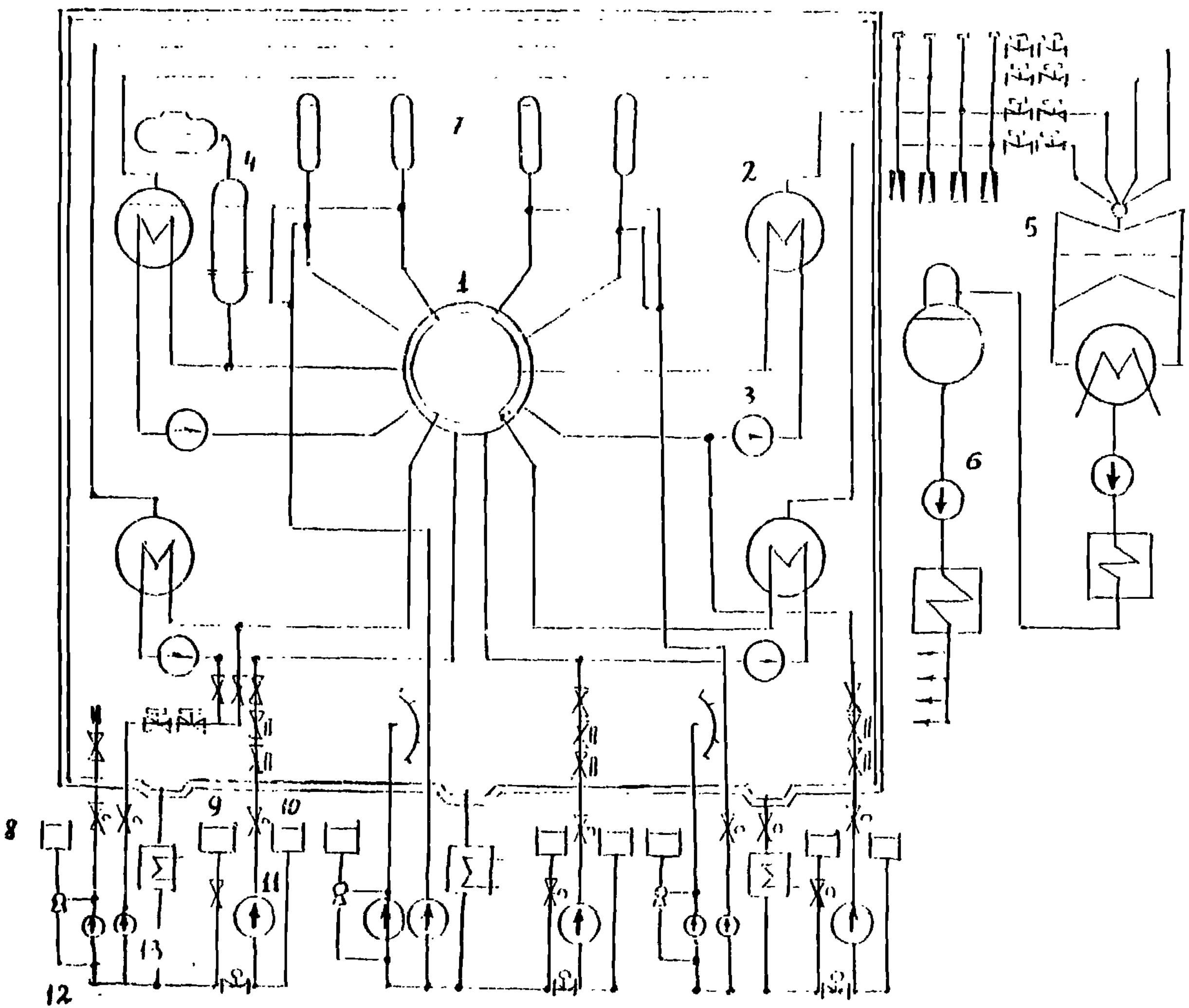


Рис.1. Схема основных систем реактора ВВЭР 1000

Fig.1. VVER-1000 Process Flow Diagram

1. Корпус реактора
   Reactor pressure vessel
2. Парогенератор
   Steamgenerator
3. Главный циркуляционный насос
   Primary coolant pump
4. Компенсатор давления
   Pressurizer
5. Турбина
   Turbine
6. Подпиточный насос
   Feedwater pump
7. Гидроаккумуляторы
   Hydroaccumulators
8. Бак запаса спринклерного раствора
   Hidrazine-hidrate storage tank
9. Бак запаса раствора борной кислоты
   Tank for emergency storage of boron acid
10. Бак запаса концентрата борной кислоты
    Tank for storage of boric acid concentrate
11. Насос САОЗ ВД
    ECCS high pressure pump
12. Спринклерный насос
    Sprinkler pump
13. Насос САОЗ НД
    ECCS low pressure pump

Table 1.

The main parameters of the Emergemcy Core Cooling Sysytem.

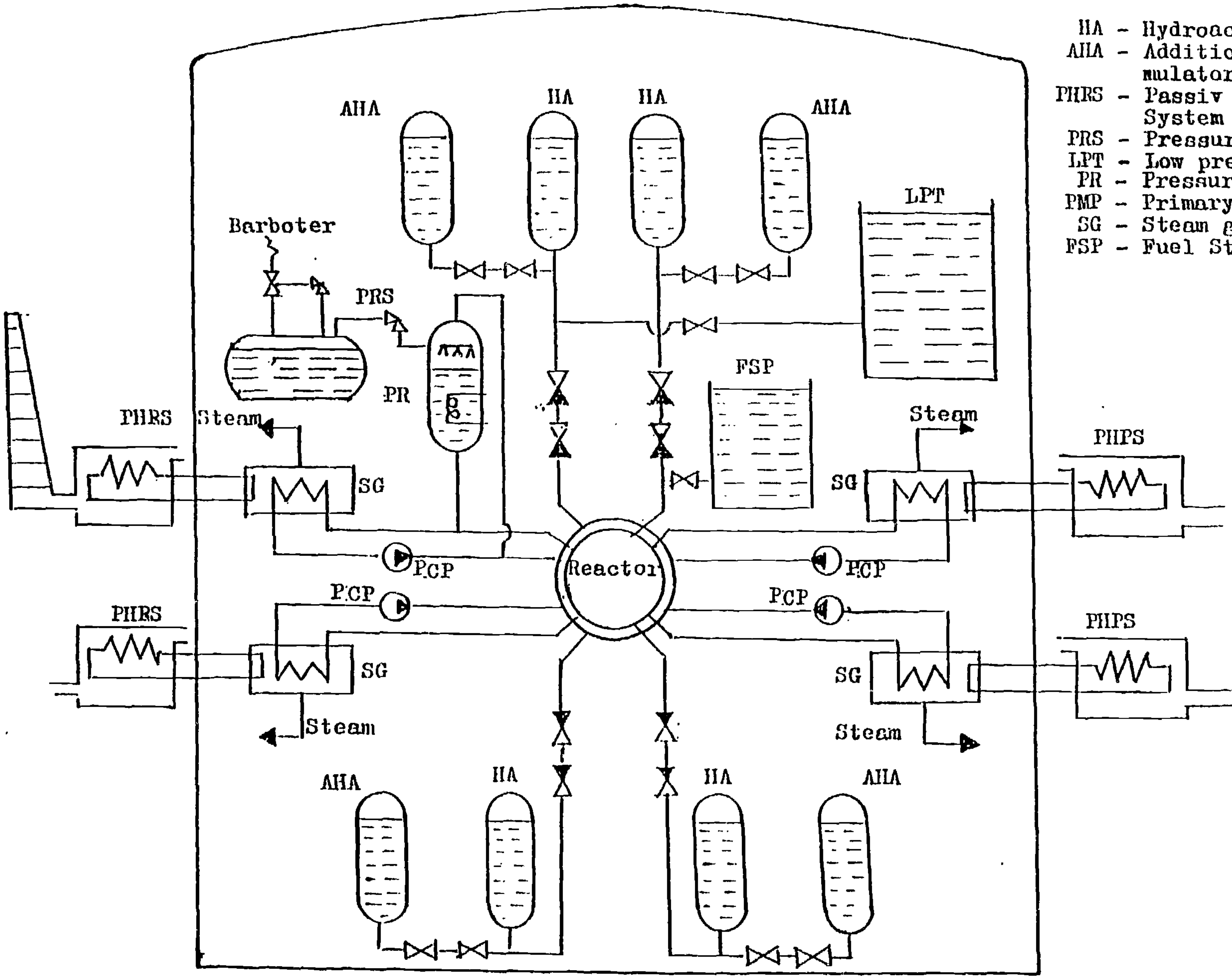
System, parameter	Value
1. ECCS hydroaccumulators	
Number	4
Total volume of Water	200 m^3
Boric asid concentration	12 g/kg
Working pressure	0,6 MPa
2. LP ECCS	
Number of independent channels	3
Channel parameters	
-volume of tank for emergency storage of boron acid (common with spray system)	700 m^3
-flow rate at 0.1 MPa	~700 m^3/hr
-flow rate at 2.2 MPa	250 - 300 m^3/hr
-boron concentration	12 g/kg
3. HP ECCS	
Number of independent channels	3
Channel parameters	
-volume of tank for storage of boron acid concentrate	15 m^3
-nominal flow rate	150 m^3/hr
-boron concentration	40 g/kg
-working pressure	10-11 MPa

The safety analyses of some NPP with VVER-1000 of V-320 model has shown that the OPB requirements can be satisfied if additional ECCS and a system of filtered venting of the containment would be introduced to the designe (so-called the VVER-92 design [2]). The additional cooling systems are passive anbd consist of (see Fig.2):

- passive secondary heat removal system;
- system of additional hydroaccumulators ensuring a long-term core cooling.

In addition to the above mentioned systems (available in the design and additional ones) NPP with VVER-1000 are provided with the systems which can be used in accidents for organisation of core cooldown, in particular:

- system of normal makeup of the primary circuit; the system can be used for makeup at high pressure and may operate in the case of HP ECCS failure;
- system of safety valves on the pressurizer; it can be used for reduction of the primary circuit pressure in small LOCA or in the plant black-out, ensuring operation of LP ECCS;
- system of emergency gas removal; it can be used for reduction of the primary circuit pressure;
- system of normal supply of feed water to the secondary circuit; it can be used for duplication of the emergency feed water supply system;
- fire system; it can be used for water supply to the steam generators in the case of falure of the normal and emergency feed water systems.



HA - Hydroaccumulator
AHA - Additional Hydroaccumulator
PHRS - Passiv Heat Removal System
PRS - Pressure Relief System
LPT - Low pressure tank
PR - Pressurizer
PMP - Primary Coolant Pump
SG - Steam generator
FSP - Fuel Storage Pool

Fig 2. VVER-92 (1000 MW) Passive Emergency Core Cooling Systems

2. Spray system.

The spray system is a safety-related system and serves the protection and accident localisation. It is made in accordance with the first seismic stability category.

This system is intended for accident localisation by condensation of the evaporated part of the coolant discharged into the containment as a result of pipe ruptures. The spray system functions are: reduction of the pressure in the containment to the outside pressure and fixing of various kinds of iodine released to the containment atmosphere during the accident.

The spray system operation is swiched on automatically by the signals from HP and LP ECCS and pressure rise in the containment up to 0.12 MPa. By the first signal the system is swiched on to recirculation with prohibition of opening the valves on the pump head. By the second signal the velves on the pump head on the line of water supply to the spray system sprinklers are opened (with prohibition of closing). When the pressure reduced below 0.12 MPa prohibition of remote valve closing is removed and at a pressure below 0.08 MPa the valves close.

The sprinkler solution is delivered to the sprinklers in 30-60 seconds after start of a big LOCA.

The system consist of three independent channels. The channel diagram is shown in Fig.3.

Each channel consist of spray pump (4), 6 m³ tank with sprinkler solution (1), water-jet pump (2) for delivery of the sprinkler solution to the borated water in ~700 m³ tank for emergency storage of boron solution (7), and sprinklers (4) installed in the form of three concentric open rings under the containment dome, cooler of the water charged from the sump and control and isolation valves.

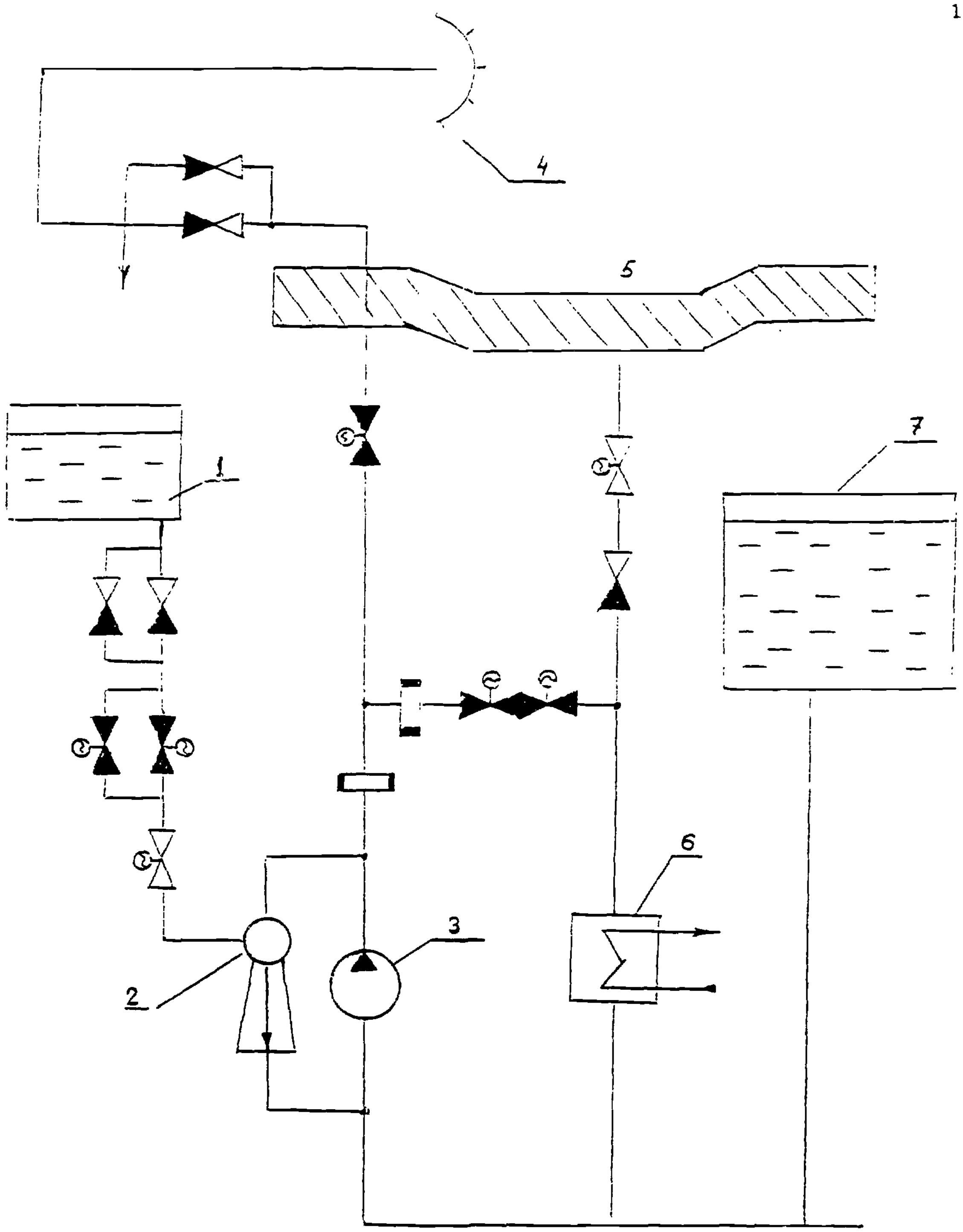


Fig.3. Spray system flow diagram: 1 - tank for storage of sprinkler solution; 2 - jet pump; 3 - sprinkler pump; 4 - sprinklers; 5 - sump; 6 - cooler; 7 - tank for emergency storage of boron solution.

Each spray system channel has 20 sprinklers. The main characteristics of this channel are presented in Table 2.

In the analyses of the accident consequences it is assumed that two of three spray system channels remain operable.

Then tank (7) is emptied the system is switched on to recirculation with makeup from sump (5). The HP and LP ECCS pumps are also fed from this sump.

