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Technical feasibility and reliability of passive safety systems for nuclear power plants

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FOREWORD

There were 432 nuclear power plants in operation in the world at the end of 1994. Over 17% of the world's electricity needs were supplied by nuclear power in 1994. The safety of nuclear power has an excellent record, with an accumulated experience of over 7200 reactor-years of operation. Few accidents have occurred in the history of nuclear energy. The two main accidents, namely Three Mile Island (TMI) and Chernobyl were caused by human error and the improper shutdown of safety systems designed specifically to prevent such accidents. The TMI accident demonstrated the importance of containment. Nearly all radioactivity was contained inside the plant and off-site releases were negligible (the highest exposure received by anyone during the accident amounted to the equivalent of a single X ray exposure). The accident at Chernobyl, however, resulted in a considerable release. Many people's scepticism about nuclear energy has either been started or magnified by this accident.

The future of nuclear power depends primarily on two factors: how well and how safely it actually performs and how safely nuclear power is perceived to perform. In response to this, designers and national and international organizations involved in nuclear power development, design and generation have paid increased attention to the safety of current and future nuclear power plants. Enormous efforts have been devoted to this subject worldwide.

Several new designs for future nuclear power plants have been developed. Many of these designs have adopted passive safety means to accomplish the required functions. Passive systems rely on natural forces and minimize the effect of human factors. The use of passive safety is also a desirable method of achieving simplification and increasing reliability.

The design, development and testing programmes of passive safety systems have reached a mature stage. Some designs, utilizing passive systems to accomplish the required safety functions, are currently in the detailed design stage. The International Atomic Energy Agency has long provided a forum for joint discussion and exchange of information on subjects of international interest. This Advisory Group meeting provided the opportunity to exchange information on the technical feasibility and reliability of passive safety systems.

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SUMMARY

One of the main goals of advanced reactor development is to further enhance nuclear power plant safety. Advanced reactor designs have generally incorporated improvements for accident prevention and mitigation which substantially reduce the potential effects of human error and which could result in significant reduction in emergency planning requirements.

The design of next generation plants has not been driven by public opinion. Even though improved, public acceptability is considered a prerequisite for the revival of the nuclear power market. For this reason, many reactor designs have chosen a new approach to gain the same and possibly higher level of safety, by system simplification and utilization of natural forces. Development activities in this area are integral parts of the advanced reactor programmes in many Member States.

Safety enhancements have in a number of the next generation plants to a considerable extent been attained by utilizing passive safety systems (for a definition of passive and active systems, see IAEA-TECDOC-626, Safety Related Terms for Advanced Nuclear Plants, 1991). These plants still use some active systems as a first line of defence for different safety functions. The tangible difference lies in designating the passive systems as the safety-graded ones for the various functions. This makes the passive systems the assumed available systems in the event of an accident. Passive safety systems for severe accident prevention and mitigation have been incorporated into some of the new designs and have been covered in previous IAEA activities. After substantial basic research and industrial development in different countries, it was felt that discussion of the technical feasibility and reliability of proposed passive systems would be very fruitful for all parties involved.

The design approaches and design descriptions of these systems have been presented at other meetings and conferences, and this meeting, therefore, concentrated on key issues connected with the technical feasibility and reliability of passive safety systems. Hence, the papers that were presented and discussed at the meeting focused on key components, features and phenomena that make passive safety systems feasible or highlighted problem areas in specific designs.

The meeting provided an overview of the key issues on passive safety. Technical problems which may affect future deployment, and the operating experience of passive systems and components, as well as, definitions of passive safety terms, were discussed. Advantages and disadvantages of passive systems were also highlighted. The philosophy behind different passive safety systems was presented and the range of possibilities between fully passive and fully active systems was discussed.

The numerous points made in the discussions support the following consensus:

- Advantages of passive systems/components outweigh the disadvantages.
- There is a need to address data collection and validation of passive systems/components, integration of passive systems in the overall safety systems, regulatory requirements and demonstrability of passive systems.
- The degrees of passivity defined in TECDOC-626 are intended for illustration of the spectrum of possibilities, but do not represent fixed categories.

- Passive components/systems have been used and currently are being used. The operating experience has been satisfactory. In-service inspection was identified to be an important area for passive systems.

In order to discuss the issues of passive systems in more detail, examples of systems/components with a varying degree of passivity were selected. The categorization of TECDOC-626 was used for the selection of the various examples and to facilitate discussion of passive systems with similar characteristics. As highlighted in the paper "Development of IAEA Description of Passive Safety and Subsequent Thoughts", this does not mean that these categorizations are to be used for special applications (e.g., regulatory aspect).

The steel pressure boundary represents a typical example of a system/component of category A; passive systems/components with no moving mechanical parts or working fluids and no signal input or external power source. Probabilistic fracture mechanics (PFM) are an important part of the reliability analysis of the steel pressure boundaries. The high reliability of relatively small population of these components represent a main difficulty for providing exact failure probability by this technique. It was, however, shown that the technique could provide upper limits for failure probability, and identify also the most sensitive parameters. PFM could also provide a useful tool to guide design changes, improve quality control or modify operation where the fracture probability is felt to be too high. It was concluded that PFM is the best approach in spite of all its shortcomings.

The second passive system category (Category B) includes systems with working fluids but no mechanical movement or signal or active power input is needed. Heat transfer for reactor emergency cooling systems based on water natural circulation, or similar systems for containment cooling systems, play an important role in passive heat removal applications. The presence or build-up of non-condensable gases drastically degrades the condensation process and hence inhibits the heat transfer mechanism. The isolation condenser of a passive containment cooling system was taken as an example for this category. It was concluded that the condensable film surrounding the heat exchanger surface is very important for the process. The presence of the non-condensable gases greatly affects the efficiency of the system. It was also concluded that the orientation and geometry of the system is an important factor for its efficiency.

Data on the effect of the non-condensable gases and on the effectiveness of heat removal systems in the presence of non-condensables are system specific and largely depending on the prevailing conditions and geometry. This makes data collection and model validation a much more difficult task.

The check valve was taken as a third example for passive safety (category C). Check valves are widely used in nuclear power plants and in advanced reactor designs. The check valve is proposed to be used for many of the passive cooling systems. Check valves are convenient for system isolation up to a given differential pressure. Below a given differential pressure the isolated system delivers a fluid which is in a stand-by state to perform a given function (e.g. cooling, make up, reactivity control). Check valves could be used at both high or low pressure conditions. At the meeting data on actual experience with check valves used under full reactor pressure, low pressure and at zero differential pressure have been reported. The reliability of this component is a key issue for passive safety systems. The component, from practical experience, is very reliable. There is some room for improvement in the check valve performance at zero differential pressure. The biased-open check valve is designed with these conditions in mind.

The intermediate zone between active and passive, where the safety function is conducted passively once external intelligence initiates the process, forms another category (D). Reactivity control systems relying on natural forces or stored energy for system operation fall under this category. Some initiation principles, however, rely on a natural process or material properties (e.g. temperature induced changes in the mechanical or magnetic properties of the control mechanism resulting in a large negative reactivity insertion). Stored energy (e.g. pre-pressurized tank) is another passive mechanism used for initiation and operation of reactivity control systems. Redundancy, diversity and fail safe operation practiced in the design and operation of the control systems, in these cases, assure the required level of performance. With regard to these systems the main outcome of the discussion was that the need for a signal for process initiation does not by any means degrade the quality of the system. Moreover, systems of this type have shown high reliability in actual operation. These systems are incorporated in the designs with some redundancy and diversity. It is also desirable to have these systems designed to operate in a "fail safe" manner.

The self-acting stabilization of the neutron reaction in gas-cooled reactors was presented. The combination of the negative fuel temperature coefficient of reactivity, the xenon-effect, the low excess reactivity, and the large design margin between the operating temperature and maximum temperature that the fuel can withstand, constitute an inherent safety characteristic with respect to a potentially hazardous reactivity insertion in an appropriate core design. These features contribute to the solution of problems of gas cooled reactor safety. The first three of these are inherent safety characteristics with regard to reactivity incidents. Large fuel temperature design margin constitute an inherent safety characteristic with regard to loss of heat sink. They, however, could not be regarded as systems since no components could be defined for such a system. This, however, does not degrade or disregard the strong and desirable effect of the inherent safety characteristics of this type of reactor.

Specific systems and components were also considered at the meeting.

For the European Pressurized Reactor (EPR) project, numerous passive safety systems have been considered and criteria with regard to design, operation, safety and cost were used for the assessment. It was concluded that for large reactors few passive features could possibly be implemented without substantial effect on the design and cost.

For beyond design basis accidents, there is a general belief that passive safety systems have advantages. For the EPR, a conceptual core catcher design for core melt retention and cooling was presented. The approach is to provide a large spreading area for the melt on a high temperature resistant protection layer. This would then be followed by flooding of the melt with water. Several experimental programmes are underway to verify some of the physical phenomena connected with melt spreading and melt interactions, some of which are being conducted with real corium.

The feasibility assessment of incorporating a Passive Residual Heat Removal System (RHRS) into a large PWR (System 80+) carried out by KAERI, Republic of Korea, was presented. Safety assessments showed some advantages specially for beyond design accidents, but there is an adverse impact to two events. One is the main steam line break and the other is the inadvertent operation of PRHRS. The combination of these two events may lead to over-cooling of the reactor; this with the negative temperature coefficient of reactivity could lead to positive reactivity insertion. PSA evaluation has shown a reduction in the Core

Damage Frequency (CDF) by implementing the PRHRS. The introduction of a large number of heat exchanger tubes results in the expansion of the pressure boundary, yielding additional LOCA paths. From a cost point of view, the implementation of the system would result in an enlargement of the containment to accommodate a PRHR heat exchanger at an elevated level to allow for natural circulation. Additional containment penetrations and larger external water storage tanks would also be needed. Licensing issues would also have to be reconsidered if the PRHR would be safety graded. The over all conclusion of the paper was not in favour of such modification at present. If the overall nuclear environment changes, PRHR would be a strong candidate for decay heat removal (DHR) in the case of beyond design basis events for a large PWR.

The design for an emergency condenser for the new KWU boiling water reactor design SWR 1000 is truly passive and could prove to be very reliable. The system consists of a parallel arrangement of horizontal U-tubes connected to two main heads. The top head is connected to the reactor vessel steam space and the bottom head is connected below the reactor vessel water level. In normal conditions the tubes are filled with cold water and no heat transfer takes place. If the water level in the reactor vessel drops the heat exchanging area is gradually uncovered and the incoming steam condenses on the cold surface and returns to the reactor vessel. Due to the simplicity of the design, the cost is expected to be a fraction of the cost of a comparable active system. For this reactor concept it was also found to be technically feasible and economically possible to adopt several other passive safety systems. Core flooding, containment cooling and other examples are presented in another paper on the passive systems used in this concept.

Safety systems of the CANDU reactor for heat removal and water make-up were presented and the ease of passive safety implementation was highlighted. Due to the concept of the design, passive safety features are possible even with units in the large power range (i.e. 900-1000 MWe). The CANDU design relies on passive and active safety systems to achieve low failure frequency and provide for the maximum diversity.

The AC600 design by China relies totally on passive safety systems to accomplish the main safety functions. The technical feasibility and reliability of the proposed systems were presented. Major research and test programmes to verify the feasibility and reliability of the proposed systems were highlighted.

Analysis of the availability of the AP600 passive core cooling system was presented. A general description of the different sub-systems was highlighted and plant operation in normal and accidental conditions was described. The methodology used for the availability analysis was outlined. System unavailability figures were reported for different options. It was concluded that the main contributor to plant downtime would mostly be due to valve unavailability.

The passive safety systems for decay heat removal for the marine reactor designed by Japan Atomic Energy Research Institute was presented. The idea of a water filled containment vessel is used to counter a LOCA accident. The natural circulation of water in the reactor pressure vessel and the water filled containment are regarded as key factors of the Marine Reactor X (MRX) safety system. Preliminary design showed the effectiveness of the water-filled containment vessel in the event of a LOCA. Additional work to investigate the maintainability of the systems/components in the water filled containment is underway.

The principle of operation of a passive heat removal system with an injector-condenser developed by the All Russian Scientific Research Institute was presented. The experimental facility used for testing verification of the operation of this system was described. Results of experiments demonstrated the simplicity and full passivity of the system. Another feature of this system is the relatively short time between accident initiation and the beginning of heat removal. The system has also been developed for the specific requirements of the VVER-440 NPP.

An out of pile experimental facility PACTEL (parallel channel test loop) was presented. The facility has been designed to simulate the major components and systems of a commercial PWR during postulated LOCA scenarios and transients. Recent modification provided for the possibility to conduct experiments modeling passive core cooling systems. An additional objective of the test facility is to enhance the understanding of the physical phenomena in passive safety systems working with low differential pressure. Experimental and theoretical results showed good agreement for most of the different physical events. The main discrepancy was due to the difficulty in calculation to predict the rapid condensation in the core make up tank. This calls for further experiments and new computational models to be developed. The reference reactor for the facility is a LOVIISA type VVER-440.

An analytical and experimental programme (ALPHA) was initiated in 1990 by the Paul Scherrer Institute. The purpose of the programme is to better understand the long-term decay heat removal and aerosol questions for advanced light water reactors utilizing passive safety techniques to accomplish the main safety functions. The ALPHA programme includes four major items:

- PANDA — Large scale, integral system behaviour test facility.
- LINX — Test facility for the thermal hydraulics of natural convection and mixing in pools and large volumes.
- AIDA — Aerosol transport and deposition in plena and tubes.

The fourth item is the development and qualification of models and validation of relevant codes using data obtained from the PANDA and LINX test facilities. A paper reviewing the above four topics and current status of the experimental facilities was presented.

The scope of the meeting covered all reactor development lines and, a review of decay heat removal systems in liquid metal-cooled reactors was presented and discussed. The paper mainly dealt with the problems of technical feasibility of passive decay heat removal for fast reactors. Classification of safety systems according to: principle of operation, location of systems in the NPP, and the mechanism of heat removal were presented. The paper further focussed on the classification of such systems by the degree of their passivity and highlighted the advantages and disadvantages. Ways to enhance the degree of passivity were described. It was concluded that RVACS and DRACS systems are the preferred passive systems for Liquid Metal Fast Reactors. The former provides for better efficiency and extended applicability with regard to power level; where the later is attractive due to the high operating temperature of the LMFR. Their future usage in advanced fast reactors was also described.

Several papers were made available by some participants but were not presented (These papers are given in Appendix I).

Based on the AGM presentations and discussions several interesting observations were made:

- Active or passive safety systems differ in the manner in which safety systems, components or structures function. In particular, they are distinguished from each other by determining whether there exists any reliance on external forces or signals. Passive safety systems rely on natural forces, properties of materials or internally stored energy.
- Passive safety systems and components have been used in the past in NPPs (e.g. Hydraulic accumulators, check valves, gravity driven control rods for emergency scram). Active safety systems were the safety graded systems that formed the front line of defence. Recent designs, however, have proposed the possibility of reliance on passive systems as the first safety graded systems to react to an emergency situation. This is seen by many vendors as an option for an economical approach to achieving a high level of safety and to prolong the period for operator intervention giving the operator much more time to understand the status of his plant and to take corrective action, if required.
- Passive safety systems rely on natural forces, and hence eliminates active pumps and valves along with their safety graded power supplies. This could provide for cost reduction and simplification.
- The reliability of a given system or component depends largely on its specific design; hence redundancy, diversity and single failure criteria should be decided on a case by case basis.
- Inherent safety features with respect to a negative temperature coefficient of reactivity combined with the large margin between the operating temperature and the temperature which the fuel can withstand without releasing fission products is seen by some designers as a category of passivity specially when connected with a passive safety system for the eventual removal of the after heat. This is the case with the HTGR design. Others look at it as engineered safety features which cannot be defined as a passive system, but is nevertheless a highly reliable and effective technique for meeting a safety requirement.
- For certain situations passive safety systems and components are less susceptible to operator intervention, making them less vulnerable to operator errors. More emphasis, however, should be put on the design and QA & QC of passive systems to ensure their operability when required. Passive systems are seen to be less flexible with regard to accident management.
- A large amount of data on the performance of passive systems and components exists worldwide. Quantification of reliability is still a difficult process for some systems due to the possibly different modes of failure of passive systems from the more familiar active systems. Causes for failure of passive systems also differ. Periodic testing of passive systems/components where possible provides for better reliability and more data in the long run. Some passive safety systems/components in current designs do take into consideration such features. The ageing effect on an active or a passive system may also be different. For active systems replacement of parts could be the solution, passive systems may require a different solution in some circumstances.
- Passive safety systems relying on a different mode of operation and power supply(compared to active systems), provide the maximum diversity when deployed in combination with active safety systems.
- In certain conditions where forceful or rapid action is required and at zero or very low power, active systems may be more suitable to attain certain safety functions.

- In the case of the very remote possibility of severe accidents such as core melt and corium management, inter alia passive safety systems are expected to provide better handling of these conditions
- Some utilities present at the meeting see little difference between deployment of active or passive systems and base their choice of a safety system mainly on the safety level that could be attained, the reliability of the system and last but not least the cost factor, regardless of the degree of passivity of the system. There is, however, a strong tendency from the vendors and R&D organizations side to employ passive safety systems to improve NPP safety in the small and medium size range.

These observations, and the discussions following the presentations, formed the basis for a final discussion on open issues, such as, modeling, experiments and benchmarks, leading to the following general conclusions and recommendations for future activities.

Conclusions

1. The safety approach for advanced nuclear power plants basically remains the same as for existing plants. All safety systems, passive or active, are based on the defense-in-depth concept. No conflict exists in the employment of passive or active systems. Passive systems, in combination with active systems, provide for diversity and do improve the safety level of advanced reactors.
2. Utilization of passive safety systems and the general relaxation of conditions (larger coolant inventories, negative temperature coefficients etc) provide for a longer grace period and relieve operators from immediate action. On the other hand, accident scenarios have been widened to include severe accidents, and passive systems are often used for the mitigation of such accidents.
3. The reliability of passive safety systems should be seen from two main aspects

    - systems/component reliability
    - physical phenomena reliability

    The first calls for well engineered safety components with at least the same level of reliability as active ones. The second aspect is concerned with the way the natural physical phenomena operate in a particular system and the long term effect of the surrounding on the properties of the system components. It calls for the identification and quantification of the uncertainties in the interaction between the phenomena, the immediate environment and the system. The latter should be complimented by the use of PSA for design optimization. Identification of modes/causes of failure and the collection of existing data from actual experience along with results from current experimental investigations would provide information on influences on the functional reliability of the passive systems.
4. Many Member States conduct substantial work on the design, modeling, and development of passive safety systems. This could be substantially enhanced by global coordination of information exchange on the subject.
5. Coordination of activities on the quantification of reliability of passive safety systems and components could be accomplished through:

    - mutual exchange of available PSA on passive systems,
    - discussions on available methods to evaluate uncertainties in the physical correlations,
    - review of different types of passive systems,

- identification of failure modes of passive components (e.g. pressure boundary),
- gathering of relevant data (e.g. experience with passive safety systems),
- validation of codes at available test facilities (e.g. influence of non-condensable gases),
- specific PSA on innovative passive safety systems (e.g. emergency condensers).

OVERVIEW OF THE KEY ISSUES ON PASSIVE SAFETY

(SESSION I)

KEY ISSUES FOR PASSIVE SAFETY

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Abstract

This paper represents a summary of the introductory presentation made at this Advisory Group Meeting on the Technical Feasibility and Reliability of Passive Safety Systems. It was intended as an overview of our views on what are the key issues and what are the technical problems which might dominate any future developments of passive safety systems. It is, therefore, not a "review paper" as such and only record the highlights.

WHAT ARE PASSIVE SAFETY SYSTEMS

- Need for a consistent definition. It is clear that the terminology has been interpreted differently in the past and this has led to confusion and, worse, to a loss of credibility for 'passive' systems. It is strongly recommended that the IAEA definition is widely adopted and used as a means of helping to alleviate this difficulty.

- The need to differentiate between 'systems' and 'features'. Also aligned to the general problem of definition, it is necessary to have a clear differentiation between 'systems' and 'features'. A 'system' is usually a complete set of engineering components and instrumentations and control systems, the ECCS or secondary shut down are examples. Clearly to engineer a totally passive 'system' is much more difficult (if at all possible) than to have passive 'features'. Features in this case are meant to include, for example, natural circulation (when used as a means for heat removal) or stored energy devices where the 'mechanism' for storing the energy is the "feature".

- The role of natural phenomena. Natural phenomena are, of course, at the heart of any device or process. However, in the context of passive safety, natural phenomena are often called upon to act without the need for other input. The obvious example is Gravity which can be relied upon to ensure rods fall into the core or that natural circulation has a driving force. However, from the point of view of quantified safety, natural phenomena have to be shown to be able to operate under all conditions eg under out of normal heat transfer conditions (boiling, two phase flow) or earthquakes. Reliance upon natural phenomena is much more complicated than simply relying upon Gravity to be there.

In any discussion of passive safety, or indeed of natural phenomena, the phrase 'inherent safety' is often used. We believe this is a potentially misleading phrase and should be avoided.

WHY USE PASSIVE SYSTEMS?

In order to answer this question, it is useful first to simply list the attractions, and, of course, potential detractions of the use of such systems.

Attractions

- Simplicity. In general passive systems do not, by their very nature, call upon complex control systems nor upon external power sources which may need to be both redundant and diverse. Because of this they should also be easier to licence once the basic processes are satisfactorily understood.

- Safety. The principal rationale for postulating passive systems is that they offer a solution to improved safety without an unacceptable increase in costs. The argument for a change, however, is extremely complex. Current (modern) reactor designs are clearly considered to be 'safe enough' by the regulatory bodies, the owners and to a large extent by the political powers in most western countries. Furthermore, with the low activity generally in the nuclear industry there is little opportunity presently for new innovative designs. The full potential of passive systems can only be realised once the demand for nuclear power is re-established and as a part of that, there is a demand for such safety systems.

Detractions

- Lack of data on important phenomena. There is a perhaps surprising lack of data on the phenomena of interest in the particular circumstance under which they would be expected to operate. This is especially true since these 'phenomena' will have to be understood to a level appropriate to nuclear safety standards.

- Need to understand performance in a wide range of conditions. This applies clearly to all conditions within the normal operating envelope for the system. However, there is now an increasing need to be able to determine system performance outside this envelope. This applies certainly to accident management procedures, but also to the requirement to understand severe accident behaviour in order to satisfy developments in regulatory requirements.

- Unknown (untested) response from regulators. In either normal (design basis) or out of normal conditions, passive systems will need to be shown to conform to the expectations of regulators. Whilst there may be clear advantages in having simpler systems, it is not clear yet whether regulators will feel able to license them. This may be of particular difficulty in those countries operating prescriptive licensing regimes since the introduction of passive systems will require a change in the regulations.

Overall, the real 'prize' to be won if passive safety features can be incorporated into the design of next generation plant is a combination of the following:

- improved safety margin at lower cost
- a tool to help improve public perception of nuclear power. (But noting that public perception is not a good driver for the design process)
- clear demonstration of the ALARA principle.

The latter of these is important since it may be argued that the passive systems are as low as achievable in an absolute sense. Given that these are the pro's and con's of passive safety systems and that we believe that the ultimate rewards justify continued efforts, what are the basic technical issues which need to be addressed? The following represent only the obvious 'high level' technical issues. The symposium addresses many of these and in much more detail.

- Quantification of reliability. The reliability of systems is absolutely key to their incorporation into any larger engineered systems. This is not just because the failure rate is important in safety calculations, but also because the requirements placed on other parts of the system will be defined by it. For passive systems, there does not exist a large database of relevant experience which can be used as a starting point, such as there is (was) for valves, electric motors etc when this kind of data was first needed to quantify systems reliabilities. It is necessary to establish what methods are available to determine the reliability of such systems. Analogous problems arise in the determination of the reliability of software based systems. Also, the reliability of this type of system needs to be established over a wide range of conditions.

- Fitness for purpose of passive systems. Whilst passive systems may seem attractive, eg for heat removal it is necessary to demonstrate that they can cope with all of the demands put upon them. For example, they may be too slow for safety grade applications. In other circumstances they may require operator intervention to initiate them violating operational rules or requiring non passive means for initiation. It is also possible that they may degrade operational performance so much that they are uneconomic. Examples of the latter could be where decay heat removal systems operate continuously, even during normal operation and the heat loss might be unsustainable.

- Plant ageing. This is one of the most important aspects of the performance of current plant. The lessons are clear in that unexpected problems always occur as the plant ages and continuing programmes are needed to ensure that plant use is optimised. The pressure on the economic aspects of nuclear power means that this will certainly continue into the future as current plant lives are extended as far as is safely prudent and that new designs will need to be able to demonstrate unequivocally that the predicted life will be achieved. In many cases this life is being extended to 60 years or more. For passive systems there is a total lack of data on the performance of the phenomena under such conditions. Examples might include - degradation of stored energy devices, the blockage of heat transfer routes with deposits, environment effects on structural materials for any changes to chemical composition of coolant systems etc. In addition, there is the question of testing of systems which may degrade, but are of themselves 'untestable'. An example of this is a heat removal system which only operates under accident conditions, and the generation of such conditions is not normally attempted, or even allowed. Further issues associated with plant ageing include such things as the effects on passive systems of maintenance or up-grading of other parts of the plant which could have a deleterious effect on the passive systems performance, or even major back fitting of, eg control systems which would need to be examined very closely for any interaction with existing passive systems.

- Technical Feasibility of Innovative Systems. If a 'passive system' involves innovative components or aspects then there is likely to be a need for an extensive demonstration of the technical feasibility. An example of this is the amount of work required to "prove" the hydraulic lock on ASEA's PIUS systems. And, even though there is now an appreciable body of evidence it has never been tested in the regulatory domain. Also, the use of vortex diodes as proposed in the SIR™ design is innovative so far as power reactor systems are concerned.

- In Service Testing. All systems, whether passive or not require some sort of in service testing regime. Passive systems either have to be 'testable', or to have an overwhelmingly powerful case that it is not required. Two examples serve to make the point.

    - passive containment cooling. Heat removal through thick concrete containment structures is poor. Therefore removal of heat requires some kind of heat exchanger system. Some designs which have been proposed are passive and simply rely upon temperature differences for their operation. These systems have to be very large if they are to cope with the heat loads associated with severe accidents. Their capability may be decreased through a number of mechanisms, eg surface degradation, oxidation etc. Full scale testing is not possible so some form of inspection plus component testing would have to be devised.

    - bursting disks. This is the most obvious puzzle when it comes to in service testing. No disk can be 'tested' since if the test is successful it will have, by definition, failed. Normally, this is circumvented by arguing for strong quality assurance during manufacture, coupled with frequent random tests from production examples.

- Maintainability. It has not been proven that passive systems can be designed for ease of maintenance, nor to minimise radiation exposure. There are cost implications for both circumstances. It is not clear that there will be such difficulties with passive systems but as with other practical aspects of their implementation this has yet to be demonstrated.

HYBRID SYSTEMS

Finally, under the heading of basic technical issues, there is the question of mixed or hybrid systems. In many cases passive systems are being proposed as add-ons or alternatives for existing plant. This brings into question the cost savings and even safety margins since 'active' engineered systems will be needed anyway, along with their safety grade back up power, diversity of operation and quality/reliability associated with nuclear plant. Unless the passive system has genuine cost and safety advantages when used alongside active equipment then it is very unlikely to be welcomed by the operators. The obvious advantage of passive systems is when they can replace the need for <u>all</u> active (safety grade) systems in a plant. Such innovative plant are clearly for the future, the question of course is how to leap-frog current designs to bring them into play on the right timescale.

CONCLUSIONS

The authors believe that the attractions of passive systems outweigh the detractions and that they should form the basis of advanced designs of reactors. However, it is not an open and shut case and there is a need for a programme covering the basic issues addressed in this paper to ensure their availability on a timely basis. The principal requirements are:

- completion of the phenomenological database
- demonstration of the operability of passive systems
- integrated, total concept designs maximising the positive contributions of passive systems
- clarification of regulatory requirements

Implementation of such a programme by either vendors and owners, or Governments (acting alone or in concert) will require an act of vision and confidence in the future prospects of nuclear energy.

DEVELOPMENT OF IAEA DESCRIPTION OF PASSIVE SAFETY AND SUBSEQUENT THOUGHTS

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Abstract

The description of passive components and systems published by the IAEA in its TECDOC-626 was developed in the course of a Technical Committee Meeting held in Sweden and two subsequent Consultants Meetings held in Vienna. This description is reviewed and discussed in terms of the philosophies behind it, alternatives considered, problems encountered, and conclusions drawn. Also discussed is an Appendix to the TECDOC, which illustrates the spectrum of possibilities from passive to active by describing four typical categories of passivity.

Subsequent thoughts on passive safety include a discussion of its advantages and disadvantages, concluding with a summary of current views and problems with it.

Introduction

The description of passive components and systems published by the IAEA in its TECDOC-626[1] was developed in the course of a Technical Committee Meeting held in Vasteras, Sweden, and two subsequent Consultants Meetings held in Vienna. During these meetings, many proposals for descriptions of the various terms were offered and discussed, and a series of drafts of the descriptions that ultimately were published as the TECDOC were prepared, repeatedly reviewed, and revised. This process iterated both on the descriptions of the various terms and on the specific terms to be included or excluded from the document. The process as it relates to passive safety is discussed here, as it may shed some light on what passive safety is and is not, and how the term should properly be used. This discussion includes consideration of alternatives that were suggested, issues that were raised and resolved, and conclusions that were drawn. Although the author participated actively in the meetings and discussions, the published description and the discussions that were part of its development represent the various views of the participants and the consensus that was reached, rather than primarily the views of the author.

Participation in this kind of analysis, discussion, and documentation sensitizes one to the issues and problems and has led to the author's subsequent observation and consideration of how passive terminology is being used by others, including particularly outside the nuclear field. Work on advanced plant designs utilizing passive safety systems subsequent to the publication of the TECDOC is also leading to a clearer understanding of the advantages and disadvantages of such systems. Such subsequent thoughts on passive safety, including current views on it, are also presented below.

Vasteras Meeting Proposals

The author did not offer a specific proposal for describing passive components and systems at Vasteras, but rather offered a framework within which a good description should fit, as follows. It should conform with the common-sense, public understanding of the term; it should agree with usage in other technologies perceived by the public as hazardous; it should agree with common, every day experience such as with automobiles, aircraft, and fire protection; it should not be at variance with dictionary definitions although it should include more refinement and specificity than those definitions; and it should be clear and easy to apply, without ambiguity, and with easy determinability as to whether any piece of hardware conforms to the description. Dictionary definitions of passive center around the negative concepts of "not acting but acted upon" and "not active". Active in turn is defined in dictionaries as "in-action, moving, causing or initiating action or change". Previous definitions of passive safety in the nuclear technical literature included the concepts of coming into action in the event of an accident without switching operations or additional energy supplies(2), or alternatively without an external and continual energy input except for the initial activation energy(3). At Vasteras, the author suggested that active engineered safety systems depend for their functioning on humans, external power sources, mechanical or electrical devices, and the like. In contrast, the functioning of passive systems depends on their inherent or self-contained properties and the laws of nature. Further, it was emphasized that neither kind of system is immune to failure(4).

Other Vasteras proposals of particular interest were those of Forsberg(5), Aritomi and Tominaga(6), and Voznesensky and Fyodorov(7). Forsberg offered the attractively simple concept that passive safety engineering avoids the use of moving parts. Aritomi and Tominaga distinguished between active and passive by focusing on whether reliance is placed on external mechanical and/or electrical signals and forces; this phrasing was ultimately accepted by the consultants for the description of a passive component in the TECDOC. While all the other participants appeared to be searching for a single, sharp criterion dividing active from passive, Voznesensky and Fyodorov offered the view that passivity has different degrees or stages, depending on which of a series of criteria are satisfied, with only the higher stages avoiding the use of mechanical, moving parts. Their suggestion led to the discussion contained in Appendix A of the TECDOC, which describes the "Range of Possibilities from Passive to Active".

Description of Passive Component and Passive System

TECDOC-626 provides the following descriptions of a passive component and a passive system. A passive component is "A component which does not need any external input to operate". A passive system is "Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way* to initiate subsequent passive operation". The asterisk on "limited way" refers to Appendix A of the TECDOC, which provides some typical additional criteria that may be imposed on the initiation process:

- Energy must only be obtained from stored sources such as batteries or compressed or elevated fluids, excluding continuously generated power such as normal AC power from continuously rotating or reciprocating machinery;

- Active components are limited to controls, instrumentation and valves, but valves used to initiate safety system operation must be single-action relying on stored energy; and

- manual initiation is excluded.

Issues in Drafting of the TECDOC Descriptions

Early drafts of the TECDOC utilized the no-moving-parts concept in the description of a passive component, but with a footnote which listed exceptions such as rupture disks, safety valves, check valves, and the like, each of which was considered passive by at least some of the consultants. This approach turned out to be unsatisfying, both because there was disagreement as to whether a specific component such as a check valve should be included in the list of exceptions, and because the list of exceptions appeared to be somewhat arbitrary and perhaps incomplete; i.e., no good criterion for permitting exceptions seemed to exist. Some of the discussion of specific components, such as check valves, brought out that the component may have low reliability, and its acceptance as passive was questioned for that reason. Further discussion led to the acceptance of passive vs. active as descriptive only of the principle of operation, without necessarily implying any judgement of reliability. Even so, the concept of no external input eventually was preferred for the description of a passive component.

The principal difficulty with the "external input" concept arises from the interpretation of the word "external", especially how it relates to the boundaries drawn to define a component and a system. If one chooses to define any particular system sufficiently all-inclusive, all component inputs could be said to be internal to the system, and systems normally considered active would fit the description of passive. For this reason, the description of a passive system was formulated in terms of consisting of passive components, rather than directly in terms of external input to the system. This led to the further question of whether it might not be desirable to develop a further term and description for certain systems which appear to the outside observer to have all the properties of being passive, but which on close examination are found to utilize active components. Several drafts of the TECDOC included the description of such automatic systems under the term "self-acting system"; i.e., without explicitly using the word passive. This description was finally deleted from the TECDOC as it was thought that self-acting was not widely used; as discussed in the penultimate paragraph of the Introduction to the TECDOC, it was desired to avoid the coining or promoting of new terms through the TECDOC. Nevertheless, from the point of view of human factors engineering, such self-acting systems exhibit the distinctive feature of passive systems of not requiring human intervention.

Advanced designs which have been termed "passive plants" are under development in the United States and to some extent also in other countries. These designs make much greater use of passive safety systems and therefore represent important lines of further development. They do not, however, use purely passive components exclusively for all safety functions; to attempt such use would be extremely difficult or perhaps impossible and would probably not be desirable in terms of the overall performance of the designs. Some of the safety systems of these designs use external signals in certain limited ways to initiate their subsequently purely passive operation; this is covered in the IAEA description of a Passive System in general terms and is described specifically in Appendix A of the TECDOC under the heading of Category D. The philosophy underlying this, as discussed in the Appendix, is that a spectrum of possibilities exists between passive and active, rather than an absolutely sharp and clear dividing line. Although the Appendix describes four different categories of passive systems, these were considered illustrative only, and the possible existence of additional categories was indicated.

Subsequent Thoughts

After the drafting of the TECDOC, the author became aware of the fact that in automobile safety in the United States, the term "passive" is now formally and legally defined as automatic; i.e., as synonymous with the term "self-acting" as described previously. For example, automobile insurance discounts are offered for Passive Restraints or Passive Belting Systems, which are defined only as automatic and which have numerous electrical and mechanical components that are actuated and powered through means external to the systems. This suggests that an additional Category E could be added to the Appendix of the TECDOC for this kind of system, as being in the intermediate zone between fully passive and fully active.

The advantages and disadvantages of passive safety from the point of view of the designers of passive plants is becoming clearer, as more experience is gained with the design and licensing of plants utilizing passive safety systems. The most important advantages are that passive systems are not vulnerable to external failures, such as failures of power sources, and that they are less subject to human errors of omission. They also can be designed for less need of operator intervention during accidents (longer grace periods) and usually permit greater simplification. A still unresolved licensing question in the United States is whether active systems performing safety functions need to be safety grade, if they are backed up by passive systems on which ultimate reliance for safety is placed. Countervailing disadvantages of passive systems as opposed to active ones include weaker driving forces which result in quantitative uncertainty in flows and perhaps just enough safety action rather than generous margins, more difficulty and time to restore normal operation after their actuation, bulkier equipment within containment, in some cases better protection of the public than of the plant investment, and in some cases inherent limits on the unit size of reactor if the effective functions of passive safety are to be retained.

When passive safety began to be strongly promoted a few years ago, it was also generally offered as a means to enhance public acceptance of nuclear powerplants. The work of Bisconti[8] has shown that the public reaction in the United States to passive safety systems is more negative than positive.

It is thought that rather than associating this term with its technical advantages, the public perceives it as identified with lazy, lethargic, or doing nothing, and probably feels that active protective measures should be undertaken in the event of an accident.

Concluding Remarks

Some passive components and systems have been used in nuclear reactors since the earliest reactors built; there can be no question about the feasibility of most such applications. In recent years, proposals have been made for new, radically different, and much broader applications of passive safety systems. Although questions of feasibility may credibly be raised with respect to some of these proposals, it is thought that on the whole most such applications are technically feasible and that the more important issue is whether such applications are technically and economically justified in comparison to alternatives employing less or even no passivity.

The different categories of Appendix A of the TECDOC were intended to illustrate the concept of the spectrum of possibilities from passive to active; they were not intended to be (and are not) either all-inclusive or to be used for applications such as categorization of specific systems. Such categorization could be misused for promotional purposes or even worse, as regulatory considerations. Passive safety should be viewed as an engineering tool -- one of a number of possible solutions to an engineering problem, not necessarily the only or the best one. Passive safety should not become an engineering or regulatory objective for its own sake. Hence, a fully passive plant (e.g., relying only on the highest or higher categories of passivity of Appendix A) may not even be desirable -- even if achievable, which is very doubtful.

The best use of passive safety systems appears to be for ultimate protection; the first line of defense is usually better served by the systems used for normal operation, which are usually active systems.

ACKNOWLEDGEMENT

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XA9743156

OPERATING EXPERIENCES WITH PASSIVE SYSTEMS AND COMPONENTS IN GERMAN NUCLEAR POWER PLANTS

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Abstract

Operating experience with passive systems and components is limited to the equipment installed in existing NPPs. In German power plants, this experience is available for equipment of the IAEA categories A, C and D. The presentation will focus on typical examples out of these three categories. An overview will be given on the number of reported events and typical failure modes. Selected failures will be discussed in detail.

Regarding piping in PWRs and BWRs about 123 defects in nuclear heat generation and reactor auxiliary systems were reported in the German national event reporting system. The analysis of defects in piping with small diameters shows no differences between PWRs and BWRs. At least 7 ruptures occurred in pipes with small diameters. Most of the defects have been through-wall cracks with subsequent leakage. The behaiviour of pipes with large diameters is different in BWRs and PWRs. In PWRs very few defects were detected in pipes with large diameter. In BWRs pipes with large diameter were affected too. In BWRs two main areas of concern have been reported: defects at WB-35 pipes and cracks in the heat affected zones near welds in austenitic pipes. Through-wall cracks with subsequent leakage did not occur. The overall operating experience with piping in safety systems shows the validity of the design assumptions and low failure probabilities.

The evaluation of operating experience shows that the reliability of check valves and butterfly valves are within the expected range. But, reported events reveal specific problems of these components. eak tightness and mechanical failures of check valves and butterfly valves are major points of interest.

Recently, significant degradation occurred in the diverse thermal actuation mechanism of fire dampers. These defects highlight the importance of regular inservice inspections of so-called "passive" actuations.

The scram system is another area of passive actuation of systems. Only about 30 events have been reported in the German reporting system regarding deficiencies in

control rod drop or possible precursors. Only 2 events occurred where single rods were not inserted. Main contributors to deficiencies in the scram system were leakages and failures in hydraulic actuation system in BWRs as well as rupture of control rod pins in PWRs.

The operating experience indicate high reliability of passive safety systems and components. The events reported in the German event reporting system reveal some possible common cause failure mechanisms for different components. The importance of regular in-service inspections has been highlighted by some events.

1 Introduction

The reliability of passive safety systems is often assessed higher than that of active systems. The evaluation of operating experience can be used to verify this assumption. Operating experience is limited to those systems and components already installed in existing NPPs.

Operating experience of German light water reactors was evaluated regarding reported events with degradation of passive components. These components belong to the IAEA categories A, C and D. The presentation will focus on typical examples of these three categories.

The presentation is based on the German licensee event reporting system. The reporting system is not intended to provide reliability numbers. But, the qualitative evaluation can give hints on the reliability of these passive components. The events reported are focused on safety systems, therefore the valuation is limited on components installed in these systems.

Based on the events reported in the German event reporting system was evalutaed regarding the number of degradations of passive components and the type of the failures. The components evaluated were pipes and valves. In addition, degradations in the fail safe scram systems were investigated.

2 Operating Experiences with Piping in German LWRs

The reported events were evaluated regarding defects in piping /1/. The period of investigation covers the period from 1974 to May 1994. The former East German reactors were not considered.

The investigation was further limited by some missing information. Several reports did not contain the diameter of the pipe or the exact cause. In single event reports the location of the deficiency was not described in detail. In these cases assumptions have been made e.g. whether the deficiency was located in the pipe or at the stud of the vessel.

Table 1 summarizes the data base of the investigation. For PWRs, fifty-six events have been reported. Twenty deficiencies have been in the nuclear heat generation systems, thirty-six events dealt with deficiencies in reactor auxiliary systems. The results of the investigation regarding BWRs are similar. Overall, fifty-nine events have been reported, twenty-nine in the nuclear heat generation systems and thirty events occurred in the reactor auxiliary systems.

With respect to the number of plants (14 PWRs and 7 BWRs) and their start of operation, the number of events reported per plant and year can be calculated. Regarding the above mentioned limitations of the data base, the general behavior of piping in PWRs and BWRs is comparable. The result of more detailed investigations is given in the tables 2-7.

The operating experience of systems regarding piping are not significantly different between PWRs and BWRs. The main causes for cracks respectively ruptures have been fatigue, corrosion and manufacturing deficiencies for PWRs and additionally for BWRs the combination of corrosion and manufacturing deficiencies. Through-wall cracks with leakages contributed most to the relevant failure modes in PWRs. These cracks occurred mainly in piping with small diameters (Tab. 6). In German BWRs through-wall cracks were the main failure mode for piping with small diameter. In piping with large diameters crack indications were detected in austenitic material.

Table 1: Systems Evaluated Regarding Piping Events

	System	Events reported	Events reported per year and plant
PWR	NHGS*	20	0.15
	RAS*	36	0.18
BWR	NHGS	29	0.24
	RAS	30	0.24

* NHGS= Nuclear Heat Generation System, RAS= Reactor Auxiliary Systems - (Events reported until May 1994)

Table 2: Causes of Reported Cracks and Ruptures*

Cause	PWR	BWR
Fatigue	26 %	15 %
Corrosion	16 %	22 %
Erosion	2 %	7 %
Manufacturing	28 %	20 %
Corrosion + Manufacturing	3 %	26 %
Fatigue + Manufacturing	9 %	3 %
Fatigue + Corrosion	2 %	2 %
Cause not reported	14 %	5 %
Events reported	64	59

* Only Nuclear Heat Generation Systems and Reactor Auxiliary Systems - (Events reported until May 1994)

Table 3: Reported Failure Modes*

Failure Mode	PWR	BWR
Rupture	0 %	2 %
Rupture with leakage	5 %	5 %
Crack	6 %	34 %
Crack with leakage	86 %	56 %
Leakage	3 %	3 %
Events reported	64	59

* Only Nuclear Heat Generation Systems and Reactor Auxiliary Systems - (Events reported until May 1994)

Table 4: Affected Piping Material*

Material	PWR	BWR
Austenitic Steel	73 %	63 %
Ferritic Steel	5 %	27 %
Nickel basis alloy	2 %	0 %
Others	6 %	7 %
Not reported	13 %	3 %
Events reported	64	59

* Only Nuclear Heat Generation Systems and Reactor Auxiliary Systems - (Events reported until May 1994)

Table 5: Location of Failure*

Location	PWR	BWR
Basis material	30 %	39 %
Weld	67 %	58 %
Flange/Seal	3 %	3 %
Events reported	64	59

* Only Nuclear Heat Generation Systems and Reactor Auxiliary Systems - (Events reported until May 1994)

Table 6: Diameters of Pipes Affected in PWRs*

Diameter	Crack	Rupture	Others
ND $\leq$ 15	25 %	3 %	-
15 > ND $\geq$ 25	11 %	2 %	-
25 > ND $\geq$ 50	13 %	-	-
50 > ND $\geq$ 100	14 %	-	-
100 < ND $\geq$ 250	3 %	-	3 %
ND > 250	2 %	-	-
Not reported	24 %	-	-
Events reported	59	3	2

* Only Nuclear Heat Generation Systems and Reactor Auxiliary Systems - (Events reported until May 1994)

Table 7: Diameters of Pipes Affected in BWRs*

Diameter	Crack	Rupture	Others
ND $\leq$ 15	14 %	3 %	-
15 > ND $\geq$ 25	14 %	-	-
25 > ND $\geq$ 50	-	-	-
50 > ND $\geq$ 100	5 %	-	-
100 < ND $\geq$ 250	7 %	-	-
ND > 250	14 %	-	-
Not reported	37 %	3 %	3 %
Events reported	53	4	2

* Only Nuclear Heat Generation Systems and Reactor Auxiliary Systems - (Events reported until May 1994)

With respect to the piping material austenitic steel was mostly affected. In BWRs ferritic steel piping contributed significantly. Two generic issues occurred in German BWRs regarding piping: Cracks in WB-35 as well as cracks in titan-stabilized austenitic steel. Weld areas including the heat affected zones contributed about 60 % to the reported German events with pipe failures. The manufacturing deficiencies mentioned above occured at the weld area.

In general, there is no need to reveal the assumptions used in PSA. For German BWRs the goal and scope of in-service inspection have to be revised.

3 Operating Experience with Passive Components in German NPPs

Three types of passive components are discussed in depth: Check valves, butterfly valves and fire dampers. It must be considered that these components are not really "passive" but acting without auxiliary supply systems like electrical or instrument air systems. Generally, the reliability of these passive components are significantly lower than those of structural elements. The analyses presented here are based on reported events in the German event reporting system. These events alone are not sufficient to set up reliability numbers.

Four different types of failures have been considerd for valve deficiencies. The failures to open and the failures to close, respectively, take into account deficiencies in the actuation of the valve, e.g. solenoid failures. Mechanical failures are deficiencies of valve internals, e.g. fastening bolts of flaps. Internal leakages are categorized as "leakage". These four categories are used in figure 1 and figure 3.

3.1 Check Valves

Regarding check valves, two major types of defects have been reported:

- Internal leakages
- Mechanical failures

The results are shown in figure 1. There are remarkable differences between the operating experience of check valves (or butterfly valves) in standby systems and operating systems, respectively. Open check valves in operating systems fail significantly

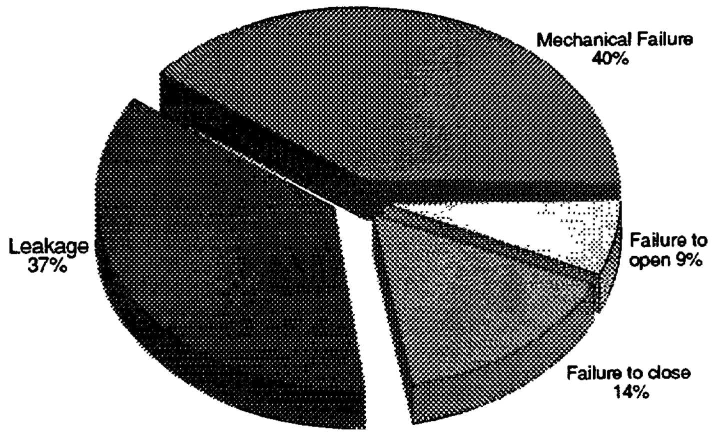

Fig. 1: Type of check valve failures reported
43 check valves affected

more often, especially due to flow induced vibration, than check valves in standby systems. These check valves are closed in normal operation.

The distribution of check valves failures with respect to their systems (see figure 2) shows that check valves of the feedwater system contribute at most. The other valves are mainly installed in stand-by systems.

3.2 Butterfly Valves

The amount of butterfly valve deficiencies reported is comparable to those of check valves. Mechanical failures were the main defect (figure 3).

At least six-teen (of thirty-eight reported) butterfly valves failures were caused by loose flaps in the pipes due to insufficient fastening. Cracks respectively ruptures of bearing bolts were another major failure type. Figure 4 shows the systems which were affected by the butterfly valve failures.

3.3 Fire Dampers

Fire dampers in ventilation ducts are part of the structural fire protection measures. They prevent the spreading of fires through the ventilation system to other fire zones. Closing of fire dampers has to be regarded as a single measure within the fire protection concept. This concept takes also into account non-closing of individual fire dampers without significant degradation of the system function.

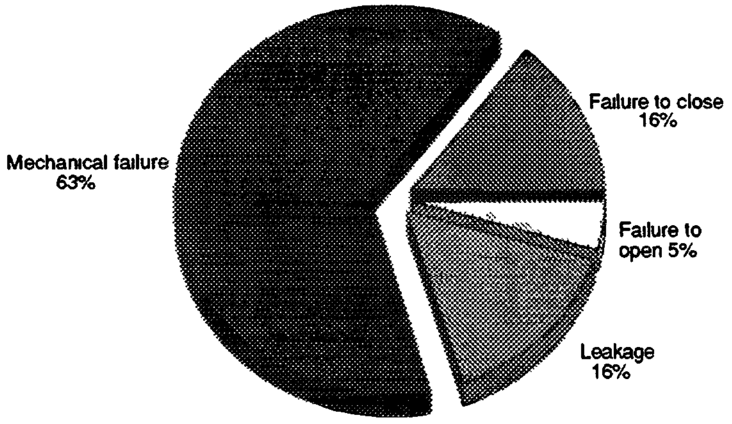

Fig. 2: Systems affected by check valve failures

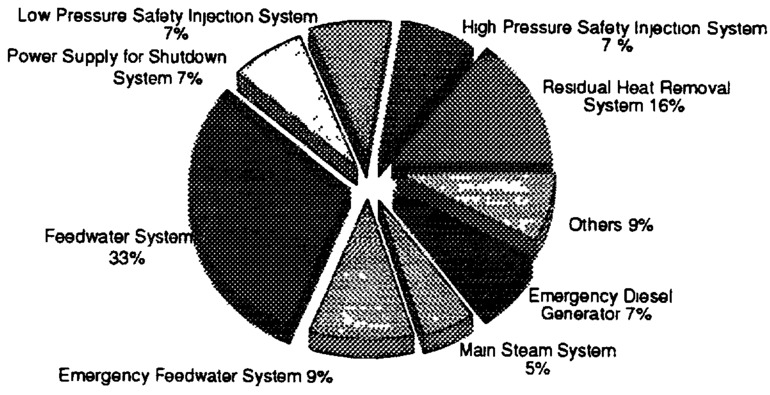

Fig. 3: Type of butterfly valves failures reported
38 butterfly valves affected

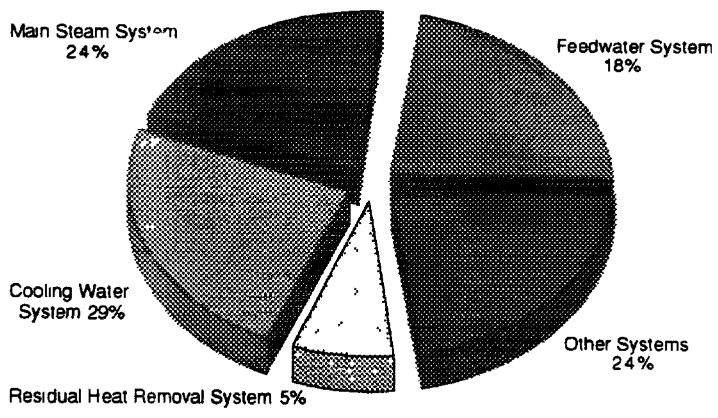

Fig. 4: Systems affected by butterfly valve failures

It should be mentioned that structural fire protection components are not licensed and supervised by the nuclear regulatory authorities, but that authorities and testing facilities are involved which usually supervise the application of fire protection components in industrial plants and conventional buildings. Therefore the licensing process and supervision is not equivalent to the procedure applied on safety systems of nuclear power plants.

The first malfunction of the thermal actuation mechanism was detected during the annual inspection of fire dampers located in the ventilation system for controlled areas in one NPP. During this inspection the function of the electrical remote actuation and the manual actuation were checked, and a visual inspection of the inner and outer parts of the dampers was performed. The visual inspection revealed deficiencies at the thermal actuation mechanism of one fire damper. To prove proper function the fusible element was removed, but the damper disc did not close.

At subsequent tests of the thermal actuation mechanism at 652 fire dampers in this NPP 109 dampers showed the same deficiencies with identical cause. It should be mentioned that all fire dampers affected closed properly at the previous check of the electrical and manual actuation.

Following this incident, GRS recommended inspections of the thermal actuation mechanism of fire dampers at all nuclear facilities in Germany. During these inspections further malfunctions were detected, including different types of fire dampers from several manufacturers. In total, up to May 1994 deficiencies were found in 8 NPPs and in one reprocessing plant. 9 different designs from three manufacturers are affected. Most of the deficiencies identified are systematic failures. However, some of the malfunctions can be regarded as single failures.

Although a variety of designs is affected, some common aspects have been observed. Up to now following root causes are identified:

- The designs of the thermal actuation mechanisms of the affected fire dampers are not robust enough. Minor irregularities in manufacturing and installation, or the operating conditions can result in malfunction.
- There are indications of deficiencies in the quality assurance process. It seems that the fire dampers are not tested as a whole by one institute, but that different parts and functions are examined by different testing facilities. A further problem is

that the combination of electrical remote and thermal actuation is almost exclusively applied in NPPs and therefore no experience is available from conventional facilities, which could have been of use for quality assurance.

- Appropriate in-service inspections were not performed in the past. The thermal actuation mechanism was usually not tested by actually melting the fusible element, but by removing it manually or even only by optical inspection. These inspections have been proved to be not sufficient.

4 Operating Experiences with Shut-Down Systems

Shut-down systems of PWRs and BWRs are differntly designed. In PWRs The operating function and the scram function are performed by operating in German NPPs have two different driving systems:

The control rod drive mechanism in PWRs is a magnetic jack assembly which ensures a step by step motion during operation and the scram function. The drive mechanism in BWRs consists of a motor-driven operational system and a hydraulic scram system. The scram functions of both reactor types is designed according the "fail-safe" principle. With respect to the IAEA category D the operating experience of the fail-safe system was evaluated for this report. Three different failure modes were taken into consideration:

- Scram function was not available or potentially not available (precursors),
- defects in the Actuation mechanism, and
- defects in control function.

The overall experience of the "fail-safe" part of the shut-down systems in German PWRs and BWRs is excellent. Nevertheless, significant differences in the deficiencies detected can be revealed between PWRs and BWRs. These differences are based on the different actuation principles.

For German PWRs only ten events have been reported. The various failure modes (see figure 5) do not show any significant main deficiency. German BWRs experienced twenty-five events (figure 6). The hydraulic actuation system was the main contribution. But, there was no event that affected more than three rods at a time. A remarkable event revealed possible ageing of Teflon seals in valves of the hydraulic actuation system. The regular inspections have to consider this effect.

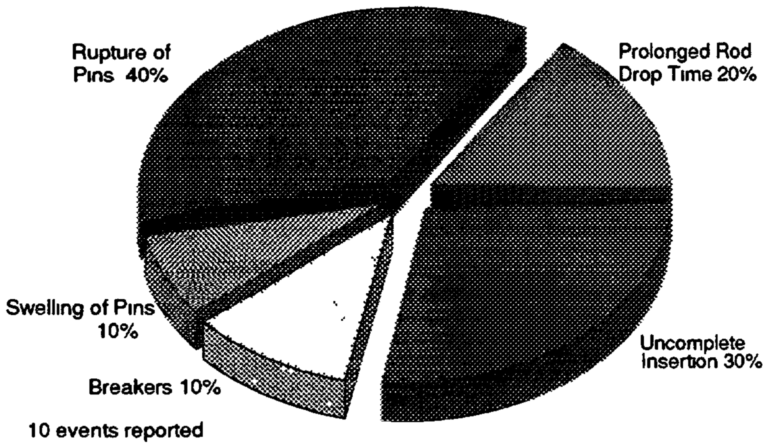

Fig. 5: Reported failures and precursors which could have prevented control rod insertion (PWR)

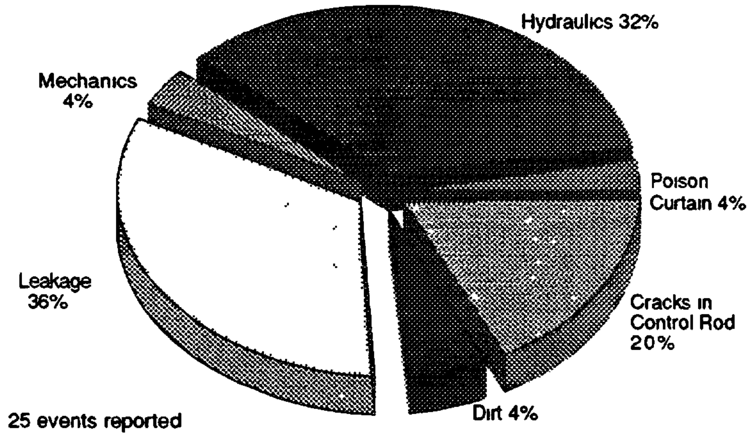

Fig. 6: Reported failures and precursors which could have prevented control rod insertion (BWR)

5 Conclusion

The operating experiences of passive system respectively components was analyzed with respect to the IAEA categories A, C and D. Basis of the analyses were the reported events in the German event reporting system. Reliability numbers cannot be drawn only from events. But, it may be concluded that the reliability numbers are well in the determined range used in risk studies.

The overall experience with passive system and components is excellent. But, There may be one general area of interest which has to be highlightened. Several events reveal degradation of components which were not detected in time (i.e. before a failure occured). This was caused by too long inspection periods or unsufficient inspection extent. Examples of these in-service inspection deficiencies are the non-destructive tests of austenitic pipings as well as the functional tests of butterfly valves and fire dampers.

REFERENCE

/1/ Bienussa, K., Reck, M.
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EXAMPLES OF PASSIVE SAFETY SYSTEMS/COMPONENTS

(SESSION II)

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PROBLEMS AND CHANCES FOR PROBABILISTIC FRACTURE MECHANICS IN THE ANALYSIS OF STEEL PRESSURE BOUNDARY RELIABILITY

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Abstract

It is shown that the difficulty for probabilistic fracture mechanics (PFM) is the general problem of the high reliability of a small population. There is no way around the problem as yet. Therefore what PFM can contribute to the reliability of steel pressure boundaries is demonstrated with the example of a typical reactor pressure vessel and critically discussed. Although no method is distinguishable that could give exact failure probabilities, PFM has several additional chances. Upper limits for failure probability may be obtained together with trends for design and operating conditions. Further, PFM can identify the most sensitive parameters, improved control of which would increase reliability. Thus PFM should play a vital rôle in the analysis of steel pressure boundaries despite all shortcomings.

1. Introduction: The Problem for Probabilistic Fracture Mechanics

In predicting the failure pressure for 134 longitudinally flawed pipes and vessels with four engineering methods the 'best' method was within $\pm10\%$ ($\pm20\%$) in only 40% (60%) of all cases /1/. This poor result can only partly be attributed to the concepts used and their mathematical formulations. The other reason is the large uncertainty introduced by insufficient material characterisation and a lack of control over the many influences. Since most of these uncertainties are of a stochastic nature one could expect probabilistic fracture mechanics (PFM) to resolve the problem. Unfortunately this is only possible in a narrow sense to be explained in the present paper. In terms of failure probability P_f any computation must be poor in principle if P_f is small as is best understood from Fig.1.

In probabilistic structural or fracture mechanics a generalised reliability index β_E may be computed with an accuracy similar to that expected for the deterministic prediction of a safety factor. Transforming β_E to failure probability $P_f = \Phi(-\beta_E)$, where Φ is the standard normal distribution function, is very sensitive and strongly amplifies any error if the reliability is high. Predicting $\beta_E = 5$ within $\pm10\%$ yields $P_f = 1.9 \cdot 10^{-8} ... 3.4 \cdot 10^{-6}$. Thus for close bounds of β_E even the order of magnitude of P_f remains questionable. Moreover, actual failure often results from gross errors in design, production or operation and may not be adequately treated probabilistically. However, problems are small for small reliability because $\beta_E = 2$ within $\pm10\%$ yields $P_f = 1.4 \cdot 10^{-2} ... 3.6 \cdot 10^{-2}$.

Assuming that β_E is in any way more precise than P_f would be a complete misunderstanding. Both are mathematically equivalent since the transformation is one-to-one and thus $\beta_E = -\Phi^{-1}(P_f)$. The deterministic safety factor as a reliability measure gives no quantitative answer and is sometimes even qualitatively wrong. It is not generally order-preserving i.e. a component with a lower deterministic safety factor may be more reliable. This is because the deterministic safety margin does not contain the uncertainty and the different behaviour of different failure modes. A convincing and easy-to-follow textbook example of the limit analysis of a portal frame is given in /2/ on pp.139-141.

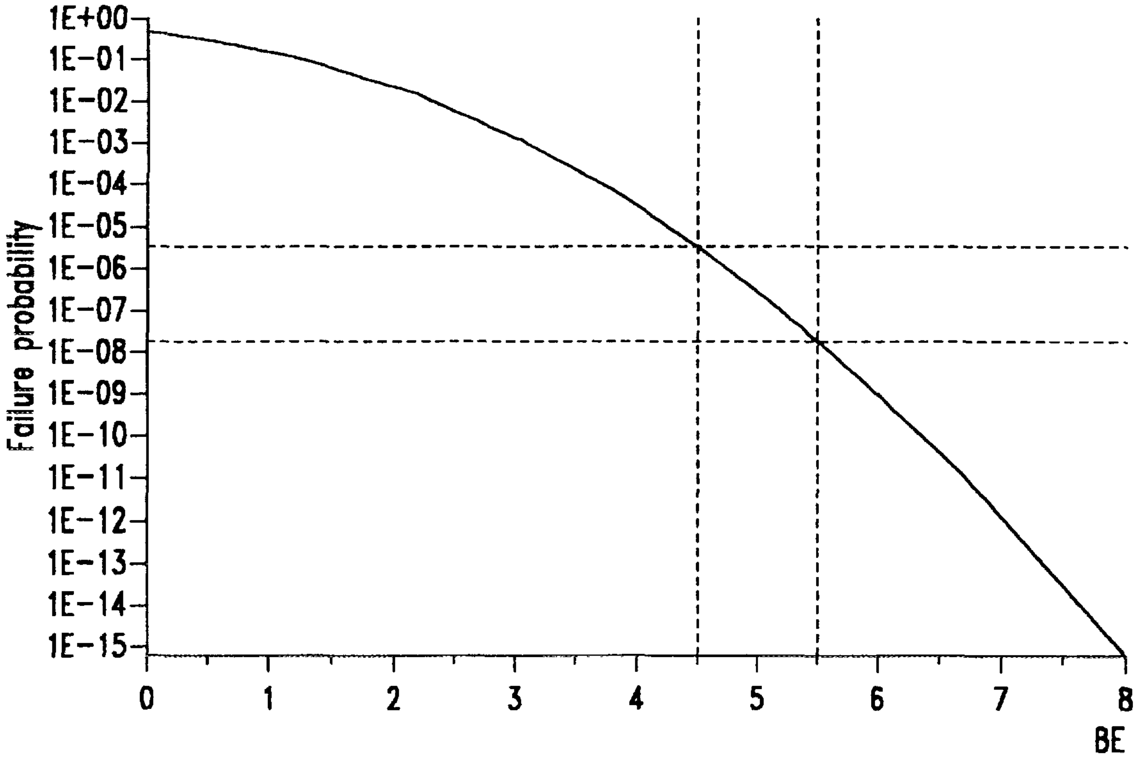

Fig. 1. Failure probability vs generalised reliability index.

Only under the conditions of mass production or an otherwise huge population of sufficient homogeneity may reliability be evaluated by the statistical treatment of direct observation. The direct observation of failure probabilities of non-nuclear pressure vessels and the transfer to nuclear ones poses further questions. But a few conclusions have been drawn /3/. Note as additional comment that the population is necessarily small and of older design and production with little knowledge of properties, operation history, and homogeneity of population. The sensitivity observed in parameter variations in PFM calculations indicates that homogeneity is highly questionable and that failure statistics hardly apply to just similar components. The decrease of P_f with improved design and quality control or its increase with particular service conditions such as stress corrosion cracking or neutron irradiation cannot be assessed by direct observation. Experimental verification of low P_f must be excluded by comparison with the effort of numerical experiments known as Monte Carlo Simulation (MCS). Even these numerical experiments are hardly feasible without limiting their number by some variance reduction, by Importance Sampling (IS) or Stratified Sampling /2/. The basic problem of small P_f of a small population persists. If P_f is assumed to be in the order of P^* and no variance reduction can be employed the number of (numerical or real) experiments may be estimated to be $N=(1-P^*)/(\epsilon^2 P^*)$ where ϵ is the desired relative error /2/. Therefore 10^6 experiments (or simulations) are needed to prove $P^* = 10^{-6}$ within $\pm 100\%$ (i.e. $\epsilon = 1$ just to check the order of magnitude). The formula says that one failure is expected in 10^6 experiments which may or may not occur. It also says that a prediction within $\pm 10\%$ needs 10^8 simulations.

It therefore becomes necessary to investigate the possible contribution of computational probabilistic fracture mechanics (PFM) to the assessment of the reliability of passive components which are not mass products in a positive manner. Despite the provoking section title there is no particular problem for PFM but rather a problem of high reliability of a small population. It is a vicious circle: collecting strength data is equivalent to observing the reliability of the tension specimen. If the population is too small it is too small for both activities.

2. Numerical Method

The failure function (limit state function) g(x) of all variables (basic variables) $\mathbf{x} = (a, a/c, K_{Ic}, R_F, C, \sigma_p, \sigma_s)^T$ used in the fracture mechanics model is defined such that g(x)<0 in case of failure, and g(x)≥0 otherwise. Since all basic variables x are uncertain they may be treated as stochastic variables X with the joint cumulative distribution function $F_X(\mathbf{x})$. Then the failure probability P_f is the probability that g(x)<0, i.e. $P_f = P(g(\mathbf{x}) < 0)$. It is computed with a code which was developed starting from the ZERBERUS code using FORM/SORM /4/,/5/.

If $\mathbf{X} = (X_1, X_2, ...)^T$ is independent but not normally distributed, the reliability problem is transformed into the space of independent standard normally distributed variables $U = (U_1, U_2, ...)^T$. The point $\mathbf{u}^*$ on the failure surface $g(\mathbf{x}) = h(\mathbf{u}) = 0$ closest to the origin is called design point. The transformation is derived from the condition $F_i(x_i) = \Phi(u_i)$ on the marginal distribution $F_i(x_i)$. Assuming the failure surface is smooth, u^* is computed iteratively with the Rackwitz-Fiessler algorithm. The design point $\mathbf{u}^*$ in standard normal space is $\mathbf{u}^* = -\beta\ \boldsymbol{\alpha}$, where the absolute value of the reliability index β is the local minimum distance from $h(\mathbf{u}) = 0$ to the origin, and $\boldsymbol{\alpha}$ is the unit normal to the failure surface in $\mathbf{u}^*$. Then:

- the failure probability $P_f = \Phi(-\beta)$ is obtained using a linear approximation of $h(\mathbf{u}) = 0$ in FORM (a quadratic approximation in SORM).
- the design point $\mathbf{x}^*$ is obtained from $\mathbf{u}^*$ by inverse transformation. Its components are the values assumed by the stochastic variables at the most probable point of failure.
- the sensitivity factors are the components of $\boldsymbol{\alpha}$, and are a measure of the influence produced by the individual stochastic variables on failure probability.
- P_f may be improved by MCS with IS around the design point, its numerical error estimated, and the design point checked.

The latest major structural changes of the code, which are relevant for the present calculations, are the completed implementation of pre-service and in-service inspection (PSI and ISI). The variables of crack depth and shape may become dependent after inspection and a Rosenblatt transformation /2/ is used to transform these variables into standard normal space.

3. Analysis Conditions for a Reactor Pressure Vessel

The Japanese round robin /6/,/7/ may serve as an appropriate starting point since it already gives some flavour of the problems and chances for PFM. However, it becomes necessary to extend its limited stochastic approach at least gradually to some material data. Similarly, its linear elastic fracture mechanics (LEFM) approach must be extended to an elastic plastic analysis. The beltline portion of a typical reactor pressure vessel is analysed as a plate of thickness t = 200mm and width 2b = 12.6m using the design data taken from /8/. All necessary data is also given in /6/,/7/ and will not be repeated here.

3.1 Failure Criteria

The fracture and leak criteria given in /6/,/7/ must be completed and modified from /9/ yielding a 'break' probability which could be more appropriately called failure probability

$$P_f = P_{break} = P(\sigma \geq \sigma_{2D}) \ . \tag{1}$$

It is the probability that the crack opening stress σ exceeds the critical stress σ_{2D} of a semi-elliptical surface crack. Instead of the simple leak criterion yielding

$$P_{leak} = P(a \geq 0.8t \; and \; \sigma < \sigma_{2D} \; and \; b/c > 1) \tag{2}$$

an upper bound is used

$$P_{80} = P(a \geq 0.8t) \geq P_{leak} \; . \tag{3}$$

This is the probability that the crack will penetrate 80% of the wall. In /6/,/7/ eq. (2) was used but only the reduced condition in eq. (3) was given /9/.

With these definitions the RPV shows leak-before-break behaviour (LBB) in a probabilistic sense if $P_{break} < P_{leak}$ i.e. if

$$P_f < P_{leak} \; . \tag{4}$$

Here the less demanding condition

$$P_f < P_{80} \; , \tag{5}$$

is used although it should be noted that other definitions may be more rational but also more critical /9/. For $P_{leak} \approx P_{break}$, no definite conclusion can be drawn since the calculated probabilities are uncertain due to unavoidable deficiencies in both the modelling and data base.

3.2 Material Data

Fatigue crack growth and fracture toughness K_{IC} at 300°C is treated deterministically in /6/,/7/ but gradual decrease due to thermal ageing

$$K_{IC} = \begin{cases} 135 \; MNm^{-3/2} & for \;\; t \leq 14.5 \, years \\ 145.95 - 9.43 \log_{10} t & for \;\; t > 14.5 \, years \end{cases} \tag{6}$$

and neutron irradiation

$$K_{IC} = \begin{cases} 135 \; MNm^{-3/2} & for \;\; F(t) \leq 0.361 \\ 3.29 - 118.71 \, F^{-0.102} & for \;\; F(t) > 0.361 \end{cases} , \tag{7}$$

where F is neutron fluence ($10^{19} ncm^{-2}$), is taken into account. "It should be noted here that the chemical contents of the material tested were slightly modified for an acceleration study of thermal ageing phenomena, and that the above K_{IC} values seem somewhat lower than actual values of operating power plants..." /7/. Therefore the given decrease is unlikely to hold for standard A533 Grade B Class 1 (Germany: 20MnMoNi55, France: 16MND5) material and cannot be transferred to the stochastic treatment of data.

At 300°C the mean values $\pm$ standard deviation for K_{IC} are taken from /8/

$$K_{IC} = 202 \pm 49 MNm^{-3/2} \; , \tag{8}$$

and for the flow stress $\sigma = 0.5(R_{p02} + R_m)$ from /5/

$$\sigma_F = 485 \pm 23 Nmm^{-2} \; . \tag{8}$$

The standard deviation of K_{IC} seems conservative for modern steel production. That of σ_F is perhaps a little optimistic. A Weibull and a normal distribution are used for both in turn.

3.3 Crack Size and Shape

All cracks found in /8/ are converted in a conservative manner to the uniform type of internal semi-elliptical surface crack. The depth a of cracks caused by manufacture was derived from experience with non-nuclear vessels to be exponentially distributed with the density

$$f_{(a)} = \lambda\, e^{-\lambda a}\,, \tag{9}$$

where $\lambda = 0.161\ \text{mm}^{-1}$, /3/,/8/.

The crack length 2c is introduced through the geometric ratio c/a as a shape variable. A lognormal distribution with the density

$$f_{(c/a)} = \frac{1}{c/a\ \sigma\sqrt{2\pi}}\, e^{-\frac{1}{2}\left(\frac{\ln(c/a) - m}{\sigma}\right)^2}, \tag{10}$$

where m = 1.336 and $\sigma = 0.538$, is assumed in /10/.

Crack size and shape are modified if all cracks found by PSI are repaired (introducing no new cracks). The probability of non-detection $P_{ND(a)}$ given in /6/,/7/ is completed from /10/ yielding

$$P_{ND(a,c)} = \varepsilon + (1-\varepsilon)\ erfc(v \ln \frac{A}{A^*})\,, \tag{11}$$

where erfc is the complementary error function and

$$A = a \min\{2c, D_B\}\,, \qquad A^* = a^* D_B\,.$$

Here $D_B = 25.4$mm is the diameter of the ultrasonic beam, $\varepsilon = 0.005$ a residual chance of overlooking deep cracks, and a^* is the crack depth at which $P_{ND} = 0.5$. This equation poses some problems for FORM/SORM since a and c/a are dependent after inspection. Alternatively

$$P_{ND(a)} = \varepsilon + (1-\varepsilon)\ e^{-\mu a}\,, \tag{12}$$

from /8/ is used with $\mu = 0.1134\ \text{mm}^{-1}$ (corresponding to $a^* = 6.11$mm) and the same ε. This widely used function has the disadvantage that the chance to detect a crack becomes independent of its length 2c.

Clearly e.g. crack depth is defined only up to wall thickness t, thus $0 \leq a \leq t$. Therefore all densities are truncated and normalised for finite lower and upper bounds of their arguments, resulting in the densities given in /6/,/7/ for the above equations. Further truncation may become necessary since limit load solutions are given in closer ranges in the literature. No correlation between the above stochastic variables is assumed before inspection.

4. Computations for a Reactor Pressure Vessel

The comparison with /6/,/7/ is also a comparison of methods. FORM/SORM gives more insights by providing design points and sensitivity factors as additional information about the problem. All probabilities refer to one crack, and no residual stress due to welding is considered in the calculations.

Crack growth is the only reason for P_f increasing with time if K_{IC} is constant in an LEFM analysis. Fig. 2 shows the FORM and SORM results together with the seven different MCS (with IS or mostly Stratified Sampling) in /6/,/7/ for years of operation under design conditions. Both FORM and SORM are sufficiently accurate; the SORM solution seems to be closer to the majority of the computations. It should be pointed out that the FORM/SORM solutions may change slightly with starting point and with convergence of the optimisation whereas the MCS results may improve with the number of samples. Similarly FORM/SORM solutions may be improved by IS around the design point. In practice, one is content if P_f is found within a factor of two and P_{leak} within a factor of five /6/. If K_{IC} is varied as a deterministic parameter it is found that P_{leak} is about one order of magnitude less than P_{80} at $K_{IC} = 135 MNm^{-3/2}$ and $P_{leak} \approx P_{80}$ at $K_{IC} = 200 MNm^{-3/2}$ /9/. It has become clear by now that one is interested only in orders of magnitude. Thus FORM results are sufficiently accurate for all computations to follow.

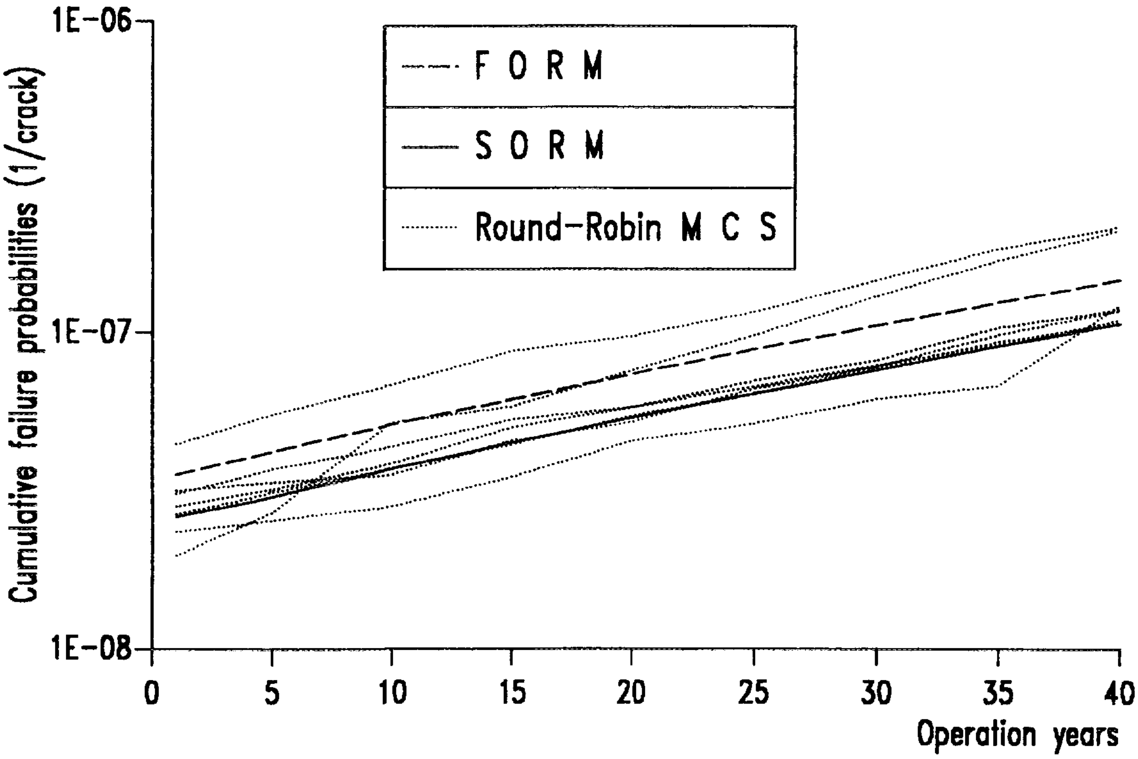

Fig. 2. Time histories for Pf of the RPV for different methods of computation.

The idea in /7/ of using PFM in a criterion for life-extension judgement is as follows:

- First, compute P_f at design life from the design loads, operating conditions and material data. Define this $P_{fdesign}$ as design criterion.
- Second, compute the time from start of operation until the designed $P_{fdesign}$ is reached under the actually measured loads, operation conditions and material data. This gives a new time until end-of-life (EOL).

This idea is used to discuss the effects of reduced neutron fluence F (by measurement or by leakage reducing fuel-charging schemes) and of different intervals for ISI. ISI may change P_f

only if followed by repair. This is possible for the RPV in principle as recently demonstrated by the FENIX project for the twenty-year-old Unit 1 at Oskarshamn, Sweden /11/ (actually the RPV itself was found to be free of cracks). For obvious reasons the frequency of such repairs cannot be high.

Here an 'old design' with F(40years) = $3 \cdot 10^{19} ncm^{-2}$ is compared with an 'evolutionary design' with F(60years) = $1 \cdot 10^{19} ncm^{-2}$ according to the limits set in /12/. The decrease of K_{IC} and increase of P_f is shown in Fig.3 and Fig.4 for the two designs together with the effect of thermal ageing. Note, that the time scale is lost and the two effects cannot be compared if one does not specify a designed lifetime. The above comparison may be used with any kind of ageing passive component. From the flat slope, the lack of data, and the sensitivity of the prediction it should be clear that no sharp time may be given but necessary actions may be indicated. The situation is not very different in deterministic lifetime predictions.

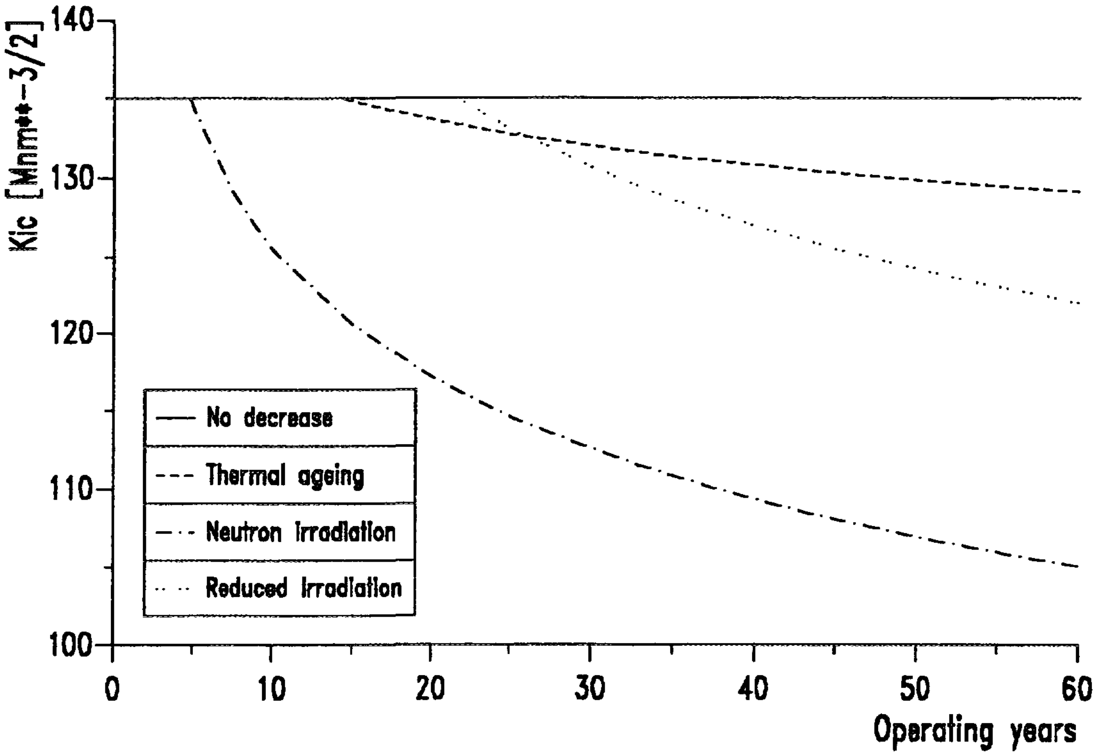

Fig. 3. Decrease of Kic with ageing and different irradiation conditions.

Results of PFM similar to those in Fig.4 may be interpreted differently if one is not primarily interested in lifetime predictions. They actually show the possible loss or gain in reliability for different scenarios. Since both interpretations use only relative changes the absolute values of P_f may be in error. Parameter variations and different stochastic assumptions should be used to discover whether these relative changes are stable.

4.2 Material Data as Additional Stochastic Variables

Assuming a Weibull distribution, but compensating by lifting K_{IC} to the usual values, changes P_f only slightly in an LEFM analysis, see Fig.5. Modelling effective PSI can reduce P_f by one or two orders of magnitude. The optimistic PSI model in eq. (12) may compensate the pessimistic distribution $f_{(a)}$, eq.(9) leading to an overall realistic statistical modelling according to

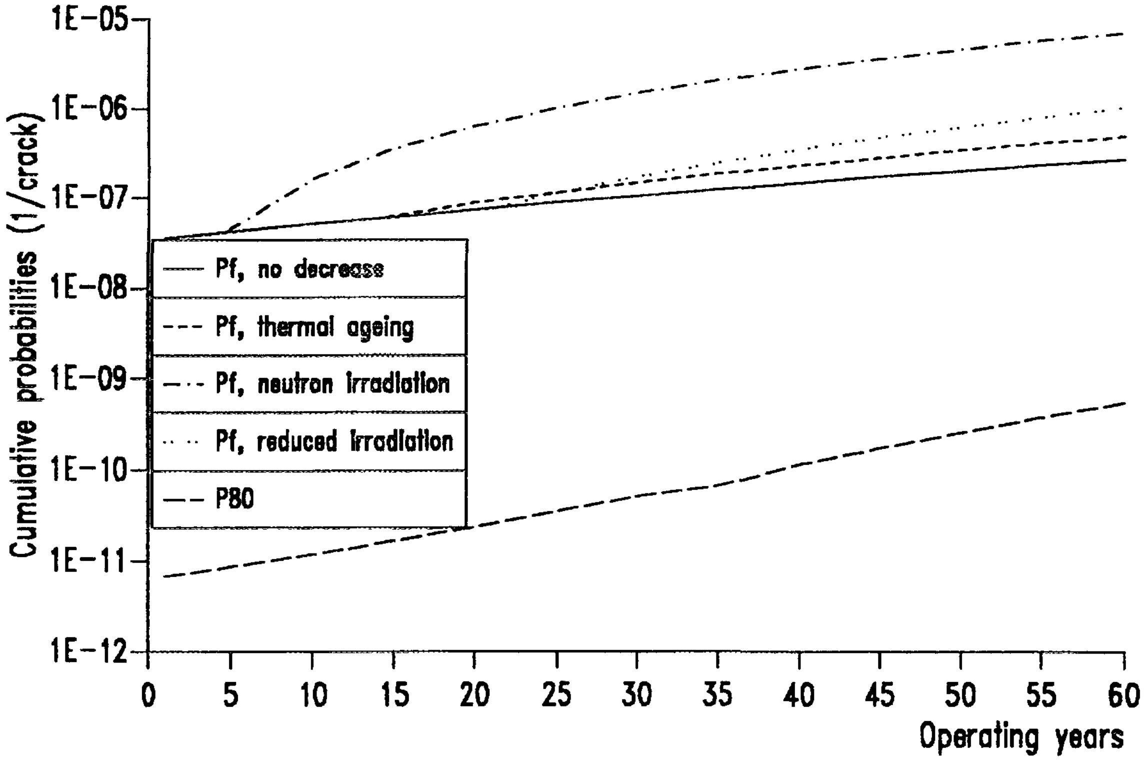

Fig. 4. Increase of Pf with ageing and different irradiation conditions.

/13/. With the μ used there is a 50% chance of finding 6.11mm deep cracks. Obviously the function used for modelling P_{ND} has a great influence since $a^* = 6.35$mm taken from /10/ reduces P_f further by one order of magnitude. /6/,/7/ are more pessimistic about PSI and ISI using $a^* = 31.75$mm for PSI, which was given in /10/ for austenitic steels.

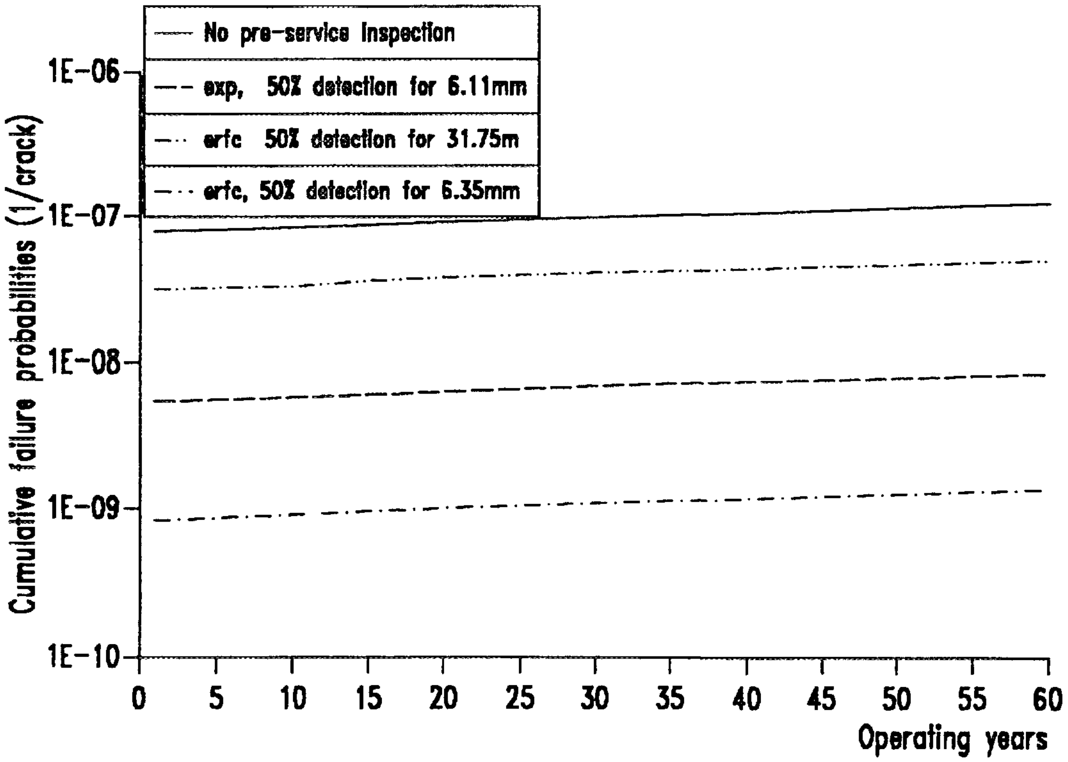

Fig. 5. Time histories for Pf of the RPV for different effectiveness of PSI.

It is important to notice that despite the uncertainty about P_f its relative increase in 60 years is between 53% and 61% for all four curves in Fig.5. This is quite stable but about one order of magnitude lower than the relative increase found in Sec.4.1 with deterministic K_{IC}. In this simplified modelling a has the greatest influence with a sensitivity factor of about -0.9. Tab.1 shows that this rôle is taken over by K_{IC} in the completed modelling (but with positive sign since P_f increases if the design point of K_{IC} decreases. This is opposite for a). Obviously the uncertainty about a stochastic variable of medium sensitivity results in moderate uncertainties about P_f and allows for stable predictions of relative changes i.e. the trends for P_f. Better control of K_{IC} reducing its standard deviation would reduce its sensitivity.