3. Effects of the safety-related system operation on radiation consequences of the accident.

3.1. Design basis accident.

Maximum DBA is postulated as instantanious rupture of the 85 cm diam primary pipe and double side leakage of the coolant. The cladding temperature increases up to 1000°C whithin a short time depending on the power density. Normally it is assumed in the calculations that this accident is accompanied by failure of the fuel element claddings and release of all iodine accumulated in the gap ($\sim$0.3% of all iodine inventory in the core or $\sim 2.5.10^{-5}$ Ci of iodine-131). In response to ECCS operation the core is flooded. At about this time the spray system is switched on. In parallel operation of the spray system and LP ECCS the borated water can be delivered for 20-30 min and after that the pumps of both systems are changed over to feeding from the sump. With allowance for the primary water flowing through the rupture and the water from the hydroaccumulators, the total amount of water circulating over one channel of the spray system and ECCS will be 1000-1200 m^3.

The first 20-30 min the spray pump delivers the pure (non-radioactive) sprinkler solution, with the molecular iodine being

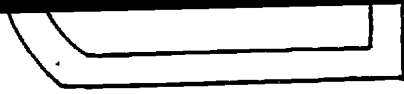


Table 2.

The main parameters of the spray system channel.

Component, parameter	Value
1. Spray pump	
Type: spiral, horizontal	
Pumped medium: non-deaerated radioactive water containing:	
boric acid	up to 16 g/kg
KOH	up to 2 g/kg
Hydrazine-hydrate	up to 200 g/kg
Nominal flowrate	700 m^3/hr
Nominal head	1,37 MPa
Maximum permissible head on the suction side:	
in operation	0,69 MPa
Liquid temperature	10-100°C
Power	500 kW
2. Tank for emergency storage of boron solution.	
Volume	707 m^3
Bottom are	~180 m^2
Minimum silution inventory	500 m^3
Boron concentration	12 g/kg
3. Water jet pump.	
Inlet pressure	0,68-1,17 MPa
Pressure of pumped water	0,098 MPa
Working liquid supply	50 m^3/hr
Sprinkler solution supply	10 m^3/hr
4. Tank for storage of sprinkler solution.	
Total volume	6 m^3
Pressure	atmospheric
Working solution temperature	40°C
Nominal level	3,1 m
5. Sprinklers	
Design pressure difference	0,098 MPa
Solution flow rate	30 m^3/hr
Spray angle	>75°
Equivalent drop diameter	800 mkm

effectively removed from the containment atmosphere (with two channels in operation the period of semiremoval of the molecular iodine is 4-5 min) and bound by chemical reagents contained in the sprinkler solution so that within 20-30 min 1-5% of molecular iodine remain in the containment atmosphere. After the spray system is changed over to recirculation the water earlier containing the iodine (~0.25 Ci/kg of iodine-131) will be supplied to the containment through the sprinklers. In accordance with the distribution law iodine will be trapped by the solution drops and simultaniously released back into the containment atmosphere. Therefore the spray system may reduce the iodine concentration in the containment atmosphere only to a value determined by the distribution coefficient depending on the rate of hydrolysis of iodine compounds in the solution and on other physical and chemical processes. For the VVER-1000 conditions the coefficient of iodine distribution between the steam and aqueous phases is about 10^{-4}, the equilibrium activity of iodine in the containment atmosphere being ~0.3% of the total amount of iodine released during the accident or ~750 Ci. Because of a large specific surface of water drops rapid kinetics of equilibrium state may be expected, with the time of reaching the equilibrium under the conditions described estimated as 1-2 hrs.

However the water from the sump is delivered not only to the spray system but also to the cooled core where it is reevaporated. As the cooling water contains iodine the latter will be carried away from the boiling water with steam and at the time of the steam-water conversion some iodine concentration is established in the steam, which is determined by the specific surface of steam bubbles, average time of bubble emersion, and temperature. It may be expected that under these conditions the distribution coefficient would be several time higher than for the containment conditions because of higher rate of the hydrolysis reaction at a higher temperature and, particulary, because of the radiolysis of iodine compounds in high radiation fields in the core. If the iodine is carried away by steam into the containment the equilibrium there will be shifted towards the steam phase of the containment atmosphere and the amount of iodine-131 there may reach several thousands of Ci.

At the design tightness of the containment the iodine release to the environment may be as high as 10 Ci/day.

3.2. DBA with unseated valves on the 40 cm diam vent pipe.

In the case of unseated valves radioactive steam-gas mixture will be carried away from the containment through the vent stack directly to the environment (direct containment by-pass). At the initial phase of the accident when the spray system removes effectively the iodine from the containment atmosphere (for 1-2 hrs) iodine release to the environment will be determined by the relation between the relative iodine release through the open pipe and the relative rate of iodine removal by the spray system. The iodine-131 release to the environment during this time will be about 25,000 Ci. Further, during spray system operation, only due to distribution law the iodine release will increase and approach 100,000Ci/day or ~40% of initial amount of iodine released to the cintainment. The possible equilibrium shift toward the steam phase due to additional iodine release in core flooding from the sump is taken into account the total iodine release to the envirinment may be close to 100%. If the time of iodine release via by-pass of the containment is limited to 1-2 hours, its release to the environment will not exeed the permissible release for beyond DBA and will be even much lower since it was assumed in the assessment that in DBA the claddings of all fuel element will be fail. Taking into account more realistic estimates of failure percentage ~2,500 Ci are obtained for iodine-131 release through by-pass whithin the first two hours. Thus if some additional design (additional isolation valves) or design and arganization (remote or manual closing of the valves during accident) measures are taken the radiation consequences of the given accident may be apprecibly reduced.

3.3. Accidents with core melting.

As noted, the beyond DBA management measures suggest organization of the protection levels ensuring the barriers integrity. The core-relating measures include:

- organization of long-term core cooling using the active and passive means to prevent the core from serious damages;
- organization of cooling of a considerably damaged core;
- retention of core melt in the reactor vessel;
- prevention of the containment basement melt-through if the core melt went from reactor vessel.

Depending on the scenario of accident propagation the time of the beginning of core melting and its duration till full melting are determined by the initial accident event, composition and parameters of the core flooding systems and residual heat removal system.

3.3.1. Fuel melting stage.

Let as consider the radiation consequences of a small LOCA (8 cm diam pipe rupture) with failure of HP and LP ECCS and of the spray system in the presence (a) and in the absence (b) of the passive heat removal in the secondary circuit. Assume that in the case (b) the primary pressure is reduced by the system of the safety valves on the pressurizer so that on the initial accident stage before beginning of core heating the processes develop practically at the same velocity.

Fig.4 present the time dependences of the airborne aerosol mass in the containment atmosphere for the cases (a) and (b); Fig.5 shows release of the iodine in the aerosol form through the design leaks in the containment shell to the environment. The total release of all iodine and caesium forms is presented in Table 3. The difference in the total release being mainly due to the difference in the release of the fission product aerosol forms.

The difference observed in Table 3 for the cases (a) and (b) is mainly due to the features of the behavior of aerosols in the containment atmosphere depending on the aerosol concentration. It is known that the rate of aerosol s edimentation is determined by the aerosol particles size depending on the rate of coagulation of particles of smaller sizes. In turn the coagulation rate, and eventually the sedimentation rate and the airborne mass of aerosol particles are determined by the square of initial aerosol particles concentration. If the aerosol release rate during core melting is relatively low then aerosol concentration and natural sedimentation rate are relatively low too. Besides in case (a) the total mass of aerosols released from core is smaller than in case (b). All these are the reason for the above mentioned difference in fission product release to the environment and, hence, in the scale of radiation consequences. Use of a safety-related systems reliably protecting the core against ladge damages in certain accidents can make consequences of accidents heavier than in the absence of such a systems.

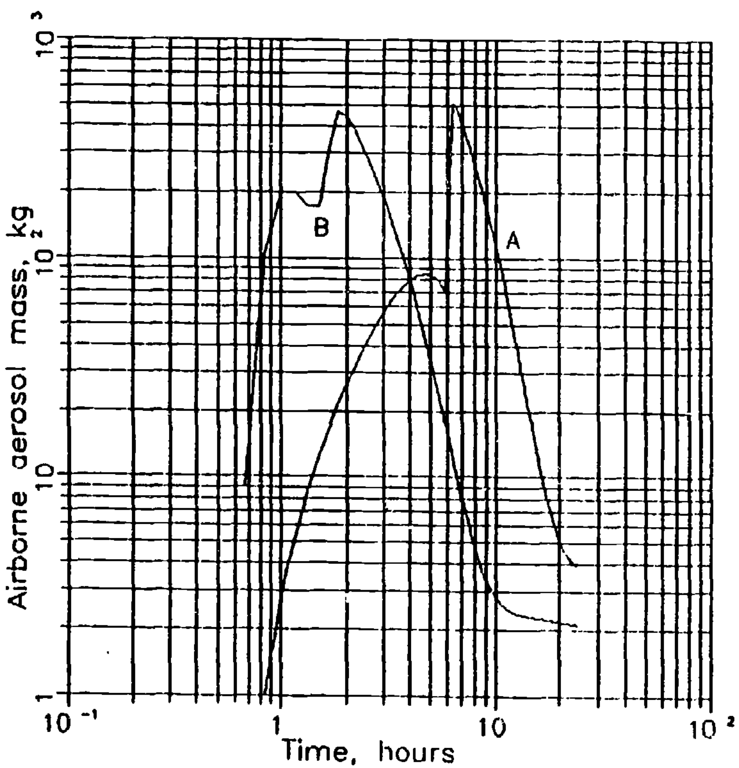


Fig.4. Airborne aerosols mass in containment vs time: (a) – in the presence and (b) – in the absence of the passive heat removal in the secondary circuit.

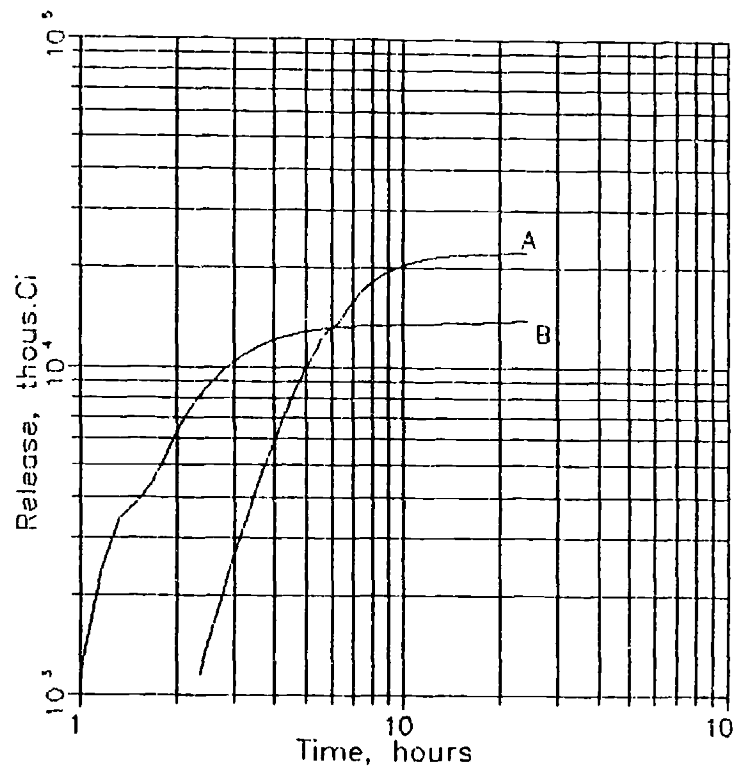


Fig.5. Cumulative release of iodine in aerosol form to the environment vs time: (a) – in the presence and (b)– in the absence of the passive heat removal in the secondary circuit.

Table 3.

The radiation consequences of a small LOCA with failure of HP and LP ECCS and of the spray system in the presence (a) and in the absence (b) of the passive heat removal in the secondary circuit.

Parameters, processes	a	b
Iodine and caesium release from the core	80-100%%	
Fission products forms:		
aerosol iodine	95%	
molecular iodine	4%	
organic iodine	1%	
aerosol caesium	100%	
Aerosols mass released from the core	180 кг	252 кг
Aerosols mass released during core-concrete interaction	500 кг	416 кг
Start of melting	~2000c	~2000c
End of melting	~16000c	~4000c
Release to the environment through the untightness thous.Ci/day		
iodine cumulative release	25,7	17,0
aerosols iodine release	22,3	13,4
molecular iodine release	1,5	1,5
organic iodine release	1,9	2,1
caesium release	2,2	1,3
iodine release with spray system		5,1

3.3.2. Stage of flooding of a severely damaged core or core melt.

One of the requirement normally imposed on the systems which can be used for flooding of a severely damaged core or core melt is sufficient removal of decay heat in order to prevent damage of the next barrier i.e. core degradation, melting of the reactor vessel, containment basement melt-through. The stage considered occurs, for example, in the accident with ECCS failure or with its malfunction (high pressure accidents). Restoration of normal ECCS performanc·· or use of other sources of water are also possible at the stage considered.

In the case when the spray system remains operable the radioactive aerosols and fission products released from the melting core will be effectively discharged from the containment atmosphere and accumulate in the sump. With HP and LP ECCS failed the spray system will supply non-radioactive sprinkler solution from the tank to the containment; in a big loca when 30-40 min are sufficient for the core to be melt, the sprinkler solution will trap more than 0.99 of radioactive iodine and caesium. After that the iodine-131 content in the containment atmosphere stabilized at a level of 1.3-1.5% (~1.2 million Ci) Of the its initial inventory in the core.

If at any time the ECCS functions restore the only source of water for core or corium flooding will be the sump and the water with ultra-high content of radioactive products (up to 100 Ci/kg of iodine-131) supplied to the superheated core. In interaction of the superheated core the water will evaporate. The fraction of radioactive products repeatedly released from the water with steam will depend on wether a sufficiently thick layer of water would form over the fuel surface with relatively slowly rising bubbles or an intence surface boiling would take place. In the first case the repeated release with steam would be relatively small and it has been already estimated (increase in the molecular iodine concentration in the containment atmosphere by several times). In the second case with insufficient water flow rate to the melt or with water supply to the overheated core with partly saved structure, a considerable release of radioactive products solved in the water should be expected. Due to high temperatures direct sublimation of chemical compounds of fission products and their considerable formchanging should be expected, which will result in intence formation of aerosols and volatile forms of fission products. If in the containment compartments exist the source of carbon compounds, organic highly volatile iodine compounds will also be produced. As a result the equilibrium concentration of fission products in the containment atmosphere will rise (with the spray system in operation the rise will amount to asa much as an order of magnitude and more) and, therefore, the fission product release rate to the environment through the leaks in the containment shell will increase. In the absence of flooding the daily release will amount to about 3,500 Ci of iodine-131. If the flooding begins in 5-6 hours after the accident the daily release may rise to 35,000-50,000 Ci and exeed the permissible value for beyond DBA. If no special measures for retaining fission products are taken it may be assumed that all fission products released from the fuel during the accident could release to the environment.

If the spray sistem fails during the accident considered and flooding of the core or the corium becomes possible the fission product release rate to the environment will rise due to increase in the equilibrium fission product concentration in the containment atmosphere. According to the estimates, at a water flow rate to ECCS of about 1,000 m^3/hr and at a rate of natural iodine deposition of $10^{-4} c^{-1}$ the amount of iodine-131 in the containment atmosphere will be more than 75% of all iodine-131 core inventory, and the daily release may be about 200,000 Ci.

The phenomenon of repeated release of fission products to the containment atmosphere, with the safety-related systems being in normal operation, may be of importance for assessment of the radiation consequences of the accident in the case of loosing of containment localization functions (containment failure or basement melt-through) and must be taken into account when choosing the working parameters of other localization systems, for example, the filtered venting system.