TABLE 1. INFLUENCE OF PSI FOR 40 YEARS OF DESIGN OPERATION
(Design points, sensitivity factors and failure probabilities, see Fig. 5.)

Case	a [mm]		c/a		K_{IC} [$MNm^{-3/2}$]		P_f	
	designp.	sensit.	designp.	sensit.	designp.	sensit.	FORM	SORM
No PSI	35.7	-0.52	2.73	-0.26	66.6	0.81	$1.1 \cdot 10^{-7}$	$8.9 \cdot 10^{-8}$
exp, $a^* = 6.11$mm	20.3	-0.47	2.67	-0.22	49.0	0.86	$7.4 \cdot 10^{-9}$	$3.9 \cdot 10^{-9}$
erfc, $a^* = 31.75$mm	26.0	-0.45	2.61	-0.24	55.4	0.86	$4.5 \cdot 10^{-8}$	-
erfc, $a^* = 6.35$mm	10.4	-0.34	3.06	-0.26	36.0	0.91	$1.2 \cdot 10^{-9}$	-

Suppose now $\varepsilon = 0.0$. Then increasing λ in a parameter variation may be interpreted as either representing the possible influence of PSI (in the sense of eq. (12)) or a shift of initial crack distribution towards shallow cracks (in the sense of eq. (9)) by extracting some deep cracks from the population with improved production /9/. The left line for $\lambda = 0.161\text{mm}^{-1}$ in Fig.6 represents the non-nuclear vessels with no PSI. It is reasonable to assume that nuclear vessels are not worse than that but can be improved by controlled production and PSI up to the right line for $\lambda = 0.161\text{mm}^{-1} + \mu = 0.2744\text{mm}^{-1}$. Thus the optimistic PSI model in eq. (12) may compensate the pessimistic distribution $f_{(a)}$, eq.(9) leading to an overall realistic stochastic modelling according to /13/.

If one uses the R6 method /14/ for interpolation between LEFM and limit analysis (LA) there are two contributions to P_f shown in Fig.6 (at 40 years of operation with design loads) and identified by inspection of the design points in Fig.7. The first failure mode caused by low toughness is not missed by LEFM. The second new one is the plastic collapse of deep half-through cracks. Since both failure modes are weakly correlated P_f is the sum of both contributions /15/. It is impossible to combine two deterministic safety factors in a similar way. The large scatter in K_{IC} data leads to a high sensitivity and slow reduction of P_f with improved vessel (i.e. increasing λ). Changing the distributions of K_{IC} and σ_F from Weibull to normal distribution with the same mean values and standard deviations reduces the low toughness contribution by about three orders of magnitude. The plastic collapse contribution is not affected because of the low sensitivity factor of 0.1 (or less) for σ_F. This is not surprising since a fairly narrow distribution was assumed for σ_F. Conservatively secondary stress was not excluded in LA.

On the basis of the simple criterion in Sec.3.1 no LBB behaviour could be demonstrated probabilistically. The situation becomes 'worse' for improved vessels because PSI followed by repair removes the large cracks thus further reducing P_{80}. The reliability is increased, however,

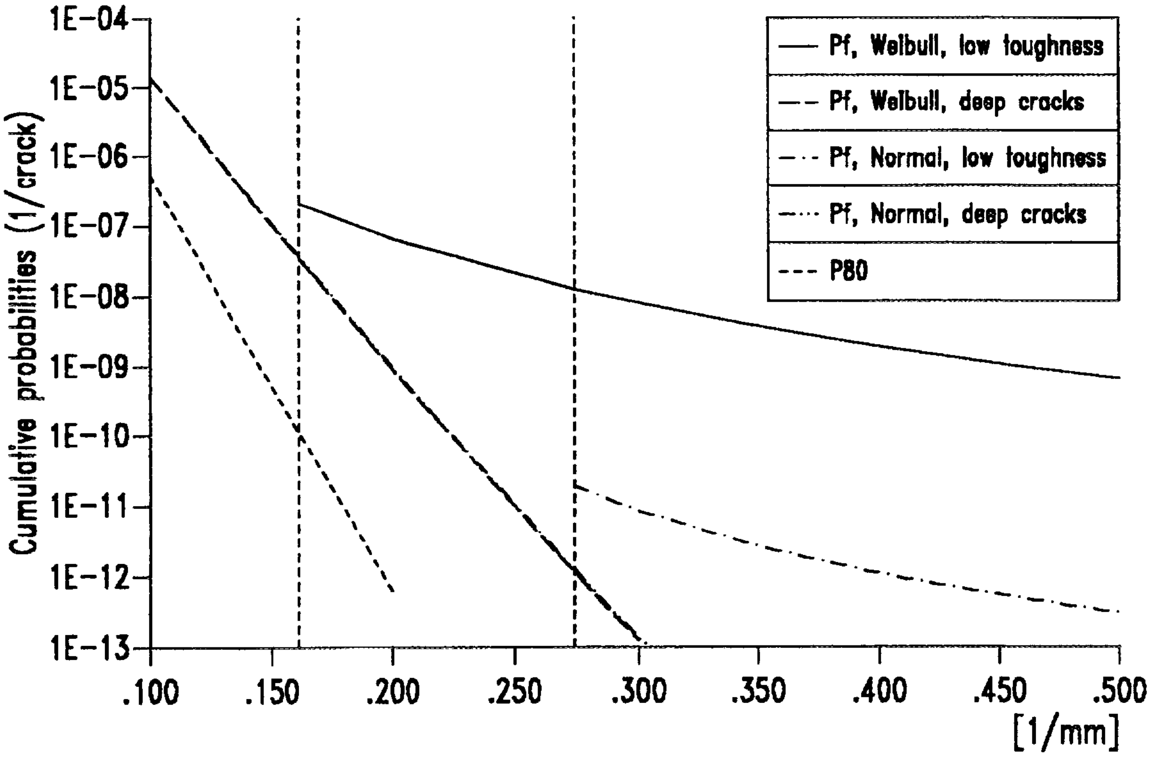

Fig. 6. **Pf vs parameter λ for different distributions** (40 years design operation).

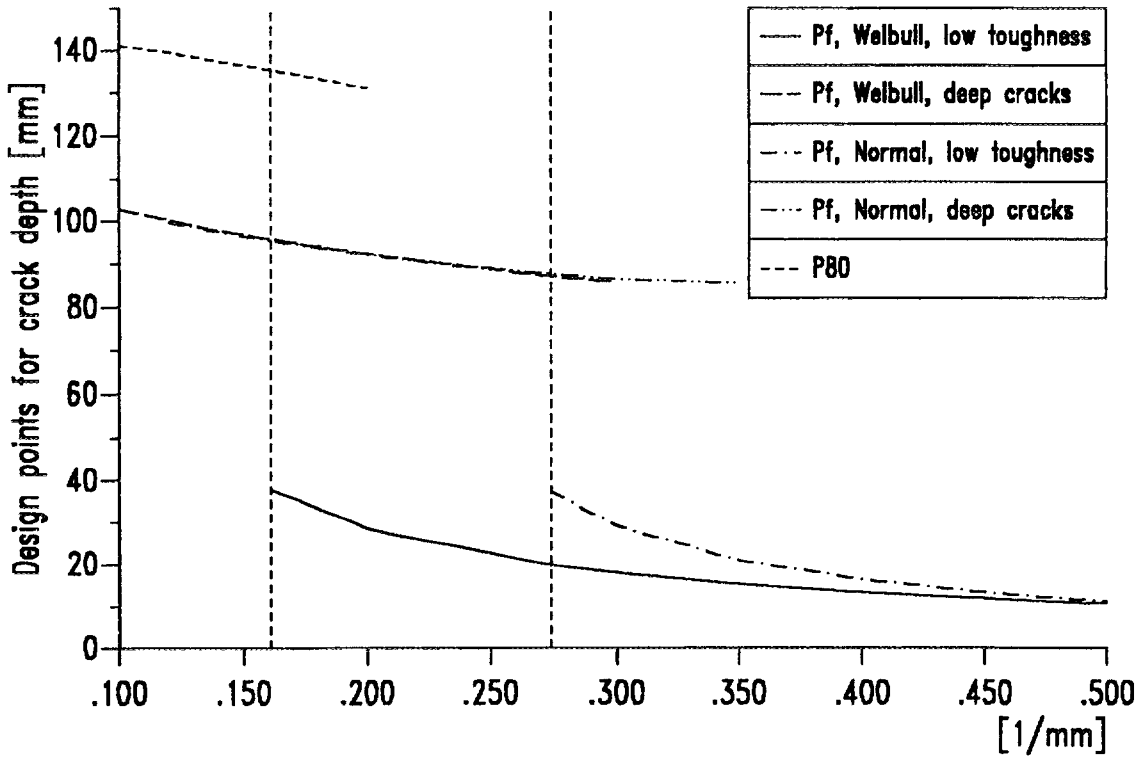

Fig. 7. **Design points for a vs parameter λ for different distributions** (40 years design operation).

for all probabilities are reduced by PSI. The RPV of the Siemens/KWU HTR-Module reactor is thinner at core level than the design point for a in Fig.7 for the P_{80} calculations. Thus the simple criterion makes LBB more probable for this RPV /9/ (in these calculations primary stress was correctly excluded in LA). But the questions should be postponed for refined criteria. Comparing the results in /5/,/15/ for the whole primary circuit pressure boundary of the HTR-Module helps identify the part and mode of most probable failure. It is found that normal operation contributes to risk more than the accident conditions in /15/. Finally, note that all probabilities come closer together as they increase with reduced quality of the RPV.

5. Conclusions: The Chances for Probabilistic Fracture Mechanics

If the population is small failure statistics, experimental and numerical predictions of safety face the same problem of high reliability. There is a particular chance for the numerical approach because it breaks failure down into all possible contributions, for which, stochastic models of the physical process can be made. Thus extrapolation from a small data base is supported by a model of the distribution functions of the stochastic variables. Although not mentioned this was done for most distributions used in the text (e.g. an exponential distribution of crack-depth may be derived from certain possible reasons for the existence of defects in welds /16/). However, it was shown that the choice of distribution for a sensitive variable has a great influence on failure probability. By the very nature of the problems identified in Sec.2 there is no optimal solution. Asking for the value of very low failure probabilities of a small population is asking too much. However, there is a clear sub-optimal solution.

Summarising, one can conclude that with all the different stochastic models and even with the more conservative assumptions about the distributions the reliability of the RPV proves to be high and P_f may even be much lower than this. For the safety of the whole plant it is not so relevant to know exactly how small P_f might be. But it is of prime concern to know an upper bound for its value and its change during operating years. Therefore target values for P_f and for its increase in time may be generated for critical passive components by probabilistic safety analyses (PSA) and it remains the objective of PFM to demonstrate that these single passive components are not worse than the demands under realistic but still conservative assumptions. If P_f is too high for a component PFM may be used to guide its improvement by changing design, improving production and quality control, or by modified operation. For existing plants there are several means to move the material back towards its original conditions (including crack distribution). PFM may demonstrate their effectiveness. Low P_f should be regarded as an operative value and should not be taken as an absolute value for the reliability properties of a component /5/,/16/.

There is a similarity and a fundamental difference between deterministic and probabilistic fracture mechanics. In deterministic analysis a crack size and shape is conservatively postulated and material data and loading are handled in the same pessimistic spirit. Then the answer 'yes' or 'no' is given with no quality of the confidence in this answer. In PFM distributions for the above data are conservatively selected and the answer 'no' is chosen. Then the confidence in this answer is quantified. Of course, both kinds of analyses can be done in a best-estimate sense as well. Besides giving the more complete answer PFM has the chance to monitor all possible realisations of data, conditions, and all possible failure modes together with all their possible interactions at one time. Finally, since P_f (and the generalised reliability index β_E) are the only rational reliability measures, PFM has the chance to help identify components, conditions, locations and modes of the most probable failure together with the possible influences of different conditions and possible actions.

What remains to be done? For the RPV stochastic models for fatigue crack-growth should be used or developed for ageing and neutron irradiation. Existing models for all variables may be checked for possible improvement. The methodology should be applied to other pressurized passive components. Other ageing phenomena may come into play for other passive components such as stress corrosion cracking /17/ or creep crack-growth /18/. Sensitivity factors as computed by FORM/SORM methods may be of some help in identifying the most influential data and in guiding research into the most productive areas. Reducing the scatter in sensitive data will reduce both failure probability and uncertainty of its prediction. The invention of some variance reduction for real experiments, thus reducing the number necessary, would be the major breakthrough. Finally the reader may consult /19/ for "The meaning of probability in probabilistic safety analysis".

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THE RESEARCH ACTIVITIES ON IN-TUBE CONDENSATION IN THE PRESENCE OF NONCONDENSABLES FOR PASSIVE COOLING APPLICATIONS

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Abstract

The introduction of nuclear power becomes an attractive solution to the problem of increasing demand for electricity power capacity in Turkey. Thus, Turkey is willing to follow the technological development trends in advanced reactor systems and to participate in joint research studies. The primary objectives of the passive design features are to simplify the design, which assures the minimized demand on operator, and to improve plant safety. To accomplish these features the operating principles of passive safety systems should be well understood by an experimental validation program. Such a validation program is also important for the assessment of advanced computer codes which are currently used for design and licensing procedures. The condensation mode of heat transfer plays an important role for the passive heat removal applications in the current nuclear power plants (e.g. decay heat removal via steam generators in case of loss of heat removal system) and advanced water-cooled reactor systems. But it is well established that the presence of noncondensable gases can greatly inhibit the condensation process due to the build-up of noncondensable gas concentration at the liquid/gas interface. The isolation condenser of passive containment cooling system of the simplified boiling water reactors is a typical application area of in-tube condensation in the presence of noncondensable. This paper describes the research activities at the Turkish Atomic Energy Authority concerning condensation in the presence of air, as a noncondensable gas.

1. INTRODUCTION

A part of our long term research and development efforts in Turkey is planned to concentrate on passive systems and advanced fuels. The research on passive systems mainly comprises the computer code assessment studies and includes the applications for both old and new generation reactor systems. The research work concerning the application of condensation in the presence of air, as a noncondensable gas, was first undertaken for a Once Through Steam Generator (OTSG) type of PWR, for which, experimental data were available. These experimental data were obtained from the 2x4 test loop of University of Maryland at College Park (UMCP) and addressing a very important safety issue so called the loss of residual heat removal system after reactor shutdown. The experimental data were used for the assessment of RELAP5 computer code and both the effect of Nusselt model, incorporated in the code as the condensation model, and the effect of nodalization were investigated. But the lacking of measurements for the inside of SG has led us to the conclusion that the separate effect test is strongly needed for the investigation of in-tube condensation and the effect of noncondensables on the condensation mode of heat transfer. Thus, an experimental study, which will enable us for the fundamental investigation of condensation in the presence of air, was planned in corporation with the Department of Mechanical Engineering of the Middle East Technical University (METU), Ankara, in the frame of a project between the Turkish Atomic Energy Authority (TAEA) and METU. The project is now in the design stage of the separate effect test facility. The planned experimental investigations will cover a wide range of mixture flow rate, i.e. nearly stagnant steam-air mixture flow and forced convection condensation, with respect to different pressure and air mass fraction values.

2. THEORETICAL BASIS

The Nusselt laminar film condensation model [1] is widely used in system analysis codes, such as RELAP5 and TRAC, and considered as a base model for our investigations. The model needs to be improved since the analytical derivation of the model is valid for situations where the steam is basically stagnant. In cases where the steam flows past cold surface, the model will predict the heat transfer rate that is significantly too low. However, one of the basic assumptions of the analytical derivation of Nusselt, that is, linear temperature distribution exists between wall and vapor conditions, can be considered reasonable in forced convection conditions since the resistance of the condensate film is further diminished because of the thinning, rippling and waviness. But since the influence of turbulence, due to waviness and rippling, is not incorporated in the Nusselt model McAdams suggested to increase the heat transfer coefficient by 20% [2].

The application of Nusselt model, incorporated in advanced computer codes, is another problematic area since the nodalization scheme can highly affect the results. The volume length (L) selected is the input for the average heat transfer coefficient ($h \propto (1/L)^{0.25}$) and it should, preferably, be the length of effective condensation. Problems may be encountered when fine nodalization model is applied since the logic behind the derivation of the model originally assumes the film re-development in each volume where the condensation prevails. But, instead, the film thickness will increase continuously in downward direction, irrespective to the nodal model selected. Thus, the model as coded in computer programs can be expected to overpredict the heat transfer coefficient when the condensation length is discritized by fine mesh model.

As mentioned above, the condensate film provides the only heat transfer resistance in case of pure vapor condensation whereas the main resistance lies in the gas/vapor boundary layer if small amounts of a noncondensable gas are also present. Minkowycz and Sparrow reported a 50% or more reduction of heat transfer rate due to an air mass fraction as low as 0.5% [3]. Many experimental investigations reveal the fact that the diffusional resistance is the dominant factor for the reduction in heat transfer in the presence of a noncondensable gas. From the computational point of view, the presence of noncondensable gas urges us to use fine node models for better axial gas mass fraction distribution which then results in above explained drawback in predicting the local heat transfer coefficient.

Traditionally the reduction factor, which is the function of partial pressure of steam in RELAP5 [4], has been used for reducing the heat transfer coefficient. This approach needs to be improved since degradation of the heat transfer coefficient strongly depends on this factor. Some recent experimental investigations have shown that the diffusive mass transfer resistance in the gas/steam boundary layer, through which the heat transfer consists of the sensible heat transfer and the latent heat given up by the condensing vapor, controls the heat transfer mechanism, that is, the presence of the noncondensable gas is the actual cause of the existence of temperature and concentration gradients. The theoretical approach of the experimental investigation (performed by using a vertical tube with 46.0-mm-i.d.), undertaken at Massachusetts Institute of Technology (MIT), Cambridge, have revealed that the general form of the local Nusselt number is the function of Reynolds (mixture), Schmidt, Jacob nondimensional number groups, and gas mass fraction [5]. The comparable work available on this subject is that of Vierow and Schrock at the University of California Berkeley (UCB), and the correlation obtained for the local heat transfer coefficient is the function of the heat transfer coefficient for pure steam condensation based on Nusselt model, condensate film Reynolds number and air mass fraction. The experiment, performed at the UCB, was made using a 22.0-mm-i.d. vertical tube, natural circulation air-steam system [5].

3. RELAP5 SIMULATIONS

The RELAP5 simulations comprise the major part of the work undertaken for the investigation of condensation in the presence of noncondensable. These simulations are aimed to study the capability of the code to capture the phenomena observed in the experiments. For this purpose, our investigations are based on two experimental data sets, that are, the data from UMCP integral test facility and MIT separate effect test facility. Apart from the simulations of test facilities, a parametric study for the Inherently Safe Boiling Water Reactor (ISBWR) is also carried out.

3.1 THE SIMULATION OF THE UMCP INTEGRAL TEST FACILITY

The operational characteristics of the test facility during the experiment, performed for the simulation of loss of residual heat removal system after reactor shutdown, may be considered to be different than those of the passive safety systems of new generation reactor types. This simulation leads us to understand the major role of the condensation phenomenon for heat removal performance and the effect of condensation on primary loop parameters. The simulation capability of RELAP5 is also assessed against the experimental data.

The integral test facility is installed at, and operated by, the University of Maryland, and is a 1/500 scaled model of a Babcock and Wilcox PWR with two loops [6]. The heat addition into the loop is accomplished by means of 15 heater rods of 2.54 cm diameter and 0.6096 m active length. The two steam generators are of Once-Through Steam Generator (OTSG) type and made of 28 tubes. The tubes are 3.905 m long and have an inside diameter of 29.97 mm and an outside diameter of 31.75 mm.

The experiment is initialized at cold conditions, i.e. the system is under atmospheric condition, temperature is 30 °C, and primary loop is drained down to pressurizer surge line connection. The hot steady-state condition is reached by the establishment of Boiler-Condenser Mode (BCM) and general characteristics of this condition are: system pressure is 440 kPa, upper parts of vessel and hot-leg are at saturation temperature (~145 °C), and SG primary level is 75%. The thermal power during BCM is 34.9 kW and heat removal via SG is mainly by means of condensation. The UMCP test facility is simulated by RELAP5 [7] and the role of condensation and the inhibiting effect of air on condensation process, as predicted by RELAP5, is given in Fig. 1 and the primary system pressure at the BCM predicted by RELAP5 code is as close as 5.5% compared to the experimental data (Fig. 2). It is to be noted that air is accumulated above the water level in the SG and the major part of heat transfer by condensation takes place in the uppermost volume.

3.2 THE SIMULATION OF THE MIT TEST FACILITY

The experimental apparatus [5] consists of an open cooling water circuit and an open noncondensable gas/steam loop. The main components of the gas/steam part of the facility are the boiler vessel (4.5 m height, 0.45 m inside diameter) and cooled vertical test section, that is, the condenser tube. The tube is 2.54 m long (effective) and has an outside diameter of 50.8 mm and an inside diameter of 46.0 mm. The condenser tube is surrounded by a jacket pipe (62.7 mm inside diameter).

Experiments are performed for air-steam mixture inlet temperatures of 100, 120, and 140 °C. At each inlet temperature setting, the steam flow rate is varied by using different boiler power levels (6, 13, and 20 kW). The inlet air mass fraction is varied from 10 to 35% for each

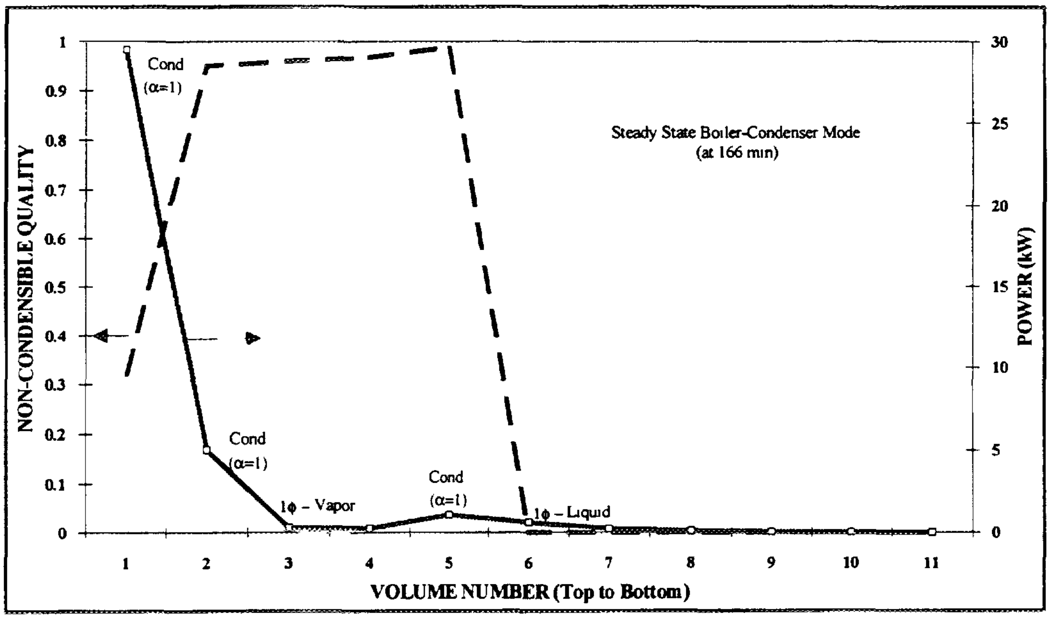

Figure 1. Heat Transfer Characteristics of UMCP Steam Generator at the BCM

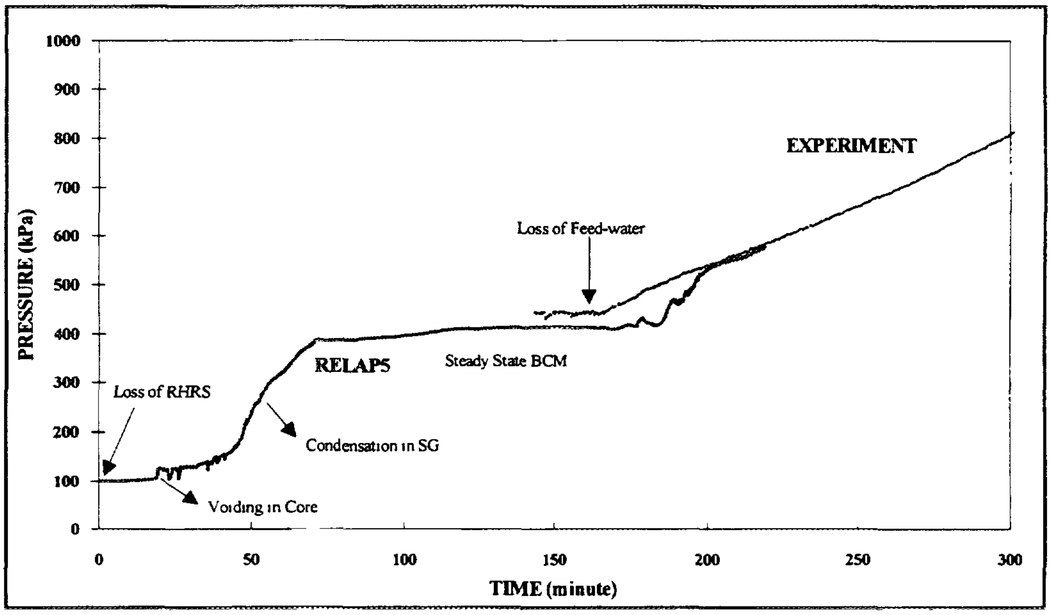

Figure 2. System Pressure of UMCP Test Facility during Loss of Residual Heat Removal System and Loss of Feed Water System Accidents

inlet temperature and power level setting. A similar test matrix is formed for helium-steam mixture, that is, for different mixture inlet temperatures (100, 120, and 140 °C) the steam flow rate is varied by using 6 and 13 kW power settings. The inlet helium mass fraction is varied from 2 to 10% [5].

The main objective of the experimental investigations, performed at MIT, is to measure local heat transfer coefficients for steam condensing in the presence of air or helium inside a tube. Moreover, this study aims to represent the operating characteristics of the Isolation Condenser (IC) which is the main component of the Passive Containment Cooling System (PCCS) of Simplified Boiling Water Reactor (SBWR) design.

A RELAP5 model of MIT test facility is prepared (Fig 3) and RELAP5 results, obtained for different cases in the test matrix, are compared to the experimental data. Since there are few forced convection in-tube condensation studies in open literature and most of them can not represent the operating characteristics of the IC, the MIT data could enable us to extend our simulation capabilities to the passive safety systems of advanced reactor designs.

Two typical RELAP5 results are compared with the experimental data in Figs. 4 and 5. The parameters selected, for the comparison, are the heat transfer coefficient and the air mass fraction with respect to the channel length. The case is characterized by the operating conditions, namely, the inlet mixture temperature is 120 °C and the inlet air mass fraction is 0.08. For fixed inlet conditions, the heat transfer coefficient decreases by the accumulation of air. The RELAP5 prediction for air mass fraction follows the trend of experiment very closely (especially at the entrance region) while the prediction for heat transfer coefficient yields an overestimation compared to the experimental data. But it is interesting to note that the heat transfer coefficient calculated for the uppermost volume is very close to the experimental data. Another important parameter to be considered is the effective condensation length which is the function of axial air mass distribution. The effective condensation length, for the case presented in Figs. 4 and 5, is overpredicted by the code.

3.3 THE SIMULATION OF THE INHERENTLY SAFE BOILING WATER REACTOR CONCEPT

The Inherently Safe Boiling Water Reactor (ISBWR) concept is a 340 MWe (1000 MWt), natural circulation, indirect cycle small boiling water reactor [8,9]. The design features a multi-cavity Prestressed Concrete Reactor Vessel (PCRV) which contains all primary loop components (i.e. reactor, steam separator, subcooler/preheater, condenser/evaporator). Fig. 6 shows a section view of the ISBWR. Under normal operation, the naturally circulated primary fluid rises vertically in chimney after exiting from the core and enters the steam separator. The separated steam in the steam separator rises through in chimney cavity and then goes through the steam by-pass orifice to the upper section of downcomer cavity where the Condenser/Evaporator (C/E) is located. The saturated water goes through the water by-pass orifice to the lower section of the downcomer cavity where the Subcooler/Preheater (S/P) is located. The secondary loop coolant flows in the S/P tubes and is heated up, and then goes to the C/E tubes and is evaporated. The ISBWR is inherently safe against any primary breaks. However, any kind of secondary fault (e.g. feed water pump trip, rupture of one of the S/P or C/E tubes) may lead to loss of heat sink. In that case, the steam driven jet injector uses the decay heat steam to pump water from suppression pool to cool the reactor core.

The primary loop operation characteristics are function of the secondary loop pressure, inlet temperature, and mass flow rate. Condensation without noncondensable occurs on the main steam generator tubes which is not a part of safety system of ISBWR. In the ISBWR, the condensation on the C/E tube outside surface plays main role for the system steady-state operation parameters.

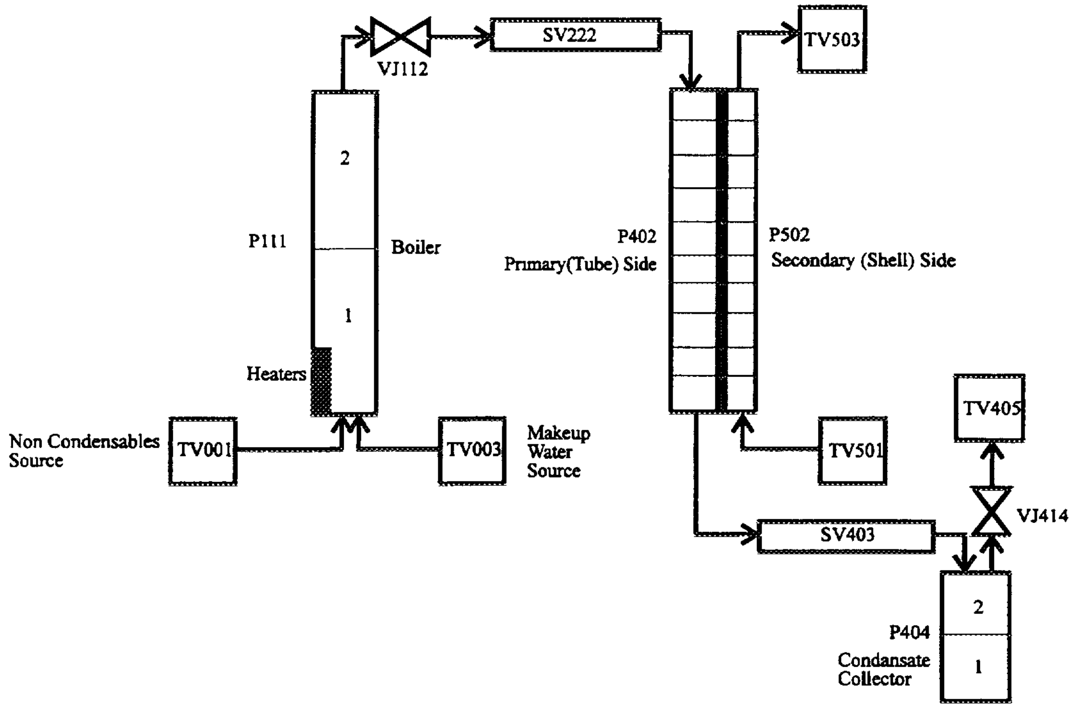

Figure 3. RELAP5/MOD3 Nodalization for MIT In-tube Condensation Experiment Setup

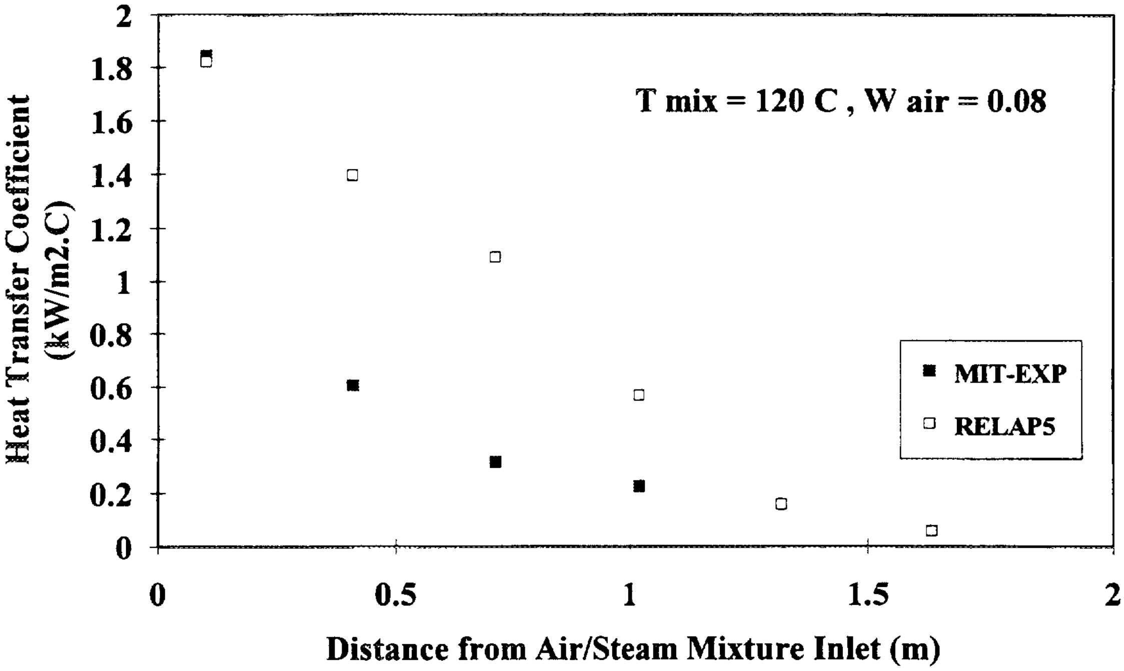

Figure 4. Axial Variation of Heat Transfer Coefficient

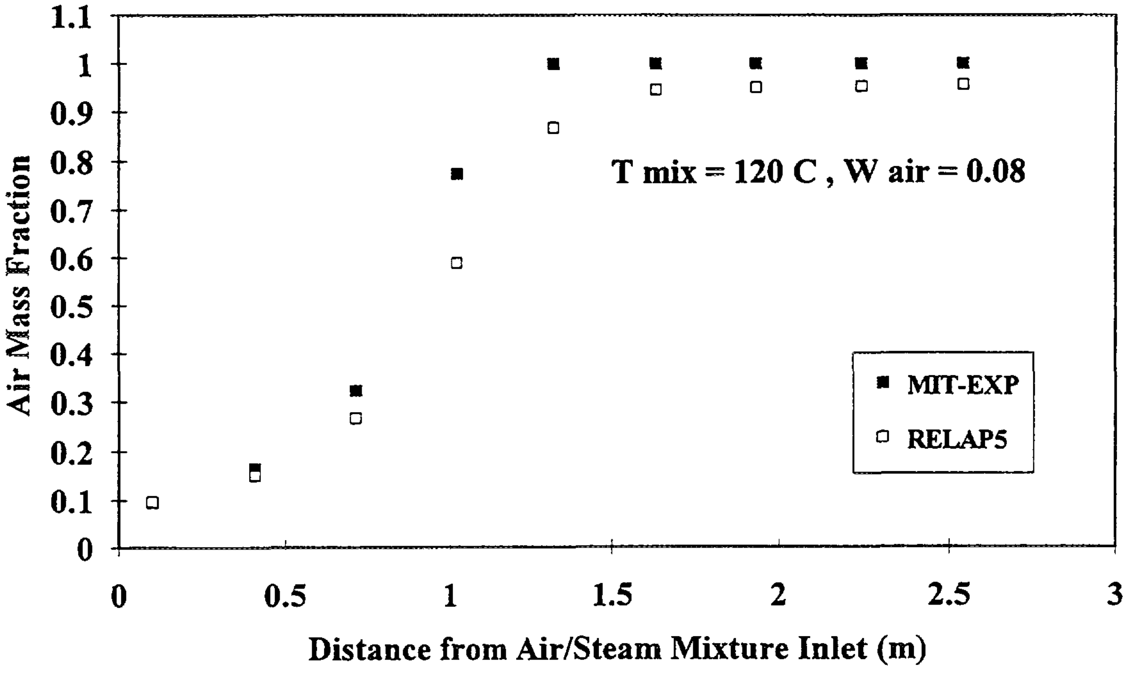

Figure 5. Axial Variation of Air Mass Fraction

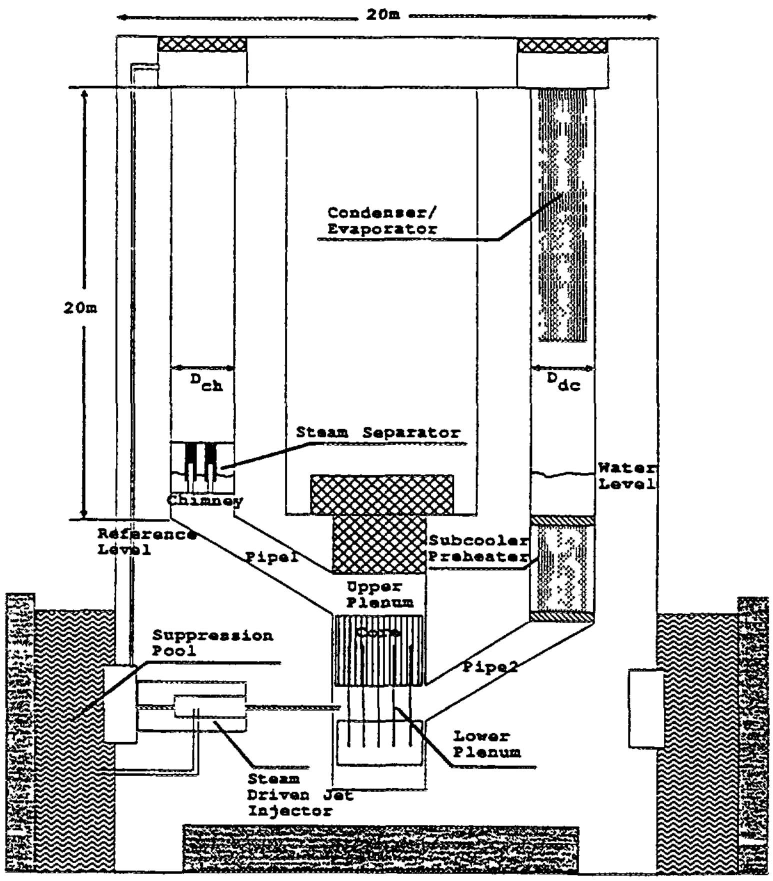

Figure 6. Section View of the Inherently Safe Boiling Water Reactor (ISBWR)

4. THE PROJECT RELATED TO THE EXPERIMENTAL INVESTIGATION OF IN-TUBE CONDENSATION IN THE PRESENCE OF AIR

The experimental data available for in-tube condensation in the presence of air are very limited and most of the experimental correlations developed are applicable for the particular system for which they were developed. It is also needed to assess the experimental data available in the open literature with some independent experimental measurements provided that the experimental conditions are similar.

An experimental program was planned in cooperation with the Department of Mechanical Engineering of the Middle East Technical University (METU), Ankara, for the investigation of in-tube condensation in the presence of air in the frame of a project between TAEA and METU. The project is now in the design stage of the facility.

The project aims to investigate the condensation phenomenon for two different cases. First, the case corresponding to the operating conditions of SG of UMCP test facility, i.e. the lowest part of the SG tube is full of water and some amount of air is accumulated just above the water level. In this case, only the vapor flows -with relatively low Reynolds number- from top of the SG tube and the water level is kept constant throughout the experiment. The mass of air accumulated above the water level can affect the effective condensation length. The second case planned is for forced convection condensation. The test matrix proposed for this case will comprise different system pressures (1-5 atm) with the variation of inlet air mass fraction for each pressure setting. The effect of mixture Reynolds number will also be considered. There are two condenser tubes, planned to be used for the experiments, with different inside/outside diameters (25.0/32.0 mm, 47.0/51.0 mm) and same total length (~2.0 m).

5. CONCLUSIONS

The annular film condensation of vapor inside vertical tubes is extremely important for applications concerning chemical and power industries. By the investigations regarding condensation in unconfined spaces it has been well established that the existence of noncondensable gases can greatly inhibit the condensation process, and in turn, the heat transfer performance of heat exchangers. Among such investigations there are also theoretical and experimental studies for plane surfaces, with different orientations, to simulate the cooling conditions of containment wall. But the experimental investigations addressing the research on in-tube condensation in the presence of noncondensable(s) for passive cooling applications of new generation reactors are very limited in the open literature.

The experimental and computational research activity for in-tube condensation in the presence of noncondensable(s) is planned and launched by TAEA to make contributions in the area of passive cooling applications. The mechanism beyond the effect of noncondensable gases for the degradation of heat transfer rates by the condensation process is rather complicated. Thus, the research program is also supported by the theoretical investigations.

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PLANT EXPERIENCE WITH CHECK VALVES IN PASSIVE SYSTEMS

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Abstract

In the design of the advanced nuclear reactors there is a tendency to introduce more passive safety systems. The 25 year old design of the GKN nuclear reactor is different from the present BWR reactors because of some special features, such as the Natural Circulation- and the Passive Isolation Condenser system. When reviewing the design, one can conclude that the plant has 25 years of experience with check valves in passive systems and as passive components in systems.

The result of this experience has been modeled in a plant-specific "living PSA" for the plant. A data-analysis has been performed on components which are related to the safety systems in the plant. As part of this study also the check valves have been taken in consideration. At GKN, the check valves have shown to be reliable components in the systems and no catastrophic failures have been experienced during the 25 years of operation.

Especially the Isolation Condenser with it's operation experience can contribute substantially to the insight of check valves in stand-by position at reactor pressure and operating by gravity under different pressure conditions.

With the introduction of several passive systems in the SBWR-600 design, such as the Isolation Condensers -, Gravity Driven Cooling - and Suppression Pool Cooling System, the issue of reliability of check valves in these systems is actual. Some critical aspects for study in connection with check valves are:

- what is the reliability of a check valve in a system at reactor pressure, to open on demand.
- what is the reliability of a check valve in a system at low pressure (gravity), to open on demand.
- what is the reliability of a check valve to open/close when the stand-by check valve is at zero differential pressure.

In this paper the plant experience with check valves in a few essential safety systems will be described and a brief introduction will be made about the application of check valves in the design of the new generation reactors.

1. INTRODUCTION

The information for this presentation is retrieved from the GKN Level 1 PSA model. The plant experience presented, comes from those systems and components which are modeled in the PSA.

The systems which will be reported are those where check valves have a passive function or where the whole system is a passive safety system.

2. HISTORY

To obtain an objective picture about check valves which are operating at low differential pressure it is worth making a historical review.

Check valves developed to perform quick action and also operate under minimum differential pressure are not new; the swing check valve with outside lever and weight is an old application.

The check valves used at GKN are mainly of the "swing-type" as shown in figure 1. Figure 2 shows some of the check valves as installed in the plant.

PSA data-analyses show only one (1) registered failure of the check valve T40-04022 (see figure 4 on page 74) over a time-window of 19 years.

Failure Mode: backflow (**leakage**) of this check valve.

No action was taken because the leakage had stopped.

CAST STAINLESS STEEL CHECKVALVE
with vertical hanging disk and membrane-seal

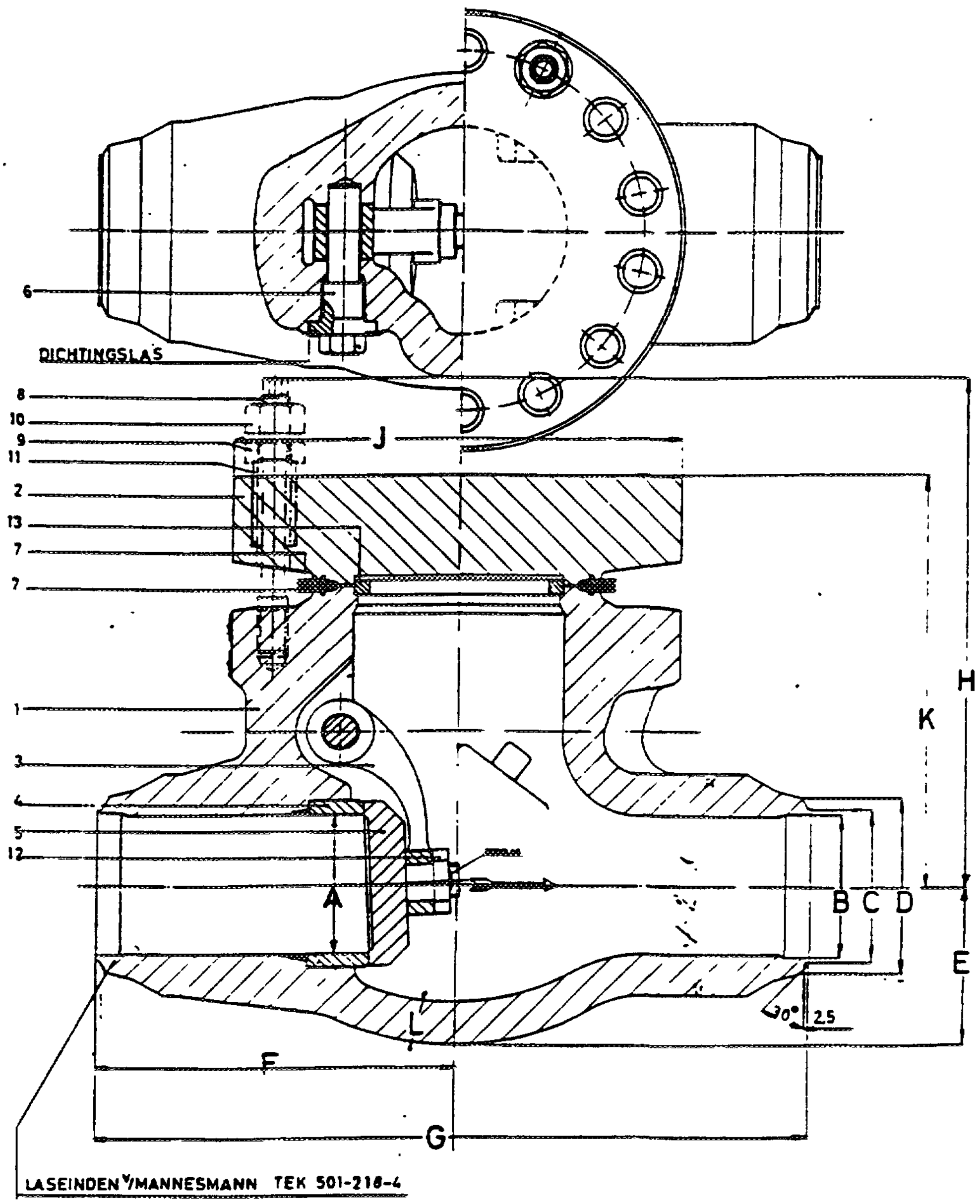
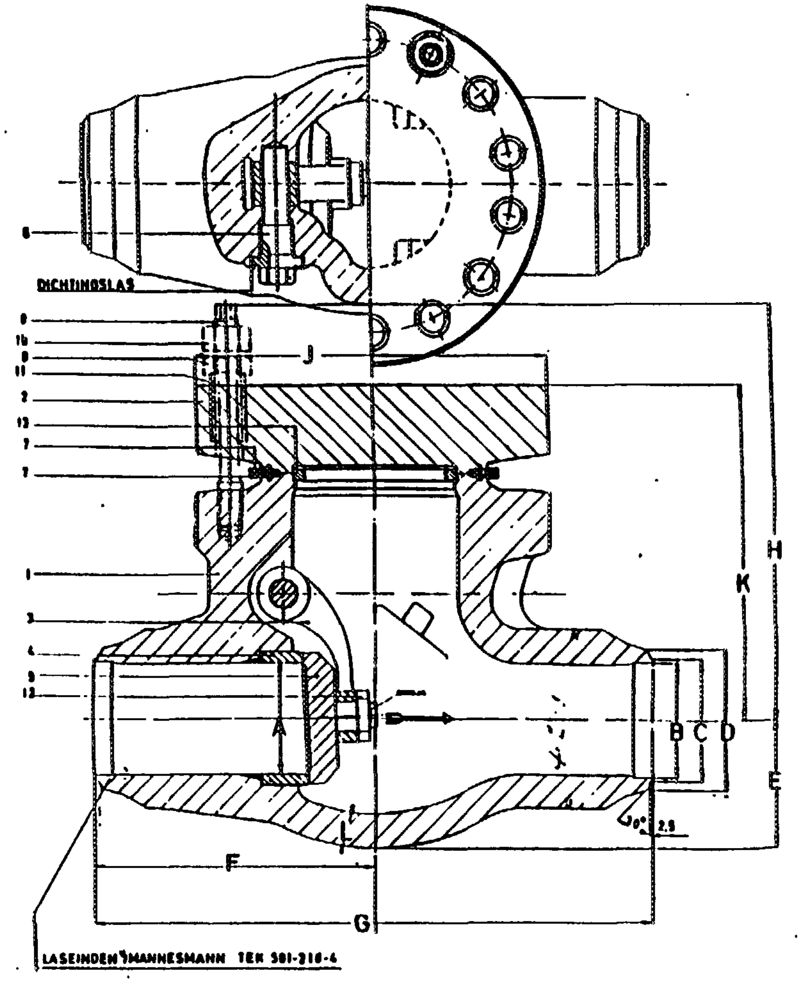


FIG. 1. Swing check valve

CAST STAINLESS STEEL CHECKVALVE
with vertical hanging disk and
membrane-seal



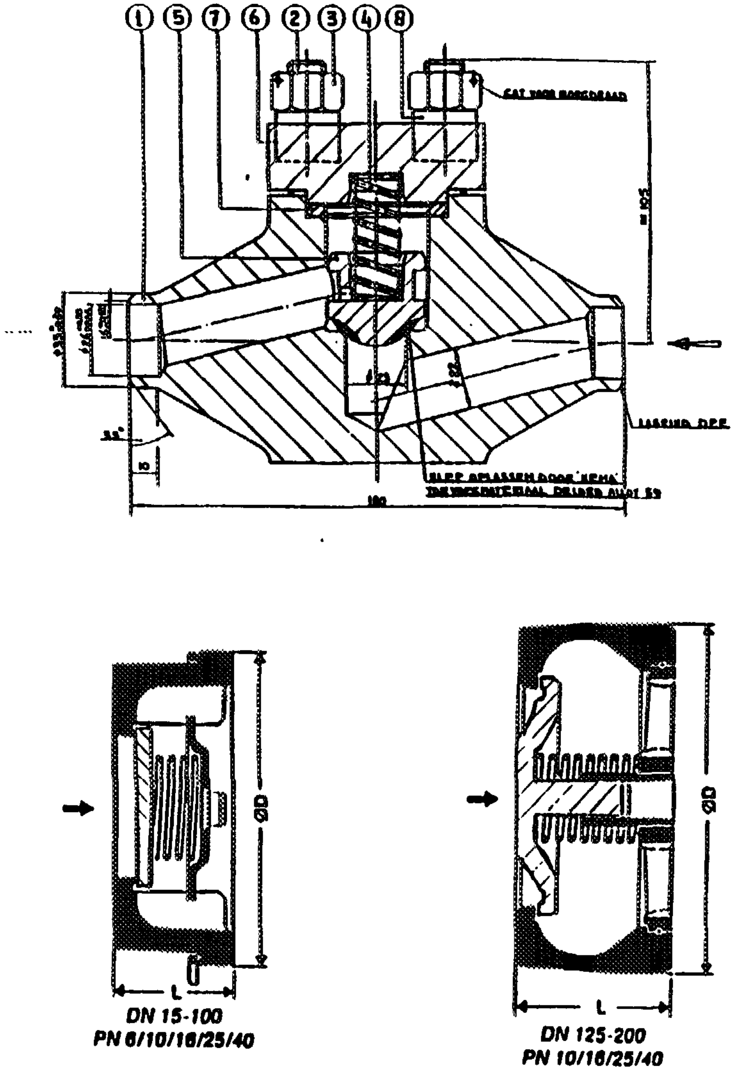


FIG. 2. GKN check valves

3. SYSTEMS WITH PASSIVE COMPONENTS

At GKN the systems which are worth mentioning in connection with passive check valves are:

- The Isolation Condenser System (NCS system):
  This system is the passive system at GKN and will be the main system in this report. The 4 inch check valve in the IC discharge line is closed and exposed to reactor pressure; it is located in the reactor chamber and installed horizontally in the piping to the reactor vessel.

  Systemdata:
  - Water Level in IC (Top) : 39.100 mm
  - Injection nozzle on Reactor Vessel : 28.468 mm
  - Water level in Reactor Vessel : 27.325 mm
  - Maximum watercolumn : 10.632 mm

- The Low Pressure Core Flooding system (KIS system):
  This system has the function of core flooding at pressures below 1.57 MPa. The four (4) main 8 inch check valves, two in series in each KIS discharge line have a passive function and are installed in horizontal and vertical positions. One of the valves in each line is exposed to reactor pressure while the maximum pressure for the second check valve is below 1.57 MPa.

  Systemdata:
  - Injection nozzle on Reactor Vessel : 28.875 mm

- The Standby Liquid Control System (NGS system):
  The function of this system is to inject a boron-solution in the reactor-vessel. The piston-type check valve has a passive function and is exposed to reactor pressure.

  Systemdata:
  - Injection nozzle on Reactor Vessel : 28.163 mm.

3.1. Isolation condenser system (NCS)

3.1.1. *Intended Function*

The Isolation Condenser System (fig. 3) is a safety-system with the design-basis that, after a scram and isolation, the decay-heat can be removed through a closed water/steam cycle (through natural circulation) assuring that the reactor pressure is maintained below the opening pressure for the Automatic Depressurization System (ADS). The system is considered as a "stand-by" passive system.

Capacity	:	(Measured) 14MW.
at:		
Operating pressure	:	5.49 MPa.
Temp. tubeside	:	270 °C
Temp. shellside	:	100 °C

3.1.2. *Failure Mode*

The failure mode for the Isolation Condenser in the GKN-PSA is:

- ISOLATION CONDENSER FAILS.

3.1.3. *Reliability*

The present system unavailability in de PSA, for the NCS-system is 3.44×E-02.

3.1.4 *Component/System availability*

The unavailability of the discharge piping of the Isolation Condenser is the main contributor to the unavailability of the NCS-system with **3.29×E-02**. The contribution of the failures of the check valve to this number is **1.7×E-03**.

3.1.5 *Lifetime*

The IC is a part of the primary reactor support systems and has to be available as long as the reactor will be in operation.

3.1.6. *Testing Requirements*

The check valve T-18020 and the motor operated valve (MOV) 18005 are yearly tested simultaneously according to plant-procedures. This test is done before refuelling and with **open** reactor vessel.

IC tests after refuelling : 26.

As shown in Table I, the IC has been seventeen (17) times in operation under different power- and pressure conditions and for various time-intervals.

3.1.7. *MMI*

The IC will automatically come in operation when reactor pressure reaches 8.14 MPa. The motor operated valve 18005 will open and control the reactor pressure at 8.14 MPa to prevent ADS actuation. The IC can also be operated manually from the control room. Manually opening of the MOV 18005 is also possible locally in the reactor building.

3.2. Low pressure core flooding system (KIS)

3.2.1. *Intended Function*

The Low Pressure Core Flooding System (fig. 4) is a safety-system for injection of water to cover the reactor-core under LOCA conditions whereby the pressure is below 1.57 MPa.

Discharge pressure	:	1.57 MPa.
Capacity	:	180 $m^3.h^{-1}$
Temperature	:	95 °C (max.)

3.2.2. *Failure Mode*

The failure mode for the Low Pressure Core Flooding system in the GKN-PSA is:

- THREE OF THE FOUR KIS PUMP STRINGS NOT AVAILABLE.

3.2.3. *Reliability*

The present system unavailability in the PSA, for the KIS system is: 3.76×E-03, (Short Term).

TABLE I. ISOLATION CONDENSER IN OPERATION

ENS95

GKN **ISOLATION CONDENSER IN OPERATION**
File: *GKNICS.XLS*
Date: 17NOV1994

FWS FEED WATER SYSTEM
I_SCRAM *ISOLATION SCRAM*
SD SHUT DOWN

	Date:	Event:	Description	Timespan: From:	Till:	Pressure (ato.) Start:	End:
1	03SEP68	SCRAM	ICS IN OPERATION AFTER SCRAM.			70.00	70.00
2	26OCT68	**TEST**	TEST ICSbundle #1 and #2 AND INCREASE POWER FROM 5MWe to 52MWe.	12:42	15:01	70.00	70.00
3	06DEC68	**TEST**	TEST ICS AT START-UP @ 15MWe.	22:35	23:07	70.00	70.00
4	04JUL69	I_SCRAM	ICS IN OPERATION TO CONTROL REACTOR PRESSURE.	00:58		68.00	68.00
5	22AUG69	**TEST**	WATERHAMMER TEST. REACTOR PRESSURE FROM "7" to "55" ato.	02:00	07:25	7.00	55.00
6	28JAN70	SD	ICS TAKEN IN OPERATION TO KEEP PRESSURE @ "4" ato.	10:04	16:01	5.40	4.00
7	01JUN70	**TEST**	TEST FROM "0" to "55" ato.	01:27	07:50	0.00	55.00
8	23SEP70	SD	ICS IN OPERATION TO REPAIR RHR.	08:00	11:00	3.20	1.50
9	30SEP70	**TEST**	TEST FROM "1" ato TO "55" ATO.	15:10	21:00	1.00	55.00
10	02OCT70	I_SCRAM	ICS IN OPERATION TO CONTROL PRESSURE @ "68" ato. (LOSP : 3 kV).	16:16	23:10	73.00	69.00
11	19OCT70	**TEST**	ICS IN OPERATION @ "1" ato.	11:34	13:19	1.00	1.00
12	04DEC73	SD	ICS IN OPERATION @ "5" ato. i.c.w. LEAKAGE RHR. (ALSO TURBINE BEARING).	08:30	21:25	5.00	5.00
13	01JUL75	FWS	ICS IN OPERATION THROUGH JULY 05. FW LINE NOT AVAILABLE.	11:58	5days	2.00	2.00
14	04FEB76	I_SCRAM	ICS IN OPERATION TO CONTROL PRESSURE (FAILURE IN 75 MVA TRF-COOLER).	02:24	03:16	70.00	70.00
15	18AUG84	I_SCRAM	ICS IN OPERATION "3 minutes" AFTER I_SCRAM ADS-CHAIN 2 FAILURE.	08:33	12:09	76.00	10.00
16	24FEB88	I_SCRAM	ICS IN OPERATION AFTER LOSP 3kV.	13:42	19:00	76.00	38.00
17	24MAY88	I_SCRAM	ICS IN OPERATION DURING ADS-TEST.	14:02	15:30	70.00	45.00

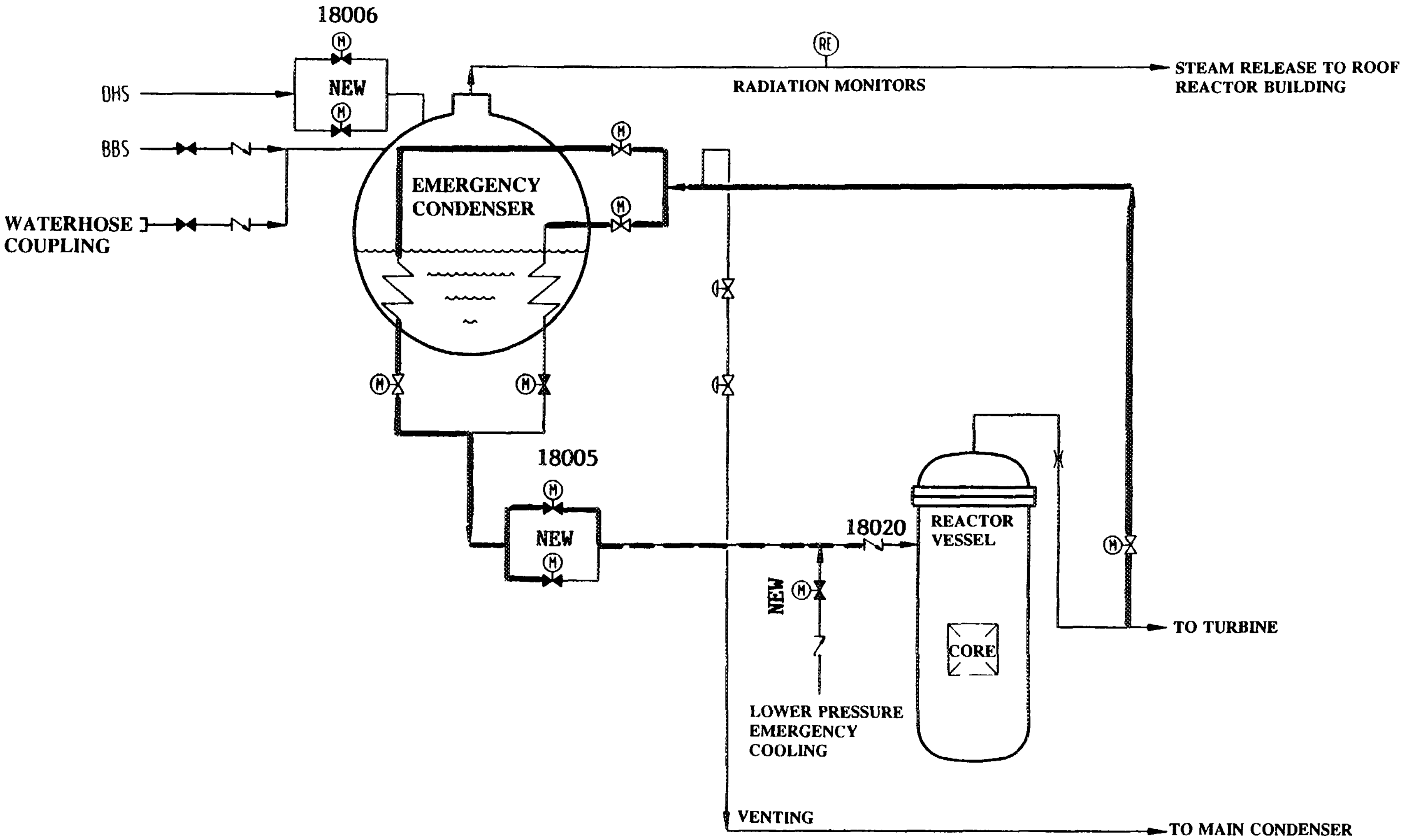


FIG. 3 Schematic of NCS - system

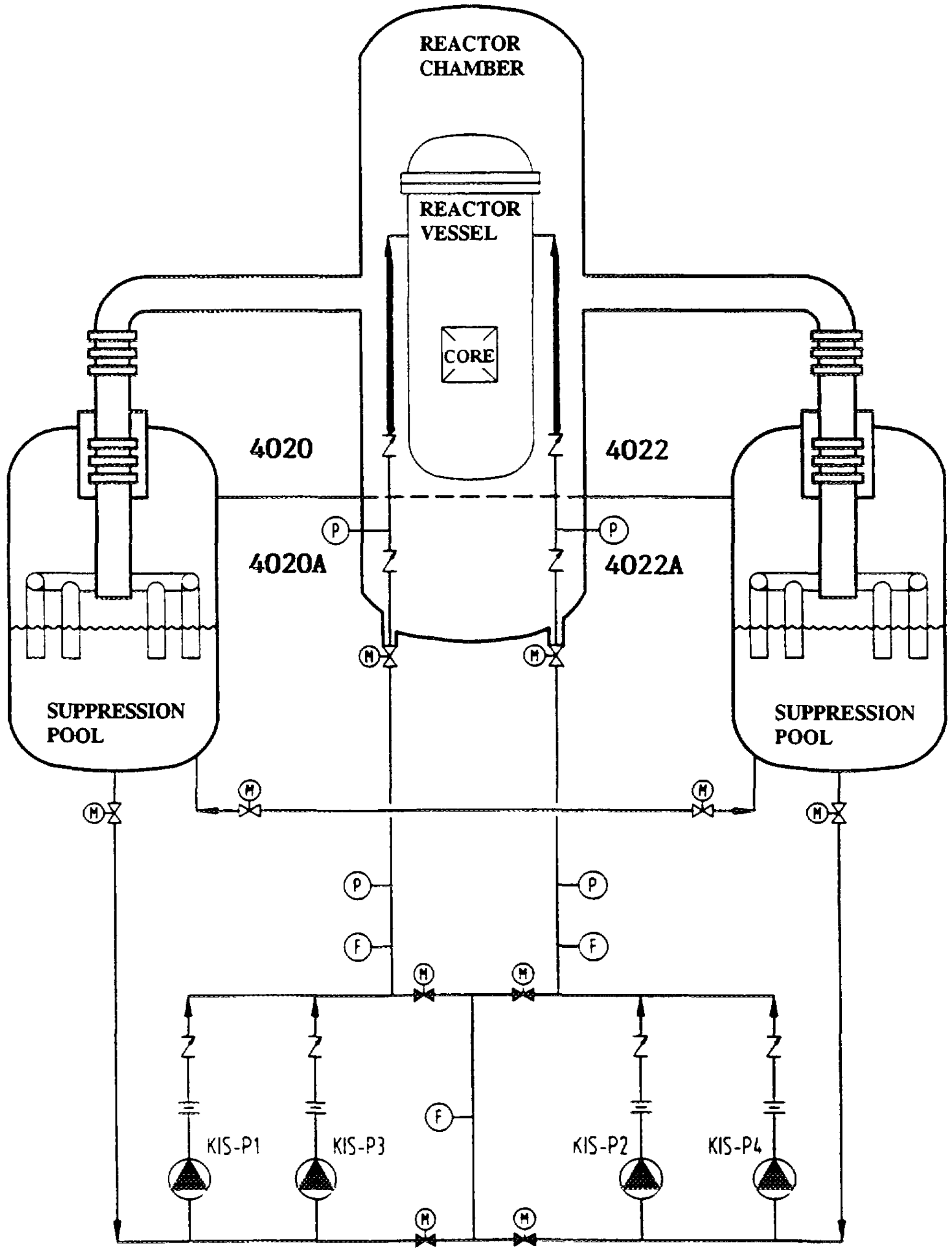


FIG. 4 Schematic of KIS-system

3.2.4 *Component/System availability*

In the unavailability of the KIS system the CCFs for the **check valves** have an important contribution. If this system has to be improved one might consider a different type of check valves in one of the KIS discharge lines.

3.2.5 *Lifetime*

The KIS as a part of the ECCS is a primary reactor support system and has to be available as long as the reactor will be in operation.

3.2.6 *Testing Requirements*

The discharge lines with check valves T40/04020 and T40/04020a resp. T40/04022 and T40/04022a are tested **yearly** after refuelling according to plant-procedures.

Discharge line tests after refuelling : 26.

The KIS system up to the isolation valves is tested **monthly**.

3.2.7 *MMI*

The KIS will automatically come in operation after either one of the following signals:

- high pressure drywell
- opening of the ADS-valves

The KIS system can be put in operation manually from the control room and the KIS-pumps can be started locally.

3.3. Standby liquid control system (NGS)

3.3.1 *Intended Function*

The Standby Liquid Control System (fig. 5) is a safety-system for injection of a boron-solution (natriumpentaboraat) in the reactor vessel to bring down the reactor from "full power" to cold "sub-critical" condition".

Discharge pressure	:	10.59 MPa.
Capacity	:	1.14 $m^3.h^{-1}$
Temperature	:	50 °C

3.3.2 *Failure Mode*

The failure mode for the NGS system in the GKN-PSA is:

- NGS FAILS TO DELIVER LIQUID POISON TO REACTOR.

3.3.3. *Reliability*

The present system unavailability in the PSA, for the NGS system is: 4.06×E-02.

3.3.4 *Component/System availability*

The unavailability of the system is mainly determined by the valves in the common discharge header. The marginal improvement which is possible in the reliability of the system is a result of the system design. The operator has to start the NGS system manually and the basic-event related to the failure of this action has a probability of **3.00×E-02**.

3.3.5 *Lifetime*

The NGS is a part of the primary reactor support systems and has to be available as long as the reactor will be in operation.

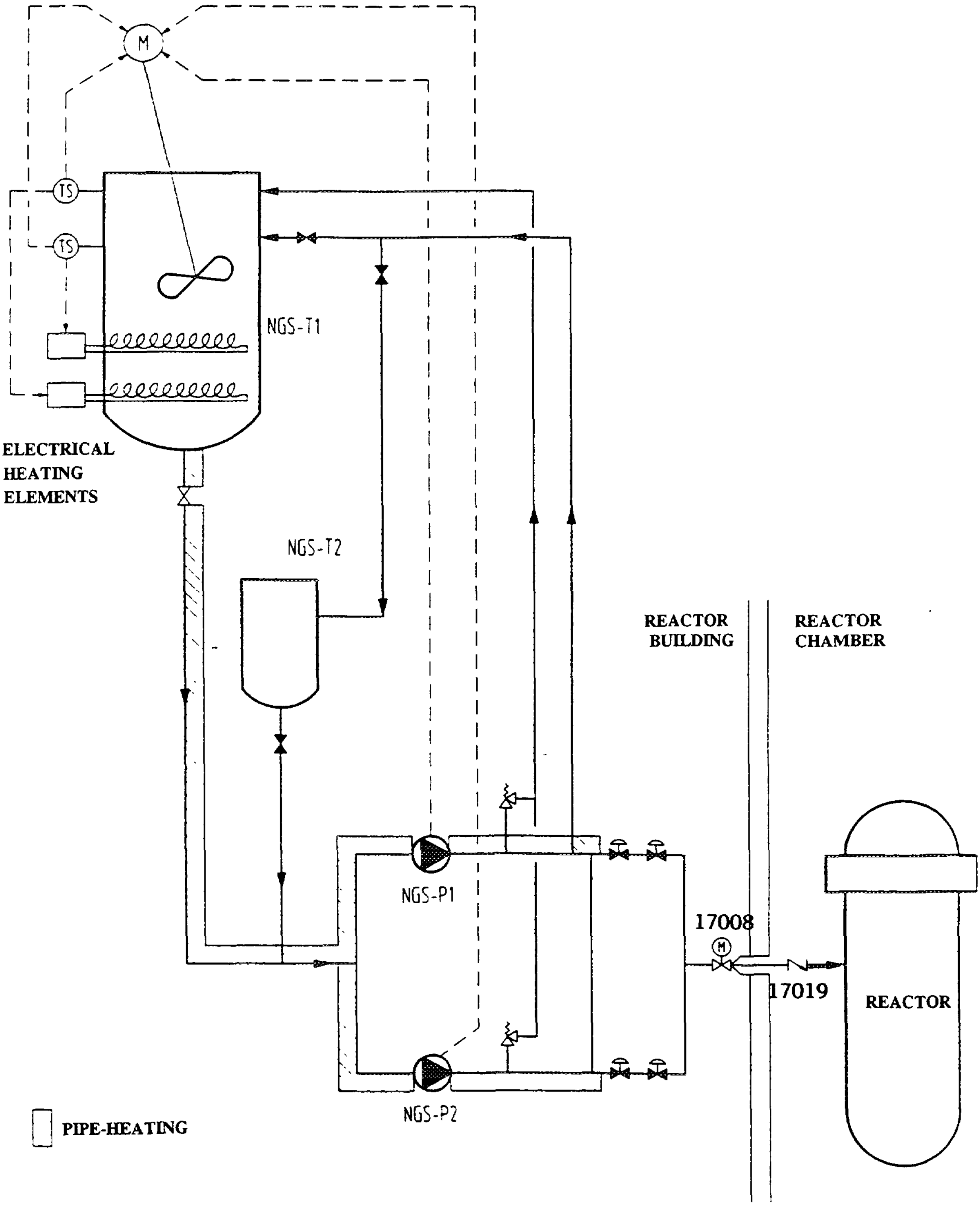


FIG. 5 Schematic of NGS-system

3.3.6 *Testing Requirements*

The check valve T-17019 and the MOV 17008E are yearly tested simultaneously according to plant-procedures.

This test is done after refuelling and with **closed** reactor vessel.

Number of test after refuelling : 26.

3.3.7 *MMI*

The NGS system is manually put in operation from the control room and the NGS-pumps can be started locally.

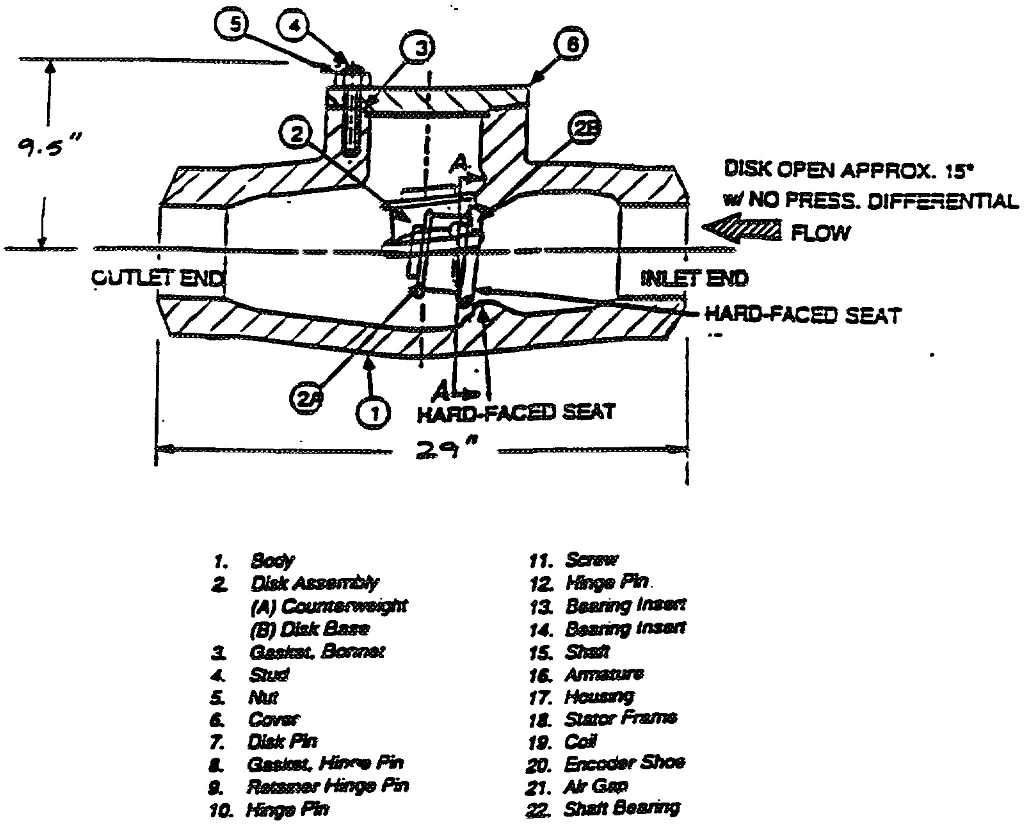


FIG. 6. Biased-open check valve

4. TECHNICAL SPECIFICATION ON REPORTED CHECK VALVES.

System + Injection level in RV	Valve Type	Valve Position	Pressure
NCS (IC) 28.468 mm	Swing	Horizontal	7.55 MPa
KIS-Line 1 28.875 mm	Swing (1st) Swing (2nd)	Vertical Vertical	7.55 MPa <1.57 MPa
KIS-Line 2 28.875 mm	Swing (1st) Swing (2nd)	Vertical Horizontal	7.55 MPa <1.57 MPa
NGS (Boron) 28.163 mm	Piston type	Horizontal	7.55 MPa

5. NEW DESIGNS

To improve the check valves to be installed in the passive systems of the new General Electric SBWR-600 reactor figure 6 shows the design of the biased open check valve for the Gravity Driven Cooling System (GDCS). This valve with a GE patent has been designed and the testing program for this model will start soon.

6. CONCLUSIONS

Plant Data-analysis shows that in the 25 years of operation no catastrophic failures with check valves have been experienced in safety systems at GKN-Dodewaard nuclear power plant.

The Isolation Condenser System with it's **"swing check valve"** has been in operation for at least 43 times (17 + 26) under various plant conditions and has shown that natural circulation in the passive Isolation Condenser has performed it's function upon request.

Automatic testing of passive components/systems during operation can also be a method to verify the function and thus guarantee the availability of these systems.

The objective for the new design for check valves is to improve the performance of the valve at low differential pressures.

The reliability of check valves is an important safety issue in the design in passive systems for the new generation reactors.

7. CONSEQUENCES OF EXTREME LOADS

All check valves presented in this report are subjected to reactor pressure. During operation or tests no catastrophic failures were detected.

8. REDUNDANCY AND DIVERSITY

Redundancy and diversity could improve the NCS-, KIS- and the NGS system. Because of accessibility problems and exposure-dose, which is expected to be substantial when installing the redundant component, other solutions where chosen to bring down the Core Damage Frequency (CDF).

BACKUP PASSIVE REACTIVITY SHUTDOWN SYSTEMS

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Abstract

The paper reviews self-actuated shutdown systems (SASSs) for liquid metal-cooled fast reactors (LMFRs). Principles of operation are described, advantages and drawbacks analyzed, and prospects for application in advanced fast reactors examined.

Ways to improve reactor self-protection via reactivity feedback amplification and related problems are discussed.

1. INTRODUCTION

A major task to be achieved in advanced reactors is to ensure reactor self-protection in the case of the most severe accidents, i.e., unscrammed loss-of-flow (ULOF), unscrammed loss-of-heat sink (ULOHS) and unscrammed transient-over-power (UTOP).

Hence, reactor safety is to be ensured in the case of any accident, even under conditions of total active system failure, i.e., by means of inherent safety features and specific self-actuated systems and elements.

Reactor shutdown is a key safety function that should be effective in any accidental event.

Reactor power can be lowered to the safety level under the conditions of active reactor protection system failure via effective negative reactivity feedback, self-actuated shutdown systems or simultaneous application of both.

Interest in self-actuated shutdown systems has never ceased. However, it was advanced reactor concepts as well as high modern safety standards that led to SASS development. A wide range of experiments to prove self-actuated shutdown systems feasibility are being conducted in parallel with theoretical studies. For instance, SASSs application in LMFRs is currently under study.

Simultaneously, the full potential of inherent LMFR safety features is being investigated. Here most comprehensive way of attaining reactor self-protection is negative reactivity feedback amplification.

The present paper discusses the requirements of self-actuated shutdown systems and LMFR characteristics, and analyzes advantages and disadvantages of some of the means as well as their prospects.

2. GENERAL REQUIREMENTS OF PASSIVE SHUTDOWN SYSTEMS

The general requirements of passive shutdown systems in sodium-cooled fast reactors are determined taking into account the following pre-requisites. The system should ensure termination of the accident and transition of the reactor to a safe condition. In practical terms, this implies the impossibility of fuel element melting and coolant boiling. Requirements for a self-actuated shutdown system of this kind for the BN-800 fast reactor are as follows:

- It should be efficient enough to shut down the reactor (about -0.8 % $\Delta k/k$).
- It should be activated by a core outlet sodium temperature of 650°C or by an in-core coolant flowrate drop below 0.6 of the nominal value.
- The delay between reacting the trip point and operation of the passive system should not be long enough to lead to multiple fuel pin cladding failure (delay not exceeding, 2 sec), nor to sodium boiling in the core (delay not exceeding 5 sec).
- It should differ in design from the standard shutdown system.
- It should be insensitive to displacements due to temperature in the upper part of the reactor.
- The absorber position in self-actuated shutdown system devices should be under control.
- Unloading of self-actuated shutdown system devices by the standard transfer machine should be practicable.

3. CLASSIFICATION OF SELF-ACTUATED SHUTDOWN SYSTEMS

Self-actuated shutdown systems can be classified according to their methods of influencing neutron flux, according to their principles of operation and the range of accidents the system is intended for, or according to the form of the absorber.

Solid absorber insertion into the core is used to influence neutron flux in virtually all backup passive reactivity shutdown systems for fast reactors except for gas expansion modules (GEMs), whose operation is based on varying the neutron leakage.

The actuation principles employed in the devices differ greatly. They are as follows:

- absorber rods or balls suspended in the coolant flow which go down into the core when the coolant flowrate drops,
- coolant expulsion from the core in the event of pump turn-off,
- temperature-induced changes in magnetic properties of the materials used in the control rod latches, which result in the rods being released;
- temperature-induced changes in the form of the materials used in the latches which result in the rods being released;
- elongation of special elements caused by coolant temperature increase and resulting in absorber rod release and insertion into the core;
- direct connection of the latch to a standard power supply.

4. BACKUP PASSIVE SHUTDOWN SYSTEMS FOR FAST REACTORS

4.1. SADE system

The EFR project employs the so-called "third shutdown level" based on passive principles in order to bring the reliability of the shutdown system to the required level. The SADE system is a part of this system. It employs passive means to block the power supply to electric magnets of some groups of DSD (diverse shutdown rods) after loss of power in the primary circuit pumps.

SADE is a passive system. The fact that it uses the rods of the standard shutdown system may be considered its major drawback. The system is applicable to ULOF and ULOHS types of accidents. A similar system is provided for the U.S. Advanced Liquid Metal Reactor (ALMR).

4.2. Control rod enhanced expansion device (CREED)

CREED is the second component of the "third shutdown level" for the EFR. It is a self-actuated shutdown system with absorber rod insertion into the core resulting from an increase in the core outlet temperature. The purpose of the device is to prevent sodium from boiling in the core.

The EFR project studied two variants of the CREED concept. One is based on thermal expansion of a fixed mass of liquid sodium (Fig. 1), while the other depends on the elongation of a stack of bimetallic washers (Fig. 2) /2/. These variants have a common principle for absorber rod delatching. A temperature rise results in displacement of one

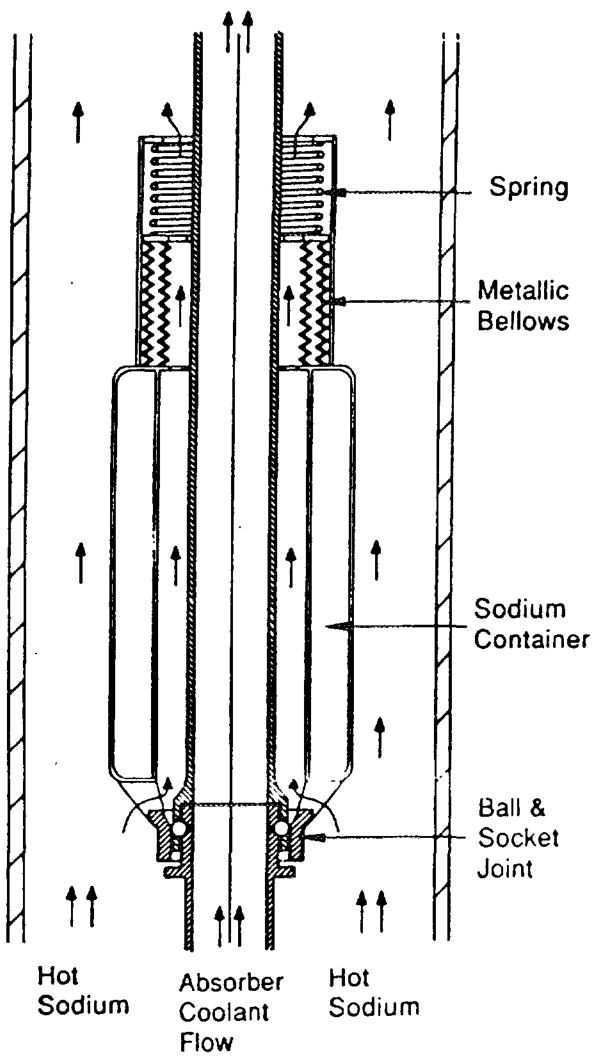


FIG. 1. Enhanced thermal expansion hydraulic ring concept.

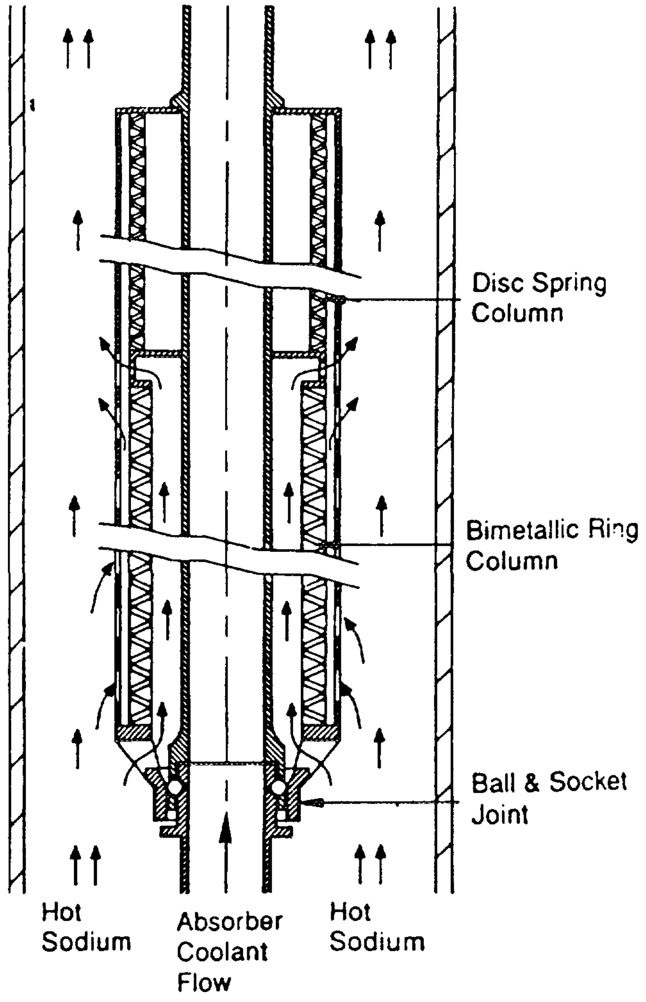


FIG.2. Enhanced thermal expansion bimetallic ring concept.

element of the socket in a ball and socket joint resulting in release of the balls and allowing the absorber rod to drop into the core.

CREED is an effective remedy for ULOF and ULOHS accidents. It ensures that the maximum permissible sodium temperature, 900°C, is not exceeded.

Among the advantages of CREED we can note its complete independence from the main reactor shutdown systems and its compact form.

Application of the principle of solid absorber rod insertion which is employed in the standard shutdown systems, constitutes the main drawback of CREED. In addition, the latch is not immune to failure, e.g., as a result of coolant impurity deposition on it, or when two elements in contact with each other get stuck together. The actuation temperature threshold for these devices may vary widely. The devices cannot be reset.

4.3. Gas expansion modules (GEMs)

These devices are based on variations in neutron leakage from the core due to changes in coolant volume (Fig.3). The device in question is intended for reactor protection in ULOF accidents /3/. A cut in primary pump power leads to a drop of core inlet pressure which in turn results in a drop in the sodium level in the GEM.

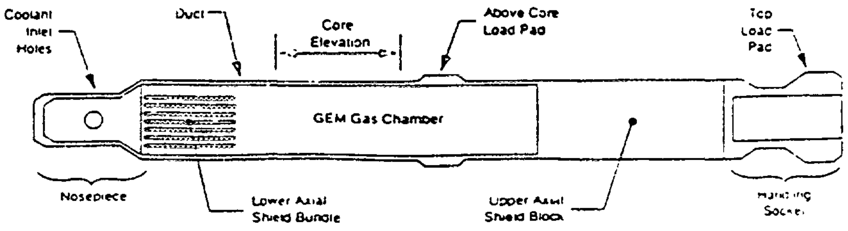


FIG. 3. Gas expansion module.

A device of this type has been tested in the FFTF reactor and has proved to be operational. At present GEMs are planned to be used in the PRISM reactor design. They are intended to provide negative reactivity insertion of about -$0.5 - $0.8.

This method of negative reactivity insertion is characterized by fast response, does not employ moving parts, can be used many times without special operations to reset it, and is easy to control.

It has the following drawbacks:

- It is not multipurpose: it is practicable in the case of only one accident, i.e., loss-of-flow through the reactor core (ULOF).
- GEMS are most efficient in small-power reactors in which neutron leakage plays a major role; their efficiency in more powerful reactors is much lower.
- The efficiency of a GEM decreases at low coolant flowrate.
- Special procedures must be developed and employed for installation of a GEM into the reactor.
- Coolant can trap the gas from GEM, especially in transient regimes (pump startup, flowrate change, accidental transient), which may lead to a decrease in its efficiency.

4.4. Flow suspended rods

The principle of operation of hydraulically suspended rods is as follows. Under the normal conditions of reactor power operation absorber rods are suspended in the coolant flow, their absorbing part being above the core. In ULOF accidents when the coolant flowrate through the core ceases, the rods go down into the core under their own weight (Fig.4) /4, 5/.

Some time ago these rods were studied as a passive safety device within the British CDFR project. They had an independent circulation loop with a special pump that turned off simultaneously with the primary circuit pumps in the case of a power cut.

Development of such devices is now under way in Russia.

The Russian variant is based on the flow-suspended rods containing boron carbide, which are placed in a standard shutdown system subassembly. The subassembly is placed in the core and its actuation is induced by the changes of coolant flowrate in the core.

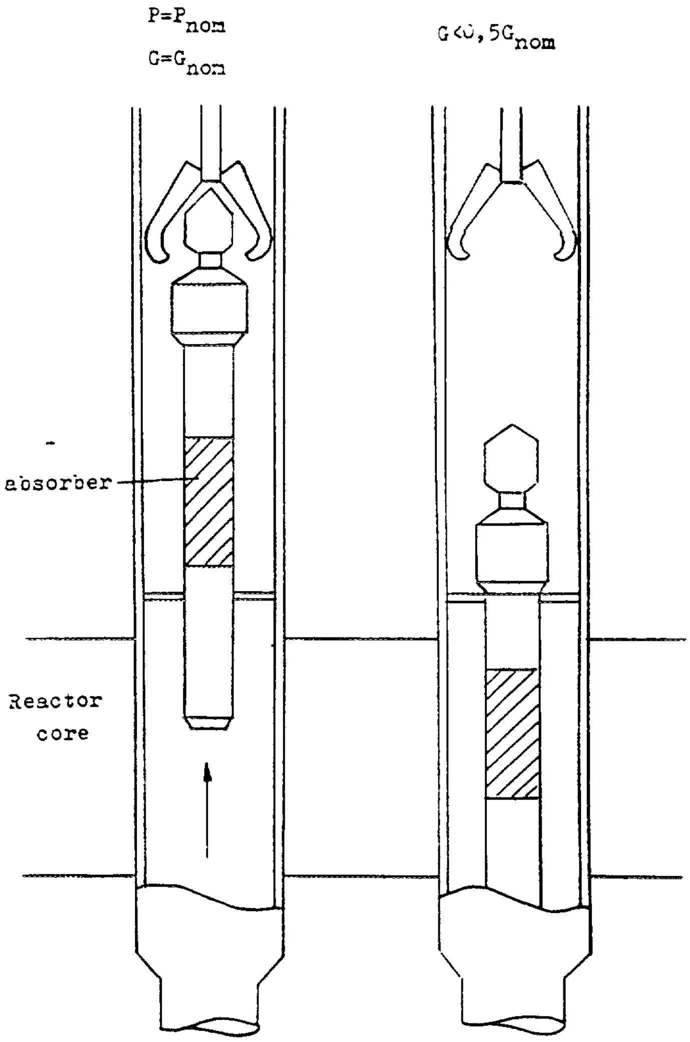


FIG. 4. Passive secondary shutdown system with flow-suspended rods for BN-800.

In order to prevent absorber rods from rising during reactor shutdown in the event of unauthorised increase in flowrate above the allowable level, provisions are made for the hydraulic relief of the actuating part. The rods are returned to the upper working position with the help of grips before power increase takes place.

Devices of this kind are being developed for BN-600 and BN-800 reactors /4/. Their flowrate actuation threshold has been set at 0.6 of the nominal value. Thus, reactors can be also run on two out of three working loops.

Numerous in-core and simulation experiments have been made in order to confirm the correct operation of these flow-suspended rods.

The actuation mode of the rods has been tested in the BR-10 reactor, inclusive of on-power operation. Design characteristics including the time of insertion into the core (0.7 sec and 1.2-sec for different variants) were substantiated. Longevity tests are underway in the BR-10 reactor; the devices in question are actually being used as the standard reactor protection system.

A full-scale simulation of such a device for the BN-600 reactor is being tested in a hydraulic (water) mock-up. Results concerning the characteristics of such devices when employed in power reactors were obtained. These include the time of insertion of the rods into the core which were 4.7 and 6.1 sec for two variants. These characteristics ensure reactor shutdown with a temperature margin to sodium boiling exceeding 200°C.

These devices have the following advantages:

- simple actuation principle employed;
- rather fast response providing a considerable temperature margin to sodium boiling;
- can be used many times with only a simple inspection and resetting procedure to restore the operational configuration of the device.

The drawbacks are as follows:

- it is not multipurpose, i.e., it can only be used in loss-of-flow accidents (ULOF).
- it employs the same principle of solid absorber rod insertion as the standard shutdown system.
- it has a limited range of allowable coolant flowrate values in the core.
- operational parameters may change due to changes in hydraulic characteristics, i.e., due to the deposition of impurities or oxides on the flow part of the device and its elements.

4.5. Absorber balls shutdown system

The self-actuated shutdown system proposed in /6/ is similar in its principle to the flow suspended rods. The difference is that it employs flow suspended absorber balls instead of rods. In addition, in order to widen the range of accidents by which the device is actuated, it contains a shutoff valve actuated by a thermal device based on a Curie point magnet. The latter is actuated by the increase of the coolant temperature and blocks the passage of coolant through the shutdown device, which also results in the insertion of absorber balls into the core (fig.5).

Due to the above, the device has some advantages over the flow suspended rods:

- it is more versatile and ensures safety not only in ULOF, but also in UTOP and ULOHS accidents;
- it uses a different principle of absorber insertion into the core as compared to the standard shutdown systems, which enhances its immunity to common mode failure;
- significant deformation of the core cannot prevent the insertion of the absorber.

We can also list the following drawbacks:

- Absorber balls can leak from the device if it loses integrity;
- The possibility of the absorber balls jamming in the upper or lower position cannot be excluded;

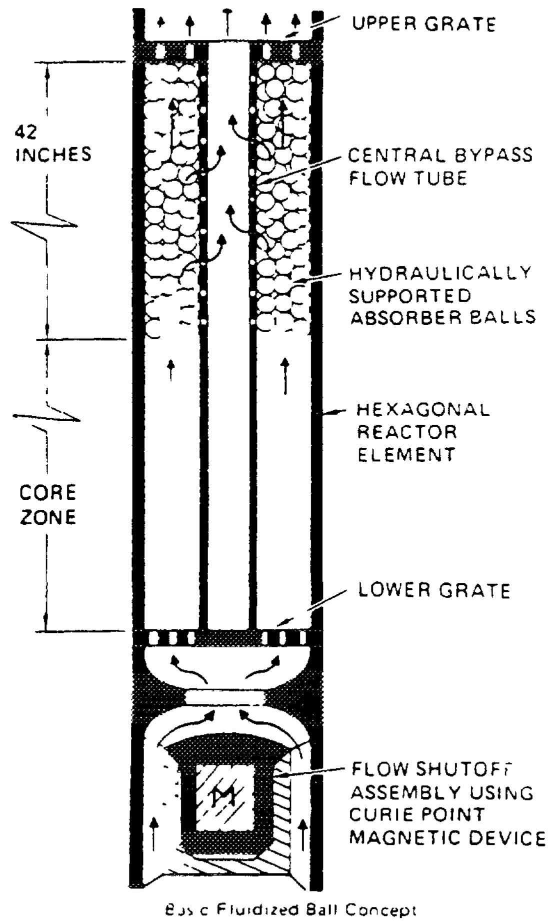


FIG 5

- The possibility of self-welding of the absorber ball cannot be ruled out.

4.6. SASS based on curie point magnets (CPM)

Development of SASSs based on Curie point magnets is underway in Russia /4, 5/ and Japan /7/.

Magnetic materials are used in the latch that holds the absorber rods above the core. The materials lose their magnetism at a definite temperature. Thus, in accidents involving temperature increase the absorber rod is released and dropped into the core.

In Japan, research on such devices is based on electromagnets, while in Russia devices of this kind are developed as fully autonomous.

A most important task is to find materials that significantly change their magnetic properties within the desired temperature range. For the device under development in Russia, the actuation temperature, i.e., the temperature of the coolant, is taken to be 650-670°C with the condition that the actuation time should not exceed 5 sec. A very important feature of the magnetic material for the autonomous variant is the weight it can hold.

In order to ensure the desired rated load the following configuration of the CPM is used. It consists of a permanent magnet made of magnetic alloy with axial magnetization,

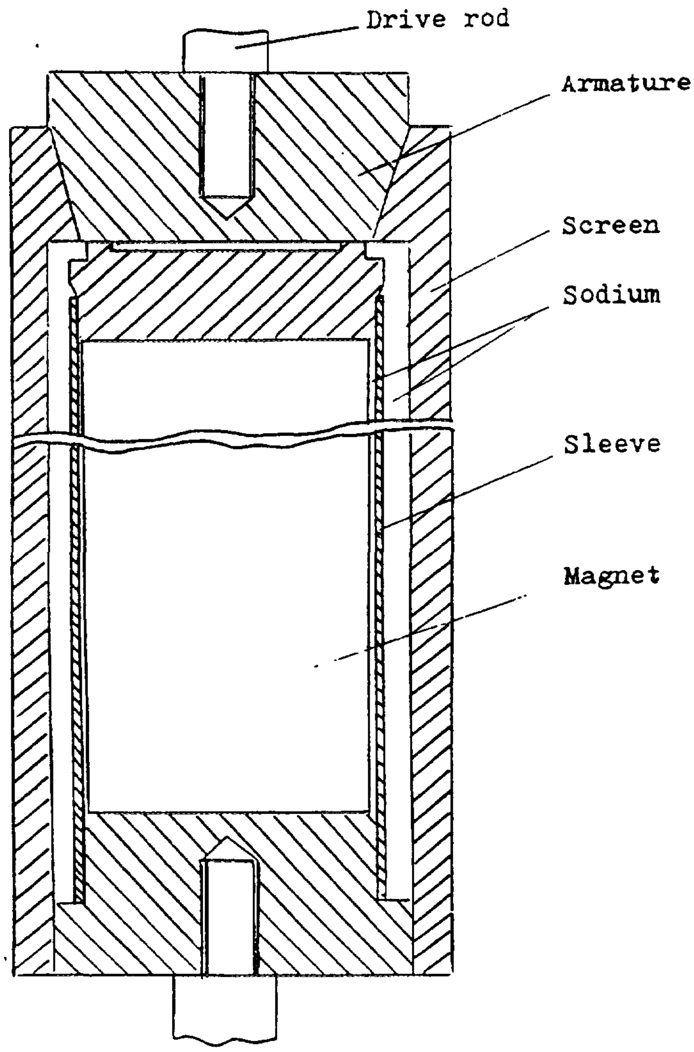


FIG. 6. Mock-up design of magnetic actuating device.

a surrounding screen made of ferrous-nickel alloy with a Curie point of 620°C, and the armature made of Armco iron which is connected to the absorber rod.

Fig.6 shows a mock-up design of the CPM device for an experiment in a sodium rig. The experiment tested its rated load in a flow of sodium in the temperature range of 300 - 680°C, with a rate of temperature increase in the device of about 12°C. The rated load of the device in a gaseous atmosphere is about 8.2kg at room temperature and 2.8kg at 680°C. To study the effects of irradiation, magnetic material specimens have been placed in the BR-10 reactor.

These SASS devices have the advantage of versatility and can be used to prevent any type of accident.

Their drawbacks are as follows:

- They employ the same principle of solid absorber insertion into the core as the standard shutdown system;

- The device may fail to demagnetize due to either insufficient temperature increase in the temperature-sensitive material or increase in its sensitive temperature. The former may result from changes in thermohydraulic features of the device and the basic, thermo-physical properties of the thermosensitive material. The latter may be due to changes in temperature / magnetic properties as a result of gap reduction, e.g., due to accumulation of some magnetic material from the coolant, swelling of the material, etc., or Curie point rise and changes in magnetic permeability.
- The latch may fail to release despite demagnetization of the material, e.g., due to adhesion of the magnet and the rod.

4.7. SASS based on shape memory alloys

Such devices are currently under development in Russia /4, 5/ and in Japan.

The absorber rod is held above the core with the help of a SASS device based on shape memory alloys. When the temperature of the latch reaches a certain level it changes its form and releases the absorber rod which falls down into the core. A method for shaping the material as applied to different rod designs is being developed.

Ti-Ta and Ti-Ta-Hf alloys with a shape recovery temperature of 650°C were selected for the purpose. Currently corrosion studies and mechanical tests are underway. Investigations of titanium alloy in the form of a disk spring pack show that these elements can provide the following characteristics - an actuation time of 1s, a stroke of 6-8 mm, and a force developed of 700 N. Also tested is a titanium-based alloy with addition of rhodium which would have the required level of shape memory temperature.

The device can be used in different types of accidents. The drawbacks are as follows:

- principle of solid absorber insertion common with the standard systems;
- possible irradiation-induced changes of the properties of the material.

4.8. General comments on SASS devices

Analysis of the SASS devices under development shows that;

a) Solid absorbers are mostly used (except for GEMS), usually in the form of rods (the only exception is the use of absorber balls).
b) Hence, they all belong to passivity category C (except for GEMS, which belong to category B) /8/.
c) The principle of solid absorber insertion, which is common to standard shutdown systems, makes most of the self-actuated devices vulnerable to common mode failure simultaneously with that of standard shutdown systems. Common mode failure may be caused by absorber rod jamming due to core deformation resulting from swelling or transient conditions, etc. (except for absorber balls and GEMS).
d) Only temperature-actuated SASS devices are applicable to different types of accidents. SASS devices adapted to coolant flowrate changes can be used for a limited range of accidents (ULOF).

As to application of SASS devices in advanced reactors, temperature-activated devices, which are applicable to all severe accidents, are considered the most promising.

SASS devices sensitive to specific types of accidents should be used in combination with other means so that the whole range of possible accidents is covered.

Development of SASS devices with a liquid absorber is very promising. It will raise the degree of passivity to category B. At present, preliminary studies of such a device using cadmium or indium as an absorber are underway. The device is temperature-actuated and, in addition, functions as a means of reactivity feedback amplification in the initial stage of the accident.

5. Analysis of the effect of reactivity feedback amplification

Reactivity feedback amplification is an effective and universal means of reactor self-protection in the case of severe beyond-design-basis accidents.

In sodium-cooled fast reactors, this implies that even in the most severe accidents sodium boiling and fuel melting do not take place. However, in some cases avoiding sodium boiling and flow melting can be seen as a conservative measure in avoiding core damage. For instance, in the case of negative sodium void effect of reactivity (SVER), sodium boiling and fuel melting in the limited central part of fuel pins does lead to rapid onset of damage to the core. At present, the SVER value in the core of the BN-800 reactor is close to zero (-0.1% $\Delta k/k$).

Analysis of the relations between reactivity feedback components that provide reactor self-protection and taking into account the dynamic factor effects, shows that, in order to ensure reactor safety in the most severe ULOF accident, it is the reactivity feedback components determined by the temperature of the coolant in the core and in the core outlet that should be amplified /9/.

These components may be as follows:

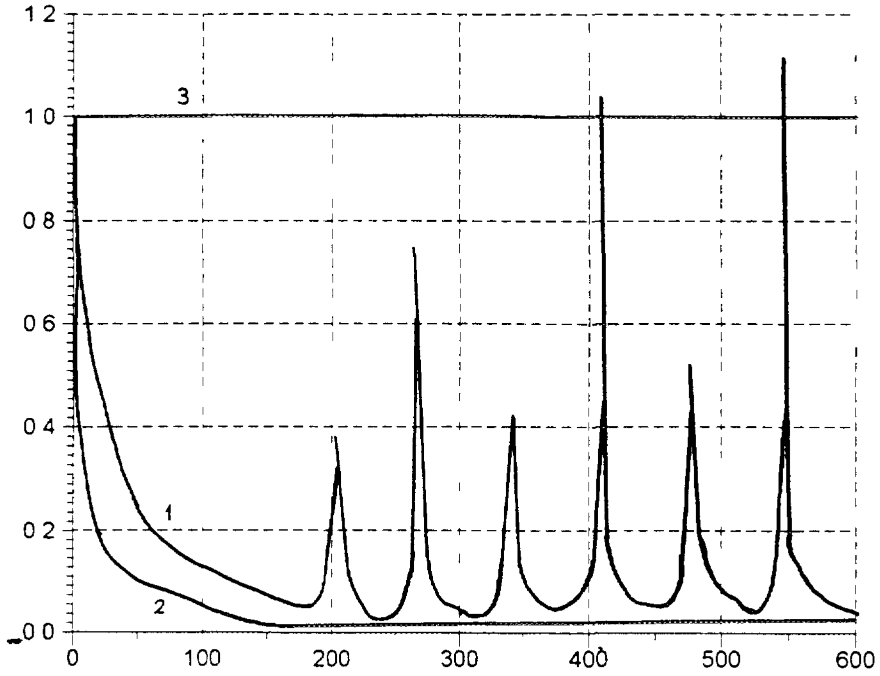
- the effect of radial expansion of the core;
- the effect resulting from the insertion of absorber, into the core, e.g., due to control rod drivers expansion;
- thermal expansion of specific sensitive elements placed in the coolant flow.

However, reactivity feedback amplification may lead to reactor instability, instability boundaries being closer than is called for by self-protection requirements.

As an example, Figs. 7-12 show the calculation results for two transients for a BN-800 reactor type under ULOF conditions.

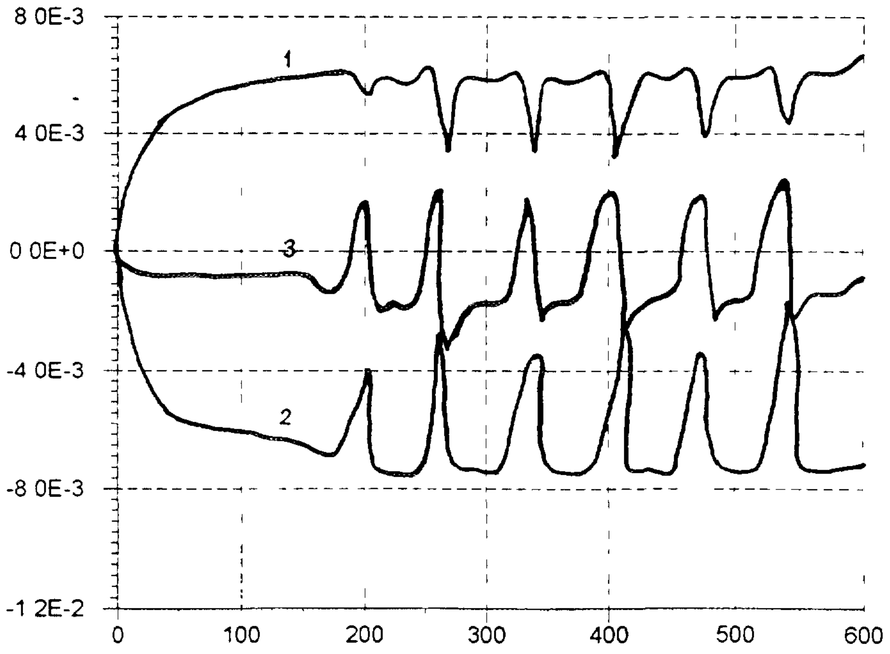
In the first transient (Figs. 7-9) an additional reactivity component, besides the usual ones is introduced. It is caused by changes in the temperature of sodium, at the outlet of fuel subassemblies in the core with a reactivity coefficient at this temperature being equal to $-2*10^{-5}/°C$. This corresponds to an increase of core radial expansion.

It is obvious that the reactivity feedback amplification is insufficient to prevent sodium boiling in the core. However, it also induces reactor instability. This is due to the fact that the response of the component in question to the initial disturbance takes place with a significant phase shift. This fact requires careful analysis of the effects caused by expansion of control rod drivers and their displacement with respect to the core.



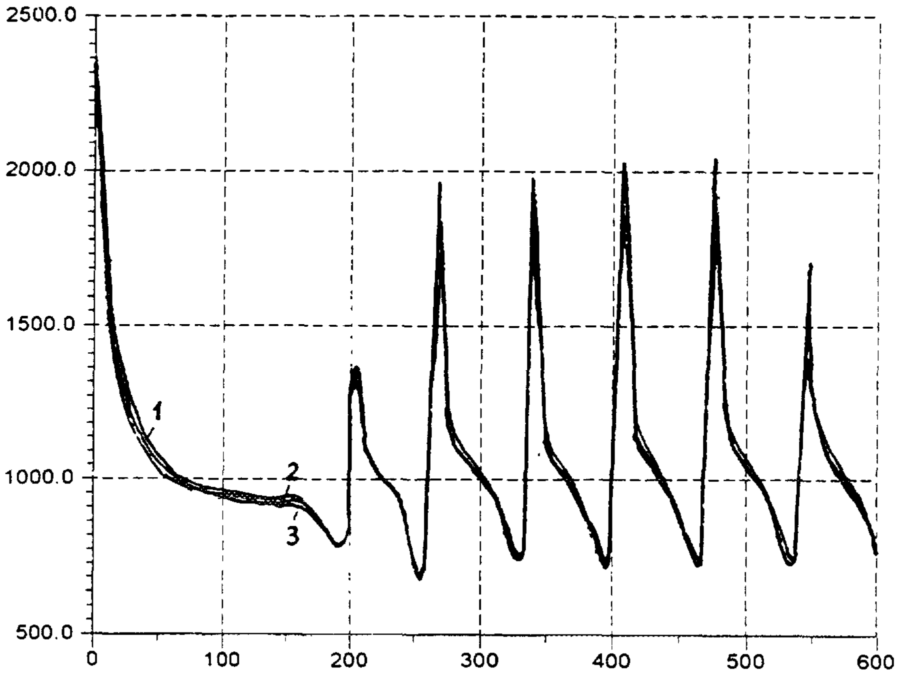
1 - Neutronic reactor power
2 - Coolant flow rate in the core
3 - Coolant flow rate in the secondary circuit

FIG. 7. Accident ULOF for the reactor BN-800. Application of negative reactivity feedback depending on core outlet sodium temperature.



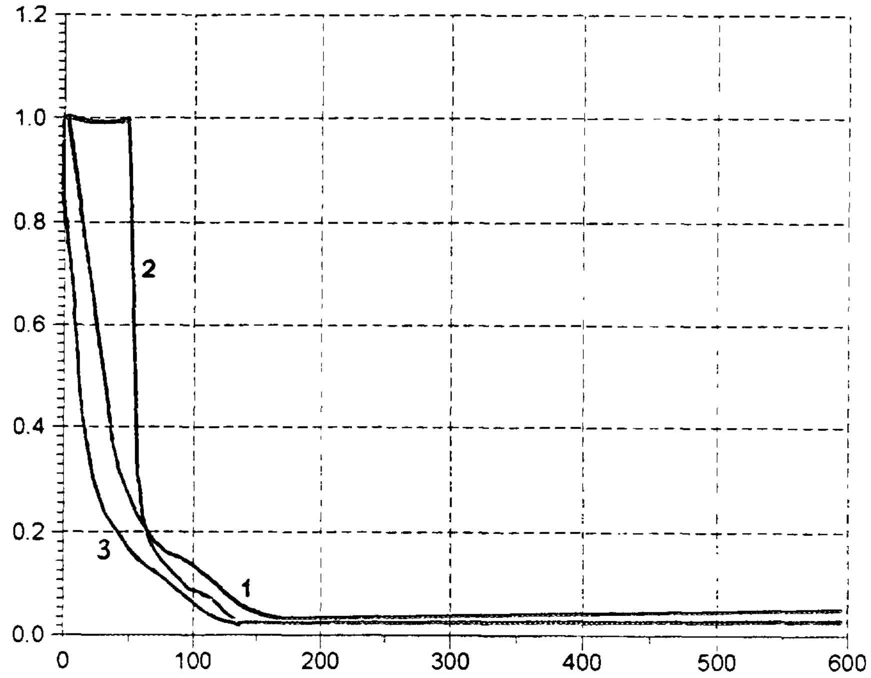
1 - Temperature reactivity effect (without core radial expansion)
2 - Amplified negative reactivity feedback
3 - Net reactivity

FIG. 8.



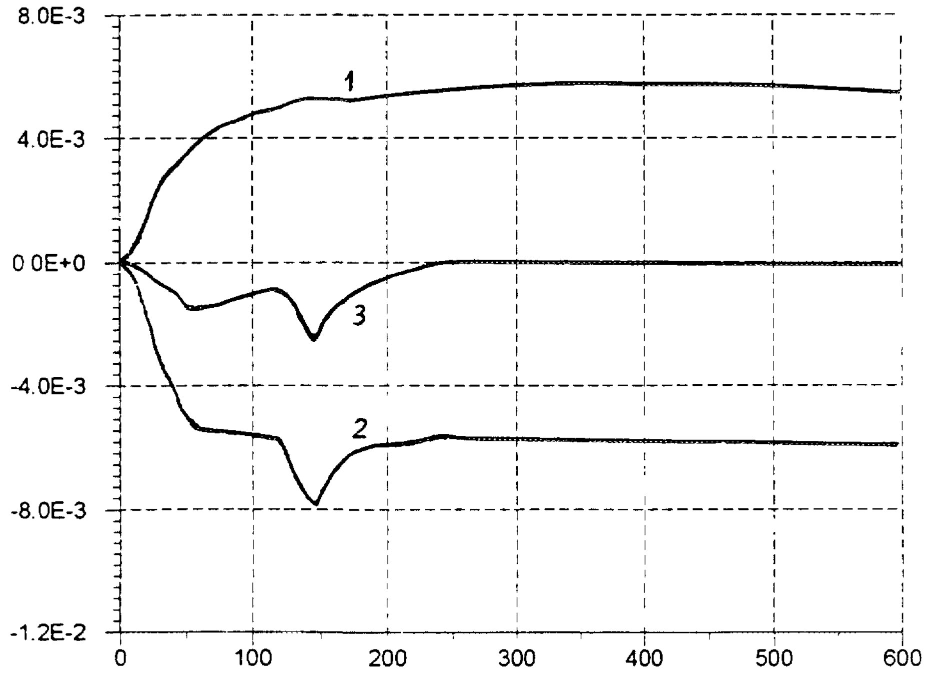
1 - Maximum fuel temperature in the pin centre (LEZ)
2 - Maximum fuel temperature in the pin centre (MEZ)
3 - Maximum fuel temperature in the pin centre (HEZ)

FIG. 9. Accident ULOF for the reactor BN-800. Amplification of negative reactivity feedback depending on core outlet sodium temperature.



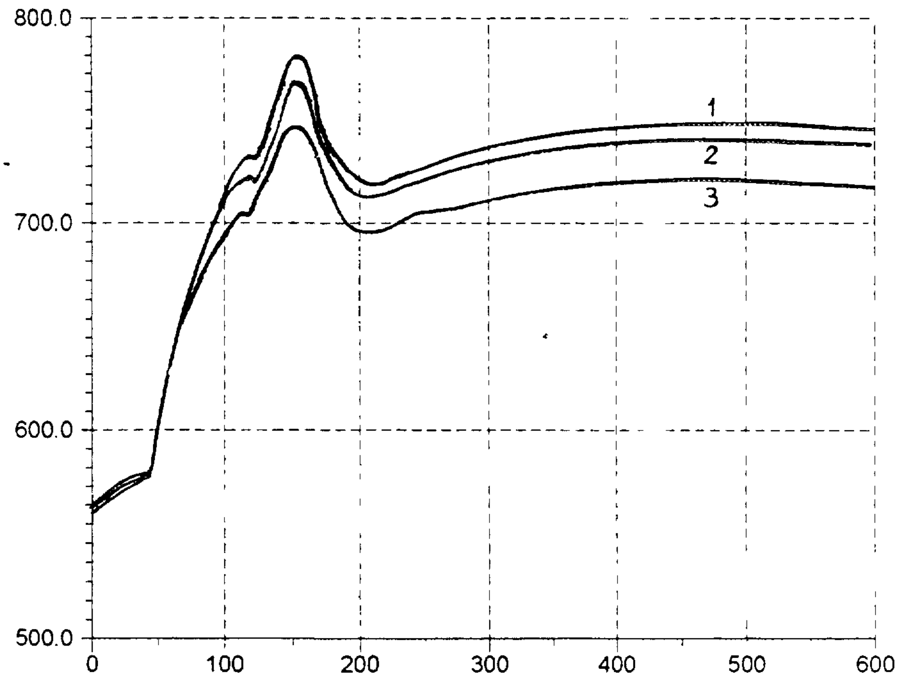
1 - Neutronic reactor power
2 - Coolant flow rate in the core
3 - Coolant flow rate in the secondary circuit

FIG. 10. Accident ULOF for the reactor BN-800. Amplification of negative reactivity feedback depending on sodium temperature in the core midplane.



1 - Temperature reactivity effect (without core radial expansion)
2 - Amplified negative reactivity feedback
3 - Net reactivity

FIG. 11. Accident ULOF for the reactor BN-800. Amplification of negative reactivity feedback depending on sodium temperature in the core midplane.



1 - Core outlet sodium temperature (LEZ)
2 - Core outlet sodium temperature (MEZ)
3 - Core outlet sodium temperature (HEZ)

FIG. 12.

In order to ensure reactor stability in the case of reactivity feedback amplification, use can be made of sensitive elements operating on coolant temperature changes averaged over the height of the core.

Figures 10-12 show the results of calculations of the variant in which the thermo-sensitive element is placed in the median plane of the core. The reactivity coefficient is $-6*10^{-5}/°C$. The process of lowering reactor power in this case is seen to take place without self-sustained oscillations. Sodium in the core does not boil.

6. Conclusions

The above SASS devices significantly enhance the reliability of the reactor shutdown function, making the reactor passively safe and ensuring its self-protection against, some or all beyond design basis accidents.

It would be expedient to use SASS devices in advanced reactor designs.

Reactivity feedback amplification is an effective means of increasing reactor self-protection in severe accidents. A thorough analysis of reactor stability is necessary in this case.

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PRESENTATIONS ON PASSIVE SAFETY SYSTEMS/COMPONENTS

(SESSION III)

REACTIVITY CONTROL IN HTR POWER PLANTS WITH RESPECT TO PASSIVE SAFETY SYSTEMS

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(Summary)

The R & D and Demonstration of the High Temperature Reactor (HTR) is described in overview. The HTR-MODULE power plant, as the most advanced concept, is taken for the description of the reactivity control in general. The idea of the "modularization of the core" of the HTR has been developed as the answer on the experiences of the core melt accident at Three Miles Island. The HTR module has two shutdown systems: The "6 rods"-system for hot shutdown and the "18 small absorber pebbles units"-system for cold shutdown. With respect to the definition of "Passive Systems" of IAEA-TECDOC-626 the total reactivity control system of the HTR-MODULE is a passive system of catagory D, because it is an emergency reactor shutdown system based on gravity driven rods - and devices -, activated by fail-safe trip logic. But reactivity control of the HTR does not only consist of these engineered safety system but does have a self-acting stabilization by the negative temperature coefficient of the reactivity, being rather effective in reactivity control. Examples from computer calculations are presented, and - in addition - experimental results from the "Stuck Rod Experiment" at the AVR reactor in Jülich. On the basis of this the proposal is made that "self-acting stabilization as a quality of the function" should be discussed as a new catagory in addition to the active and passive engineered safety systems, structures and components of IAEA-TECDOC-626. The requirements for a future "catastophe-free" nuclear technology are presented. In the appendix the 7th amendment of the atomic energy act of the Federal Republic of Germany, effective 28 July 94, is given.

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CANDU PASSIVE SHUTDOWN SYSTEMS

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Abstract

CANDU incorporates two diverse, passive shutdown systems (Shutdown System No. 1 and Shutdown System No. 2) which are independent of each other and from the reactor regulating system. Both shutdown systems function in the low pressure, low temperature, moderator which surrounds the fuel channels; the shutdown systems do not penetrate the heat transport system pressure boundary.

The shutdown systems are functionally different, physically separate, and passive since the driving force for SDS1 is gravity and the driving force for SDS2 is stored energy. The physics of the reactor core itself ensures a degree of passive safety in that the relatively long prompt neutron generation time inherent in the design of CANDU reactors tends to retard power excursions and reduces the speed required for shutdown action, even for large postulated reactivity increases.

All passive systems include a number of active components or initiators. Hence, an important aspect of passive systems is the inclusion of fail safe (activated by active component failure) operation. The mechanisms that achieve the fail safe action should be passive. Consequently the passive performance of the CANDU shutdown systems extends beyond their basic modes of operation to include fail safe operation based on natural phenomenon or stored energy. For example, loss of power to the SDS1 clutches results in the drop of the shutdown rods by gravity, loss of power or instrument air to the injection valves of SDS2 results in valve opening via spring action, and rigorous self checking of logic, data and timing by the shutdown systems computers assures a fail safe reactor trip through the collapse of a fluctuating magnetic field or the discharge of a capacitor. Event statistics from operating CANDU stations indicate a significant decrease in protection system faults that could lead to loss of production and elimination of protection system faults that could lead to loss of protection.

This paper provides a comprehensive description of the passive shutdown systems employed by CANDU.

1.1 OVERVIEW DESCRIPTION

The CANDU reactor is equipped with two physically independent safety shutdown systems (SDS1 and SDS2) that are physically and functionally independent of each other and of the reactor regulating system.

Shutdown System No. 1 (SDS1) consists of mechanical shutdown rods which drop by gravity, enhanced by springaction, into the core (between the columns of fuel channels) when a trip signal de-energizes clutches which hold the shutdown rods out of the core during normal plant operation.

Shutdown System No. 2 (SDS2) injects a concentrated solution of gadolinium nitrate into the low pressure moderator between the rows of fuel channels to quickly render the core subcritical. The injection is initiated by de-energizing fast acting valves, which are held closed during normal plant operation, to pressurize the individual poison tanks associated with each of the injection nozzles with helium.

A computerized monitoring and test system provides the operator with indications of all shutdown system parameters and automates testing. The system prompts the operator, executes the testing, and records the test results. For each shutdown system, trip parameter instrumentation and logic is tested in such a way that the complete system, from variable sensing to final trip action is tested (for example, each shutdown rod tested by partial drop into core). A test is automatically terminated if another trip channel goes into a tripped state. The test frequency assures that the reliability requirement of not more than one shutdown system failure in one thousand demands is satisfied.

1.2 PASSIVE/FAIL-SAFE FEATURES

The shutdown systems are functionally different, physically separate, and passive in that the driving force for SDS1 is gravity and the driving force for SDS2 is stored energy.

Passive features are extensively applied throughout the detailed implementation of the shutdown systems. The design of each component of both systems and the response of the systems to loss of supporting services such as electrical power or pneumatics is subjected to detailed failure modes and effects analysis which is used to ensure passive, fail safe features are inherent in the mechanical, electrical and pneumatic aspects of the design. For example, loss of electric power to the electromagnetic clutches that hold the shut-down rods poised above the core, result in a rod drop of SDS1 shutdown rods. Loss of the instrument air supply or electrical power to some or all of the valves associated with the poison injection for SDS2 will result in fail safe action, achieved by "air-to-close", "spring-to-open" valves. Failure of any critical component of the trip logic or the physical shutdown mechanism also results in fail safe action.

Passive concepts have been embedded at many levels of the detailed design and realized in innovative ways. For example, the patented CANDU, PWdog passive watchdog timer, incorporated into the design of the independent computers that perform the trip detection logic for each of the two shutdown systems (see Appendix I). This device achieves extremely high reliability by combining a very low electronic component count with the use of passive, capacitive stored energy discharge, to ensure that fail safe action continues if extensive computer self checks indicate potential faults. Essentially, if the computer finds no faults in rigorous periodic self checking, it opens and closes digital output contacts . The continuous, opening and closing of these contacts on a rigidly uniform and precise time increment schedule, creates an alternating current magnetic field that in turn ensures energy transfer through a simple transformer, coupled to a capacitor (a transformer coupled RC circuit). The energy in the capacitor maintains closed shutdown contacts. Even a slight deterioration in the consistency of the execution time performance of the computer, or a non compliance in any one of hundreds of trip computer internal and external hardware and software self checks will result in the collapse of the magnetic field, the discharge of the capacitor and a reactor trip. The only conceivable failure mode is the simultaneous welding of the redundant shutdown contacts which initiate the passive systems.

1.3 OPERATIONAL HISTORY

Historically, CANDU was among the first reactors to include fail safe computers in safety systems with the PDCs (Programmable Digital Comparators) used in the CANDU 6 reactors (early 1980s). PDCs were used to implement the trip decision logic for the process trip parameters.

CANDU plants have had excellent operating statistics. The four CANDU 6 stations (Wolsong 1, Embalse, Gentilly 2 and Point Lepreau) have a total of 45 years of operating history without a single unsafe failure reported. All PDC failures have been safe failures which can be contrasted with the experience with the conventional portions of the system where about 1/4 of the failures are potentially unsafe; i.e. temporarily diminish the redundancy of protection, until corrected. This is due largely to the design that employs features such as self-checks, "continuous" testing, hardware watchdog timers, etc., which convert unsafe faults into safe failures (i.e. trip the channel). From the production reliability viewpoint, there have been no spurious reactor trips attributed to protection logic related failures. Table I summarizes the operational data.

This experience has confirmed the original reasons for using fail safe digital logic; that it enhances safety availability (converts unsafe failures into safe ones), and improves production reliability.

Table I

Lepreau I G.S.
Protection System Reliability Data
1982 July 25 to 1987 September 30

	Conventional Equipment	Computers
Potential Production Loss	35	23
Potential Protection Loss	63	0

The detailed descriptions of SDS1 and SDS2 in the following sections provide more insight into the design and it's passive features.

2. SHUTDOWN SYSTEM NO. 1 (SDS1)

2.1 General Arrangement

The first method of quickly terminating reactor operation when certain parameters exceed defined operating limits, is the release of the spring-assisted gravity-drop shutdown rods of SDS1. SDS1 employs a logic system having three independent channels which detect the requirement for reactor trip and de-energize direct current clutches to release the shutdown rods into the moderator region of the reactor core.

The mechanical shutdown rod units, which are part of shutdown system no. 1, rapidly insert neutron-absorbing rods into the core to shut the reactor down. Each mechanical shutdown unit, shown in Figure 1, comprises a shutdown rod, a vertical guide

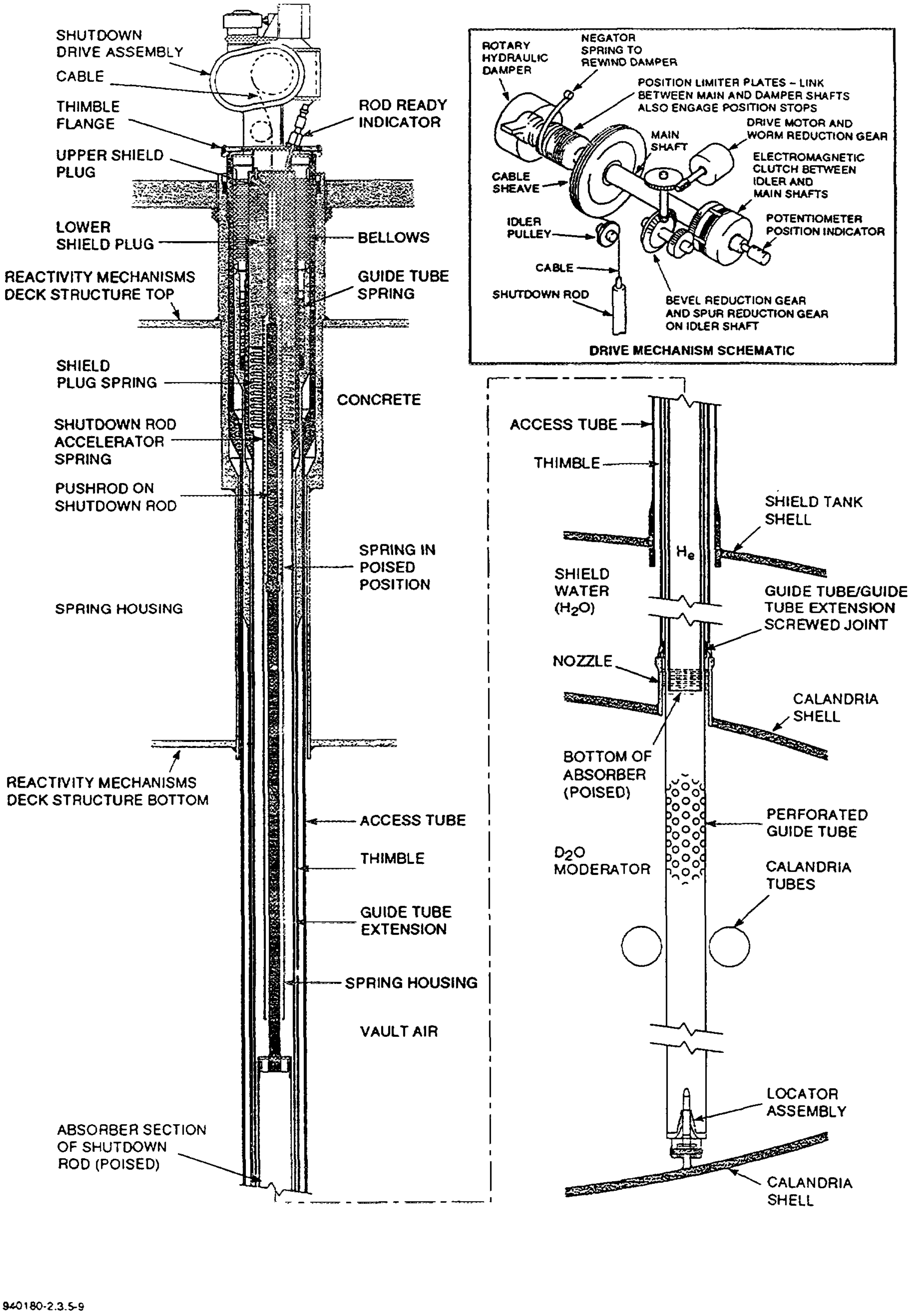

Figure 1 Mechanical Shutdown Rod Unit

tube, and a winch-type drive and release mechanism, along with its shield plugs and accelerator spring assembly. The tubular element section of each shutdown rod is a thin cadmium absorber tube sheathed in stainless steel. It is supported on a tubular push-rod which is suspended from a stainless steel wire rope, that is wound onto the sheave of its drive mechanism. The shutdown rod cable is passed straight through a vertical hole in the shield plug. Radiation streaming is blocked by the push-rod, since the rod is always in the shield plug when the reactor is operating. The sheave is coupled by an electromagnetic friction clutch to its electric motor through a gear train. When the clutch is de-energized by a shutdown trip signal, the sheave is released and the rod falls under the force of gravity. A supplementary initial acceleration is imparted directly to each rod by the compressed accelerator spring when the sheave is released. The fall of the rod is arrested near the end of its travel by hydraulic braking on the shaft within the drive mechanism. After the reason for the trip has been established and when the operator decides to restart the plant, the following occurs: When the clutch is energized, upon clearance of the trip signal, the rod is raised by the motor-driven sheave. The vertical position of each rod is measured by an electrical sensor on the sheave shaft. The travel of the rod is physically limited by dog-plate mechanisms and shaft rotation end stops inside the drive. A second position sensor, the "rod ready" indicator, directly monitors the presence of the rod in the withdrawn position, to verify that it is ready for use. Magnetic reed switches mounted in the shield plug sense a permanent magnet mounted in the top of the rod. Consistent with the CANDU safety philosophy of separating safety and regulating functions, the clutch is part of the safety system. The winch motor used for shutdown rod withdrawal is controlled by the regulating system, but cannot engage the shaft to withdraw the rod when the clutch de-energized.

The design is based on triplicated measurements of each variable, with protective action initiated when any two of the three trip channels are tripped. A single loop component or power supply failure will not incapacitate or spuriously invoke the operation of the safety system. As indicated in Table II, there are nine types of measured variables which can initiate a reactor trip through SDS1. The selection of variables ensures that there are redundant sensing parameters for all categories of process failures identified. A SDS1 reactor trip can also be manually initiated by the operator.

Table II

Shutdown System No. 1 Trip Parameters

Item	Trip Parameter	Detector Type
a.	Neutron Power	Vertical In-Core Detectors
b.	Rate Log Neutron Power	Ion Chambers
c.	Heat Transport System Flow	Differential Pressure Transmitters
d.	Heat Transport System Pressure	Pressure Transmitters
e.	Reactor Building Pressure	Differential Pressure Transmitters
f.	Pressurizer Level	Differential Pressure Transmitters
g.	Steam Generator Level	Differential Pressure Transmitters on each Steam Generator
h.	Steam Generator Feedline Pressure	Pressure Transmitters on Individual Feedlines
i.	Moderator Level	Differential Pressure Transmitters

There are three independent channels, D, E and F, having completely independent and physically separated power supplies, trip parameter sensors, instrumentation, trip computers, and annunciation. SDS1 uses general coincidence voting logic; i.e. the shutdown rods are dropped when any two of the three channels trip, regardless of the parameters causing the channel trips. A simplified block diagram of one channel is shown in Figure 2. The trip system is monitored to provide a positive indication of the state of the trip logic, by verifying the correct operation of all contacts when each channel is tested. The shutdown rods are divided into two banks: each bank is supplied with dual 90 volt dc power supplies for the clutches. Each clutch coil is held energized by the contacts of a separate relay. The high volt-ampere rating of the clutch dictates this arrangement to ensure no relay contact overrating. For each variable monitored, a test capability is provided by which a trip condition is simulated establishing that the channel and parameter trip logic function as designed. The testing frequency is determined based on the target unavailabilities for each parameter.

2.2 Instrumentation Power Supplies

2.2.1 Displays

All the information required on the tripping parameters and the status and operation of the system can be displayed on CRTs in the main control room and the secondary control area, at the operator's command.

2.2.2 Power Supplies

Separately channelled to Group 2A, Class I and Class II power supplies are connected to each of the SDS1 channels. The logic and instrumentation have been designed so that a channel trips on loss of power. Fuse failure or loss of power to individual transmitters results in a channel trip, and is annunciated.

2.2.3 Annunciation

Annunciation for SDS1 is provided in the secondary control area and in the main control room (using buffered outputs from the secondary control area). The SDS1 control room panel contains window equivalent alarms which indicate the state of trip parameters. When a parameter reaches the trip level, these windows show an alarm state. The parameter and channel trip statuses are connected to the plant display system through a fibre-optic link for annunciation and event sequencing. During upset conditions, the time and the sequence of SDS1 parameters exceeding their limits may be printed out on demand. Helium tank pressure, valve position, poison tank level, helium makeup supply pressure and poison front position are indicated in the main control room and secondary control room. The quick-opening valve limit switches are also used to monitor the valve stroking time during channel test.

3. SHUTDOWN SYSTEM NO. 2

3.1 General Arrangement

SDS2 is the second method of quickly terminating reactor operation when certain parameters exceed defined operating limits. Reactor shutdown is via the rapid injection of concentrated gadolinium nitrate solution into the moderator through horizontal nozzle

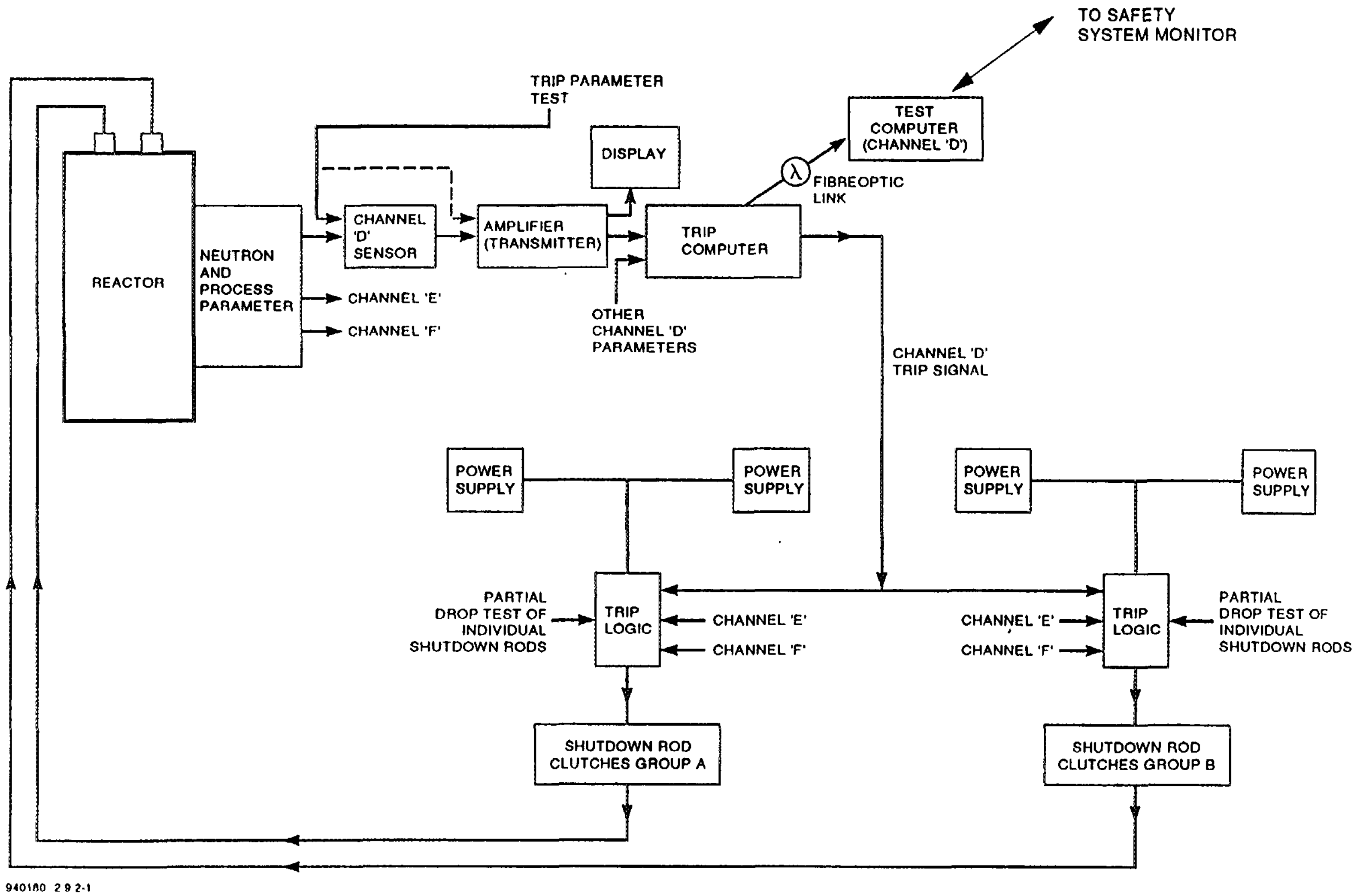

Figure 2 Shutdown System No. 1 – Block Diagram

assemblies. SDS2 employs a logic system having three independent channels which sense the requirement for shutdown and release fast-acting valves which release high pressure helium to inject the gadolinium poison into the moderator.

Figure 3 shows a simplified schematic diagram of the system. A vessel containing high pressure helium supplies is the source of energy for rapid poison injection. The tank is connected, through six quick-opening valves arranged in three successive pairs, to a helium header which services the poison tanks. The quick-opening valves are air-to-close, spring-to-open, so that they fail safe on loss of air supply or electrical power. The cylindrical poison tanks are mounted on the outside of the reactor vault wall. Each of these poison tanks contains gadolinium nitrate solution. The nominal solution concentration is 8000 mg of gadolinium per kg heavy water and is verified by an on-line recirculating sampling system. Each poison tank is connected by a stainless steel pipe to a horizontal in-core injection tube nozzle which spans the calandria between rows of fuel channels, and is immersed in the moderator. The Zircaloy-2 nozzles penetrate the calandria horizontally and at right angles to the fuel channel tubes. Holes are drilled into the nozzle along its length to form four rows of jets which facilitate complete dispersion of the poison into the moderator. There is a liquid-to-liquid interface between the poison solution and the moderator. Movement of the interface is caused by the poison very slowly migrating by diffusion from an area of high concentration to an area of low concentration. Also, physical motion of the liquid back and forth in the line causes a small amount of mixing of the poison solution with the moderator. This motion is caused by variations in the moderator level. The interface movement results in a periodic requirement for back flushing (moving of poison back to design position) approximately twice per year. This is because of the slow diffusion process and because the moderator level is maintained constant during warmup and upgrading by bleed and feed respectively from the D2O supply system. Also moderator temperature is maintained constant during operation. Two conductivity probes are installed in each poison injection line downstream of the poison tank. One is located close to the bottom of the poison tank to monitor the poison concentration and alarm on low poison concentration. The second probe is located close to the shield tank to detect when the poison solution reaches the downstream top of the U-Section. Upon alarm from any of the latter probes, the associated poison injection line must be backflushed to pull the poison interface back to the ball valve or drain line inside the vault. In the backflushing procedure, the affected poison tank is partially drained to the mixing tank with ball valve in the poison injection line closed. This valve is then opened and the interface moves towards the poison tank, refilling it.

The design is based on triplicating the measurement of each variable and initiating protective action when any two of the three trip channels are tripped, regardless of the parameter (i.e., general coincidence logic). A single loop component or power supply failure will not incapacitate or spuriously invoke the operation of the safety system. There are nine measured parameters which can initiate a reactor shutdown through SDS2, as shown in Table III. The selection of parameters is such that there are redundant sensing parameters for all categories of process failures identified. The system can also be manually tripped by the operator from the main control room or the secondary control area.

There are three independent channels, G, H and J. A six valve configuration ensures a very simple and reliable design. Figure 4 shows a simplified block diagram of one channel of SDS2. One manual trip button and parameter test controls for each channel are mounted on the SDS2 control console in the main control room. Interlocks

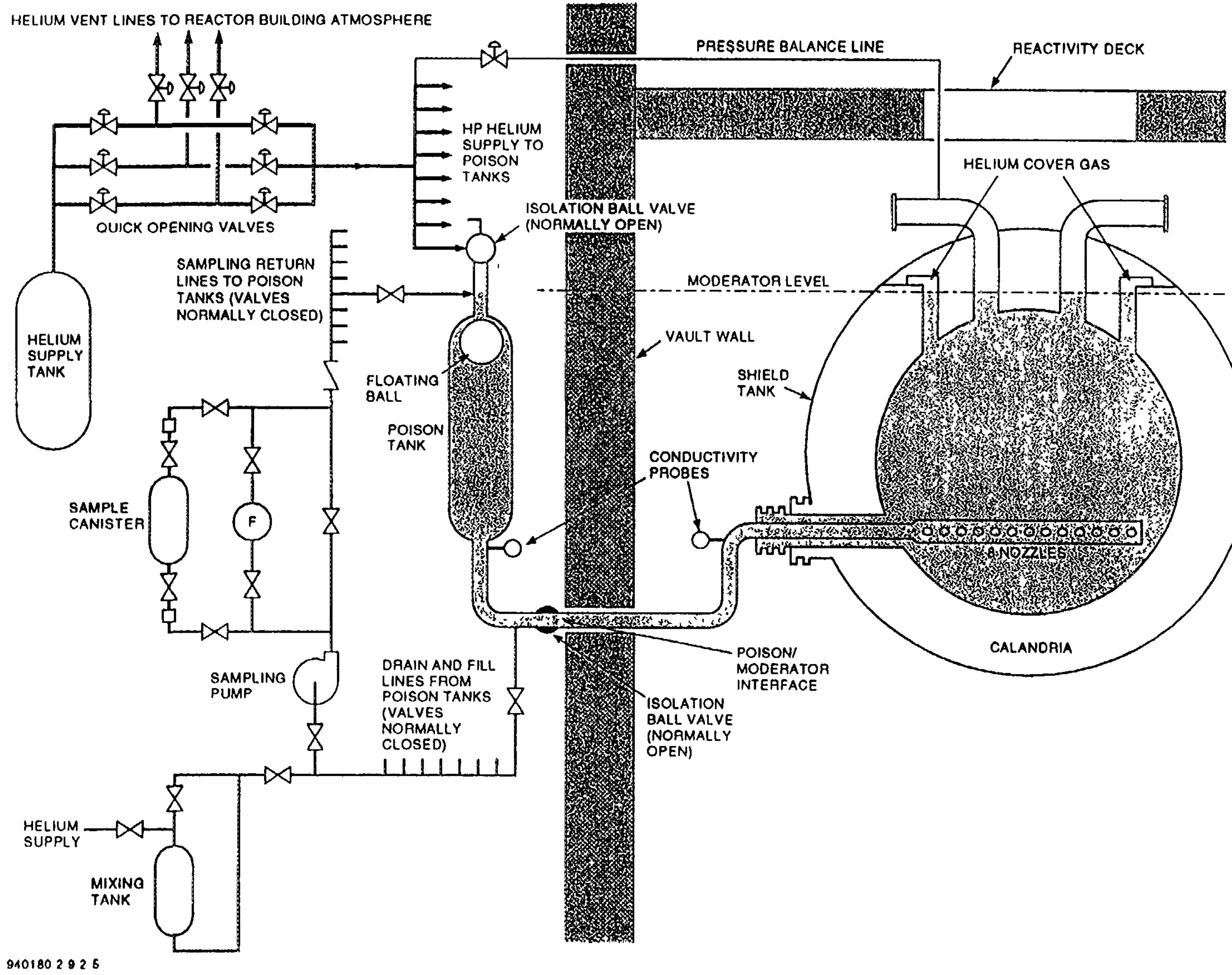

Figure 3 Shutdown System No. 2 Liquid Poison Injection System

are provided so that a channel may not be tested if any other channel is in the tripped state. There are three alternate helium paths, each with injection valves and an interspace vent valve. Special logic prevents the testing of two different channels within ten minutes of each other. This ensures that the interspace between the helium injection valves is depressurized and it prevents spurious, partial injections.

Table III

Shutdown System No. 2 Trip Parameters and Detectors Used

Item	Trip Parameter	Detector Type
a.	Neutron Power	Horizontal In-Core Detectors
b.	Rate Log Neutron Power	Ion Chambers
c.	Heat Transport System Flow	Differential Pressure Transmitter
d.	Heat Transport System Pressure	Pressure Transmitter
e.	Reactor Building Pressure	Differential Pressure Transmitter
f.	Pressurizer Level	Differential Pressure Transmitter
g.	Steam Generator Level	Differential Pressure Transmitter on each Steam Generator
h.	Steam Generator Feedline Pressure	Pressure Transmitter on Individual Feedlines
i.	Moderator Level	Differential Pressure Transmitter

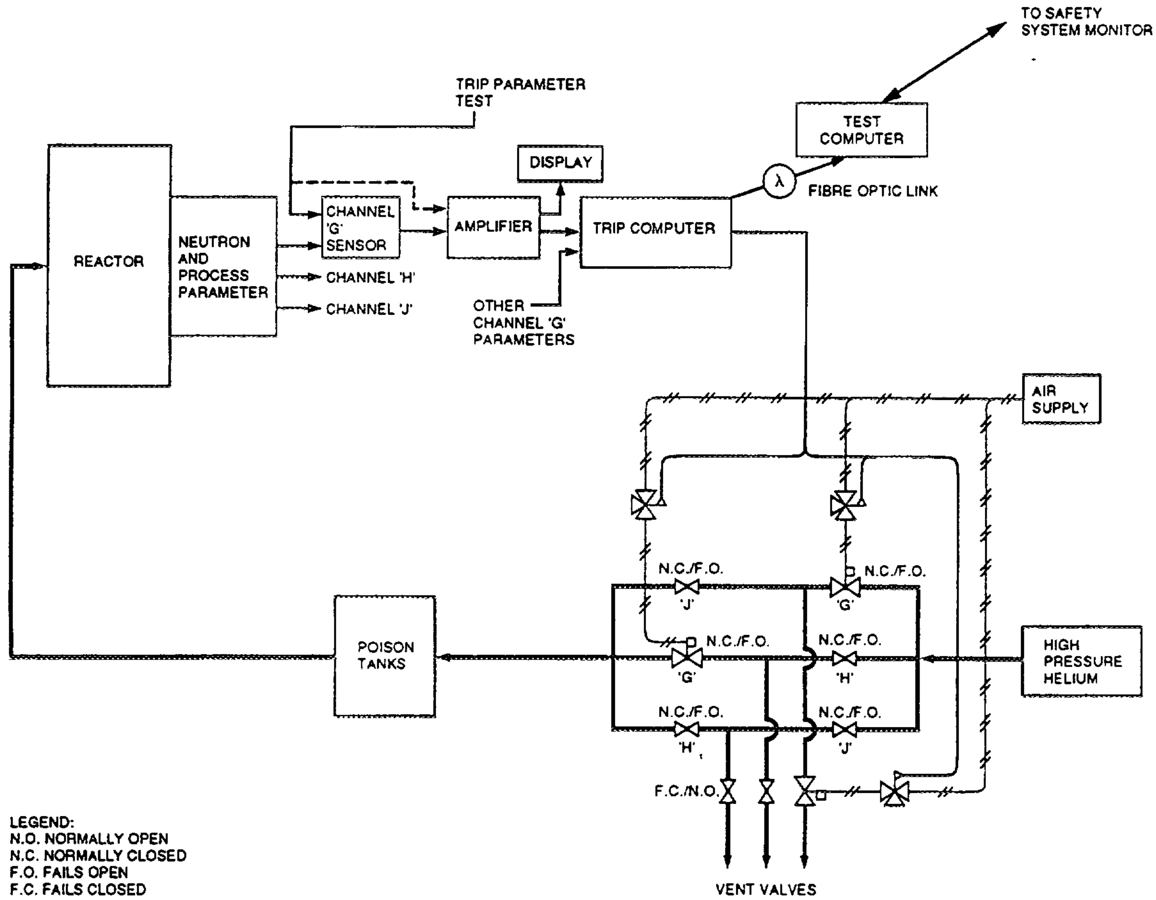

Figure 4 Shutdown System No. 2 – Block Diagram

For each variable monitored, a test capability is provided by which a trip condition is simulated establishing that the channel trip logic for that parameter functions as designed. The complete system operation can be tested with a single input from the sensor to the helium injection valves to mimic the condition of an actual injection.

In addition, trip parameter measurements are tested by a check for insulation resistance for the in-core flux detectors and by a monitoring and channelized cross-comparison process for process pressure measurements. Similar channelized measurements of routine pressure fluctuations are cross-compared using the trip monitoring computers to determine if any one of the transmitters is not responding properly to variation in the measured variables. Testing frequencies are determined on the basis of target unavailabilities for each parameter.

3.2 Instrumentation and Power Supplies

The approach to instrumentation and power supplies for SDS2 is the same as that for SDS1, described in Section 2.2. The instrumentation and power supplies for SDS2 are, however, independent and separate from those for SDS1.

4. SUMMARY

CANDU incorporates two independent, passive and fully capable shutdown systems that function based on passive principles, both in their basic mode of operation and in their fail safe operation. The two shutdown systems, each functionally and physically separate from each other assure that the probability of CANDU not tripping as required is less than 1 in 10^6 demands. The passive operation of the shutdown systems in combination with rigorous on-power testing assures this level of performance.

APPENDIX I

The Watchdog as a Passive Safety Device in CANDU Computerized Shutdown Systems

CANDU reactors use computers to carry out the trip decision logic portion of the shutdown systems. That is, the computers (channelized and triplicated) initiate a reactor shutdown based on the state of a number of trip parameters; such as pressurizer level, heat transport system pressure, steam generator levels, etc. Many fail-safe features are incorporated in these shutdown system computers. One such feature is the "watchdog".

The watchdogs are totally independent devices from the trip computers. There is one watchdog per shutdown system channel. The output from each trip channel's watchdog is a relay contact that sits in the hardwired channel trip voting logic in series with the channel trip contact from the trip computer. That is, the channel will trip if either the trip computer or watchdog contact opens.

The watchdog is a passive device that requires a specific signal from the trip computer to supply power for its output relay. The signal typically is a constant square wave that toggles between 24 or 48 Vdc and 0 Vdc. This device is powered by the signal from the trip computer in such a manner that if the output from the trip computer gets stuck (either high or low) the device becomes unpowered. This is achieved by the use of a transformer.

Referring to Figure 5, the signal from the trip computer is a square wave coming into the primary side of a transformer, shown on the left side of the figure. A transformer requires a signal that is changing (toggling) in order to transfer power from the primary coil (input) to the secondary coil (output). This is a fundamental principle of electromagnetism. Should the input signal cease toggling, even if it remains in the high state (24 or 48 volts) then there will be no output from the secondary side of the transformer. This in turn means that there will be no power to the relay coil which will cause the contact to open. Figure 5 shows the simplest form of such a watchdog. A capacitor is typically added to provide a constant voltage to the relay coil.

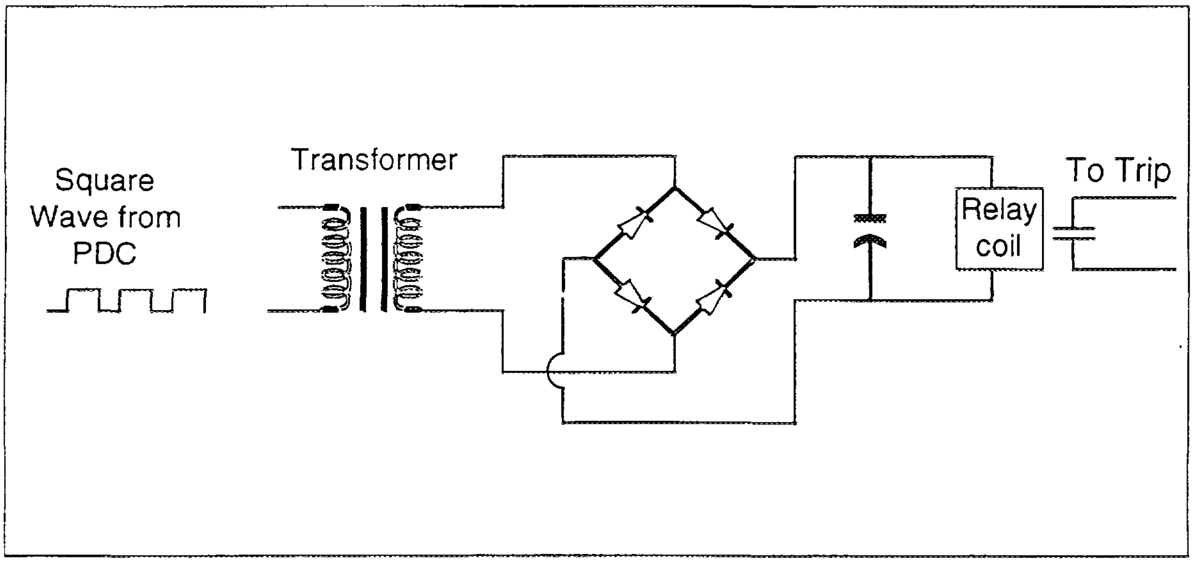

FIGURE 5 CANDU TRIP COMPUTER WATCHDOG

Further refinements to this basic circuit (i.e. a frequency sensitive circuit) have been added such that the watchdog will only accept certain update frequencies. In this way, the watchdog not only detects stuck outputs from the trip computer but also incorrect output frequencies from the trip computer, thereby increasing the types of trip computer failure modes that will result in a fail-safe channel trip.

ACTIVE OR PASSIVE SYSTEMS? THE EPR APPROACH

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Abstract

In attempting to review how EPR is contemplated to meet requirements applicable to future nuclear power plants, the authors indicate where they see the markets and the corresponding unit sizes for the EPR which is a generic key factor for competitiveness. There are no reason in industrialized countries, other than USA (where the investment and amortizing practices under control by Public Utility Commission are quite particular), not to build future plants in the 1000 to 1500 MWe range.

Standardization, which has been actively applied all along the French program and for the Konvoi plants, does not prevent evolution and allows to concentrate large engineering effort in smooth realization of plants and achieve actual construction and commissionning without significant delays.

In order to contribute to public trust renewal, a next generation of power reactors should be fundamentaly less likely to incur serious accidents To reach this goal the best of passive and active systems must be considered without forgetting that the most important source of knowledge is construction and operating experience.

Criteria to assess passive systems investigated for possible implementation in the EPR, such as simplicity of design, impact on plant operation, safety and cost, are discussed.

Exemples of the principal passive systems investigated are described and reasons why they have been dropped after screening through the criteria are given.

INTRODUCTION

Nuclear plant designers throughout the world are addressing the potential inclusion of passive features to meet higher safety standards and reduced costs for future plants. The intent is to simplify designs based upon the current operating knowledge and provide safer, simpler and less expensive designs. These goals are chosen to allow future nuclear plants to remain economically competitive with other power production alternatives.

The designers of future plants must make difficult decisions concerning system designs in order to assure a high level of safety while also addressing public perception of nuclear power. In order to improve public perceptions, the use of a greater number of passive systems in a plant design is encouraged. However, the limited experience and testing of such systems can raise additional questions about the economic operation of such nuclear plants in the future.

The following discussion addresses these issues for the EPR project. The discussion includes expectations of market conditions (including base assumptions on plant size and potential locations of customers) and the assessment of passive design features that have been considered thus far in the EPR design. The discussion is directed at defining the factors which affect future plant designs and then describes the EPR approach to the design choices.

Specific discussion is provided concerning the major passive features considered by EPR and the logic used in assessing their inclusion in the EPR design As is noted throughout the discussion, many of the assessments are qualitative in nature but address the need to find economic alternatives in the future plant designs while maintaining the benefits of current plant operating experience The results of the design process to date have indicated a need to maintain current plant design features, in favor of newer and untested passive design concepts to avoid unnecessary costs and system complications It is expected that as passive system designs mature, many of the decisions made to date will be re-examined as the plant designs continue to evolve

1 MARKET AND UNITS SIZES

Nearly 112 LWR's are operating in Europe (54 in France, 21 in Germany) and another 300 or so have been operating worldwide for tens of thousand of reactor years with only one significant accident That one, TMI-2, demonstrated remarkably how serious an accident could be without affecting public safety The core and the entire plant were badly damaged, yet little radiation escaped the containment

Not withstanding that record, further use of reactors as a base energy resource is moribund in most countries but a very few such as France, Japan and South Korea which would otherwise be extremely, if not totally, dependent on external sources to fulfil their primary energy needs

And yet most responsible officials of industrialized countries believe that considerations of economic strength, environmental protection and security of procurement of energy resources dictate that their countries should still rely on nuclear energy in the next few decades

In attempting to review how Framatome, Siemens and NPI contemplate meeting tomorrow's requirements, it is first necessary to indicate where we see our markets and what will likely be the corresponding unit sizes

1.1 FRENCH AND GERMAN PLANTS

The German utilities started operation of the last 3 PWR's (Konvoi plants) that were ordered in the late 1980's

The French national utility EDF has a program for 1450 MWe plants (called N4) to be put in operation, during the 1990's Four such plants have already been firmly decided Chooz B1 and B2, Civaux 1 and 2 which will enter commercial operation respectively in winter 95/96, 97 and 98

Both the N4 and Konvoi plants introduce a number of new features which makes them so called "evolutionary" reactors, compared to the previous generations of plants It may be that they fail to satisfy some of the minor specifications of the EPRI requirements, but they have a major feature, which consists in being effectively built and at known prices The experience that this gives us is a part of our preparedness for the future

The French and German markets will then be dependent on the expected life of the plants built earlier Extensive studies were made to determine how long their life could be extended , 40 years lifetime is often mentioned, but no formal decision of that sort has yet been accepted No plant has yet reached such a life and a number of units have been deliberately shut down much before

Because of this prudency, with regard to life extension it would not be reasonable to schedule the replacement plants over a short period Therefore, NPI and its Parent Companies plan to be ready for orders in the first decade of the next century, when the cost of fissile materials will likely be much more of a concern than today

Hence, the design of these plants, which is now being initiated through our cooperation with French and German utilities, will have to be made with economic assumptions somewhat different from those now prevailing

The 54 French PWR plants and 12 German PWR plants which will be under operation when, soon, a few old units will have been shut down, are all in the 900 to 1400 MWe range We see no reason why the future plants should not be in that range too

1.2 NON-DOMESTIC PLANTS

As far as the industrialized countries are concerned, we consider that the same determining factors will apply and that they will keep investing in plants of 900 MWe or more Within the frame of our association with Siemens in the NPI Company, this policy obviously applies to the development of the EPR

The export customers of Framatome and Siemens have been eager, in the past, to adopt the technical solutions used in the French or German plants, taken as references The many practical advantages of this, such as cost, construction schedule, licensability, operation and maintenance, will keep applying in the future , the technical choices for the EPR will consequently be made with an eye towards their acceptability by this group of customers This may be achieved even if the domestic and foreign plant sizes are different the important point is to have a unified technology for components, for systems for answers to safety requirements and for maintenance practices

We recognize that certain local conditions tend to call for smaller unit sizes We do plan to try to find acceptable solutions, but are concerned with the ability to remain competitive with fossil energy sources, while there are many uncertainties, in the long term, on the price of fossil fuel and on the intensity with which governments will protect the environment against emissions We have also experienced that, by the time they become prepared to make a decision, potential customers for 600 MWe plants will be ready to invest at the 900 MWe level because of evolving needs

The above remarks on reactor size do not totally apply to the USA, where the situation is quite different The investment amortizing practices and control by PUC's give an incentive to certain utilities to invest in small plants Vendors are under intense pressure to develop concepts allowing competitiveness and to claim that they will succeed Should this turn out to be true, the impact on certain countries outside the USA should not be dismissed This is why Framatome and Siemens evaluated carefully the feasibility of such concepts

2 COMPETITIVENESS AND STANDARDIZATION

Costs considerations are important when considering future nuclear plant purchases against alternative energy sources The element of cost may override any publicly perceived gains in safety and preclude the nuclear option in some cases It would be useless to expend large efforts to improve the safety and public acceptance of nuclear technology, if the resulting designs are so costly that no customer could financially justify the purchase of them A basic strategy of NPI, strongly supported by its Parent Companies, is to contain costs of future PWR plants in order to maintain their viability relative to alternative power sources

When talking of costs, it is again necessary to distinguish between generic problems and those specific to certain countries For instance, in the USA, the high costs experienced in the past result from a licensing environment generating unpredictible delays and requests for modifications In many other countries, like France and Germany the licensing processes, although using similarly severe safety criteria, are such that actual construction and commissionning is achieved without significant delays We will be careful to do whatever is necessary, on the vendor side, to preserve such processes , this will likely include the necessity to design the plants in some detail before construction contracts can be obtained

The unit size is a key factor for competitiveness. Looking at the breakdown of costs for a plant shows that a majority of items follow the iron-rule of increasing costs per kW for decreasing power. Only a few items have a fixed cost per kW : for instance when decrease of power is obtained through the use of smaller numbers of standardized components. In a very limited number of instances may one expect to achieve a function in a cheaper way, by using solutions which would not work for larger power units.

Claims have been made that a number of safety functions may fall in this category when 600 MWe unit sizes are considered. We consider that such claims should be taken with due caution as long as extensive, detailed, designs of plants are not yet completed. It is the intent of NPI to contribute to the clarification of this essential question.

Standardization has been actively applied all along the French program and for the Konvoi plants. We think it should be pursued in the future. As there will be, proportionally, less domestic plants and more export plants than in the past, the requirements of the latter should be taken into consideration while designing the former ; but export customers should also refrain from unecessary "nationalistic" requirements, if they wish to benefit from standardization ; consultants and engineering organizations, when working on the preparation of customers specifications, should remember that they bear specific responsibilities in this respect.

The maximum benefits of standardization will be obtained if it is extended to the whole plant ; when not possible, one should at least apply it to the nuclear island. When differences must exist between plants, for instance in terms of unit power, most of these benefits of standardization will be preserved if one chooses to use existing designs for components, fluid systems, instrumentation and control, and to apply already used lay-out and building principles.

The concept of standardization should in no way be considered as opposed to evolution. Evolution in designs may be necessary for a number of reasons. Standardization is a way to take care of these needs in an organized manner ; in particular, designers will be wise to select solutions which will remain licensable and commercially attractive for the longest possible time. Standardization, then, allows concentration of large engineering efforts in R & D, design, manufacturing practices, maintenance toolings and procedures ; this concentration is the key to economics, smooth realization of plants and scrutiny of safety matters.

3 SAFETY AND PUBLIC ACCEPTANCE

Many people in the nuclear industry feel that our difficulty with respect to public acceptance is primarily the fault of the public, or the media, or the schools, or the anti-nuclear groups.

But we must agree that in a broad sense the public's distrust has its foundations. We said we were designing and building plants in which a core meltdown was essentially impossible ... and then came TMI-2. We then argued that we could have meltdowns but not energetic reactivity accidents, that we might contaminate a power plant but not its neighborhood ... and then came Chernobyl. The public come away doubtful and with a feeling of having been mislead.

In that respect, our specific contribution should be to prepare the future nuclear technology in a rigorous and responsible manner, without announcing objectives which cannot be reached in earnest. Others may have to simplify and to discuss presently fashionable ideas, so as to explain them in an easier way ; their work deserves respect. Ours is to contribute to public trust renewal by demonstration of our professionalism.

If they are to be built, a next generation of power reactors should be reactors with designs that, in both perception and reality, are fundamentally less likely to incur a serious accident.

The utilities would probably like to see in such next generation plants which will operate more reliably, with higher capacity factors, capable to the greatest practicable degree of thwarting maloperation by negligent or imprudent operators, and which in addition, when operating limits are exceeded, have inherent tendencies to return to safe, stable, undamaged conditions without operator intervention or external power sources.

The public and the professionals require that safety matters be reexamined, with an open mind However, we should avoid over-simplification , safety matters are complex and, even if they have to be simplified for public presentation, the professionals, among themselves, must keep a rational, balanced and comprehensive approach It is not a war between passive and active systems , both have virtues and the best of each must be considered The most important source of knowledge is Construction and Operating Experience , this should be the foundation on which to build the future , it has shown us that many other matters must be considered to provide the best overall safety Let us remember the over-emphasis given to large LOCA twenty years ago, preventing a sufficient analysis of Small Break accidents then TMI came

A balanced and comprehensive approach is needed with regard to the extent to which safety functions should be achieved by passive systems There are reasons to think that they will increase the probability that such functions will be achieved However, one should remember that there are limitations to such increases passive systems do have failure modes , frequently, passive systems need an active triggering , they work well only if they are correctly aligned while dormant, etc The coexistence of active and passive systems is also an interesting issue, in terms of cost and of safety balance if one does not want to decrease plant availability, operators, concerned by plant availability, will likely wish to keep their hands on the plant during perturbed situation, and this can be done only through active systems , the management of long-term post-accident situations will probably also require active systems The gain in safety, marginal or significant, can only be determined by extensive and detailed studies , they still have to be done and we intend to provide our share of them

But, we consider that we should also expend significant efforts in drawing lessons from the past The careful analysis of construction problems of operation and maintenance incidents, suggests many precise improvements, the sum of which can bring a significant contribution to safety enhancement, but, of course, only if our future plants design does not depart radically from present ones An area where significant progress is at hand concerns maintenance it is now possible to design a plant in a way which will facilitate its maintainability and hence reduce the probability (which is not negligible today) that safety relevant problems will be induced by maintenance operations or faults
Also, the concept of forgiveness to transients and to operator errors, which must be extended to maintenance errors (in parallel with improved maintainability), is also an important guide in our design efforts of the EPR

Among the other objectives we pursue, are the reduction (by design) of the personnel exposure, the simplification of systems and the effort to design a no-release reactor containment

Safety improvements can be achieved through all these means and we are careful not to weaken these by wild innovations which will induce "youth problems" for the future plants, with devastating impacts on public acceptance We think the detailed and highly professional work is what is expected from us in that respect, we view the vast engineering potential available to NPI as a significant contributor to safety enhancement

4 POTENTIAL PURPOSE OF PASSIVE SYSTEMS

According to their proponents, the main reasons for considering passive features in system design are to achieve simple designs and to improve safety or the confidence that adequate safety is ensured, thus enhancing public acceptance For plant design, the "defense in depth" approach will be maintained and reasons for introducing passive features will be examined at each level of defense in depth

- **First level : High quality in design, construction, operation**

  There is no direct impact due to adopting passive design The use of plant features and designs that ensure increased margins is our prefered approach to achieve smoother plant response and greater allowed time delays for safety system operation

- **Second level : Limit operating disturbances through proper design of plant response**

  No major use of passive features is made or anticipated at this level. In certain cases, however, passive features introduced at third or fourth level may also be relied upon at the second level (e.g. decay heat removal).

- **Third level : Provide engineered safety features to mitigate accident consequences**

  Passive safety features can be used to replace or reinforce safety functions, introduce diversity, simplify system design and/or reduce redundancy in active systems.

- **Fourth level : Severe accidents (beyond level 3)**

  Passive safety systems are introduced to reduce the risk of a major release by preventing core melt, mitigation of core melt consequences or preventing basemat meltthrough.

  The question for the design of the EPR is : can the use of passive features be applied without losing the safety advantages of presently operated PWR's ?

5 INVESTIGATION OF PASSIVE FEATURES

5.1 DEFINITION OF "PASSIVE"

The IAEA definition of a passive component is :

"a component which does not need any external input to operate. It may experience a change in pressure, temperature, radiation, fluid level and flow in performing its function. The function is achieved by means of static or dormant unpowered or selfacting means".

The associated definition of a passive system is :

"a system which is composed of passive components and structures" (1) .

The so-called "passive reactors" being developed are not strictly based on this definition, but more on a definition proposed by EPRI (2) :

"passive system : systems which employ primarily passive means (i.e. natural circulation, gravity, stored energy) for essential safety functions - contrasted with active systems. Use of active components is limited to valves, controls and instrumentation"

Therefore passive features according to the EPRI definition were included in the review at the beginning of the EPR conceptual phase.

5.2 CRITERIA TO ASSESS PASSIVE SYSTEMS

Passive features or systems should be subject to a systematic assessment with respect to the criteria of simplicity of design, impact on plant operation, safety, cost. The first two assessment criteria concern, in more general terms, simplicity.

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(1) IAEA 622-I3-TC-633 "Description of passive safety-related terms

(2) EPRI ALWR Requirements document

Firstly, the design should be simplified, or at least not complicated by the implementation of passive features In this context, proven technology of the components employed is requested Furthermore, the degree of passivity shall be investigated where does a proposed solution rely on active equipment like valves or on active auxiliary systems like cooling or ventilation ? And the overall system configuration shall also be simplified If possible, an active system should be removed, or at least simplified by the implementation of a passive system In addition, the overall system configuration should be simplified Indicators for that could be for instance the necessity of system interconnections

Secondly, the operation of the plant and of the passive system should be simple Normal operational modes like power operation startup, shutdown, refuelling, and maintenance should not be affected by a passive system Spurious actuation of passive systems would have to be investigated, as well as the possibility to detect it and to take straight-forward recovery actions to avoid undue consequences on overall plant operation The operation of the passive system itself should also be simple this includes initiation which should be based on plant status and not on a perhaps difficult diagnosis of an accident scenario, as well as system operation (e g need for adjustment of operational modes as a function of plant status or operating situation should be avoided)

As a rule, passive features to be implemented should be inspectable and have in-service testing capability with the testing mode as close as possible to the operational mode of the system

The last two assessment criteria concern safety and cost As already mentioned, the implementation of a passive system should allow clear safety and economic advantages

New accident scenarios should not be introduced by passive systems This should fit with the well-proven defense in-depth concept and allow for a gradual response to incidents or accidents The incident consequences should not be aggravated by the system operation Furthermore, the multi-barrier concept (strong reactor coolant pressure boundary, control of containment leakages by double containment) presently existing in French and German PWR should not be weakened by the introduction of passive systems

5.3 PASSIVE FEATURES INVESTIGATED FOR THE EPR

The idea of performing safety functions by passive means is not new All existing PWRs employ successfully passive features like accumulators, gravity-driven control rod insertion or natural circulation in the primary circuit Besides these, more passive features have been included in the EPR such as

- larger SG and pressurizer volumes to slow plant response to upset conditions ,
- initial SIS line-up (suction from IRWST and discharge to hot and cold legs) fits long term cooling needs without realignment ,
- simultaneous hot and cold leg low pressure safety injection to limit fuel failure risk in case of large break LOCA ,
- lower core elevation relative to the cold leg cross over piping which limits core uncovery during small break LOCAs ,
- absence of lower head penetration on the RPV for in core instrumentation, thus eliminating one potential failure mechanism and failure location ,
- passive pressurizer safety valves for both overpressure protection and prevention of spurious opening (passive opening under pressure increase, passive closing under pressure decrease) ,
- a large dedicated spreading area outside the reactor cavity to prevent the molten core-concrete interaction by spreading and subsequent flooding of the corium

- a large water source in the IRWST located inside the reactor building, draining by gravity into the reactor cavity and the corium spreading area ;
- a double wall containment with a reinforced concrete outer wall and a prestressed concrete inner wall and an intermediate space maintained passively under small subatmospheric pressure.

In addition to the above features, about twenty passive features were evaluated at the beginning of the conceptual phase of the EPR. The depth of evaluation of specific features depended on interest for their application. About half were briefly examined and dropped without further evaluation, the others were assessed in more detail. The principal passive features which were investigated for possible implementation in the EPR are given in Figure 1. They are briefly described in the subsequent paragraphs.

5.3.1 Passive high pressure residual heat removal system (Figure 2)

The objective of this system is to remove the residual heat for events where existing designs take into account secondary side cooling, so as to replace the emergency feedwater system (EFWS). The primary water flows by natural circulation through the RHR heat exchanger located in an elevated water filled pool. The RHR heat exchanger is cooled by the pool water which evaporates into the containment. A containment cooling system becomes necessary or the pool must be cooled by an active cooling system. Active measures are required such as opening of valves for RHR system flow and start of heat removal from the pool or containment. The main results of the assessment of this system were the following :
Flow rate through the RHR system depends (a) on the elevation between levels of the reactor coolant system (RCS) loops and the RHR heat exchanger and (b) on diameter of RHR pipes.
This concept leads to a significant extension of class 1 equipment.

LIST OF PRINCIPAL PASSIVE SYSTEM APPROACHES INVESTIGATED

PRIMARY-SIDE RHR

1 (HIGH PRESSURE) HEAT EXCHANGER CONNECTED TO PRIMARY SIDE, PASSIVE COOLING

SECONDARY-SIDE RHR

2 CONDENSER CONNECTED TO SECONDARY SIDE, WATER COOLING ON TERTIARY SIDE
3 PASSIVE EMERGENCY FEEDWATER SYSTEM (EFWS)
4 SECONDARY DEPRESSURIZATION AND PASSIVE FEED

PRIMARY-SIDE INJECTION OR MAKE-UP

5 MEDIUM-HEAD SAFETY INJECTION BY ACCUMULATORS
6 GRAVITY-DRIVEN LOW-HEAD SAFETY INJECTION FROM TANK/SUMP BY PRIMARY SYSTEM DEPRESSURIZATION

CONTAINMENT HEAT REMOVAL

7 METAL CONTAINMENT OUTSIDE COOLING (WATER/AIR)
8 SUMP COOLER WITH PASSIVE COOLING CHAIN
9 CONDENSER COOLERS (PASSIVE ON THE CONDENSING SIDE)

FIGURE 1

PRINCIPLE SCHEME OF PASSIVE RHR-SYSTEM

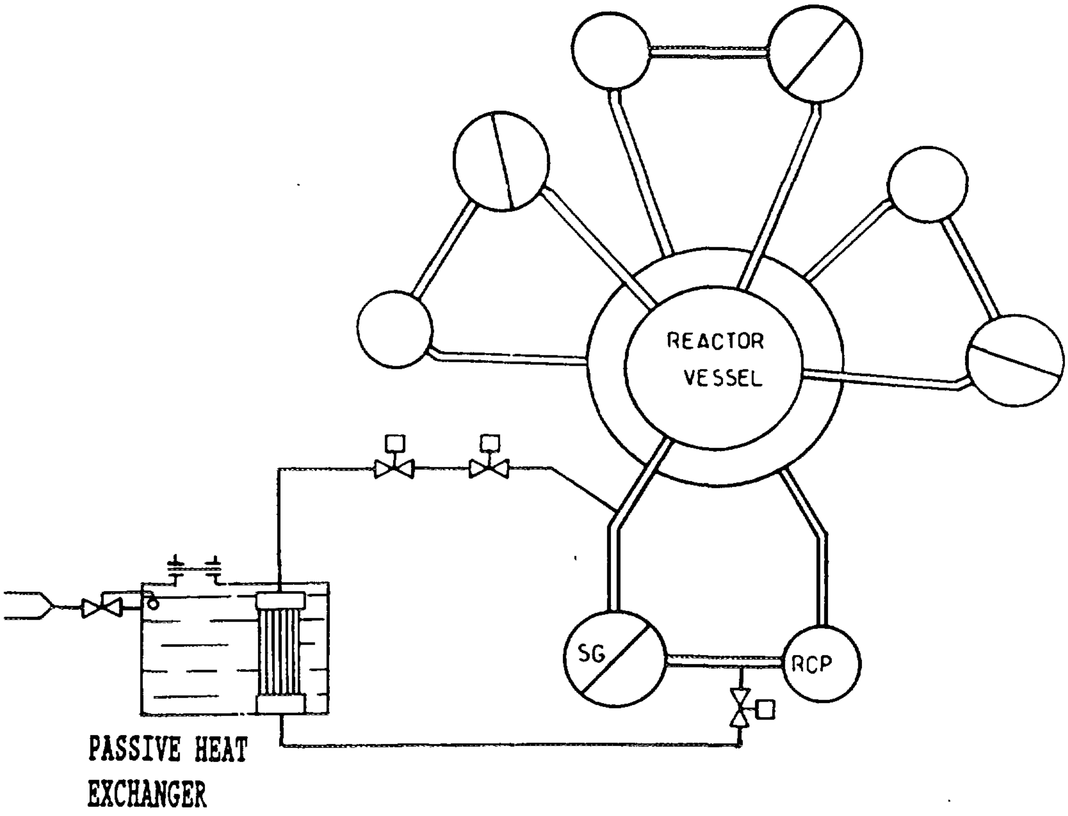

FIGURE 2

Installation of water pool including RHR heat exchanger at about the same level as the operating floor and assuming that more than one train, including pool, would be necessary lead to a complex arrangement of the reactor building.

An operational system would also be necessary to bring the plant to cold shutdown conditions for refueling. Although the passive RHR system presents the potential advantage of easy operation, it was not retained for the EPR because it failed to pass the criteria of design simplicity, safety improvement and cost reduction among the selected criteria.

5.3.2 Safety condenser (Figure 3)

The objective of this concept is to constitute an autonomous, self fed secondary-side residual heat removal system.
The main element of this system is the safety condenser itself, located outside the containment and connected to the steam generator on the steam side and on the water side, and the demineralized water pool, which is connected to the shell side of the safety condenser.