6. Discussion and conclusions.

The assessment analysis has revealed that during the NPP accidents resulting in pipe ruptures, containment bypass and possible core degradation situations may occur when the goals of some emergency safety-relating systems for a given method of realization of these functions may contradict to that of accident management - mitigation of the radiation connsequences of the accidents to the lowest level. In other words, a situation is possible when the local goals of accident management may not coincide with the strategic aim. Attempts to prevent core melting, to retain the corium in the reactor vessel or in the core-catcher, use of additional systems giving a real gain in some cases may restore the NPP control but in doing so the processes will be initiated that increase fission product release to the environment as compared with that without such systems and such attempts. The design features of particular NPP may be the reason why the systems which are, in principle,highly effective (spray system) at some accident stage not only stop to perform their design functions but becomes sources of radioactive products. Such a highly efficient system for the reactor and station barriers protection as ECCS is an intence source of radioactive products in flooding a degradated core or core melt. Therefore restoration of NPP control and bringing it to the stable state from the viewpoint of taking measures on protecting the retention functions of the barriers, requires these special actions for reliable trapping of radioactive products and excluding the possibility of their cyclic release to the containment atmosphere and increased release to the environment to be made. For this purpose the following technical and organization actions can be proposed:

pro	contra
1.Use of external sources of water for ECCS and spray system	Problem of handling large volumes of water inside the containment
2.Use of the sprinklers solutions dealing fission products in insoluble compounds or essentially decreasing the iodine distribution coefficient between steam and water	Lack of protection against repeated release of fission products to the containment atmosphere
3.Effective decontamination of the sump water	Considerable expenses for ion-exchange filters
4.Initiation of additional aerosols release to the containment atmosphere for effective deposition of radioactive aerosols	Repeated fission products release during ECCS operation
5.Installation of additional isolation valves on the venting lines for avoiding direct containment by-pass	
6.Use of the normal containment vent system for removal of fission products or depressurezation of the containment with decontamination of the medium on the normal vent filters	Possible only at the accident stage at low temperature and pressure; otherwise the filters may fail with direct bypassing of the containment
7.Organization of forced recirculation venting of the containment with decontamination of the steam-gas mixture on the filters designed for the accident conditions (e.g., on the existed accident filters)	

The latter measure enables, in principle, all problems associated with use of safety-relating systems and with need of localisation of radioactive products to the containment in operation of these systems to be solved, thus preventing considerable fission product release to the environment.

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UTILIZATION OF THE SYSTEMS OF POWER PLANTS WITH RBMK REACTORS TO MITIGATE ACCIDENT CONSEQUENCES

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At present there are fourteen 1000 MW(e) power plant units with the RBMK-1000 reactor and two 1500 MW(e) units with the RBMK-1500 reactor in operation in the USSR [1]. Unit 5 of the Kursk nuclear power plant - which also incorporates an RBMK-1000 - is being prepared for commissioning.

The RBMK nuclear power reactor is a heterogeneous channel-type thermal reactor in which graphite is used as the moderator and boiling light water as the coolant. It has 1660 fuel channels located in vertical perforations in the graphite columns. These channels are zirconium alloy tubes 80 mm in diameter. The channel contains a fuel assembly with 18 fuel rods each having a diameter of 13.6 mm in a zirconium alloy cladding. The heat flow diagram is typical of single-circuit power plants with boiling water reactors (Fig. 1). The multiple forced circulation circuit (MFCC) consists of two parallel loops in each of which half the fuel channels of the reactor are cooled. The coolant is circulated with the help of the electrical main circulation pumps (MCP). Sub-cooled water is supplied to and steam-water mixture removed from each channel through individual pipes. The water and steam are separated in horizontal separators at a pressure of about 7 MPa. The saturated steam is sent to two turbines and its condensate, after heating and de-aeration, returns to the separators from where, mixed with the separated saturated water, it is pumped to the reactor inlet by MCP.

At the time of designing the first power plants with RBMK-1000s, a list of initiating emergency events was prepared and the worst scenarios were analysed [2]. On the basis of operating experience and in the light of the increasingly strict safety requirements for nuclear power plants throughout the world in general, the original list of initiating events has been expanded considerably. The list of design-basis initiating events for RBMK power plants today includes about 50 situations which can be divided into five main types:

(1) Situations involving reactivity changes - fall or spontaneous removal of scram rods, failures in the instrumental part of the reactor control and protection system (RCPS), water loss in RCPS channels and spurious actuation of the emergency core cooling system (ECCS);

(2) Accidents in the core cooling system - MCP trip, rupture of the non-return valve plate or isolation valve disk of MCP, failures in the feedwater system, etc.;

(3) Accidents caused by breaks in MFCC piping, in steam lines and in feedwater lines;

(4) Accidents with disconnection or failure of equipment - loss of auxiliary power supply, turbogenerator trip, opening of the main safety valves followed by subsequent disturbance of their setting, etc.;

(5) Other accidents - during refuelling by the fuelling machine, fire, water flooding, etc.

The following systems of the reactor and the power plant as a whole ensure that the designed limits of fuel element damage and the permissible level of release of radionuclides into the environment in normal operation and in accident conditions are not exceeded:

- Reactor control and protection system (RCPS), which, in addition to the conventional devices, includes local automatic regulators, local emergency protection groups and a fast-acting emergency protection system;

- Power density monitoring and control system, which carries out continuous digital control of radial and vertical power density in the core;

- Emergency core cooling system (ECCS), which removes heat from the core during accidents caused by pipe rupture or equipment failure. It goes into operation automatically in accordance with five independent algorithms, depending on the type of accident;

- System of monitoring cladding integrity for each fuel channel separately and in groups;

- System of individual monitoring of the integrity of fuel channels and RCPS channels;

- System for protection against overpressure in the circulation circuit - group of safety valves;

- System of monitoring and control of coolant flow in the fuel channels and RCPS channels;

- System for monitoring temperature in reactor metal structures and in the volume of the graphite structure;

- Centralized monitoring system, which acquires and monitors analogue and digital inputs, calculates parameters for the safe conduct of the operating processes, and signals deviations of the monitored and calculated parameters from their set values;

- System for dumping the steam and gas mixture from the reactor enclosure in the event of reactor channel rupture in an accident situation;

- Fuelling machine, which is used for on-load changing of spent or ruptured fuel assemblies;

- Accident containment system designed to receive coolant ejected in an accident situation, to condense steam and to hold up non-condensed radioactive gases;

- Water supply system for the main reactor circuits and for the ECCS;

- Station auxiliary power supply system;

- Stand-by diesel generator set.

The above-mentioned systems of the reactor can ensure safe development of accident processes in accordance with the adopted list of design-basis accidents [3]. Upon deviation of the parameters, the reactor control and protection system automatically controls reactor power, actuates the core cooling and injection systems, the separator pressure and level controllers, and automatically maintains the thermohydraulic characteristics within safe limits, the ECCS ensures emergency cooling of the reactor and so on. Failure-proof operation of these systems in accident conditions is ensured, first of all, by emergency power supply and water supply and by redundancy. Therefore, most important from the standpoint of power plant safety are the auxiliary power supply system, the water supply system for the reactor safety injection system and the coolant ejection containment system.

In the present paper we describe these systems for power plants using the second-generation RBMK-1000 reactor and the principles of their operation under normal and emergency conditions. We consider the emergency condition created by total loss of the auxiliary power supply and the beyond-design-basis accident coupled with loss of the auxiliary power supply and failure of the stand-by diesel generator plant.

1. EMERGENCY POWER SUPPLY SYSTEM

The station emergency power supply system is designed to run the equipment needed for after-heat removal both under normal and emergency conditions.

All users of the station auxiliary power supply are divided into three groups in accordance with their reliability requirements:

Group 1: These are devices whose power supply may not be interrupted (for more than the fractions of a second needed for automatic switch-over) in accordance with the safety requirements for the entire duration of the accident process, including the situation with total loss of station auxiliary power supply.

This group comprises the instruments and servodrives of the reactor control and protection system, the computer system, reactor instrumentation, electrical drives of the valves and fittings of the emergency core-cooling system, accident containment system, circuit safety injection systems, etc.

Group 2: This includes devices whose power supply may be interrupted for a period determined by the safety requirements (from tens of seconds to several minutes) and which must have power after actuation of the reactor protection system. The group includes the mechanisms that ensure after-heat removal under emergency conditions accompanied by total loss of voltage in the station auxiliary power supply buses (cooling pond emergency pump, pumps for the cooling circuit of the reactor safety and protection system, emergency core cooling system pumps, emergency feed pumps, clean condensate pumps, fire-fighting pumps, service water pumps and sprinkler cooling system pumps).

Group 3: This consists of devices whose power supply may be interrupted during the time of emergency connection of the stand-by and which do not require compulsory power supply after actuation of the reactor protection system. They include the main circulation pumps, electrical feedwater pumps, condensate pumps, mechanisms of the auxiliary systems of the reactor and turbine hall, and other equipment which ensures the operation of the unit under normal conditions.

Group 1 is supplied from the emergency power system, whose source is static inverters. These are continuous power supply units, which are operated from batteries in the event of station blackout.

Group 2 is also supplied from the emergency power system which, in the event of station blackout, is supplied from diesel generators. Not more than 40 s are required by the diesel generators before they can take over the full load.

For Group 3, the power source is the operating and the stand-by unit transformers.

The emergency diesel generator plant is intended for supplying power to the safety system devices under emergency conditions. For each unit of the power station there is provision for three 6.3 MW 6.3 kV generators located in three isolated structural compartments. Each compartment is a self-contained one-generator power plant functioning as one channel of the protective safety system. Each compartment is equipped with independent systems of fuel, oil and starting-air supply, together with cooling, heating, auxiliary power supply, control and monitoring systems.

Diesel fuel is supplied by pipelines from the base store to 100 m^3 capacity underground tanks with a minimum of two days' stock of fuel. Each compartment contains a service tank of oil sufficient for operation of the emergency diesel plant for 20 days.

The diesel generators are started by compressed air, which is kept in two cylinders. The reserve of air is sufficient for six successive starts. The cylinders are refilled from two automatic compressors.

Each compartment has its own 6.3/0.4 kV single-transformer sub-station, connected to the 6.3 kV station auxiliary power system buses, for users of auxiliary power supply.

The automatic control circuits which ensure the operation of the stand-by diesel plant in the event of station blackout in an accident situation are supplied from an independent source - 24 V batteries on floating charge from a rectifier. The stand-by plant is fully automated and is designed to start and operate for 240 h without requiring constant attendance by personnel.

In normal operation of the power plant unit the diesel generators remain on stand-by (ready to start automatically at any moment), and are switched on only for testing the safety systems. Under stand-by conditions the mechanisms of the station auxiliary power supply are supplied with 6.3 and 0.4 kV AC from the emergency power supply buses, so that the temperature of the internal-circuit water and oil is maintained within specified limits.

Under emergency conditions requiring the operation of safety systems the diesel generators start automatically. The time of automatic start of a diesel generator is about 10 seconds - from the instant when the instruction to start is given to the instant when it is ready to take the load. As soon as the rated revolutions and voltage are reached, the diesel generators are connected to the de-energized emergency power supply buses, followed by stepped increase of load.

The stand-by diesel generator plant has three completely independent channels, each of which is capable of carrying out the required functions in full.

2. WATER SUPPLY FOR THE SAFETY INJECTION AND EMERGENCY CORE COOLING SYSTEMS

The system of intake and filling of the main circuits of the unit and the reserves in the pressure suppression pool of the accident containment system are used for supplying water to the emergency core cooling system and the emergency feedwater system.

The system which is most important for reactor safety is the clean condensate system. It is used for filling the main and auxiliary circuits and for make-up under normal operating conditions, for safety injection in the event of breaks in the feedwater supply system and for emergency core cooling during accidents caused by pipe breaks in the circulation circuit.

The clean condensate system includes storage tanks and pumps. The storage tanks (CCST) have a capacity of 3000 m^3 for two units of the station. During power operation of the unit these tanks have a minimum reserve of 2000 m^3 of water. The tanks are refilled with chemically desalinated water from the chemical water purification system at a maximum flow rate of 110 t/h and with the secondary steam condensate of the evaporators of the floor drains purification system at a maximum flow rate of 45 t/h. The condensate is supplied from two monitor tanks with a total capacity of 400 m^3. Under emergency conditions the four circuit-water collecting tanks with a capacity of 750 m^3 each can be used as the CCST reserve.

During operation of the reactor the following pumps are connected to the CCST:

- The ECCS pumps which supply water to the unaffected half of the reactor in the event of circuit pipe break accidents;
- The pumps for independent make-up of the circulation circuit (clean condensate pumps 1, 2, 3), which supply clean condensate directly to the suction of the electrical emergency feedwater pumps or to the de-aerator equalizer line under conditions associated with steaming of the electrical feedwater pump, break of the latter's suction lines and loss of power;
- The pumps for filling and make-up of tanks of the reactor auxiliary systems and water purification systems (clean condensate pumps 4, 5, 6). These pumps ensure the operation of the safety-related systems of the reactor and the normal operation systems.

All pump units supplying water to the clean condensate storage tanks are powered from the diesel generators in the case of an emergency condition coupled with loss of plant auxiliary power supply. The pumps on the line connecting the clean condensate storage tanks and the circuit water collecting tanks are also connected to the diesel generators. Therefore, in the event of an accident accompanied by loss of plant auxiliary power supply, condensate can be supplied from these tanks to the pressure header of the clean condensate pumps for circuit make-up or to the de-aerators or to the suction of the electrical emergency feedwater pumps for delivery directly to the moisture separators.

Another important source of water for reactor cooling during an accident caused by a circulation circuit pipe break is the reserve of water in the pressure suppression pool of the accident containment system (ACS). This pool has a reserve of 3200 t of water. In an accident caused by a circuit pipe break, the water and steam from the ruptured section go to the above pool, where the steam is condensed. From this pool the emergency core cooling pumps deliver water to the affected half of the reactor, ensuring its cooling and make-up. The necessary temperature conditions in the pool are maintained by a special sprinkler cooling system, which has circulation pumps and heat-exchangers. The circulation pumps of this system are also powered from the diesel generators in the event of loss of plant auxiliary power supply.

Thus, the total reserves of clean condensate are about 5000 m^3. In accidents with uncompensated coolant loss, such as steamline break, this reserve of water will be sufficient for reactor water make-up for seven days. By this time, the steam production in the reactor due to after-heat release drops to 10 t/h.

Compensating reactor water make-up can be arranged from the chemical water purification system or from the floor drains purification system or from other special water purification systems of the power station.

3. ACCIDENT CONTAINMENT SYSTEM (ACS)

This is designed to localize radioactive releases during accidents involving rupture of any reactor cooling circuit piping except the steam-water lines and the upper part of downcomers, which are situated in the compartments of the drum-type steam separators (DSS). The ACS is a system of leaktight compartments and includes a condensation-type pressure suppression system, a system for heat removal from the latter and from the leaktight compartments, a system of shut-off and isolation valves and a system of hydrogen removal from the leaktight compartments.

The system of leaktight compartments (see Fig. 2) includes the following reactor compartments:

- Sealed pressure-resistant enclosures (items 1 and 2 in Fig. 2) located symmetrically in relation to the reactor axis and designed for an overpressure of 0.265 MPa;

- Compartments of the distribution group headers and lower water lines, which are also located symmetrically in relation to the reactor axis and separated from each other by the supporting cross-piece of the reactor with passages having a total area of 5 m^2. According to the strength requirements for the reactor structural components, a pressure rise above 0.2 MPa is not permitted in these compartments;

- Compartment of the steam distribution space (item 5);

- Compartment of the condensation-type pressure suppression system (CPS), a part of which is filled with water and the rest with air (item 8).

The leaktight compartments are connected with each other by means of valves of the following three types:

- Non-return valves (item 9) located in the floor separating the compartment of the distribution group header and lower water lines from the steam distribution space;

- Non-return valves (item 10) located in the openings of the floor separating the air-filled part of the pressure suppression pool and the sealed pressure-resistant enclosures;

- Panels of non-return valves (item 11) located in the partitions separating the steam distribution space and the sealed pressure-resistant enclosures.

The pressure-resistant enclosures and the steam distribution space are connected with the water-filled part of the CPS through the steam extraction channels (item 17), the lower parts of which are located 1.2 m below the water level.

In normal operation the system of leaktight compartments and the CPS remain on stand-by. Under emergency conditions the system functions in the following manner. In the event of a rupture of circulation circuit pipes, the boiling coolant is ejected into the pressure-resistant enclosure. The steam formed raises the pressure inside it. The non-return valves of the panels between this enclosure and the steam distribution space (item 11) open at a pressure difference exceeding 0.02 MPa. When the pressure in this enclosure attains the value sufficient for displacing the water column in the steam extraction channels, the steam-air mixture begins to enter the condensation systems. In bubbling through the water layer the steam condenses, while the air is collected in the air-filled part of the condensation system. When the pressure in it goes above 5 kPa, the bleed valves, which connect the air-filled part of the condensation system and the unaffected pressure-resistant enclosure, open and a part of the air goes into this enclosure. Thus, the volume of the latter enclosure is used for reducing the pressure in the affected enclosure. In the given emergency condition the non-return valves (item 9) remain closed.