During normal plant operation the system is on standby and is separated from the SG by the closed isolation valves in the condensate line. The valve in the steam supply line is locked in the open position, so that the condenser is full of cold condensate on the tube side and is at main steam pressure. On the shell side, the condenser is partially filled : the closed control/isolation valve prevents the inflow of demineralized water from the demineralized water pool. To start up the system at demand, the redundant, diverse condensate drain valves and the isolation valve in

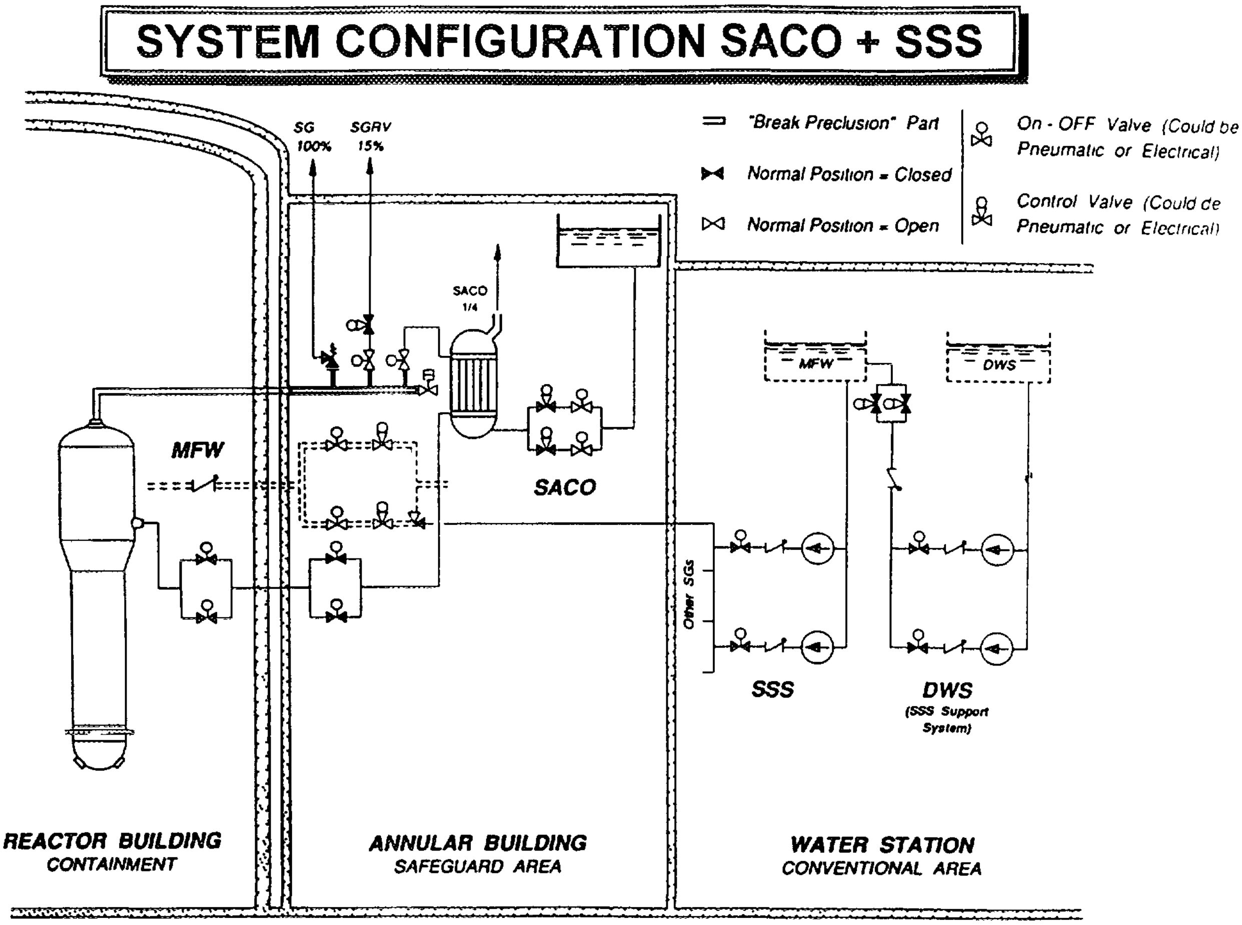

FIGURE 3

the demineralized water supply system are opened and the load controller activated Draining of the secondary side of the condenser exposes heat transfer surface , heat exchange from the steam generator to the condenser takes place when level on the tube side falls below that on the shell side The cold condensate flowing from the condenser into the SG absorbs energy, before heat removal by the condenser actually begins After the system run-up time, which is governed chiefly by the draining characteristic of the condenser, cooling begins This is achieved by the admission, via the redundant, diverse control stations, of demineralized water from the demineralized water pool, which is at a higher static head This results in evaporation to the atmosphere acting as a heat sink

The power supply for the valves required in normal operation and to ensure operation even under emergency conditions is provided by a battery-backed emergency supply bus Since only a small electrical power is required a grace period of several hours is conceivable for restoring the function of any a c generators which may have failed

Although the safety condenser concept presents potential reduction in activity release in the case of a SGTR, this concept was not retained either for the EPR because it failed to pass all the selected criteria Specifically, the system does not meet simplicity and cost criteria

5.3.3 Passive EFWS (Figure4)

The objective of this system is the same as that of the safety condenser The heat exchanger and the demineralized water pool are combined in a single component instead of two separated components

PASSIVE SAFETY CONDENSER

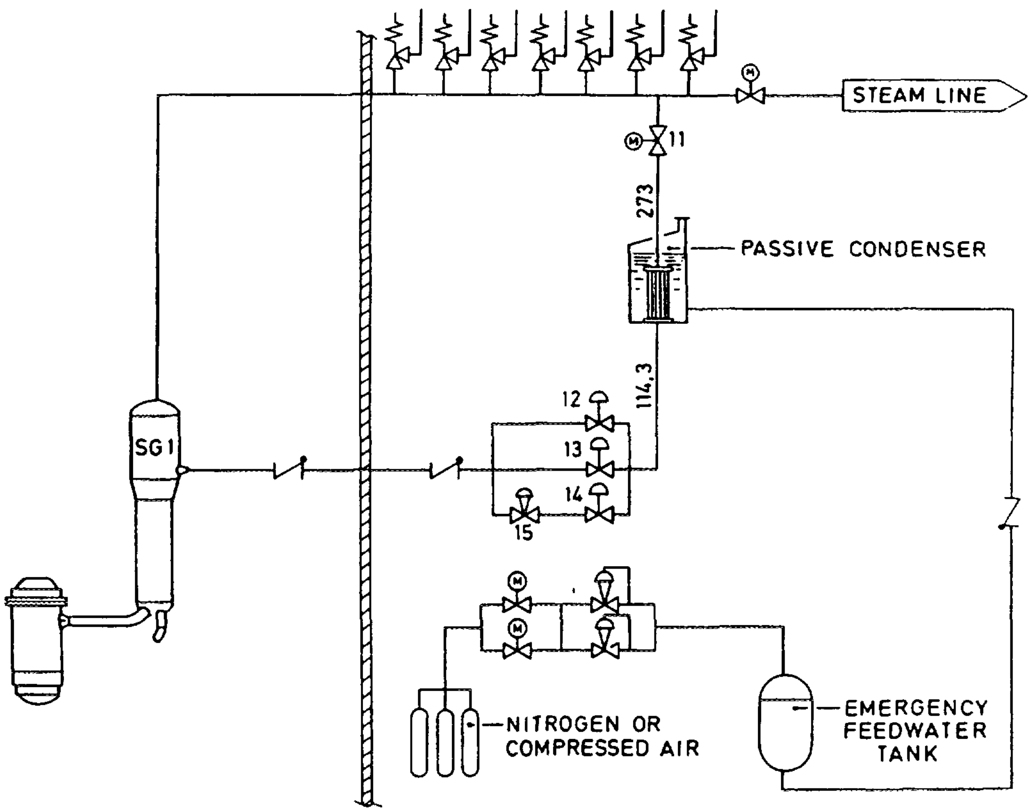

FIGURE 4

In order to avoid elevated storage of large inventory of water an emergency feedwater tank under nitrogen or compressed air located at ground level allows to replenish the passive condenser as and when required, according to water evaporation. This system failed to meet the simplification, operation and cost criteria.

5.3.4 Secondary side residual heat removal and passive feed (Figure 5)

The objective of this system is to remove the residual heat from the core for events such as station blackout and complete loss of feedwater by providing a passive mean to supply water to the steam generators (SG).

Elevated demineralized water pools, large enough to supply the SGs during several hours (station blackout duration), provide flow by gravity once the associated control valves have been open and after closure of the SG main steam isolation valves. The cooldown is performed by steam release to the atmosphere through dedicated relief valves.

The main drawback of this concept is the elevated pools, which must be protected against external events, particularly earthquakes. Movements of large inventories of water and the design of their supporting structures are major safety and cost challenges.

This concept was also dropped because it did not pass any of the four categories of evaluation criteria.

RHR SECONDARY SIDE
SECONDARY DEPRESSURIZATION &
PASSIVE FEED

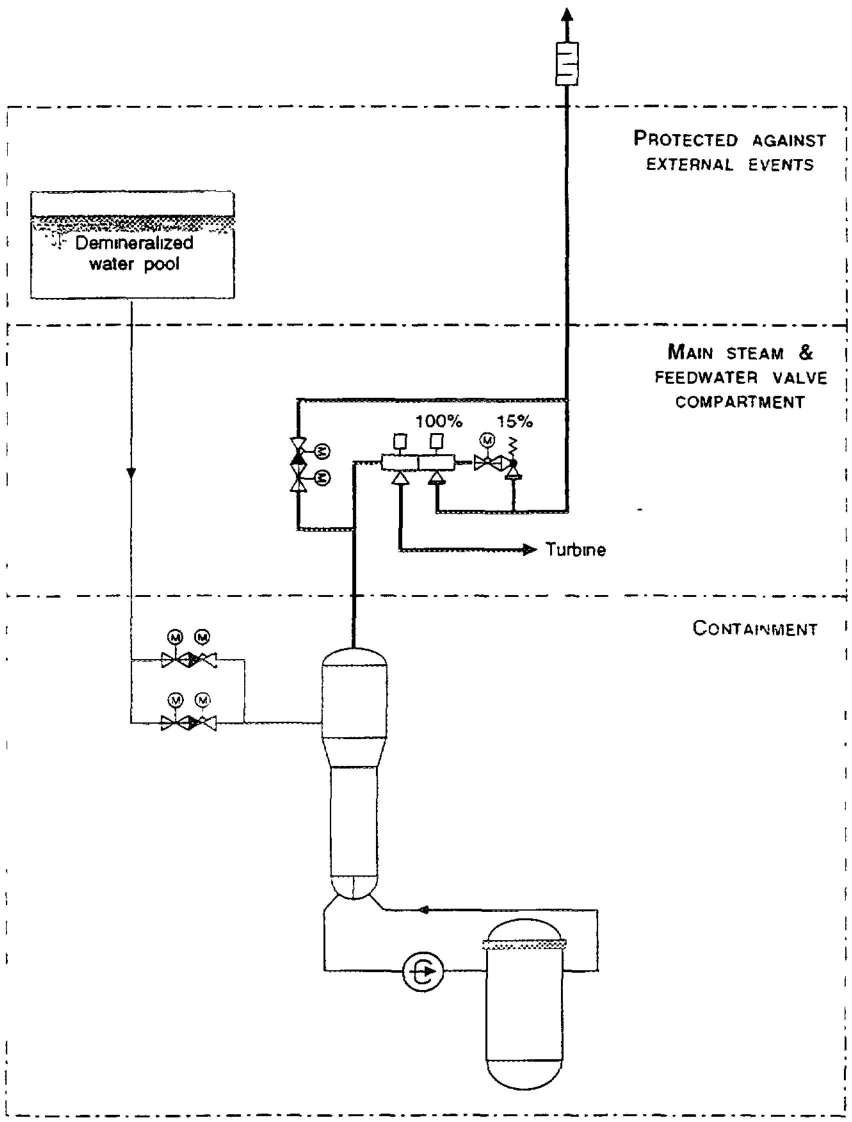

FIGURE 5

5.3.5 Medium-head safety injection by accumulator (Figure 6)

The objective of this system is to simplify the safety injection system (SIS) without reduction of safety level with respect to existing plants. The idea was to delete the medium head safety injection (MHSI) pumps so as to reduce the SIS cost, to require less maintenance and to simplify operation of this system.

In order to fulfill the MHSI functions it is necessary to provide for an automatic depressurization system and high pressure accumulators. Potential difficulties arose during the assessment of this concept. A safety grade boration system appeared to be necessary for non-LOCA events and steam generator tube rupture (SGTR). The management of this accident would have to be reconsidered and questionable operating modes were discovered.
This concept was also dropped because it would finally lead to extra cost with respect to conventional active safety injection systems, thus failing to meet the primary objective

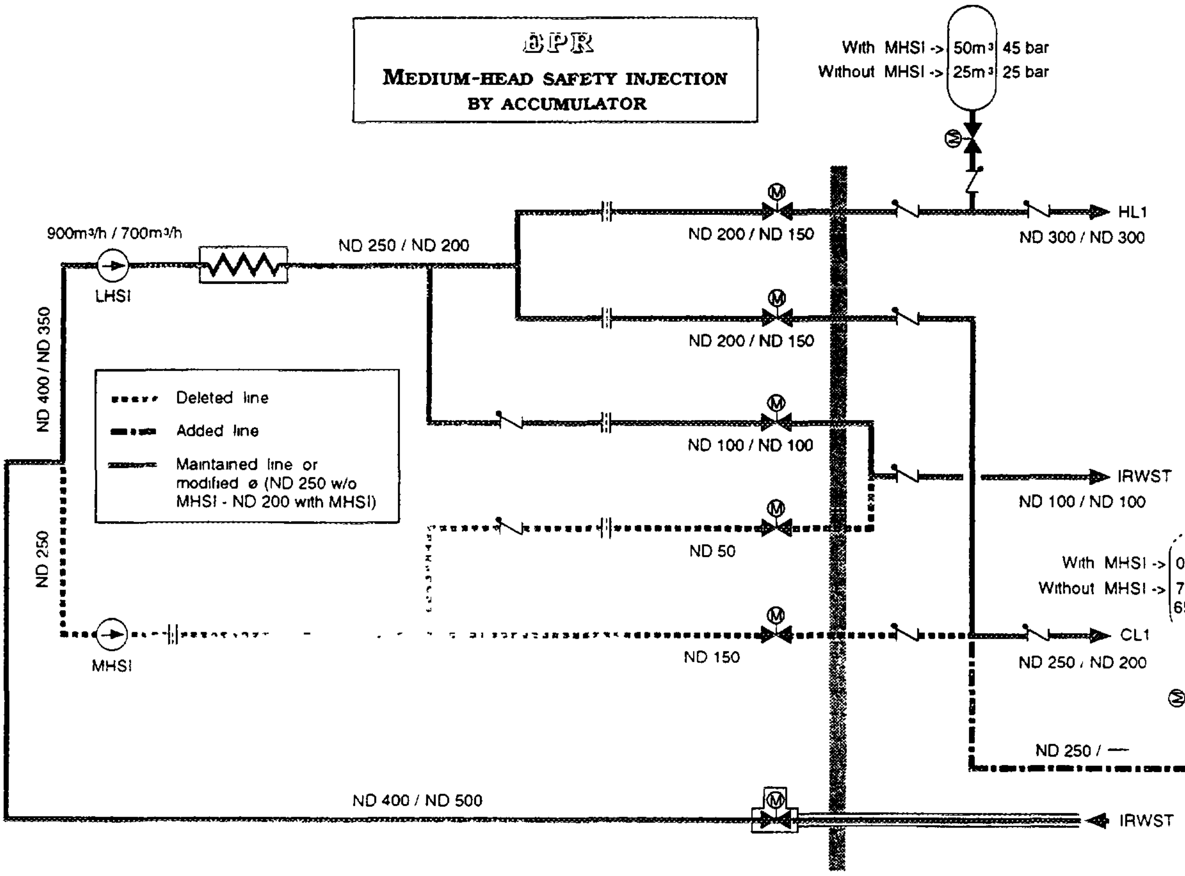

FIGURE 6

5.3.6 Gravity-driven low-head safety injection from tank/sump by primary system depressurization (Figure 7)

The objective of this system is to provide an efficient ultimate back-up for injection of water at low pressure in the long term. The RCS is flooded with water above the loop level and water flows by gravity from sumps, through check valves, into the reactor vessel. The decay heat is removed to the containment atmosphere by evaporation of the flooded water. The steam produced inside the containment condenses on cooled surfaces of a containment cooling system and the condensates flow back to the RCS.

Active measures are required, like other systems . opening of isolation valves, opening of RCS discharge and feed line from the sumps, and start of heat removal system from the containment. The principle results of the assessment of this system were the following :

A large amount of water is necessary to flood the RCS and depends on the reactor building lay out. For the French 4 loops plants with cylindrical prestressed concrete containment, this volume may vary between 4700 m^3 and more than 10000 m^3.

Large diameter of discharge line(s) and small flow resistance check valves are necessary to allow gravity flow to the reactor vessel. Spurious opening of valves in the discharge line(s) would have to be avoided. Additional connections to the RCS are required for discharge line(s) and feed line (s). A full scale test to verify the concept would be extremely costly.

Although the passive low head safety injection system presents advantages, such as providing a back-up to low head safety injection pump and avoiding long term recirculation outside the containment, it was not retained for the EPR because it also failed to pass the criteria of design simplicity, safety improvement and cost reduction among the selected criteria.

BACK-UP OF LOW HEAD
SAFETY INJECTION SYSTEM

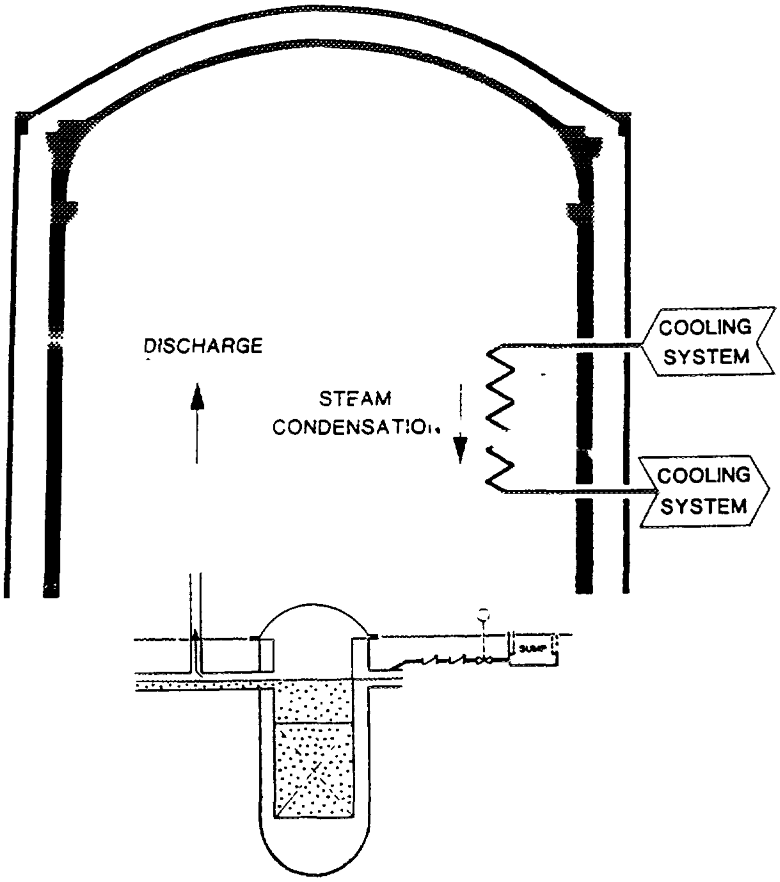

FIGURE 7

5.3.7 Metal containment outside cooling (Figure 8)

For metal containment structures, a concept in which heat removal is ensured by conduction through the containment wall is in principle feasable. Inside containment, heat transfer is by natural convection in the containment atmosphere and condensation on the containment wall inner surface. Outside containment, several alternate cooling schemes can be envisaged. A completely passive concept, using natural circulation air cooling, is only possible for small unit sizes and in the long term, after decay heat is sufficiently reduced. Thus, additional means based on water spray on the outside containment surface is required at least in the short term. For the larger unit size of the EPR, such water-circulation assisted outside cooling is required also in the long term. Use of water cooling outside, without evaporation and based on an active cooling circuit with pump and heat exchanger, then also maintains a double containment barrier, which is not possible in case of a natural air circulation cooling mode. However, in such a containment heat removal concept the passive means of heat removal is provided only inside containment. Furthermore, the heat transfer capacity by condensation on the inner containment surface in the presence of noncondensable gases is limited and, for the larger EPR unit size, not capable of avoiding relatively elevated containment pressure (several bar), for the medium term following an accident.
It is for these reasons, as well as for the fact that the EPR uses a concrete rather than a steel containment concept, that this option has not been retained for EPR.

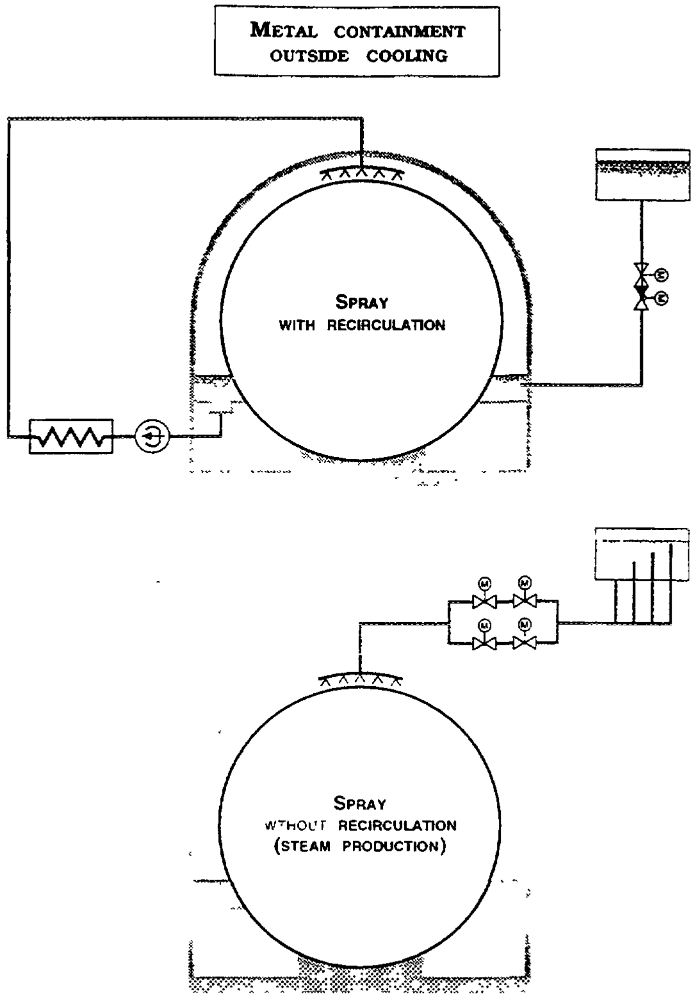

FIGURE 8

5.3.8 Sump cooler with passive cooling chain (Figure 9)

The objective of this concept is to remove the decay heat following a LOCA by natural circulation from the reactor building sump via submerged coolers and a secondary cooling system to the atmosphere.
Like many other passive concepts, opening of valves is necessary to start operation of this system.

Additional measures to transfer the heat from the containment sump were estimated to be necessary during the evaluation of this concept. A large heat transfer suface for sump cooler (a minimum of 1000 m²) was found to be required.

The passive sump cooling feature was dropped because the height difference between the ultimate heat sink and the sump cooler to secure natural circulation (a minimum of 20 m) would lead to unbearable costs.

5.3.9 Containment condenser coolers (Figure 10)

The objective of this system is to provide a passive mean, at least inside the reactor building, to remove the decay heat in the long term after a severe accident, in order to avoid the internal pressure exceeding the containment design pressure.

SUMP COOLER ARRANGEMENT EXAMPLE

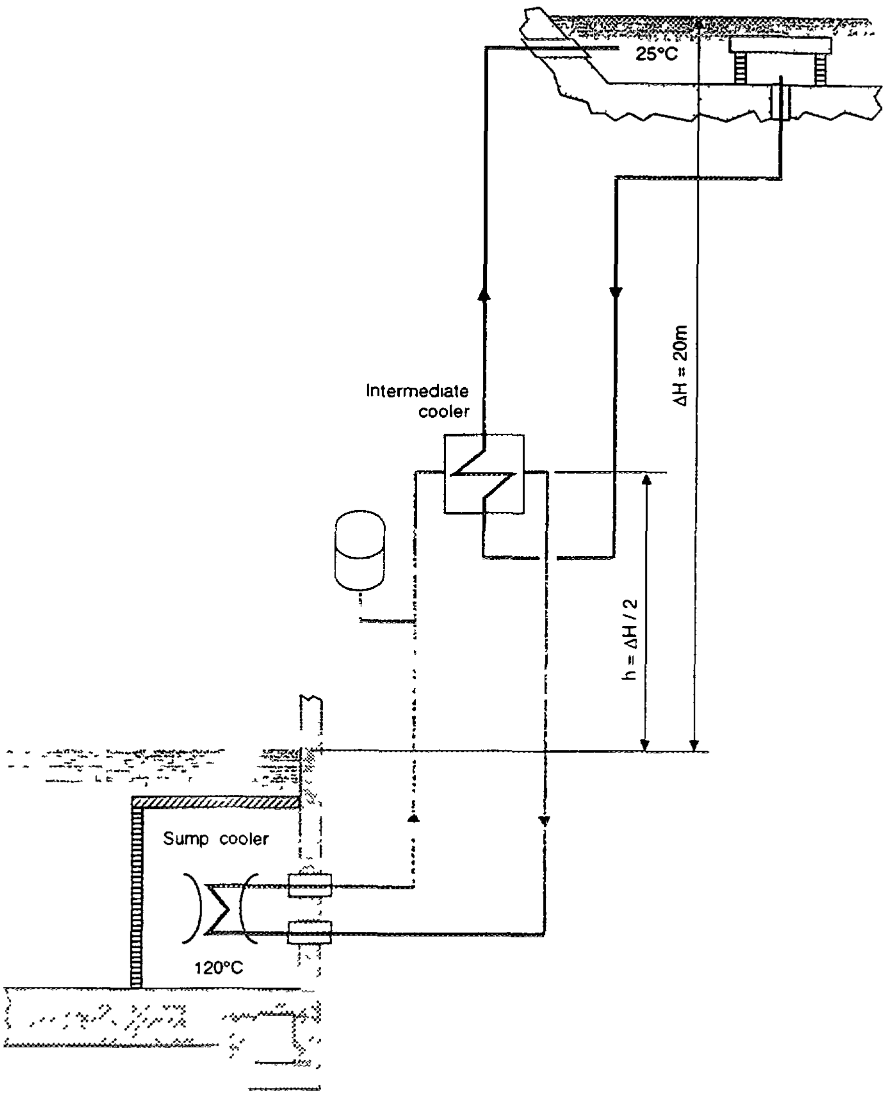

FIGURE 9

Steam, driven by natural circulation, condenses on the outside surface of coolers. Cooling water circulates inside the coolers surface. The cooling system is active and located outside the containment.

The major drawbacks of this system, in comparison to existing spray systems, are the following :

- Heat transfer and containment pressure reduction capability are strongly dependent on the presence of non-condensable gas and on general convection movements inside the containment.

- Condensers must be located in the upper part of the containment where hydrogen is likely to accumulate. They constitute hydrogen traps, thus reducing heat transfer capability and increasing the risk of explosion

- Large room for lay-out is required above operating floor which is a congested area during maintenance and refueling

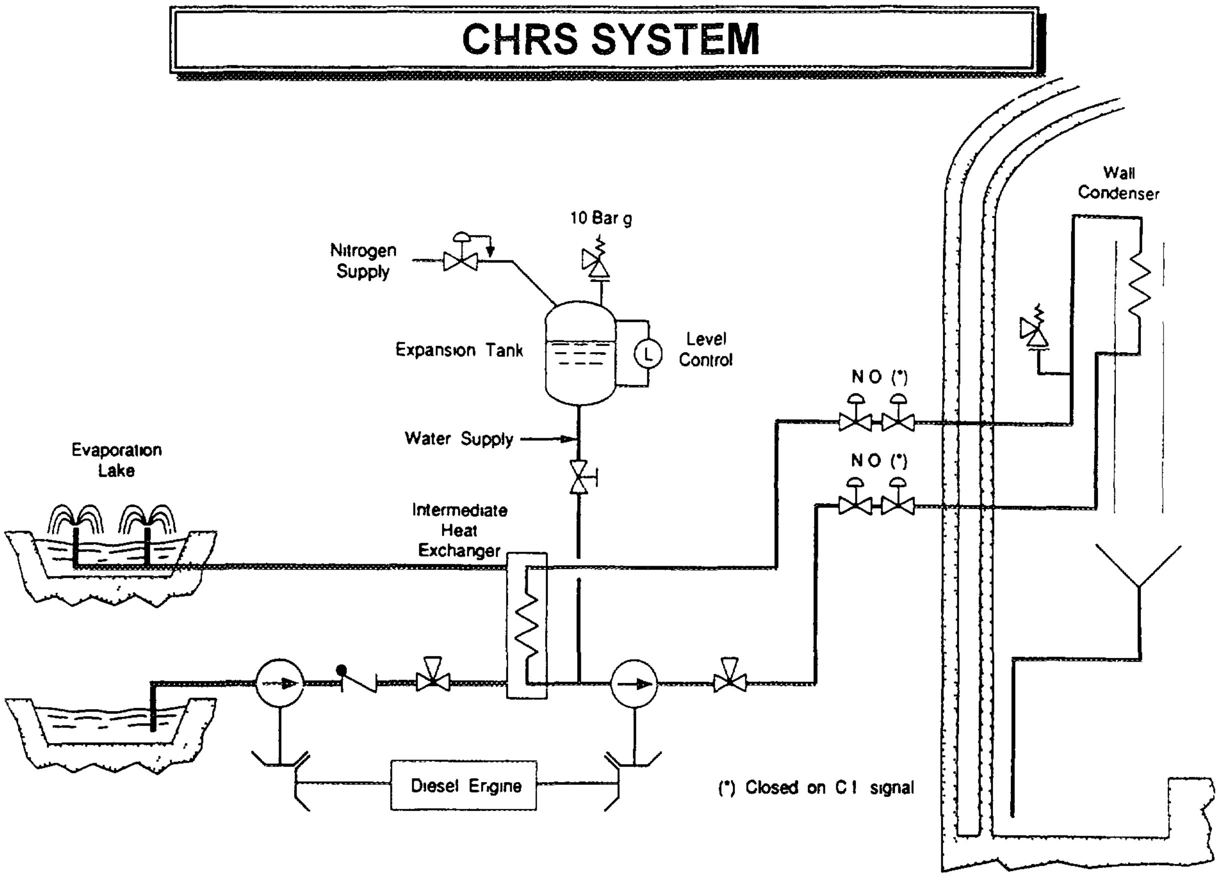

FIGURE 10

- The condensers are of no help in reducing source term outside containment because they have no effect on aerosols and they decrease containment pressure slower than a spray system

However, this system offers several advantages, the major one with respect to a containment spray system being that it avoids recirculation of highly radioactive water outside the containment after a severe accident

The Figure 11 summarizes the assessment done of the principal passive systems listed in Figure 1 All of them were dropped after screening through the criteria mentioned in paragraph 5 2 The containment condenser coolers might be reconsidered to solve one of the severe accident challenge how to remove heat from a building without circulating any fluid through its walls and without impairing its leak tightness ?

Many engineers have come around to the idea that hybrid systems combining active and passive features represent an attractive alternative to existing designs This is supported by the utilities which contribute to the development of the EPR They consider that more important than passive features are simplicity, reliability and less complicated control and automation

CRITERIA FOR EVALUATION OF "PASSIVE SYSTEMS"

CRITERIA FOR EVALUATION	PRIMARY SIDE RHR	SECONDARY SIDE RHR		
	H.P. RHRS	Safety Condenser	Passive EFWS	Second.dep. & feed
DESIGN SIMPLICITY				
System configurations simplified relative to actual solutions ?	No	No	No	Yes
Number of "Active" components within "Passive system"	Nihil	2 valves (1)	1 valve (1)	Nihil
Passive components based on proven technology ?	small experience base	small experience base	Same except design against earthquakes	No
Triggering of system actuation by plant status or operator	Operator	Plant status	Plant status	Plant status
IMPACT ON PLANT OPERATION				
Negative impact on normal plant operation, including refueling and maintenance, avoided ?	No	No	No	Yes
Consequences of a spurious activation ?	Plant trip	Plant trip	Plant trip	No
Is a recovery possible ?	Yes	Yes	Yes	
SAFETY				
Does the solution provide diversity ?	Yes	Yes, if EFWS is kept	Yes, if EFWS is kept	Yes
Reliability trend for the proposed solution ?	Neutral	Neutral	Neutral	Theoretically positive
Is periodic testing possible ?				
- Only during start-up shutdown	Yes	Yes	Yes	Partially
- During all operating modes	No	No	No	No
Are new accident paths avoided ?	No	Not completely	Not completely	Yes
COSTS				
Development costs for design, verification, computer codes test facilities	High	Medium (design against earthquakes)	Medium (design against earthquakes)	Low/ medium
Overall cost estimate	Increased	Increased	Increased	Moderate
Cost benefits due to savings in systems replaced or simplified by the proposed solution ?	EFWS savings	EFWS eliminated	EFWS savings	EFWS savings

(1) Control valves (2) 4 LHSI pumps + 2 boration pumps

FIGURE 11

CRITERIA FOR EVALUATION OF "PASSIVE SYSTEMS"

CRITERIA FOR EVALUATION	PRIMARY SIDE INJECTION OR MAKE-UP		CONTAINMENT HEAT REMOVAL		
	MHSI by accumulat.	Gravity LHSI	Metal cont. cooling	Sump coolers	Cont. wall condensers
DESIGN SIMPLICITY System configurations simplified relative to actual solutions ?	No	Yes	No	No	No
Number of "Active" components within "Passive system"	6 pumps (2)	Nihil	Nihil only for small power units	Nihil only for small power units	at least 2 pumps
Passive components based on proven technology ?	Yes	No	No	Yes	No
Triggering of system actuation by plant status or operator	Plant status	Plant status	Plant status	Operator	Operator
IMPACT ON PLANT OPERATION Negative impact on normal plant operation, including refueling and maintenance, avoided ?	No	Yes	Yes	Yes	No
Consequences of a spurious activation ? Is a recovery possible ?	Plant trip + SI Yes	No No	No No	No Yes	No Yes
SAFETY Does the solution provide diversity ?	Yes	Yes	Yes	Yes	Yes
Reliability trend for the proposed solution ?	<Active SIS	Neutral	Positive	Positive	< CSS
Is periodic testing possible ? - Only during start-up shutdown - During all operating modes	 Yes No	 Partially No	 Partially No	 Partially No	 Partially No
Are new accident paths avoided ?	No	Yes	Yes	Yes	No
COSTS Development costs for design, verification, computer codes test facilities	Small/ medium	Low	High	Low	Medium
Overall cost estimate	Increased	High	Not feasable for large power units	Not feasable for large power units	Increased
Cost benefits due to savings in systems replaced or simplified by the proposed solution ?	SIS savings	Small	No	No	R B coolers eventual savings

(1) Control valves (2) 4 LHSI pumps + 2 boration pumps

FIGURE 11 (cont.)

6 CONCLUSION

The environment surrounding the nuclear power industry is changing, both due to economic factors and greater impact of public perception on nuclear plant designs throughout the world. it has become imperative to address both the technical and public perception issues now more than ever before.

The design process begins by defining the intended plant size for future plants. Current experience in France, Germany, and the rest of the world indicate that the new plant sizes, at least for the foreseeable future, will be large, in the 900 MWe and above range. This favoring of larger plants sizes is based on proven operating experience and the economic advantages of size, using available proven technology.

The EPR is being developed to address all issues of safety, public perceptions and economics. The experience to date in both France and Germany on standardization of the plant design is a large factor in the overall EPR design.

Although the current standard designs have performed well, evolution of the design, using better active and passive features add to the overall plant safety and economics. Additionally, the development of a standard plant that is nearly fully designed prior to obtaining a construction contract, allows for the concentration of large engineering efforts in R&D, design, manufacturing practices, maintenance tooling and procedures to meet the market demands for safety, availability and economy.

In general, the majority of passive features considered thus far are still unproven through test or operation. As such, the features remain economically unjustified or actually lead to plant complications that may degrade rather than enhance safety. For these reasons, NPI has not yet embraced a large number of new passive features for use in the EPR. As technology and experience evolves, NPI will continue to pursue both active and passive features that improve plant safety as well as ensure that nuclear power remains cost competitive with alternative power production sources.

DIMENSIONING OF EMERGENCY CONDENSERS IN ACCORDANCE WITH SAFETY REQUIREMENTS

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Abstract

The emergency condensers are heat exchangers consisting of a parallel arrangement of horizontal U-tubes between two common heads. The top header is connected via piping to the reactor vessel steam space, while the lower header is connected to the reactor vessel below the reactor vessel water level. The heat exchangers are located in a pool filled with cold water. The emergency condensers and the reactor vessel thus form a system of communicating pipes. At normal reactor water level, the emergency condensers are flooded with cold, non-flowing water. No heat transfer takes place in this condition. If there is a drop in the reactor water level, the heat exchanging surfaces are gradually uncovered and the incoming steam condenses on the cold surfaces. The cold condensate in returned to the reactor vessel.

In this way, heat is removed from the reactor vessel and water simultaneously supplied to the reactor vessel. This means that the emergency condensers function as a heat removal system while at the same time serving as HP and LP coolant injection systems. The emergency condensers operate with the highest possible degree of passivity imaginable, namely through a drop in the reactor vessel water level alone, requiring neither control systems nor power supply. The design of the emergency condensers must meet the requirements dictated by the thermal and the hydraulic conditions.

Taking into consideration a redundancy degree of N + 2, a specific thermal rating of 63 MW per emergency condenser results for a reactor with an output of 2778 MW. the total performance of the emergency condenser system in thus 252 MW, or 9.1% of reactor output.

The probability of failure of the emergency condenser of Siemens SWR 1000 is approximately 10^{-4} per demand, while that of older emergency condenser designs is approximately 2 to 3 x 10^{-3} per demand.

1 Introduction

The Power Generation Group (KWU) of Siemens AG and the German electrical power utilities - particularly those operating boiling water reactor plants - are together developing a new reactor type which is characterized in particular by its passive safety systems.

These passive safety systems, which have been described in a separate paper on this subject, are the following:

- 4 emergency condensers
- 4 containment cooling condensers

- 8 passive pressure pulse transmitters
- 6 gravity-driven core flooding lines
- 8 rupture disks arranged in parallel to the relief valves
- 2 scram systems

There are a variety of reasons for introducing passive safety systems, the most important of which are the following:

a) Increasing the safety of future nuclear power plants is a prerequisite to increasing public acceptance of nuclear power.

b) Reactor safety systems must be simplified in order to reduce capital costs.

2 Emergency Condensers (figure 1)

Whether accident conditions involve loss of coolant or not, the emergency condensers play a central role in accident control.

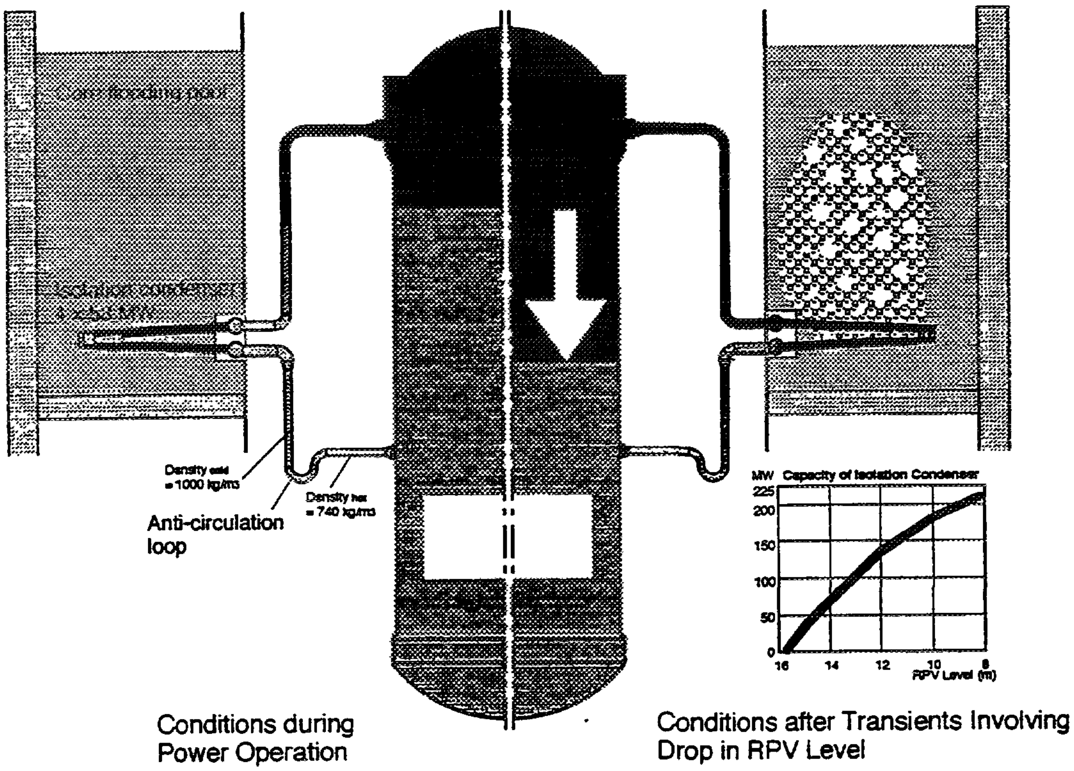

Figure 1
SWR 1000 - Isolation Condenser

The emergency condensers are heat exchangers consisting of a parallel arrangement of horizontal U-tubes between two common heads. The top header is connected via piping to the reactor vessel steam space, while the lower header is connected to the reactor vessel below the reactor vessel water level. The heat exchangers are located in a pool filled with cold water. The emergency condensers and the reactor vessel thus form a system of communicating pipes. At normal reactor water level, the emergency condensers are flooded with cold, non-flowing water. No heat transfer takes place in this condition. If there is a drop in the reactor water level, the heat exchanging surfaces are gradually uncovered and the incoming steam condenses on the cold surfaces. The cold condensate is returned to the reactor vessel.

In this way, heat is removed from the reactor vessel and water simultaneously supplied to the reactor vessel. This means that the emergency condensers function as a heat removal system while at the same time serving as HP and LP coolant injection systems. The emergency condensers operate with the highest possible degree of passivity imaginable, namely through a drop in the reactor vessel water level alone, requiring neither control systems nor power supply.

3 Previous Know-How and Experience with Emergency Condensers

The first generation of boiling water reactors built by General Electric and under licenses from GE were equipped with similar emergency condensers (figure 2).

In Germany, the Gundremmingen A nuclear power plant unit, which began operation in 1966, is provided with a system of this type.

The emergency condenser in this design is a tank filled with water containing two tube bundles, to which the connecting piping from the main steam line is connected. The inlet valves in the supply lines to the tube bundle are always open; the outlet valves are normally closed such that the tube bundles are filled with condensate. In the event that emergency condenser operation is initiated as a result of excessive pressure in the reactor, the outlet valves open automatically and natural circulation is established. The primary steam enters the tube bundles, condenses and is returned by gravity force to the reactor vessel. During the condensation process, the water in the condenser is heated and starts to boil; the resulting steam is discharged from the condenser to the atmosphere. The water inventory is sufficient to last for a period of up to approximately four hours subsequent to reactor scram without makeup supply to the emergency condenser.

The emergency condenser installed at Gundremmingen A has a thermal duty of approximately 48 MW at a reactor pressure of 70 bar and a water temperature in the

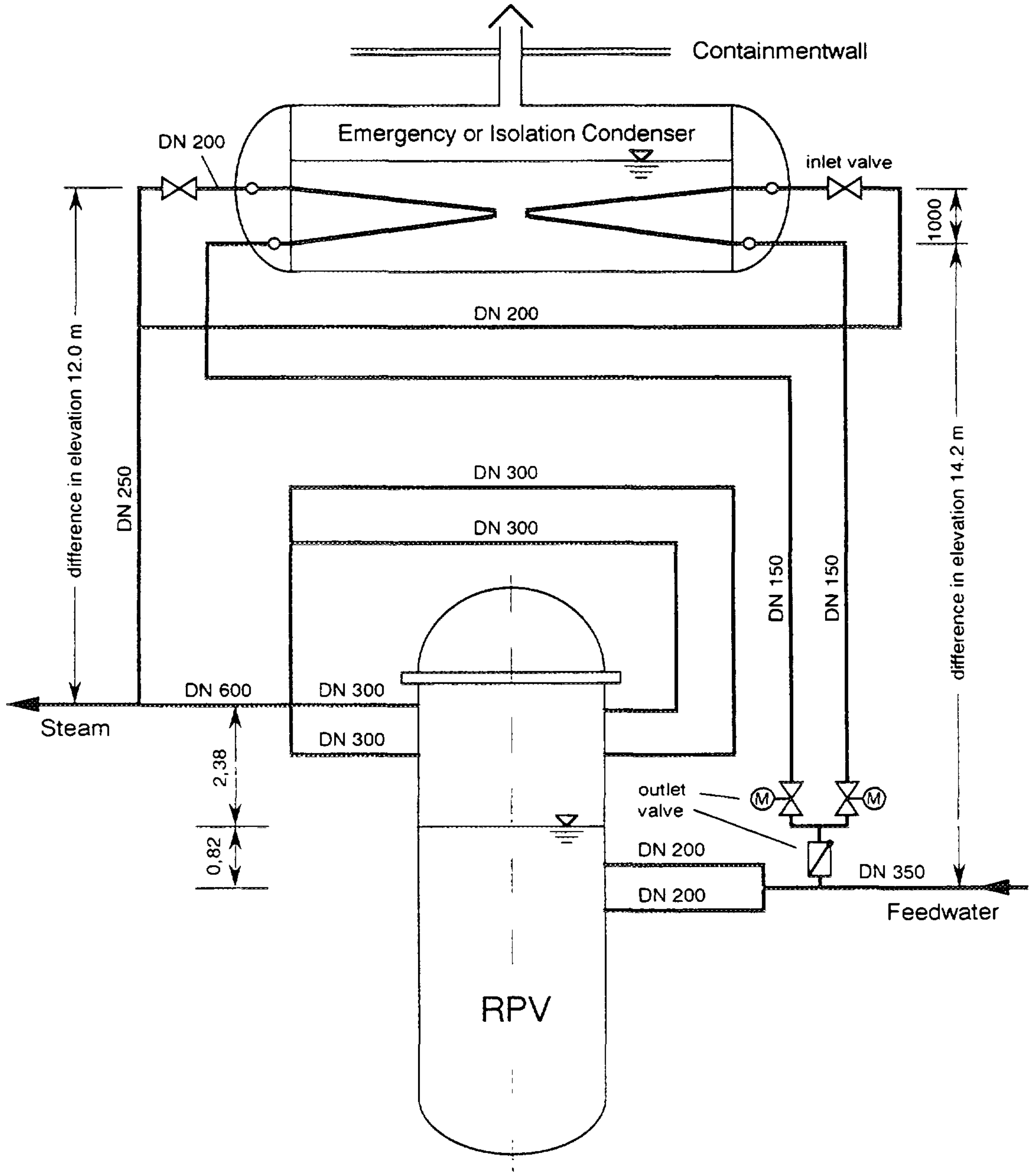

Figure 2
KRB - A Emergency condenser diagram

emergency condenser of 120 °C (corresponding to about 2 bar and saturated conditions).

The elevation differential between the emergency condenser and the steam lines is approximately 12 m.

The functional capability of Gundremmingen A's emergency condenser system was tested and measured several times in the course of the units service life. The attached figure 3 shows the results of testing conducted on May 10, 1975. Here, emergency condenser performance is presented as a function of reactor pressure. Two curves are indicated, one for the cold water (approximately 30 °C) and the second for the boiling water condition (approximately 116 °C) in the emergency condenser. The measuring points are identified with numbers which indicate the measuring sequence. At the beginning (measuring point 1), cold water is in the emergency condenser; the water heats up in the course of time until the boiling temperature is finally reached. The design point is also given for the purpose of comparison.

The heat exchangers employed in our new emergency condensers are identical to those used at Gundremmingen A, i.e. a well known component with proven operating experience.

4 Design of Emergency Condensers for SWR 1000

The design of the emergency condensers must meet the requirements dictated by the following two conditions:

- Thermal conditions, and
- Hydraulic conditions.

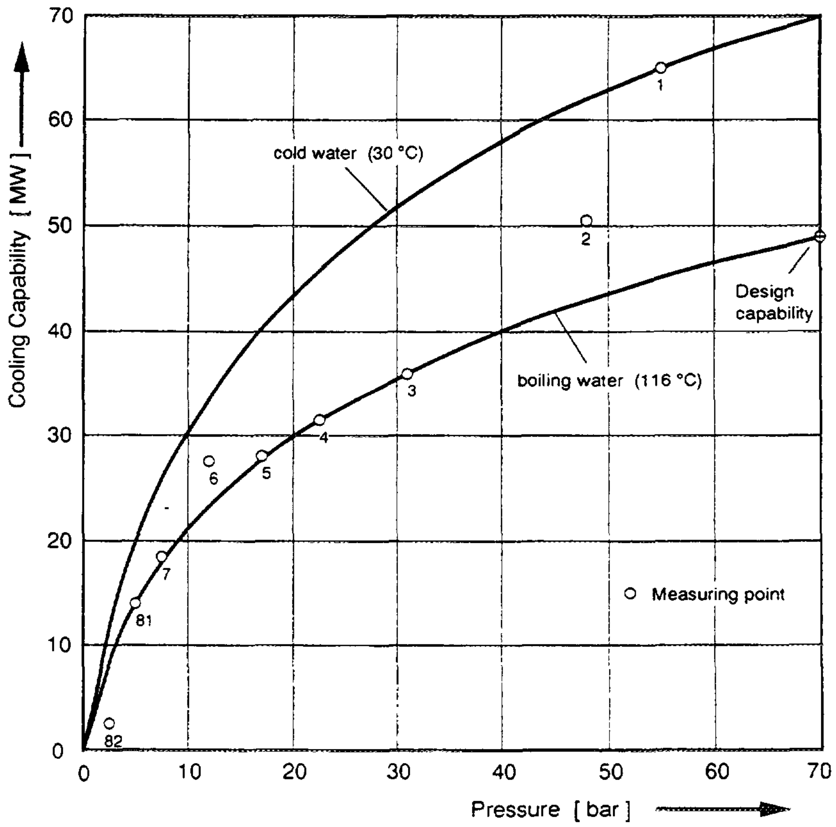

Figure 3
KRB - A
Cooling capability of the emergency condenser
Measurement on May 10, 1975

The effects of the thermal condition parameters are relatively well known to us from the evaluation of emergency condenser testing conducted at Gundremmingen. As we have altered the elevation conditions in the radial direction in comparison to Gundremmingen A, new sizing calculations have been performed. An emergency condenser test rig was constructed at the Jülich nuclear research center in order to provide experimental verification of our calculations. We will visit Jülich for the purpose of viewing the test rig. The elevation conditions for the BWR 1000 are shown in Figure 4.

In contrast to the emergency condenser at Gundremmingen, the performance of our emergency condenser is dependent not only on the primary system pressure, but also on the reactor vessel water level.

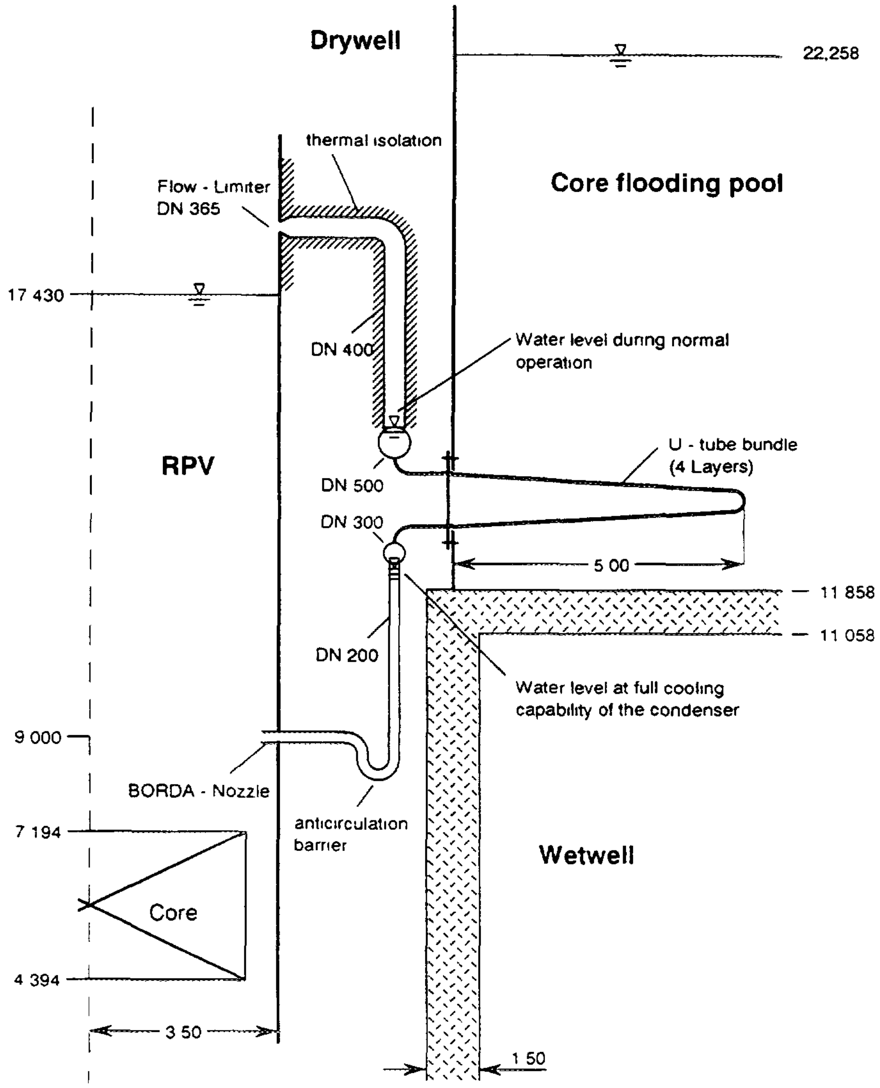

Figure 4
SWR 1000 - Isolation condenser (schematic)
(Height in m above ± 0.0 elevation in RPV)

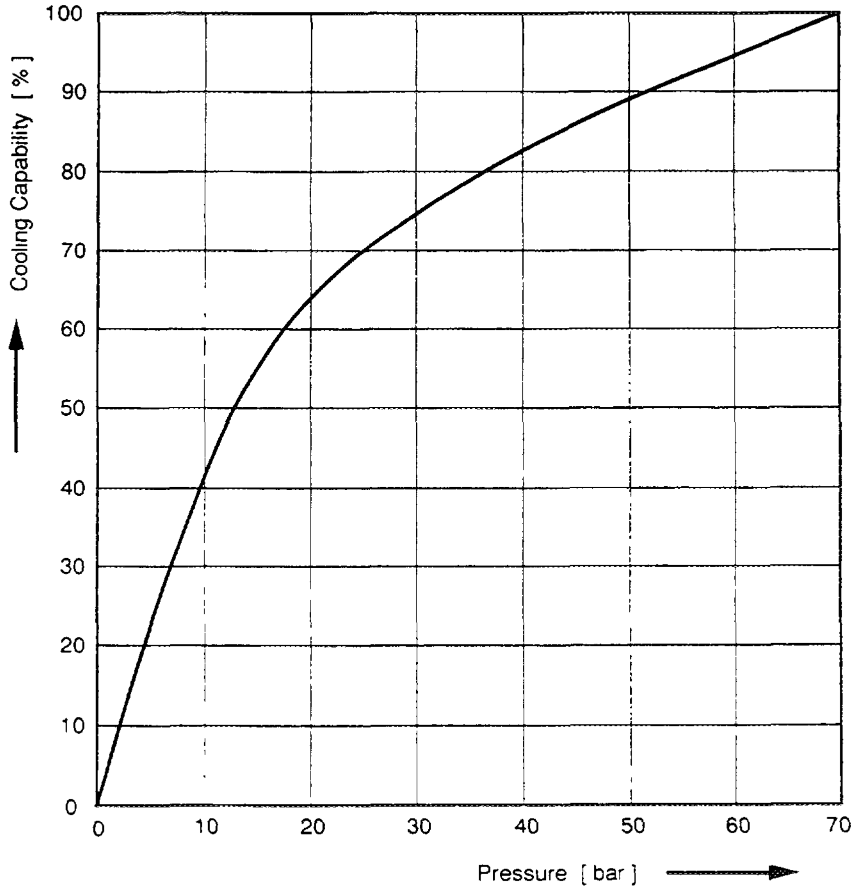

Figure 5
SWR 1000 - Emergency condenser.
Cooling capability as a function of the pressure in the RPV

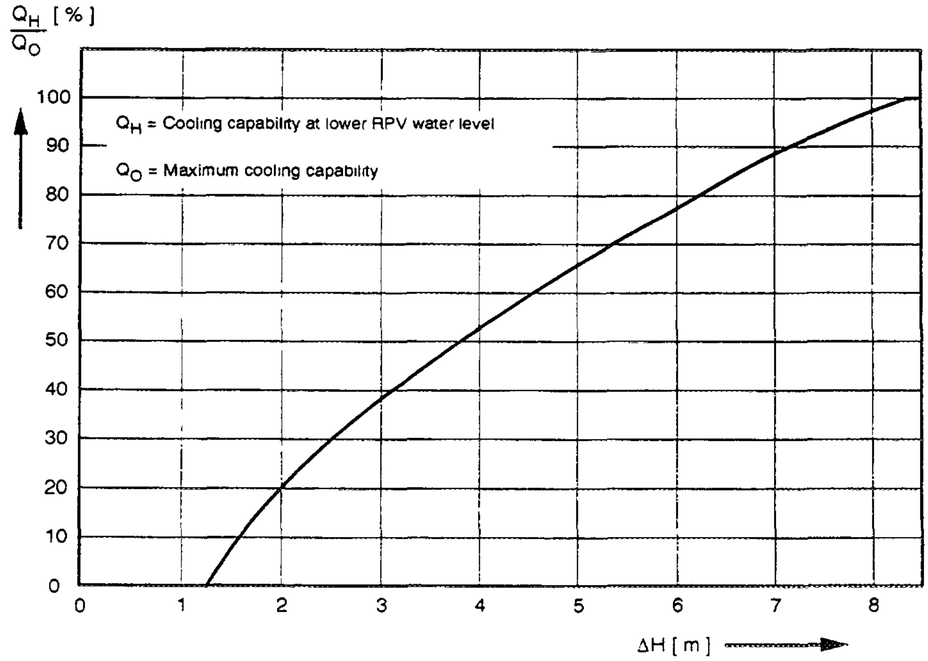

Figure 6
SWR 1000 - Emergency condenser.
Cooling capability as a function of loss of water level in the RPV. (ΔH in m)

The interdependencies between emergency condenser performance and the pressure and water level in the reactor vessel are illustrated in Figure 5 and 6. An initial estimation shows that the interconnection of these two parameters is a multiplicative. The principal data for the emergency condenser system are shown in Table 1.

The emergency condensers are mainly employed for accidents involving transients (loss of main heat sink), whereas in the case of loss-of-coolant accidents (LOCA) only a limited accident control scope can be assumed by these components. The most important passive systems in the case of a LOCA are the pressure pulse transmitters and the gravity-driven core flooding system. These means that - from the point of view of safety - the emergency condenser system must only accommodate the decay heat. Taking into consideration a redundancy degree of N + 2, a specific thermal rating of

Table 1: Principal Data of Emergency Condenser System

Number of emergency condensers	4
Design	1 tube bundle comprising a four-pass U-tube configuration, with connection to two common headers
Performance of each emergency condenser	63 MW given a primary system pressure of 70 bar, a core flooding pool temperature of 40 °C, and a reactor vessel water level drop of 8.20 m
Heat transfer area per condenser	138 m², comprising 104 tubes; tube diameter: 44.5; wall thickness: 2.9 mm
Design conditions:	
- Primary side	88 bar, saturated water
- Secondary side	0 - 10 bar
Temperature:	
- Primary side	300 °C
- Secondary side	40 - 180 °C
Size (diameter) of connected piping	- Supply steam line: 400 mm - Condensate discharge line: 200 mm - Headers o steam side: 500 mm o condensate side: 300 mm

63 MW per emergency condenser results for a reactor with an output of 2778 MW. The total performance of the emergency condenser system is thus 252 MW, or 9.1 % of reactor output.

Given this emergency condenser rating, accident control of some transients becomes very interesting:

- The heat removal capacity in the lower pressure range corresponds to that of 2 to 3 relief valves (see figure 7).

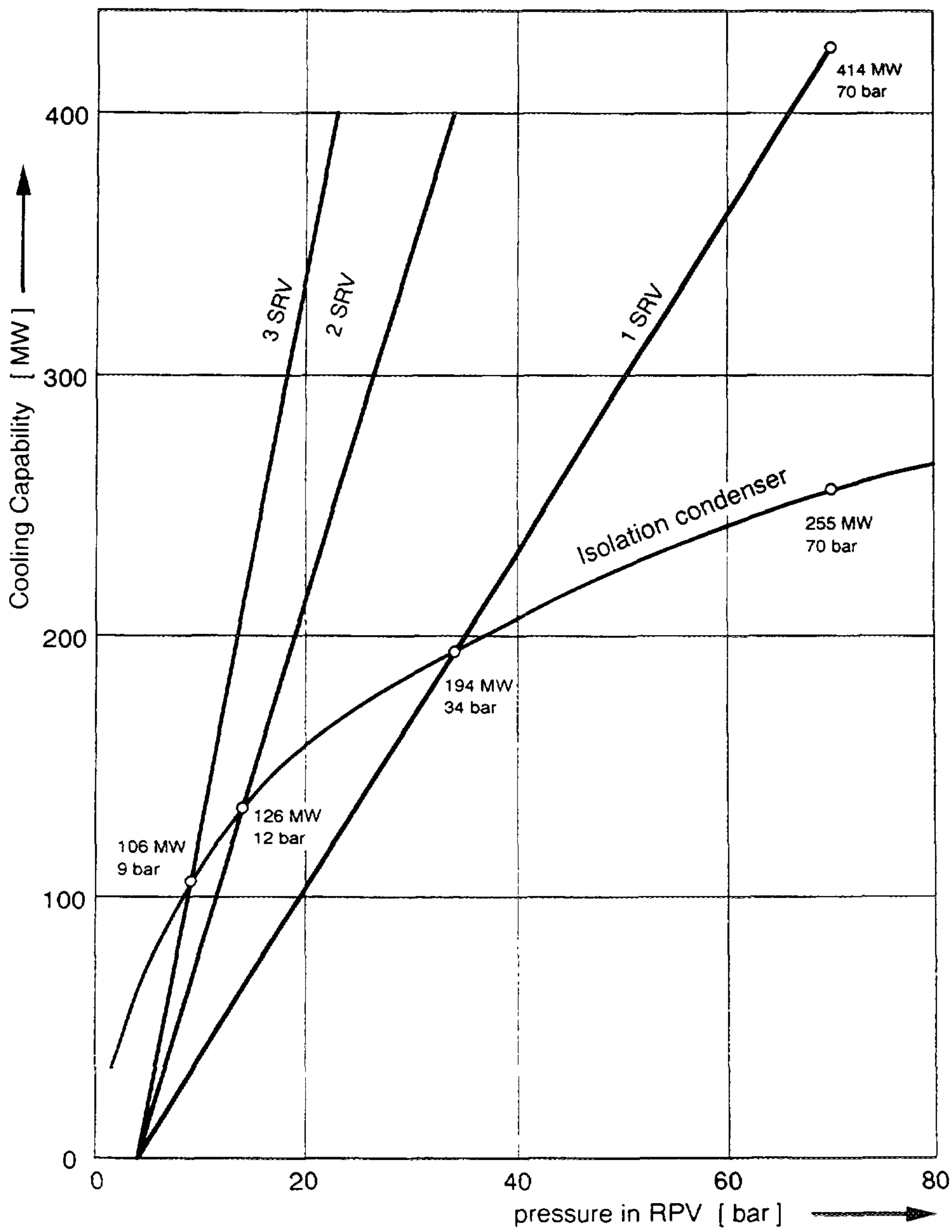

Figure 7
SWR 1000 - Comparison between the cooling capability of the isolation condenser and those of the safety relief valves

- In the event of a stuck-open relief valve with simultaneous failure of all reactor vessel injection possibilities, the core will not become uncovered until some 24 hours after the onset of accident conditions.

Dimensioning of the emergency condenser

- **thermal condition**

$$Q = ka \,.\, A \,.\, (t_{RVP} - tc) = m\,(h_v + c\,(t_{RVP} - t_{out}))$$

nomenclature

Q	=	Rate of heat transfer
ka	=	Average overall heat-transfer coefficient
A	=	Heat-exchanger surface area
m	=	mass flowrate, tube inside
hv	=	Heat of vaporization
c	=	specific heat of condensate
t_{RVP}	=	Temperature of steam in the RPV
t_{OUT}	=	Temperature of condensate of the EC outlet
t_c	=	Temperature of water in the EC (tube outside)

- **hydraulic condition**

$$DP = (P_O \,.\, h_C - P_{RPV} \,.\, h_{RPV})\, g = 1/2 \,.\, m^2\, (\Sigma\, Z_i/P_i A_i^2)$$

momenlature

DP	=	Difference of pressure
P_{RPV}	=	Density of water in RPV
P_O	=	Density of condensate at the EC Outlet
h_C	=	Elevation differential between water level inside of the EC and Inlet in RPV
h_{RPV}	=	Elevation differential between water level inside of the RPV and Inlet of condensate in the RPV
g	=	Gravity acceleration
m	=	Mass flow, tube inside
z_i	=	Resistence coefficiente in section i
P_i	=	Density in section i
A_i	=	Cross-section area of piping in section i

5 Reliability of Passive Emergency Condenser System

As little data are available on the reliability of passive components, I would just like to make a comparison with some systems which are either similar or which are intended to perform approximately the same tasks, such as:

- The emergency condenser system at Gundremmingen A, and
- The residual heat removal systems at Gundremmingen Units B and C (initial startup in 1984), which constitute the Siemens KWU Group's ABWR series.

I know that these three systems are not completely comparable, because they do not perform identical functions. Nevertheless, I would like to point out the failure probabilities of functions - according to groups - for the purpose of orientation. These failure frequencies are shown in Table 2.

6 Conclusion

With this information, I hope I have adequately introduced a very interesting component for passive heat removal from the reactor vessel. Of this component, the following can be said:

a) It is more reliable that components designed for comparable functions.

Table 2: Comparison of Magnitude of Failure Probabilities per demond of Various Residual Heat Removal Systems

Failure Probability / System	BWR 1000 Emergency Condenser	Gundremmingen A Type Emergency Condenser	Siemens-KWU Series ABWR Active RHRS
Signal acquisition and processing	-	1 E - 3	1 E - 4
Startup failure (valves, pumps, etc.)	-	1 E - 3	1 - 3 E - 2
Failure during accident (7 days)	-	1 E - 4	1 E - 2
Failure of piping and heat exchanger tubing	1 E - 4	1 E - 4	1 E - 3
Failure of power supply (from this, failure of emergency power supply)	-	1 E - 5	2 E - 5 (2 E - 2)

b) It is considerably less expensive than the residual heat removal systems implemented to date, which comprise a primary circuit, a component cooling system and a final cooling system, each equipped with pumps, valves and heat exchangers, etc. These latter systems are provided with a diesel generator as a redundant power supply system. The cost of one train (without considering infrastructure elements such as the building, etc.) can be assumed to amount to some DM 100 million. In contrast to this, the cost for an emergency condenser system (comprising four emergency condensers) is estimated to cost between somewhere between DM 10 and 20 million.

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PASSIVE HEAT REMOVAL IN CANDU

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Abstract

The Three Mile Island accident spurred a world-wide interest in severe accidents. The initial reaction was to increase the *preventative* measures in existing designs, followed by development of predictive capabilities to improve the *management* of severe accidents[1]. Recently, emphasis has been placed in new designs on *mitigative* measures which slow down or contain the progression of a severe accident. U.S. requirements for Advanced Light Water Reactor designs must now:

- provide reactor cavity floor space to enhance debris spreading
- provide a means to flood the reactor cavity to assist in the cooling process[2]

This paper describes how CANDU Pressurized Heavy Water Reactors (PHWRs) have severe accident prevention and mitigation[3] inherent in the design; in particular, the U.S. severe accident requirements can be met without significant change to the design of current CANDUs.

2. AVAILABLE WATER NEAR THE FUEL

CANDU is a horizontal pressure-tube reactor, with the fuel bundles located inside several hundred 10.5-cm diameter, 0.48-cm thick pressure tubes[1]. Twelve 0.5 m-long fuel bundles reside within each pressure-tube. The 37-element fuel bundle is in close proximity to the pressure tube, separated from it by means of 1.1-mm high bearing pads on the outer fuel elements. The heavy-water coolant flows over and through the fuel bundles and is contained by the pressure tubes within the core.

Since the pressure-tube operates at approximately the coolant temperature (300°C), it is thermally insulated during normal operation from the heavy water moderator (65°C) by the carbon dioxide filled annulus formed between the concentric pressure tubes and calandria tubes. The calandria tube forms the outer boundary between the gas and the moderator (Figure 1). The assembly of fuel, pressure-tube, gas annulus and calandria tube is collectively called the fuel channel. The total radial distance between the fuel and the moderator is 1.5 cm.

[1]Unless otherwise specified, specific numerical values refer to the CANDU 9 reactor. However the relationships between the values and the conclusions are generic to all CANDUs.

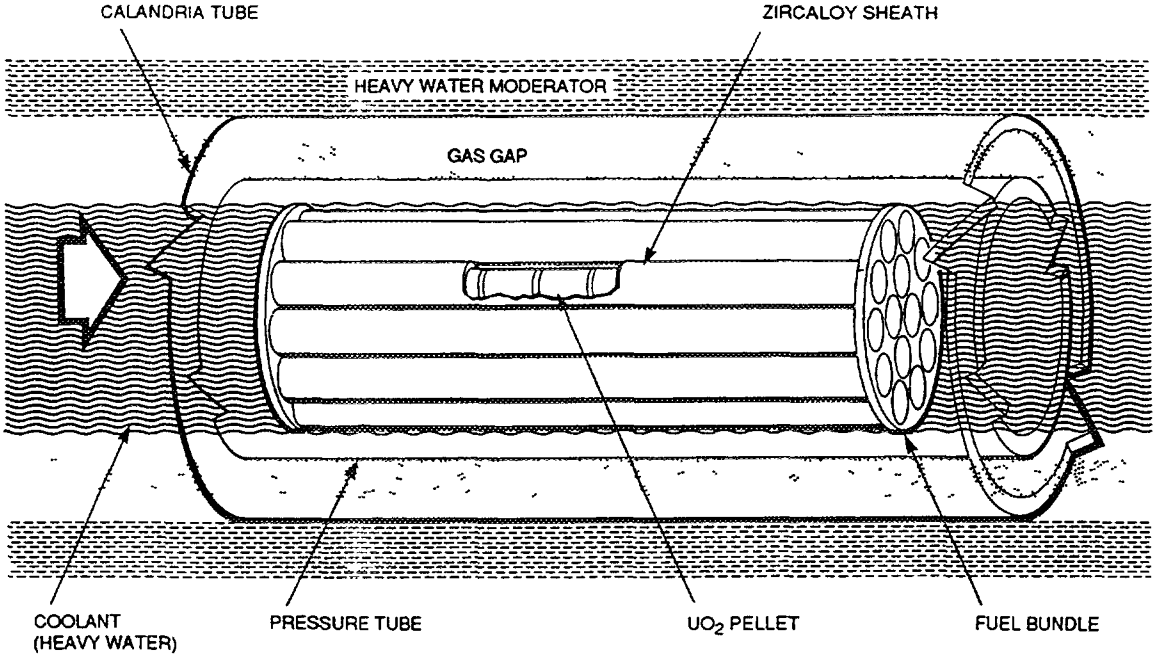

FIGURE 1 Separation of coolant and moderator

The moderator is contained within a low-pressure tank, called the calandria. During normal operation, about 4.4 % of the thermal output of the core is deposited in the moderator, a small amount by conduction from the channels, but mostly by direct deposition of fission gamma rays. This heat is removed via dedicated external moderator heat exchangers; external pumps circulate the moderator through the heat exchangers and provide momentum to mix the moderator within the calandria. They are powered by normal Class IV electrical power, backed up by Class III emergency diesel power when required.

The moderator role as an emergency heat sink for the fuel in a severe accident is discussed below. In this role, its active heat removal capability is enough to continuously remove all fuel decay heat following 15 seconds after reactor shutdown. The moderator specific volume is typically 8 litres/kW(th) at 1% decay power, or enough to absorb (through heat-up and boil-off) over 5 hours of decay heat from the fuel, assuming no heat removal from the moderator fluid.

The calandria vessel is in turn contained within a shield tank, which provides biological shielding during normal operation and maintenance (Figure 2). It is a large steel or concrete tank filled with ordinary water. During normal operation, about 0.4 % of the thermal output of the core is deposited in the shield tank and end shields, through conduction from the calandria structure and fission heating. This heat is removed via the end shield cooling system, consisting of pumps and heat exchangers.

The shield tank's role as an emergency heat sink for the fuel in a severe core damage accident is discussed below. In this role, its active heat removal capability is enough to continuously remove all fuel decay heat a few days after reactor shutdown. The shield tank specific volume is typically 16 litres/kW(th) at 1% decay power, or enough to absorb (through heat-up and boil-off) more then ten hours of decay heat from the fuel, assuming no heat removal from the shield tank water.

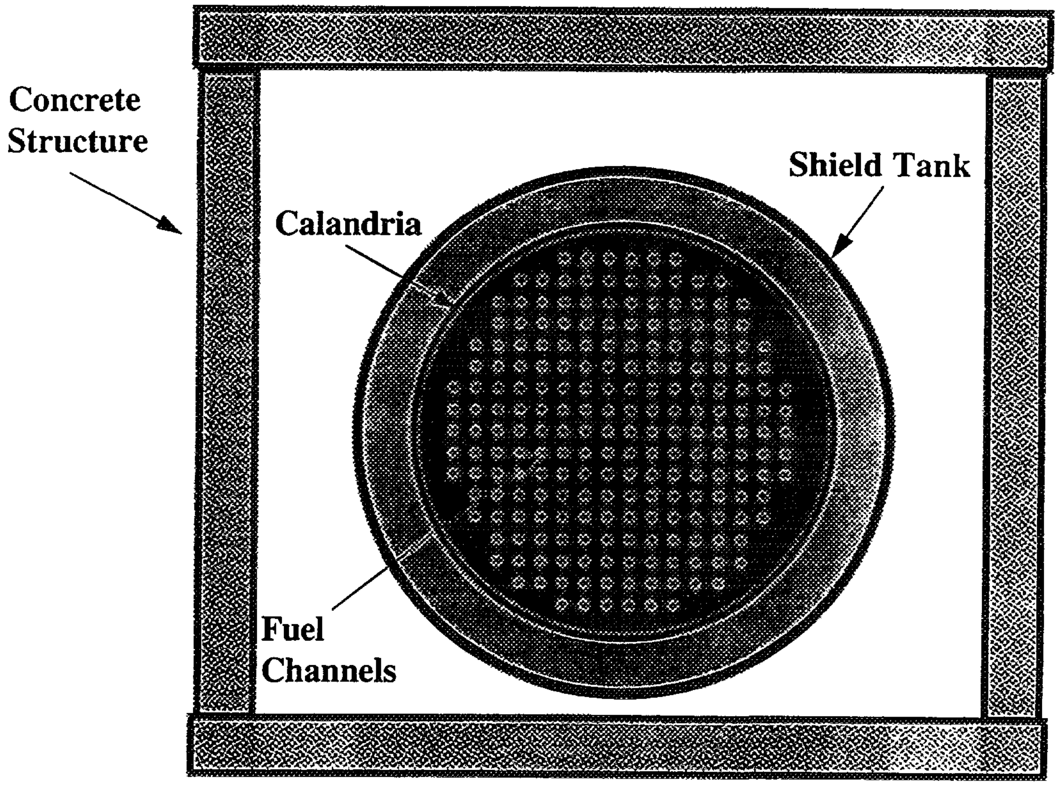

FIGURE 2: Shield tank as core catcher

TABLE I
CAPABILITIES OF MODERATOR AND SHIELD TANK IN SEVERE ACCIDENTS

System	Continuous Heat Removal Capability (% full thermal reactor power)	Specific Fluid Volume at Decay Power	Time to Heat Up and Boil off All Fluid By Fuel Decay Power, No Heat Removal
Moderator	4.4 %	8 litres/kW @1%	> 5 hours
Shield tank	0.4 %	16 litres/kW @ 1%	>10 hours
		32 litres/kW @ 0.5%	> 20 hours

3. MODERATOR AS HEAT SINK FOR THE CHANNEL

All large pipes in the CANDU Reactor Coolant System (RCS) are above the core. They consist of headers, or collectors, to which each channel is connected via a 6-cm to 8-cm diameter inlet and outlet feeder pipe; plus pump suction and discharge piping and steam generator inlet and outlet piping. A large break in one of these pipes would cause rapid voiding of the pressure tubes. As with other water-reactor designs, the emergency core cooling system (ECC) provides high-pressure injection of water to refill the core. In CANDU ECC water is supplied to all the reactor headers.

A failure of ECC in light-water reactors, will, if uncorrected, lead to a meltdown of the core. In CANDU, a loss of coolant with a failure of ECC will be arrested by the moderator short of UO_2 melting. The mechanism is as follows[4]:

The fuel will heat up due to decay power, since no heat is being removed by the RCS. Since the pressure-tube is close by, it will also heat up, by conduction and radiation from the fuel, and convection by the steam remaining in the channel. At about 800°C, the pressure tube will start to plastically deform under the loads from the weight of the fuel and any residual coolant pressure. If the coolant pressure is high (for example, for medium-sized breaks with failure of ECC), typically above 1 MPa, the pressure tube will strain radially outward until it contacts the cool calandria tube (Figure 3). If the pressure is below 1 MPa, the pressure tube will preferentially sag, until again it contacts the cool calandria tube. As long as the calandria tube remains cool, it

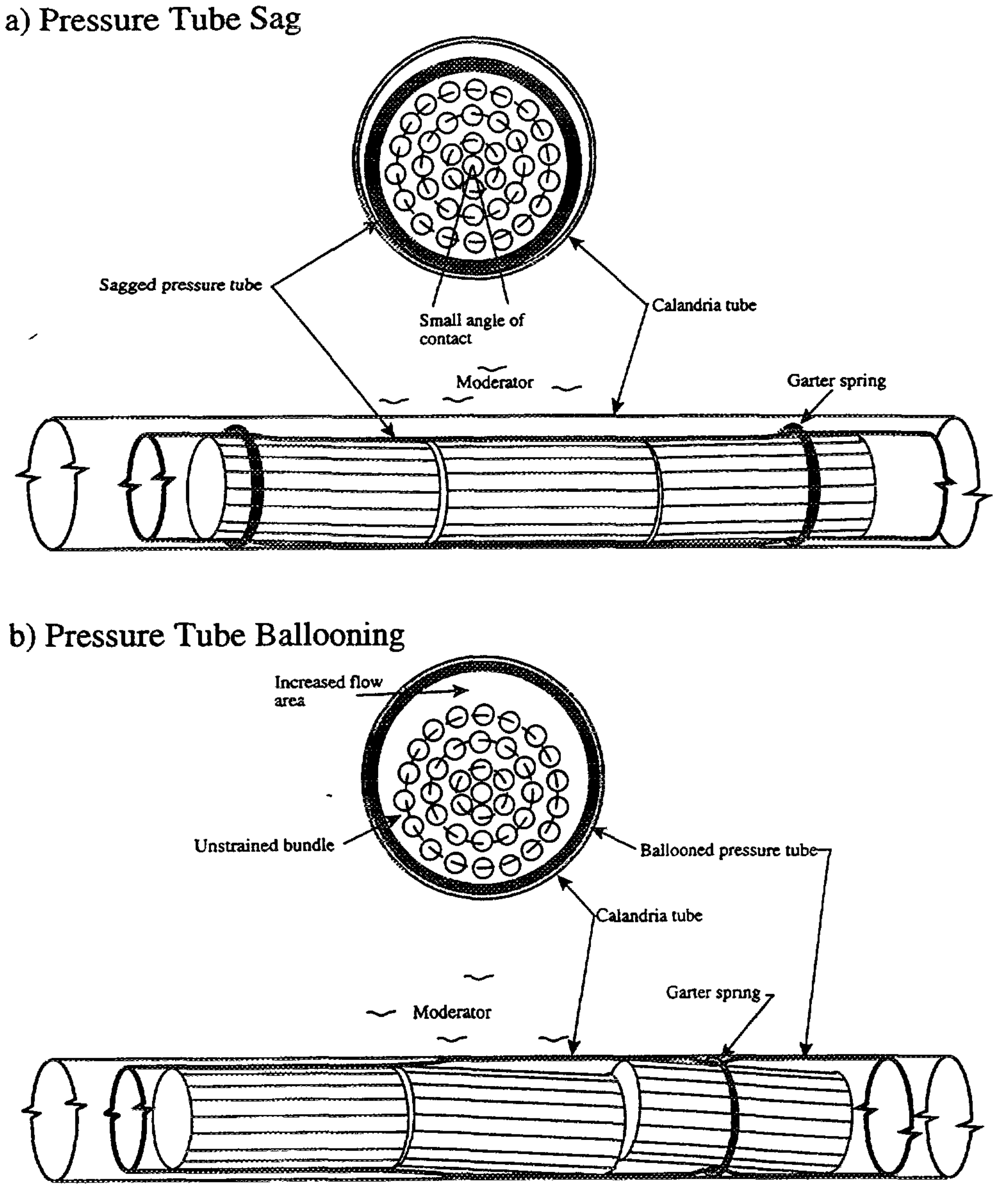

FIGURE 3: Pressure Tube Deformation Modes During a Postulated Loss-of-coolant Accident: a) Sagging, b) Ballooning

is strong enough to arrest the deformation of the pressure tube. Heat can then be removed from the fuel, by conduction and radiation to the pressure tube and calandria tube, and then by convection to the bulk moderator. From there it is removed by the moderator cooling system. The pressure-tube thus acts as a passive **fuse**, deforming only when it overheats in an accident, and so creating a low-resistance heat transfer path to the moderator. This path can remove decay heat from the fuel without the UO_2 melting even with no coolant in the pressure-tube. This is due to the short physical distance from the fuel to the pressure-tube, the relatively thin walls of the pressure-tube and calandria tube, and the enhanced heat transfer through the two tubes when they touch.

The calandria tube can be kept cold by preventing dryout on the outside surface at the time of pressure-tube contact. The surface heat flux at contact is determined by the pressure-tube temperature, the interface heat transfer coefficient and the moderator subcooling. The former cannot practically be controlled, but the latter two can. For existing CANDU reactors, a moderator temperature of about $70^{\circ}C$ is sufficient to prevent calandria tube dryout.

The pressure-tube "fuse" is sensitive to moderator temperature, but NOT to active moderator heat removal - it is truly passive. However the moderator pumps and heat exchangers are used to bring the severe accident to a controlled steady-state.

Measures are taken to assure that the pressure tube does not fail before it reaches the calandria tube. Although such failure would not prevent the moderator from performing its emergency role, the sequence is less complex if the pressure tube remains intact. Pressure tube integrity depends on the pressure at which the pressure-tube strains - the higher the pressure, the more sensitive is the strain to non-uniformities in pressure tube temperature, and the higher the chance of failure before contact with the calandria tube. The pressure parameter varies slightly with the design of the RCS.

Another severe accident results from assuming *all* heat sinks for the RCS are lost. This is an unlikely sequence because the following systems are each capable of removing decay heat from an intact RCS:

- the main feedwater system
- the auxiliary feedwater system
- the shutdown cooling system (which can be brought in at full RCS temperature and pressure)
- the Group II emergency feedwater system (this is a separate means of adding water to the steam generators, taking its supplies from a separate seismically qualified source and using independent seismically-qualified power)
- A gravity supply of water to the steam generator from the high level dousing or reserve water tanks

If however they are all lost, the RCS will pressurize and the fluid will gradually be lost through the relief valves, and the fuel will overheat. Since this sequence occurs at or above operating pressure, typically 10 MPa, the overheated pressure tubes will start to fail before they contact their respective calandria tubes. The higher powered channels will fail first, and the pressure tubes will relieve the rest of the RCS fluid. This will reduce the RCS pressure and allow the moderator to act as an emergency heat sink as described above.

Section 6 describe the Research programme which develops and verifies the models for these sequences[5].

4. SHIELD TANK AS HEAT SINK FOR THE CALANDRIA

Use of the moderator as an emergency heat sink for severe accidents has been extensively studied in Canada both theoretically and experimentally. The driving force has been the AECB requirement that certain severe accidents be considered within the Design Basis. This set includes all combinations of a reactor system failure *and* the unavailability of a safety system - for example, the previous example of a large LOCA and failure of ECC injection. **Severe accidents** within this set, i.e., those for which the fuel heat is not removed by the RCS, result in damaged fuel, but do not lead to loss of pressure-tube geometry. Accidents which combine yet further failures are generally outside the design basis. They may result in loss of core geometry, in which case they are called **severe core damage** accidents. The two types of accidents are usually synonymous in other reactor types, but because the moderator can arrest severe accidents before the core geometry is lost, in CANDU they are distinct.

Severe core damage accidents in CANDU include sequences such as:

- loss of all feedwater and loss of cooling to all alternate heat sinks including the moderator
- loss of coolant, loss of ECC injection, and loss of moderator cooling.

The frequencies of such combinations[6] are of the order of 10^{-7}/year, and are thus not within the scope of licensing analysis. They are, however, examined in the context of Probabilistic Risk Assessment[7]. Because of the low frequency, the emphasis has been on scoping calculations[8] rather than extensive experimental verification of detailed codes.

For such sequences, the moderator water will heat up and boil off. This will take some hours, during which time the pressure tubes will start to fail and the debris will collect in the bottom of the calandria. As long as there is water in the shield tank, the calandria shell will remain intact; the heat generated by the debris is less than the critical heat flux on the outer surface of the calandria[9]. However as is apparent from Table I, the shield tank heat removal rate is insufficient to keep up with the decay power until a few days have passed, so the shield tank water will boil off and the calandria shell will be penetrated. Nonetheless, the heat-up and boil-off of the moderator and shield tank buys valuable time, up to 24 hours, so that accident management can be put into effect before the debris even reaches the concrete floor of the containment.

5. DESIGN ENHANCEMENTS TO IMPROVE PASSIVE HEAT SINKS[10]

Based on the previous description, it is obvious how to extend the passive heat sinks provided by the moderator and the shield tank - simply add water. The advanced evolutionary CANDU 9 family[11] (single unit plants in the power range from 900 - 1300 MWe) has done just that. *An elevated reserve water storage tank in containment* provides emergency makeup water to the moderator and permits passive heat removal by thermosyphoning from the shield tank (Figure 4). The amount of water is sufficient for more than 40 hours of decay heat removal. During or after that time, a recovery pump collects water from the building sumps and returns it to the reserve water tank. The heat is removed from containment through a combination of passive conduction through the

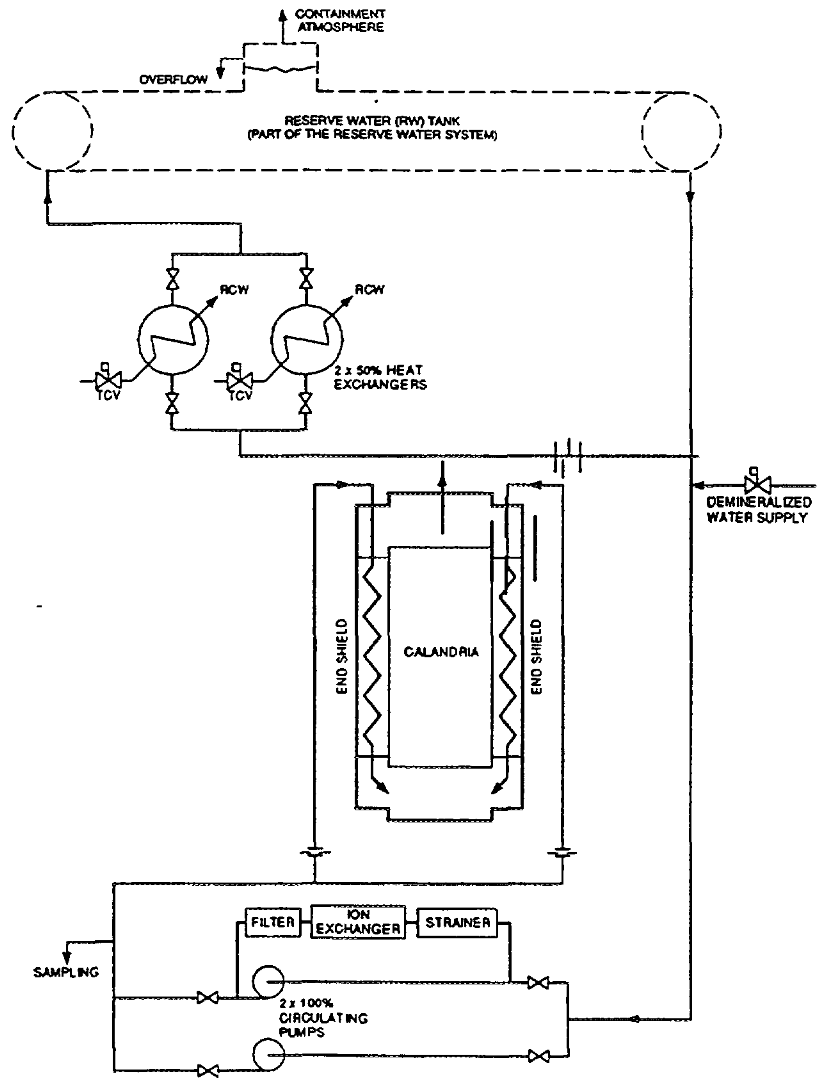

FIGURE 4: Shield cooling system flow diagram

building walls and actively by containment air coolers. A severe accident can thus be arrested either by the moderator or by the shield tank, contained therein, and stabilized. The same approach is being considered for the smaller CANDU 3 (a single-unit 450 MWe plant)[12].

To ensure that steam is adequately relieved from the shield tank without overpressurizing the vessel, engineered relief paths have been provided on the newer designs, sized to take the steam flow generated by decay heat removal.

Finally the pressure-tube/calandria-tube heat transfer has been fine-tuned. First, to reduce the sensitivity to initial moderator subcooling: When the two tubes contact in an accident, the stored heat in the pressure tube is transferred in a "pulse" through the calandria tube. To reduce the magnitude of this pulse, which sets the margin to CHF of the calandria tube, the inner surface of the latter has been roughened slightly (50 micron ridges). This "smears out" the heat transfer over a longer period of time and reduces the peak inter-tube heat flux. Second, to enhance the heat transfer after sag contact, and so to reduce the quasi-steady-state fuel temperatures, the inside of the calandria tube has been blackened.

In short, the provision of emergency water to the moderator and shield tanks gives an effective, and cost-effective, way of arresting severe accidents. Moreover the U.S. requirements for severe accidents, described in Section 1, are met inherently by existing CANDU structures -the calandria shell (backed up by the shield tank) provides both the "floor" area for spreading of debris **and** passive debris cooling through the shield tank water.

6. R&D IN SUPPORT OF PASSIVE HEAT REMOVAL

6.1 General Methodology

The operation of the moderator as a heat sink when normal and emergency cooling is lost to the fuel has been described above. The verification of such behaviour under a wide range of accident scenarios has been provided by an extensive and on-going research programme. This research programme covers fuel channel behaviour throughout the LOCA transient. Phenomena such as coolant boiloff in the channel, thermal-chemical behaviour of the fuel channel at elevated temperatures and pressure-tube deformation have all been extensively studied. The general methodology used in the research programme has been to perform small scale separate-effects experiments to develop and validate mathematical models to describe the phenomena. These validated models are then integrated into a code linking the various phenomena to characterize the fuel channel response to a loss of coolant accident.

For example, small scale experiments have been performed to study pressure-tube deformation at elevated temperatures. These high-temperature creep experiments characterized the plastic deformation mechanisms which control the ballooning and sag behaviour of the pressure tubes when they heat up. The end product was a set of constitutive equations describing transverse and longitudinal pressure-tube deformation[13][14].

The small scale experiments permitted the development of pressure-tube deformation models to describe the ballooning behaviour of a full size fuel channel. These models were validated through experiments on full size sections of pressure tubes in a simulated fuel channel mock-up including a calandria tube and a surrounding water tank to simulate the moderator[15] [16] [17]. Experiments covered both sag and ballooning of the pressure tubes (Figure 3).

6.2 Moderator as a Heat Sink

When an overheated pressure tube deforms and contacts its surrounding moderator-cooled calandria tube, the thermal and mechanical responses of both tubes change rapidly. Prior to contact, the pressure tube is at a much higher temperature than the calandria tube. Upon contact, stored heat in the pressure tube is transferred across the interface of the contacting tubes, through the calandria tube and then to the surrounding moderator. This process results in a large increase in the heat flux to the outside surface of the calandria tube. The magnitude of this heat flux is determined by the internal pressure, pressure-tube contact temperature and the interface thermal contact conductance. The magnitude of the peak heat flux relative to the critical heat flux governs the type of boiling which would occur at a given location on the calandria tube surface.

If the sudden rise in surface heat flux does not initiate film boiling on the outside surface of the calandria tube, the stored heat in the pressure tube is transferred to the moderator. If the critical heat flux on the surface of the calandria tube is exceeded in a particular area, the surface will dry out and film boiling will occur. Since film boiling is less efficient at heat removal than nucleate boiling, the stored heat in the pressure tube is only partially removed, and the calandria tube heats up. If the incident heat flux to the pressure tube is high, the tubes could overheat sufficiently to jeopardize fuel channel integrity.

The relationship between subcooling and critical heat flux on the outside surface of the calandria tube has been investigated over the years through small scale pool-boiling experiments with horizontal banks of tubes. Information from these small-scale experiments fed into full-scale contact boiling experiments using reactor-typical pressure and calandria tubes. These contact boiling experiments covered a wide range of moderator subcooling, pressure-tube internal pressures and pressure-tube heatup rates[18]. The current moderator subcooling requirements are specified for CANDU reactors to avoid the calandria tube being forced into film boiling upon contact with its deforming pressure tube. Figure 5 schematically represents a collection of experimental data from several contact boiling experiments. The broad hatched line marks the boundary separating the film and nucleate boiling regimes. From this, it is apparent that a moderator local subcooling of 26 to 28°C is sufficient to prevent extensive dryout on the calandria tube external surface during ballooning.

This moderator subcooling requirement may be reduced significantly by reducing the pressure-tube to calandria-tube thermal contact conductance as demonstrated by Sanderson et. al.[19] In this experiment, the tube to tube contact conduciance was

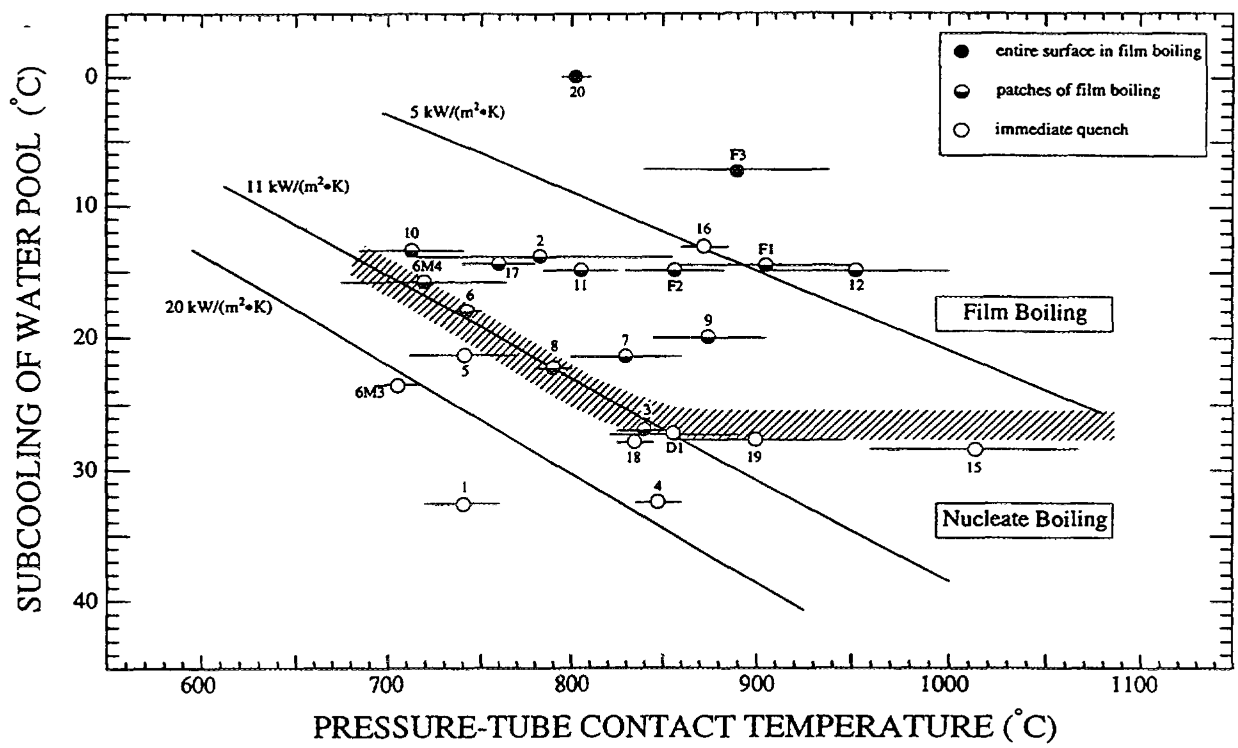

FIGURE 5: Moderator Subcooling Requirements as a Function of Pressure-tube Contact Temperature. The hashed line marks the boundary between film and nucleate boiling regimes.

reduced from its nominal value of 11 kW/(m^{2}K) upon ballooning contact to less than 1 kW/(m^{2}K) through contact limiters placed between the two tubes. This reduction in contact conduction has the potential to significantly reduced the moderator subcooling requirements.

Having demonstrated the sufficient conditions for good heat transfer following ballooning contact, the R&D programme focussed on determining if there were any mechanisms by which the pressure tube would fail prior to coming into contact with the calandria tube or cause the ballooned fuel channel itself to fail after contact in spite of general nucleate boiling.

The likelihood of pressure-tube failure prior to ballooning-contact involves both thermal and mechanical considerations. Since Zr-2.5 Nb pressure tube material is ductile at high temperatures, failure only occurs as a result of severe necking or thinning of the material at a given location. Given the geometric configuration of a pressure tube and its surrounding calandria tube, this is possible only if the pressure tube experiences a highly localized strain before it contacts its calandria tube. Such conditions could arise from either stratified coolant conditions in which the top half of the pressure tube is exposed to steam while the bottom is kept relatively cool by the presence of liquid, or by contact with the fuel element. The latter case can result from conduction through the bearing pad or by contact with bowed, sagged or melted cladding.

6.3 Fuel Channel Behaviour Subjected to Stratified Flow

Water in the horizontal fuel channels of a CANDU reactor may boil off slowly in some postulated LOCA scenarios. This would expose the upper portion of the fuel bundle and pressure tube to superheated steam as the water level drops (Figure 6). The pressure tube would become hot at the top because of thermal radiation and steam convection while it remained near the saturation temperature below the liquid level. The resulting pressure-tube circumferential temperature gradient would induce localized thermal stresses and plastic deformation at the top of the tube. Such conditions may cause nonuniform pressure-tube ballooning and the pressure tube could possibly rupture before coming into contact with the surrounding moderator cooled calandria tube.

A series of experiments have been performed to investigate the circumferential temperature distribution and resultant deformation that may develop around the pressure

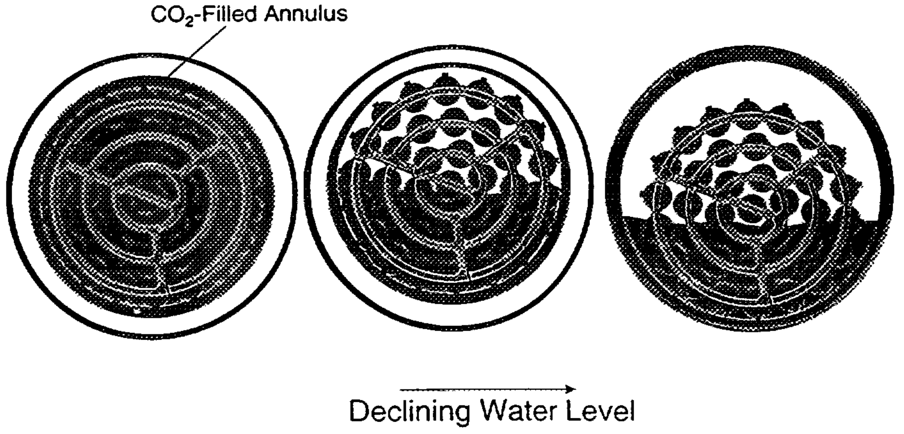

FIGURE 6: Schematic Showing Pressure-tube Behaviour Under Coolant Boiloff Conditions

tube of a CANDU fuel channel under such conditions[20] [21]. These experiments have shown the benefit of steam flow in the uncovered portion of the fuel channel. The steam helps distribute heat circumferentially across the top of the pressure tube, reducing thermal gradients and the likelihood of localized hot spots. The reduction of localized hot spots limits localized strain and the likelihood of a pressure tube failure. These full-scale experiments have provided a substantial data base of experimental results for use in the validation of fuel channel codes used in the analysis of fuel channel behaviour during a LOCA.

6.4 Fuel Channel Behaviour Subjected to Localized Hot Spots

In some postulated LOCAs, the interior of the pressure tube can become completely dry in a matter of seconds after flow stagnation occurs. As the RCS depressurizes, the surface temperature of the fuel bundle can exceed 1000°C. Most of the pressure tube circumference will be heated by thermal radiation, except at locations where the bearing pads are in contact with the pressure tube. Here, conduction and thermal radiation are the dominant modes of heat transfer. Therefore, local hot spots can develop on the pressure tube under the bearing pads. Whether the pressure tube would fail at these hot spots before contacting the calandria tube depends on the temperature and pressure transients it experiences.

An extensive series of small[22] and full[23] [24] scale experiments have been performed to investigate this phenomenon. These experiments demonstrated that the interaction between adjacent bearing pads in contact with the pressure tube tended to smooth out the circumferential temperature gradients. During heatup, the pressure-tube temperature increased more rapidly opposite the ring of bearing pads. This resulted in greater axial temperature gradients than circumferential gradients. The thermal contact conductance between the bearing pad and the pressure tube increases during heatup then decreases during ballooning. This decrease in conductance during ballooning helps minimize the magnitude of the bearing-pad induced hotspot, minimizing the risk of pressure tube failure under the bearing pad.

Another recent series of experiments[25] looked at the effect of molten zirconium that might be created by an overheated fuel bundle end plate if it were to fall onto the surface of a ballooned pressure tube and create a hot spot. The experiments and subsequent analysis demonstrated the resilience of the fuel channel to intense localized hot spots, as might occur in a severe accident. The ballooned fuel channel was resilient if the outside surface of the calandria tube was well cooled (i.e. it was in nucleate boiling) prior to molten Zr-4 (up to 90 g) making contact with the ballooned pressure tube. This in turn required sufficient moderator subcooling to prevent the critical heat flux from being exceeded, on the outside of the calandria tube during the ballooning transient. Fuel channel integrity was maintained in all experiments where the subcooling was adequate to prevent film boiling upon ballooning contact. Figure 7 shows typical calandria tube temperature transients from one of these experiments.

6.5 High Temperature Thermal-Chemical Behaviour of the Fuel Channel

Several small and large scale experiments have been performed over the years to investigate the high-temperature thermal-chemical behaviour of a CANDU fuel channel. These experiments have provided data on the high-temperature thermal properties (emissivity, thermal conductivity and solid to solid heat transfer)[26] [27] , material interactions[28] and oxidation characteristics[29] [30] of various fuel channel

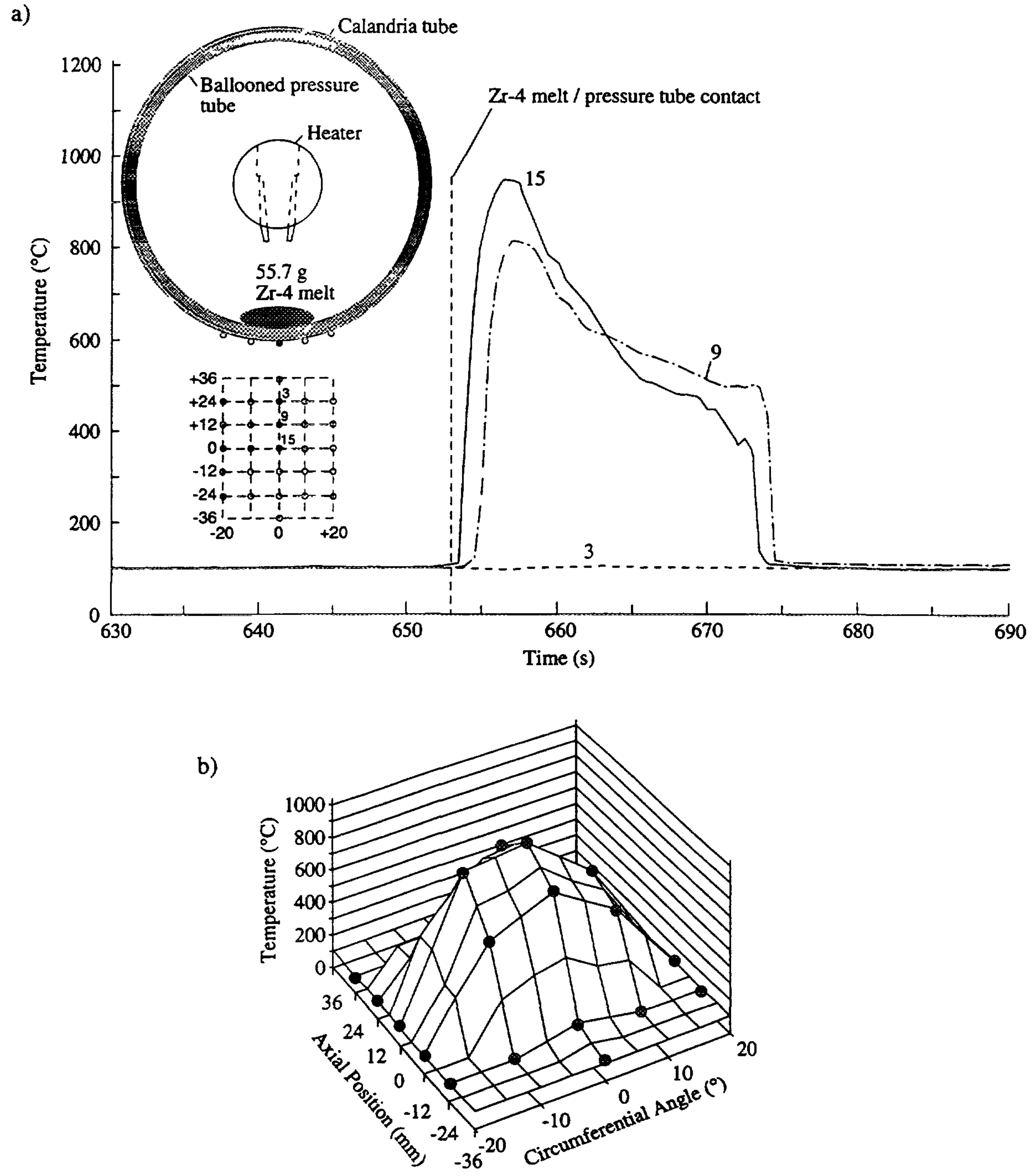

FIGURE 7 a) Calandria-Tube Temperatures Recorded Beneath the Zr-4 Melt, During Test 13 of the Molten Zr-4/Fuel Channel Interaction Program [25]. b) A Three-Dimensional Representation of Maximum Calandria-Tube Temperatures Beneath the Molten Zr-4.

components. Data from the single-effect tests were used to develop mathematical models describing the underlying phenomena. These models are then coupled into an integrated code to predict fuel channel behaviour under accident conditions. Data for validation of the integrated codes come from various full-scale experiments involving the complex interaction of pressure, temperature, material properties, heat transfer and oxidation kinetics on fuel channel components subjected to severe temperature transients.

In one such validation exercise[31], data from a high-temperature (>1600°C) thermal-chemical experiment was used to validate the multi-purpose code CATHENA. The validation exercise demonstrated the capability of CATHENA to model the thermal-chemical behaviour of a 28-element fuel channel when high-temperature steam was the only coolant available within the channel.

6.6 Core Melt Retention

A number of severe accident sequences involving loss of core geometry and core melting have been analyzed by Rogers[32]. They involve sequences where along with loss of normal and emergency cooling the moderator heat sink becomes unavailable. The level in the calandria will drop as the moderator boils and the fuel channels will heat up and collapse onto channels below that are still submerged. As the moderator level continues to drop, more channels will collapse, resulting in a pile of debris at the bottom of the vessel. Roger's analysis shows that at this stage, molten debris may exist but the shield tank water which surrounds the calandria vessel will be able to cool the debris sufficiently that the melt will be contained in the vessel. The peak heat flux into the shield tank for the sequences studied was 50 W/cm^2, well below the estimated critical heat flux of 280 W/cm^2.

The light water reactor community is now showing interest in this concept and are considering the merits of containing a core melt in a severe accident by external flooding of the pressure vessel. A research programme at the Kurchatov institute in Russia has been initiated to develop data and codes to verify this concept for pressure vessel reactors. It is cost shared 50% by Russia and 50% by fourteen OECD countries including Canada. Canada is participating since the technology derived from this study will be useful in improving our capability to analyze the shield tank capability to contain a melt.

7. SUMMARY

CANDU reactors possess two supplies of water surrounding the core - the moderator which surrounds the fuel channels and the shielding water which surrounds the calandria, that can function in emergencies to prevent or contain severe core damage. The moderator capability has been verified by small-scale and full-scale channel tests; the shield tank capability has been assessed analytically, and will be supported by international tests in which Canada is participating. The capability to stop severe accidents can be enhanced by the provision of emergency water to the moderator and shield tanks. This capability exceeds developing international requirements on the mitigation of severe accidents.

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DIVERSIFIED EMERGENCY CORE COOLING IN CANDU WITH A PASSIVE MODERATOR HEAT REJECTION SYSTEM

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Abstract

A passive moderator heat rejection system is being developed for CANDU reactors which, combined with a conventional emergency-coolant injection system, provides the diversity to reduce core-melt frequency to order 10^{-7} per unit-year. This is similar to the approach used in the design of contemporary CANDU shutdown systems which leads to a frequency of order 10^{-8} per unit-year for events leading to loss of shutdown.

Testing of a full height 1/60 power-and-volume-scaled loop has demonstrated the feasibility of the passive system for removal of moderator heat during normal operation and during accidents.

With the frequency of core-melt reduced, by these measures, to order 10^{-7} per unit year, no need should exist for further mitigation.

1. Introduction

This paper concerns the employment of diversity in redundant safety systems so as to eliminate common-mode failures. Common-mode failures are single failures that dissable multiple systems.

Contemporary CANDU reactor designs employ redundancy and a considerable level of diversity in the safety systems. Thus redundancy and diversity exist in the two shutdown systems and in the reactor regulating system. Both shutdown systems make use of the low-pressure moderator environment (the calandria) but shutoff rods enter the calandria from above whereas the poison injection system enters the calandria from the side, as shown in figure 1. Each system has its own initiating signals. Note that the two shutdown systems are passive in that no operator action and no external power are needed for shutdown action to occur.

Redundancy also exists in the emergency-core-cooling systems. In the event of a loss-of-coolant accident, the Emergency Coolant Injection (ECI) system uses pressurized gas (or pumps in some plants) to inject light water into the heat transport system. The water is eventually recovered from a sump in the reactor building, cooled in a heat exchanger and pumped back into the heat transport system.

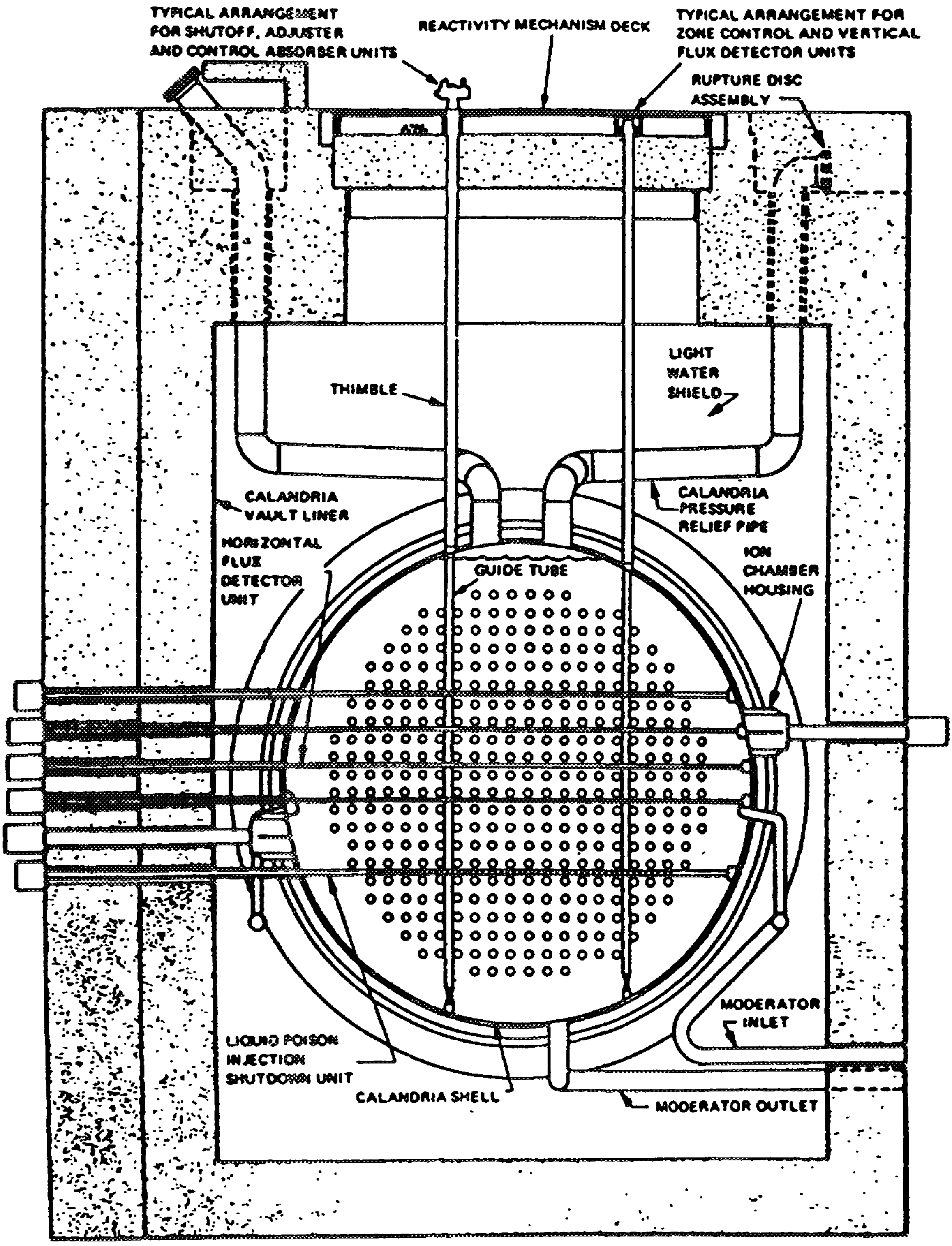

Figure 1: CANDU Reactor Core

The low-pressure low-temperature moderator serves as a redundant emergency-core-cooling system. Its availability during normal operation is apparent and it acts passively to accept heat from the fuel channels in an accident involving loss of coolant with loss of ECI. However moderator heat, which eventually has to be rejected, is done with pumps which detracts from the potential independence between the moderator and ECI systems.

Core melt in CANDU can occur only with a loss of coolant with loss of ECI plus loss of the moderator heat sink. Common failures in these systems, e.g. loss of service water to both the moderator and the ECI heat exchangers, are the main contributors limiting the core-melt frequency to about $4x10^{-6}$ per year (ref. 1). Elimination of common failures would reduce the core-melt frequency to order 10^{-7} per year, a level at which further core-melt mitigation should be unnecessary. The basis for the 10^{-7} figure is given below and compared to the figure for loss of shutdown.

A conceptual CANDU design is under study which employs a conventional ECI system with a passive moderator heat rejection system. Thus passive design techniques are used to advantage in enhancing the diversity in the two emergency-core-cooling systems. Progress on the passive moderator system development is described.

2. Passive Moderator Heat Rejection

Progress on the passive moderator system development was last given in reference 2. Figure 2 illustrates the concept. A heavy-water natural-circulation loop transfers heat to heat exchangers cooled by light water. During normal operation, the light water can be feedwater en route to the steam generators. During accidents the light water can be supplied by natural circulation from a large elevated tank.

The idea is to run the heavy water in the calandria at a temperature near the boiling point but to allow the water to flash to steam as it rises in a pipe from the calandria to an elevated heat exchanger. Subcooled heavy water would be returned to the calandria. The difference in density between the two-phase flow in the riser and the liquid in the downcomer would provide the buoyancy force to drive the flow.

Reference 2 gives results of simulations using the CATHENA transient thermalhydraulics code which demonstrate that the normal heat load to the moderator can be transferred in a stable manner with such a design. Note that the peak heat load to the moderator during a loss of coolant accident, with the reactor at decay power, is only 30% of normal full power.

More recently, further CATHENA simulations have been done at reduced powers. They show a flow oscillation at low power but stable flow at full power. Also tests have been done in a full elevation loop having a scale of about 1/60 in power, volume and flow area. They confirm the CATHENA predictions. The tests will be reported in more detail in reference 3.

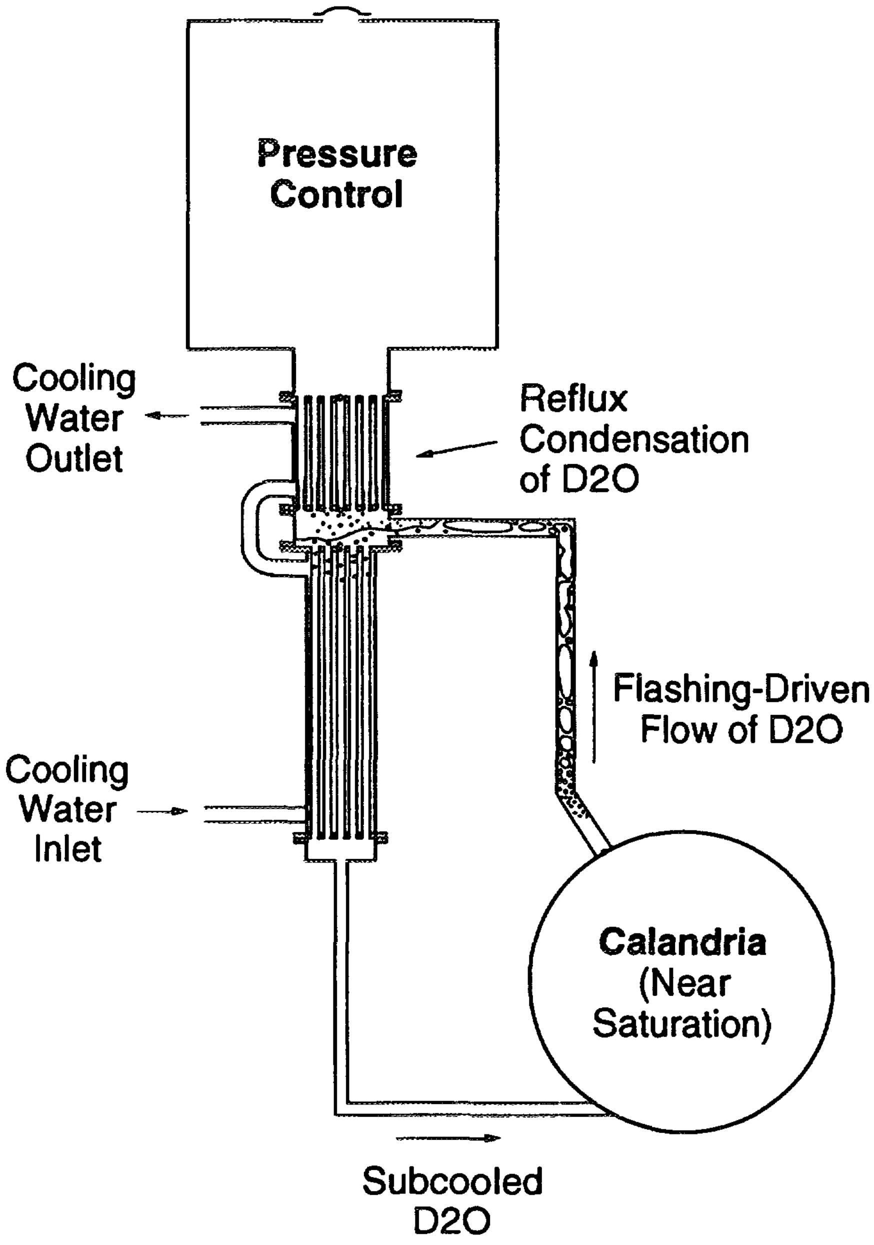

Figure 2: The Heavy-Water Natural-Circulation Loop

As power is increased, flashing is first observed in the transparent glass riser at upper elevations. The flow is oscillatory with the riser being liquid filled after the high-flow part of a cycle. No untoward effect of the oscillations is evident. As the power is increased, the oscillation amplitude decreases and the flow becomes stable. During a rapid increase of power, only one or two oscillations are seen as the flow overshoots before returning to the steady-state value. Thus the feasibility of the flashing-driven design is considered to be established both for the normal operating condition and for accidents.

3. Loss of Shutdown

The careful attention given to diversity in the design of CANDU and its two safety shutdown systems makes common-mode failures very unlikely. This enables a simple product of the failure frequency of events leading to overpower[1] and the unavailabilities of the two shutdown systems to properly reflect the order of magnitude of frequency for loss of shutdown events.

From the operating record of about 200 unit years, from improvements made to earlier control systems and from the more recent operating experience, the frequency of challenges to the shutdown systems is now about 10^{-2} per unit-year. Also the two shutdown systems are each required to have an unavailability of 10^{-3} and continued operation requires testing to ensure that this figure is met. Thus the frequency of loss of shutdown is of order

$$10^{-2} \times 10^{-3} \times 10^{-3} = 10^{-8} \text{ per unit-year.}$$

See also the figure of 2.5×10^{-8} per unit-year calculated in more detail in reference 1.

4. Core Melt

Core melt can occur in CANDU only with a loss of heat transport system inventory (whether caused by pipe failure, valve failure or loss of the steam-generator heat sink), a loss of ECI and a loss of the moderator as a heat sink. Reference 1 gives a detailed account of the accident sequences leading to core-melt for CANDU 6. The sequences are dominated by common cause events such as loss of service water and loss of electrical power. These contribute most of the core-melt frequency of 4.4×10^{-6} per unit-year. The triple failure events involving LOCA with loss of ECI and loss of the moderator heat sink contribute only 0.6×10^{-6} per unit year and even this figure is dominated by the continuing need to keep moderator pumps running in the longer term.

With diversity in the heat transport system, the ECI system and the moderator system, the core melt frequency reduces to a simple product of the failure frequency of the heat transport system and the unavailabilities of the ECI and the moderator systems. From the 200 unit-year operating record for CANDUs and the single event (Pickering unit 2, 1994) which required actuation of ECI, the failure frequency is order 10^{-2} per unit year. As with each safety shutdown system, the ECI system is required to

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[1]beyond the capability of the reactor control system

have an unavailability on demand of 10^{-3}. However the ECI pumps have to keep running for an extended period and an unavailability of 10^{-2} has been assigned (reference 1) over the mission period.

The moderator heat rejection system will be designed to operate continuously so that its availability on demand is assured. Also a passive moderator heat rejection system, which does not rely on continued operation of pumps, should be more reliable than the ECI system. An unavailability target of 10^{-3} is thought to be achievable.

The frequency of core melt becomes simply

$$10^{-2} \times 10^{-2} \times 10^{-3} = 10^{-7} \text{ per unit-year.}$$

5. Conclusion

A passive moderator heat rejection system is being developed for CANDU which, combined with a conventional emergency-coolant-injection system, will provide the diversity to reduce core-melt frequency to order 10^{-7} per unit-year. This is similar to the approach used in contemporary CANDU designs of redundancy and diversity in the shutdown systems which results in a frequency of loss of shutdown as low as 10^{-8} per unit-year.

Testing of a full height 1/60 power-and-volume-scaled passive-moderator-heat-rejection system has demonstrated its feasibility for removal of heat during normal operation and during accidents.

With the frequency of severe accidents reduced, by these measures, to order 10^{-7} per unit year, no need should exist for further measures to mitigate core melt.

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SURVEY OF THE PASSIVE SAFETY SYSTEMS OF THE BWR 1000 CONCEPT FROM SIEMENS

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Abstract

The Power Generation Group (KWU) of Siemens AG and the German nuclear utilities are currently working together to develop a boiling water reactor of the next generation which is to feature a significantly improved safety concept and possess an electrical generating capacity of about 1000 MW.

Through the use of passive safety systems and components for accident control in addition to the active systems required for plant operation, a higher degree of safety against core-endangering conditions is achieved which is no longer ruled by complex system engineering dependent on power supply and activation by I&C systems. A low core power density and large water inventories stored inside the reactor pressure vessel (RPV) as well as inside and outside the containment ensure good plant behavior in the event of transients or accidents. These passive safety systems - which require neither electric power to function nor I&C systems for actuation, being activated solely on the basis of changes in process variables such as water level, pressure and temperature - provide a grace period of more than 5 days after the onset of accident conditions before manual intervention becomes necessary. The concept features the following passive safety systems:

- Four 50% capacity emergency condensers for removal of decay heat from the RPV to a large water inventory (core flooding pool) stored inside the containment. These condensers commence functioning without valve actuation, solely on the basis of the drop in the RPV water level.

- A gravity-driven core flooding system (four flooding lines exiting the flooding pool) which becomes effective when check valves, one located in each line, self-actuate by deadweight after depressurization of the RPV.

- Four 25% capacity containment cooling condensers for removal of heat from the containment to a water inventory stored outside the

containment. These condensers require no valve actuation, rather commence functioning solely on the basis of temperature increase, e.g. due to steam formation in the containment, and thereby limit pressure rise.

- Rupture disks arranged in the bypass to safety-relief valves.

- Passive pressure pulse transmitters for actuation of the following functions:

  - Reactor scram
  - Isolation of main steam line penetrations
  - Fast RPV depressurization.

  These components generate pressure in a heat-exchanger secondary circuit as reactor water level drops which is used directly to actuate pilot or main valves.

1 Introduction

Siemens, in cooperation with Germany's electric utilities, is developing a design concept for an innovative boiling water reactor plant with a net capacity of approximately 1000 MWe.

In this design concept, the primarily "active", highly redundant safety equipment of today's operating plants are replaced by "passive" safety equipment. These function according to basic laws of physics such as gravity, natural convection and evaporation. For this reason these systems require neither "active" energy sources such as a permanent power supply nor associated I&C equipment.

The main goal of this development work is an enhanced safety concept in which functional capability and reliability are ensured by simple and less sensitive safety equipment. In this way the effects of human error are to be diminished, reactor safety is to be improved even further, and capital cost as well as maintenance reduced.

The development goals specified for designing this innovative boiling water reactor - such as passive or inherent system characteristics, good operating and accident control behavior, and ease of operation both under normal operating conditions and during accidents have resulted in a concept characterized by the following key features:

- Cooling of the reactor core when the plant is in the shutdown condition following the occurrence of abnormal events is reliably assured by making use of the natural force of gravity. Provision of a large water inventory inside the reactor pressure vessel as well as of a large source of water inside the containment makes active, fast-response safety equipment, pumps and electric power unnecessary in the event of disturbances in the reactor coolant system.

- The safety systems all operate by passive means. The provision of diverse valve designs as well as the actuation of such valves using system fluid or stored mechanical or hydraulic energy serve to supplement their effectiveness in strategic areas.

- A drop in the water level inside the reactor pressure vessel initiates automatic depressurization, allowing core flooding systems that operate according to the principle of gravity flow to be activated and preventing core melt scenarios from occurring at high reactor pressure levels. Furthermore, facilities are provided for retaining and cooling a molten core.

- Although the occurrence of a core melt accident is not to be realistically expected, the containment is nevertheless protected against such an event by being designed with sufficient capacity to accommodate any hydrogen generated by a zirconium-water reaction as well as by being inerted during plant operation.

- All systems and components employed for plant operation are based on the extensive operating experience gained from the boiling water reactor plants currently in service in Germany as well as on the proven system and component designs implemented in these plants.

In the case of a nuclear reactor designed according to this concept, the residual risk associated with melting of the reactor core is much lower than in the nuclear power plants which have gone on line most recently. Even if such a hypothetical core melt accident were to occur, the consequences of this accident would remain restricted to the plant itself; i.e. no emergency response actions outside the plant such as evacuation or relocation would become necessary.

Even for capacity ratings of around 1000 MWe, power generating costs are equal to those of large-capacity plants currently on line.

The high degree of safety achieved by using passive safety equipment should have a positive effect on political and public opinions concerning this nuclear plant design concept.

2 Nuclear Steam Supply System (Fig. 1)

2.1 Reactor Pressure Vessel and Internals

The reactor pressure vessel, which is relatively large in relation to the thermal output of the nuclear steam supply system (NSSS), contains a large water inventory above the core which allows depressurization of the RPV without simultaneous makeup of water inventory and thereby reliably prevents core uncovery.

The RPV internals differ from those of existing BWR plants with respect to the following aspects.

The active core height is 2.8 m with a core power density of 48 kW per liter. Above the core, a high, narrow chimney is situated which supports the moisture separators. The control rod drives mechanisms can be removed upwards into the RPV for servicing, which allows the RPV to be arranged lower within the containment. Among other resultant advantages, this design facilitates flooding of the RPV environs.

2.2 Containment with Pressure Suppression System, Passive Safety Equipment and Active Non-Safety-Related Systems

The RPV and the piping systems belonging to the pressure retaining boundary (PRB) are surrounded by a cylindrical concrete containment with steel liner. The containment contains a pressure suppression system comprising a pressure suppression chamber and dedicated vent pipes. Situated above the pressure suppression chamber is a large core flooding pool which serves on the one hand to provide gravity-driven flooding of the core under accident conditions subsequent to depressurization of the RPV, and on the other as a heat sink for the emergency condensers. Located above the core flooding pool are the

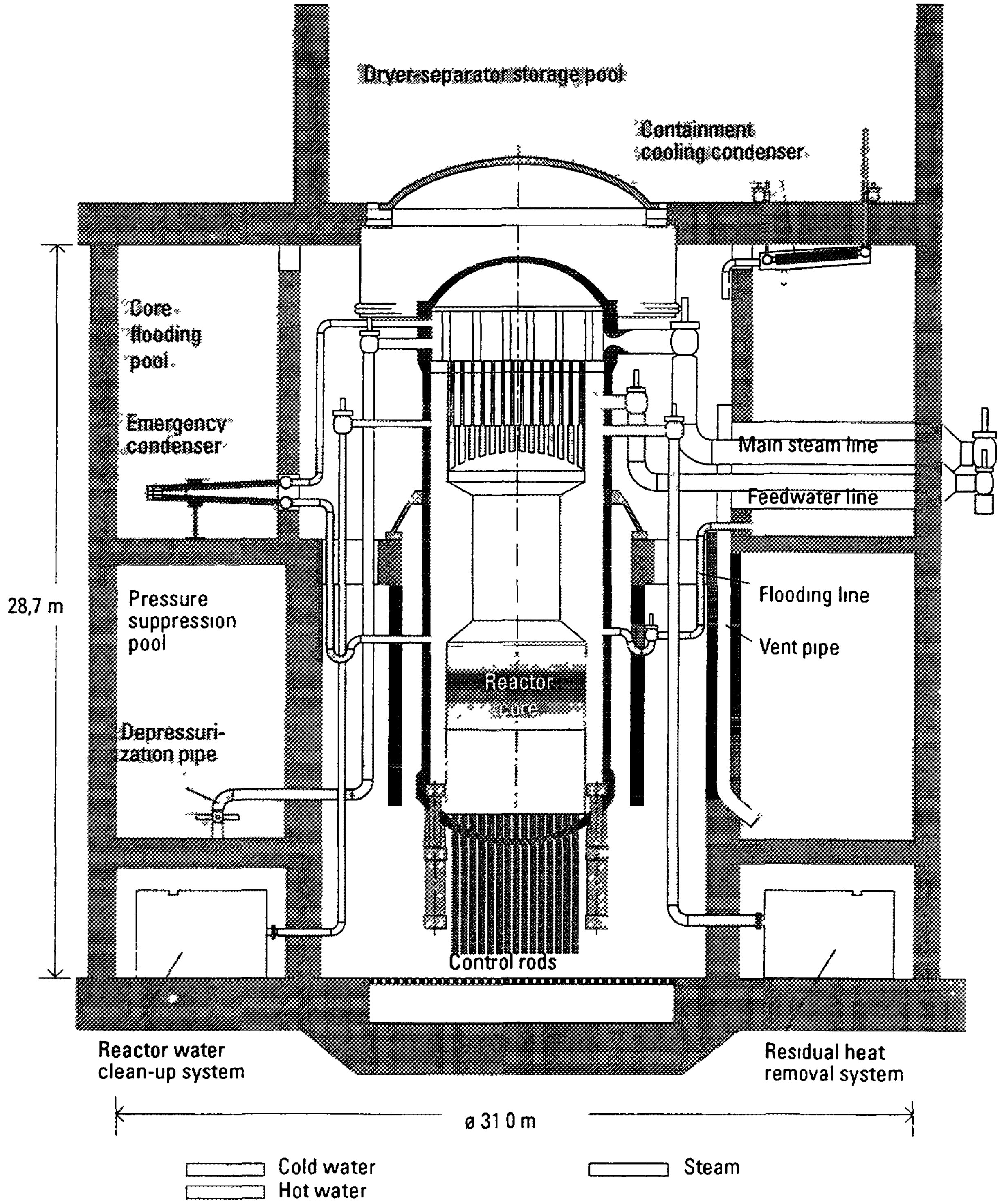

FIG. 1. BWR 1000 Contanment

containment cooling condensers which condense any steam released inside the containment and return the condensate to the core flooding pool while transferring the heat to the water of the dryer-separator storage pool situated outside the containment.

Compartments are located below the pressure suppression pool which house the two-train shutdown cooling system, the reactor water cleanup system, the valves of the reactor scram system, as well as pressure and level transmitters, etc. This arrangement is therefore such that all systems carrying reactor water are located inside the containment. The main steam and feedwater lines penetrating the containment are each equipped with three isolation valves, one of which is of diverse design. The depressurization system, equipped with system-fluid-actuated main valves and diverse pilot valves, is also located inside the containment. The main valve blowdown lines end in nozzles arranged inside the pressure suppression pool. Lines carrying steam (condensation pipes and blowdown lines of the safety-relief valves) are routed outside the pressure suppression chamber air space.

3 Safety Systems

3.1 Overview

The primary objective followed in developing the BWR 1000 is to enhance the quality of safety by introducing additional passive systems for performing safety-related functions in the event of transients or accidents. Compared to the reactors of today, the technology employed in these systems is much simpler, operation of the equipment being independent of a power supply and activation by I & C systems.

Passive systems are characterized by the fact that they utilize the laws of nature (e.g. gravity) to perform their safety functions and dispense with active components (e.g. pumps and drives). The supply of coolant to the depressurized reactor by gravity flow from an elevated water pool is a classic example of a passive system.

The most frequent events requiring control of reactor coolant inventory and residual heat removal comprise anomalies in plant operation, so-called transients. In the event of a LOCA, heat from the reactor is discharged as steam to the containment atmosphere via a postulated primary system pipe break. In order to prevent core uncovery, passive systems for supplying reactor water makeup must operate.

The following safety functions must be assured in the case of most transients as well as in the event of accidents:

- Reactor scram
- Containment isolation
- RPV pressure relief and depressurization
- Heat removal from the RPV
- Reactor water makeup and control of core coolant inventory
- Heat removal from the containment.

A short description now follows of the passive systems planned for these tasks.

3.2 Passive Safety Systems (Fig. 2)

Reactor Scram, Containment Isolation and Reactor Depressurization

The tank opening valves of the scram system, the main steam isolation valves and the relief valves of the depressurization system are all actuated by dedicated diaphragm pilot valves.

The switching operations required for these safety functions are carried out by passive and diverse means without electric power, actuating fluids or I & C signals as follows.

- Passive Pressure Pulse Transmitter (Fig. 3)

The RPV is connected via a non-isolatable line to a heat exchanger which acts as a passive pressure pulse transmitter. When the water level in the RPV is normal, the tubes inside the heat exchanger are filled with water and do not transfer any heat. If the water level in the RPV starts to drop, the water drains from the heat exchanger tubes and is replaced by steam that is continuously condensed. The heat transferred during this process causes a buildup of pressure on the shell side of the heat exchanger. This rise in pressure automatically and passively actuates pilot valves, which in turn initiate the safety functions of reactor scram, containment isolation and RPV depressurization.

Pos.		Number
1	Emergency condenser	4
2	Safety-relief valve	8
3	Spring-loaded pilot valve	8
4	Diaphragm pilot valve	8
5	Passive pressure pulse transmitter	2 x 4
6	Rupture disk	8
7	Flooding line	4
8	Containment cooling condenser	4
9	Core flooding pool	
10	Pressure suppression pool	
11	Vent pipes	15
12	Scram system	2

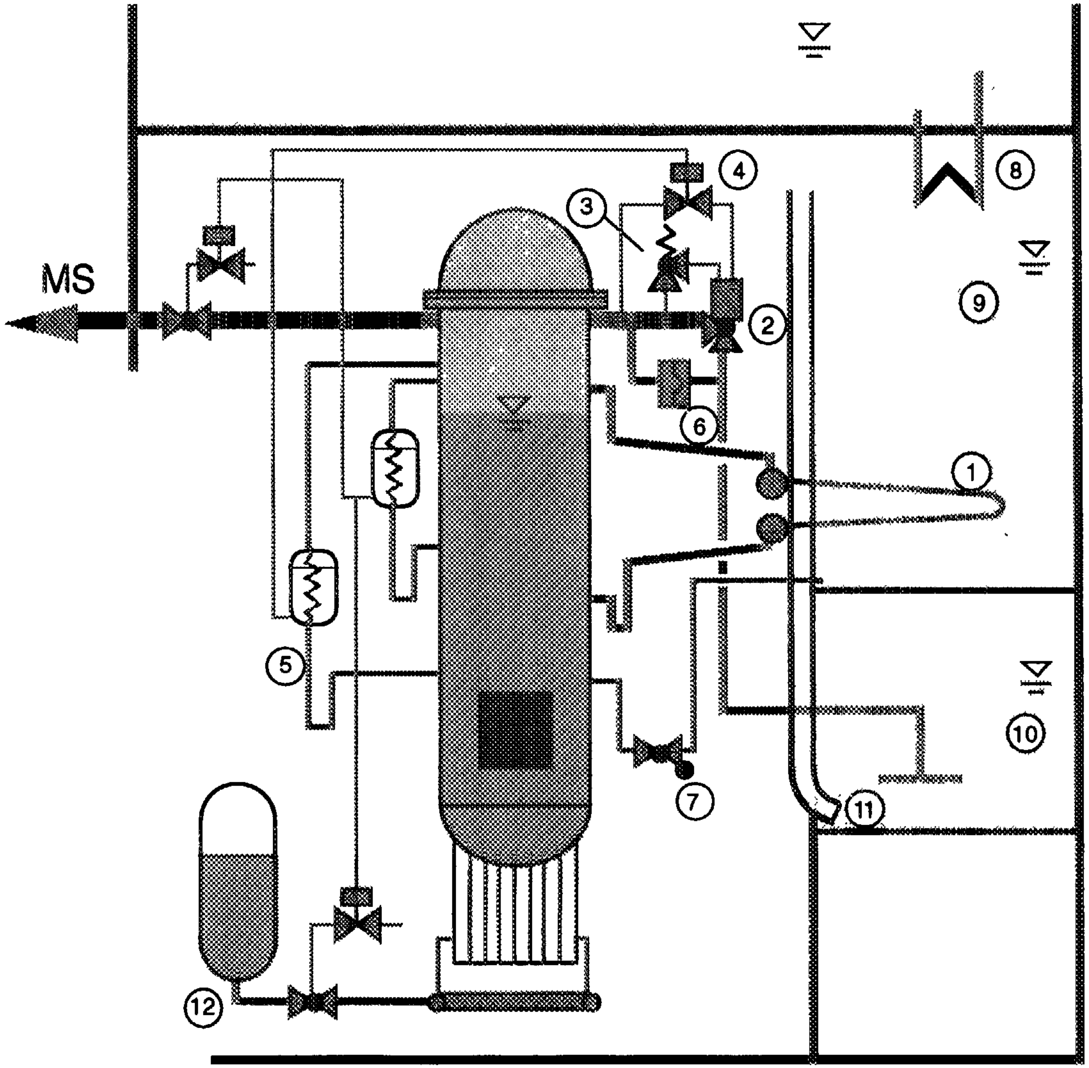

FIG. 2. Passive systems

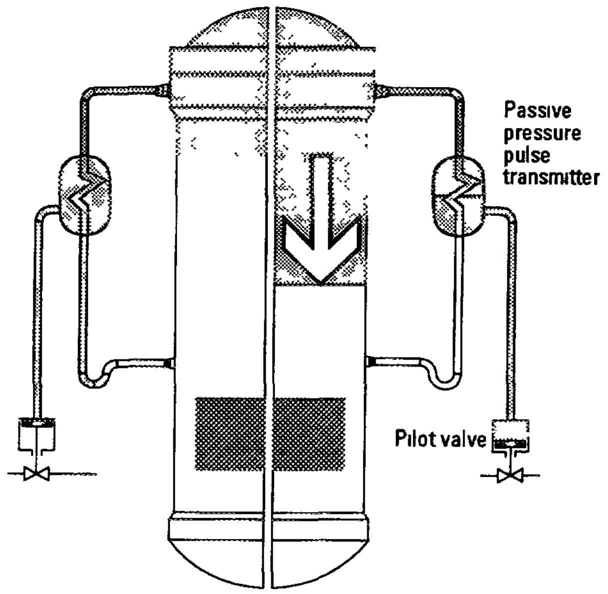

FIG. 3. BWR 600/1000
Passive pressure pulse transmitter

Reactor and Containment Heat Removal and Makeup of Reactor Coolant Inventory

- Emergency Condenser (Fig. 4)

A passive safety feature of particular importance, especially for controlling transients, are the four emergency condensers (each rated for 63 MW at a pressure of 70 bar) which are located in the core flooding pool and are connected to the RPV by non-isolatable steam discharge and return lines. The circuit of each emergency condenser contains an anti-circulation loop so that practically no circulation of condensate takes place through the open lines to the reactor during normal plant operation. Only when there has been a drop in the RPV water level does steam enter the condenser, with the resulting condensate being returned to the RPV.

The heat removed by the emergency condensers is transferred to the water of the core flooding pool and slowly raises its temperature. It takes over 12 hours for the water in the pool to reach a temperature at which it starts to evaporate. The resulting steam then causes a buildup of pressure inside the containment.

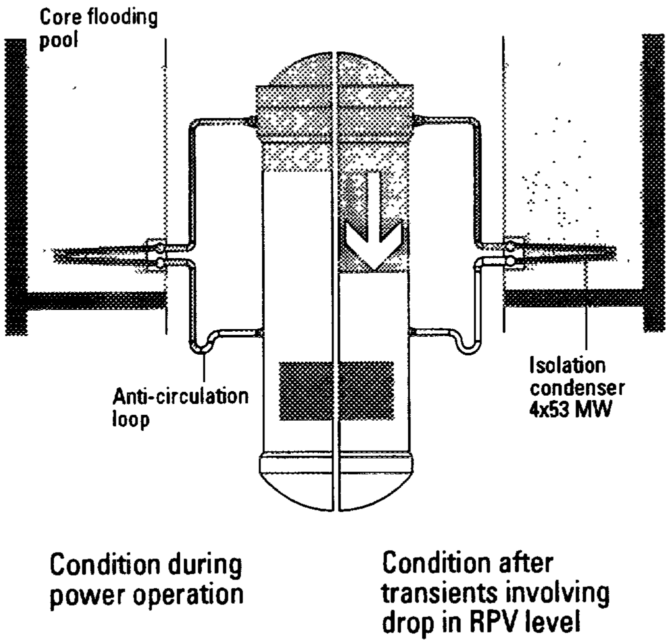

FIG. 4. BWR 600/1000
Isolation Condenser

- Gravity Core Flooding (Fig. 5)

In the event of a LOCA, steam or flashing water is discharged into the containment atmosphere. To prevent core uncovery in such a situation, passive systems are activated for supplying reactor water makeup. For this, the RPV must first be depressurized. The large wa-

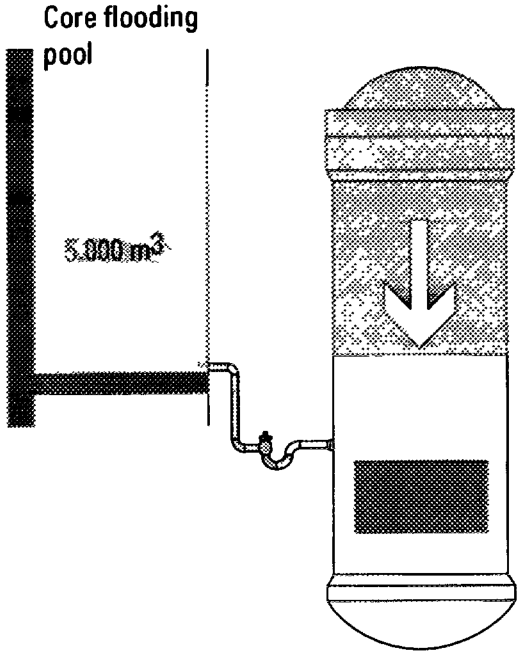

FIG. 5. BWR 600/1000
Core flooding line

ter inventory inside the RPV enables this to be done without the core becoming uncovered. Water from the elevated core flooding pool can then discharge to the RPV by gravity flow via four supply lines and self-actuated check valves. The core flooding pool contains a water inventory of approximately 5000 m^3. This volume of water is sufficient after the occurrence of a LOCA to fill both the RPV and the drywell of the containment up to a level which is then equal to that in the core flooding pool, this level being situated above the feedwater nozzles on the RPV. This not only provides a water cover over potential pipe breaks but also ensures effective cooling of the RPV exterior.

- Containment Cooling Condenser (Fig. 6)

At later stages in the accident sequences following both transients and LOCAs, the generation of steam can lead to a rise in temperature and pressure inside the drywell. In order to condense this steam and thus limit containment temperature and pressure, containment cooling condensers are provided. These discharge their condensate to the core flooding pool while the heat that they remove is rejected to the water in the dryer-separator storage pool situated above the containment. A closed, passive cooling circuit for residual heat removal is thus available inside the drywell.

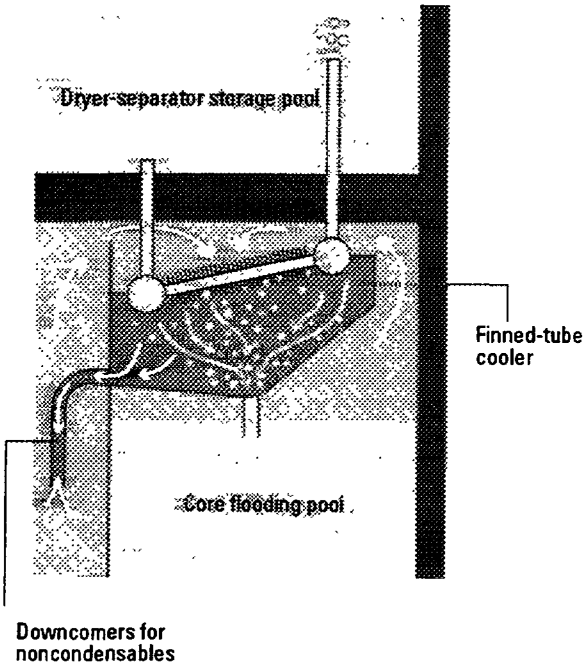

FIG. 6. BWR 600/1000
Containment cooling condenser

The containment cooling condensers provide the plant with a grace period of up to five days before there is any need for external intervention. After this period it will be necessary to make up the water inventory of the dryer-separator storage pool outside the containment, something which can be effected by simple actions.

The characteristics of these various safety features enable the reactor to stabilize itself in the event of a transient or LOCA without the need for open- or closed-loop control equipment or external actuation (e.g. by emergency power or compressed air). Existing active systems required for normal plant operation supplement the passive features provided for accident control.

4 Accident Sequences

The BWR 1000 design concept provides for accident control by both active systems usually required for normal plant operation and activated by I & C equipment, and by passive safety features which are not controlled by I & C systems. Since the accident control functions executed by active equipment (coolant makeup and heat removal) are traditional functions that are generally well known, the following descriptions will concentrate on the passive safety features.

4.1 Transients (Fig. 7)

Undesirable plant transients can be caused by, for example, the following:

- Loss of the main heat sink
- Loss of the normal feedwater supply
- Closure of the main steam isolation valves
- Stuck-open safety-relief valve.

Upon occurrence of a transient, first the reactor is scrammed either by signals from I & C systems or by passive means when the water level in the RPV starts to drop.

The spring-loaded pilot valves of the safety-relief valves operate, enabling the safety-relief valves to discharge the steam still being generated after reactor scram to the pressure suppression pool if it should not be possible to dump the steam to the main heat sink. In

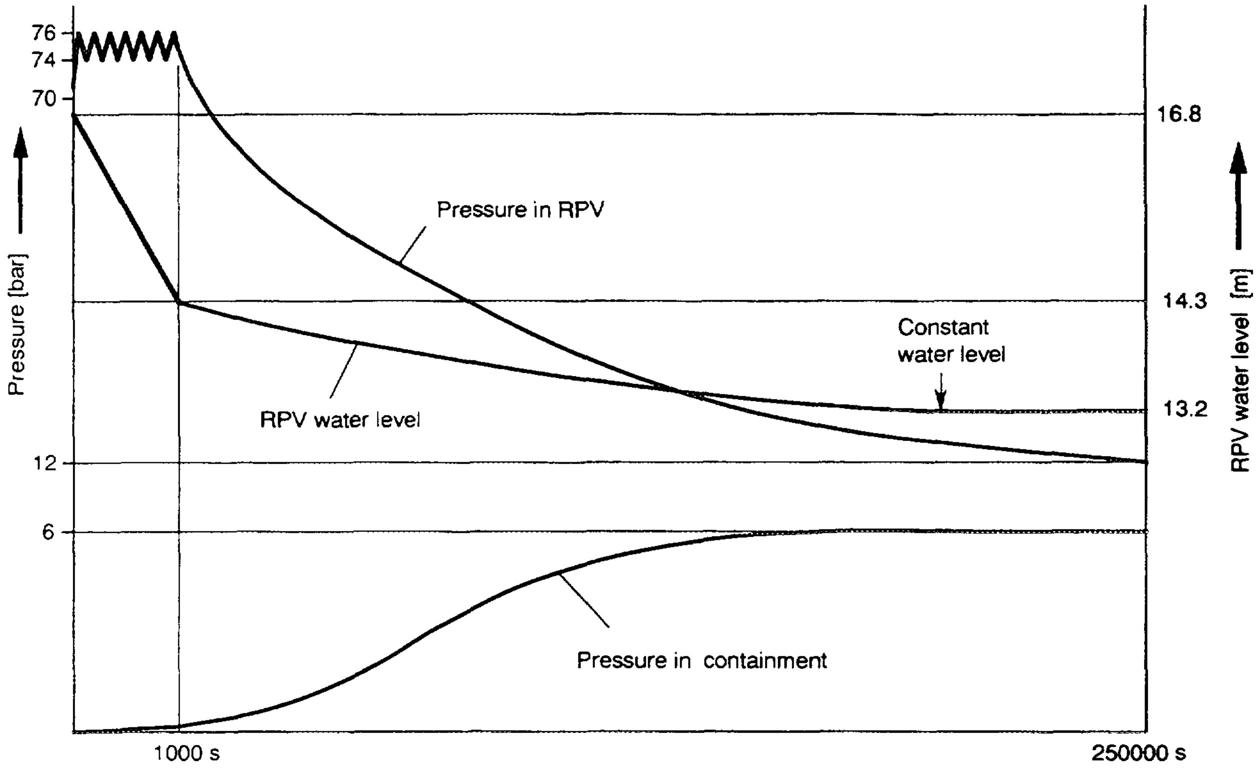

FIG. 7. BWR 1000 Time Histories of RPV Pressure, RPV Water Level and Containment Pressure During "Loss of Main Heat Sink" Transient

doing so, the valves maintain the pressure inside the reactor at its normal operating level. For higher reactor pressure levels, rupture disks are provided as a diverse means of pressure relief.

The discharge of steam to the pressure suppression pool causes the reactor water level to drop until the point is reached at which the emergency condensers can intervene to remove the decay heat generated in the reactor core to the water of the core flooding pool. The safety-relief valves then remain closed and the water level inside the RPV stops decreasing.

Medium- and long-term residual heat removal is normally performed by the active RHR trains which cool the water inventories in the pressure suppression chamber and the core flooding pool. If these active systems should fail, the containment cooling condensers start to remove heat from the core flooding pool to the atmospheric dryer-separator storage pool above the containment after approximately 12 hours, as soon as the water of the core flooding pool starts to evaporate.

This process can continue for five days without any external intervention. After this period, makeup water must be supplied to the dryer-separator storage pool, something which can be effected by means of simple temporary provisions.

If a plant transient should involve inadvertent depressurization of the reactor down to the pressure level of the pressure suppression chamber, the core flooding pool can be used to prevent core uncovery in the manner described below.

4.2 Loss-of-Coolant Accidents (Fig. 8)

Loss of coolant can be caused by pipe breaks of various sizes postulated to occur in lines carrying reactor coolant inside the containment. Provisions must additionally be made to control the effects of a 15 cm^2 leak from the bottom head of the RPV.

The good accident control characteristics of the BWR 600 described earlier in connection with plant transients likewise apply in the case of LOCAs. Occurrence of a LOCA immediately initiates reactor scram and containment isolation. This can be effected either by signals from the I & C systems or by passive means when the water level in the RPV starts to drop. This drop in the reactor water level then leads to automatic depressurization being initiated, either actively or passively. The buildup of pressure inside the drywell is reduced through discharge to the pressure suppression pool via the horizontal vents.

The loss of water via the break as well as the safety-relief valves causes the pressure and water level inside the reactor to drop until a point is reached at which passive coolant makeup by gravity discharge from the core flooding pool starts automatically. This supply of makeup water requires no I & C signals or switching operations and prevents core uncovery solely on account of the elevation differential existing between the elevated pool and the reactor.

In the event of a LOCA the coolant is retained inside the containment, the water lost through the break finally leading to equalized water levels in the core flooding pool, drywell and RPV.

In the postulated event of failure of the active RHR systems, the coolant inside the containment eventually starts to boil, leading to a further buildup of pressure. Once the containment cooling condensers start to remove heat from the containment and the resulting

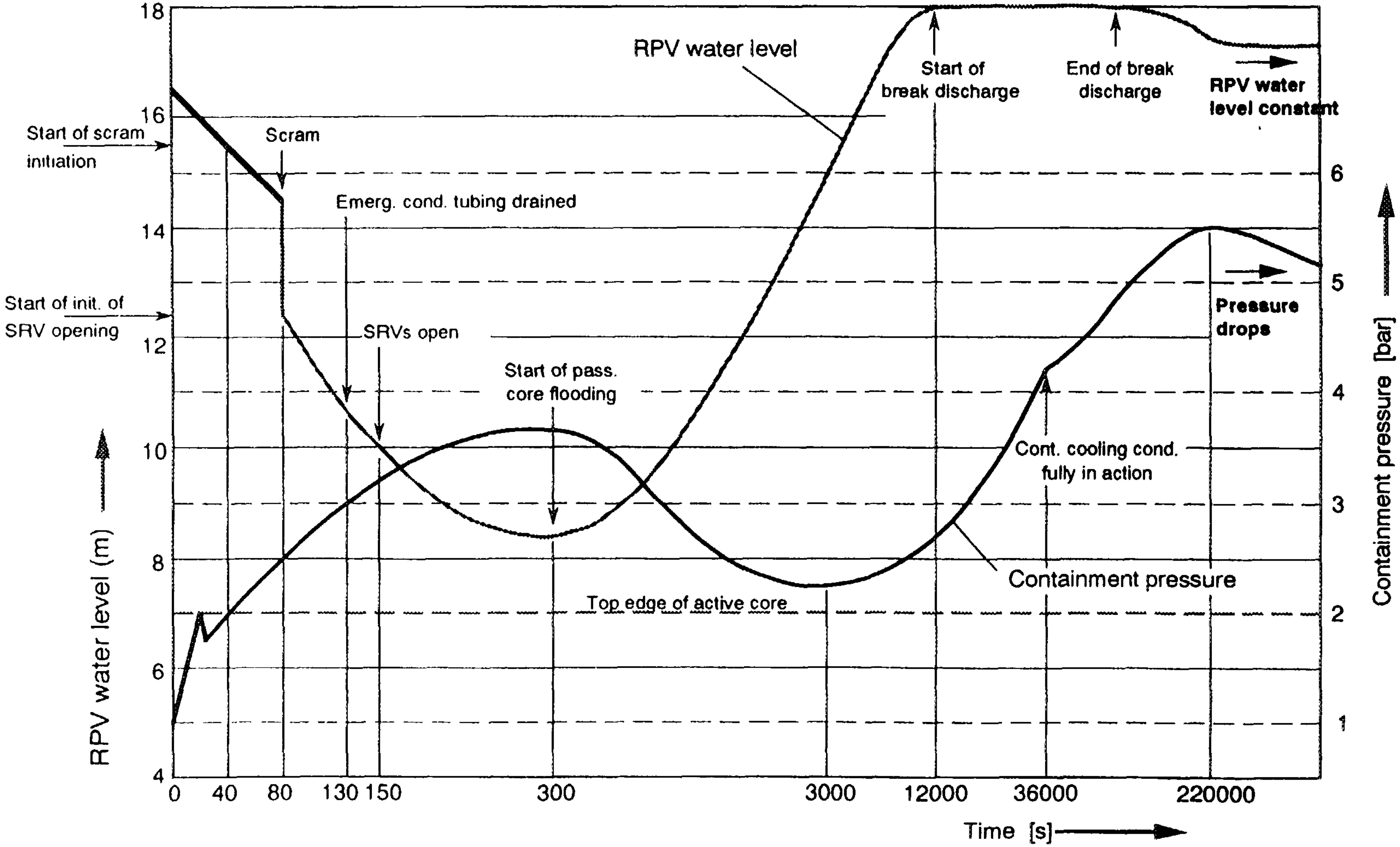

FIG. 8. BWR 1000 LOCA Due to Main Steam Line Break
Time histories of RPV water level and containment pressure

condensate begins to flow to the core flooding pool, a closed cooling circuit is established which limits any further rise in containment pressure.

As already mentioned, the containment cooling condensers discharge the heat to the external, atmospheric dryer-separator storage pool. The water contained in this pool takes at least five days to evaporate. Makeup of the storage pool water inventory using very simple provisions (e.g. fire hose connections) can extend passive control of this accident for an indefinite period of time.

5 Conclusion

As a result of the characteristics of the BWR 1000 already described, it is capable after a transient or LOCA to stabilize itself by passive means over a period of many days without intervention by active control equipment or operating personnel.

Existing coolant supply and residual heat removal systems provided for normal plant operation serve as a diverse means of accident control.

TECHNICAL FEASIBILITY AND RELIABILITY OF PASSIVE SAFETY SYSTEMS OF AC600

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Abstract

The first step conceptual design of the 600 MWe advanced PWR (AC-600) has been finished by the Nuclear Power Institute of China. Experiments on the passive system of AC-600 are being carried out, and are expected to be completed next year.

The main research emphases of AC-600 conceptual design include the advanced core, the passive safety system and simplification. The design objective of AC-600 is that the safety, reliability, maintainability, operation cost and construction period are all improved upon compared to those of PWR plant. One of important means to achieve the objective is using a passive system, which has the following functions whenever its operation is required.

- providing the reactor core with enough coolant when others fail to make up the lost coolant,
- reactor residual heat removal,
- cooling and reducing pressure in the containment and preventing radioactive substances from being released into the environment after occurrence of accident (e.g LOCA).

The system should meet the single failure criterion, and keep operating when a single active component or passive component breaks down during the first 72 hour period after occurrence of accident, or in the long period following the 72 hour period.

The passive safety system of AC-600 is composed of the primary safety injection system, the secondary emergency core residual heat removal system and the containment cooling system. The design of the system follows some relevant rules and criteria used by current PWR plant. The system has the ability to bear single failure, two complete separate subsystems are considered, each designed for 100% working capacity. Normal operation is separate from safety operation and avoids cross coupling and interference between systems, improves the reliability of components, and makes it easy to maintain, inspect and test the system.

The paper discusses the technical feasibility and reliability of the passive safety system of AC-600, and some issues and test plans are also involved.

1. DESCRIPTION OF THE PASSIVE SAFETY SYSTEMS OF AC-600

The AC-600 design is based on the Qinshan phase II standard PWR nuclear power plant (2X600 MWe). Successful experience derived from QS-II is incorporated in the AC-600 design as far as possible, but the AC-600 plant will be an improvement on QS-II. AC-600 will become a major type of reactor for the next generation 600 MWe nuclear power plants in China. It has a large safety margin of operation because of the small power density of the reactor core. The high natural circulation cooling ability due to the small flow resistance of the primary system loop is very useful for reactor core decay heat removal during accidents. The major design goals of AC-600 are (1) to enhance reactor safety and reliability, (2) to improve economics, (3) to increase nuclear plant availability, and (4) to shorten the construction schedule and lengthen plant life time.

The AC-600 advanced PWR thermal power is 1930 MW with an electrical power of 600 MW. The total reactor height is 19.1 m with a maximum out side diameter of 5.04 m. The total coolant flow rate is 32100 t/ h. The reactor coolant system consists of two loops with an operation pressure of 15.6 MPa.

Three key approaches, i.e, advanced core, passive safety systems and simplified systems, have been adopted in AC-600 design. The design features are as follows:

(1) Advanced Core

- Addition of grey control rod and utilization of Gadolinium burnable poison,
- Arrangement of stainless steel reflector,
- Reduction of core power density.

(2) Passive Safety Systems

In AC-600, the safety systems, except for the low pressure safety injection pump which is active to carry on the long-term recirculation during LOCAs, are all passive safety systems, including:

- Emergency Core Residual Heat Removal System(ECRHRS)

The passive emergency core residual heat removal system on the secondary circuit side is mainly used in the event of station blackout, main steam line rupture or loss of feedwater. An independent emergency core residual heat removal train consists of one emergency feedwater tank and one emergency air cooler as well as associated piping, valves and instruments. Each reactor coolant loop has one train. Two trains constitute the AC-600 ECRHRS. When station blackout occurs, the decay heat generated in the reactor core can be removed through use of natural circulation flow in the primary coolant system, the secondary coolant and the atmosphere.

- Safety Injection System

The AC-600 safety injection system, similar to that of the existing PWR plant, is divided into a HP injection subsystem, a MP injection subsystem and a LP injection subsystem as well as the corresponding recirculation systems. The HP injection subsystem mainly consists of two core makeup tanks in which water pressure is the same as the reactor coolant pressure. The MP injection subsystem mainly consists of two accumulators with an operating pressure of 5.2 MPa. The HP and MP injection subsystems are all passive.

- Containment Cooling System

This system, utilizing completely passive equipment, consists of the containment cooling water storage tank located on the top of the containment and cooling water sprayers. It eliminates the need for the containment spray system in the current PWR plant as well as the need for an intermediate circulation cooling medium such as the component cooling water, resulting in a saving in investment and in an improvement in system operational reliability. The system is used to remove the heat from the inside to the outside of the containment during LOCA or main steam line rupture located inside the containment. First, the water in the tank on the top of the containment will be sprayed onto the surface of the steel shell of the containment by gravity, cooling the shell so as to decrease the pressure and the temperature. After emptying the tank, the natural circulation flow of air through the annulus between the steel shell and the concrete shell can remove the heat continuously. At the same time, the low head safety injection/recirculation pumps which are installed in the containment sumps can withdraw the borated water from the sumps into the reactor coolant system. The water absorbs the core decay heat and flows out through the break point (in LOCA conditions).

These passive systems guarantee the completion of the safety functions - residual heat removal, RCS inventory control, short-term LOCA safety injection, long-term LOCA recirculation, containment spraying and cooling following a transient and/or accident.

(3) Simplified System

- The SG and pumps are connected into a single structure, eliminating the U-shape cross-over leg of the coolant pipes, improving the post-LOCA safety, decreasing the resistance in the primary circuit and enhancing the natural circulation capability of the primary circuit;
- The use of the passive safety systems and the decrease of boron concentration in the RCS eliminate and/or simplify such system as the auxiliary feedwater system, and the boron recycle system as well as the HP safety injection pump.
- Most of safety grade components are arranged within the containment, resulting in the reduction in the safety graded buildings volume and capital cost.

Except the three major features mentioned above, the advanced I&C technology and modular construction approach are also employed to improve the AC-600 performance and reduce the construction schedule and cost.

The schematic diagrams and drawing of the passive system of AC-600 are shown in Fig.1, Fig.2, and Fig.3. Nomenclature, list and quantity of the main components included in the system are given in Table. 1.

2. Feasibility and reliability

The reliability of the passive safety system refers to the ability of the system to carry out function under the prevailing condition when required. The feasibility shows the reliability, maintainability, availability and the economic advantages of the system. The feasibility includes technical realization, economics, public acceptance and political support by the government, etc.

Feasibility and reliability are closely related to each other and should be considered comprehensively in the system design. The reliability is not the best as it is set the highest. In China, the development of nuclear power plant is restricted by its economies and technology level in the country. The economic estimation of the system is one important part of its feasibility research.

Based on the specialized safety systems of the standard nuclear power plant, the passive safety system design is feasible for a developing country. So this design can absorb

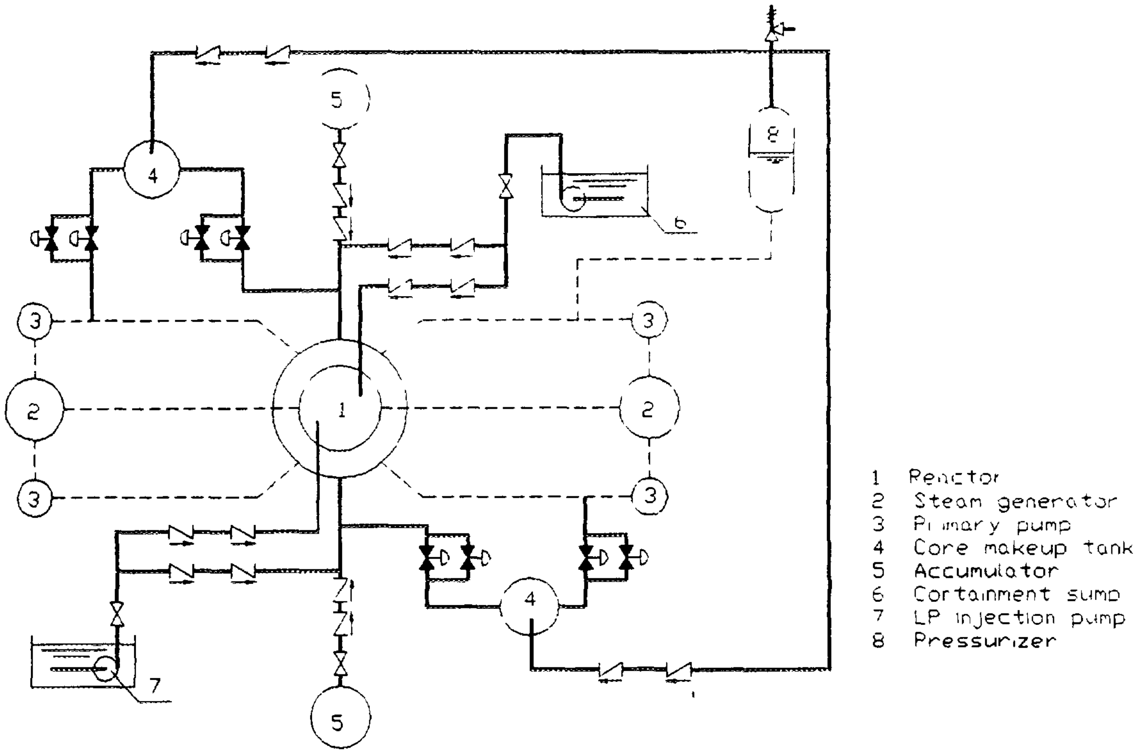

FIG. 1. Safety Injection System.

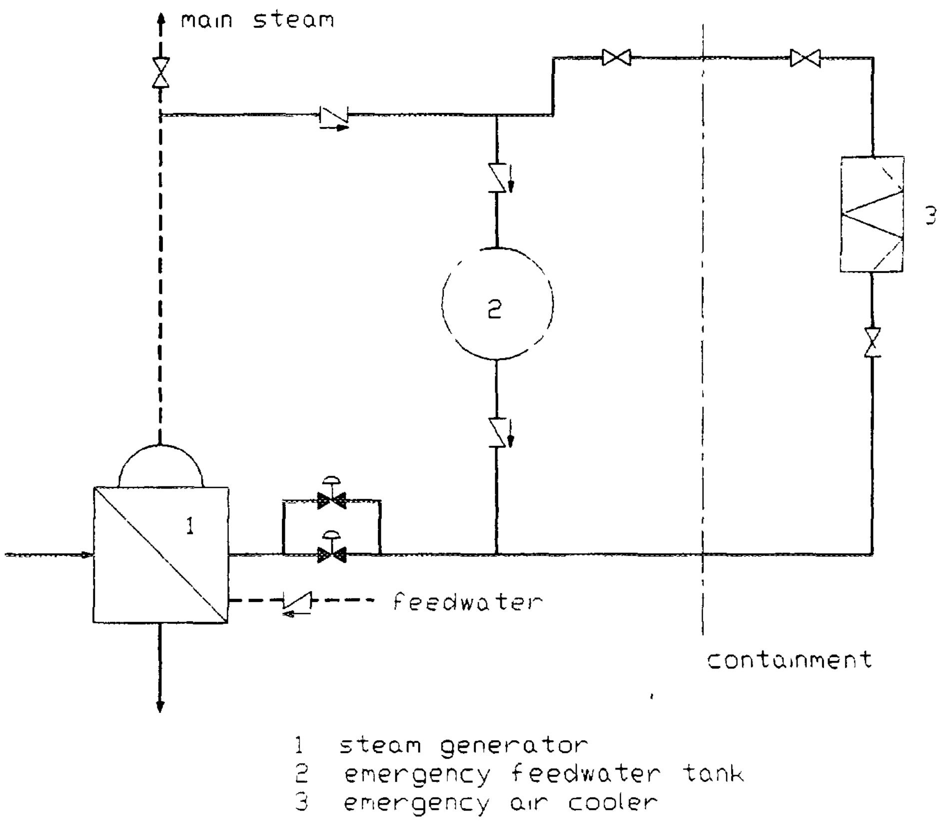

FIG. 2. Emergency Core Residual Heat Removal System.

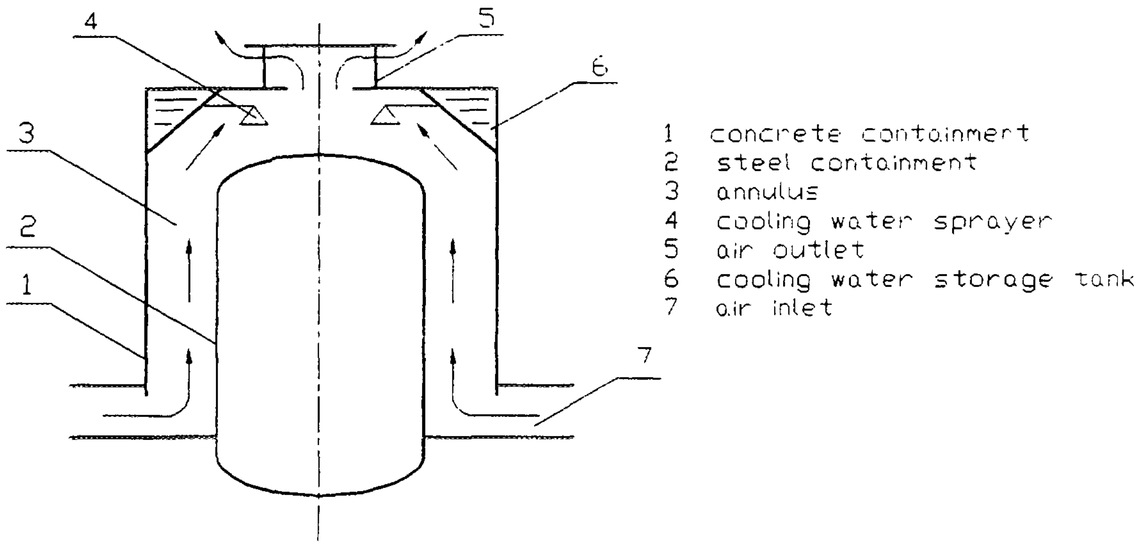

FIG. 3. Containment Cooling System.

Table 1

No:	Name	Quantity
1	core makeup tank	2
2	accumulator	2
3	emergency water tank	2
4	special sump	2
5	low pressure safety injection pump	4
6	emergency air cooler	2
7	water storage tank	1

successful experiences derived from the standard NPP and lessens the large research investment. In the system design, in order to improve the reliability of the system, the following subjects must be considered:

(1) Under the precondition of ensuring the function, the system should be designed simplified and standardized as much as possible. The simplification of the system is involved not only in its components but also in its operation.

(2) To meet the single failure criterion, it is necessary to avoid the cross and intersection between systems and to make the systems independent.

(3) The system is designed to a high level of safety which is obtained by an adequate level of redundancy in key components. All of these improve the reliability of the key components as well as the system. The key components include:

- The active components in the passive safety systems, such as the LP injection pump;
- The equipment with mechanical action when it is operating, such as the fail open valves of the ECRHRS;
- Instruments for inspecting and monitoring.

(4) The monitoring and control system provides an automated diagnosis of the state and operating conditions of the NPP. Monitoring and presentation of the information on the reactor coolant system, on all the safety-related systems, on the containment, on all operating conditions of the NPP and remote control of these systems are carried out. A post-accident monitoring system is provided to estimate the state of the NPP facilities and to present information important to safety.

(5) Integrity of the passive safety systems is provided by appropriate design and provisions, such as in-service inspection, monitoring, test and quality control. All components of the passive safety systems are subject to strength calculation under design conditions, and to stress, strain and seismic analyses under design basis accidents. The detection of leak before break is also used in the operation of system.

All the above mentioned are not the only subject for improving the system reliability, there are still other items such as power source, inspecting signal, etc. whose failure may lead to decreasing the system reliability. To improve the reliability of individual components within a system is to improve the system reliability as a whole.

3. FEASIBILITY AND RELIABILITY OF AC-600

3.1. Features of the system design of AC-600

(1) The design of the system follows some relevant rules and criteria used by current PWR plants. The system has the ability to withstand single failures. Two complete subsystems have been designed, each designed for 100% working capacity.

(2) The separation between normal and safety operation makes the systems dedicated to their function and avoids the use of common components in different systems, improves the reliability of components and makes it easy to maintain, inspect and test the system.

(3) The passive core makeup tank (CMT) takes the place of the HP injection system used in current standard PWR nuclear plants, so the HP injection pump and its mechanical/electric system are all eliminated. The CMT can operate at full pressure of the reactor core coolant, injecting rapidly and efficiently plenty of cold water into the core by gravity, avoiding the failure caused by electric power or other active component failure and the difficulty of water injection into the core due to the high coolant system pressure.

(4) The passive emergency core residual heat removal system replaces the secondary side residual heat removal system which consists of an emergency feedwater system and emergency steam release system. This replacement avoids failure due to a secondary system accident. Because the system is designed to operate with closed natural circulation by natural laws, such as evaporating, convection and gravity, its reliability is improved and its operation is not restricted by volume of a water source or by time.

(5) The passive containment cooling system replaces the containment spray system, and eliminates the boron injection subsystem used in standard NPP. The containment sump takes the place of the refueling water tank and the recirculation sump which serve both as the water sources of LP injection and recirculation, and this replacement

eliminates the exchange between two operation models mentioned above, simplifies the specialized safety facilities, makes the operation convenient and reliable and improves the inherent safety of the system.

(6) The components of the passive safety system, located on the primary coolant system side, are arranged in the steel shell of the containment; such a measure reduces the possibility of radioactive medium leakage from the system and/or component, and improves the inherent safety of the NPP.

(7) In order to improve the feasibility of the safety injection systems, AC-600 design is based on the principle of a combination of passive and active features. There are two full pressure CMTs, two accumulators and four low head safety injection /recirculation pumps which are installed in the containment sumps. In a large LOCA, the flow rate into the RCS from a CMT is larger than that from a high head safety injection pump in the conventional design. It is necessary for AC-600 to use an active pump which can perform the functions of the low head safety injection/recirculation system.

(8) The measures of increasing the vertical distance between the steam generators and the reactor core and reducing the primary flow resistance are used in the AC-600 design to increase the natural circulation cooling flow rate of the primary coolant system. If the reactor operates at 25% of the rated power, the natural circulation flow rate is about 4852 t/h (15.12% of the rated flow rate) after the reactor coolant pumps shut down. The natural circulation flow rate increase is a very important part of AC-600 passive safety.

(9) For operation of the safety injection system, except the subsystems of low pressure active safety injection and recirculation, sources of alternating current are not required. The air-operated valves needed for the function of emergency heat removal are air driven by compressed air from compressed air storage tanks. The power supply of the subsystems for low pressure active safety injection and recirculation are provided by diesel-generators or by offsite power source (during the recirculation stage after LOCA). In the passive emergency core residual heat removal system, the fail open valves on the piping are driven also by compressed air.

(10) The passive safety systems of the AC-600 design are based on the specialized safety systems of current NPP. The design and manufacture of the components of the passive safety systems such as tanks, valves, are all mature. The operating conditions of the components are good and based on previous experience and the economy is also good due to reduced need for research and development investment. All the above mentioned prove that the feasibility of the system is improved as measured by its reliability.