If the circulation circuit break occurs in the compartment of the distribution group headers and lower water lines, a pressure rise above 0.02 MPa opens the non-return valves connecting this compartment with the steam distribution space. From the tunnel, through the steam discharge channels, the steam-air mixture goes to the water-filled part in the central section of the condensation system located under the steam distribution space. A rise in pressure in the air space of the condensation system opens the bleed valves connecting this part with the two pressure-resistant enclosures. In this emergency condition the volumes of the two enclosures are used for reducing the pressure in the affected compartment, while the valves in the panels (item 11) remain closed.

All the leaktight compartments and the CPS are lined with 4 mm sheet steel and are subjected to check tests for local and integral leaktightness.

The necessary temperature conditions in the CPS are maintained during an accident by the operation of its heat removal system, which includes pumps (item 14) and heat-exchangers (item 15). The pumps take water from the pressure suppression pool, deliver it to the heat-exchangers for cooling and return it to the pool through the sprinkler system (item 13).

The temperature conditions in the pressure-resistant enclosures are maintained by spray coolers (item 12).

A surface-type condenser (item 16) is used for additional heat removal from the CPS during a design-basis accident.

Upon receiving the signal for circulation circuit pipe break, the shut-off and isolation valves automatically disconnect the pipelines intersecting the leaktight circuits (system of floor drains, water make up, water purification of the CPS, etc.) in order to prevent the escape of radioactive substances outside the leaktight circuit of the accident containment system.

The purpose of the hydrogen removal system (HRS) is to preclude a rise in hydrogen concentration in the leaktight compartments above the permissible value under normal, emergency and post-emergency conditions. This system continuously monitors the concentration of hydrogen in the compartments and removes it through the purification system to the station ventilation stack. In the initial period of a design-basis accident, the HRS is disconnected automatically from the leaktight compartment system in order to reduce radioactive releases into the atmosphere. When the maximum permissible concentration of hydrogen is attained (in about two hours) the HRS goes into operation and removes hydrogen from the leaktight compartments.

4. TOTAL LOSS OF STATION AUXILIARY POWER SUPPLY

This is possible during system failures coupled with breakdown of the power system. It is one of the most serious emergency conditions since it is accompanied by a sharp fall in the coolant flow rate in the core and an appreciable rise in pressure in the drum-type steam separators (DSS). Upon loss of station auxiliary power supply, the main circulation pumps and the feedwater pumps trip, the reactor protection system is actuated and the emergency stop valves before the turbines close. The turbogenerator trip leads to a pressure rise in the circuit and opens the main safety valves, as a result of which the pressure in the circuit begins to fall and the safety valves close. Forty seconds after the power loss the emergency feedwater pumps are switched on and begin to supply water to the DSS from the clean condensate tanks at the rate of 500 m^3/h and a temperature of 40°C. Thereupon, as was demonstrated earlier on the simulator and verified in full-scale tests, a stable natural circulation regime is established in the circuit and core after-heat removal causes no complications. Figures 3 and 4 present the results of the calculation of changes in plant parameters in the initial period of power loss coupled with simultaneous failure of one independent component of the safety system (non-closure of one main safety valve of the first group).

The following conclusions may be drawn on the basis of our studies:

- The thermal parameters under emergency after-heat removal conditions in the event of loss of station auxiliary power supply do not exceed the safe limits;

- The reactor core is cooled reliably, first by the coasting of the main circulation pumps and then by natural circulation of the coolant;

- The pressure in the DSS, even in the event of failure of one main safety valve, does not rise above the permissible value (115% of the rated value) owing to the operation of the other seven.

In the case where during loss of auxiliary power supply the settings of two main safety valves are disturbed after they are actuated, the emergency core cooling system (ECCS) is used to prevent a break in natural circulation and to ensure reliable core cooling. The prolonged cooling sub-system of the ECCS goes into action when the pressure in the DSS falls to 4.2 MPa and supplies water by the ECCS pumps from the clean condensate tank of the pressure suppression pool to the group distribution headers of both halves of the reactor at the rate of 500 m^3/h to each.

Apart from the ECCS pumps, the feedwater injection in the DSS is performed by the emergency feed pumps, which also have emergency power supply and can ensure a delivery of 1500 m^3/h. After the level in the DSS has been restored, the water flow rate into the circuit falls to the value needed for making up the leakage of the coolant.

During loss of auxiliary power supply the operation of the reactor safety injection system, automatic control system, process instrumentation system, etc., is ensured by the emergency power supply system. Figure 5 shows the power supply diagram for the emergency feed pumps. The power supply diagram for the ECCS pumps is similar to that of the emergency feed pumps.

5. REACTOR COOLING BY NATURAL AIR CIRCULATION

A feature of the RBMK reactor is that the steam and water lines with no heat insulation have a developed heat-exchange surface. The compartments containing such lines have knock-down panels in the upper and lower parts to prevent pressure rises above the limiting value in the event of circuit pipe break accidents (Fig. 6). This feature of the reactor equipment arrangement enables after-heat removal from the core by means of air cooling. It is based on the principle that when the knock-down panels are opened in the upper and lower parts of the drum-type steam separator (DSS) compartments, this creates natural circulation of the surrounding air which is brought into motion by the difference in density between the cold atmospheric air and the heated air in contact with the hot steam and water lines.

This air cooling feature of the RBMK reactor can be utilized during a beyond-design-basis accident associated with total loss of auxiliary power supply and failure of the stand-by diesel generator plant.

In such an accident, the emergency feed pumps cannot deliver feedwater. There is only one possibility of supply from the ECCS water cylinders of about 40 t at a temperature of 40°C by injection into the circuit. After-heat removal and reactor cooling will take place only through evaporation of water from the reactor circuit and heat losses. During the first hour after shutdown of the reactor about 140 t of water evaporates. By then the power plant personnel will have time to force open the knock-down panels of the DSS compartments and to put into operation the "air heat-exchanger" formed by the steam and water line pipe bundles. At the latter's temperature of 280°C the "heat-exchanger" has a capacity of 22 MW for an ambient air temperature of 30°C.

The reactor after-heat removal capacity will fall to the level of the maximum capacity of the "air heat-exchanger" five hours after reactor shutdown. In the preceding four hours with the heat-exchanger in operation 48 t of water will evaporate. For removal of the heat accumulated in the graphite structure and amounting to 278 x 10^6 kJ, it is necessary to evaporate 110 t of the circulation circuit water and 40 t of the ECCS water with a temperature of 40°C. Thus, the total quantity of water evaporating from the circulation circuit will be 295 t, i.e. somewhat less than the initial stock of water in the four DSS (314 t).

On this basis it can be concluded that, under conditions of a beyond-design-basis accident caused by total loss of station auxiliary power supply and failure of the stand-by diesel generator plant, reliable cooling of the reactor core will be ensured for an unlimited time. This is achieved by the use of a structural feature of the RBMK reactor and its layout in the power station.

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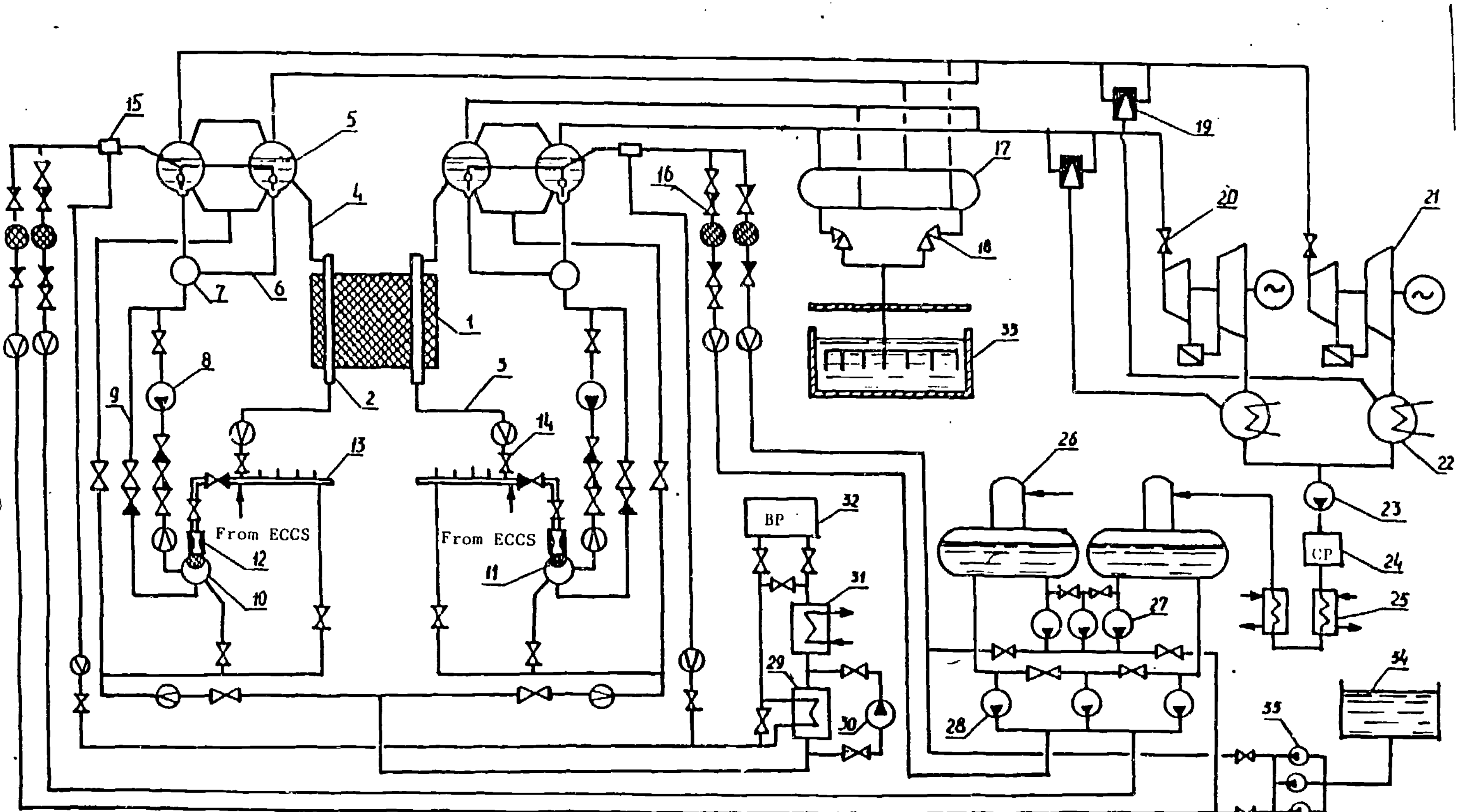

Fig. 1. Flow diagram of a unit with the RBMK-1000 reactor: (1) reactor; (2) fuel channel; (3) water lines; (4) steam and water lines; (5) drum-type steam separators; (6) downcomer; (7) MCP suction header; (8) MCP; (9) MCP header bypass; (10) MCP pressure header; (11) mechanical filter; (12) limiter; (13) group header; (14) isolation and control valve; (15) mixer; (16) feedwater equipment; (17) steam header; (18) main safety valves; (19) fast-acting steam dump system; (20) turbine stop and control valve; (21) turbogenerator; (22) condenser; (23) condensate pump; (24) condensate purification (CP); (25) heater; (26) de-aerator; (27) electrical emergency feed pumps; (28) electrical feed pumps; (29) blowdown regenerator; (30) coolant pump; (31) blowdown aftercooler; (32) bypass purification (BP); (33) condensation:

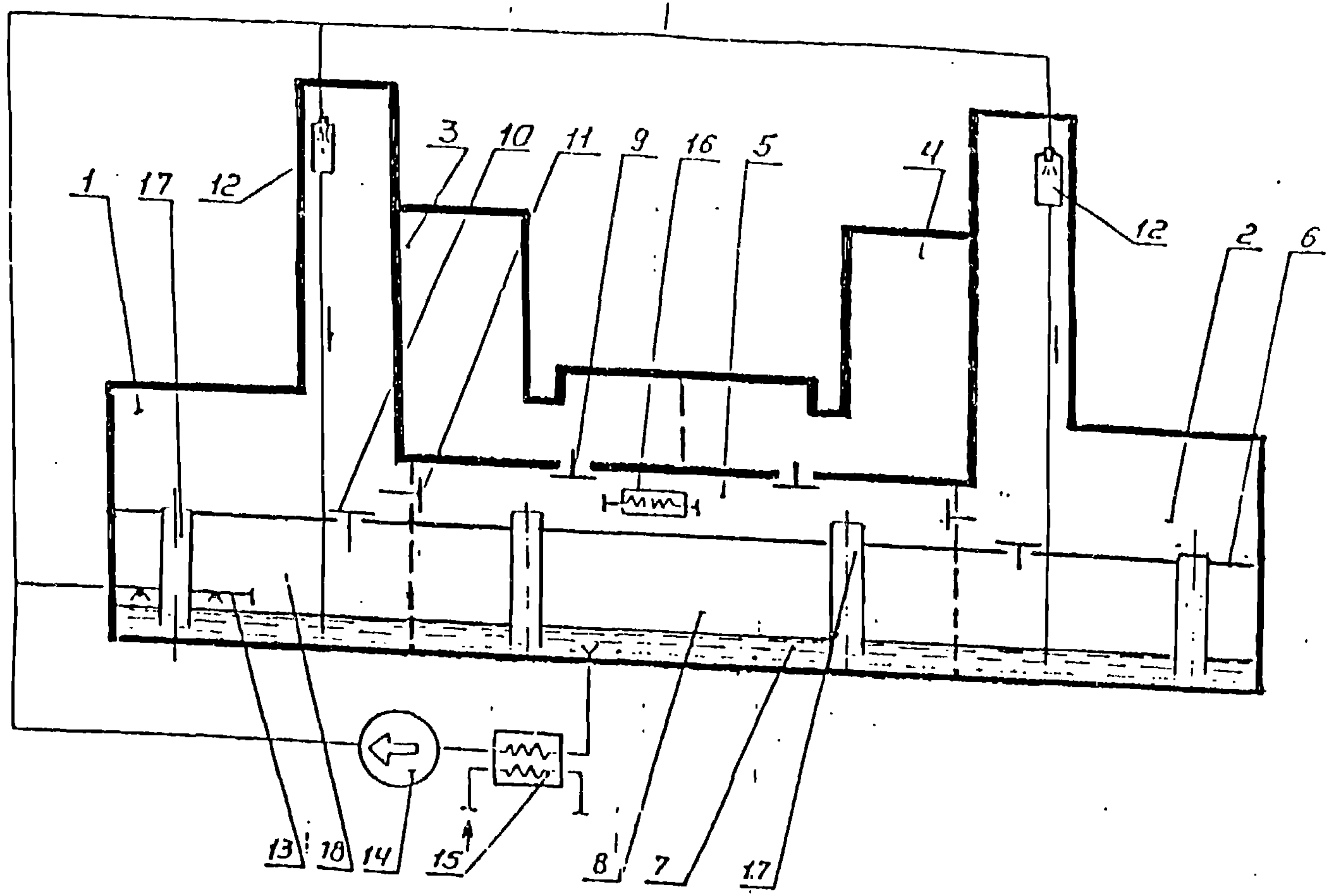

Fig. 2. Diagram of the system of leaktight compartments

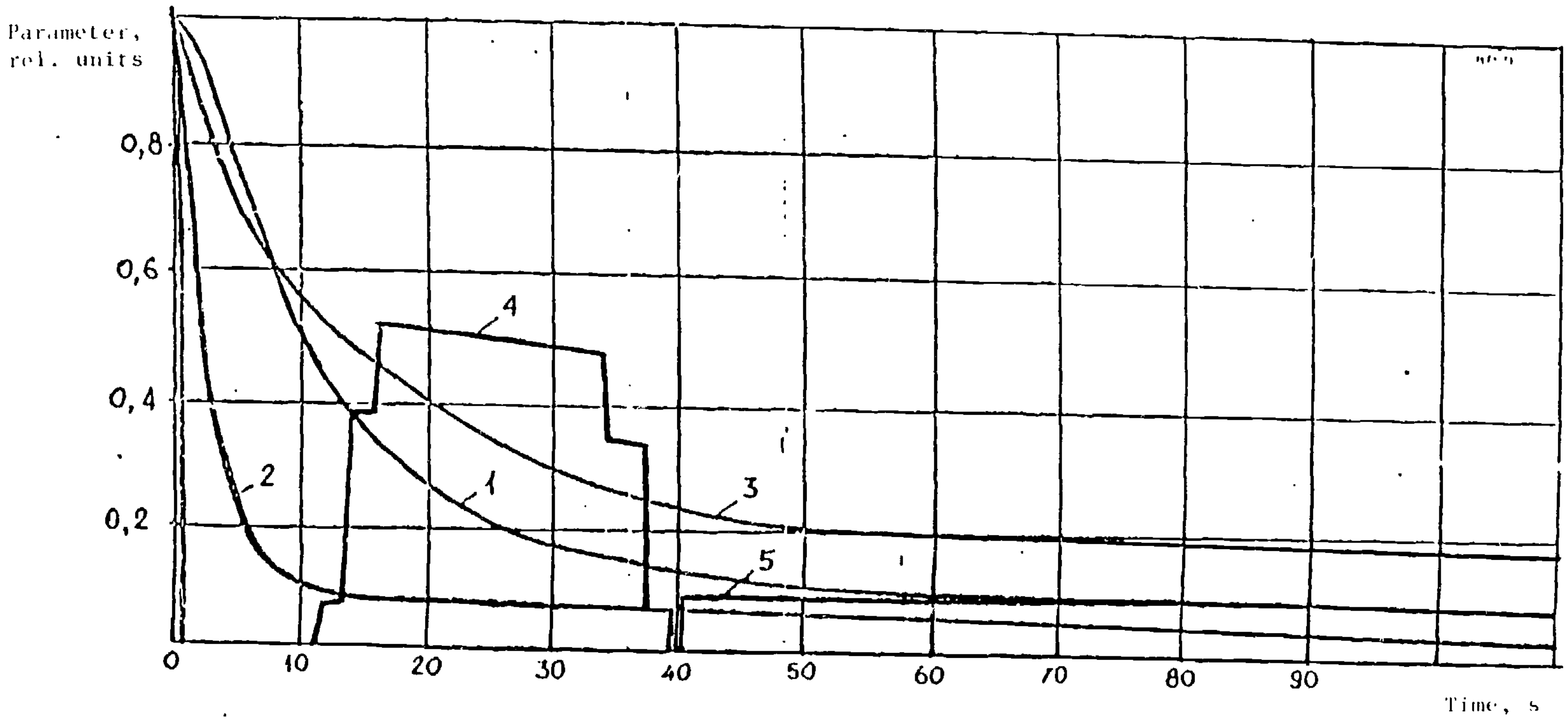

Fig. 3. Changes in the parameters of the unit during total loss of station auxiliary power supply: (1) thermal power; (2) neutron-flux-determined power; (3) circulating water flow rate; (4) steam flow rate; (5) water flow rate from emergency feed pump.

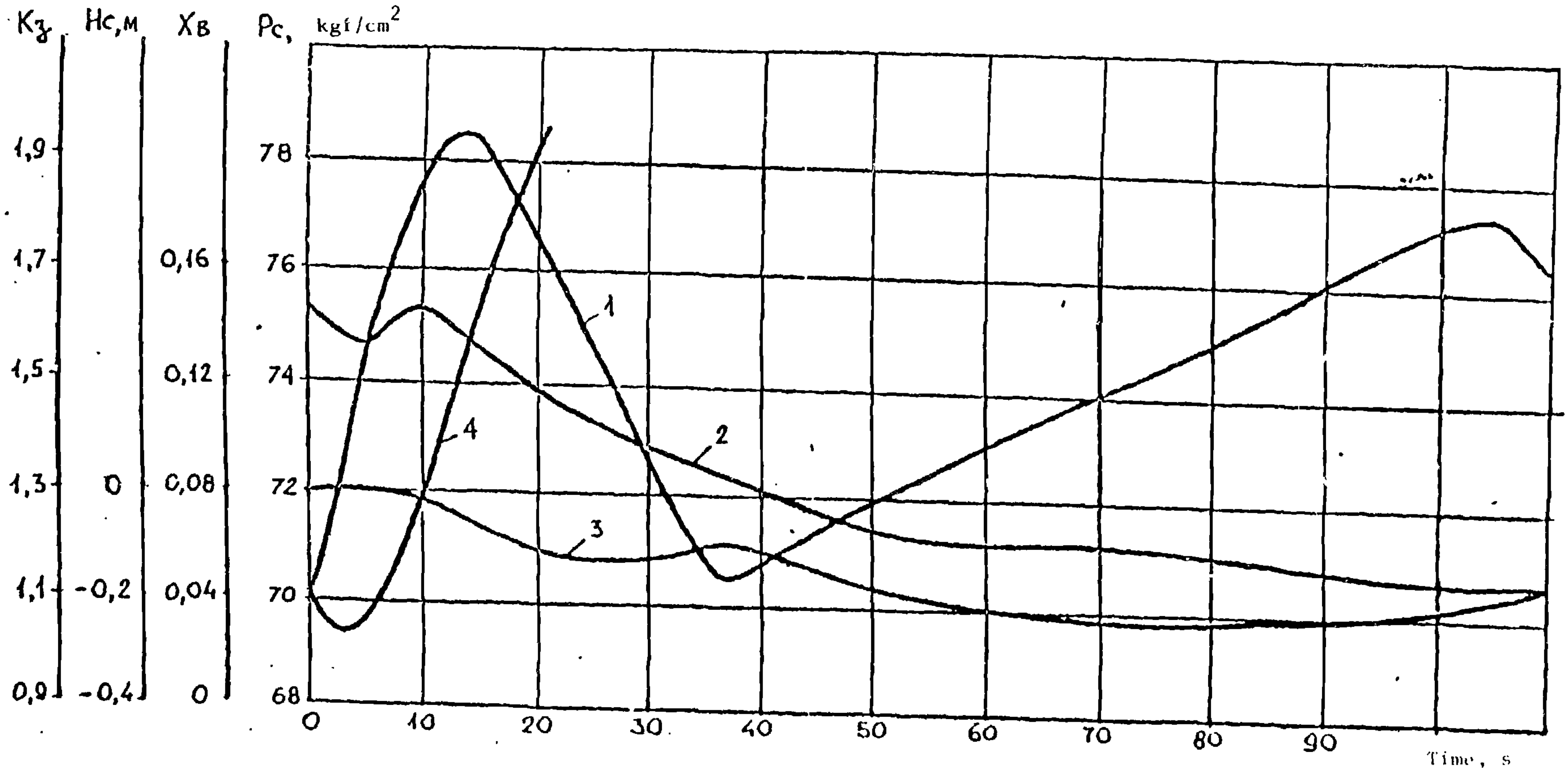

Fig. 4. Changes in the parameters of the unit during total loss of station auxiliary power supply: (1) pressure in drum-type steam separator (P_c); (2) steam quality at core outlet (X_B); (3) deviation of steam quality from steady-state value (H_c); (4) margin up to the critical limits of heat exchange (K_z).

Power supply diagram for emergency feed pumps

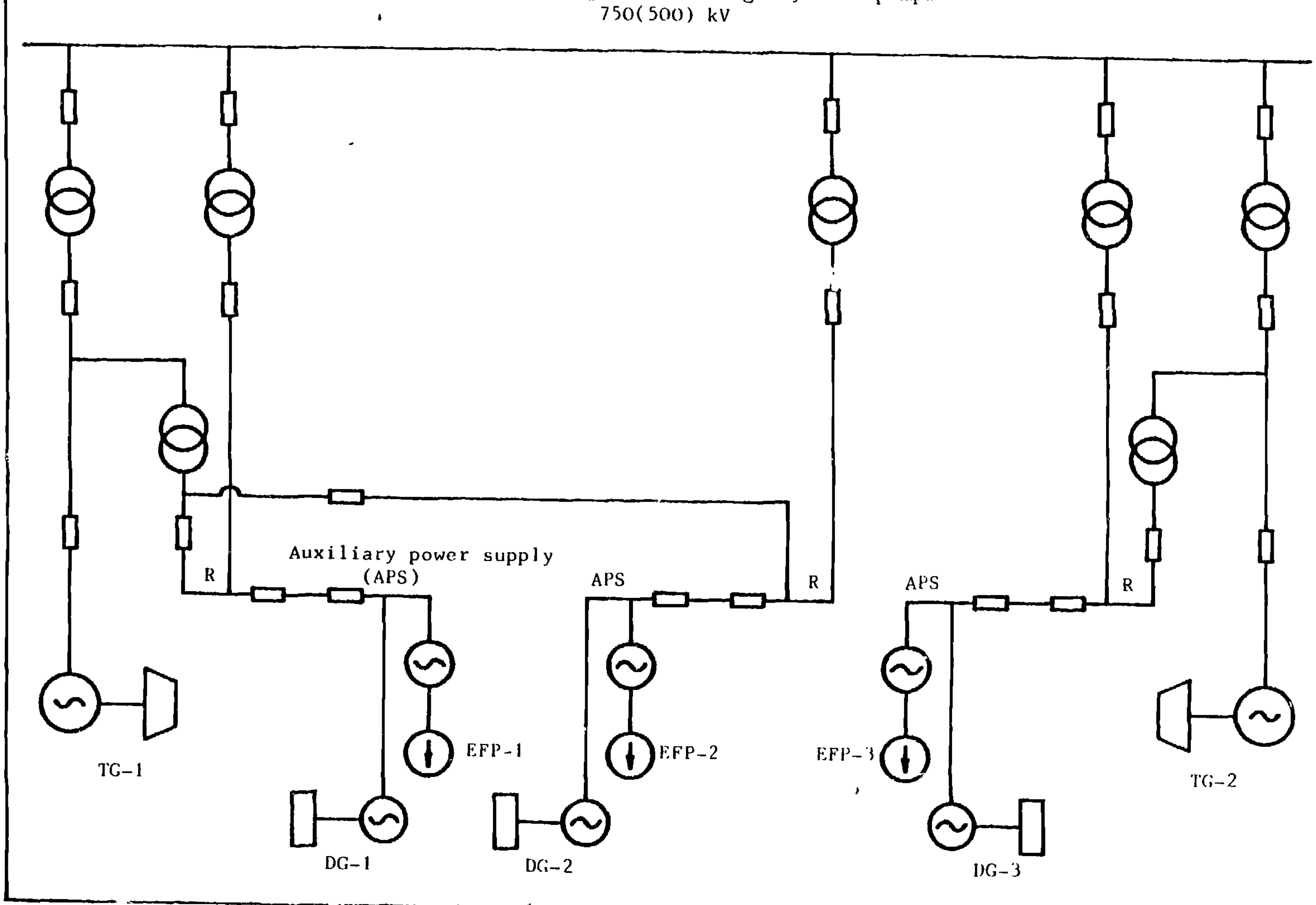

Fig. 5

Diagram of air cooling

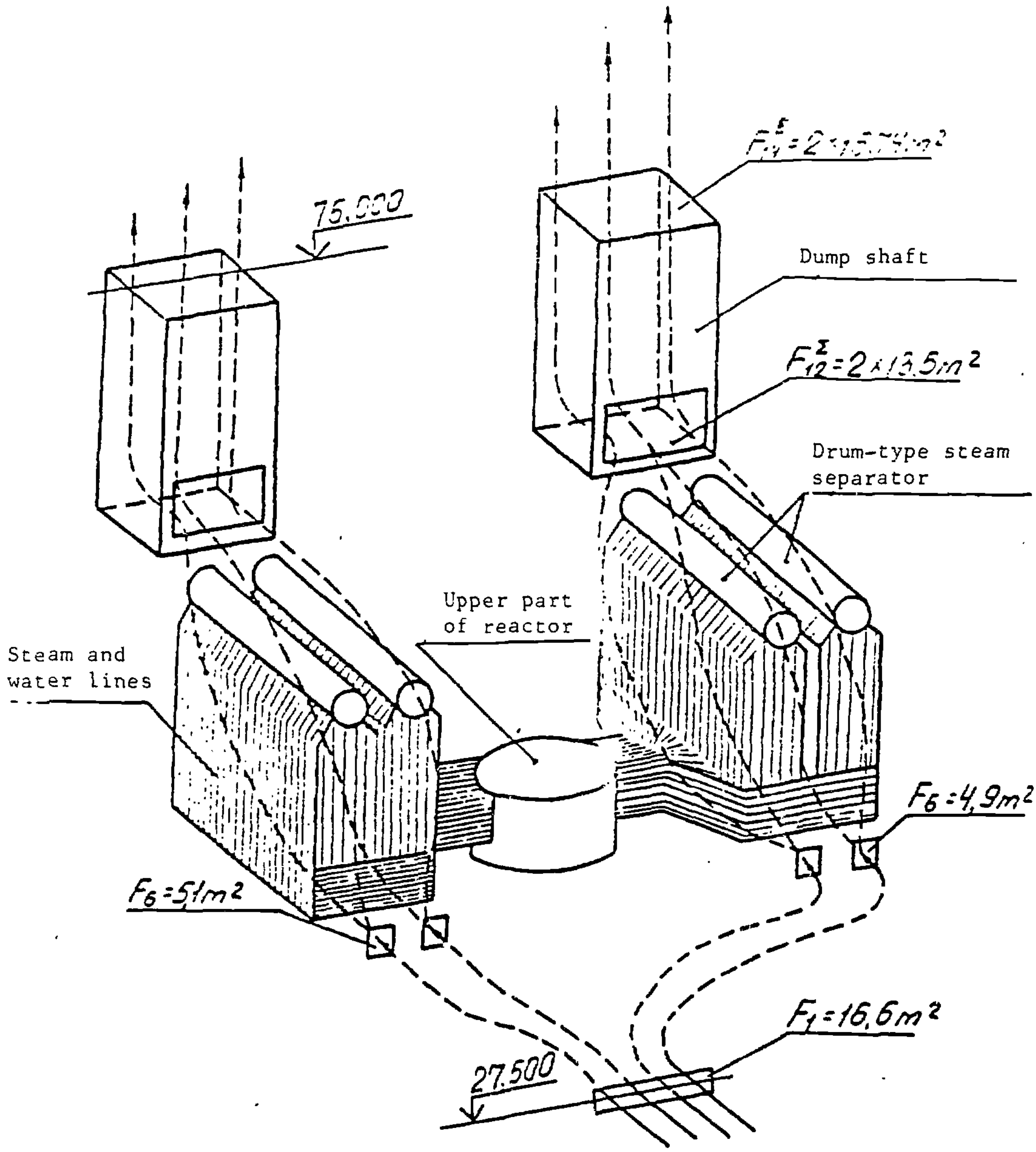

Fig. 6

ИСПОЛЬЗОВАНИЕ СИСТЕМ АЭС С РЕАКТОРАМИ РБМК ДЛЯ СМЯГЧЕНИЯ ПОСЛЕДСТВИЙ АВАРИЙ

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В настоящее время в СССР находятся в эксплуатации I4 энергоблоков с реакторами РБМК-IOOO единичной электрической мощностью IOOO МВт и 2 энергоблока с реакторами РБМК-I500 единичной электрической мощностью I500 МВт /I/. Готовится к пуску 5 блок Курской АЭС с реактором РБМК-IOOO.

Ядерный энергетический реактор типа РБМК является гетерогенным канальным реактором на тепловых нейтронах, в котором в качестве замедлителя используется графит, а в качестве теплоносителя - кипящая легкая вода. I660 топливных каналов размещаются в вертикальных отверстиях графитовых колонн и представляют собой трубу диаметром 80 мм из циркониевого сплава. Внутри канала установлена тепловыделяющая кассета, имеющая в сечении I8 стержневых твэлов диаметром I3,6 мм в оболочке из циркониевого сплава. Тепловая схема является типичной для одноконтурных энергетических установок с кипящим реактором (рис.I). Контур многократной принудительной циркуляции (КМПЦ) состоит из двух параллельных петель, в каждой из которых осуществляется охлаждение половины топливных каналов реактора. Циркуляция теплоносителя осуществляется с помощью главных циркуляционных электронасосов (ГЦН). Подвод недогретой воды и отвод пароводяной смеси от каждого канала осуществляется по индивидуальным трубопроводам. В корпусных сепараторах горизонтального типа при давлении около 7 МПа происходит разделение пара и воды. Насыщенный пар направляется в две турбины, а его конденсат возвращается после подогрева и деаэрации в сепараторы, откуда, смешиваясь с отсепарированной насыщенной водой, подается ГЦН на вход в реактор.

В процессе проектирования первых АЭС с реактором РБМК-IOOO был сформирован перечень исходных аварийных событий и проанализированы наиболее неблагоприятные пути их развития /2/. На основе опыта эксплуатации и по мере ужесточения требований к безопасности АЭС, которое имеет место в мировой атомной энергетике вообще, первоначальный перечень исходных событий был значительно расширен. Перечень планируемых исходных событий аварий для АЭС с реакторами РБМК, сегодня включает в себя около 50 ситуаций, которые могут быть разделены на 5 основных типов:

1) ситуации с изменением реактивности - падение или самопроизвольное извлечение стержня СУЗ, отказы в аппаратурной части СУЗ, обезвоживание каналов СУЗ, ложное включение системы аварийного охлаждения реактора;

2) аварии в системе охлаждения активной зоны - отключение ГЦН, обрыв тарелки обратного клапана или диска запорной задвижки ГЦН, отказы в системе подачи питательной воды и т.д.;

3) аварии, вызванные разрывом трубопроводов КМПЦ, паропроводов и питательных трубопроводов;

4) аварии с отключением или отказом оборудования - обесточивание собственных нужд, отключение турбогенератора (ТГ), открытие главных предохранительных клапанов (ГПК) с последующей непосадкой и т.д.;

5) прочие аварии - при перегрузке топлива разгрузочно-загрузочной машиной, пожар, затопление при наводнении и т.п.

Непревышение проектных пределов повреждения твэлов и допустимого уровня выхода радиоактивных нуклидов в окружающую среду в режимах нормальной эксплуатации и в аварийных ситуациях обеспечивается следующими системами реакторной установки и АЭС в целом:

- системой управления и защиты (СУЗ), в состав которой кроме традиционных устройств входят локальные автоматические регуляторы, группы локальной аварийной защиты и быстродействующая аварийная защита;

- системой контроля и регулирования энергораспределения, которая осуществляет непрерывный дискретный контроль энерговыделений по радиусу и высоте активной зоны;

- системой аварийного охлаждения реактора (САОР), обеспечивающей теплоотвод от активной зоны при авариях, вызванных разрывом трубопроводов или отказом оборудования, которая автоматически включается в работу по 5 независимым алгоритмам в зависимости от типа аварии;

- системой поканального и группового контроля герметичности оболочек твэлов топливных каналов;

- системой индивидуального контроля целостности топливных каналов и каналов СУЗ;

- системой защиты от превышения давления в циркуляционном контуре - группы предохранительных клапанов (ГПК);

- системой контроля и регулирования расхода теплоносителя в топливных каналах и каналах СУЗ;

- системой контроля температуры металлоконструкций реактора и графита по объему кладки;

- системой централизованного контроля, осуществляющей сбор и контроль входной аналоговой и дискретной информации, расчет параметров безопасного ведения технологических процессов и сигнализацию об отклонениях контролируемых и расчетных параметров от заданных уставок;

- системой сброса парогазовой смеси из реакторного пространства при аварийной разгерметизации каналов реактора;

- разгрузочно-загрузочной машиной, позволяющей производить перегрузку выгоревших или потерявших герметичность топливных кассет без останова реактора;

- системой локализации аварий, предназначенной для приема аварийных выбросов теплоносителя, конденсации пара и выдержки неконденсирующихся радиоактивных газов;

- системой водоснабжения основных контуров реактора и систем аварийного охлаждения реактора;

- системой электроснабжения собственных нужд АЭС;

- резервной дизель-электростанцией.

Перечисленные выше системы реакторной установки (РУ) позволяют обеспечить безопасное протекание аварийных процессов в соответствии с принятым перечнем проектных аварий /3/. Система СУЗ автоматически по отклонениям параметров воздействует на мощность реактора, системы охлаждения и подпитки активной зоны, регуляторы давления и уровня в сепараторах, автоматически поддерживают теплогидравлические характеристики в безопасных пределах, САОР обеспечивает аварийное расхолаживание реактора и т.д. Безотказная работа этих систем в аварийных ситуациях обеспечивается прежде всего надежным электроснабжением, водоснабжением и резервированием оборудования. Поэтому наиболее важными системами АЭС с точки зрения обеспечения ее безопасности являются системы электроснабжения собственных нужд и водоснабжения систем аварийной подпитки реактора и локализации выбросов теплоносителя.

В данном докладе приводится описание этих систем АЭС с РУ РБМК-1000 второго поколения, принципов их работы в нормальных и аварийных режимах. Рассмотрена аварийная ситуация, вызванная полным обесточиванием собственных нужд, и запроектная авария, сопровождающаяся обесточиванием собственных нужд и отказом резервной дизельной электростанции.

I. СИСТЕМА НАДЕЖНОГО ЭЛЕКТРОСНАБЖЕНИЯ

Система надежного электроснабжения АЭС предназначена для обеспечения электроэнергией оборудования, которое необходимо для расхолаживания реактора как в нормальных, так и в аварийных режимах.

Все потребители собственных нужд АЭС по требованиям к их надежности разделяются на 3 группы:

I группа - потребители, не допускающие по условиям безопасности перерывов электропитания (более чем на доли секунды для осуществления автоматических переключений) весь период протекания аварийного процесса, включая ситуацию с полным обесточиванием собственных нужд АЭС.

К I группе потребителей относятся: аппаратура и сервоприводы СУЗ, информационно-вычислительная система, контрольно-измерительные приборы, электроприводы задвижек и арматуры САОР, СЛА, систем аварийной подпитки контура и т.д.

2 группа - потребители, допускающие перерывы питания на время, определяемое условиями безопасности (от десятков секунд до нескольких минут) и требующие обязательного наличия питания после срабатывания АЗ реактора. К потребителям этой группы относятся механизмы, обеспечивающие расхолаживание реактора в аварийных режимах, сопровождаемых полной потерей напряжения на шинах собственных нужд АЭС (аварийный насос пруда охладителя, насосы контура охлаждения СУЗ, насосы САОР, АПН, насосы чистого конденсата, пожарные насосы, насосы технической воды, насосы спринклерно-охладительной системы).

3 группа – потребители, допускающие перерывы питания на время аварийного ввода резерва и не требующие обязательного наличия питания после срабатывания АЗ реактора. К ним относятся ГЦН, питательные электронасосы (ПЭН), конденсатные насосы, механизмы вспомогательных систем реактора и машинного зала и другое оборудование, обеспечивающее работу блока в нормальных режимах.

Потребители I группы запитываются от сети надежного питания, источником питания которой являются статические инверторные преобразователи. Они являются агрегатами бесперебойного питания и в случае обесточивания АЭС питаются от аккумуляторных батарей.

2 группа потребителей запитывается также от сети надежного питания, которая, в свою очередь, в случае обесточивания АЭС питается от дизель-генераторов. Время разворота дизель-генераторов до приема или полной нагрузки составляет не более 40 с.

Источником питания потребителей 3 группы являются рабочие и резервные блочные трансформаторы.

Резервная дизельная электростанция (РДЭС) предназначена для снабжения электроэнергией потребителей систем безопасности в аварийных режимах. Для каждого блока АЭС предусматривается установка в трех изолированных строительных ячейках по одному дизель-генератору мощностью 6,3 МВт напряжением 6,3 кВ. Каждая ячейка РДЭС представляет собой автономную одноагрегатную электростанцию, выполняющую функции одного канала обеспечивающей системы безопасности. Каждая ячейка оборудуется автономными системами обеспечения топливом, маслом и воздухом для пуска, а также системами охлаждения, отопления, электроснабжения собственных нужд, управления и контроля.

Дизельное топливо по трубопроводам подается с базисного склада в подземные баки неснижаемого двухсуточного запаса топлива емкостью 100 м^3. В каждой ячейке размещается расходный бак масла из условия обеспечения работы РДЭС в течение 20 суток.

Запуск дизелей осуществляется сжатым воздухом, который хранится в двух баллонах. Запаса воздуха достаточно для 6 последовательных пусков, пополнение баллонов сжатым воздухом предусмотрено от двух автоматизированных компрессоров.

Для питания потребителей собственных нужд в каждой ячейке РДЭС предусматривается своя однотрансформаторная подстанция 6,3/0,4 кВ, подключенная к шинам 6,3 кВ надежного питания АЭС.

Питание цепей автоматики дизель-генератора, обеспечивающих выполнение функций РДЭС в режиме аварийного обесточивания АЭС, осуществляется от независимого источника электроснабжения - аккумуляторной батареи 24 В, работающей в режиме постоянного подзаряда от выпрямительного устройства. РДЭС полностью автоматизирована и предусматривает запуск и работу без постоянного обслуживания оперативным персоналом в течение 240 ч.

При нормальной работе блока дизель-генераторы находятся в режиме "ожидание" (постоянная готовность к автоматическому пуску) и включаются только для опробования систем безопасности. В режиме "ожидание" электропитание механизмов собственных нужд осуществляется переменным током напряжением 6,3 и 0,4 кВ от шин надежного электроснабжения, при этом поддерживается в заданных пределах температура воды внутреннего контура и масла.

В аварийных режимах АЭС, требующих работы систем безопасности, производится автоматический запуск дизель-генераторов. Время автоматического запуска дизель-генератора с момента подачи команды на пуск до момента готовности к принятию нагрузки составляет около 10 с. По достижению номинальных оборотов и напряжения дизель-генераторы включаются на обесточенные шины надежного питания с последующим ступенчатым набором нагрузки.

РДЭС имеет три полностью независимых канала, каждый из которых способен выполнить требуемые функции в полном объеме.

2. ВОДОСНАБЖЕНИЕ СИСТЕМ АВАРИЙНОЙ ПОДПИТКИ И ОХЛАЖДЕНИЯ РЕАКТОРА

Для организации водоснабжения систем аварийного охлаждения реактора и аварийной подачи питательной воды используется система приема и заполнения основных контуров блока и запасы воды в бассейне-барботере системы локализации аварий.

Наиболее важной, с точки зрения безопасности работы реакторной установки, является система чистого конденсата, которая служит для заполнения основных и вспомогательных контуров блока и их подпитки в режимах нормальной эксплуатации, для аварийной подпитки реактора при нарушениях в системе подачи питательной воды, для аварийного охлаждения реактора при авариях, вызванных разрывом трубопроводов циркуляционного контура.

Система чистого конденсата включает в себя баки-накопители и перекачивающие насосы. Объем баков-накопителей (БЧК) составляет 3000 м^3 на два блока АЭС. При работе блока на мощности запас воды в БЧК составляет не менее 2000 м^3. Подпитка БЧК осуществляется химобессоленной водой, поступающей из системы химводоочистки (ХВО) с максимальным расходом 110 т/ч, и конденсатом вторичного пара выпарных аппаратов системы очистки трапных вод с максимальным расходом 45 т/ч. Конденсат подается из двух контрольных баков чистого конденсата общим объемом 400 м^3. В аварийных режимах резервом БЧК могут служить 4 бака сбора контурных вод (БСКВ) объемом 750 м^3 каждый.

К БЧК при работе реактора подключены:

- насосы САОР подающие воду на неаварийную половину реактора при авариях связанных с разрывом трубопроводов контура;

- насосы автономной подпитки циркуляционного контура (НЧК-I,2,3), обеспечивающие подачу чистого конденсата непосредственно на всас АПЭН или в уравнительную линию деаэраторов в режимах, связанных с запариванием ПЭН, разрывом всасывающих трубопроводов ПЭН, обесточиванием;

- насосы заполнения и подпитки баков вспомогательных систем реакторного отделения и систем водоочистки (НЧК-4,5,6), обеспечивающие работу систем реакторного отделения важных для безопасности и систем нормальной эксплуатации.

Все насосные агрегаты, подающие воду из БЧК, имеют электропитание от дизель-генераторов на случай аварийной ситуации, сопровождающейся обесточиванием собственных нужд блока. К дизель-генераторам подключены и насосы на линии связи БЧК и баков сбора контурных вод. Поэтому при аварии, сопровождающейся обесточиванием собственных нужд блока конденсат из этих баков может быть подан или в напорный коллектор НЧК для подпитки контура, или в деаэраторы, или на всас АПЭН для подачи воды непосредственно в сепараторы пара.

Другим важным источником воды для охлаждения реактора при авариях, вызванных разрывом трубопроводов циркуляционного контура, является запас воды в бассейне-барботере системы локализации аварии (СЛА). Бассейн-барботер СЛА имеет запас воды в количестве 3200 т. При авариях, вызванных разрывом трубопроводов контура, вода и пар из сечения разрыва поступает в бассейн-барботер СЛА, где пар конденсируется. Из бассейна-барботера насосы

САОР подают воду в аварийную половину реактора, обеспечивая ее охлаждение и подпитку. Необходимый температурный режим в бассейне-барботере поддерживается специальной спринклерно-охладительной системой, имеющей циркуляционные насосы и теплообменники. Циркуляционные насосы этой системы также имеют электропитание от дизель-генераторов на случай обесточивания собственных нужд.

Таким образом, суммарные запасы чистого конденсата составляют около 5000 м^3. При авариях с некомпенсируемой утечкой теплоносителя, например, при разрыве паропровода, данного запаса воды будет достаточно для подпитки реактора в течение 7 суток. Паропроизводительность реактора за счет остаточных тепловыделений к этому времени снижается до 10 т/ч.

Компенсирующая подпитка реактора может быть организована или от системы ХВО, или от системы очистки трапных вод или от других систем спецводоочистки АЭС.

3. СИСТЕМА ЛОКАЛИЗАЦИИ АВАРИЙ

Система локализации аварий предназначена для локализации радиоактивных выбросов при авариях с разгерметизацией любых трубопроводов контура охлаждения реактора, за исключением трубопроводов пароводяных коммуникаций (ПВК) и верхней части опускных труб, которые находятся в помещениях барабанов-сепараторов. СЛА представляет систему герметичных помещений и включает в себя барботажно-конденсационное устройство (БКУ), систему отвода тепла из БКУ и герметичных помещений, систему отсечной и герметизирующей арматуры, систему удаления водорода из герметичных помещений (СУВ).

Система герметичных помещений (см.рис. 2), включает в себя следующие помещения реакторного отделения:

- прочно-плотные боксы (поз.I и 2), расположенные симметрич относительно оси реактора и рассчитанные на избыточное давление 0,265 МПа;

- помещения раздаточных групповых коллекторов и нижних водяных коммуникаций (РГК-НВК), также симметричные относительно оси реактора и разделенные между собой опорной крестовиной реактора, имеющей проходные окна общей площадью 5 $м^2$. Эти помещения по условиям прочности элементов конструкции реактора не допускают роста давления выше 0,2 МПа;

- помещение парораспределительного коридора (поз.5);

- помещение барботажно-конденсационного устройства, часть которого заполнена водой (поз.7), а остальная часть - воздухом (поз.8).

Герметичные помещения соединены между собой с помощью клапанов следующих трех типов:

- обратные клапаны (поз.9), установленные в перекрытии между помещением РГК-НВК и парораспределительным коридором;

- обратные клапаны (поз.10), установленные в проемах перекрытия, разделяющего воздушное пространство барботера и прочно-плотных боксов;

- панели обратных клапанов (поз.11), установленные в перегородках, разделяющих парораспределительный коридор и прочно-плотные боксы.

Помещения прочно-плотных боксов и парораспределительного коридора соединяются с водяным объемом БКУ пароотводящими каналами (поз.17), нижняя часть которых заведена под уровень воды на глубину 1,2 м.

При нормальной эксплуатации система герметичных помещений и БКУ работают в режиме "ожидания". В аварийных ситуациях система функционирует следующим образом. При разгерметизации трубопроводов циркуляционного контура в прочно-плотном боксе происходит выброс вскипающего теплоносителя. Образующийся пар вызывает рост давления в помещении. Обратные клапаны панелей между прочно-плотным боксом с парораспределительным коридором (поз.11) открываются при перепаде давления более 0,02 МПа. Когда давление в прочно-плотном боксе достигает величины, достаточной для вытеснения столба воды в пароотводящих каналах, начинается поступление паровоздушной смеси в конденсационные устройства. При барботаже через слой воды пар конденсируется, а воздух собирается в воздушном объеме конденсационного устройства. При достижении давления в нем более 5 кПа открываются перепускные клапаны, соединяющие воздушное пространство конденсационного устройства с неаварийным прочно-плотным боксом, и часть воздуха перепускается в этот бокс. Таким образом, объем неаварийного прочно-плотного бокса используется для снижения давления в аварийном прочно-плотном боксе. В данной аварийной ситуации обратные клапаны (поз.9) остаются закрытыми.