3.2. Failures of AC-600

The following accidents will be analyzed for the AC-600 design in order to provide some important parameters for AC-600 engineered safety system design and safety assessment.

- Increase of heat removal by the secondary system;
- Decrease of heat removal by the secondary system;

- Decrease of reactor coolant system flowrate;
- Reactivity and power distribution anomalies;
- Increase in reactor coolant inventory;
- Decrease in reactor coolant inventory;
- Radioactive release from a system or component;
- Anticipated transients without scram.

List of beyond design basis accidents (severe accidents):

- Total loss of ultimate heat sink;
- Loss of main and auxiliary feedwater;
- Station blackout;
- Loss of safety injection pumps.

4. MAJOR RESEARCH SUBJECTS OF AC-600

As it is a new concept and replaces the specialized safety systems used by the standard NPP by passive safety systems, there are still many problems about the feasibility and reliability of the systems to be researched.

During the 8th five-year plan (1991-1996), NPIC undertook the AC-600 overall design and research and AC-600 key technology test and research development subjects assigned by the State Scientific and Technological Commission and CNNC. Some of these subjects relating to the passive safety systems are as follows:

1) Integrated design and research on AC-600 main equipment, passive safety systems and simplified systems;
2) Full-pressure core makeup tank test and research;
3) Passive containment cooling system wind tunnel test and research;
4) Secondary side passive emergency core residual heat removal system test and research;
5) Passive LP safety injection/recirculation system test and research;
6) Research on the system redundant principle;
7) Test and research on instruments for inspection and control;
8) The system failure model and reliability research.

During the 9th five-year plan (1996-2000), NPIC will place the emphasis on the key technology peculiar to AC-600 and engage in design and test on the design technology, advanced nuclear power techniques and key equipment encompassing 33 subjects. By the year 2000, NPIC will have completed the AC-600 nuclear plant preliminary design, key technology research and key tests with a good knowledge of the complete design technology to the extent that a utility order for an AC-600 plant can be accepted and AC-600 engineering conditions will be essentially prepared. The major design and test subjects on the passive safety systems are as follows:

(1) Tests and research on the passive containment cooling system entirely;
(2) Mock-up test and research on the passive safety systems;
(3) Typical nuclear grade valve development;
(4) Tests and research on pumps immersed in water.

The research on the above subject is proceeding well. By the end of the 8th five-year plan, NPIC will have completed not only the research on some subjects but also the AC-600 PWR plant overall design.

In the design and research subjects, many up-to-date techniques are used for core design optimization, the passive safety system design and the simplified system design. The computer codes and data base will start to be established for AC-600 accidents analysis.

In the test subjects, emphasis is placed on the emergency core residual heat removal system, passive containment cooling system, etc, together with corresponding tests, and research reports to provide a test data base for use in the safety review.

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A FEASIBILITY ASSESSMENT FOR INCORPORATION OF PASSIVE RHRS INTO LARGE SCALE ACTIVE PWR

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Abstract

A feasibility study was carried out for the possible incorporation of passive RHRS (Residual Heat Removal System) into a large-scale of active PWR plant. Four kinds of system configurations were considered. For each case its performance and impacts on plant safety, cost, licensing, operation and maintenance were evaluated. The evaluation came up with a finding of PRHRS with a gravity feed tank as most probable design concept. However, considering rearrangement of structure and pipe routing inside and outside containment, it is concluded that implementation of the PRHRS concept into well developed active plants is not desirable at present.

1. Introduction

The efforts of worldwide nuclear industries toward improving the safety of nuclear reactors can be categorized into two basic but quite different approaches; namely, evolutionary improvements to the current large LWRs and revolutionary path with passive safety concepts, The key objectives of these two approaches, however, lies in the same target; that is, achieving the increased general public acceptance in nuclear power plants by enhancing the safety itself of those reactors compared with the currently existing reactors.

In active plants decay heat is removed by the active Residual Heat Removal Systems(RHRS) in case of design basis events, but for beyond design basis events such as loss of active RHRS, total loss of feed water and station blackout, alternative path of decay heat removal would be necessary. At the case of beyond design basis events in advanced evolutionary plant designed in accordance with the EPRI-URD(USA), core decay heat removal is accomplished by a one-through cooling process in a way that water is injected directly into the reactor vessel downcomer via normal safety injection system and is bleeded through vent valves of SDS(Safety Depressurization System) to the IRWST(Incontainment Refueling Water Storage Tank). IRWST cooling is provided by the safety grade component cooling water system through RHRS heat exchanger or containment spray system heat exchanger[1]. Once feed and bleed operation starts, the contaminated reactor coolant would be relieved to the containment. The contamination of inside containment from feed and bleed operation may have plant operator be reluctant to actuate the safety depressurization valves even when required. On the other hand, PRHRS does not discharge reactor coolant inside containment. Thus to reduce operators' burden of SDS operation and to reduce the impacts of SDS operation by operator mistakes, PRHRS needs to be considered as a supporting system to the existing active one. PRHRS design concept has been widely used in small passive plants such as AP600, SIR and MS300/600. The PRHRS in AP600 is designed as safety-related system and removes residual and/or decay heat from the primary coolant system to the containment through PRHR heat exchangers(HXs) submerged in the IRWST[2]. In the SIR and MS300/600 design , the decay heat removal system is not the same as the system in AP600, but the design concept of the decay heat removal system is based on passive functions.

The purpose of this study is to look for the possible incorporation of the PRHRS into the large-scale active plant to back up the existing active system by assessment its performance, safety benefits, and impacts on cost and licensing.

2. System design and performance analysis

The existing safety-related systems of active plants(advanced evolutionary PWR) satisfy all the acceptance criteria of licensing regulations for the licensing design basis events and even for the beyond design basis events, e.g. total-loss-of-feedwater, etc. In active plants, PRHRS can be designed as a safety-related or non-safety-related system. Considering the satisfactory level of the active plant safety and the cost for development of PRHRS, the PRHRS need not to be a safety-related system. PRHRS designed as non-safety-related system can provide an additional capability for the removal of residual or decay heat from RCS in the case of beyond design basis events before actuation of the SDS isolation valve for core cooling.

To implement PRHRS into the large-scale active plant, System80+ design is selected as a reference plant. The basic concept of the PRHRS considered for the study is to transfer the primary heat load by natural circulation through PRHRS heat exchanger to secondary system. When the normal decay heat removal system including steam generator is no more available, the reactor coolant flows from hot leg to PRHRS and return to steam generator outlet chamber by density difference and gravity force. The PRHRS then transfers the primary decay heat to the secondary system via PRHRS heat exchangers. However for the heat removal from primary loop to the ultimate heat sinks through PRHRS heat exchanger, there may be various design concept to be considered. For comparison with system performance and cost impacts of each approach, all the case of PRHRS heat exchanger in the primary system design concept is located at the elevation of 146′ and the the specification of heat exchanger is similar to AP600′s, that is, 0.75″ in tube outside diameter, 0.11″ in tube thickness and 1.5″ pitch of each tube. Furthermore Westinghouse test results of AP600 PRHRS HX are used as for the boiling heat transfer coefficients. The primary pipe is selected 14″ nominal diameter with schedule 160 which is the same as the

System80+ design safety injection pipe. The system performances were analyzed by RELAP5 MOD3 computer code by modelling the reactor vessel, one-loop of RCS and connected PRHRS.

For heat removal of primary system to the ultimate heat sink, the most easily applicable approach is to remove the containment heat via containment spray system. Since the containment heat removal performance by the containment spray system already has been verified in operating active plants, the system performance would be very effective. However, if the containment spray system actuates, the sprayed water with a high boron concentration may significantly contaminate the equipments inside containment. Thus this concept has little advantage compared with SDS.

To prevent the equipments inside containment from exposure to highly concentrated boron, CCW(Component Cooling Water) can be supplied directly to the PRHRS heat exchanger which is a very similar approach to the active normal RHRS except using RHRS pump. Since the mechanical pump seal is very weak against the high RCS temperature for a long time, the primary heat is removed by natural circulation without RHRS pump(Fig-1). The PRHRS heat exchanger is located elevation 146′ in horizontal direction. The CCWS is supplied at 8000 gpm with 120°F which is the same condition for the active RHRS operation. For three cases of heat exchanger with 4m in tube length but different number of tubes(1000,1500,2000), the performance was evaluated. As shown in Fig-4, The primary temperature reaches the normal RHRS operating condition within 14 hrs. Considering the active RHRS performance of System80+ which takes 6 hrs for cold shutdown after RHRS actuation with single train, this approach accomplishes the safe shutdown requirements even for the beyond design basis events. The secondary temperature always lies below 100C. This means that CCWS design pressure and temperature of reference plant are not affected by this concept. However, CCWS heat load is almost twice the normal operation condition so that the heat removal capacity of CCWS should be resized for this design concept.

Removing decay heat from PRHRS to the ultimate heat sink without using the existing active system, the gravity feed concept from an external tank to the PRHRS heat exchanger would be applicable(Fig-2). When PRHRS actuated, hot reactor coolant comes into PRHR HX primary side and the heat transfer at outside heat exchanger will then be made by boiling mechanism. The steam generated in the heat exchanger goes out to external tank upperside to make dynamic pressure balance. Five cases were evaluated for heat exchanger sizing. As shown in Fig-5, the hot leg temperature of all the cases reaches normal RHRS operating condition within 14 hrs. The preliminary sizing of components necessary for this concept estimates 4m in minimum heat exchanger tube length and about 2.5m^2 in cross section area of heat exchanger. The required external tank size is dependent on PRHRS operation time as shown in Fig-7.

Another concept considered is to adopt a secondary circuit which has external heat exchanger and external water tank located highly elevated level outside

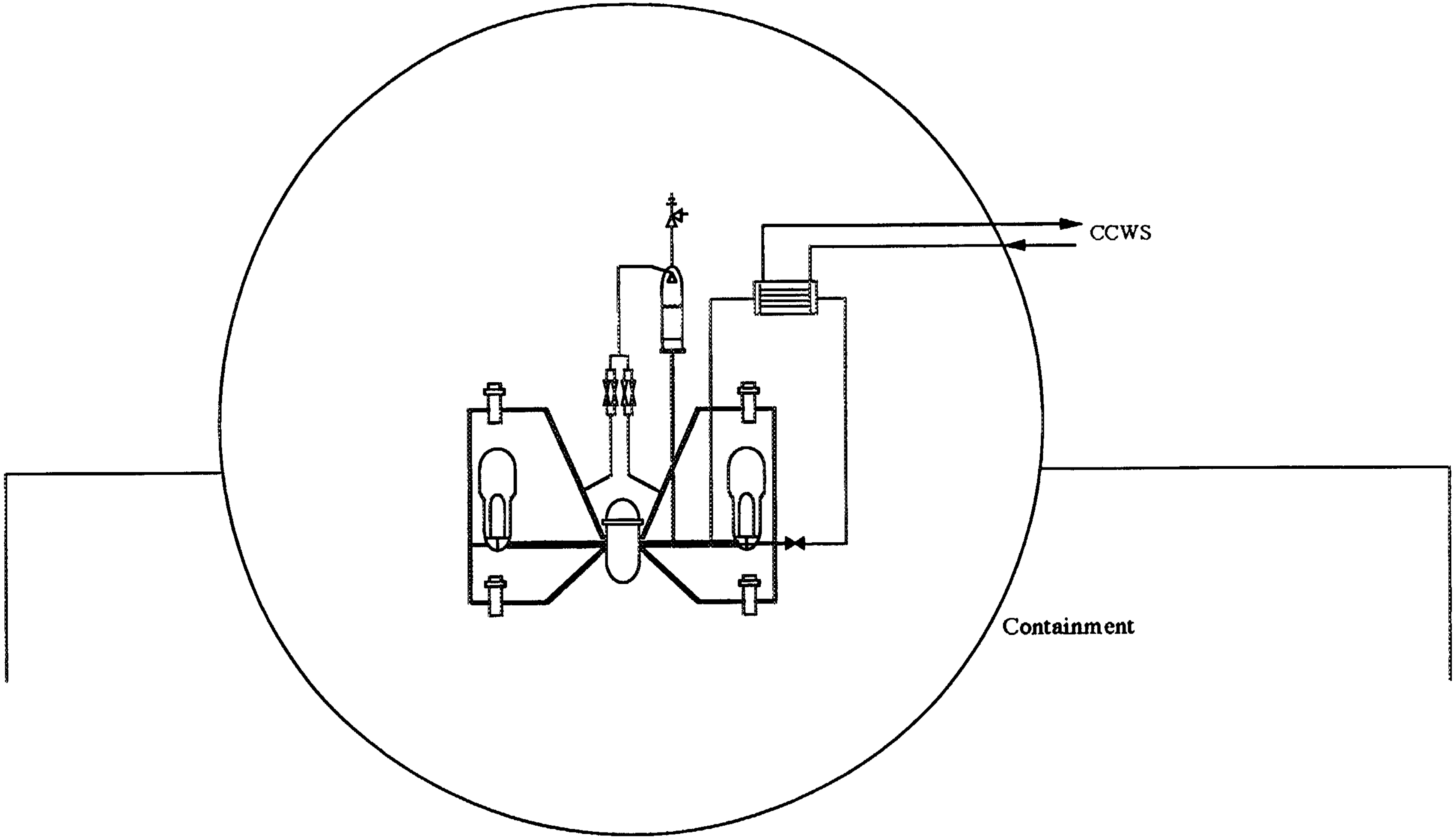

Fig-1 PRHRS supported by CCWS

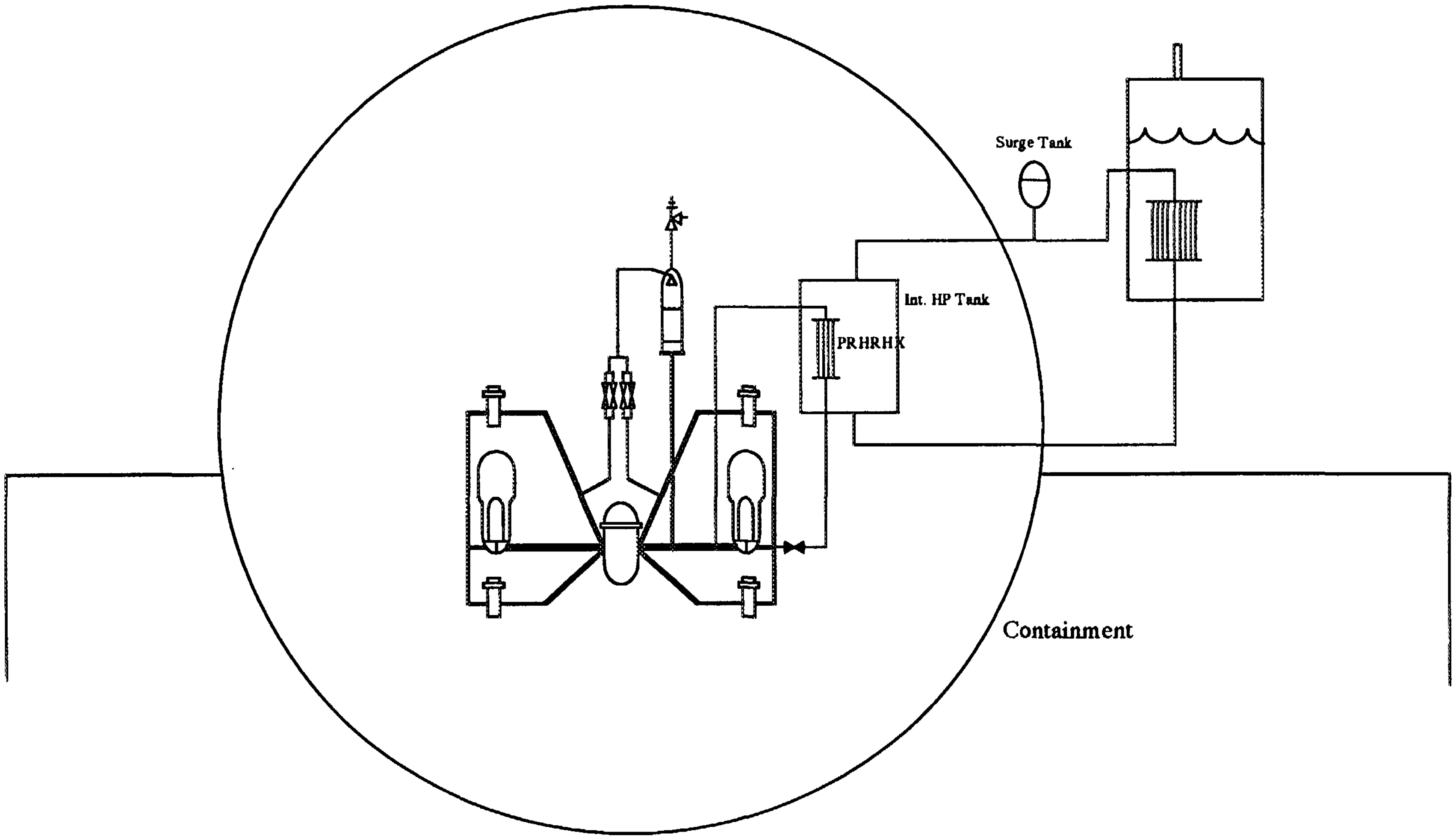

Fig-3 PRHRS with Secondary Circuit

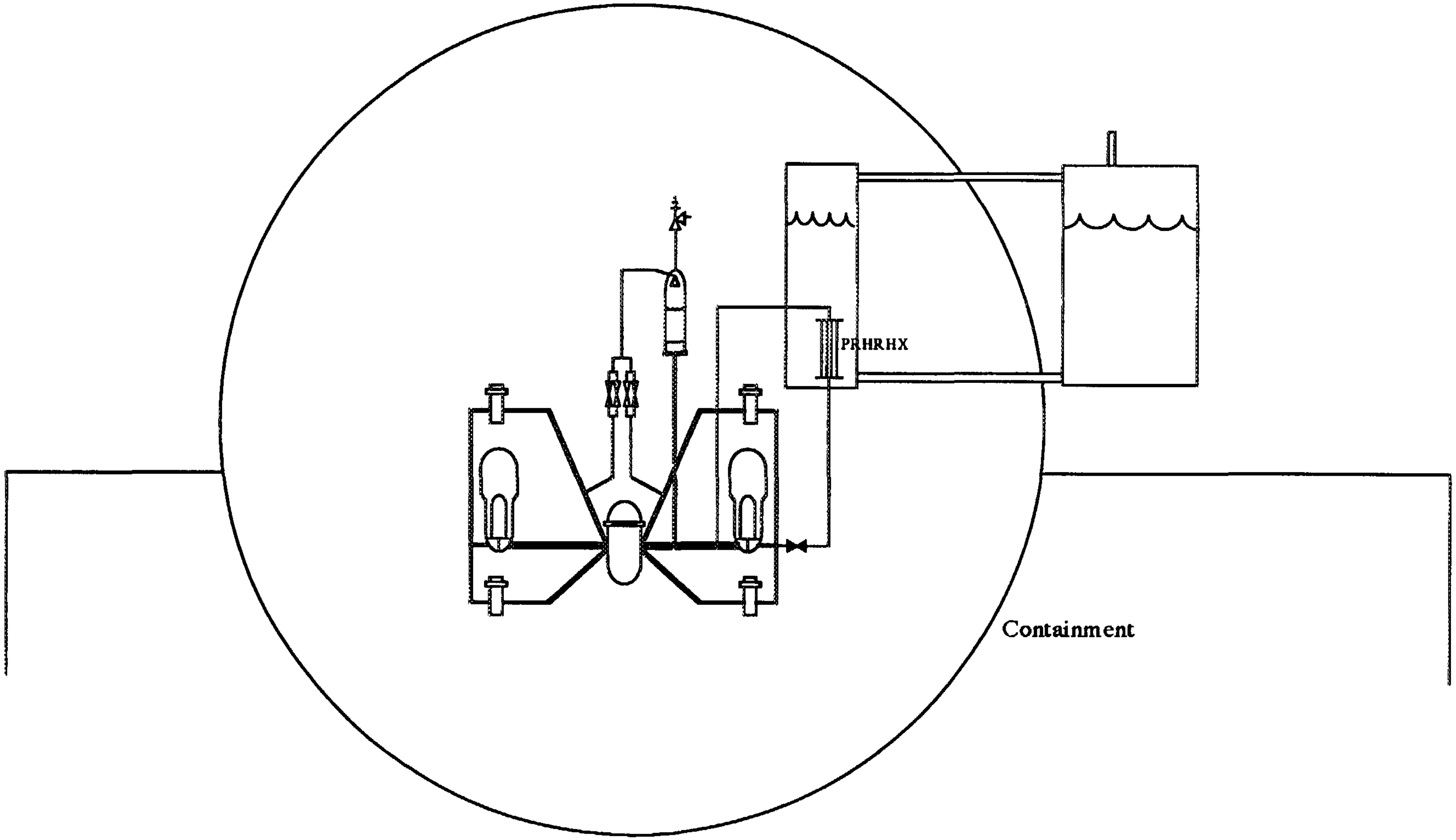

Fig-2 PRHRS with Extenal Gravity Feed System

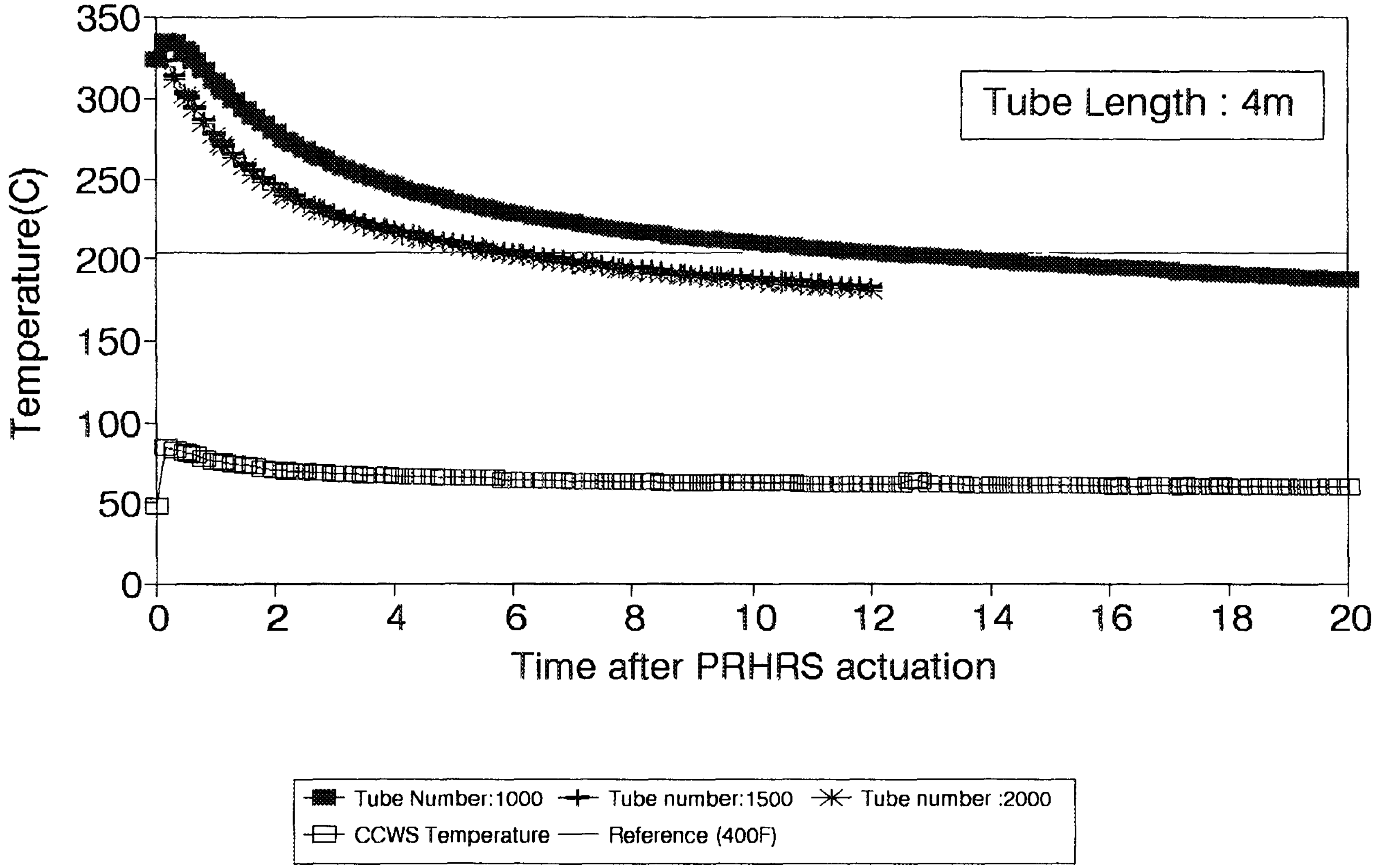

Fig-4 PRHRS Performance Curve
(PRHRS with CCWS)

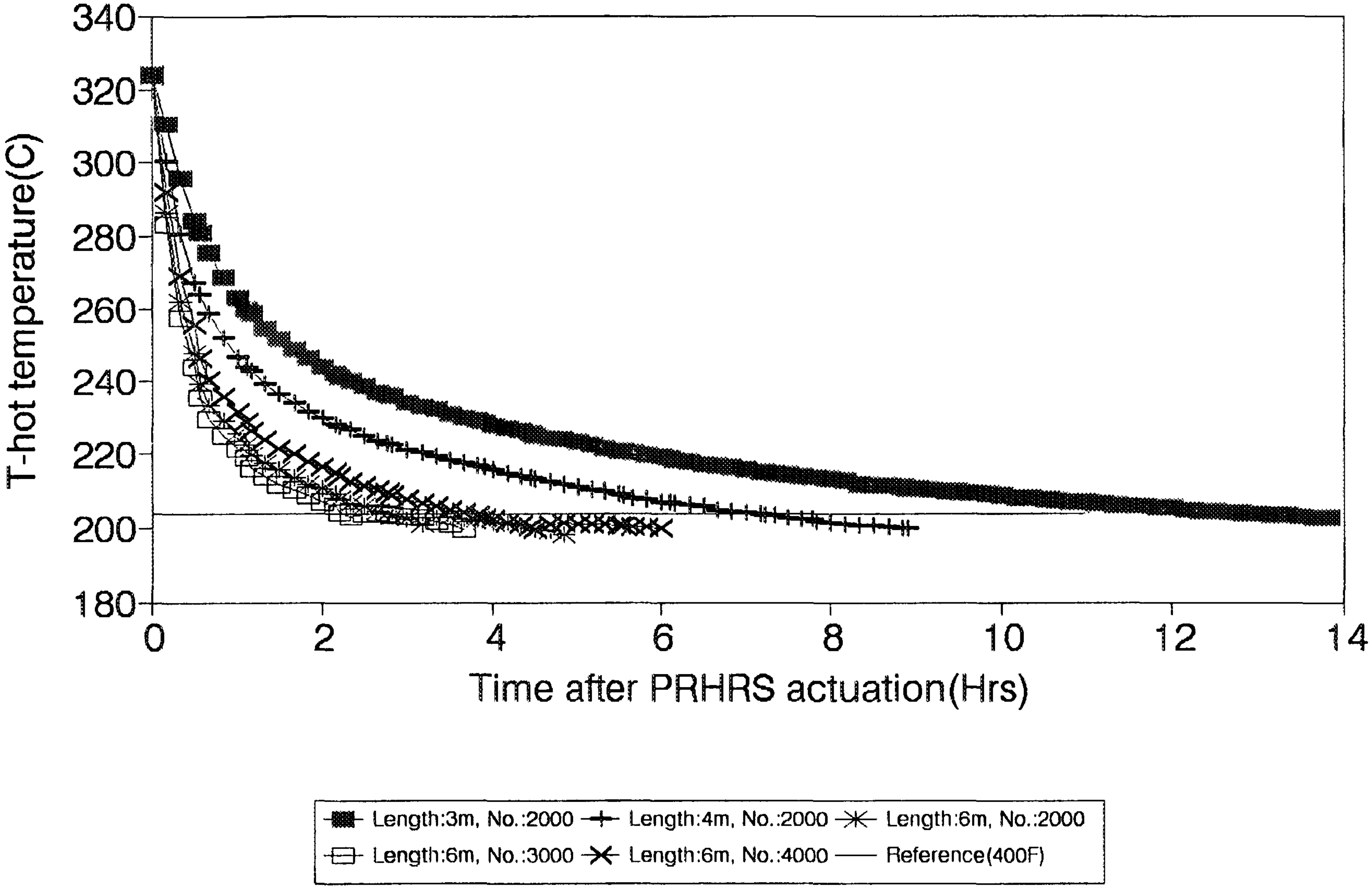

Fig-5 PRHRS performance curve
(PRHRS with Gravity Feed System)

containment(elevation 170′) (Fig-3). The transferred heat to the PRHRS heat exchanger inside containment is rejected to the ultimate heat sink through secondary loop by natural circulation. Inside PRHRS heat exchanger single phase heat transfer occurs between primary and secondary coolant but boiling heat transfer will occur in the external water tank where an external heat exchanger is submerged. The preliminary sizes of primary and secondary circuit heat exchangers for this concept were assessed. The external heat exchanger consists of 4000 tubes with 6m in length whose bottom is located at 170′ elevation. Three cases for the number of heat exchanger tubes such as 2000,3000 and 4000 were evaluated. All the case can meet safe shutdown requirements for the passive plant specified EPRI ALWR-URD but the performance is not better than the gravity feed system concept(Fig-6). During the PRHRS operation of this concept, the high temperature and pressure of secondary system should be maintained to achieve effective heat transfer in the secondary circuit. The required volume of the external water storage tank with respect to operation period is shown in Fig-7.

Comparing the performance analysis results of all the design concepts with the same heat transfer area, The PRHRS supported by gravity feed tank and PRHRS supported by CCWS shows tbe better performance for cooling of the primary system. However, in system design point of view, PRHRS supported by CCWS may has impacts on the CCWS heat removal capacity and PRHRS with secondary circuit needs more components than the system with gravity feed system.

3. Assessment of Impact

The implementation of PRHRS into the well designed active plants may cause some impacts on safety, economy, licensing and etc., either in a positive or negative direction. The assessment of impacts was thus carried out focussing mainly on the safety and cost aspects.

3.1 safety benefit

PRHRS will provide as positive effects on the beyond design basis events such as,

0 Loss of active residual heat removal system including steam generator like total loss of feed water,

0 Station blackout events.

In the case of total loss of feedwater event, the actuation of safety depressurization valves can be avoided by PRHRS actuation. PRHRS also removes the residual heat from RCS at the event of station blackout so that the possibility of RCP seal failure can be reduced. accordingly the above two events, PRHRS will improve the plant safety.

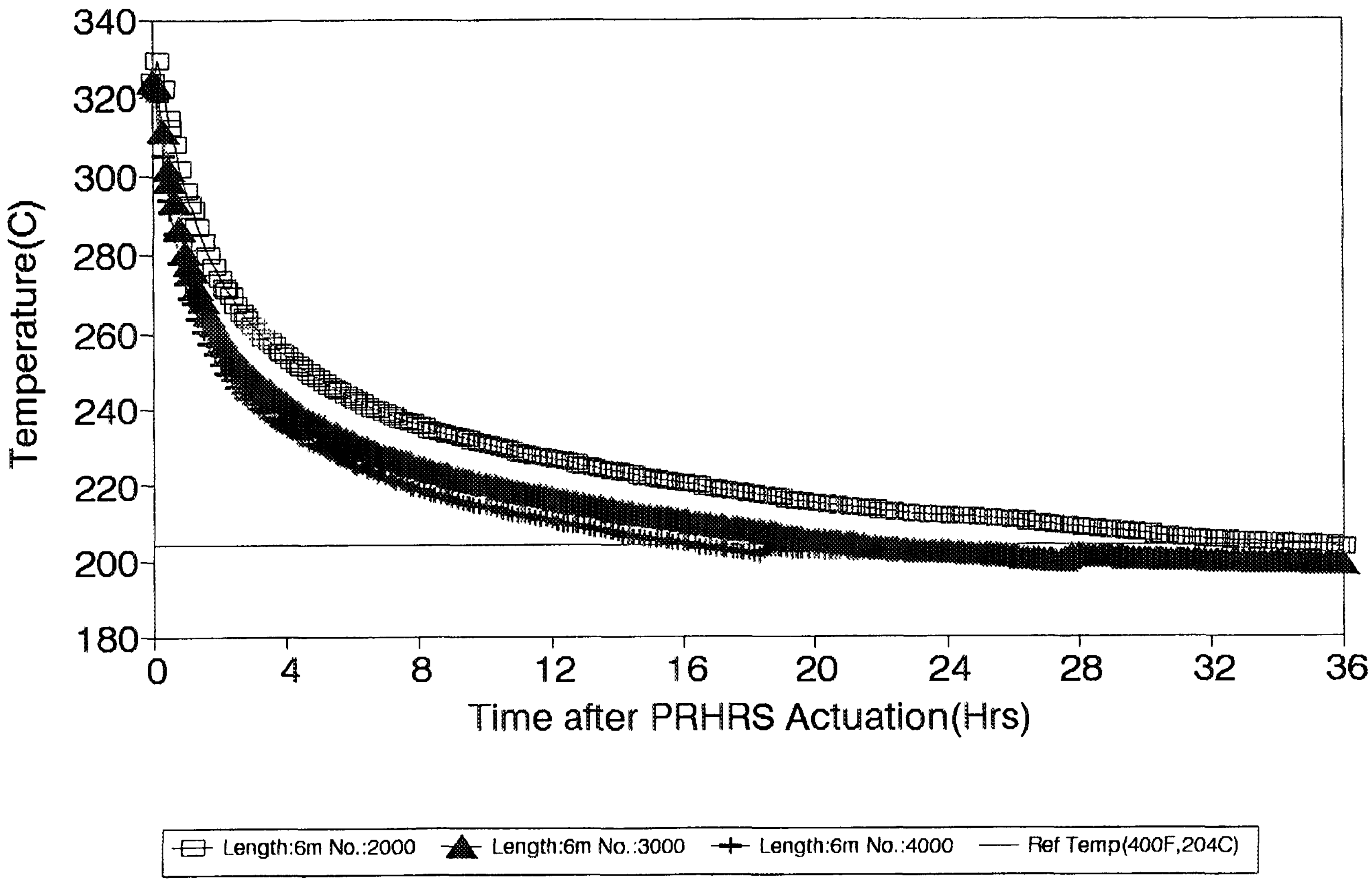

Fig-6 PRHRS Performance Curve
(PRHRS with Secondary Loop)

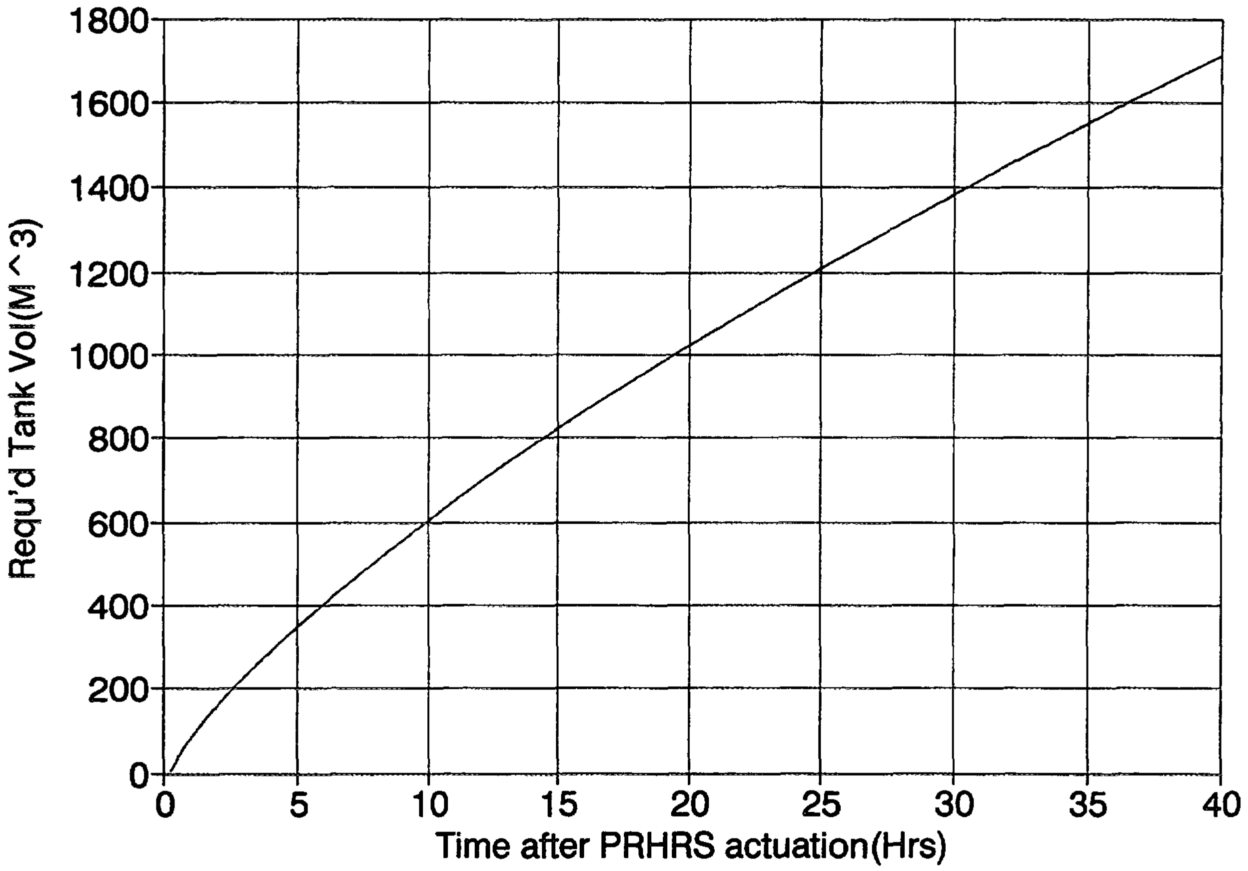

Fig-7 Required External Water Supply wrt PRHRS Operation

However, in the plant performance aspects, there would be two types of events which are affected by implementing the PRHRS. One is the main steam line break event and the other the inadvertent operation of PRHRS. The main steam line break(MSLB) with simultaneous operation of PRHRS may overcool the RCS and thus result in the return to power of the core. But this event can be excluded by proper design consideration such a way that during normal operation the PRHRS isolation valves are at closed state with proper interlock. When the system required, the PRHRS can be actuated by manual action in accordance with the administrative procedure. The above design and operational provisions make the probability of inadvertent PRHRS operation extremely low, and thus the MSLB with simultaneous operation of PRHRS can be excluded from a licensing basis event. In the case of an inadvertent operation of PRHRS during normal operation the nuclear power increases quickly, but the plant is expected to respond safely by proper actuation of the plant protection system.

The Core Damage Frequency(CDF) reduction effect by implementing PRHRS in a large-scale active plant based on System 80+ Probabilistic safety assessment(PSA) was evaluated with the assumptions that, in transients, PRHRS makes reactor safe for 24 hours and PRHRS is normally isolated from RCS by closed valves. From this evaluation, the CDF reduction effect turns out to be about 20% in CDF of the reference design[1,6]. The PSA results are summarized in Table-1. The CDF reduction effect by implementing PRHRS is, however, evaluated to be small because CDF of the reference design is very low even to ALWR design requirements. For a reference, USNRC mentioned that the goal of CDF is 1.0E-4 per reactor year[3,4] and EPRI ALWR URD requires 1.0E-5 per reactor year as CDF goal[5].

While PRHRS has a reduction effect for CDF, there is also adversary effect on CDF. The introduction of large number of heat exchanger tubes which would result in the reactor coolant pressure boundary expansion, yields additional LOCA paths. The primary circuit of PRHRS consists of suction and discharge pipe lines and heat exchanger with huge number of small tubes. This configuration results in increasing the probability of LOCA occurrence. The possible LOCA types will be medium and very small LOCA's such as PRHRS loop pipe break and the PRHRS heat tube break. This effects will also increase the CDF probability.

Table-1. Comparison of CDF reduction effect with/without PRHRS in Sys80+ design

Event	System 80+ CDF	CDF Reduction	CDF Expectation
Internal	1.7E-6	2.0 - 4.0E-7	1.3 - 1.5E-6
External	3.3E-7	1.0 - 2.0E-7	1.3 - 2.3E-7
Shutdown	8.3E-7	1.7 - 3.4E-7	4.9 - 6.6E-7
Total	2.9E-6	4.7 - 9.4E-7	2.0 - 2.4E-6

3.2 Cost impacts

The implementation of PRHRS into the well developed existing plants would have impacts on the containment configuration rearrangement to accomodate a PRHRS heat exchanger inside containment at the elevated level to maintain natural circulation of primary fluid. PRHRS also requires extra containment penetration paths to remove primary residual heat to outside containment.

PRHRS supported by CCWS may have impact on the CCWS heat removal capacity and requires long run of CCWS pipe line with large size. Redesign of CCWS and system piping extension will require additional capital costs.

For the PRHRS with passive support system, there needs also large size of external water storage tank and/or heat exchanger outside containment building. It is also requisite to test PRHR HX performance. The tests need to be made for performance of PRHR heat exchanger and operability. The whole activity for test is to be estimated nearly one year. Approximately 10M$ of capital costs including system development, containment configuration change and tests is expected.[6]

For maintenance and operation, not great effects are anticipated due to PRHRS because PRHRS is composed of passive components such as tanks and heat exchanger, etc. It is generally estimated about 2% of capital cost

3.3 Licensing impacts

The other impacts due to PRHRS implementation may related to the licensing. In licensing aspects, there may be some discussions on the adversary effects of PRHRS on safety. However, if the PRHRS designed as non-safety related system serves as a back-up system to support the active one, there would be no arguing problems because the system does not have any safety functions.

4.0 Conclusion

The overall assessment based on the more or less quantitative analysis came up with that a PRHRS concept with a gravity feed tank outside containment is a most preferable design as a back-up system to support the existing active system. This concept can be implemented by adding a PRHRS heat exchanger inside containment or by exploiting steam generator with gravity feed tank outside containment. However, to implement PRHRS into well developed active plant such as System80+, large amount of development and capital cost could be expected. Thus it is concluded that the incorporation of the PRHRS concept into large scale active plants, is not desirable at present. However if nuclear environment changes such as safety policy and public acceptance of SDS operation, PRHRS concept will be one of strong candidates for decay heat removal means at the case of beyond design basis events for active pressurized water reactor.

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PASSIVE SAFETY SYSTEMS FOR DECAY HEAT REMOVAL OF MRX

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Abstract

The MRX (marine Reactor X) is an advanced marine reactor, its design has been studied in Japan Atomic Energy Research Institute. It is characterized by four features, integral type PWR, in-vessel type control rod drive mechanisms, water-filled containment vessel and passive decay heat removal system.

A water-filled containment vessel is of great advantage since it ensures compactness of a reactor plant by realizing compact radiation shielding. The containment vessel also yields passive safety of MRX in the event of a LOCA by passively maintaining core flooding without any emergency water injection. Natural circulation of water in the vessels (reactor and containment vessels) is one of key factors of passive decay heat removal system of MRX, since decay heat is transferred from fuel rods to atmosphere by natural circulation of the primary water, water in the containment vessel and thermal medium in heat pipe system for the containment vessel water cooling in case of long term cooling after a LOCA as well as after reactor scram.

Thus, the idea of water-filled containment vessel is considered to be very profitable and significant in safety and economical point of view. This idea is, however, not so familiar for a conventional nuclear system, so experimental and analytical efforts are carried out for evaluation of hydro-thermal behaviors in the reactor pressure vessel and in the containment vessel in the event of a LOCA. The results show the effectiveness of the new design concept. Additional work will also be conducted to investigate the practical maintenance of instruments in the containment vessel.

Introduction

Design of an advanced marine reactor, MRX(1)-(3), which has been studied in Japan Atomic Energy Research Institute (JAERI), is characterized by four features, integral type PWR, in-vessel type control rod drive mechanisms, a water-filled containment vessel and passive decay heat removal system. The standard type of MRX is designed with an output of 100 MWt, but the concept can be applied to reactors with wide range of output from 50 to 300 MWt. A conceptual view and major parameters ofthe 100 MWt MRX are show in Fig. 1 and Table 1, respectively.

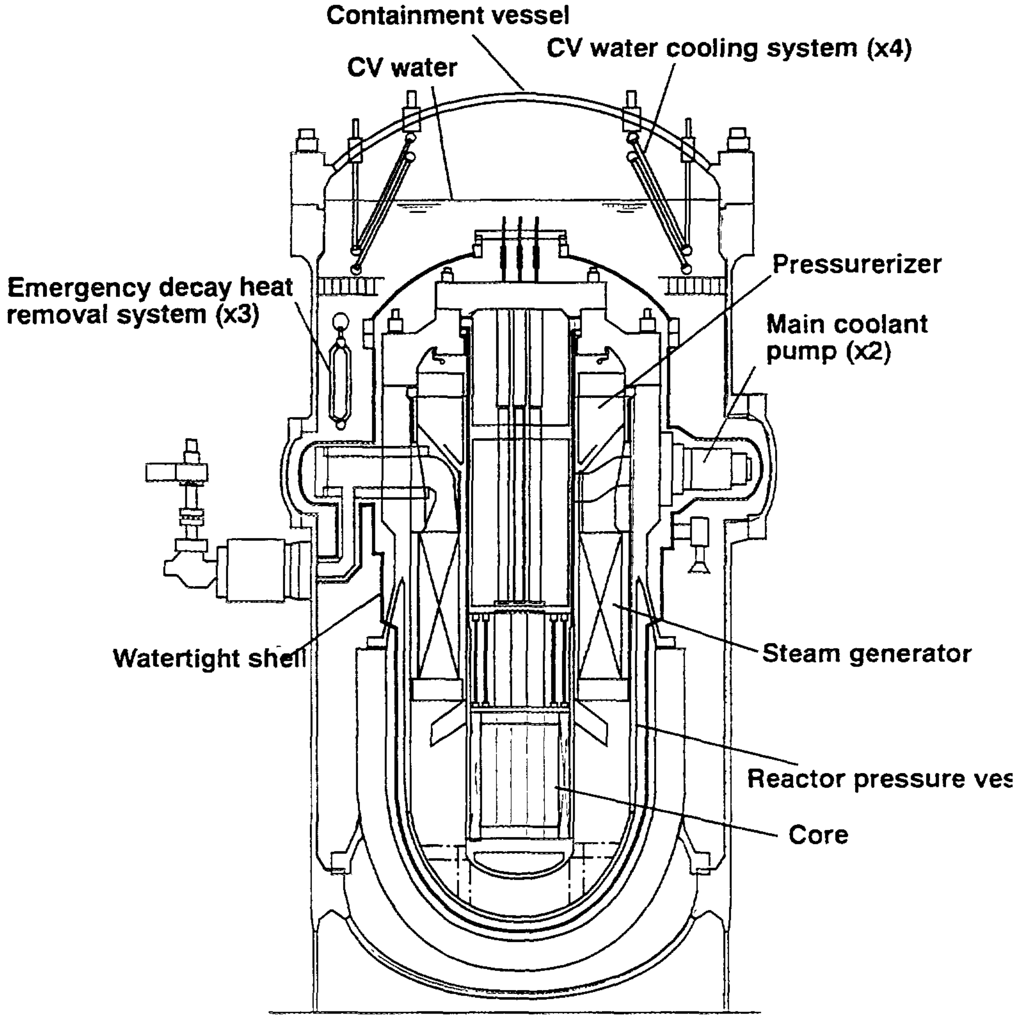


Fig.1 A View of an Advanced Marine Reactor MRX.

A marine reactor should have smaller weight than that of a land based reactor from the economic viewpoint of commercial ships. Most part of the weight of the preceding marine reactors is contributed by the radiation shielding. For instance, the total shield weight of the Nuclear Ship MUTSU (N.S. MUTSU) is more than 70% of that of the reactor plant. In the design of MRX, no bulk shield outside the containment vessel is required due to the adoption of the integral PWR and the water-filled containment vessel. As a result, MRX is considerably lighter in weight, more compact in size and hence more economical than the reactors equipped in previously constructed nuclear merchant ships(4). For instance, the plant weight and volume of the containment vessel of MRX are 50% and 70% of those of the N.S. MUTSU's reactor, in spite of the fact that the power of MRX is 2.8 times greater than that of N.S. MUTSU's reactor.

A marine reactor should be able to be operated and also managed even in case of accidents by limited number of crew. Therefore, passive safety system is very profitable for a marine reactor. MRX has two kinds of passive safety functions. The first kind is passively maintaining core flooding

TABLE 1. MAJOR PARAMETERS OF THE MRX

Reactor Power	100 MWt
Reactor Type	Integral type PWR
Operation Pressure	12 MPa
Core Inlet/Outlet Temperature	282.5 / 297.5 ℃
Number of Main Coolant Pump	2
Core Equivalent Diameter	1.45 m
Core Effective Height	1.4 m
Control Rod Drive System	In-Vessel Type
Number of CRDMS	13
Type of Steam Generator	One-through Helical Coil Type
Inner/Outer diameter of SG tube	19 / 14.8 mm
Steam Flow Rate	160 t/h
RPV Inner Diameter/ Height	3.7 m / 9.3 m
Type of the CV	a water-filled type
Inner Diameter/Height of the CV	6.8 m / 13.0 m
Design Pressure of the CV	4 MPa

function in the event of a LOCA, and the other is passive decay heat removal function after reactor shut-down. The water-filled containment vessel satisfies both functions, therefore it is clearly profitable and significant for MRX. Since the idea of the water-filled containment vessel is not so familiar for conventional nuclear systems, the feasibility of the idea is being examined experimentally and analytically.

This paper describes the present status of experimental and analytical works for evaluation of hydro-thermal behaviors in the reactor vessel and in the containment vessel in the event of a LOCA.

Passive Emergency Core Cooling System of MRX

Emergency core cooling system (ECCS) of MRX consists of emergency decay heat removal system (EDRS), containment water cooling system (CWCS) and the water-filled containment vessel (CV). The EDRS is a closed system which conveys decay heat from the core to the containment water by the natural circulation of primary water. It includes three trains, each of which has a full capacity to remove the core decay heat. Each train consists of a hydrogen reservoir tank, a cooler, two valves and piping. If reactor trip signal is generated, the battery operated valves of each train are opened. The CWCS is a heat pipe system which transfers the heat of the containment water to the atmosphere. It includes four trains, each of which has 1/3 capacity. A basic idea of the ECCS of MRX is shown in Fig. 2.

A large-scale LOCA cannot occur in MRX, since only small size pipes (≤ 50 mm) exist in the primary system. In the case of a small-scale LOCA, primary water blows down into the CV or a watertight shell (WS). In the case of blow down into the CV, primary water vaporizes immediately and then condenses rapidly. In the case of blow-down into the WS, primary water evaporates within WS to cause WS inner pressure rise. After relief valves are opened, vapor within WS flows into the CV. In both cases, RPV pressure drops, CV pressure rises by compression of N_2, cover gas in the CV, and temperature rises in the CV. Accordingly, CV pressure is equalized with RPV pressure to stop primary water discharge in around a half hour. By the appropriate setting of the initial cover gas volume in the CV, it is possible to maintain passively core flooding in MRX without any water injection. The decay heat is transferred passively by the natural convection of primary water in EDRS, CV water and CWCS working fluid after the blow down termination. Therefore, no active operation except an active signal to open valves in EDRS, is required to maintain reactor in cold shut down state even in the event of a LOCA.

In the event of any other accident in MRX, including loss of electric power, the reactor is automatically tripped and the EDRS valves are opened with active signals and passive power. Thus the reactor attains cold shut down state passively without any other operations.

LOCA Analysis

The peak pressure in the CV is desired to be less than 2.0 MPa to assure enough safety margin, although the maximum design pressure of the

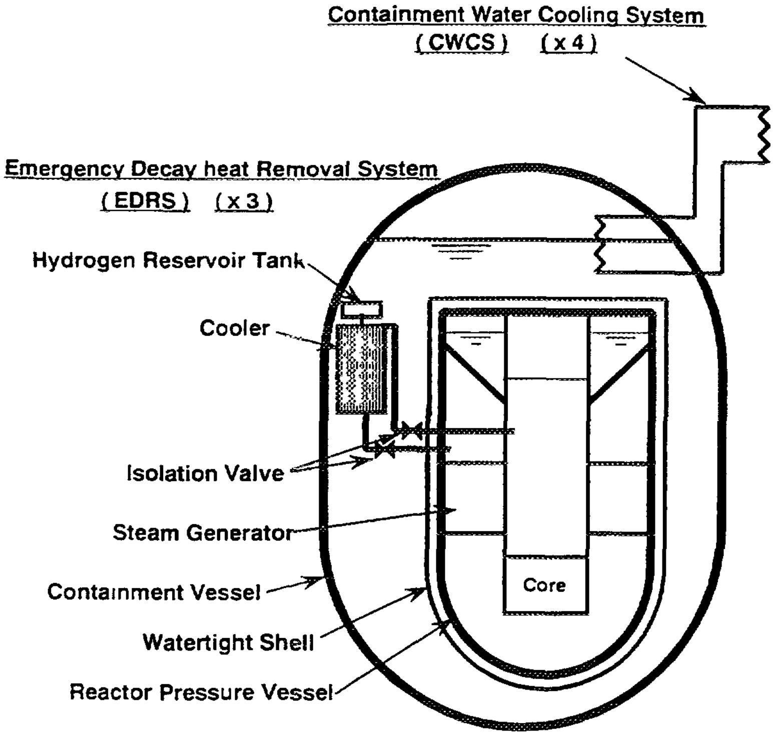


Fig.2 A Basic Idea of the ECCS of MRX

CV in the MRX is 4.0 MPa. The minimum water level in the RPV must be 50 cm above the top of the core to maintain adequate core flooding even when the ship inclines by 30 degrees (IMO Code of Safety for Nuclear Merchant Ships). If initial cover gas volume in the CV is too large, the core may not be covered with water before reaching pressure equilibrium in the event of a LOCA. On the other hand, if too small, CV pressure might exceed design pressure. Therefore, LOCA analysis has been conducted with RELAP5/Mod2 assuming 50 mm dia. pipe break, changing the initial CV cover gas volume.

The LOCA analysis model is shown in Fig. 3. The initial temperature and pressure in the CV are 60 ℃ and 0.1 MPa, respectively. The volume of the gap region between the RPV and the WS is 3 m³. The atmospheric

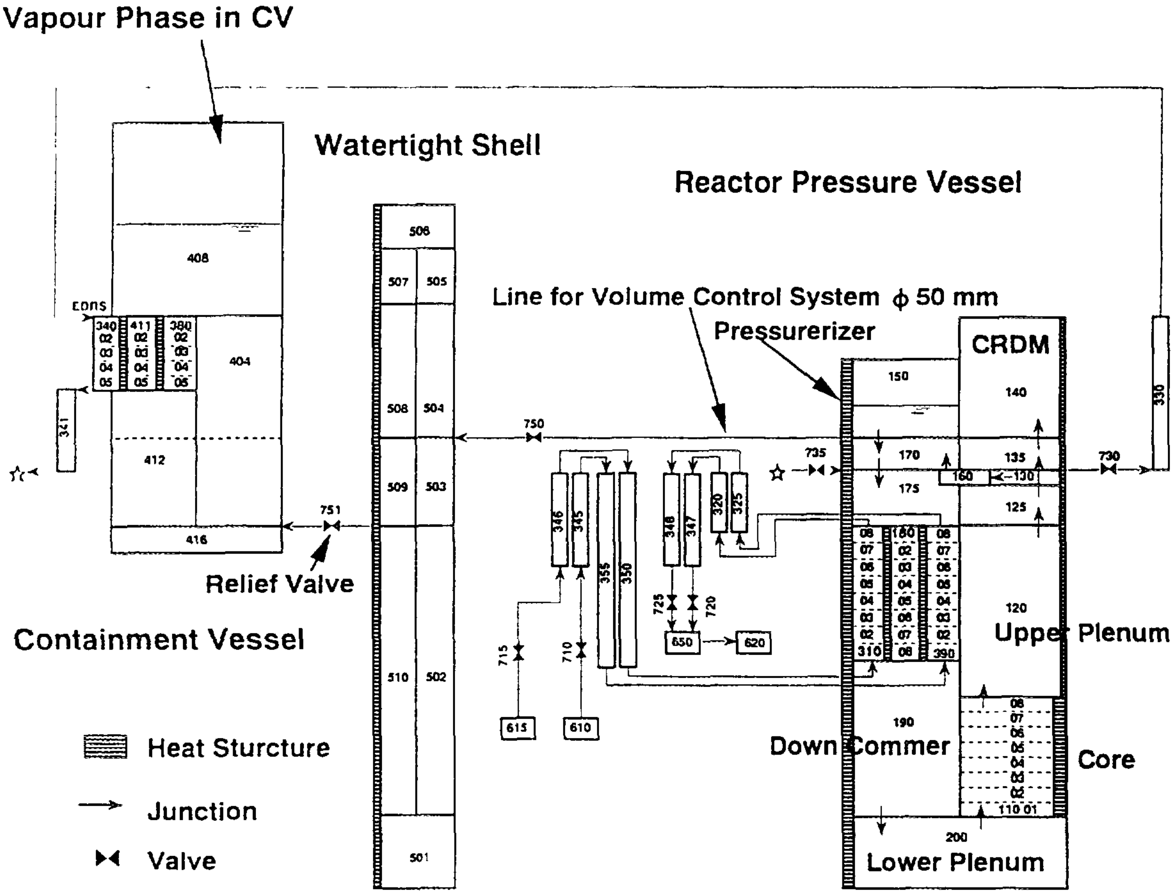


Fig. 3. The LOCA analysis model.

temperature is 35 ℃. Operating pressure and effective diameter of relief valves on the WS wall are 0.2 MPa and 300 mm, respectively. A pipe break is assumed to occur in the CV (case A) to evaluate the maximum CV pressure, and in the WS (case B) to evaluate the minimum water level in the RPV. In case A analysis the WS are assumed to be intact to evaluate CV pressure conservatively.

The transient CV pressure obtained in case A analysis as a parameter of the initial CV cover gas volume is shown in Fig.4(a), while the transient RPV water level in case B analysis is shown in Fig.4(b). In the case of the

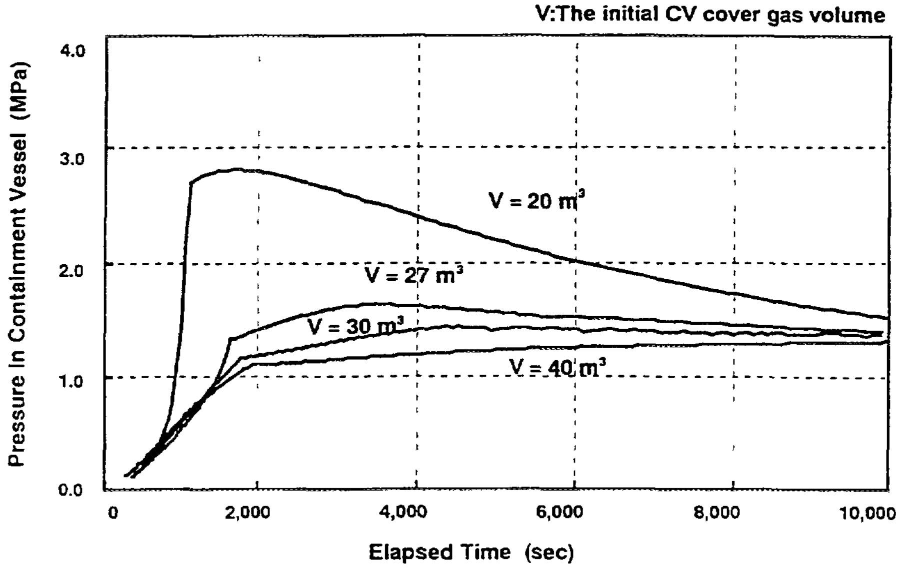


Fig.4(a) Transient CV Pressure Obtained in Case A Analysis.

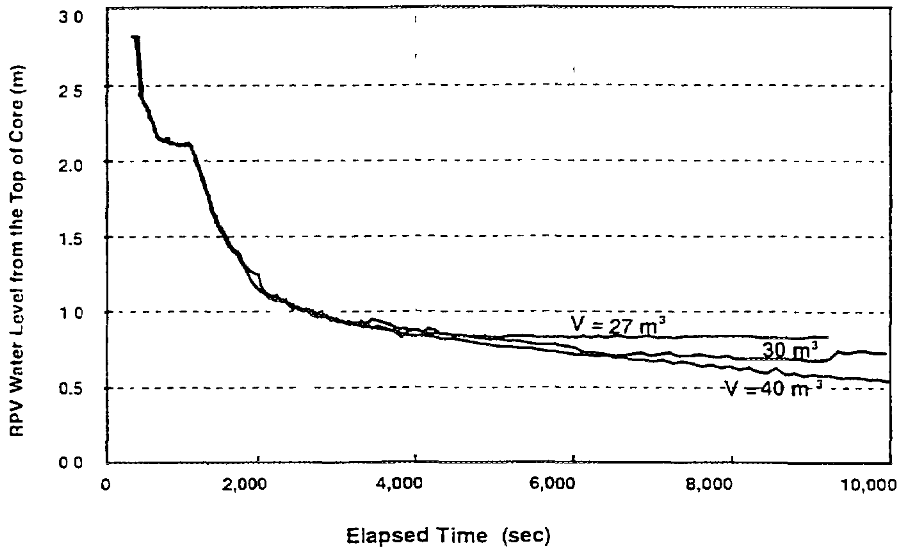


Fig. 4(b) Transient RPV Water Level in Case B Analysis.

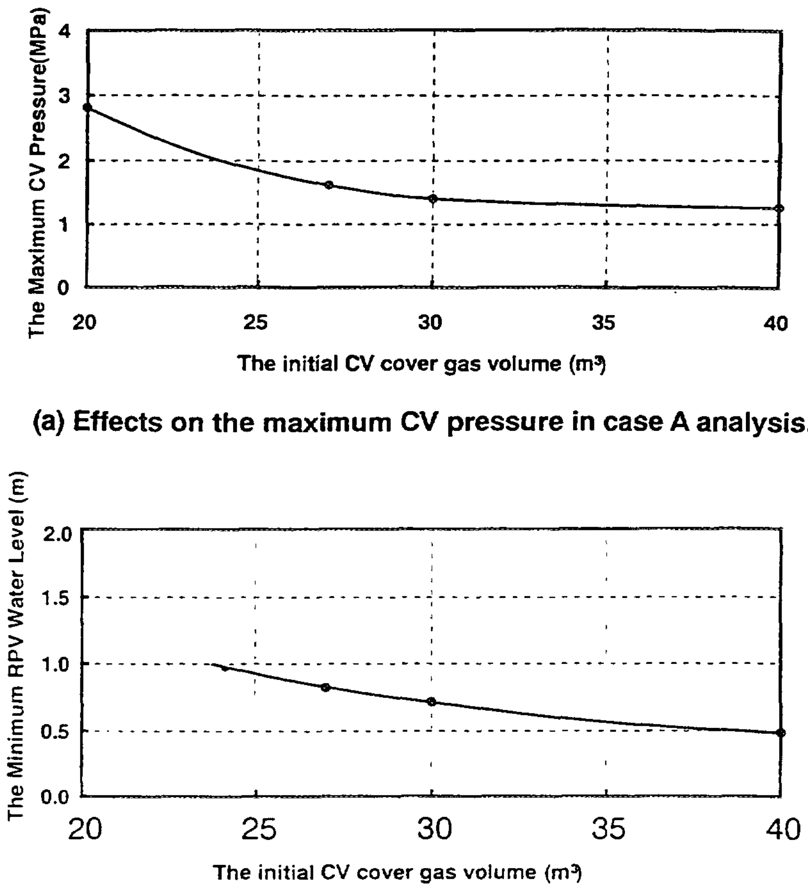
smaller CV cover gas volume, CV pressure goes past the equilibrium value due to the higher compression rate of N_2, as shown in Fig. 4(a). Figure 4(b) shows that the RPV water level, when the initial CV cover gas volume exceeds 30 m³, continues to decrease even after considerably long time, because additional discharge is caused by cooling of CV water after CV pressure reaches the same level as RPV pressure.

The maximum CV pressure obtained in case A analysis as a function of the initial CV cover gas volume, and the minimum RPV water level in case B analysis are shown in Fig. 5(a) and (b), respectively. According to these figures it can be concluded that the initial CV cover gas volume of 27 m³ yields allowable CV pressure and RPV water level with enough margin of safety.

Experiment on Condensation in highly Subcooled Water

The present analytical results show that the maximum CV pressure is low enough to assure the integrity of the CV while still guaranteeing core flooding during a LOCA. It can be also expected that no serious dynamic load on the CV structures is induced by condensation in subcooled water in the MRX, since CV structures are designed to endure high design pressures. However, it might be possible for vapor injected out of the RPV not to be sufficiently condensed in the CV water and to flow up into the cover gas region, thus causing CV pressure to increase severely. Therefore, we are conducting a small size experiment of the pipe rupture in highly subcooled water to study the condensation rate and the CV pressure behavior.

Experimental apparatus consists of a primary vessel, a simulated CV, a quick operating valve, a connecting pipe and a replaceable orifice which allows various rupture sizes to be employed, as shown in Fig.6. Volume of the primary vessel is scaled to 1/300 of a 100 MWt MRX RPV. The maximum operating pressure is 5 MPa, and the center line of the connecting pipe locates at 1,200 mm from the bottom of the primary vessel. The primary vessel is filled with saturated water at the initial pressure, while the simulated CV contains water and nitrogen at room temperature. By opening a



(a) Effects on the maximum CV pressure in case A analysis.

(b) Effects on the minimum RPV water level in case B analysis.

Fig. 5 Effects of the Initial CV Cover Gas Volume.

quick operating valve hot water and/or steam are discharged from the primary vessel to the simulated CV through the orifice.

Main experimental parameters are direction of discharge, initial primary vessel pressure and water level, initial water level in the simulated CV and orifice diameter, as shown in Table 2. The major measured parameters are the temperatures at five points, the pressure at one point in the primary vessel, the temperatures at nineteen points, the pressure at four points in the simulated CV and the water level of each vessel.

The measured transient behaviors of the pressure and the cover gas temperature in the simulated CV are shown in Figs. 7(a) and (b). Figure 7(a)

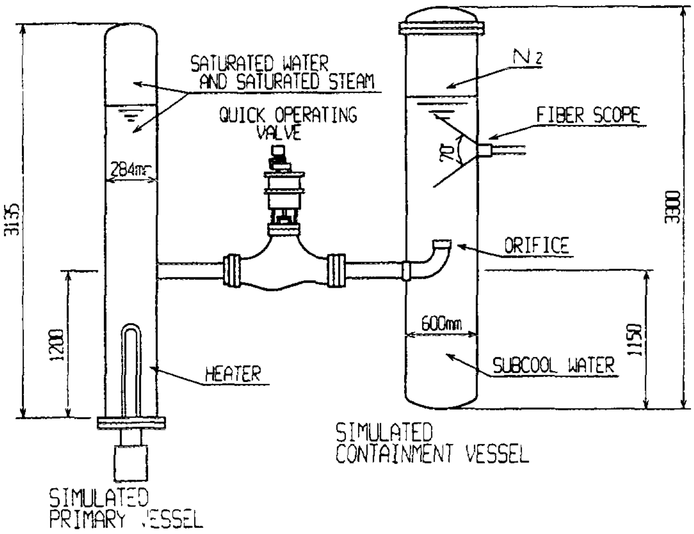


Fig. 6. Experimental apparatus.

TABLE 2. EXPERIMENTAL PARAMETERS

Direction of Discharge	vertical or horizontal
The Initial Pressure in the Primary Vessel	2.0, 3.0,4.0 MPa
The Initial Water Level in the Primary Vessel	900 or 2,850 mm
Initial Water Level in the Simulated CV	0 ~ 3,000 mm
Diameter of the Orifices	3, 12.5, 20, 24, 35 mm

shows the experimental results in the case of initial water level in the simulated CV of 30 mm height from the position of the orifice and Fig. 7(b) of 600 mm height. The other conditions in both experiments are the same; vertical discharge, initial primary vessel pressure of 2.0 MPa, initial primary vessel water level 200 mm from the bottom, that is vapor blow down, and orifice diameter of 35 mm. The lower water level produces larger overshooting of the pressure, that is, a considerable amount of uncondensed vapor exist transiently in the cover gas region.

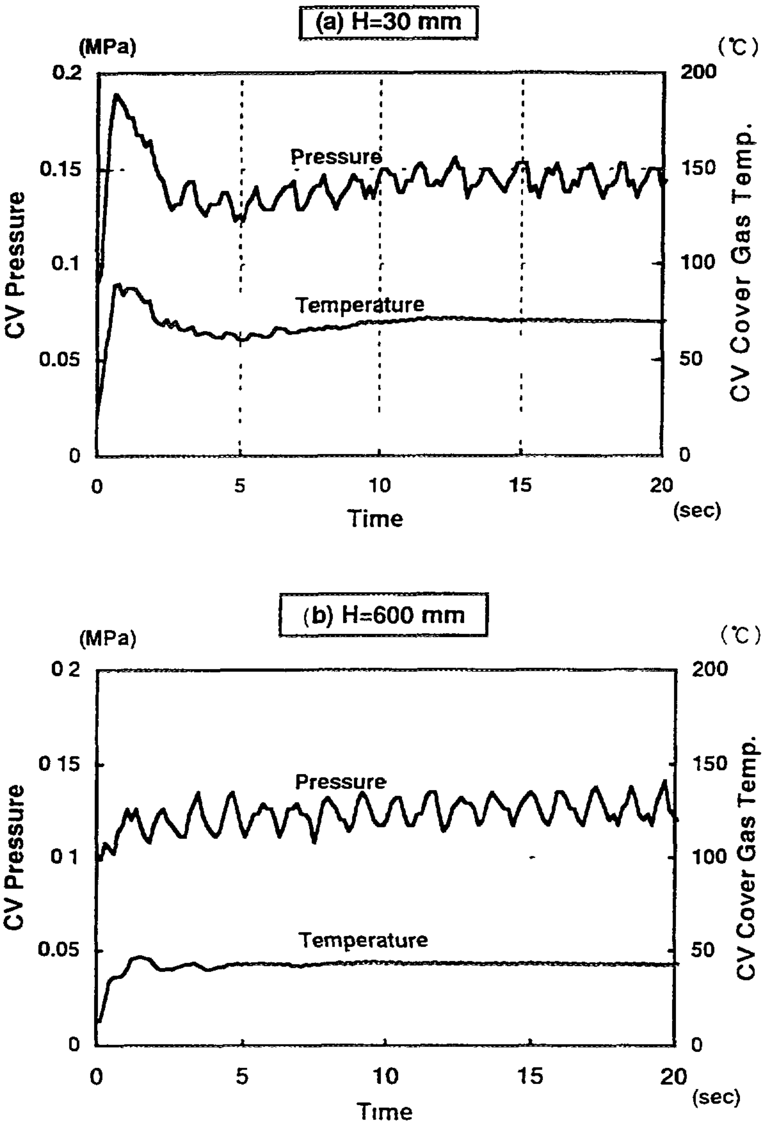


Fig. 7. Transient behaviours of the pressure and cover gas temperature in the CV.

A simple calculation has also been conducted to obtain the amount of uncondensed vapor using measured values of the primary vessel pressure and water level, the liquid and gas temperature in the simulated CV, and neglecting the vapor in subcooled water in the simulated CV, that is, assuming that all vapor in the simulated CV is saturated vapor at the measured gas phase temperature. The calculation also yields the simulated CV pressure. Figures 8(a) and (b) show the comparison of the calculated and the measured pressures in the simulated CV; very good agreement is obtained. These figures also show the amount of calculated vapor in the simulated CV.

The uncondensation rate(η) as a function of the CV water level(H) from the orifice position is shown in Fig. 9. The uncondensation rate η is defined as the maximum vapor amount in the cover gas region divided by the total amount of injected water until the amount of vapor in the cover gas

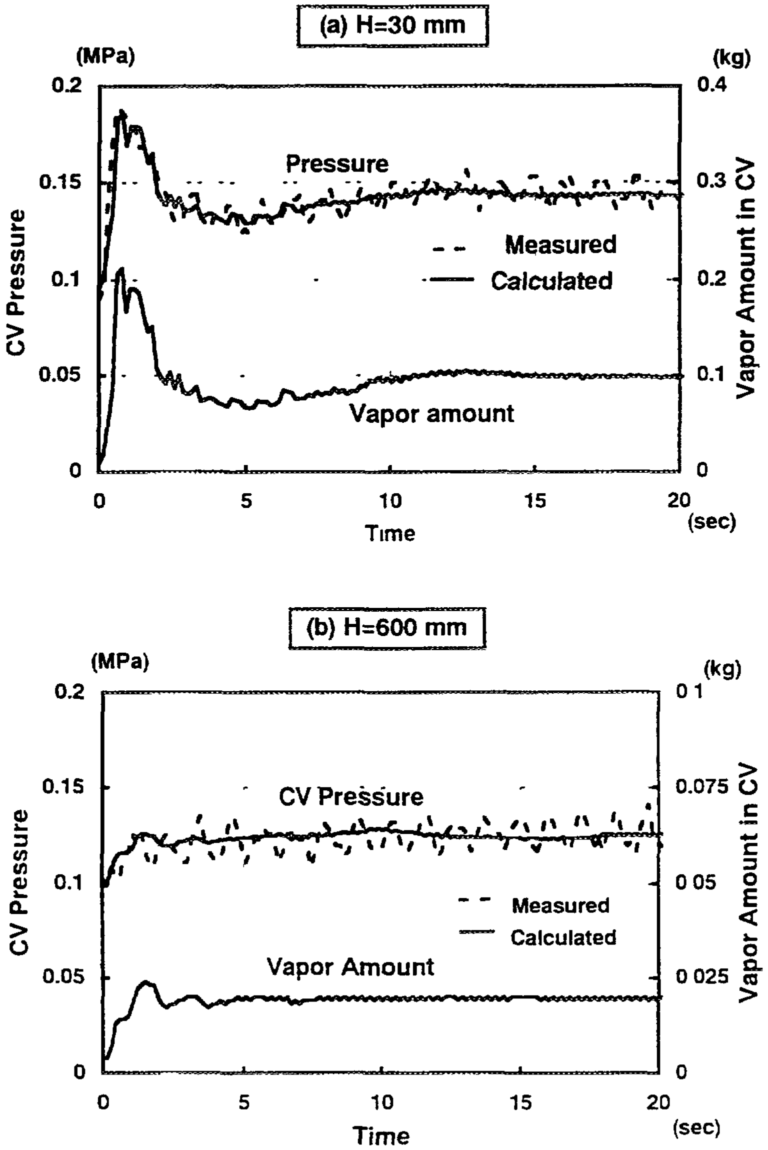


Fig. 8. The calculated and the measured CV pressure with the calculated vapor amount in th CV.

region reaches the maximum value. For both experiments (vapor blow-down and liquid blowdown) it was found that a single correlation, as given by Eq.(1), can be used to represent the uncondensation rate. This correlation was obtained using curve fitting techniques.

$$\eta = 8.4 \exp(-5.74H) \qquad (1)$$

Although more experiments are scheduled, test results to date do not show the possibility of severe CV pressure rise in the MRX. Additional correlations describing the uncondensation rate as a function of the orifice diameter, RPV pressure etc., will be obtained from the scheduled experiments. These will then be introduced into the TRAC code.

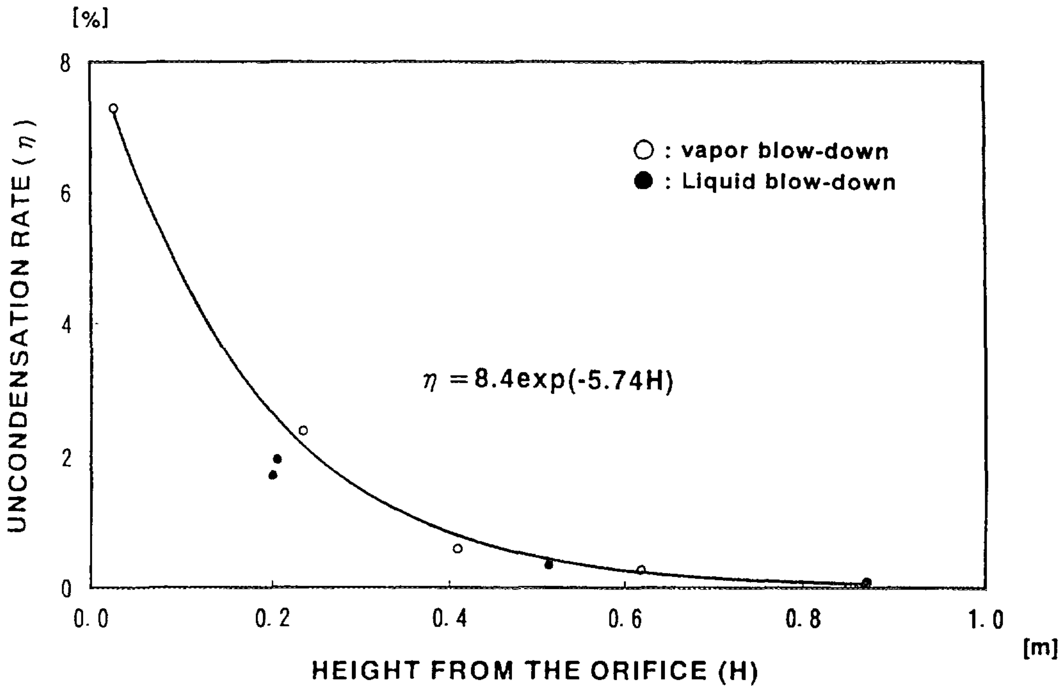


Fig. 9. Correlation of the uncondensation rate.

Conclusions

An advanced marine reactor, MRX, which has been studied in JAERI is considerably lighter in weight, more compact in size and hence more economical than the reactors equipped in previously constructed nuclear merchant ships. The passive safety system adopted in the MRX design is very profitable for a marine reactor since it must be operated by limited number of crew. The new idea of a water-filled containment vessel gives rise to better economy and passive safety to the MRX. This new concept requires analytical and experimental evaluation to assure the feasibility of the water-filled CV. The following tentative conclusions are drawn from the present work;

(1) According to the LOCA analysis, the initial cover gas volume in a water-filled containment vessel of 27 m^3 yields allowable CV pressure while maintaining core flooding without any water injection with enough safety margin.

(2) According to the condensation experiment, possibility of severe pressure rise in a water-filled containment in the event of a LOCA was not observed.

(3) A relationship describing the uncondensation rate was obtained from the experiments. This is helpful to analyze in detail hydraulic phenomena in the water-filled containment vessel in the event of a LOCA or MSLB(main steam line break).

Further work will be conducted on safety analysis and condensation. In addition to these, we will study practical maintenance method of instruments installed in a water-filled containment vessel.

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GRAVITY DRIVEN EMERGENCY CORE COOLING EXPERIMENTS WITH THE PACTEL FACILITY

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Abstract

The gravity driven emergency core cooling (ECC) systems are utilized as important components of passive safety coolant systems of advanced reactors. Most of the published investigations have been primarily concerned with the presentation of new concepts, a few of their computational analysis and even fewer studies have been addressed to the experimental investigation of these systems.

PACTEL (Parallel Channel Test Loop) is an experimental out-of-pile facility designed to simulate the major components and system behavior of a commercial Pressurized Water Reactor (PWR) during different postulated LOCAs and transients /1/. The reference reactor to the PACTEL facility is Loviisa type VVER-440. The recently made modifications enable experiments to be conducted also on the passive core cooling. In these experiments the passive core cooling system consisted of one core makeup tank (CMT) and pressure balancing lines from the pressurizer and from a cold leg connected to the top of the CMT in order to maintain the tank in pressure equilibrium with the primary system during ECC injection. The line from the pressurizer to the core makeup tank was normally open. The ECC flow was provided from the CMT located at a higher elevation than the main part of the primary system. A total number of nine experiments have been performed by now.

A preliminary series of experiments with gravity driven core cooling was conducted with PACTEL facility in November 1992 /2/. The simulated transient was a small break loss-of-coolant accident (SBLOCA) with a break in a hot leg. In these tests a rapid condensation of vapor interrupted the emergency core cooling flow several times. This behavior was found very difficult to model in the RELAP5 analysis of the experiments of the first phase. In order to investigate this behavior more precisely, a second series of experiments with an improved instrumentation of the facility was performed in November 1993 with a small break in a cold leg. The tests indicated the that steam condensation in the CMT can prevent continuous ECC and even lead to partial core uncovery. However, it should be underlined that these tests presented here are not directly applicable to the safety analyses of any suggested design, because of the major differences in the geometry between these concepts and PACTEL. Our objective has been only to simulate the gravity driven ECC and thus to enhance the understanding of the physical phenomena important in passive safety systems working with low differential pressures.

1. INTRODUCTION

Along with the normal evolution in LWR reactor designs several new interesting concepts have been presented. These ALWR designs aim at plant simplifications and safety and operability improvements. The principal tool being used to achieve a safer and simpler reactor is the use of passive system designs. Unfortunately, it is not easy to confirm that passive safety systems operate as intended under all the relevant conditions. More work is needed to evaluate the extent of improvements in safety which can be realized.

Experiments conducted with thermal hydraulic test facilities are of fundamental importance in nuclear power plant safety research. At the Lappeenranta University of Technology (LUT) work has been carried out in this field in co-operation with the Technical Research Centre of Finland (VTT), for over fifteen years. This paper provides the presentation of gravity driven emergency core cooling experiments with PACTEL, their analysis and discussion of the phenomena related to the experiments. The recently made modifications enable experiments to be conducted also on the passive core cooling.

2. PACTEL FACILITY

The PACTEL facility simulates the major PWR components and systems during small- and medium-size break LOCAs. The facility consists of a primary system, the secondary side of the steam generators, and emergency core cooling systems (ECCS). The reactor vessel is simulated by a U-tube construction including downcomer, lower plenum, core and upper plenum.

The facility is a volumetrically scaled model of the 6-loop VVER-440 PWR (The Finnish Loviisa plant being the reference) with three separate loops and 144 full-length, electrically-heated fuel rod simulators arranged in three parallel channels. The fuel rod simulators are heated indirectly.

The reference reactor has certain unique features differing from other PWR designs. The VVER-440 has six primary loops with horizontal steam generators. Due to the construction of the steam generators the driving head for the natural circulation in small break LOCAs is relatively small. The primary loops have loop seals in both the hot and cold legs. The loop seal is a U-shaped bend in the leg piping connecting the steam generator to the pressure vessel. It is interesting to note in the current context that the basic design of the VVER-440 reactor exhibits certain inherent safety features that are again found by the new designs. The reactor core has low power density and the primary circuit water inventory is large relative to the power. These characteristics ensure smooth behaviour in transient conditions.

Volumetric scaling (1:305) preserving the elevations has been applied in the PACTEL design. Maintaining system component heights and elevations is important for realistic simulation of small break and natural circulation transients. The main characteristics of the facility are presented in Table I.

3. PACTEL MODIFICATIONS AND TEST MATRICES FOR PASSIVE CORE COOLING EXPERIMENTS

The passive core cooling system used in the experiments consists of one core makeup tank and pressure balancing lines from the pressurizer and from a cold leg connected to the top of the core makeup tank in order to maintain the tank in pressure equilibrium with the primary system during injection. The line from the pressurizer to the core makeup tank is normally open. The core makeup tank is located above the reactor coolant loops and steam generators, so the motive

TABLE I. MAIN CHARACTERISTICS OF PACTEL FACILITY

Reference power plant	VVER-440
Volumetric scaling factor	1:305
Scaling factor for elevations	1:1
Number of primary loops	3
Maximum heating power	1 MW
Number of fuel rod simulators	144
Outer diameter of fuel rods	9.1 mm
Heated length of fuel rods	2420 mm
Axial power distribution	chopped cosine
Axial peaking factor	1.4
Maximum temperature of fuel rods	800 °C
Maximum primary pressure	8.0 MPa
Maximum operating temperature	295 °C
Maximum secondary pressure	4.65 MPa

force for injection is the gravity head, Figure 1. The makeup tank used limits the primary pressure to 5 MPa in the experiments. Since the PACTEL facility is not a model of any of the proposed passive ALWR designs, the modifications in the facility are intended only to simulate the gravity driven flow to the primary system. Neither automatic depressurization system (ADS) nor special valves are simulated. The primary system is depressurized from the pressurizer relief valve.

3.1 First series of experiments

The first stage of each experiment involved heating the facility to the proper temperature. Before the tests the core power was set to 80 kW corresponding to 1.8% of the 1375 MW thermal power of the Loviisa reactor. The fluid temperature and pressure reached a quasi steady state near 220 °C and 40 bars and at this point the pressurizer heater power was reduced to 2 - 4 kW. These conditions were maintained for about half an hour permitting the fluid to attain a more uniform temperature and allowing the heat losses through flanges and support structures to approach an equilibrium. The SG feedwater injection was adjusted manually to keep the water level in the SGs constant. Because of the large water inventory on the secondary side no fast automatic control was needed. Before each experiment, the CMT was filled to the top with water at a temperature and pressure of about 40 °C and 38 bar, respectively.

The experiments were started by opening the break simulation valve in hot leg number 1 at time t = 0s. Three different break sizes (Ø 2, 4 and 6mm) were used. Simultaneously with the break valve opening, the ECC line valve and the cold leg PBL valve were opened. The power of the pressurizer heaters was turned off. The first two tests, GDEØ1 and GDEØ2, were terminated when a rapid condensation of vapor in the CMT vapor space depressurized the CMT. Check valves prevented the collapsed vapor space in the CMT to be filled with liquid drawn from the ECC line. In order to investigate the flow restrictions in the ECC line the armature of the line was varied during the three first tests. Neither the primary system nor the secondary system were depressurized by the operator in the GDEØ1 and GDEØ2 tests.

In tests GDEØ3, GDEØ4, and GDEØ5 the secondary side valve was also opened. The primary system was depressurized in stages through the pressurizer relief valve before the anticipated

CMT flow interruption in the GDEØ3 and GDEØ4 tests. For the large 6 mm (in dia.) break in the GDEØ5 test no extra depressurization was needed. These three tests were terminated when a thermal hydraulic status quo and a low pressure level was reached.The first series consisted of five experiments, Table II.

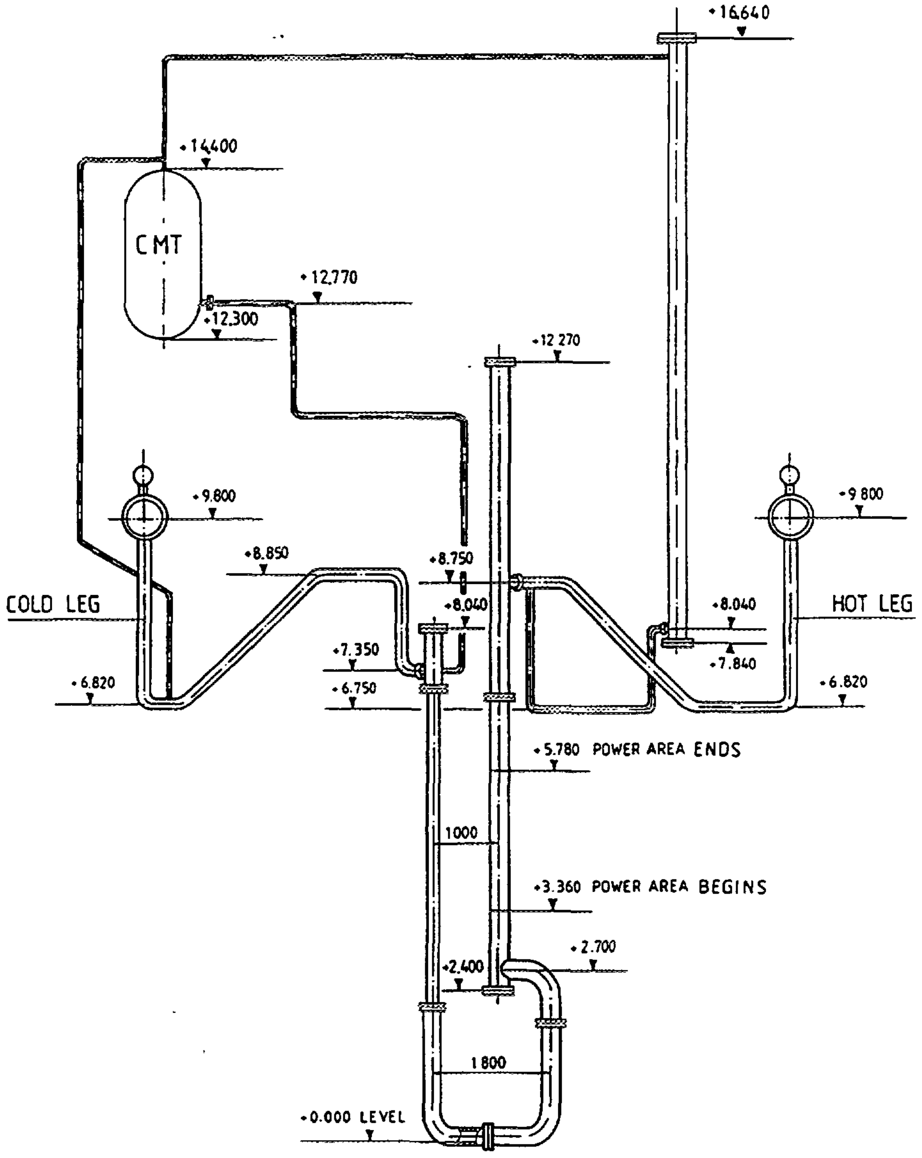

Figure 1. Component elevations in the passive ECC system of the PACTEL.