Если разгерметизация циркуляционного контура происходит в помещении РГК-НВК, то рост давления в нем выше 0,02 МПа приводит к открытию обратных клапанов,соединяющих помещения РГК-НВК и парораспределительный коридор. Из коридора паровоздушная смесь по паросбросным каналам поступает в водный объем центральной части конденсационного устройства, расположенного под парораспределительным коридором. Рост давления в воздушном пространстве конденсационного устройства вызывает открытие перепускных клапанов, соединяющих воздушное пространство конденсационного устройства с двумя прочно-плотными боксами. В этой аварийной ситуации объемы обоих прочно-плотных боксов используются для снижения давления в аварийном помещении, а клапаны панелей (поз.II) остаются закрытыми.

Все герметичные помещения и БКУ облицованы листовой сталью толщиной 4 мм и подвергаются проверочным испытаниям на локальную и интегральную герметичность.

Для поддержания необходимого температурного режима БКУ при аварии включается в работу система отвода тепла из БКУ и герметичных помещений, включающая в себя насосы (поз.I4) и теплообменники (поз.I5). Насосы забирают воду из бассейна-барботера, подают ее в теплообменники для охлаждения и возвращают в бассейн через спринклерную систему (поз.I3).

Температурный режим прочно-плотных боксов обеспечивается эжекционными охладителями (поз.12).

Для дополнительного отвода тепла из БКУ при максимальной проектной аварии используется конденсатор поверхностного типа (поз.16).

По сигналу аварии с разрывом трубопроводов циркуляционного контура с помощью отсечной и герметизирующей арматуры автоматически отключаются трубопроводы пересекающие контур герметизации (система трапных вод, подпитки, очистки вод БКУ и т.д.) с целью предотвращения выхода радиоактивных веществ за пределы контура герметизации системы локализации аварий.

Для исключения повышения концентрации водорода в герметичных помещениях СЛА выше допустимой в нормальных, аварийных и послеаварийных режимах используется система удаления водорода. Данная система осуществляет непрерывный контроль концентрации водорода в помещениях и его отвод через системы очистки в вентиляционную трубу АЭС. В начальный период при максимальной проектной аварии СУВ автоматически отключается от системы герметичных помещений с целью снижения радиоактивных выбросов в атмосферу. При достижении предельно допустимой концентрации водорода через ~ 2 часа СУВ вводится в работу и осуществляет удаление водорода из герметичных помещений.

4. ПОЛНОЕ ОБЕСТОЧИВАНИЕ СОБСТВЕННЫХ НУЖД АЭС

Обесточивание собственных нужд АЭС возможно при системных авариях, сопровождающихся "развалом" энергосистемы. Это одна из наиболее тяжелых аварийных ситуаций, поскольку она сопровождается резким снижением расхода теплоносителя через активную зону и значительным ростом давления в БС. При обесточивании собственных нужд АЭС отключаются главные циркуляционные насосы, питательные насосы, срабатывает аварийная защита, закрываются стопорные клапаны перед турбинами. Отключение турбогенераторов приводит к повышению давления в контуре и открытию главных предохранительных клапанов, вследствие чего давление в контуре начинает снижаться и предохранительные клапаны закрываются. Через 40 с с момента обесточивания процесса включаются аварийные питательные насосы (АПН), которые начинают подавать воду в БС, из баков чистого конденсата (БЧК) с расходом 500 м^3/ч и температурой около 40 $^{\circ}$С. При этом, как было установлено ранее на моделирующем стенде и проверено в натурных испытаниях, в контуре устанавливается устойчивый режим естественной циркуляции и расхолаживание активной зоны не вызывает каких-либо осложнений. Результаты расчетных исследований изменения параметров блока в начальный период режима обесточивания с наложением одновременного отказа одного независимого элемента системы безопасности (незакрытие одного главного предохранительного клапана первой группы) показаны на рис. 3,4

На основании проведенных исследований можно сделать выводы:

- теплотехнические параметры в режиме аварийного расхолаживания реактора при обесточивании собственных нужд блока не выходят за безопасные пределы;

- активная зона реактора надежно охлаждается сначала за счет выбега ГЦН, затем за счет естественной циркуляции теплоносителя;

- давление в БС даже при отказе одного ГПК за счет работы оставшихся семи ГПК не возрастает выше допустимого значения (115 % от номинального значения).

В случае, когда при обесточивании собственных нужд происходит непосадка двух ГПК, после их срабатывания, для исключения срыва естественной циркуляции и обеспечения надежного охлаждения активной зоны реактора используется система аварийного охлаждения реактора. Подсистема длительного расхолаживания САОР включается при снижении давления в БС до 4,2 МПа и подает воду насосам САОР из БЧК бассейна-барботера в раздаточные групповые коллекторы обеих половин реактора с расходом 500 $м^3/ч$ в каждую.

Кроме насосов САОР подпитка реактора в БС осуществляется насосами АПН, которые также имеют надежное электроснабжение и могут обеспечить подачу до 1500 $м^3/ч$ воды. После восстановления уровня в БС расход воды в контур снижается до значения, необходимого для восполнения утечки теплоносителя.

В режиме обесточивания собственных нужд работа систем аварийной подпитки реактора, автоматики, контрольно-измерительных приборов и т.д. обеспечивается системой надежного электроснабжения. На рис. 5 представлена схема электроснабжения аварийных питательных насосов. Схема электроснабжения насосов САОР аналогична приведенной схеме электроснабжения АПН.

5. ОХЛАЖДЕНИЕ РЕАКТОРА ЕСТЕСТВЕННОЙ ЦИРКУЛЯЦИЕЙ ВОЗДУХА

Особенностью реактора РБМК является наличие развитой поверхности теплообмена трубопроводов пароводяной коммуникации, не имеющих теплоизоляции. Помещения, где расположены трубопроводы ПВК в нижней и верхней части имеют вышибные панели для исключения повышения давления выше предельного, при аварийных разрывах трубопроводов контура (см.рис.6). Данная особенность компановки реакторного оборудования позволяет организовать отвод остаточных тепловыделений от активной зоны реактора путем организации воздушного охлаждения. Принцип воздушного расхолаживания заключается в том, что при открытии вышибных панелей в верхней и нижней частях помещения барабан-сепараторов организуется естественная циркуляция окружающего воздуха, движущим напором которой является разность плотностей холодного атмосферного воздуха и нагретого воздуха при контакте с горячими трубопроводами ПВК.

Возможность организации воздушного расхолаживания реактора РБМК позволяет использовать это качество реакторной установки в условиях запроектной аварии, связанной с полным обесточиванием собственных нужд и отказом резервной дизельной электростанции.

При такой аварии не может быть обеспечена подача питательной воды насосами АПН. Существует единственная возможность подачи воды от гидробаллонов САОР в количестве ~ 40 т с температурой 40 °С за счет ее передавливания в контур. Съем остаточных тепловыделений и расхолаживание реактора будет осуществляться только за счет испарения воды реакторного контура и тепловых потерь. В течение первого часа после заглушения реактора испаряется около 140 т воды. К этому времени персонал АЭС имеет возможность принудительно открыть вышибные панели помещений барабан-сепараторов и включить в работу "воздушный теплообменник", образованный пучками трубопроводов ПВК. Мощность "воздушного теплообменника" при температуре трубопроводов пароводяных коммуникаций 280 °С составляет около 22 МВт при температуре окружающего воздуха 30 °С.

Мощность остаточных тепловыделений реактора снизится до уровня максимальной мощности "воздушного теплообменника" через 5 часов после останова реактора. В течение предыдущих 4 часов при введенном в работу теплообменнике испарится 48 т вод Для съема тепла, аккумулированного в графитовой кладке в количестве $278 \cdot 10^6$ кДж, потребуется испарить 110 т воды циркуляционного контура и 40 т воды САОР с температурой 40 °С. Таким образом, общее количество испарившейся из циркуляционного контура воды составит 295 т, т.е. несколько меньше начального запаса воды в четырех БС (314 т).

На основании этого можно сделать вывод, что в условиях протекания запроектной аварии, вызванной полным обесточиванием собственных нужд АЭС и полным отказом резервной дизельной электростанции, обеспечивается надежное охлаждение активной зоны реактора в течение неограниченного времени. Это достигается за счет использования конструктивной особенности реакторной установки РБМК и ее компановки в помещениях АЭС.

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2. Адамов Е.О., Василевский В.П., Черкашов Ю.М. Основные принципы и критерии обеспечения безопасности канальных уран-графитовых реакторов. Международный симпозиум по изучению опыта безопасной эксплуатации АЭС. Париж, Франция 16-20 мая 1988 г.
3. Адамов Е.О., Василевский В.П., Петров А.А., Черкашов Ю.М. Техническое обоснование безопасности канальных уран-графитовых реакторов РБМК. МАГАТЭ, Вена, Австрия, май 1989 г

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- 368 -

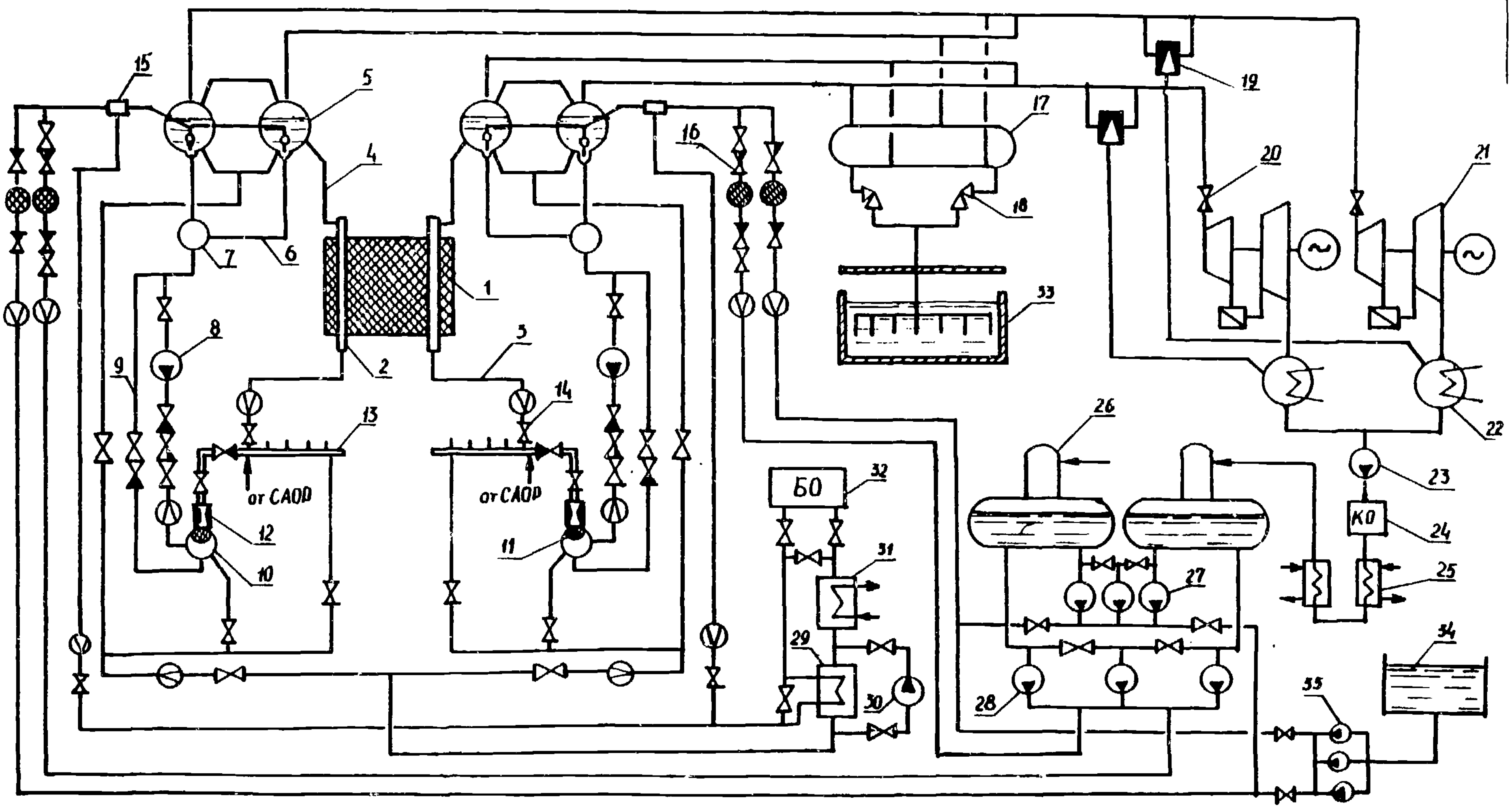

Рис. I Принципиальная схема блока с реактором РБМК-1000

I-реактор; 2-топливный канал; 3-трубопроводы ВК; 4-трубопроводы ПВК; 5-БС; 6-опускная труба; 7-всасывающий коллектор ГЦН; 8-ГЦН; 9-байпас коллекторов ГЦН; I0-напорный коллектор ГЦН; II-фильтр механический; I2-вставка ограничительная; I3-групповой коллектор; I4-запорно-регулирующий клапан; I5-смеситель; I6-питательный узел; I7-паровой коллектор; I8-ГПК; I9-БРУ-К; 20-СРК турбины; 2I-турбогенератор; 22-конденсатор; 23-конденсатный насос; 24-конденсатоочистка; 25-подогреватель; 26-деаэратор; 27-АПЭН; 28-ПЭН; 29-регенератор продувки; 30-насос расхолаживания; 3I-доохладитель продувки; 32-байпасная очистка; 33-конденсационное устройство СЛА; 34-бак с запасом воды; 35-насос аварийной подпитки БС.

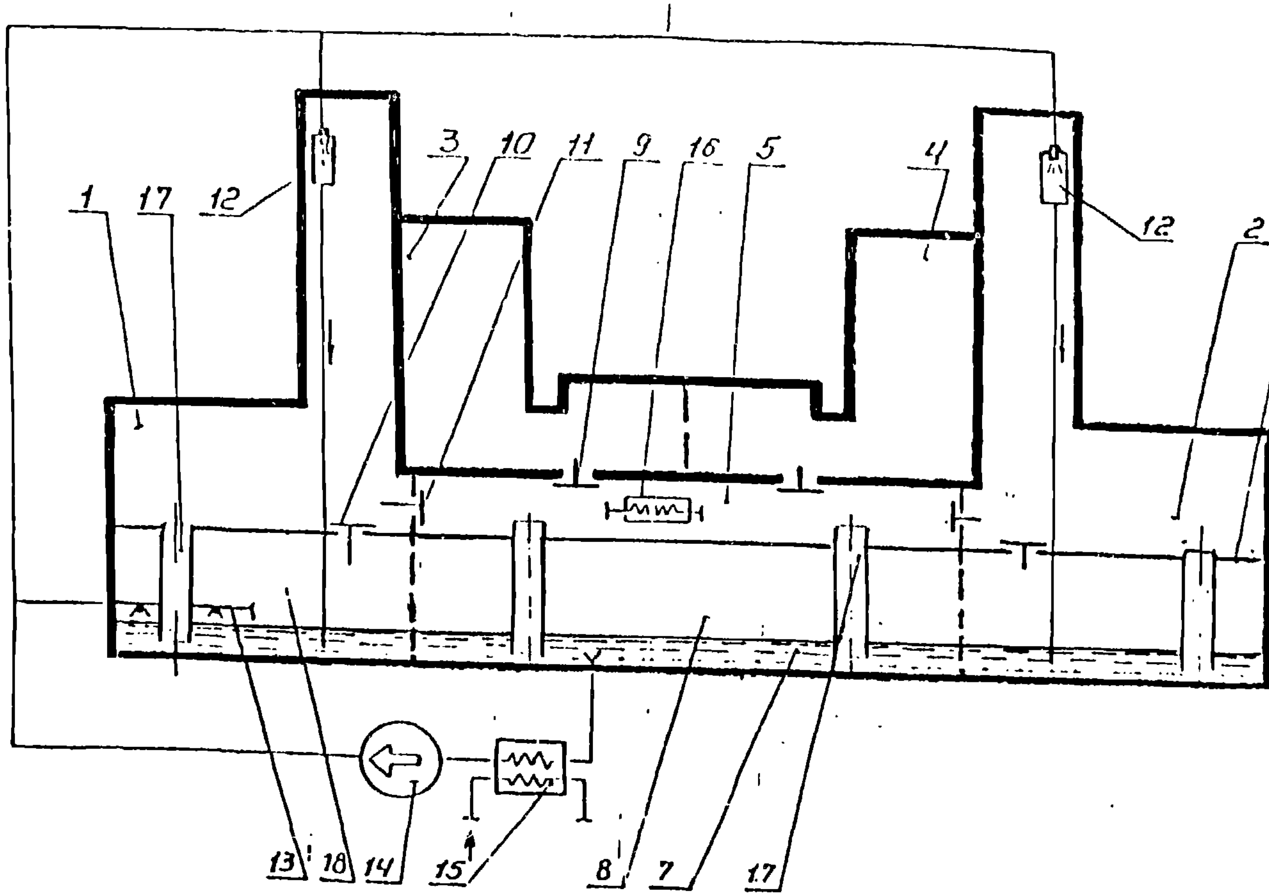

Рис. 2. Принципиальная схема системы герметичных помещений..

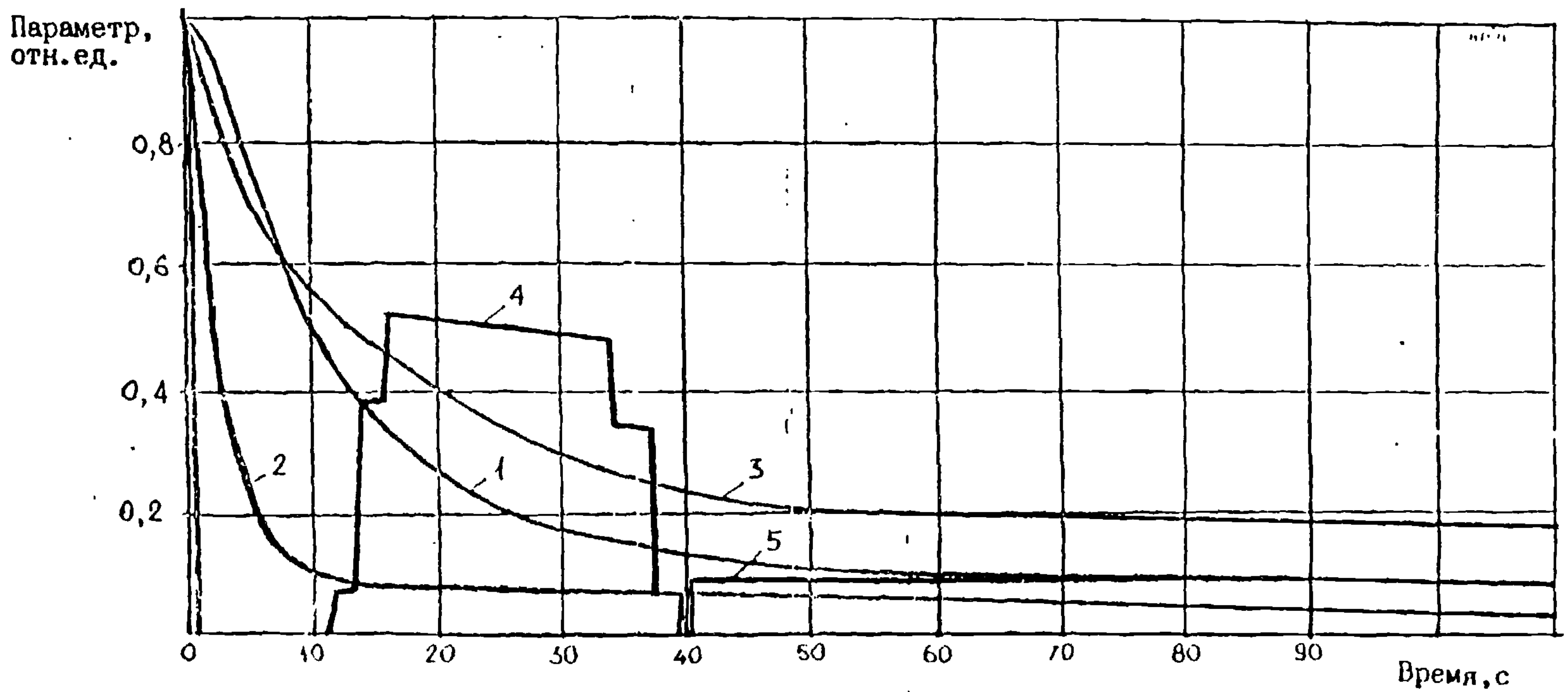

Рис. 3 Изменение параметров блока в режиме полного обесточивания собственных нужд блока:

I-тепловая мощность; 2-нейтронная мощность;

3-расход циркуляционной воды; 4-расход пара;

5-расход воды от АПН

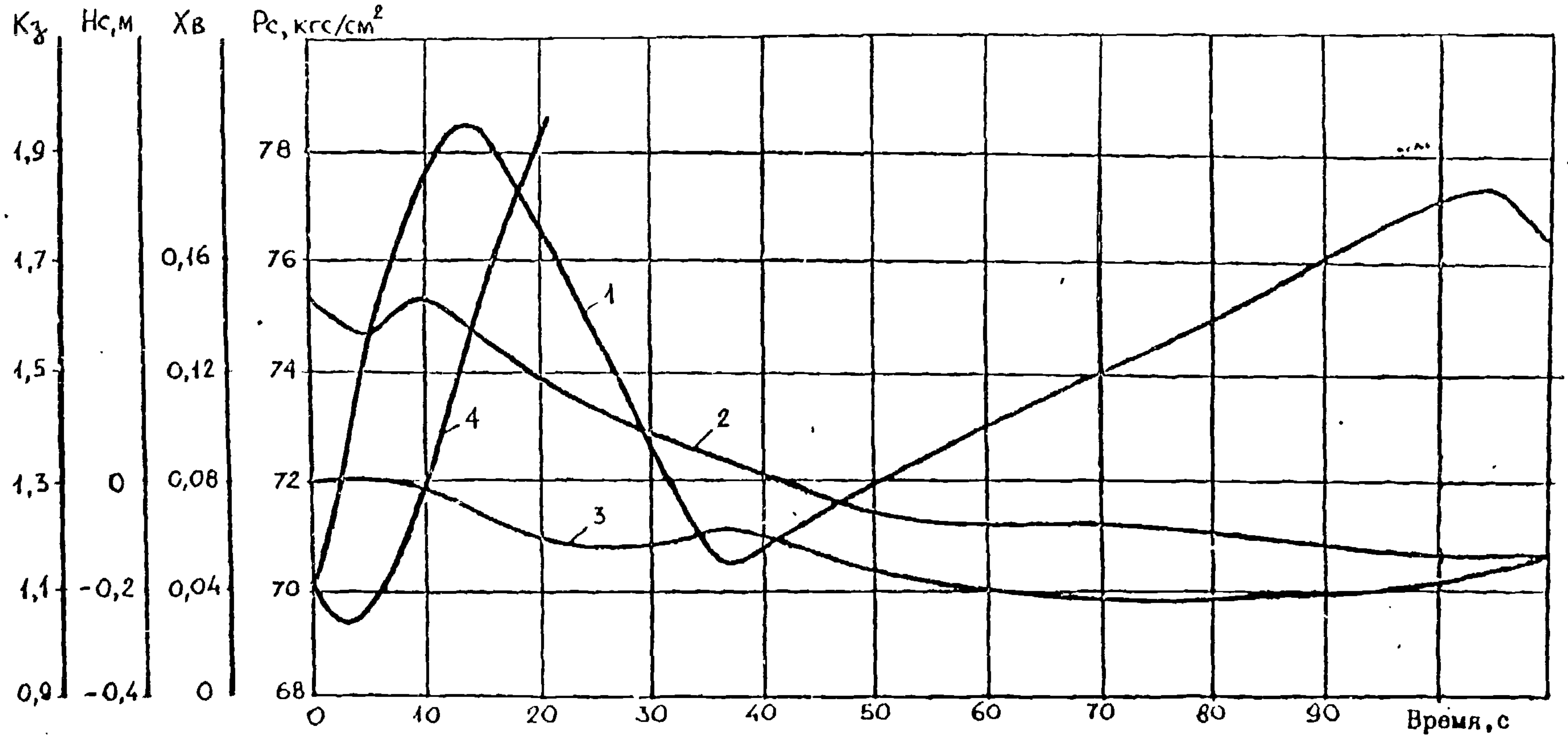

Рис. 4 Изменение параметров блока в режиме полного обесточивания собственных нужд блока:
I-давление в БС(Pc); 2-массовое паросодержание на выходе из активной зоны(Xв); 3-отклонение массового уровня от стационарного значения(Hc); 4-запас до кризиса теплообмена(Kз).

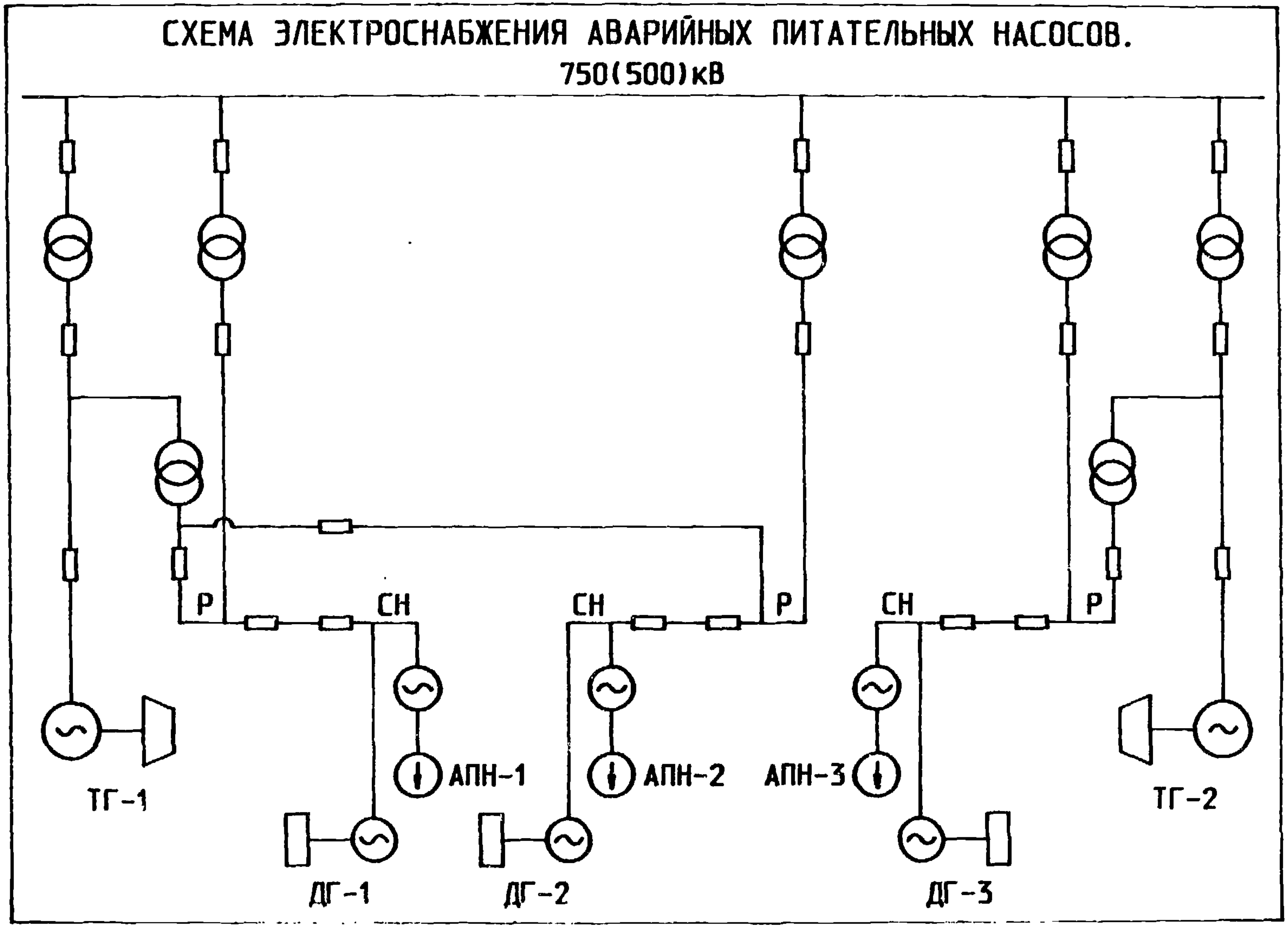
СХЕМА ЭЛЕКТРОСНАБЖЕНИЯ АВАРИЙНЫХ ПИТАТЕЛЬНЫХ НАСОСОВ.
750(500)кВ
Р
СН
СН
Р
СН
Р
АПН-1
АПН-2
АПН-3
ТГ-1
ТГ-2
ДГ-1
ДГ-2
ДГ-3

Рис. 5

Принципиальная схема
воздушного расхолаживания

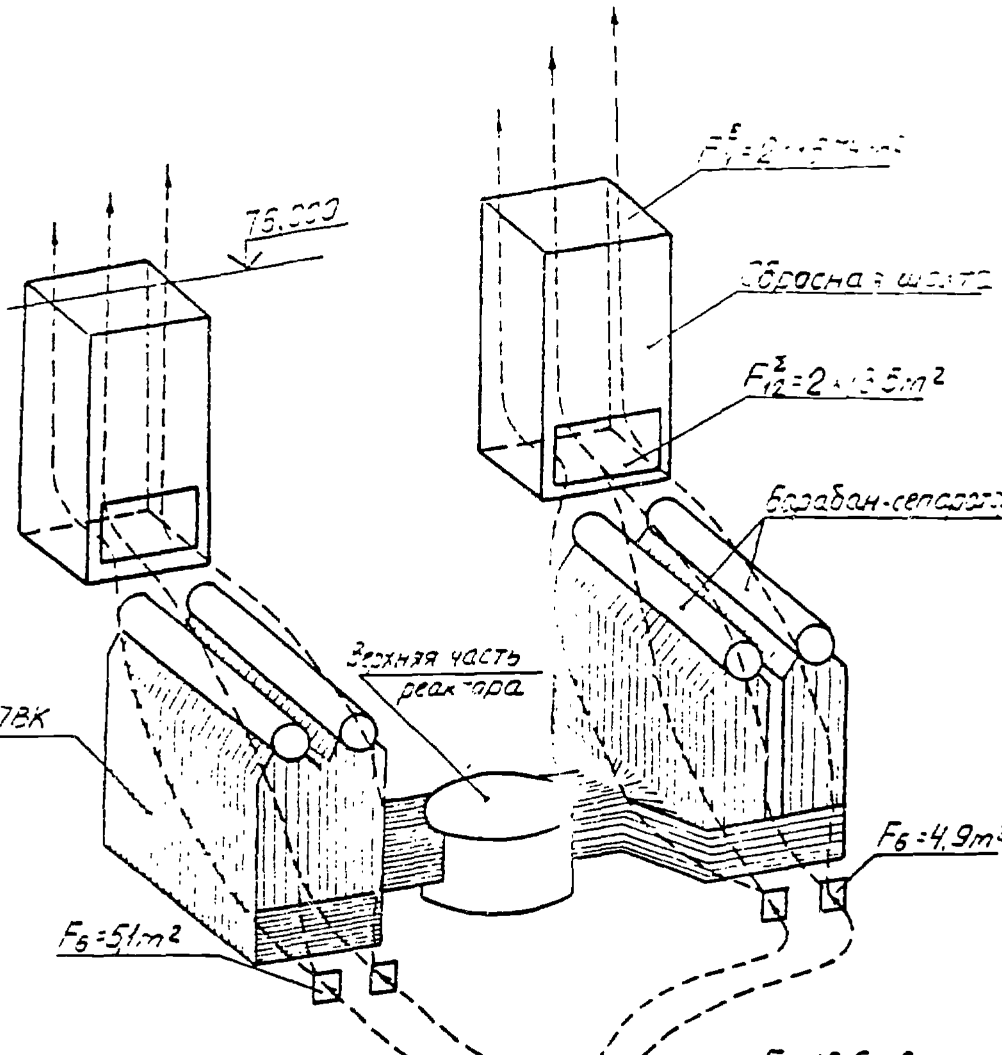

Рис. 6

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Department of Nuclear Energy and Safety
Division of Nuclear Safety

Issue No. 2
1990-11-26

NOTIFICATION OF AN AGENCY-SPONSORED MEETING

Title of meeting: TCM on Plant System Utilization for Accident Mitigation

Dates of meeting: 26-30 November 1990

Place of meeting: Garching, Germany

Scientific Secretary: H. Mauersberger

Secretary: W. Heilig

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