3.2 Second series of experiments

The gravity-driven emergency core cooling system (ECCs) behaviour was investigated more in the second phase of the tests with particular emphasis on break location, pressure reductions, reproducibility of the condensation manouvered experiments and system operation for the case of a small break LOCA. The major parameters and phenomena of concern during experiments are the break mass flow rate and the associated total primary coolant mass inventory, coolant distribution, different types and alternate paths of natural circulation in the loops, condensation and related heat transfer characteristics. For the second set of experiments the instrumentation

TABLE II. TEST MATRIX/PHASE 1

DESCRIPTION	FORESEEN PHENOMENA/AIMS	NOTES
GDE01/ low power (< 80 kW) 2.0% (4 mm) hot leg break with gravity driven ECC.	*Investigation of the blowdown process at low pressure.Gravity driven ECC effectiveness. Flow regimes. Phase separation and stratification. Condensation. Break flow. Coolant distribution.*	*For all of PCC-tests initial primary pressure = 40 bars. Steady-state initial conditions.*
GDE02/ 2.0% (4 mm) hot leg break with gravity driven ECC.	*As above.*	*Initial conditions as above. Minimized flow restricitions in the ECC line.*
GDE03/ 2.0% (4 mm) hot leg break. Gravity driven ECC with sudden depressurization of primary system.	*Effect of primary side depressurization.*	*As above. Primary and secondary side depressurization. Optimized flow restrictions in the ECC line.*
GDE04/ 2.0 % (2 mm) hot leg break. Gravity driven ECC with sudden depressurization of primary system.	*Effect of break area.*	*As above.*
GDE05/ 4.4% (6 mm) hot leg break. Gravity driven ECC.	*As above.*	*As above. No depressurizations.*

of the facility was improved. In order to investigate the temperature stratification in the CMT ten thermocouples were installed to the upper part of the CMT, Fig 2. The water level in the CMT was measured with a pressure difference transducer. One loop of the three loop facility was isolated. When compared to the first series of experiments, the main differences are that the second series was carried out with two active loops, insulated PBLs and an improved instrumentation in the CMT.

The experiments were started by opening the break simulation valve in cold leg number 1 at time t = 0s. Two different break sizes (Ø 4 and 2mm) were used. Simultaneously with the break valve opening, the ECC line valve and the cold leg PBL valve were opened. The power of the pressurizer heaters was turned off. The first two tests, GDE11 and GDE12, were terminated by operator at t= 3000s. Neither the primary system nor the secondary system were depressurized by the operator in the GDE11 and GDE12 tests.

In test GDE13 the secondary side valve was also opened and the primary system was depressurized in stages through the pressurizer relief valve before the anticipated CMT flow interruption. This test was terminated at t=2000s by the operator.

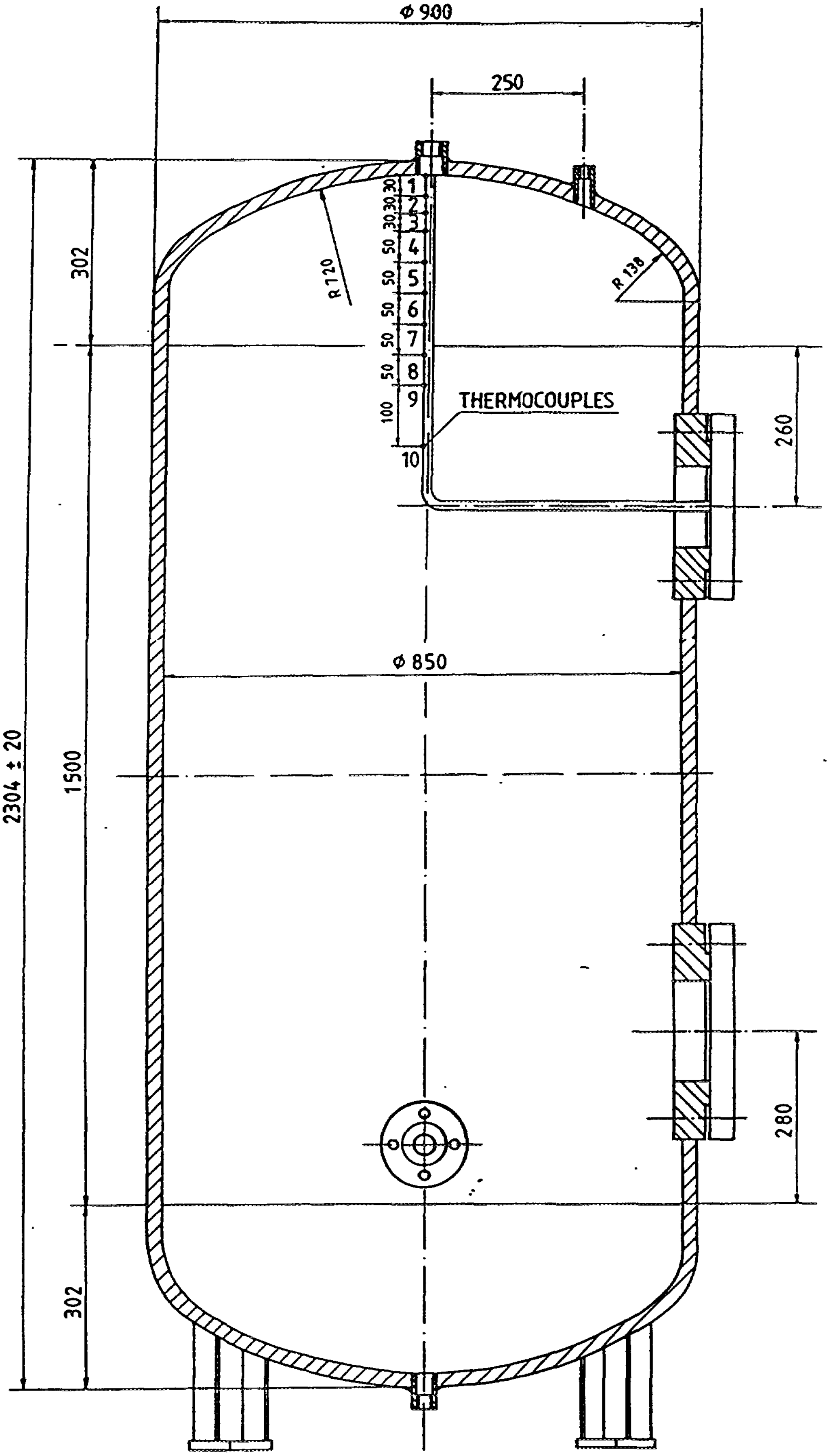

Fig. 2. Temperature measurement in the CMT tank.

For the small 2 mm (in dia.) break in the GDE14 test no depressurization was used. A high water level in the pressurizer was used in the initiation of the test in order to achieve circulation through the CMT in the early stage of the transient. The test was interrupted immediately after the condensation initiation at t=1170s. The test matrix is presented in Table III.

TABLE III. TEST MATRIX FOR PASSIVE CORE COOLING EXPERIMENTS

TEST MATRIX/PHASE II

TEST/ DESCRIPTION	*FORESEEN PHENOMENA/AIMS*	*NOTES*
GDE11 / low power (<80kW), 2.0% (4 mm) cold leg break with gravity driven ECCS.	*Gravity driven ECC effectiveness. Flow regimes. Phase separation and stratification. Condensation. Break flow. Coolant distribution. Investigation of the blowdown process at low pressure. Break location.*	*For all PCC-tests pressure = 40 bars. Steady-state initial conditions. No depressurization of the primary or the secondary system.*
GDE12 / 2.0% (4 mm) cold leg break with gravity driven ECCs.	*As above. Reproducibility.*	*As above. Initial conditions as well as possible same as above.*
GDE13 / 2.0% (4 mm) cold leg break with gravity driven ECCs. Sudden depressuriza-tion of the primary system.	*Effect of depressurization.*	*As above. Depressurization of the primary and the secondary systems.*
GDE14 / 0.5% (2 mm) cold leg break with gravity driven ECCs.	*CMT natural circulation. Effect of break size.*	*As above without depressurizations. High pressurizer level at test initiation.*

4. THE RELAP5 REPRESENTATION OF PACTEL

A base RELAP5/Mod3 input deck for PACTEL was modified to include the gravity driven emergency core cooling system. The additions included the CMT and associated pressurizer and cold leg pressure balancing connections. The model was composed of 257 hydrodynamic volumes, 284 junctions, and 394 heat structures. Although this input served as a starting point for the calculations, many modifications were made to it during the course of the analysis. Revisions were made to the original model as new information became available and as input deficiences were discovered. Those modifications that were expected to have the most effect on these calculations, and the corresponding input changes are discussed next.

In the CMT, modelled as a cylinder, the effect of nodalization was investigated by changing the number of CMT nodes. These calculations showed that there was no significant difference between 2, 5 and 10 node CMT models for the overall CMT behavior. However, the amount of rapid depressurizations of the CMT varied between 2, 5 and 10 node models and none of the models corresponded to the amount or timing of the depressurizations in the tests. The results with a CMT modelled as a branch did not give any prediction for rapid pressure drops in the CMT.

It was also found that the modelling of pressure losses in the PBLs had a significant effect on CMT depressurization behavior. Unfortunately no measured data was available for pressure losses in the PBLs. A sensitivity study on pressure losses in the cold leg PBL, pressurizer PBL, and the ECC line was performed and it was found that depressurization modelling was very sensitive especially for the value of pressure loss in the cold leg PBL.

The junctions between the cold legs and the downcomer, and between the upper plenum and the hot legs were at first modelled as crossflow junctions, but later modified as normal junctions in order to achieve realistic flow paths and water levels in the upper plenum and the upper part of the downcomer. The modelling of these junctions also had an effect on the heat loss distribution in the primary system and this way to the primary pressure when the coolant flow was near stagnation.

The subcooled discharge coefficient at the break was also varied for a better presentation of the leak mass flow of the experiments.

5. COMPARISON OF TESTS AND COMPUTATIONS OF THE FIRST TEST SERIES

The test results from the transients performed in the PACTEL loop were compared to computer simulations by the RELAP5/Mod3 program. The actual starting steady state conditions in individual tests were used as input to the computer simulations. All the calculated transients began with the opening of the break valve. Also the ECC line valve and the cold leg PBL valve were opened simultaneously. Condensation of steam in the CMT was observed in all experiments.

In the calculation of the GDEØ1, there were five rapid pressure peaks against the measured one at 1860 s, Fig 3. The experiment was terminated after this. It was found that changing the maximum time step had an effect on the peak appearance. On the other hand, RELAP5 changed the flow chart from vertically stratified flow to bubbly flow in the CMT at the initiation of condensation. Also the pressure of the pressurizer, the ECC flow and the vapor content of the upmost CMT node increased at the condensation initiation.

The best approximation for the condensation induced pressure peaks was achieved in the modelling of GDEØ3 experiment, where also the oscillatory period after the condensation was modelled, Fig 4. However, there were extra pressure peaks also here.

6. EXPERIMENTAL RESULTS FROM THE SECOND TEST SERIES

In the second series of experiments condensation behavior differed a lot from that observed in the preliminary tests. As the ECC flow in the first tests stopped totally several times because of rapid and very short condensations there was now only one condensation phase which lasted much longer. Good reproducibility was achieved in GDE11 and GDE12 test. The CMT pressures in GDE11 and GDE12 tests are shown in Fig. 5. In both experiments there was a condensation phase starting at about 1700s and lasting for 300s.

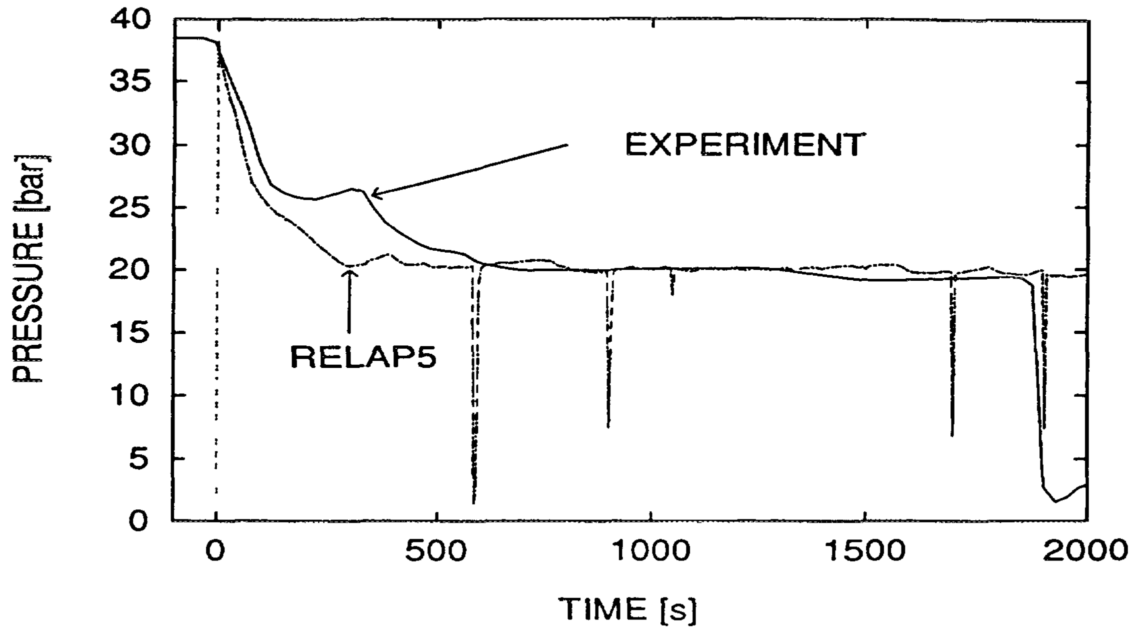

Fig. 3. The CMT pressure in GDEØ1

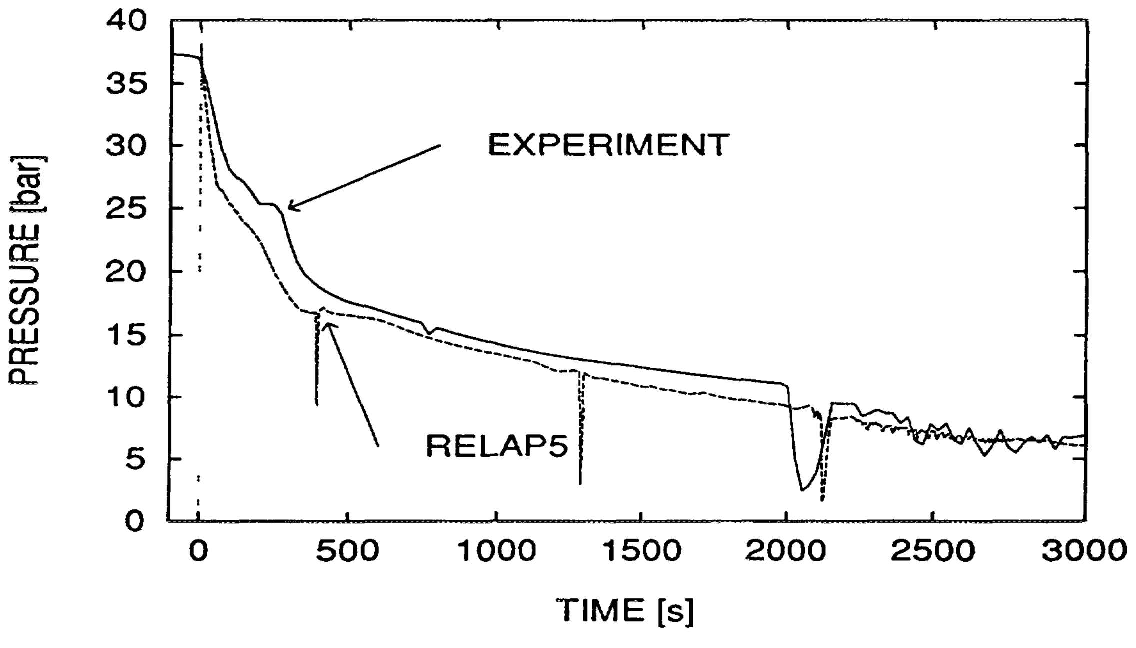

Fig. 4. The CMT pressure in GDEØ3

The operator activated primary system depressurization in stages affected to the total collapse of the vapour space, because in the GDE13 test there were three short condensations observed in the CMT, Fig. 6.

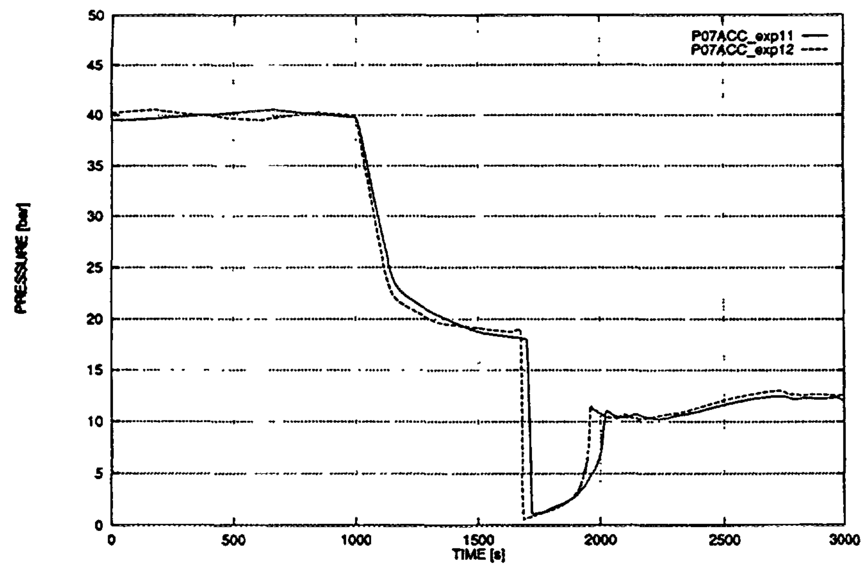

Fig. 5. The CMT pressures in GDE11 and GDE12 tests

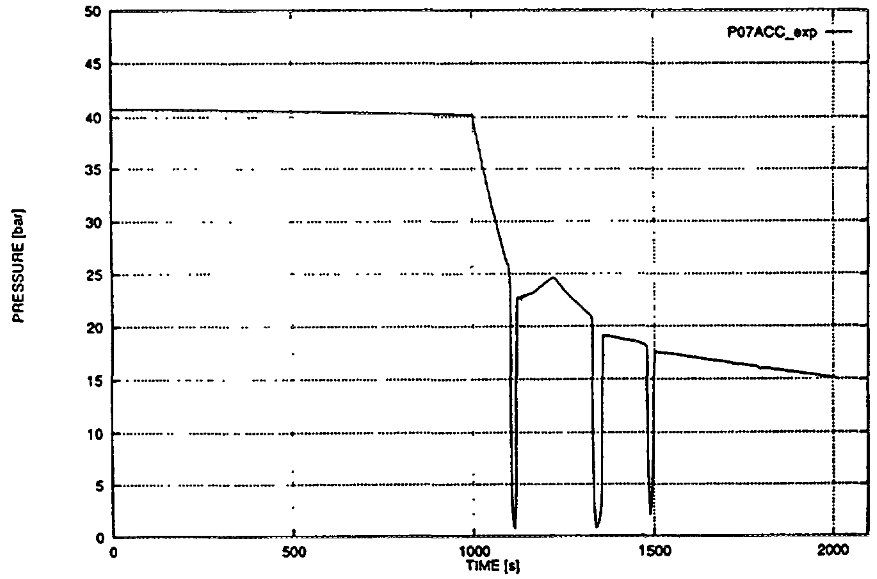

Fig. 6. The CMT pressure in GDE13

The first condensation was already at t= 1100 straight after the depressurization initiation. Similar period of short condensations were observed in the experiments of first series in both experiments with or without depressurizations. This behavior was found difficult to model in the RELAP5 analysis of the experiments of the first phase /3/. During the long condensation period in the GDE11 and GDE12 experiments the water level decreased to the top of core and even slightly below. The uncovery lasted only a short time and no significant heat-up in the core was found. In the GDE13 and GDE14 experiments no core uncovery was found.

A very steep vertical temperature gradient was formed inside the CMT in all tests. Fig. 6. shows that the temperature difference just before the condensation in the GDE11 experiment was 180 K in a water layer 0.15 m thick (the thermocouple numbering corresponds to that shown in Fig. 2.).

An effort for preventing the rapid condensation was done by carrying a thick, insulating level of hot water to the CMT with a natural circulation loop formed between the CMT and the primary system via the cold leg PBL and ECC line. For this reason the water level in pressurizer was set high and a small break size was chosen at the GDE14 test initiation. This natural circulation phase of the CMT was also in the ROSA-V/LSTF experiment /4/. With these preconditions a short natural circulation phase was then observed in the GDE14 experiment. However, this natural circulation phase was not effective enough to form a sufficient layer of hot water in the CMT. In PACTEL the total water volume above the CMT is small since there are horizontal steam generators.

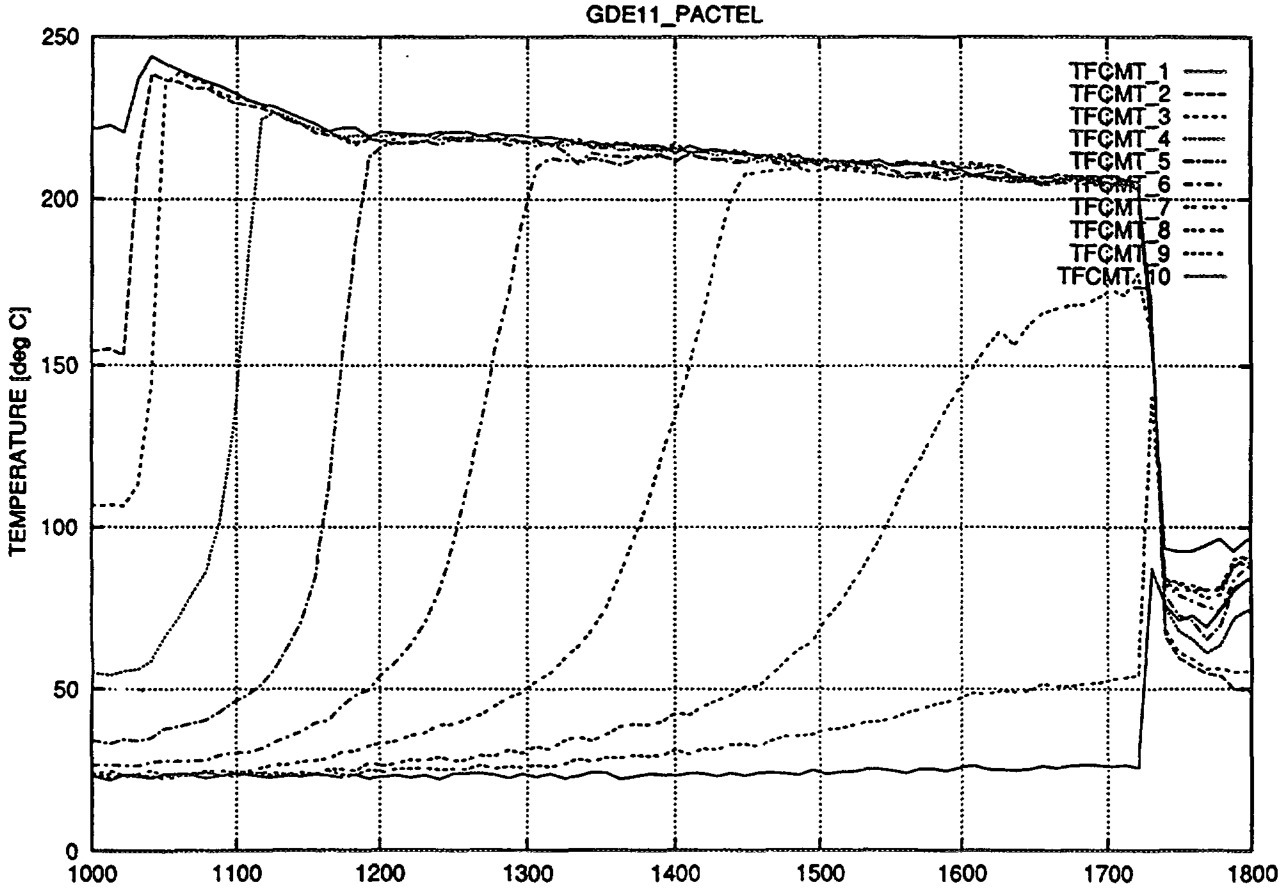

Fig. 7. Temperature distribution in the CMT at GDE11 test

7. CONCLUSIONS

No core uncovery was found in any of the tests of the first series. However, the emergency core cooling flow from the core makeup tank was stopped when rapid condensation collapsed the core makeup tank pressure. The tank repressurized rather quickly and the emergency core cooling flow was provided until the next condensation phase.

In the second series of experiments only two of the three loops of the facility were used as in the first series of experiments all the loops were active. The break was now located to the cold leg and two different break sizes were used. In one of the tests both the primary system and the secondary system were depressurized. In all the four experiments performed steam was flowing into the CMT and then later condensed to the cold water of the CMT. There were striking changes in the vertical temperature gradient of the CMT. It was experienced that condensation was then initiated easily by steam or water flow from the PBLs as the steep stratification in the CMT was broken. Especially the changes in water level in the pressurizer seemed to be responsible for most of the condensation periods.

We have also simulated the first five gravity driven core cooling experiments with RELAP5/mod3.1. The comparison of calculations and experiments show a good agreement both in magnitude and time of occurrence for most of the different physical events. The main observed discrepancy was due to limitations in the RELAP5 code to accurately predict rapid condensation in the CMT. The most critical aspect in the calculational results was that the appearance of condensation was dependent also on computational features, such as the time step and the nodalization.

Condensation of steam in the CMT could be avoided with some technical arrangements in the test facility. However, even though improvements were made to gravity driven ECC systems, we cannot guarantee that current computational models will provide accurate answers. Therefore, to build this confidence more experimental data has to be obtained and new computational models developed.

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XA9743172

ALPHA - THE LONG-TERM PASSIVE DECAY HEAT REMOVAL AND AEROSOL RETENTION PROGRAM

XA9743172

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Abstract

The Paul Scherrer Institute initiated the major new experimental and analytical program ALPHA in 1990. The program is aimed at understanding the long-term decay heat removal and aerosol questions for the next generation of Passive Light Water Reactors. The ALPHA project currently includes four major items: the large-scale, integral system behavior test facility PANDA, which will be used to examine multidimensional effects of the SBWR decay heat removal system; an investigation of the thermal hydraulics of natural convection and mixing in pools and large volumes (LINX); a separate-effects study of aerosol transport and deposition in plenum and tubes (AIDA); while finally, data from the PANDA facility and supporting separate effects tests will be used to develop and qualify models and provide validation of relevant system codes. The paper briefly reviews the above four topics and current status of the experimental facilities.

I Introduction

The Paul Scherrer Institute has recently initiated the major new experimental and analytical program ALPHA Advanced Light Water Reactor Passive Heat Removal and Aerosol Retention Program), which is aimed at understanding the long-term decay heat removal and aerosol questions for the next generation of Passive Light Water Reactors. The ALPHA project currently includes four major items: the large-scale, integral system behavior test facility PANDA (Passive Nachwaermeabfuhr und Druckabbau Testanlage; a separate effect test facility LINX (Large Scale Investigation of Natural Circulation and Mixing) for an investigation of the thermal hydraulics of natural convection and mixing in pools and large volumes; a separate-test facility AIDA (Aerosol Impaction and Deposition Analysis) for the aerosol transport and deposition in plena and tubes; while finally, data from the PANDA facility and supporting separate effects tests will be used to develop and qualify models and provide validation of the relevant system codes.

This paper presents the design concepts and scaling rationale used to define the PANDA facility, and briefly reviews the separate effects programs LINX and AIDA. The supporting system calculations for PANDA are being used to understand the behavior of the facility, relate this to similar calculations for the relevant full scale reactor.

II PANDA - An integral Containment Simulation Facility

IIA Introduction

A good understanding of the behavior of the relatively novel containment concepts proposed for the future advanced passive LWRs is of importance when assessing their safety. These concepts rely on natural circulation cooling modes; their long-term behavior includes the mixing of steam and non-condensable gases, condensation of such mixtures in parallel condenser units, large open tanks and water pools, and the mixing of fluids in large pools, air volumes, etc. Integral containment system behavior may exhibit multi-dimensional effects, due, for example, to incomplete mixing and varying modes of operation of parallel units. The PANDA facility has been designed to address such questions at a relatively large scale.

The PANDA facility consists of a 1.5 MW steam source and a number of large pressure vessels, typically 4 m in diameter and 8 m high, which can be interconnected by external piping and may contain internal structures, representing the various compartments of a variety of reactor containments. The vessels are fitted with instrumentation to measure fluid temperatures, levels, pressures and flows as well as steam and gas concentrations.

Currently the PANDA facility is to be used to examine multidimensional effects for the General Electric Simplified Boiling Water Reactor (SBWR) decay heat removal system. The SBWR utilizes two types of condenser units (Fig. 1) to remove the reactor decay heat, following a Loss-Of-Coolant Accident, from the reactor containment to an outside water pool. First, there are three Isolation Condensers (IC) connected to the reactor primary system, which are to remove the decay heat during a reactor isolation at full pressure. The PANDA facility includes scaled models of these units to investigate their behavior during an accident; it will not, however, simulate their high pressure, reactor isolation, decay heat removal function. Second, there are, currently, for the SBWR and PANDA, three low-pressure condenser units connected directly to the reactor containment (Drywell), referred to as Passive Containment Coolers or PCC units. The experimental facility PANDA will examine, on a large scale (1/25 volumetric), the system interactions between the multiple condenser units, and their heat removal capacity in the presence of non-condensable gases such as nitrogen and helium (as a simulant of hydrogen). The PANDA system behavior tests will extend the data base of previously performed experiments [2] to a much larger scale, study the interaction between the various PCC and IC units, and provide verification of integral system behavior under a variety of conditions.

The PANDA simulation of the SBWR (Fig. 2) consists of a representation of the reactor pressure vessel (RPV), reactor containment (Drywell) and suppression pool (Wetwell), as well as the Isolation Condenser and Passive Containment Cooler units and their associated water pools. Finally, condensate will be collected in a "condensate catch tank" simulating the Gravity Driven Cooling System (GDCS) pool in the SBWR. The PANDA facility is already constructed. The commissioning tests are near completion. An experimental test matrix is defined with the aim to provide necessary information for the US-NRC's certification process.

IIB General Guidelines

Early during the conceptual design phase of the facility, it was recognized that it is neither possible nor desirable to preserve exact geometrical similarity between the reactor containment volumes and the experimental facility. On the other hand, multidimensional containment phenomena such as mixing of gases and natural circulation between compartments may depend on the particular geometry of the containment building. The general philosophy followed in designing the experimental facility was to allow such multidimensional effects to take place by dividing the main containment compartments in two and by providing a variety of well-controlled boundary conditions (e.g. imbalances) during the experiments, so that the various phenomena could be studied parametrically under well-established conditions, and a behavior envelope of the system established. Carefully conducted parametric experiments can also provide more valuable data for code validation than attempts to simulate geometrically, but to an insufficient degree, the rather complex reactor system. Boundary conditions and the behavior of the interconnections between the various containment volumes can be controlled externally by software to study various system scenarios and alternative accident paths.

Beyond the general considerations stated above, in designing the PANDA facility and, in particular, the main vessels, the following general guide lines were followed:

- Full vertical height should be preserved, to correctly represent the various gravity head driving forces.

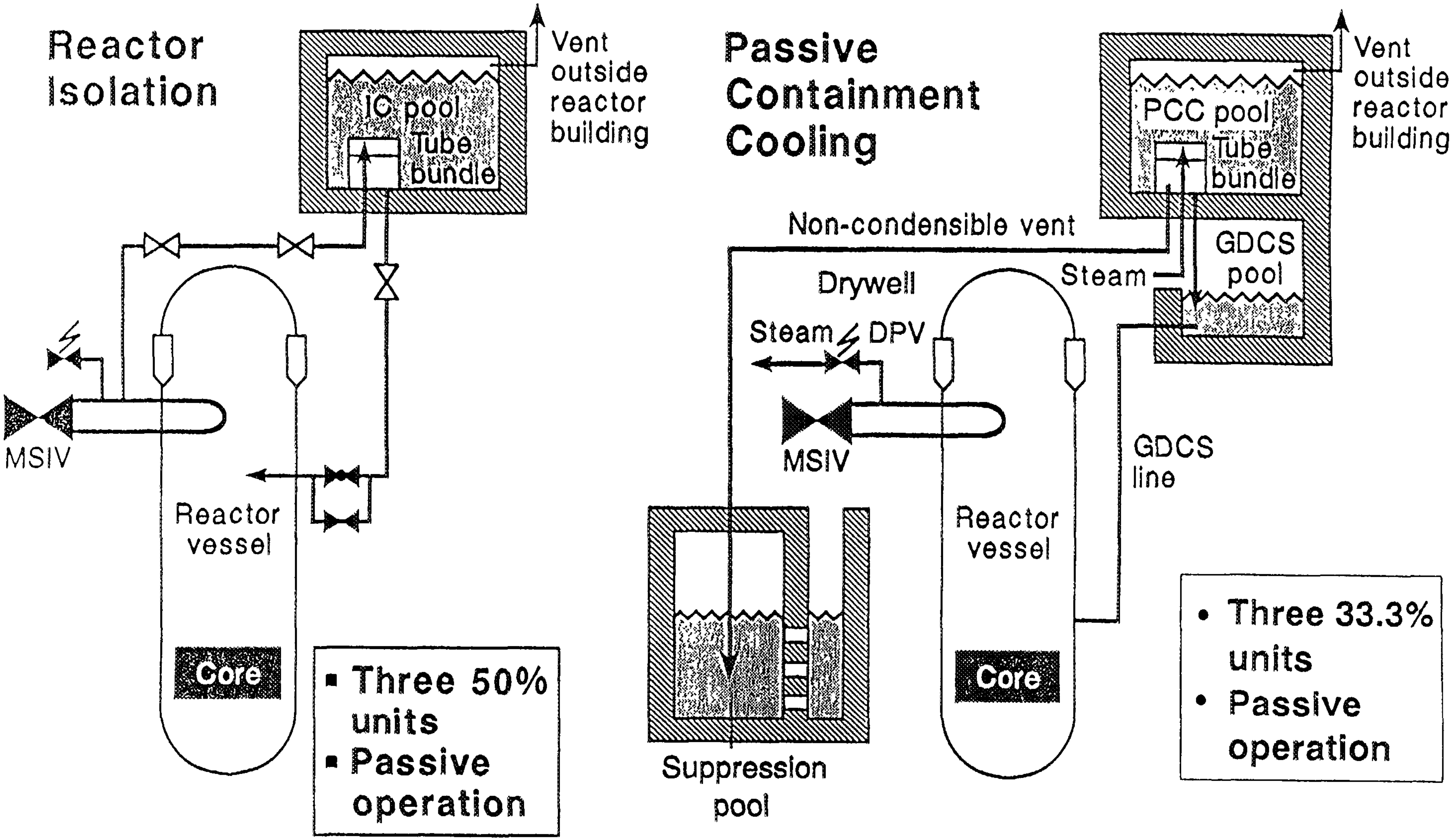


Figure 1 SBWR Isolation Condensers and Passive Containment Coolers

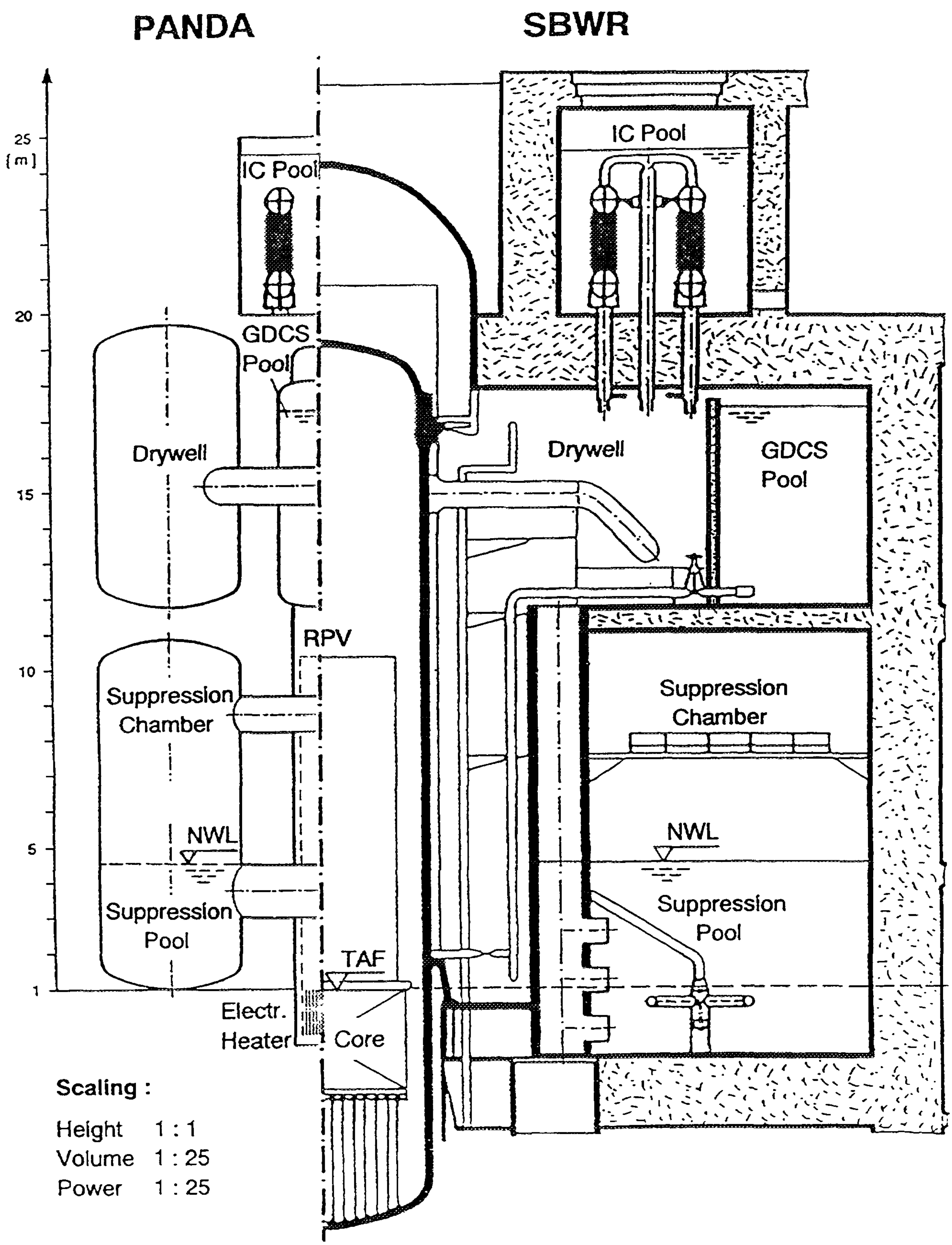


Figure 2 SBWR Containment and PANDA Comparison

– The system should be modular and use simple interconnected cylindrical vessels to simulate possible 3-dimensional effects in the SBWR annular geometry.

– Volumes should be minimized to the extent compatible with the preservation of the scaling factor chosen and the system behavior.

– The power-to-volume scaling ratio should be preserved and should be as large as practically possible.

– The experiments will be conducted under reactor pressure and temperature conditions. (The facility is designed for nominal operation at 10 bar and 180°C).

Figure 3 shows the current geometrical arrangement of the proposed PANDA facility with two interconnected Drywells, two interconnected Wetwells, the reactor pressure vessel (RPV), and a tank (GDCS Pool) to collect the steam condensate prior to returning it to the RPV. It was decided to represent the SBWR Drywell and Wetwell with two units in the PANDA facility, in order to better examine, in a systematic manner, the possible spatially non-uniform mixture of nitrogen and steam flowing through the condenser, IC and PCC units. It was considered necessary to be able to investigate the venting and purging of each of the condenser units for different mixtures of nitrogen and steam flowing into the venting of uncondensed steam, under such asymmetric conditions. The volumetric scaling of the PANDA facility shown in Fig. 3 is 1/25. Figure 2 shows the elevations of PANDA relative to those of the SBWR containment. All the SBWR heights are represented except those below the top of the active fuel (TAF). The argument for reducing the facility height by eliminating the fluid below the TAF was that this liquid is essentially inactive and is not required to correctly simulate the gravity heads. Similarly the large volume of water which is present at the bottom of the Wetwell and is only functional during reactor blowdown phase is not considered in the PANDA simulation since PANDA simulates the SBWR transients after this phase is over. Therefore, for a given facility budget it was considered preferable to eliminate these two volumes from each unit, and also to examine the energy deposition and distribution in the Wetwell pool, resulting from PANDA and so increase the overall scale of the facility. Eliminating dead volumes also decreases preconditioning times and fluid inventories and increases experimental flexibility.

II.C Scaling: The IC and PCC Condenser Units

The 1:25 scaling factor chosen for the PANDA facility is, of course, a compromise between several factors. On the one hand there is the requirement to keep the PANDA vessels within a manageable size and cost, while at the same time the desire is to construct as large a facility as possible, to provide a meaningful basis for extrapolation from the previous 1:400 scale Isolation Condenser decay heat removal experiments [2] to the full reactor scale. A critical factor that led to the choice of a 1:25 scale was the requirement that the condenser unit secondary side behavior should be representative of the units to be used in the SBWR. Figure 4 provides a schematic of a condenser module; there are two such modules per condenser unit in the SBWR (see [3] for more details). We can also see in Fig. 4 how it is possible to construct a unit at the PANDA scale by taking a slice from an SBWR condenser. Having made the decision to fabricate the PANDA condensers from a slice of the SBWR units, the only question then is how wide this should be, and Fig. 4 and shows how a 3-tube-wide slice corresponds to a scale of 1:25. This is the minimum width that will permit some tubes to be totally surrounded by other tubes. In all other respects (height, pitch, diameter, and wall thickness) the PANDA condenser tubes are identical to those to be used in the SBWR.

Adopting the above procedure for the IC produces a single unit in PANDA that has two times the tube area, at the 1:25 scale, of that of an SBWR IC. This means that the PANDA

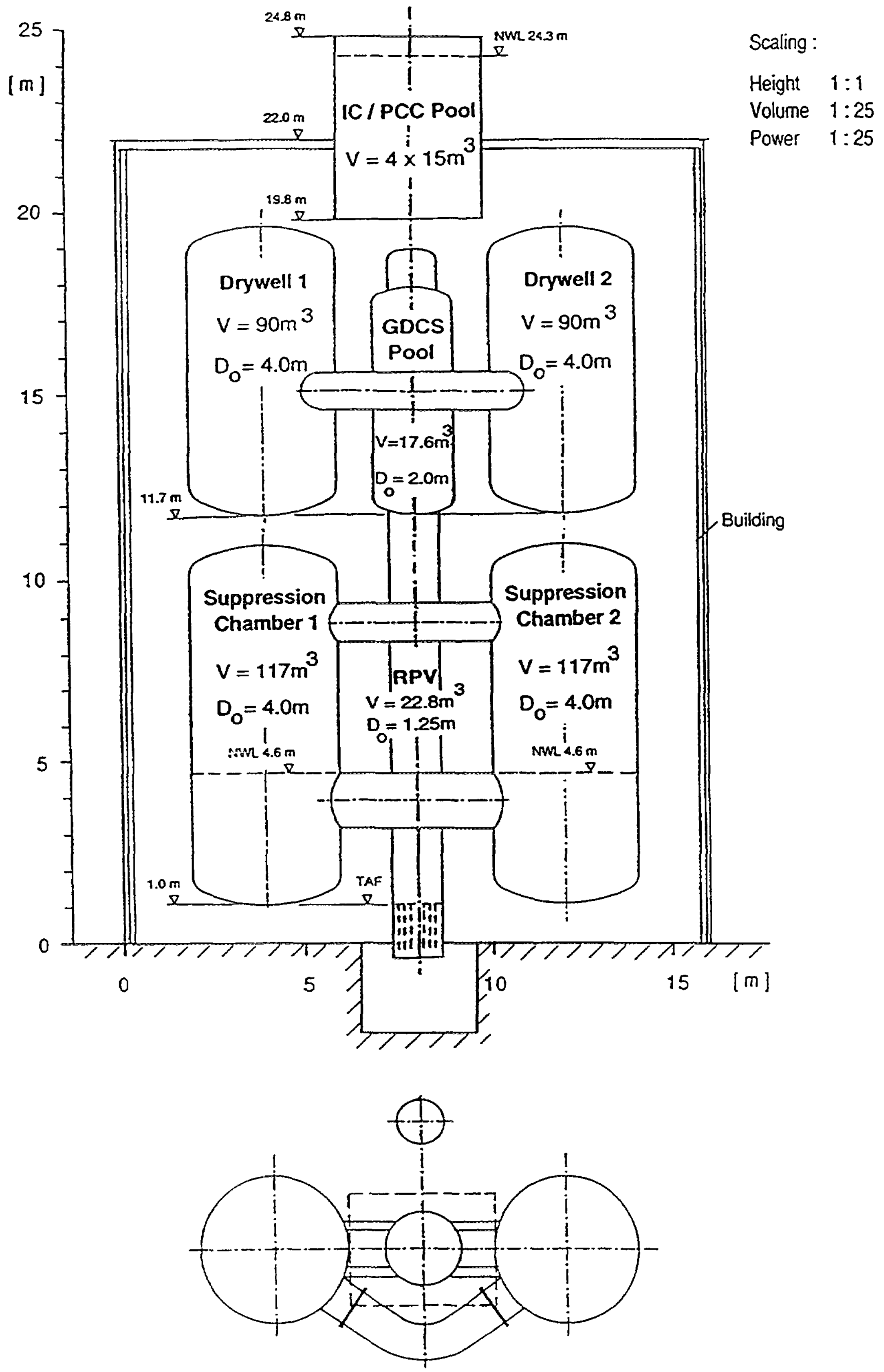


Figure 3 PANDA Experimental Arrangement

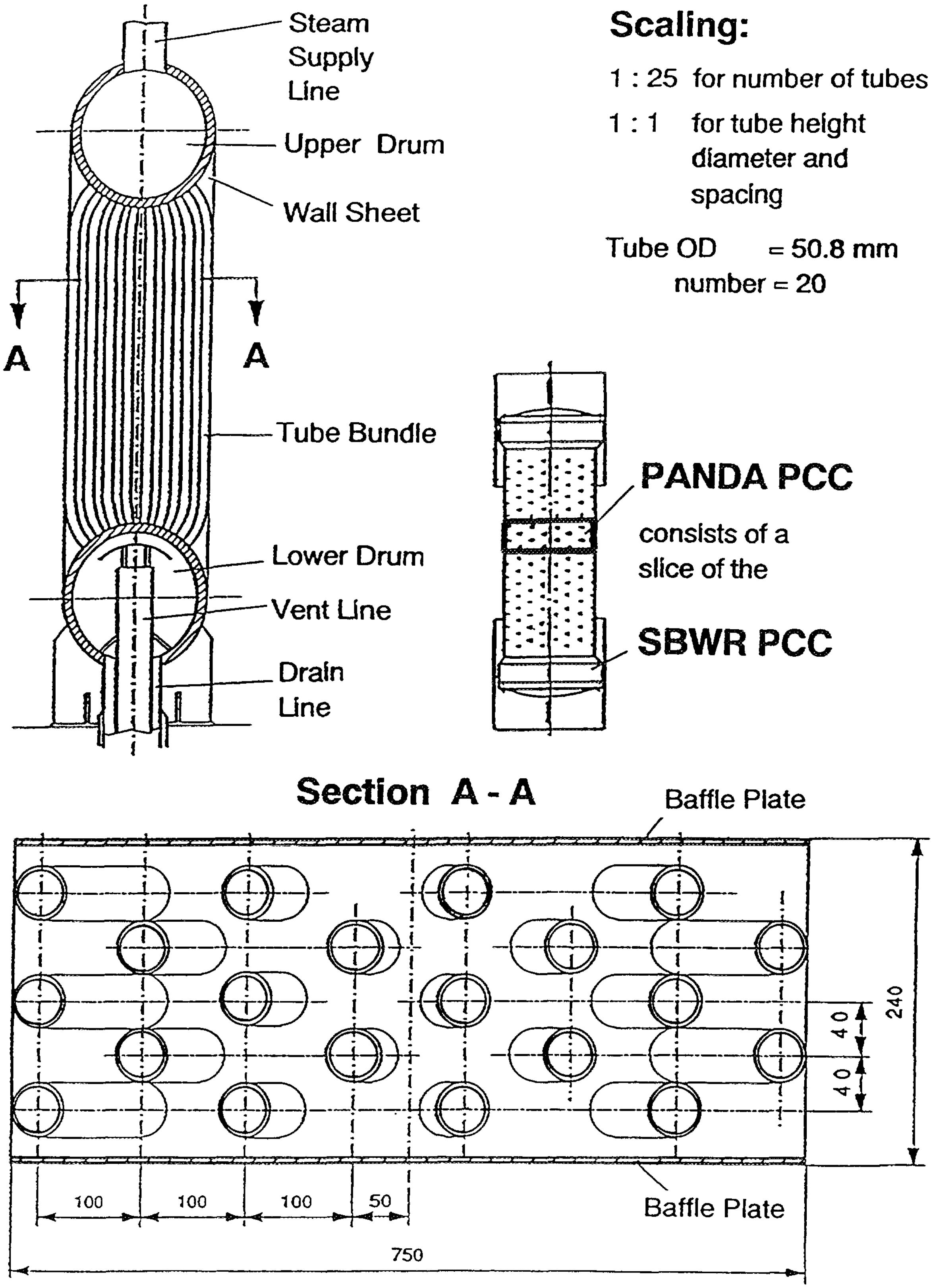


Figure 4 PANDA and SBWR Condensers

facility has four condenser units, three equivalent to the three SBWR PCC and one equivalent to three SBWR ICs.

IID The PANDA Vessels and Power Source

A schematic of the PANDA vessels is given in Fig. 3 while anisometric view is shown in Fig. 5.

As a example of the application of the general guidelines stated above, as well as of other secondary considerations, the design of the PANDA Wetwell vessel is outlined as follows:

- In order to preserve the pressure response of the entrapped non-condensable gas, it is necessary to scale the net Wetwell vapor space.

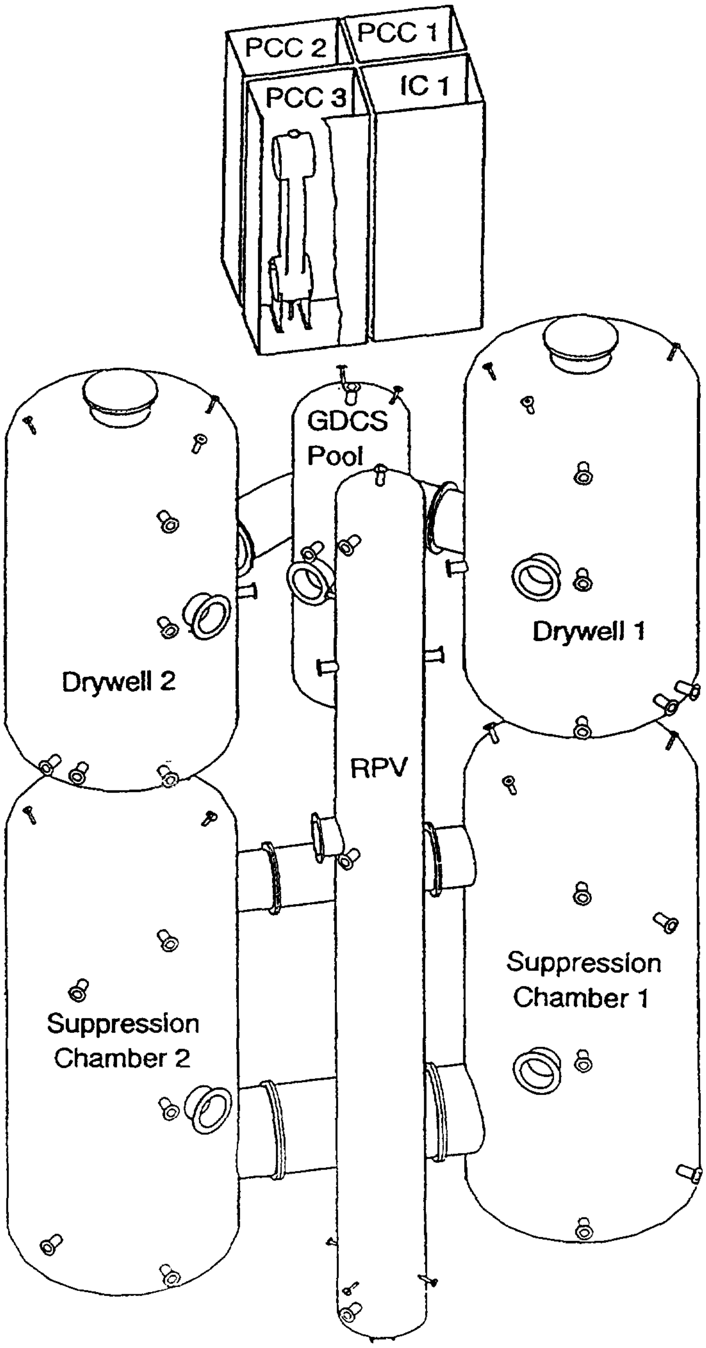


Figure 5 Isometric View of PANDA Vessels

- To have a correct representation of the evaporation/condensation processes at the pool surface, it is necessary to correctly scale the total Wetwell pool surface area.

- To provide a representative volume of water with which the uncondensed steam vented into the suppression pool can mix; the water pool depth must extend sufficiently below the condenser vent line. The suppression pool depth was also required to be large enough to accommodate at least the topmost main (horizontal) vent and the Wetwell-to-RPV equalization line. This was, in fact, the limiting factor in determining the pool depth.

In this manner it was possible to define the Wetwell dimensions. Similar procedures were also used to define the Reactor Pressure Vessel (RPV) and Drywell. In the case of the Drywell, the most important parameter to scale (for a well-mixed system) is the total volume, since this and the power level determine the venting time of the Drywell nitrogen to the PCC units.

The lower part of the Drywell volume surrounding the RPV was not included in the height of the PANDA Drywell volume, since it was felt that possible natural circulation phenomena taking place in this annular volume (heated on one side by the RPV) could not be adequately modeled. The volume of the annular space was, however, included in the PANDA Drywell volume.

For ease of construction it was considered desirable to have the Drywell and Wetwell tanks of the same diameter. Not all processes, and in particular the detailed mixing of the nitrogen and the steam from the RPV in the Drywell and the mixing of the uncondensed steam with the suppression pool water, can be accurately simulated in a scaled facility such as PANDA. In these instances separate effects studies, both experimental and analytical (see Section 3), will be used to guide parametric studies in the PANDA facility.

For example, nitrogen may be injected into the Drywell to simulate the slow convection of trapped nitrogen from a compartment with a restricted connection to the main Drywell.

The last two vessels shown in Figs. 2, 3 and 5, are those of the condensate catch tank (labeled GDCS pool) and the IC/PCC water pool. The requirements for these two vessels are somewhat different from those of the RPV, Drywell, and Wetwell. For example, for the PANDA IC/PCC water pool, in addition to providing sufficient water to keep the condenser tubes covered for a reasonable time (say 24 hours), the main requirement was one of flexibility. An element of the design was the requirement that the IC/PCC units could be re-configured in as many ways as possible, to follow possible changes in the SBWR design, without major impact on the program cost and/or time schedule. Also, there was a requirement to have the capability of re-filling the pool, during the course of an experiment, with water at different temperatures, in order to examine a variety of possible SBWR long-term depressurisation strategies. As can be seen from Fig. 5, the IC/PCC pool has four inter-connected compartments and are placed on the roof of the PANDA building (Fig. 3).

The power to the PANDA facility is provided by electrical heaters placed near the bottom of the RPV (Fig. 6). The heaters are not designed to represent the reactor core, but are placed so that their tops have the same relative elevation as the top of the active fuel (TAF) in the SBWR. The power level required for PANDA was determined on the basis that a PANDA transient would be initiated after reactor blowdown and follow the emptying of the GDCS water into the RPV. These events are predicted to occur within one hour of accident initiation and reactor scram, and so the required PANDA power level was set to be equal to the scaled decay heat one hour after scram. For a 1800 MW reactor, the decay heat after one hour is approx. 24 MW and so, for PANDA, approx. 1 MW of power is required. In order to provide flexibility of operation, the PANDA heaters have a maximum installed capacity of 1.5 MW. A controller is provided to follow accurately any given decay heat curve.

II.E Valves, Piping, and other components

The piping configuration of the PANDA facility is shown in Fig. 6, and a number of features of the design are worthy of explanation.

- All the lines (pipes) are valved to provide maximum flexibility and ease of re-configuring the system with minimum cost and time delay.
- The schematic (Fig. 6) shows the steam line, drain line and vent line to each of the condenser units, and the PANDA simulation of the main (horizontal) vents. The main vents are not be fully scaled, since they are not predicted to clear during the course of a PANDA transient due to the small Drywell to Wetwell pressure drop, which results from the fact that the PANDA transients are not initiated until one hour after scram.
- Also shown are two vacuum breakers, each one connecting one of the two Drywell-Wetwell vessel combinations. The vacuum breakers are predicted [3]to have a major influence on the behavior of the PANDA facility and are therefore a critical element in both the design of the SBWR containment and PANDA. Programmable control valves are therefore used in PANDA to simulate the SBWR vacuum breakers; this will allow a variety of SBWR vacuum breaker designs to be tested with only software, rather than hardware, changes.

Finally, Fig. 6 shows the water and gas supply lines that are used to initialize any given PANDA experiment. Sufficient flexibility is built into the facility to investigate the effect on the transient behavior of, for example:

- A variety of suppression pool water temperature distributions, e.g. well mixed, stratified,...
- Water pools in the Drywell to simulate liquid line breaks, e.g. GDCS or IC return line breaks.
- A variety of IC/PCC water pool temperature distributions.

II.F Heat Losses and Heat Capacities

Major factors that can influence the behavior of a small scale test facility, in comparison to the reactor, are the relative magnitude of heat losses and system heat capacities. In a wide range of integral test facilities it has been necessary to go to significant sophistication, including the use of guard heaters, to reduce heat losses to an acceptable level. In general, heat losses increase in inverse proportion to the scale of the facility, as the surface area to volume ratio increases at smaller scales. In this respect, PANDA at 1:25 scale is in a relatively good position. In addition, test facilities may have extra heat losses associated with additional valves, instrument penetrations, etc. Two design goals have been set for the PANDA facility:

- The heat losses, at all times during any transient, should be less than 10% of the prevailing decay heat level. Initial estimates indicate that this is achievable using commercially available insulation and that guard heaters are not required.
- All the piping, RPV, Drywell, Wetwell, etc. should be capable of being configured to separately estimate their individual heat losses, for the range of power levels expected during the course of a transient.

Heat losses from the SBWR containment during the first 1 to 3 days were evaluated, and found to be very small i.e. less than 1%. The pipes and the vessels are insulated in order to bring the experimental heat loses to the values found for the SBWR.

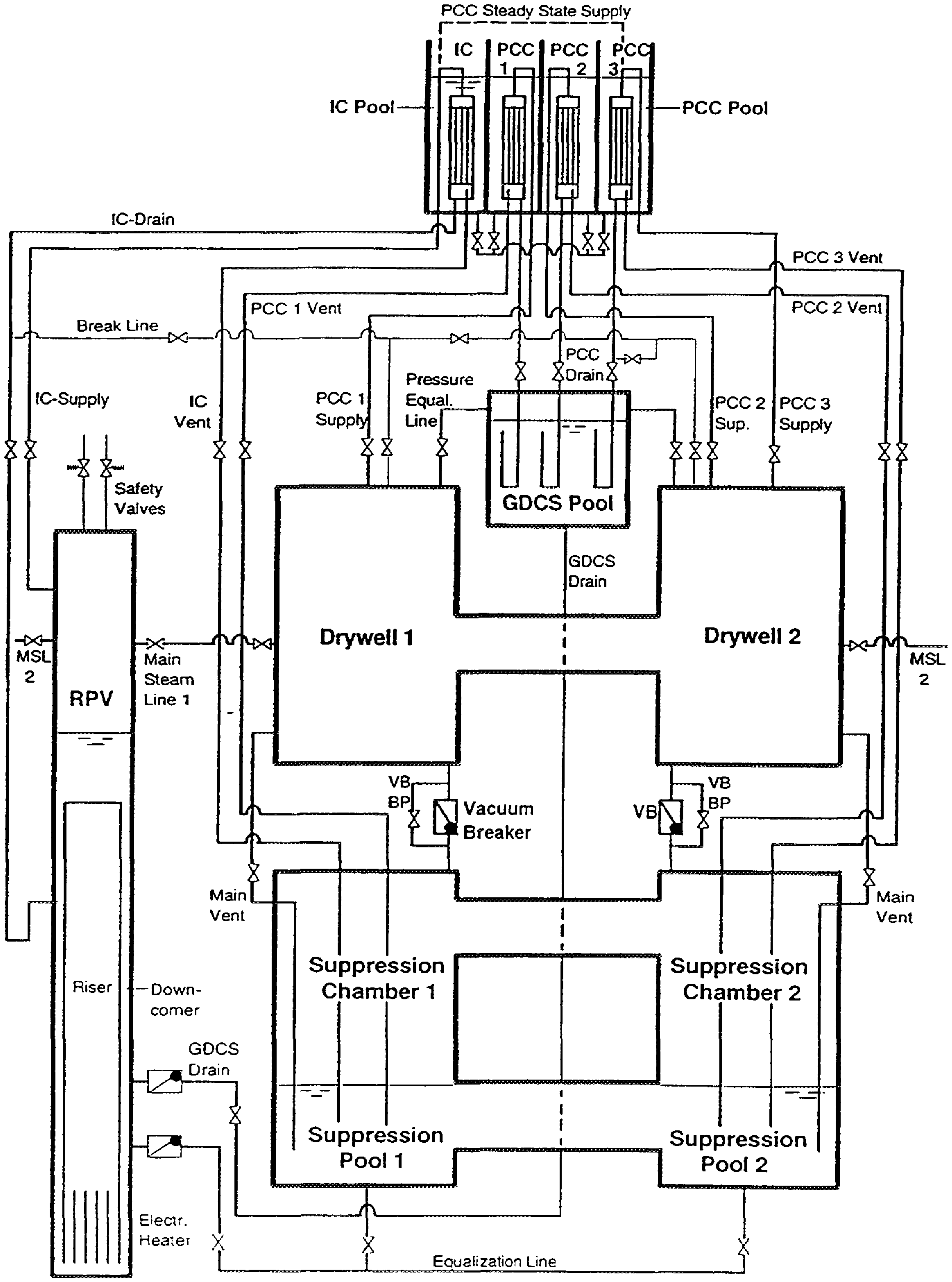


Figure 6 PANDA Schematic Including Piping Configuration

IIG Instrumentation

For basic types of measurements are made to monitor the behavior of the PANDA facility and to provide information for analytical code qualification. These are:

- vessel wall, vapor space and liquid temperatures
- absolute and differential pressures, differential pressures to measure water levels
- vapor and liquid mass.

As an example of the number and location of instrumentation to be used, Figures 7 and 8 show the distribution of the mass flow, phase detectors and oxygen sensors. As can be seen the mass flow measurements are concentrated in the steam/nitrogen and water pipes.

II.H Initial Conditions

As was described above initial conditions will be established in PANDA equivalent to those in the SBWR containment about 1 hour after reactor scram. During the first hour of an SBWR transient, the RPV will blowdown through the depressurisation system, and the emergency core cooling water in the GDCS will pools drain into the RPV. As the blowdown proceeds a large fraction of the nitrogen in the Drywell will be swept into the Wetwell leaving typically less than 10% in the Drywell, while the transfer of this nitrogen and the compression of the Wetwell gas space will raise the pressure to between 2.0 and 3.0 bar.

The following provides an example of the conditions that might be expected at the beginning of a PANDA transient.

	Wetwell	Drywell	RPV
P_{TOTAL} (bar)	3.0	3.0	3.0
P_{N2} (bar)	2.8	0.3	—
T_{LIQ} (°C)	60	—	—
T_{SAT} (°C)	—	130	133.5

Parameter variations of the above will include

– absolute pressure (bar)	2.0 to 4.0
– Drywell Nitrogen fraction (%)	1 to 40
– Wetwell pool water surfaces temperature (°C)	30 to 70

III The LINX Program

In support of the large-scale integral system behavior PANDA tests, an investigation of natural circulation and mixing phenomena in single- and multi-phase/multicomponent systems in large pools will be conducted. This work will rely heavily on the application of Computational Fluid Dynamics (CFD) tools adapted for multiphase flow and verified against a range of both large- and small-scale separate-effects mixing and natural circulation experiments to be performed at PSI. The areas of interest and investigation include the mixing of hot and cold liquids in open pools, the mixing and energy distribution within liquid pools resulting from the submerged injection (venting) of steam and gas mixtures, and the mixing of steam, nitrogen and, possibly, other gases in large, interconnected volumes.

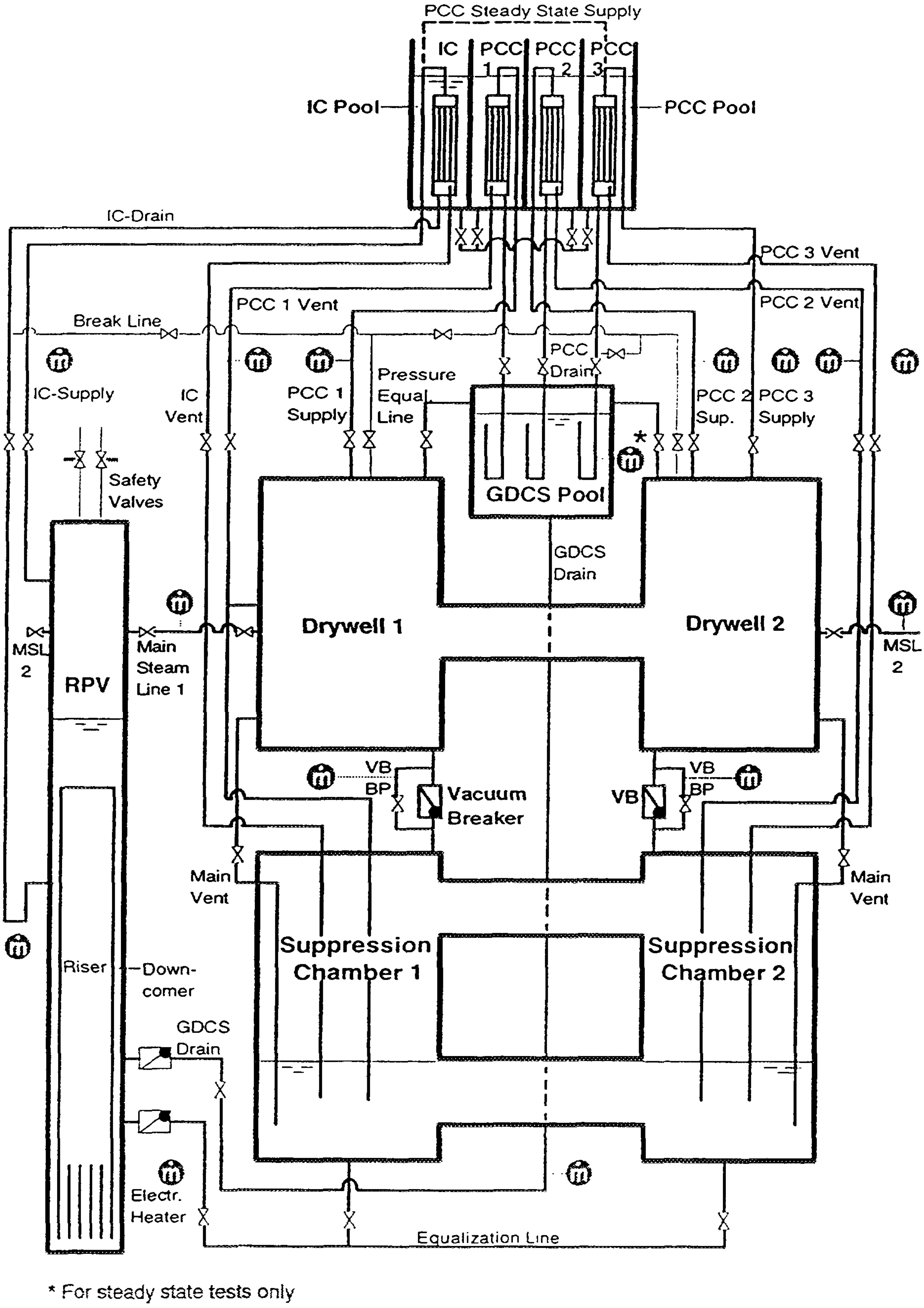


Figure 7 PANDA Instrumentation (Mass flows/flow indicators)

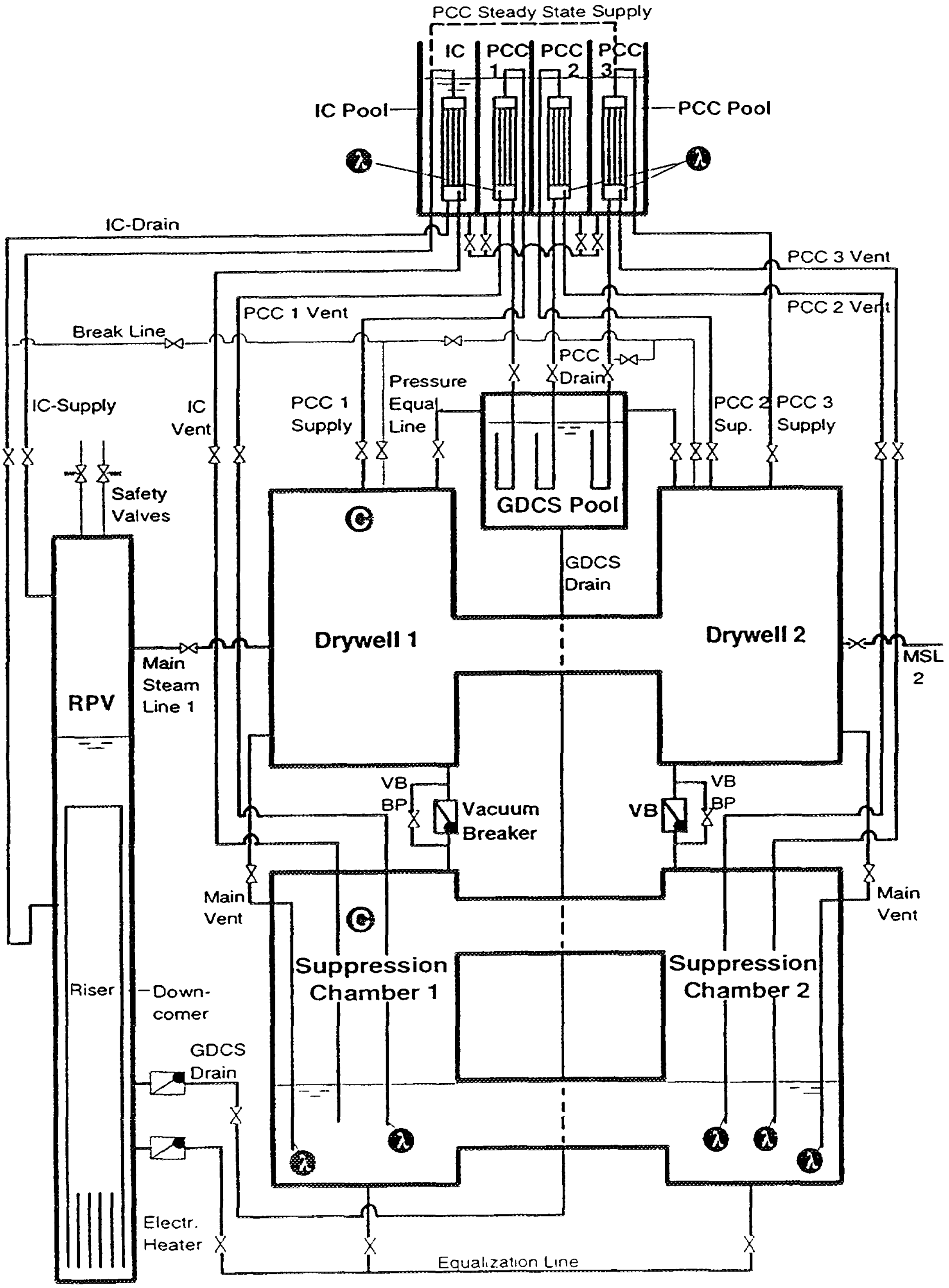


Figure 8 PANDA Instrumentation (Phase detectors and oxygen sensors)

In particular, this program of work will support the PANDA experiments and provide additional help in scaling the PANDA results to the SBWR, in two broad areas. These are: the condensation and mixing of the uncondensed steam that flows into the suppression pool from the IC and PCCs, and the mixing of steam and nitrogen in the Drywell. In the first of the two areas described, there are several phenomena that will need to be investigated separately. For example, there is the condensation of the steam initially in the presence of the non-condensable gas (nitrogen), and then there is the mixing of the resultant hot water with the bulk of the suppression pool as the hot water rises in a narrow buoyant plume to the pool surface. An initial investigation of the last of these effects was initiated at PSI [4]with the performance of some small-scale thermal plume mixing experiments. Figure 9 shows both a schematic of the plexiglas tank and electrical heater used in these experiments, and examples of the resultant rise in the water temperature as the water heated by the electrical heater rises in a very narrow plume to the pool surface and then spreads down in a 1-dimensional manner as the hot water replaces the cold water entrained in the rising plume. The LINX facility, schematically shown in Figure 10, composed of a large pressurized tank, a complex piping system for non-condensable gas and steam injection and a comprehensive data measurement and acquisition system, is currently under construction.

IV The AIDA Program

Under severe accident conditions, fission products in the form of aerosols may escape from the RPV into the various compartments of the reactor containment. It is therefore possible that the PCC units, which remove the decay heat, may be subjected to aerosols. The possible formation of an aerosol layer at the tube entrance (reduction of free flow area at the tube entrance) and on the inside tube surface (reduction of free flow area in the tube) may cause a new flow distribution into the tubes. This may dynamically change the heat removal characteristics of the system. This change may appear as a result of a) the number of tubes which are properly active becomes reduced therefore, b) some of such tubes (reducing in number with time) will continuously receive more steam than they can condense, and hence, the condenser efficiency may substantially be degraded. The long-term pressurization of the SBWR containment, following a postulated severe accident, depends on the continued function of the PCC units, and this in turn on their aerosol behavior. The goals of the AIDA program are to:

a) Experimentally determine the degree of PCC condensation degradation in the presence of aerosols.

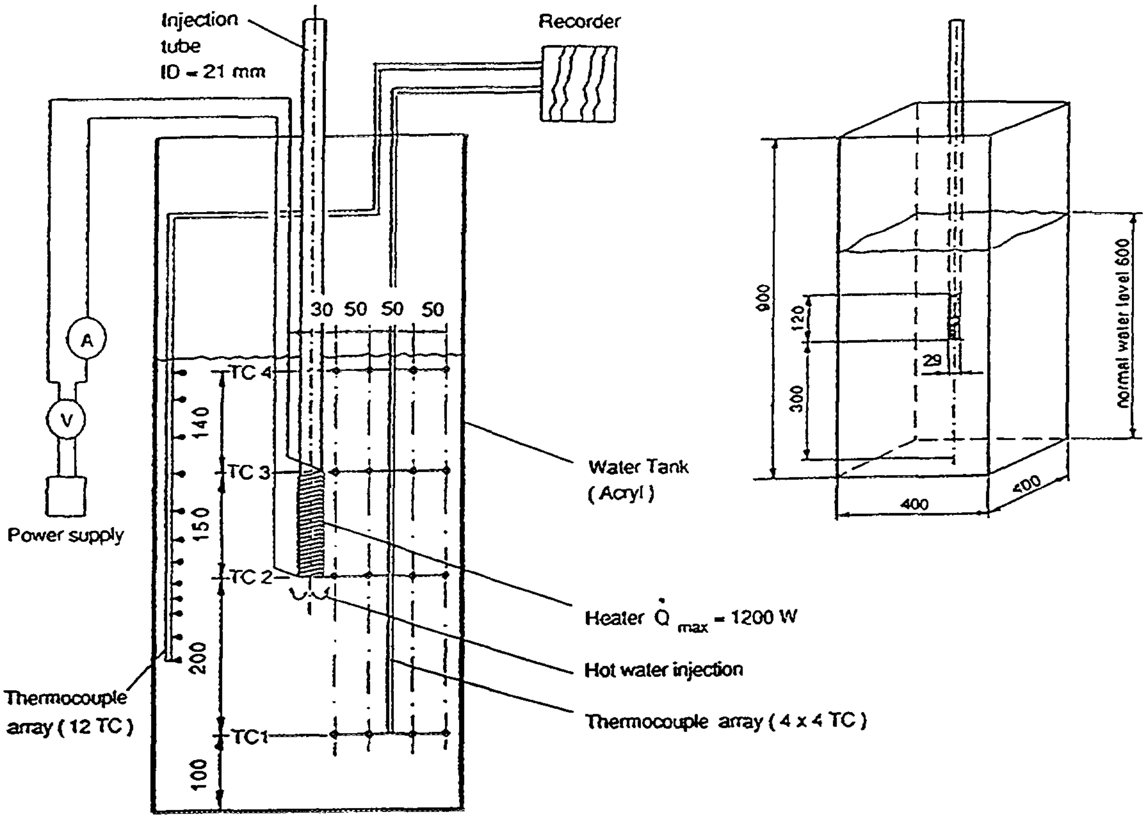
b) Investigate aerosol behavior in the upper dome.

c) Investigate aerosol behavior under strong condensation in condenser tubes.

d) Investigate aerosol transport out of the condenser unit with the condansate and non-condensable gas flow.

e) Provide the basis for the development of a physical model for aerosol behavior and its effects on the thermal-hydraulics in the PCC units.

A versatile and multiple purpose aerosol testing facility was constructed at PSI in conjunction with other aerosol programs. Two plasma torches, two reaction chambers, a mixing tank, steam and non-condensable supply systems are the main components of the facility. The system is computer controlled and can produce aerosol particles at a desired steady mass flow rate and concentration. The particles are carried with a carrier gas, composed of steam and non-condensable gas at a desired composition. The plasma torches used for aerosol generation produces aerosol particles composed of up to three components from CsI, CsOH and MnO or SnO_2 with a maximum concentration of 20 g/m^3. Experiments can be performed with 0 to 95% steam to total gas (steam and non-condensable) ratio, a steam flow rate of up to 250 kg/hr and non-condensable gas flow rate of up to 280 kg/hr and a system pressure of up to 5 bar.



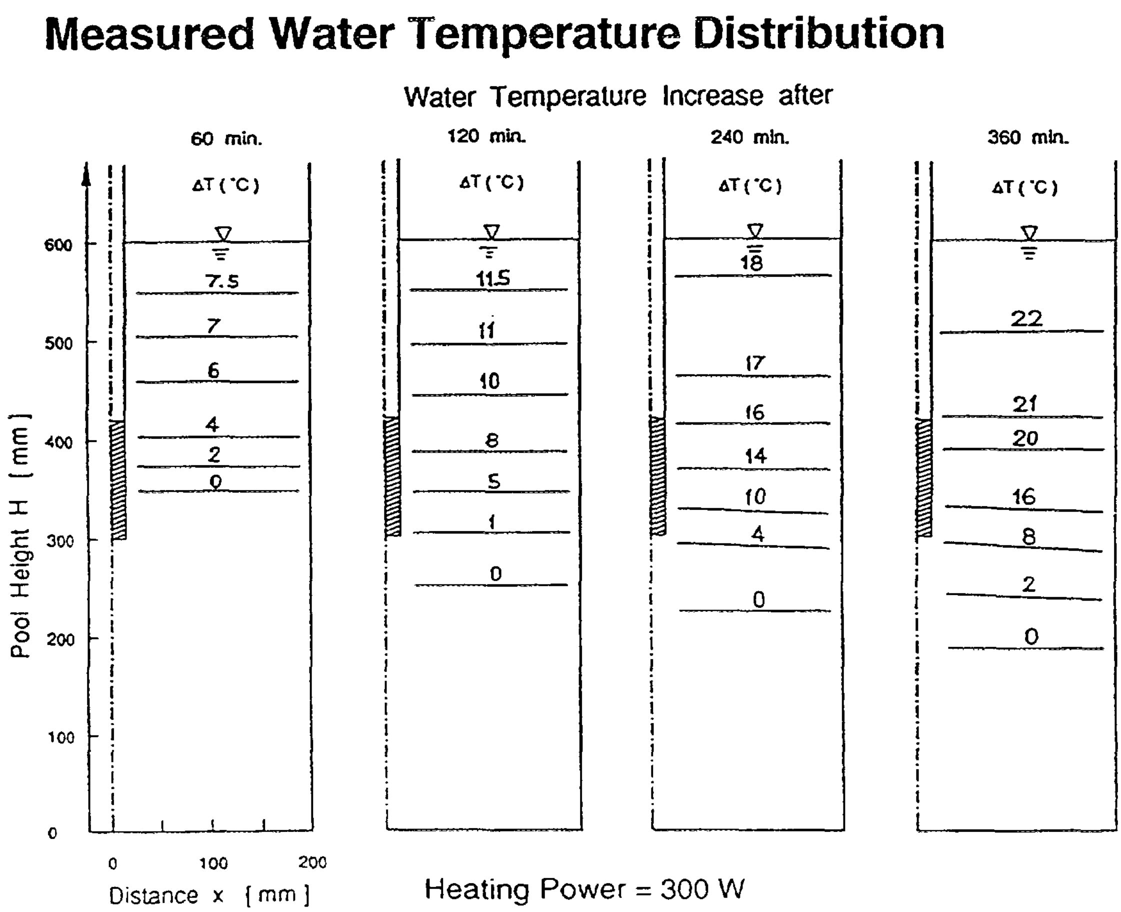


Figure 9 Mixing Experiments; Facility and Results

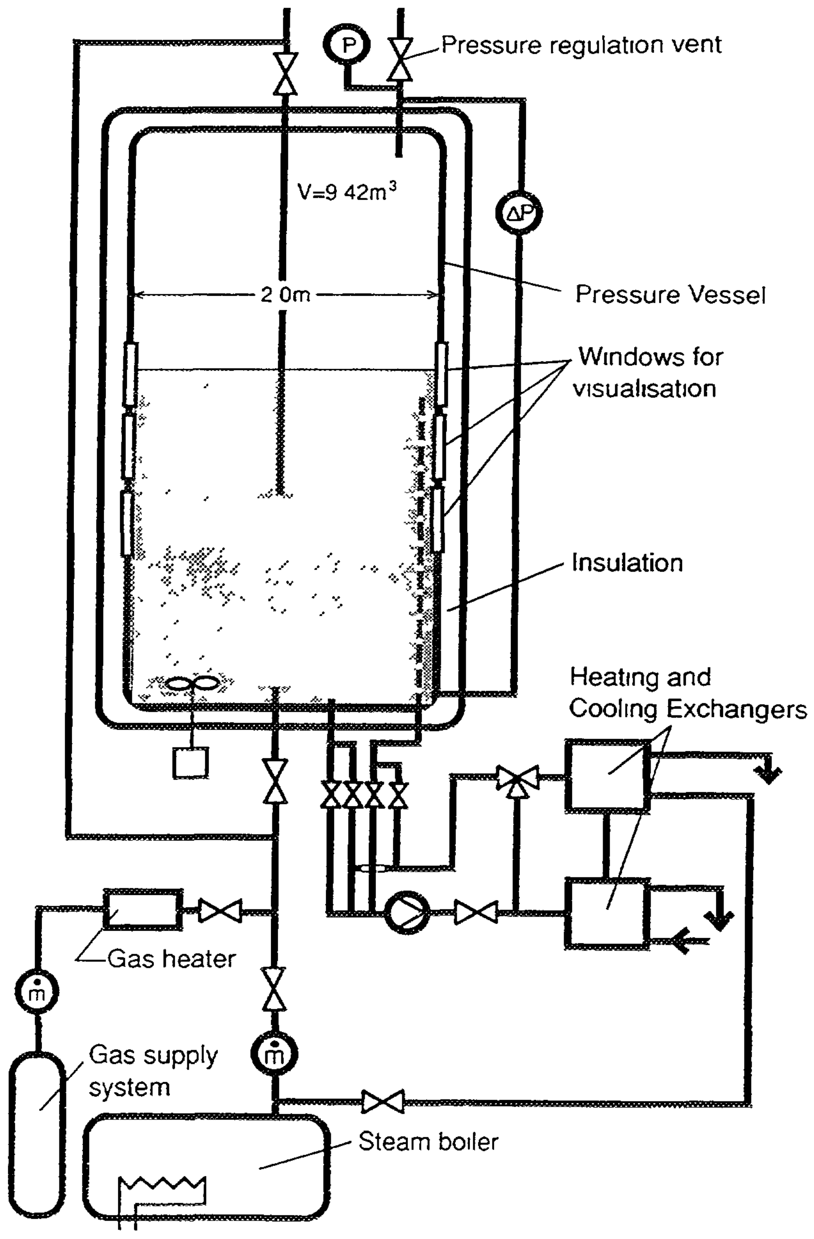


Figure 10 LINX Experimental Arrangement

A slice of the SBWR's PCCS condenser unit, containing full height 8-tubes, full diameter lower and upper dome was constructed. Figure 11 schematically shows the main components of the AIDA facility. Figure 12 presents the AIDA condenser unit. The AIDA condenser tubes are either made of glass or steel. The glass tubes are intended mainly for the visualization of the phenomena. The tubes are heavily instrumented with thermocouples to measure the gas and wall temperatures as well as the heat flux across the tube wall. The coolant channel, surrounding the tubes contain glass windows to facilitate visualization of the possible aerosol deposition-transport phenomena in the glass tubes. The water which is flowing in the coolant at a desired small velocity and at a predifined temperature of up to 80 °C simulates the heat transfer conditions expected to occur in the PCCS pool. The condensed water is collected in a tank (Condensate tank) simulating the GDCS pool. The non-condensable gas and uncondensed steam flow in a tank (Scrubber tank) which condenses the steam and scrubs the aerosol particles carried with the steam-gas flow. The condensate which is produced in Scrubbing tank is collected in another tank (Collection tank). Scrubbing and Collection tanks simulate the behavior of the Wetwell. The pressures in the condensate and collection tanks

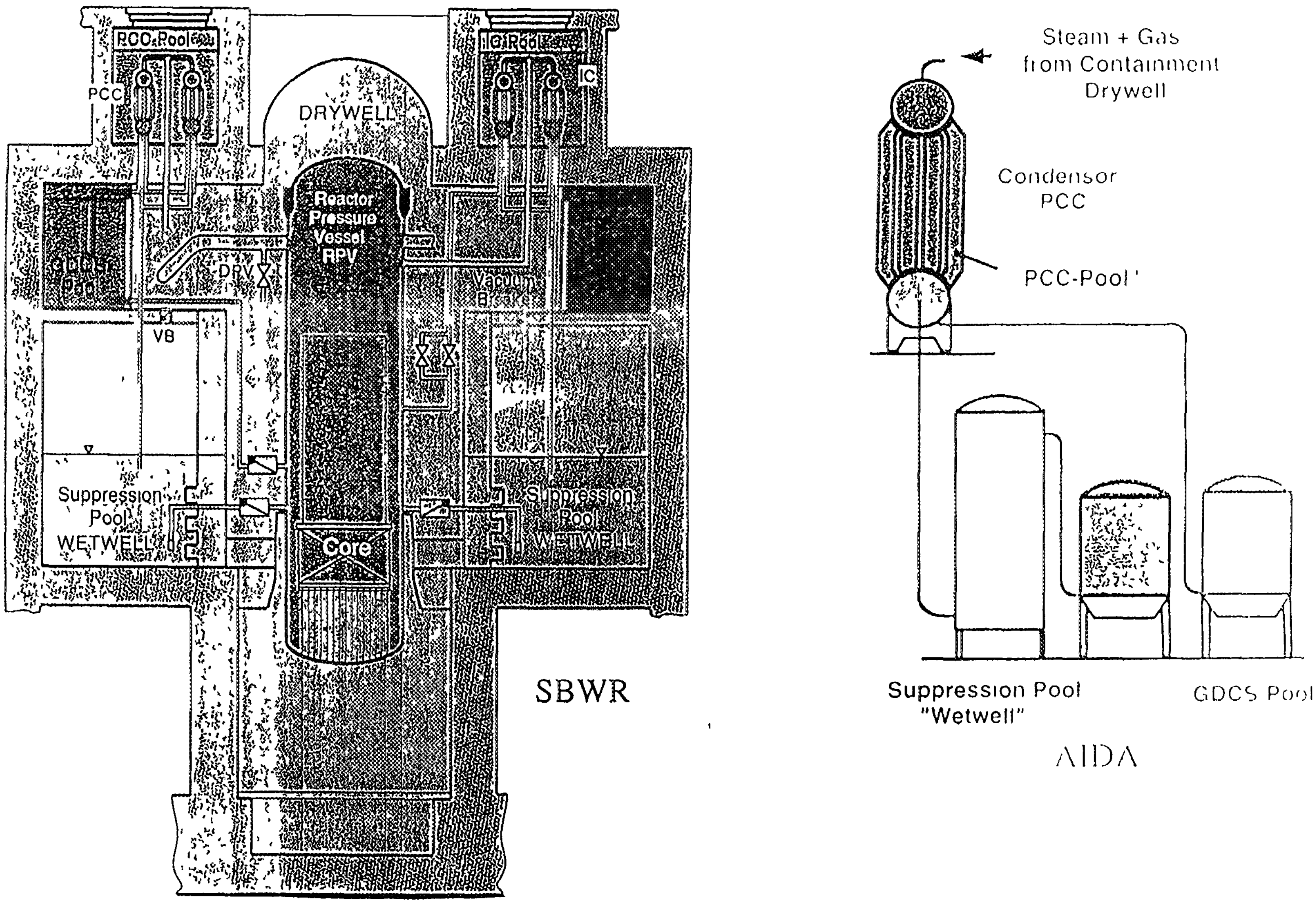


Figure 11 Schematic representation of SBWR and AIDA Test Rig

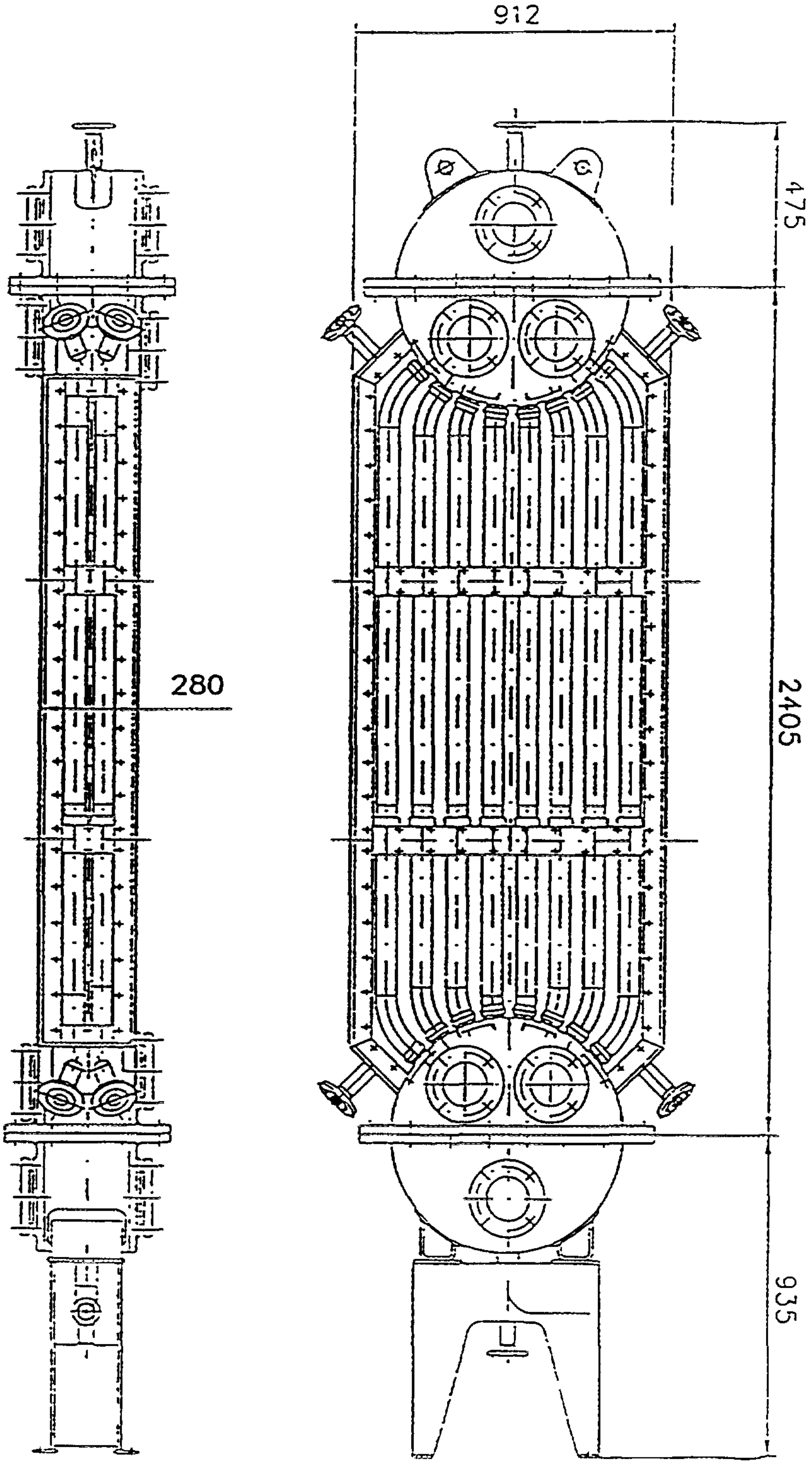


Figure 12 AIDA Condenser

are regulated to obtain the system pressure which simulates the Drywell and Wetwell pressures. The facility is instrumented with several devices to provide information on a) energy transfer due to steam condensation in and outside of the condenser, b) steam mass balance due to steam condensation in and out of the condenser, c) aerosol mass balance. The instruments provide on-line data on thermal-hydraulic behavior. The data is displayed on a computer screen to continuously monitor the system response with or without the presence of the aerosol particles. The aerosol instrumentation comprises of a) off-line devices, like filters, impactors, and deposition coupons and b) on-line devices, like, photometers, ion detection devices. A special data acquisition system is developed. Commissioning phase is close to the completion. A test matrix is prepared.

V Summary and Conclusions

The major new experimental and analytical program ALPHA initiated at the Paul Scherrer Institute has been briefly described. This program is aimed at understanding long-term decay heat removal and aerosol questions for the next generation of Light Water Reactors. The ALPHA project includes four major items: the large-scale, integral system behavior test facility PANDA; an investigation of the thermal hydraulics of natural convection and mixing in pools and large volumes (LINX); a separate-effects study of aerosol transport and deposition in plena and tubes (AIDA); while finally, data from the PANDA facility and supporting separate effects tests will be used to develop and qualify models and provide validation of relevant system codes.

PANDA consists of a 1.5 MW heat source and a number of large pressure vessels that can be interconnected by external piping to represent a variety of reactor containments. In the first instance, PANDA will be used to simulate the response of the SBWR containment to a Loss-Of-Coolant Accident. PANDA represents a 1/25 volumetric scale, the SBWR RPV, Drywell, Wetwell, and condensers. The PANDA facility has been designed to capture the asymmetric behavior of the various IC and PCC units, arising from the non-uniform spatial distribution of non-condensable gases, and their influence on the condensation process. It therefore uses two large tanks to represent the SBWR annular geometry of the Drywell and Wetwell. The facility is erected and is in commissioning phase. A test matrix has been designed to aid General Electric to provide data for the SBWR certification process.

It is recognized that no scaled experiment can possibly provide a perfect simulation of all aspects of the physical behavior of a full scale system. In response to this, and to the fact that two areas of particular importance in determining the SBWR containment pressure are the mixing of the nitrogen (and other non condensable gases) and the steam in the Drywell and the mixing of the uncondensed steam flowing into the Wetwell water pool, a companion, separate-effects program (LINX) was also initiated. LINX comprises both small- and large-scale experiments and analytical work, using simple 1-dimensional methods and 3-D CFD codes, to investigate natural circulation and mixing, of single- and multi-phase/multicomponent systems in large pools. The facility is currently under construction.

Under severe accident conditions, fission products in the form of aerosols may escape from the RPV into the various compartments of the reactor containment. It is possible that the PCC units which remove the decay heat, may be subjected to aerosols. The possible formation of an aerosol layer at the tube entrance (reduction of free flow area at the tube entrance) and on the inside tube surface (reduction of free flow area in the tube) may cause a new flow distribution into the tubes. This may dynamically change the heat removal characteristics of the system. This change may appear as a result of a) the number of tubes which are properly active becomes reduced therefore, b) some of such tubes (reducing in number with time) will continuously receive more steam than they can condense, and hence, the condenser efficiency is reduced. The long-term pressurization of the SBWR containment, following a postulated severe accident, depends on the continued function of the PCC units, and this in turn on their aerosol behavior. The AIDA program is being set up to investigate these phenomena using a scaled down PCCS condenser, associated collection tanks simulating GDCS pool and the Wetwell and the existing aerosol generation facility. The facility is erected and is in commissioning phase.

In conclusion, it is considered that the various elements of the ALPHA program will greatly enhance the understanding of the response of the SBWR containment and other similar concepts to

Loss-Of-Coolant and other accidents, and will provide a large-scale experimental facility that can be used for similar studies of other reactor systems.

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CORE MELT RETENTION AND COOLING CONCEPT OF THE ERP

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Abstract

For the French/German European Pressurized Water Reactor (EPR) mitigative measures to cope with the event of a severe accident with core melt down are considered already at the design stage.

Following the course of a postulated severe accident with reactor pressure vessel meltthrough one of the most important features of a future design must be to stabilize and cool the melt within the containment by dedicated measures. This measures should - as far as possible - be passive.

One very promising solution for core melt retention seems to be a large enough spreading of the melt on a hightemperature resistant protection layer with water cooling from above. This is the favorite concept for the EPR.

In dealing with the retention of a molten core outside of the RPV several "steps" from leaving the RPV to finally stabilize the melt have to be gone through. These steps are

- collection of the melt
- transfer of the melt
- distribution of the melt
- confining
- cooling and stabilization

The technical features for the EPR solution of a large spreading of the melt are:

- dedicated spreading chamber outside the reactor pit (area about 150 m^2)
- high temperature resistant protection layers (e. g. Zirconia bricks) at the bottom and part of the lateral structures (thus avoiding melt concrete interaction)
- reactor pit and spreading compartment are connected via a discharge channel which has a slope to the spreading area and is closed by a steel plate, which will resist the core melt for a certain time in order to allow a collection of the melt
- the spreading compartment is connected with the In-Containment Refuelling Water Storage Tank (IRWST) with pipes for water flooding after spreading. These pipes are closed and will only be opened by the hot melt itself.

It is shown how the course of the different steps mentioned above is processed and how each of these steps is automatically and passively achieved.

Finally a short overview of research areas to be addressed and a priority list of analytical and experimental support together with typical analytical and first experimental results are presented.

1 Introduction

The Defense-in-depth concept of safety has led to a very high safety standard for nuclear reactors. Great emphasis has been laid in improving features and measures for severe accident prevention. But nevertheless additional features to cope with the consequences connected to severe accidents with core melt down are discussed for future nuclear reactors. For the French/German European Pressurized Water Reactor (EPR) measures for mitigation of severe accidents are considered already at the design stage.

To cope with the consequences of a severe accident means to deal with different phenomena which may threaten the integrity of the containment or may lead to an enhanced fission product release into the environment (see fig. 1). Following the course of a postulated accident with core melt down and reactor pressure vessel melttrough one of the most important features of a future design must be to stabilize and cool the melt within the containment by dedicated measures. This measures should - as far as possible - be passive.

A lot of different concepts for retention and stabilization of the core melt has been investigated in the recent years by the cooperative partners of the EPR. The basic concept proposed is the spreading of the melt on a large area outside the reactor pit (see fig. 2), covered with a high-temperature resistant protection layer

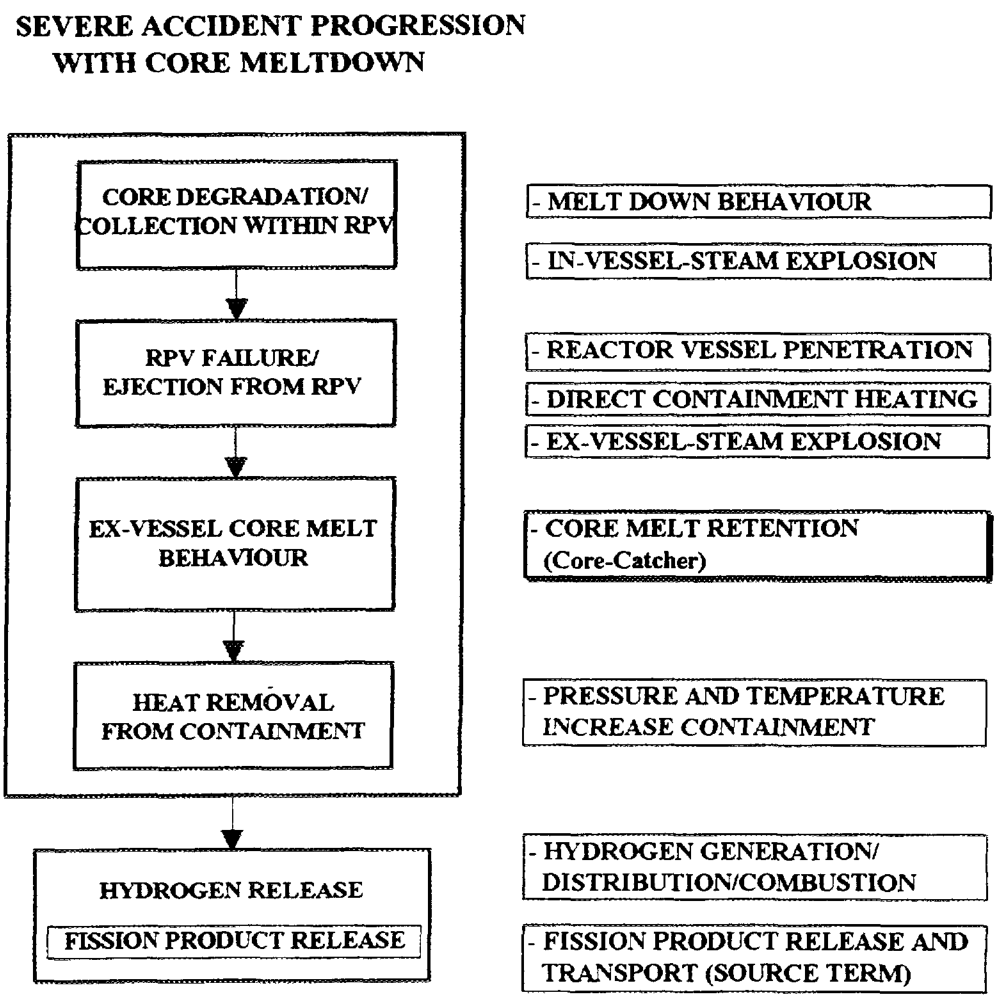

FIG. 1. Severe accidents: Phenomena to be addressed.

to prevent molten core concrete interaction. The cooling of the melt is achieved by covering it with water from the In-Containment Refuelling Water Storage Tank (IRWST).

In choosing a concept for melt retention one has to be aware that the measures taken shall be in good compliance with the overall design features of the plant and with normal operational needs and that the different mitigation measures

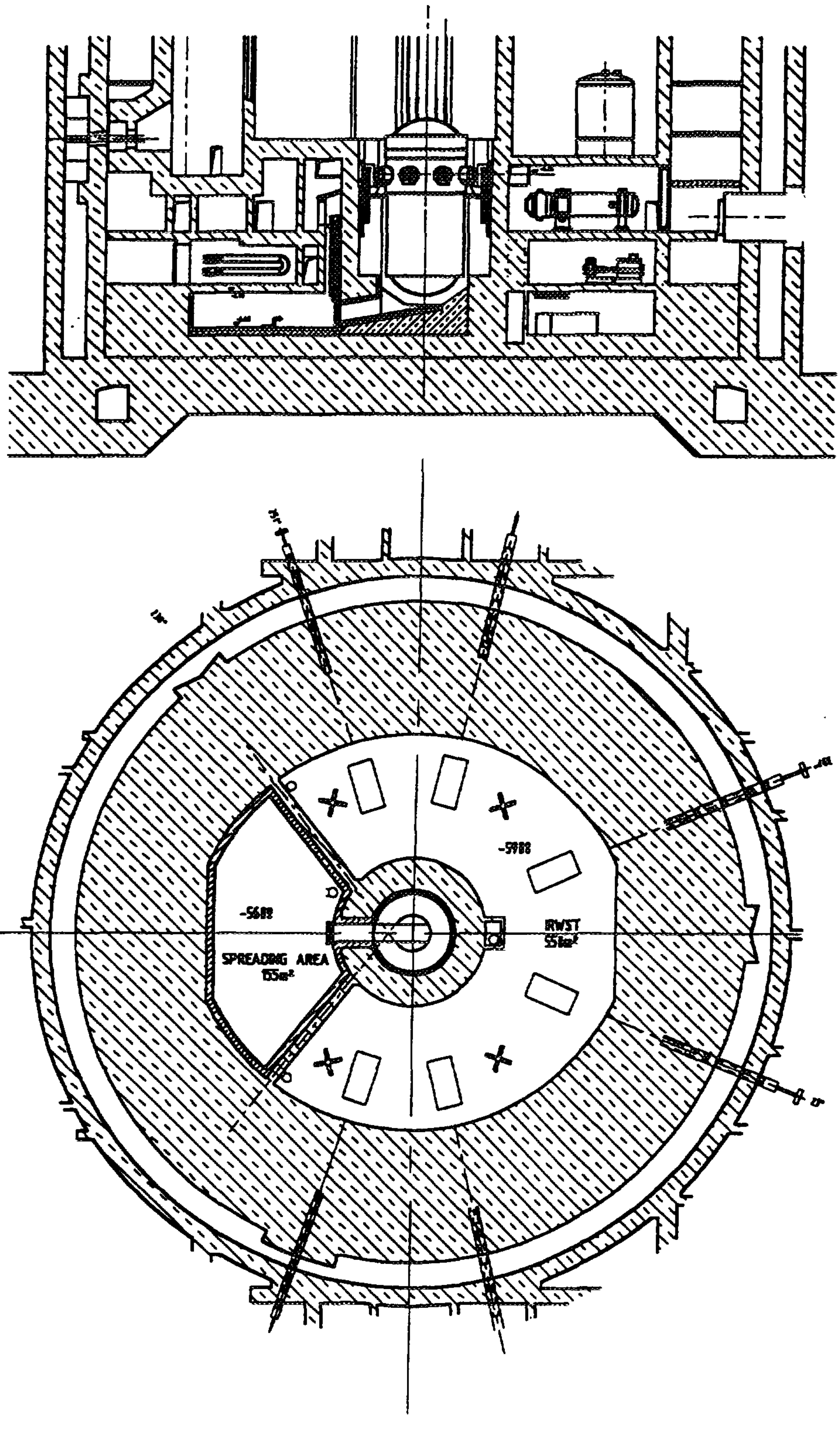

FIG. 2. ERP layout for spreading and stabilization of core melt.

implemented are not independent from each other. Further on the retention device should as far as possible be simple in construction, to minimize the physical and technological problems connected with it. This (mostly) guarantees at the same time that the costs will stay within reasonable limits.

2 Technical Features

The basic concept for the core melt retention and stabilization proposed for the EPR is the sprading of the melt on a large area outside the reactor pit and cooling from above with water. The main characteristics of the concept are the following (fig. 2):

- dedicated spreading area of about 150 m^2
- bottom and lateral structures of the compartment have protection layers designed for thermal and (if applicable) mechanical loads
- reactor pit (with the initial mechanical loads) and spreading compartment are connected via a melt discharge channel, which has a slope to the spreading area, and are in the beginning separated by a steel plate (possibly covered with refractory material), which will be molten after a certain delay time, thus allowing an accumulation and heat up of the melt.
- the spreading compartment is connected with the IRWST with pipes for water flooding after spreading; these pipes are closed during normal operation and accident conditions by plugs which will only be opened by the hot melt itself.
- the produced steam escapes via an open flow channel to the upper containment compartments

The spreading area is initially dry respectively covered with a very shallow water layer which can form as consequence of condensing steam on the walls of the spreading compartment in the case of a Loss-of-Coolant Accident. Thus energetic melt water interaction during the spreading process is prevented.

Due to the outside arrangement of the spreading area a separation of short-term mechanical and thermal loads caused by the reactor pressure vessel (RPV) failure and the long-term thermal loads caused by the spreaded melt is achieved.

3 Course of the Accident

In designing a core melt retention device one has to look not only at the melt retention and stabilization capability itself but also at the boundary conditions arising from the course of the accident - see fig. 3. (So it will, for instance, be important to know how the composition and constitution of the melt in respect to oxidic and metallic mass, its temperature and heat source distribution is.) On the other hand one has to keep in mind the goals to be fullfilled by the retention device, namely

- prevention of basemat meltthrough
- limitation of fission product release to the containment

If molten core concrete interaction can be prevented the production of additional hydrogen and other non-condensable gases is strongly limited, thus giving help to the hydrogen and pressure build-up mitigation measures.

In dealing with the retention of a molten core outside of the RPV several "steps" from leaving the RPV to finally stabilize the melt have to be gone through. These steps are (fig. 4)

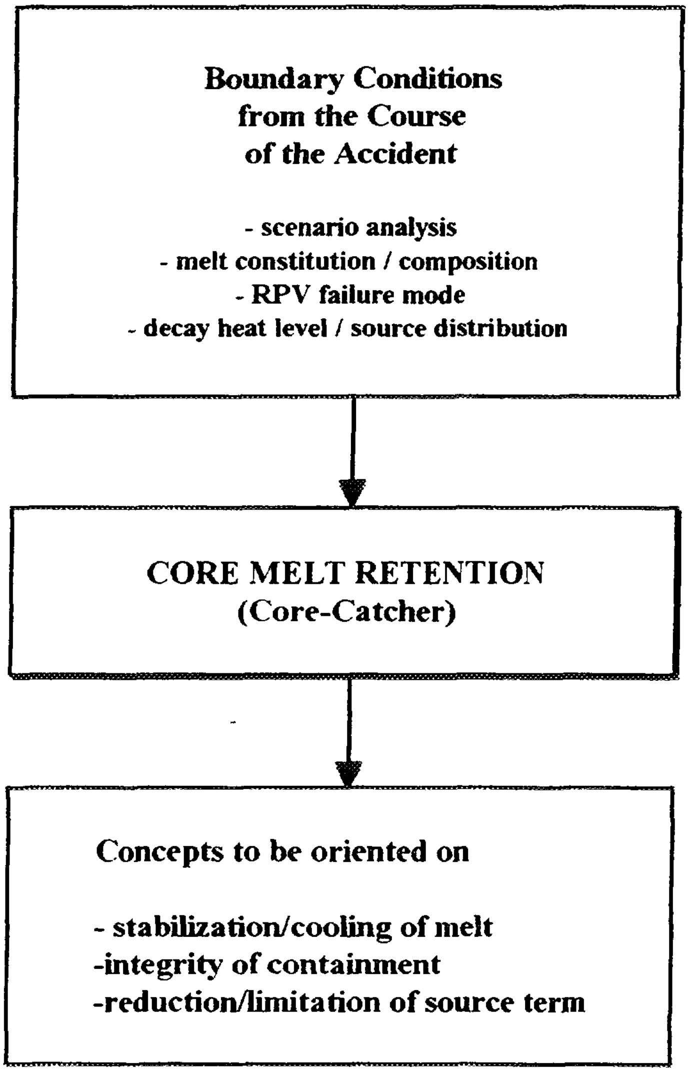

FIG. 3. Severe accident progression with core melt down.

- Collection of the melt
- Transfer of the melt to the retention device
- Distribution of the melt within the retention device
- Confining (retention) of the melt
- Cooling of the melt (by special means)
- Stabilization in the long term

These different steps apply to nearly all retention concepts, one time more one time less pregnant, and should be covered solely by passive means.

In the following it is shown how the course of these steps is processed for the EPR and how each of these steps is automatically and passively achieved. In fig. 5 the accident progression during a large spreading of the melt is indicated.

Core Degradation

As mentioned above the composition and mass of the melt when leaving the failed RPV deals as boundary condition for the further progress of the melt. The delay time of RPV failure to the first attack of the hot melt at the RPV bottom plays a significant role in a first collection of the melting core. Core degradation in

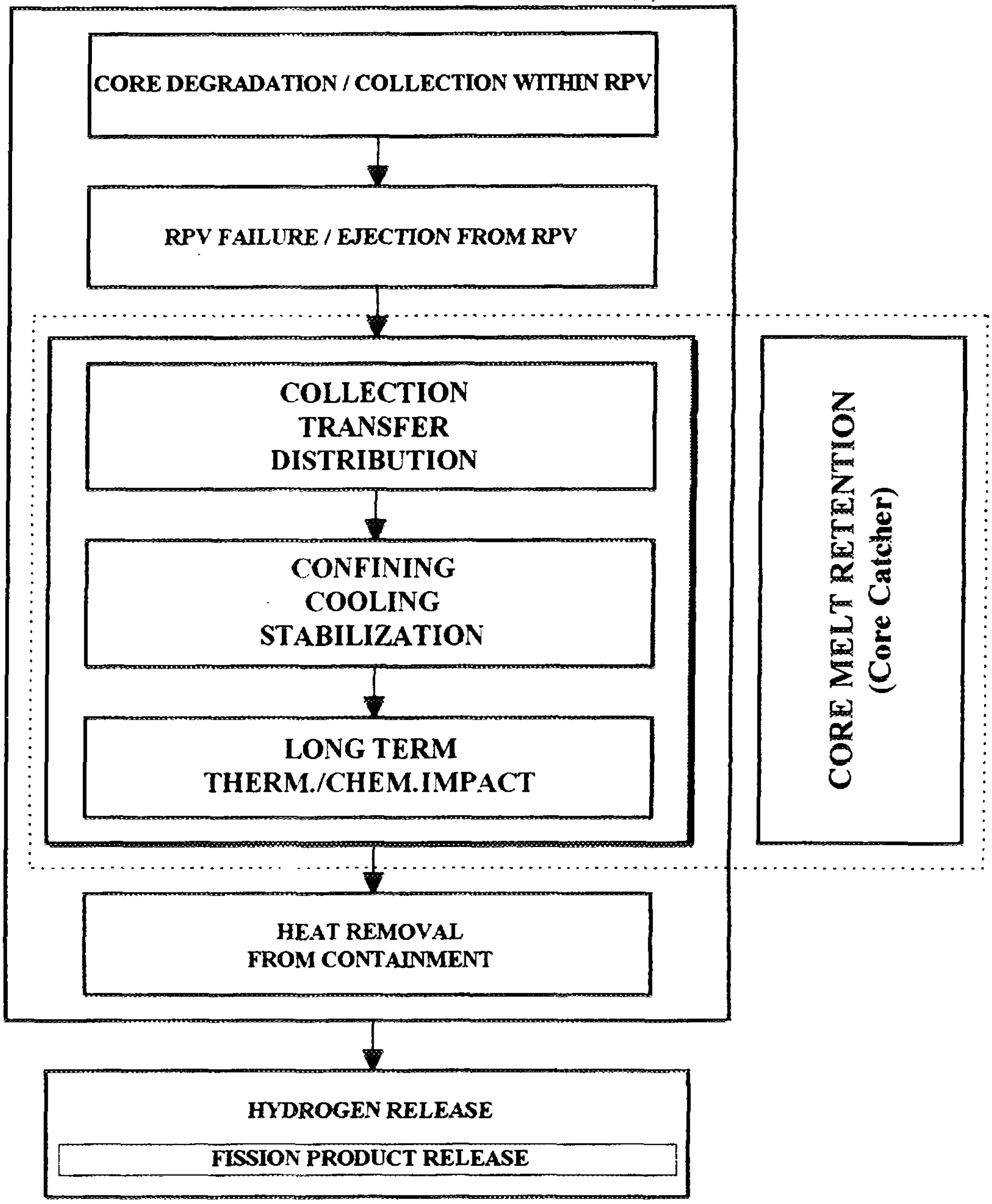

FIG. 4. Ex-vessel core melt behaviour - Phases of core melt retention.

respect to amount and slumping time depends on the scenario contemplated. To gain more knowledge for this accident phase a broad study using mostly MAAP3.0 for the time until core slumping has been performed for different characteristic scenarios. The time period from slumping to RPV failure has been addressed by special investigations.

From a scenario analysis, grouping different initiating events and boundary conditions during core degradation (e.g. water injection) into seven characteristic damage states, one can see (fig. 6) that in most cases about 155 to 200 tons of core melt are to be expected, whereas only in a small number of cases less and even more (additional mass of core internals, shrouding etc.) has to be reckoned with.

For the further evolution of the accident the kind of RPV failure is of minor importance since the melt will be collected in the reactor pit. Due to the

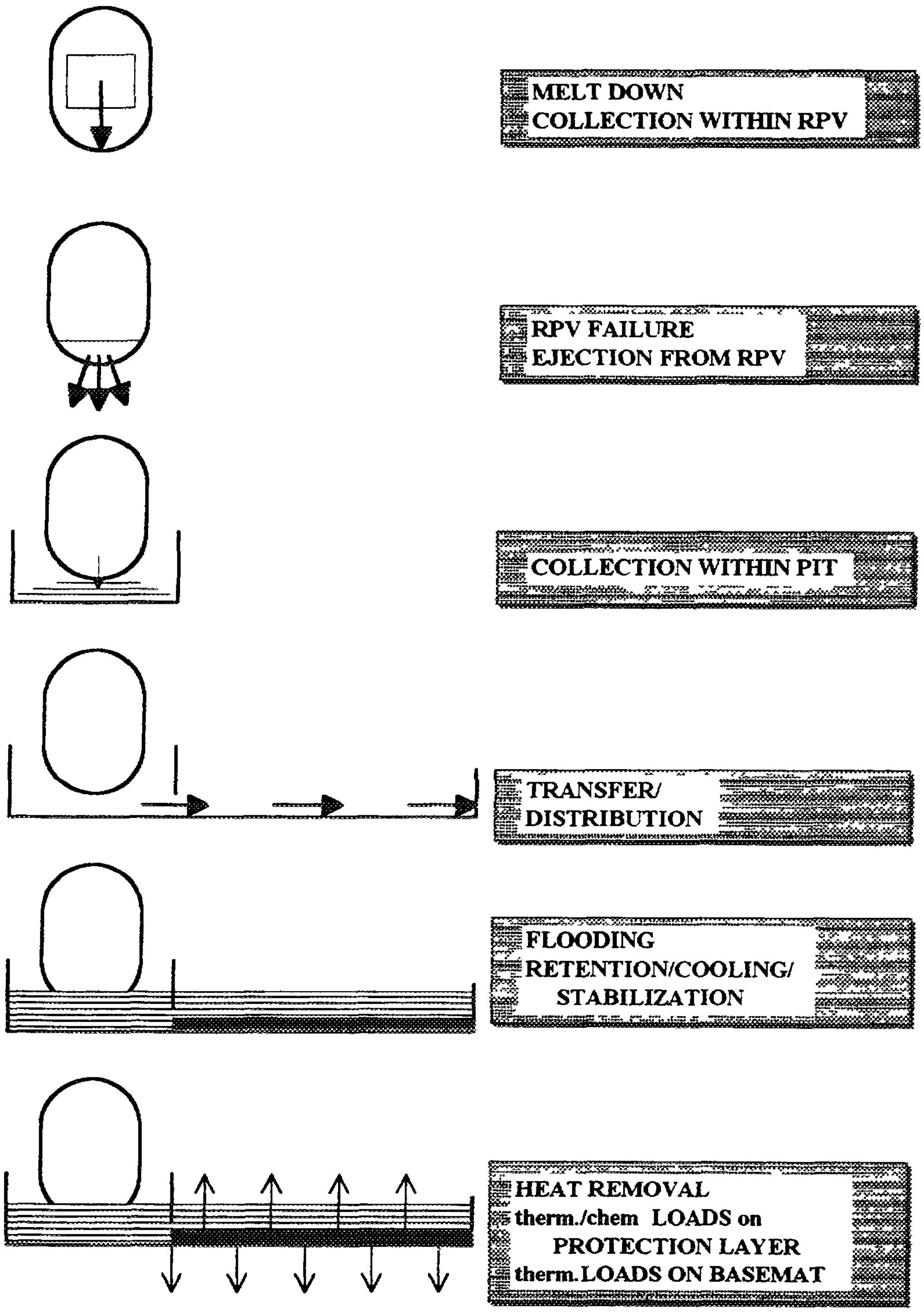

FIG. 5. Accident progression large spreading of melt.

mitigation concept for high pressure RPV failure prevention the primary pressure will be in most cases well below 20 bar (20 bar taking as layout value for dealing with RPV failure consequences).

Melt Collection within Reactor Pit

Reactor pit and spreading area are separated by a steel plate closure, covered eventually by refractory material. This closure has to be heated up and molten through by the melt, thus giving rise to an additional delay time before the melt is pouring into the spreading area. This delay time allows to further melt down the

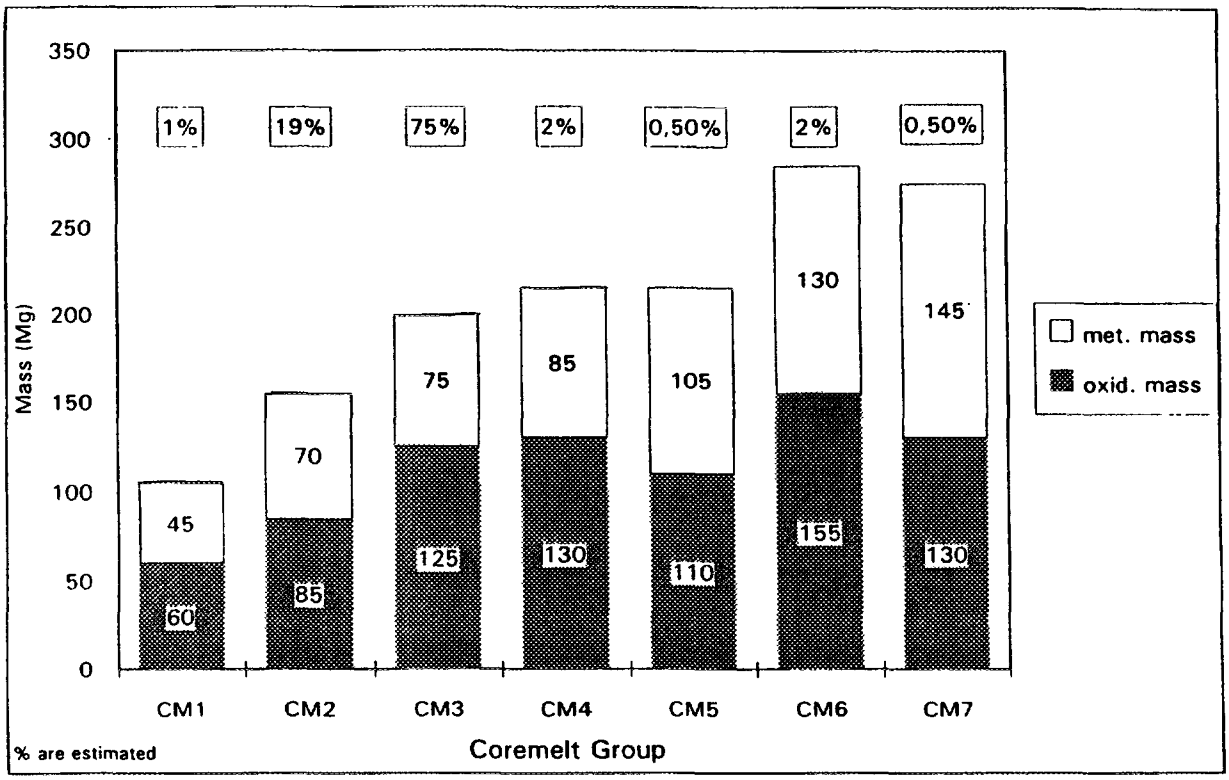

FIG. 6. Melt propagation scenarios.
Core-melt groups immediately after RPV failure.

core and deliberately collect the melt before spreading in order to cope also with scenarios where core degradation lasts over an extended period of time, with greater parts of the core coming down in a later phase, and to increase the spreading ability of the melt by increasing its temperature. No active measure is needed for this process.

Spreading of the Melt (Transfer, Distribution)

The reactor pit has one opening for melt discharge to the spreading area. The melt is guided by a melt discharge channel which has a slope to the spreading area and is cladded with protective material. The spreading area of about 150 m^2 has been chosen to ensure complete spreading on the one hand and sufficient coolability on the other hand. An example for a spreading calculation with the code CORFLOW is shown in fig. 7. One can see very clearly the quick propagation of the melt and the sloshing when hitting the opposite wall.

Small scale experiments (1 x 1 m) using 30 to 50 kg thermite of 2200°C have been performed during Winter 92/Spring 93 (fig. 8). Test series with different surface conditions (concrete, ceramic protection layer) and different amount of water present (dry to 40 cm) have shown that even under water a sufficient spreading of melt is achieved, whereas in some cases an energetic melt water interaction occurred (water trapped below the melt, e.g. initially wet concrete). For the EPR the spreading compartment is dry during normal operation, eventually covered by a very shallow layer of condensate in case of LOCA as depicted in the previous chapter.

First spreading tests with 150 kg thermite (80 kg metallic Fe, 70 kg oxidic Al_2O_3) performed in the frame of the KATS test series at KfK have shown the high spreading ability of the metallic constituent, whereas the oxidic component

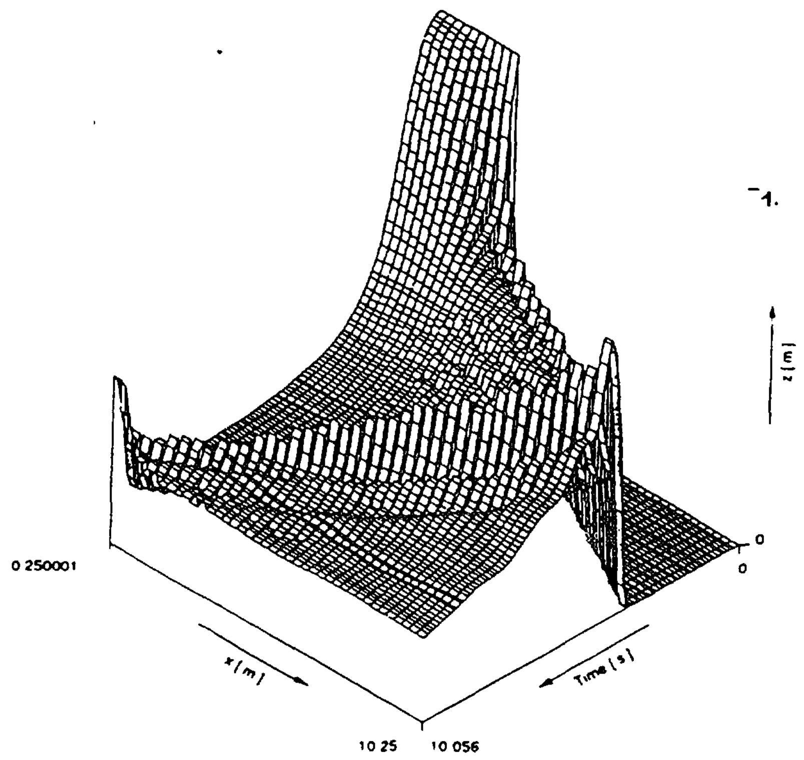

FIG. 7. Simulation of melt-spreading (Oxidic melt, Isothermal boundary conditions).

spread out only to a 5 to 10 cm thick layer - due to a too low temperature of the thermite. Further experiments with more adequate melt temperatures performed on dry and wet surfaces will follow.

Confining of the Melt

The basemat concrete is protected by an arrangement of different layers, e.g. a high-temperature resistant protection layer of Zirconia bricks and an insulating layer of refractory concrete thus avoiding melt concrete interaction (fig. 9). The spreading compartment acts therefore as confining boundary for melt retention. In addition to the initial thermal loads when the melt is flowing over its surface the long term thermal-chemical stability has to be looked at.

Experiments dealing with the thermal shock stability and first interaction tests between real corium and Zirconia bricks, performed for Siemens at St. Petersburg, lead to the conclusion, that Zirconia bricks are a possible means to fulfill the confinement function expected.

Cooling of the Melt

The spreaded melt is flooded with water by passive means via connection pipes to the water of the IRWST. These pipes are closed during normal operation and accident conditions by plugs, which will be melted by the hot melt itself (see as an example fig. 10). The flooding rate is in the order of approximately 50 kg/s, to ensure on the one hand a moderate flooding time and to limit on the other hand the amount of water for a possible energetic melt water interaction.

Since there is up to now not enough experimental evidence in respect to a (partial) quenching of the melt, it is assumed for lay-out considerations that the heat transfer from the melt to its surrounding is solely governed by heat conduction. This is in respect to the thermal loadings of the protection layer and the structural concrete below a penalizing assumption.

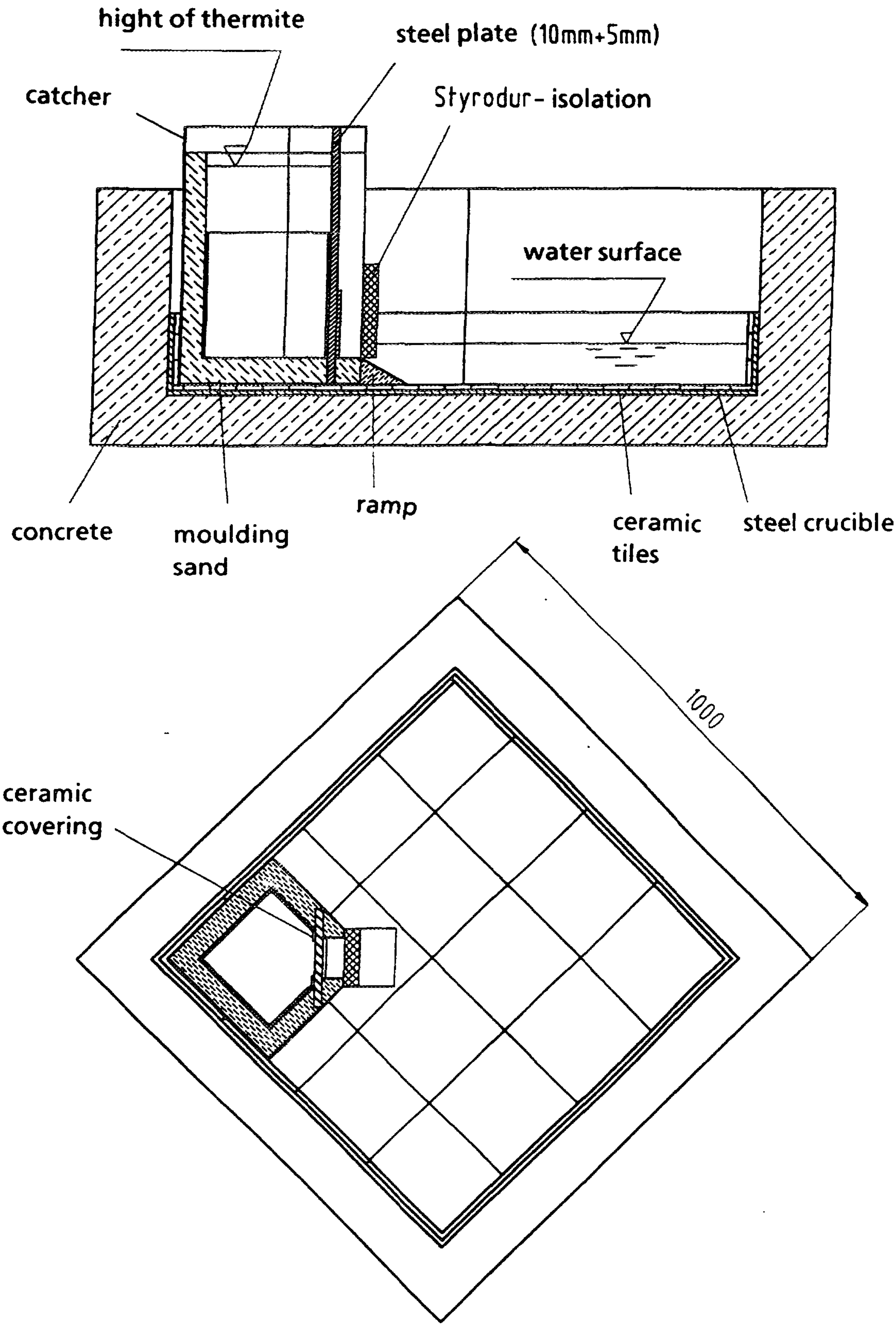

FIG. 8. Test facility 3-2.

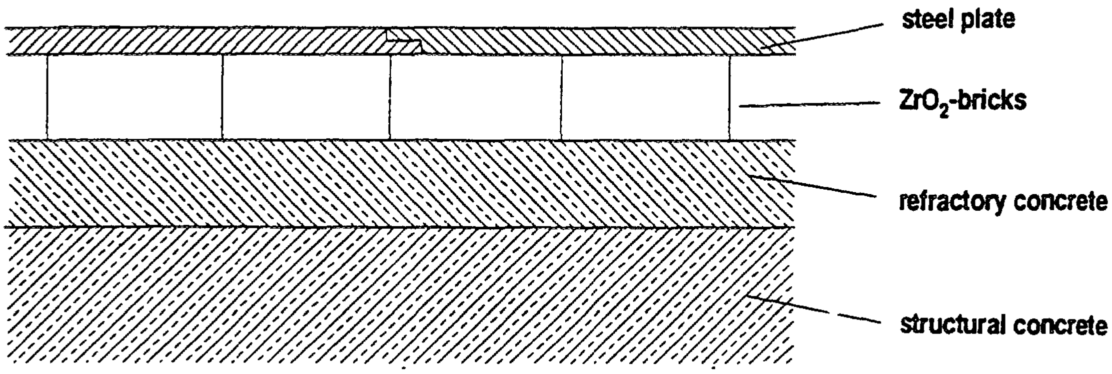

FIG. 9. Layer.

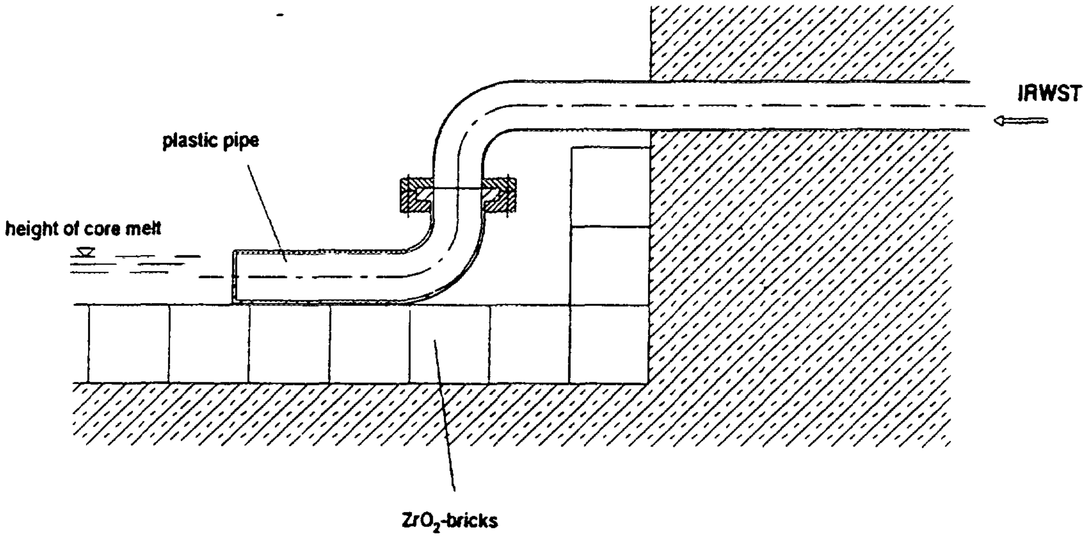

FIG. 10. Spreading area opening device for interconnecting pipe to IRWST plastic pipe.

The stored heat in the melt and the decay heat are transferred to the water lying above. The steam generated thereby escapes via the openings in the spreading compartment into the containment and will there be condensed on the walls and after the specific grace time for initiating the dedicated containment heat removal system (CHRS) - in the case of the EPR a spray system - in addition on the cold water of the CHRS thus decreasing the containment pressure again. The condensate and the spray water are flowing back to the IRWST and from there to the spreading compartment, closing the cycle.

Stabilization in the long term

The heat removal from the melt is established as depicted before. In the long term a recirculation mode of the CHRS can be chosen which leads to a subcooling of the water on the melt thus limiting very strongly further fission product escape.

The melt will be solidified in less than two days (depending very strongly on the solidification temperature assumed: 2 d for T = 1900 °C, 5 hours for T = 2200 °C; note that pure heat conduction is assumed). In any case a strong crust will be established within a few hours. This time reflects the necessary "survival" time for the protection layer. In the long term it is not needed any more.

The downwards directed heat flux from the melt leads to an increase in temperature of the basemat. This is a very slow process which will finally be reversed due to the decrease of the decay heat. A long term temperature profile in the basemat can be seen from fig. 11.

4 Experimental and analytical support

In the last years the discussions about the phenomena and potential threats connected with severe accidents had gained importance and worldwide experimental and analytical efforts are underway.

To get more certainty in respect to the feasibility of an envisaged concept one has to identify the key issues connected with the problem confronted. For the large spreading concept of the melt these key questions are:

- what is the time-dependent evolution of the accident in respect to composition, mass and temperature of the melt as initial condition for the spreading itself, taking into account the deliberate delay time needed to melt the gate between reactor pit and spreading area (scenario analysis)

- how will during the spreading process the metallic and oxidic constituents of the melt be distributed (the completeness of the spreading process is a minor issue).

- what is the thermochemical stability of the protection layer in the first hours of contact with the melt (in the long term the melt is solidified)

- is there any quenching of the melt when flooded
  (note: for the lay-out considerations this has not been taken into account, but would help very much in reducing long term thermal loads)

- what is the energetic interaction between melt and water during the flooding process

- what is the fission product release during the different stages of accident progression: this is a question for f. p. release in the short term, but especially a question of preventing a long term source of fission products

This list is not complete but addresses main items.

To solve the most stringent problems connected with this EPR-concept concerted actions have been started in the frame of cooperations between the French and German partners:

- CEA and KfK cooperating closely together
- CEA with the French industry
- KfK with the German industry
- and all together in regularly information exchanges and working groups to be established.

In addition to that BMFT-sponsored activities in the frame of the AGIK-group and the FARO work program of the JRC Ispra will help to get more insight in the processes involved.

Main experimental programs are:

- VULCANO (CEA, where already a lot is done in the frame of the CORINNE experiments)

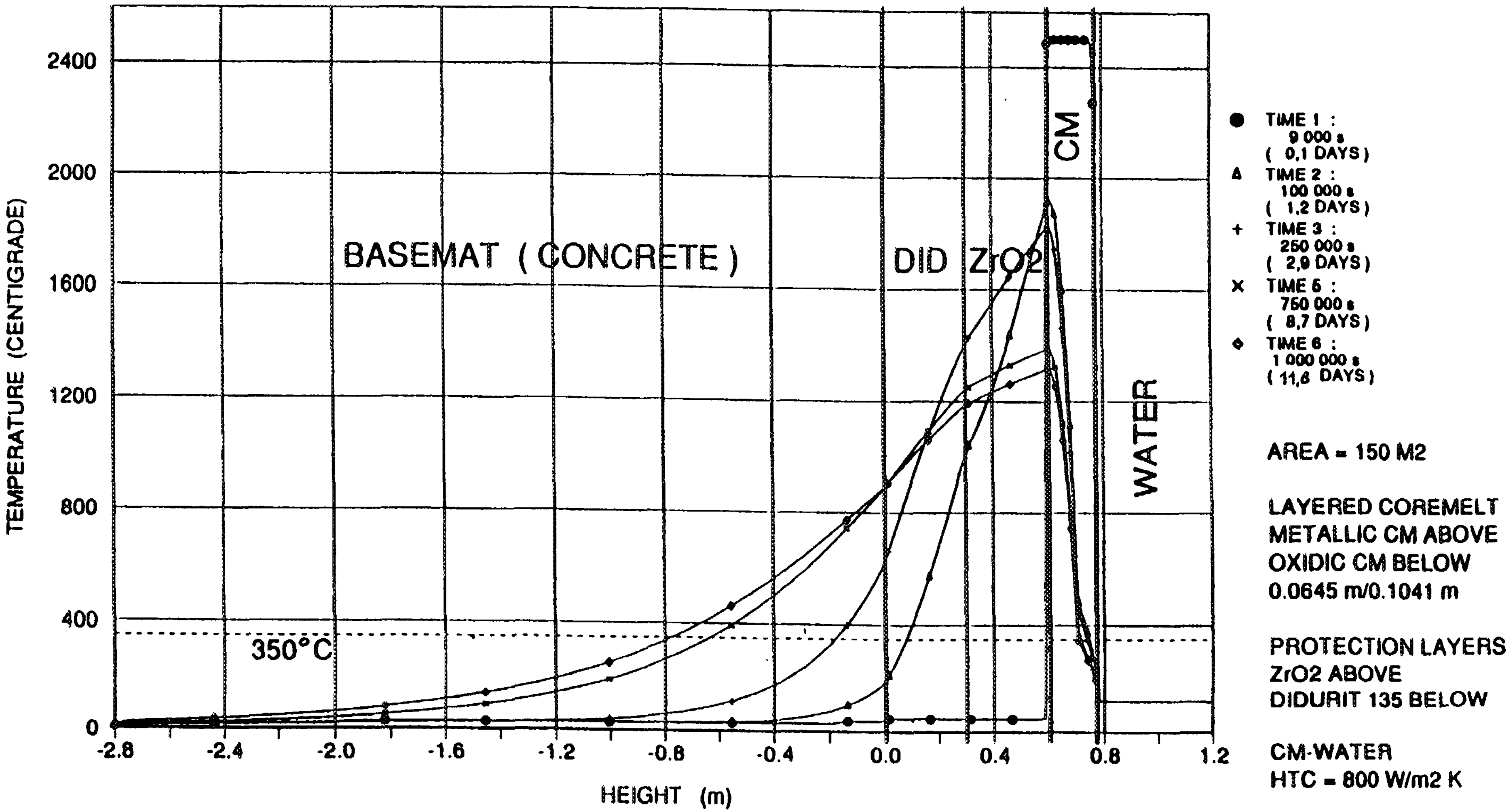

FIG. 11. Temperature profiles - Base case.

- spreading tests with real corium (1 t), delayed melt spreading and accumulation, water addition

- KATS (KfK, with parallel investigations to other core-catcher concepts - COMET)
  - spreading tests with thermite on dry and wet surfaces

- COMAS (AGIK)
  - spreading tests with real corium (3 t) on different surfaces

- FARO (JRC Ispra)
  - melt water interaction tests

- CIRMAT (SIEMENS)
  - thermal-chemical interaction tests corium with protective material

5 Conclusions

For the EPR core melt retention/stabilization is proposed to be achieved by a large spreading of melt on a dedicated spreading area of about 150 m^2, with high temperature resistant protection layers and flooding of the melt with water from the IRWST after spreading. The different steps involved in the course of the accident till finally the melt is confined, cooled and stabilized are achieved solely by passive means. Decay heat removal is established via a closed thermohydraulic circuit, where as ultimate heat sink the dedicated Containment Heat Removal System (spray system) comes in action with a sufficient delay time, governed by the need to limit the containment pressure, until system operation is required.

To show the feasibility of the melt retention concept cooperations have been established with an ambitious work program, including tests with real corium.

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FEASIBILITY OF PASSIVE HEAT REMOVAL SYSTEMS

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Abstract

This paper presents a review of decay heat removal systems (DHRSs) used in liquid metal-cooled fast reactors (LMFRs). Advantages and the disadvantages of these DHRSs, extent of their passivity and prospects for their use in advanced fast reactor projects are analyzed. Methods of extending the limitations on the employment of individual systems, allowing enhancement in their effectiveness as safety systems and assuring their total passivity are described.

1. INTRODUCTION

Decay heat removal after a reactor shutdown is one of the most important safety functions. The degree of reliability of fulfillment of this function affects decisively the safety level of a nuclear power plant (NPP) as a whole. One of the main points in the concept of designing new-generation advanced nuclear power plants of any type is to aim at a maximum use of the inherent safety properties characteristic of this reactor type. Therefore, design workers tend, to an ever-increasing degree, to use as safety systems those based on passive principles. As applied to decay heat removal systems it means using natural convection of coolant as a motive force.

This paper deals mainly with the problem of technical feasibility of passive decay heat removal systems for fast reactors.

It should be noted that in the field of fast reactors, considerable experience has been already gained of using natural-convection coolant systems for decay heat removal. Not all of these systems can be formally classified as fully passive ones in compliance with the terms and recommendations adopted in [1] because many of them are based on a combination of passive and active principles for starting up the system and for heat removal. However, available experience of the operation of these systems and experimental studies can be of value for designing fully passive DHRSs for advanced fast reactors. And this experience can be useful for other types of advanced reactors as well.

2. GENERAL ASPECTS OF THE TECHNICAL FEASIBILITY OF PASSIVE DHRSs

Let us try to formulate general criteria for what should be understood by technical feasibility of passive DHRSs?

In the technical feasibility problem of passive decay heat removal systems there can be noted the following general points:

A. To show the basic possibility for realization of a particular passive heat removal method. The problem is divided into two tasks: assurance of passive start-up of the system and of its subsequent passive operation.

B. The limits of applicability of a particular passive heat removal method should be shown both as to the applicable reactor types and for their power range. Note, that a reactor type determines the possible range of main reactor plant parameter variation during operation of the system under consideration.

C. A spectrum of accidental events in which the use of the considered heat removal method is possible should be defined. This aspect also is dependent on reactor type.

D. The stability of passive heat removal system characteristics should be validated relative to the possible effect of external and internal factors, including relative to the initial state of the reactor and system, and relative to single failures and common mode failures as well.

3. MAIN TYPES OF DHRSs USED IN LMFRs

In most accidental events in fast reactors a system of normal heat removal to the third circuit through steam generators is used for decay heat removal. Usually these systems are not safety-grade systems.

As a rule, in LMFRs, for decay heat removal in the most severe accidents, special systems are provided based on heat removal to air as a final heat sink. Heat removal in them is accomplished through forced or natural convection of coolant and air, or by a combination of driving forces. These systems can be classified in different ways:

- by the degree of their dependence on power supply sources, i.e. on the principle of their operation, either passive or active;
- by the location of these systems in the nuclear power plant layout;
- by the method of heat transfer to air: the use of sodium-air heat exchangers (AHXs) based on convection or heat removal through the reactor vessel - a combined use of convective and radiative means for heat removal, etc,.

At present the most widespread method of classification of these systems is by their position in the nuclear plant layout (Fig.1).

So there can be decay heat removal systems which remove heat directly from the reactor vessel - direct reactor auxiliary cooling systems (DRACS). Such systems are used in pool-type reactors (SPX-1 in France, the European Fast Reactor Project (EFR) (Fig.2), the BN-1600 project in Russia, etc.), or in so-called top-entry loop-type fast reactors (the DFBR project in Japan). In DRACS specially provided loops with immersed sodium-sodium heat exchangers are used for heat removal from the reactor vessel. These systems are fully independent of the normal heat removal systems.

In contrast to DRACS, systems also removing decay heat from the primary circuit - so-called primary reactor auxiliary cooling systems (PRACS) - use a heat exchanger in the IHX and so in the normal flow path for primary sodium. Heat is then removed to air through special loops, as in DRACS (PFR in Great Britain).

For loop-type LMFRs (Monju in Japan) and in some pool-type reactors (SPX- I in France, the BN-800 project in Russia (Fig.3)) there are decay heat removal systems connected to the secondary circuit - intermediate reactor auxiliary cooling systems (IRACS). This version of DHRS is characterized by integration some equipment of the safety system with the normal operational system functions.

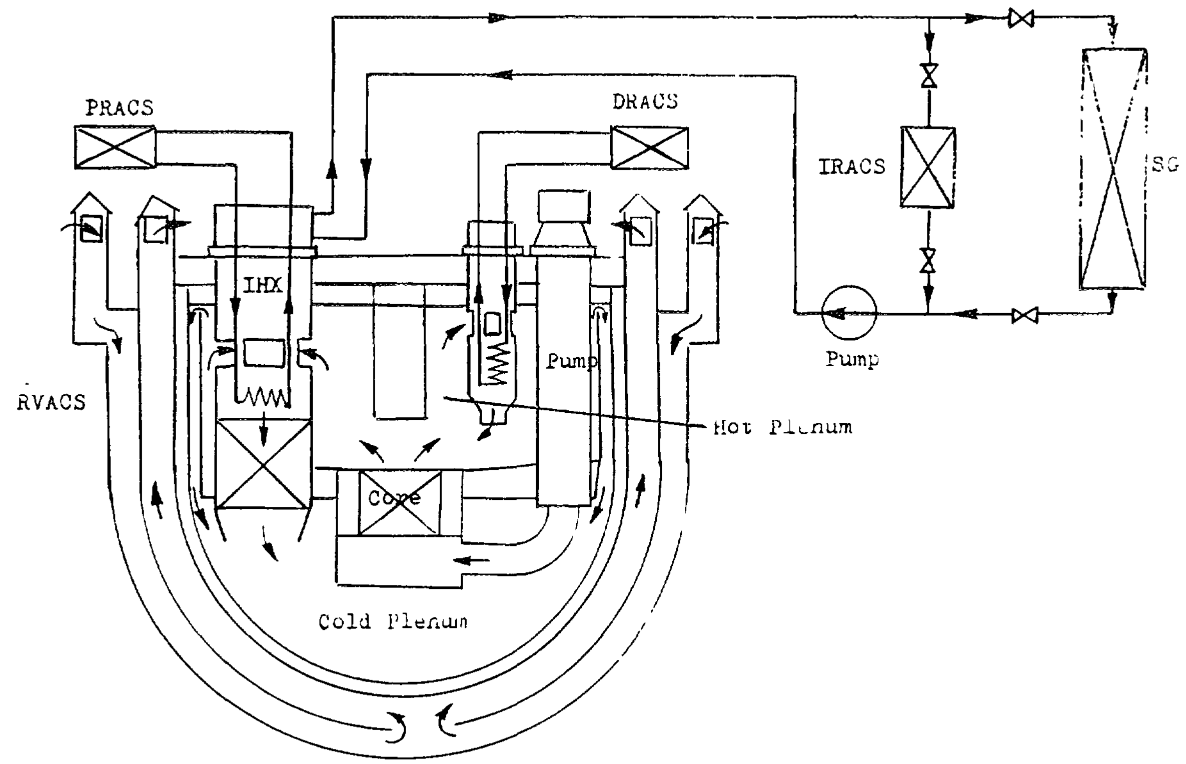


Fig. 1. Principle scheme of different types of DHRS.

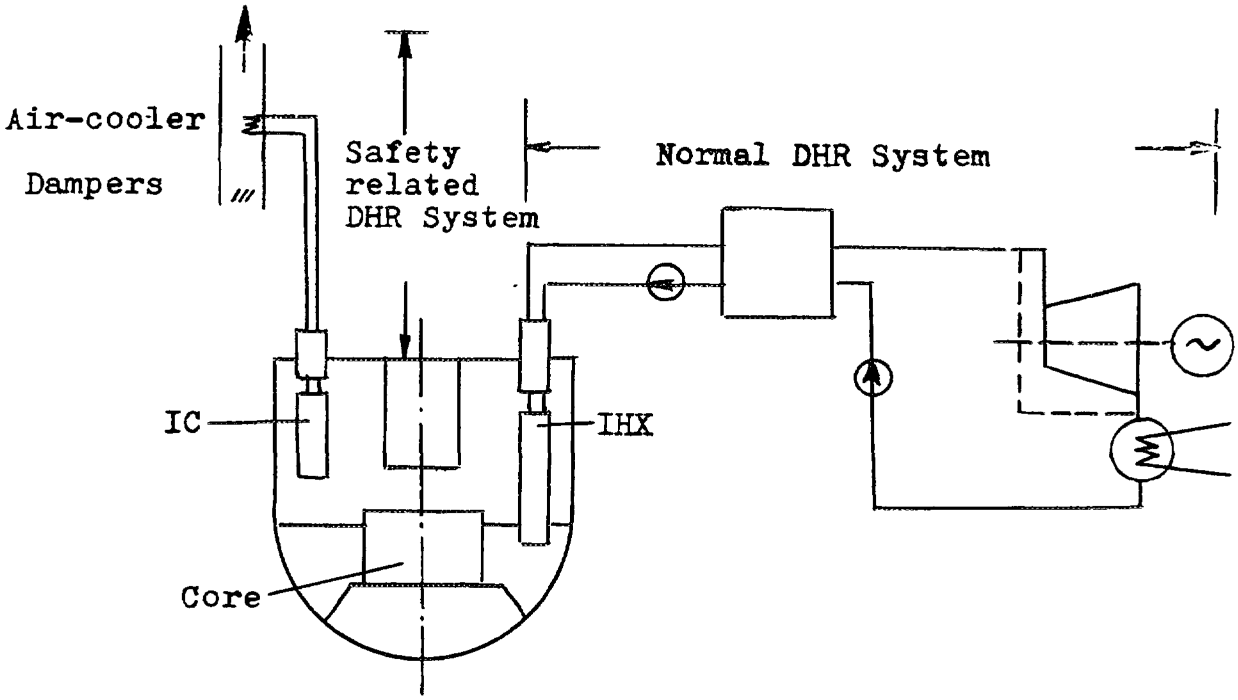


Fig. 2. DHR-Normal/Safety Related systems (EFR).

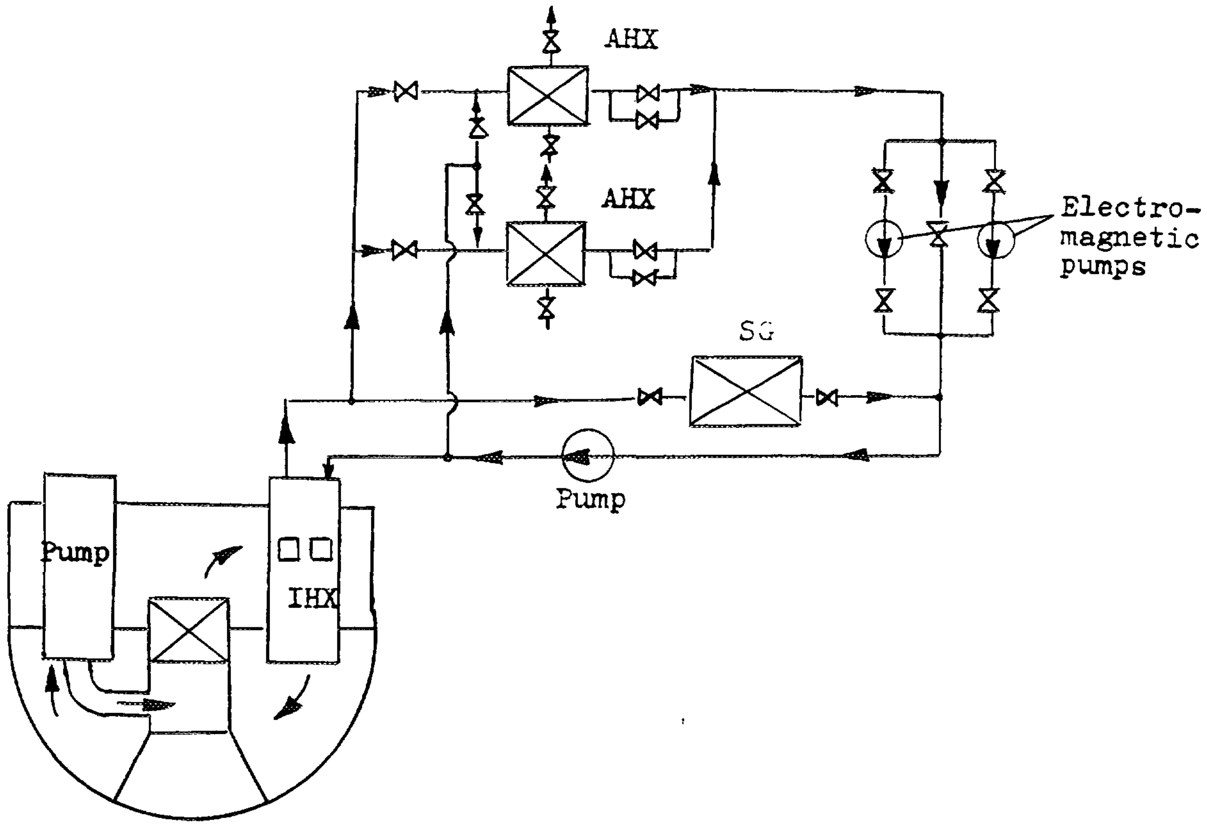


Fig. 3. DHR Safety Related System (BN-800).

All the above DHRSs use sodium-air heat exchangers for heat transfer to air. In some reactors, the systems for air cooling, of the external surfaces of steam generators (SGs) are used as IRACS (Phenix in France, BN-350 reactor in Kazakhstan, FBTR in India).

Within the framework of the American modular liquid metal-cooled fast reactor PRISM project, a qualitatively different way of heat removal from the reactor was proposed through the reactor and guard vessels to air passing in the gap between the guard vessel and the reactor silo lining or supply header (Fig.4).

It should be noted that systems for heat removal through the reactor vessel exist in the Phenix and SPX- I reactors. In these, heat is removed not to air but to the water cooling system placed within the reactor silo concrete. The water cooling system is an active one. As it is difficult in practice to make these systems passive, we shall not analyze them below.

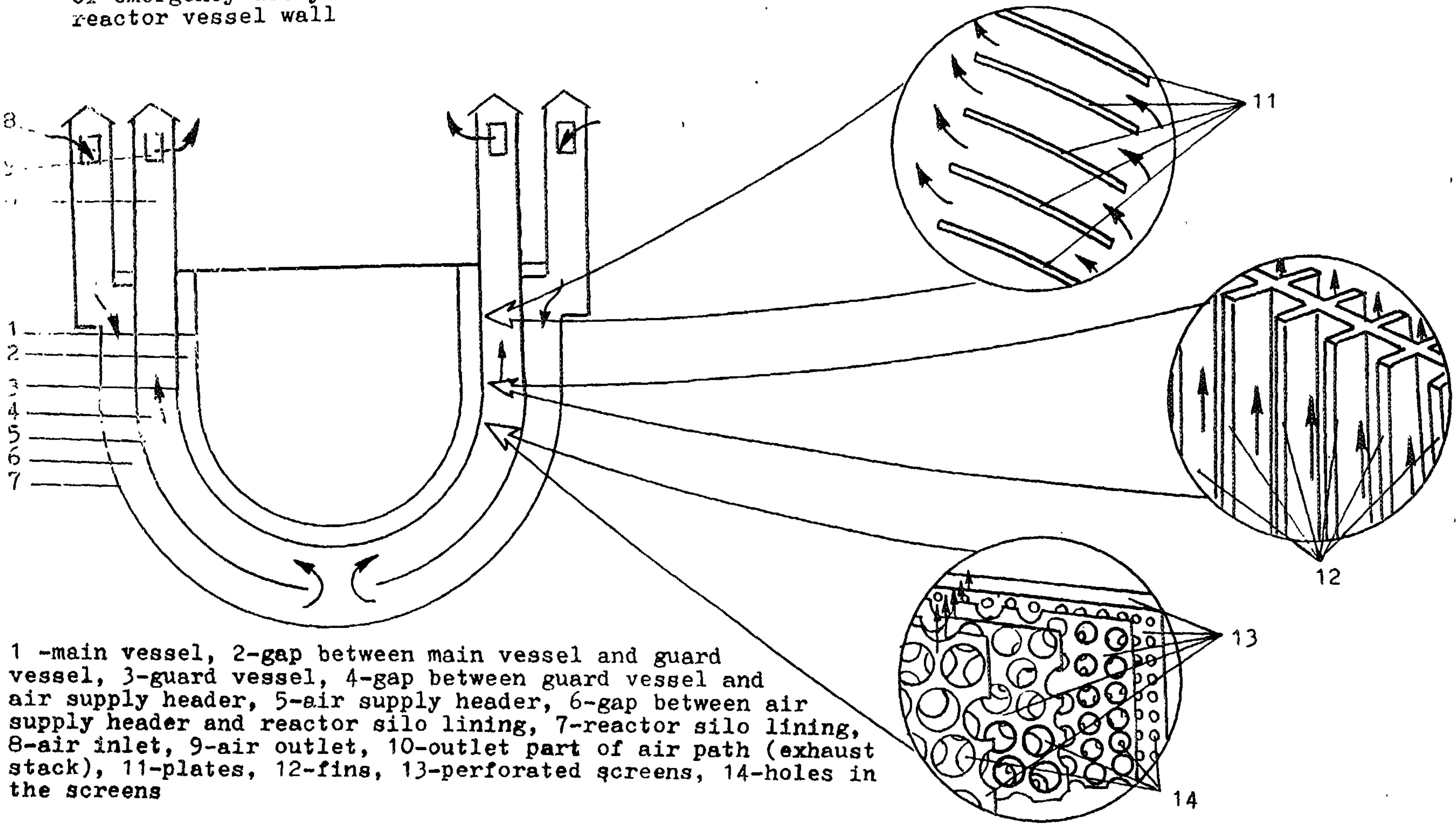
However, from the viewpoint of safety requirements, the classification of the systems under consideration by their degree of passivity is an important approach.

Thus the system for heat removal through the reactor vessel - reactor vessel auxiliary cooling system (RVACS) - is a fully passive one and is classified according to the degree of passivity of classification adopted in [1] as category B.

In many DHRSs with AHX, forced circulation of coolant in some circuits is provided for normal operation and the natural convection conditions provide stand-by cooling for the case of forced circulation failure. However, such systems were always designed on the basis

Principle scheme of traditional system of emergency decay heat removal through reactor vessel wall

Proposals to improve the traditional system



1 -main vessel, 2-gap between main vessel and guard vessel, 3-guard vessel, 4-gap between guard vessel and air supply header, 5-air supply header, 6-gap between air supply header and reactor silo lining, 7-reactor silo lining, 8-air inlet, 9-air outlet, 10-outlet part of air path (exhaust stack), 11-plates, 12-fins, 13-perforated screens, 14-holes in the screens

← Arrows indicate air flow direction

FIG. 4.

that the DHRS should also ensure decay heat removal with the required efficiency under natural convection conditions. Nevertheless DHRSs with AHX cannot be classified as wholly passive systems, because some active components (air dampers and, perhaps, sodium valves) are used in their start-up. At best the degree of passivity of these systems can be brought to category D, being intermediate between the passive and active systems.

The degree of passivity of DHRSs with AHX can be enhanced up to category B by designing for a permanent heat loss through the DHRS at a level which would ensure reactor plant safety even in the event of total failure of all active heat removal systems. Such a solution has been adopted in the EFR project. On three of six DRACS loops the dampers are in a fixed open position during normal power operation ensuring a permanent heat loss of about 30 MW.

It should be noted that natural convection cooling is also used in fast reactors for ensuring core debris confinement, within the reactor vessel. A so-called core catcher at the bottom of the reactor vessel specially provided for this purpose (SPX-1, the BN-800 project) is designed so as to assure efficient decay heat removal from the destroyed core using natural convection of the coolant.

4. PROSPECTIVE DHRSs FOR ADVANCED LMFRs

Let us analyze some advantages and disadvantages of various DHRS designs revealed as a result of experimental and calculational studies aimed at the determination of the prospects of their use in advanced LMFR projects. An analysis of the maximum attainable degree of passivity for a particular DHRS design and of its applicability limits is presented.

1) RVACS

This system has been designed for advanced PRISM project and is a fully passive one (category B). The system is extremely simple and considerably reduces the number of elements which must meet safety requirements [2,3].

Its main disadvantage is its restricted applicability. At present this system has been adopted and investigated as applied to low-power reactors. The determining parameter for RVACS is not the absolute value of reactor power but its ratio to the heat capacity of the reactor vessel (or to the reactor vessel volume) and to the reactor vessel-to-air flow heat - transfer surface.

The calculational studies conducted confirm this. Reactors such as BN-1600 and SPX-1 prove to be more suitable for using these systems than, e.g., the BN-800 reactor. For the BN-1600 reactor during RVACS operation, the maximum level of average sodium temperature in the reactor vessel will not exceed ~ 730-820°C, whereas for the BN-800 reactor it will be ~ 770-870°C, all other things being equal.

Studies on RVACS efficiency enhancement are currently under way. The following are possible:

- an increase of the exhaust stack height;
- the use of special radiation collectors with modified heat transfer surface located in the gap between the guard vessel and reactor silo lining (Fig.4);

- optimization of the gap width between the guard vessel and reactor silo lining;
- abandoning of the guard vessel and passing its functions to the reactor silo lining (such abandonment was studied for the French SPX-2 reactor project);
- filling the gap between the reactor main and guard vessels with sodium.

Methods of surface modification for the heat radiation collector have been proposed as follows:

- the use of porous material [4];
- the use of longitudinal fins [5];
- the use of plates placed at an angle to the guard vessel surface [6];
- the use of a pack of semipermeable screens which can be made in the form of grids, perforated screens or semipermeable films [6].

In Figs.5-9 the calculation results of the use of the RVACS applied to the BN-800 reactor are presented. In Fig.5 the results of optimization of the width of the gap between the guard vessel and the reactor silo lining for conventional RVACS (as in PRISM) are presented.

In Figs.6-7 the results of optimization of the gap width and the number of screens for an advanced RVACS with semi-permeable screens are shown. Fig.8 illustrates the dependence of the maximum level of the average mixed coolant temperature in the BN-800 reactor vessel on the initial power level. In Fig.9 the results of gap width optimization for a version without the guard vessel are presented.

In these figures the following symbols are used:

H - height of air path exhaust stack,
ϵ - emissivity of reactor vessels and reactor silo lining materials,
N - the number of semipermeable screens in the heat radiation collector,
INA=O - a system operation mode when the gap between the reactor vessel and the guard vessel contains no sodium.
INA= I - a system operation mode when the gap between the reactor vessel and the guard vessel is filled with sodium.

These figures illustrate possibilities of improving RVACS characteristics and of extending their applicability limits. Investigations have shown that RVACS can also be used for larger power reactors but the relationship between power, vessel size and sodium temperature under RVACS operation leads to a limitation or reactor power.

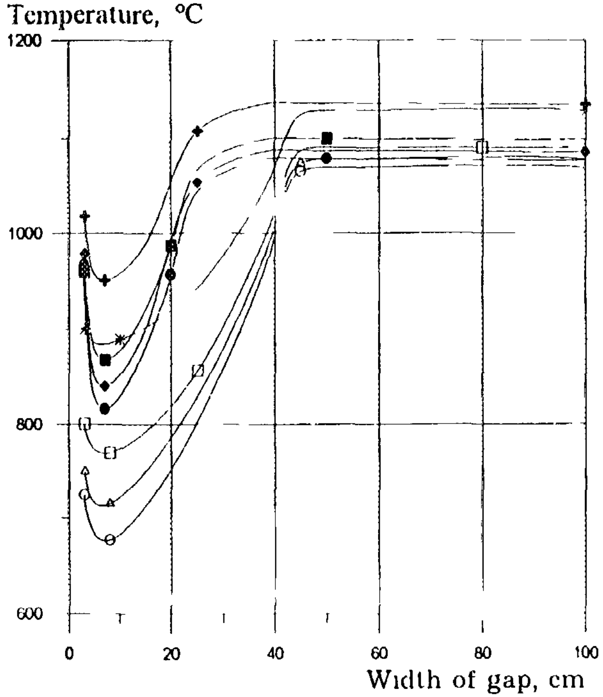
In connection with the above, there arises a problem of experimental, and calculational study of RVACS for larger power reactors, in particular, the problem of an investigation of natural coolant convection in large vessels during RVACS operation.

2) IRACS

Air heat exchangers have considerable hydraulic resistance so that there is only one practical possibility for their attachment to the secondary circuit, i.e. on a by-pass to the main normal heat removal line pipework. This can be a by-pass of the main pipe section with a check valve on it (SPX-1), or a by-pass relative to the steam generator (BN-800). In any case, their will be active components such as sodium valves and air dampers within the DHRS.

Maximum coolant temperature in the reactor vessel as function of width of the gap between guard vessel and reactor silo lining

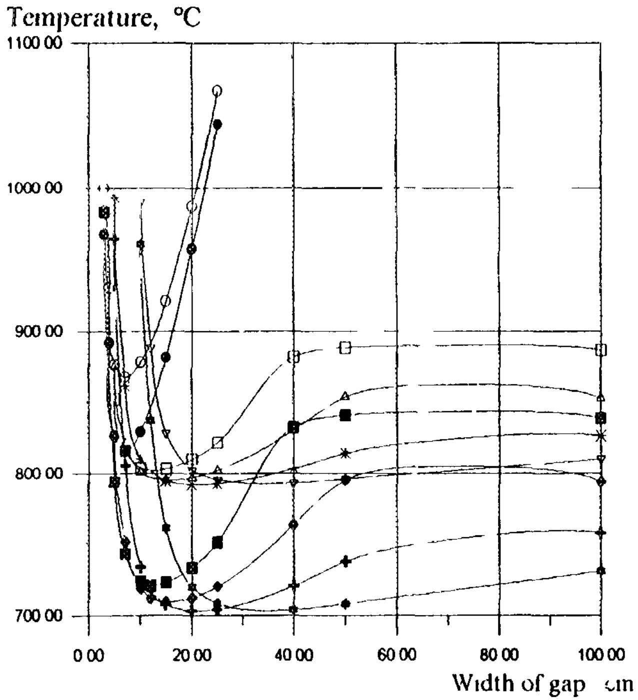
Traditional RVACS



○ - H = 60 m , INA = 1, ε = 0.85
□ - H = 60 m , INA = 0, ε = 0.85
△ - H = 60 m , INA = 1, ε = 0.5
• - H = 60 m, INA = 0, ε = 0.5
● - H = 8 m , INA = 1, ε = 0.85
■ - H = 8 m , INA = 0, ε = 0.85
◆ - H = 8 m , INA = 1, ε = 0.5
✚ - H = 8 m, INA = 0, ε = 0.5

FIG. 5.

Improved RVACS with screens



H = 8 m, ε = 0.85,

○ - N = 0 , INA = 0
□ - N = 1 , INA = 0
△ - N = 2 , INA = 0
• - N = 5 , INA = 0
▽ - N = 20 , INA = 0
● - N = 0 , INA = 1
■ - N = 1 , INA = 1
◆ - N = 2 , INA = 1
✚ - N = 5, INA = 1
✱ - N = 20, INA = 1

FIG. 6.

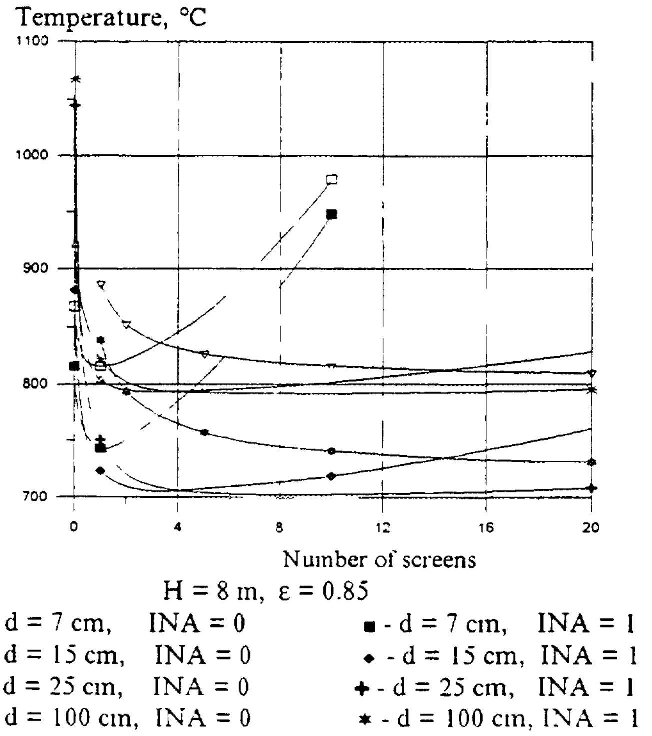


FIG. 7. Maximum coolant temperature in the reactor vessel as function of number of perforated screens (improved RVACS with screens)

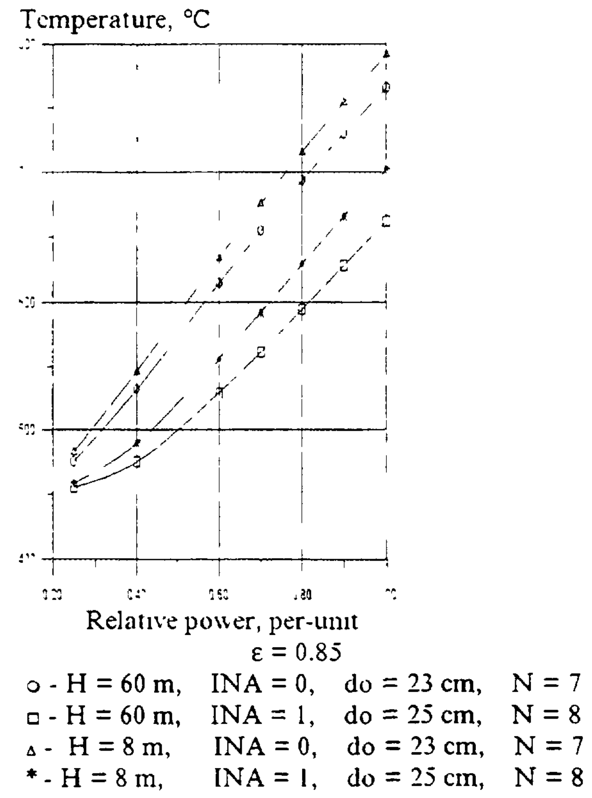
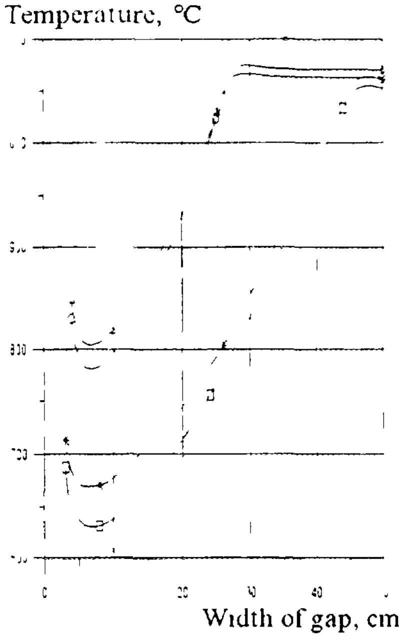


FIG. 8. Maximum coolant temperature in the reactor vessel as function of reactor power (improved RVACS with screens)



ɔ - H = 8 m , ε = 0.85
□ - H = 60 m , ε = 0.85
Δ - H = 8 m , ε = 0.5
* - H = 60 m , ε = 0.5

FIG. 9. Maximum coolant temperature in the reactor vessel as function of width of the gap between vessel and reactor silo lining (RVACS without guard vessel)

The use of air cooling system on the outside of a steam generator as an IRACS has the following advantages:

- such active components as sodium valves are eliminated;
- the coolant circulation circuit has a more simple arrangement, there are no by-passes.

It is seen that even in the case of natural circulation in all DHRS circuits there is a dependence on the serviceability of active components. In addition, the stability of circulation depends on the initial temperature condition of the DHRS and of the reactor plant as a whole, on the transient conditions of such active components as the main primary and secondary pumps and on the procedure for putting the DHRS into operation. Making such systems fully passive is possible by means of partial opening of sodium valves and air dampers during NPP normal operation. However, it results in deterioration of NPP economic factors. In addition, uncertainty in the conditions of passing from forced to natural circulation, which are accompanied by main primary and secondary pumps coast-down, cannot be eliminated in principle. Therefore, it cannot be eliminated in principle a chance of occurrence of conditions impeding the esstablishment of natural convection of coolant in the DHRS and even its reversability.

In addition IRACS have the following disadvantages. Unclosed sodium valves on the main pipe line lead to marked deterioration of decay heat removal efficiency. In IRACS a great number of components must meet the enhanced requirements put on safety grade systems.

3) DRACS

DRACS removes heat directly from the reactor vessel with the use of special loops independent of the main heat removal loops. Such independence allows elimination of sodium valves performing a switch-over of heat removal from the main equipment to the emergency circuit. Air dampers are the only active components. However, realization of a fully passive DHRS is possible in principle at the expense of the provision of a continuous air leak through AHXs as has been done in the EFR. In this case of the NPP economic factors have to be optimized and some losses in economy must be borne.

Important problems that exist for such systems are the problems of experimental investigation of such items as coolant flow pattern in the reactor vessel under natural convection, coolant stratification, interaction between the upper plenum and the core, and optimization of the location of immersed heat exchangers. A large number of investigations in this field have already been carried out in Germany [7,8] and in Japan [9]. This work should be continued.

In Russia calculation studies on optimization of immersed heat exchanging have been carried out [10]. It has been shown that at their location in the hot plenum some difficulties with the development of natural convection in the intermediate loops are possible. At present the possibility of locating the immersed heat exchangers in the cold plenum is being analyzed.

In Fig.10 a version is proposed where heat from the reactor vessel is transported by coolant to an AHX directly connected on one side to the reactor vessel by two pipes and on the other side to the expansion tank. During NPP normal operation the loop with the AHX is filled with argon and there is no heat loss through the AHX. In an accidental event accompanied by a failure of all heat removal systems, heating-up of coolant in the reactor vessel results in filling the loop incorporating the AHX due to a pressure rise in the reactor gas space. Coolant natural convection through the AHX takes place and decay heat is removed to the outside. Thus the system is fully passive both in its start-up and operation. The possibility of a system failure in the case of a reactor vessel loss of gas tightness can be overcome by various means, including by supplementing this system with a similar one but located lower down and filled only due to a rise of the coolant level in the reactor vessel.

One of the advantages of DRACS is relatively small number of components included in the safely-grade systems. As compared to RVACS they can remove more power upto a level which is virtually unlimited.

4) PRACS

PRACS have basically the same solution as DRACS to the heat removal problem and, accordingly, have the same advantages and disadvantages.

5. CONCLUSIONS

An analysis of the present DHRSs and of the possibilities improving their characteristics including their degree of passivity reveals the following.

The most preferred systems to be used in advanced LMFR projects are RVACS and DRACS (or PRACS). These systems permit full realization of passive principles in their start-up and operation.

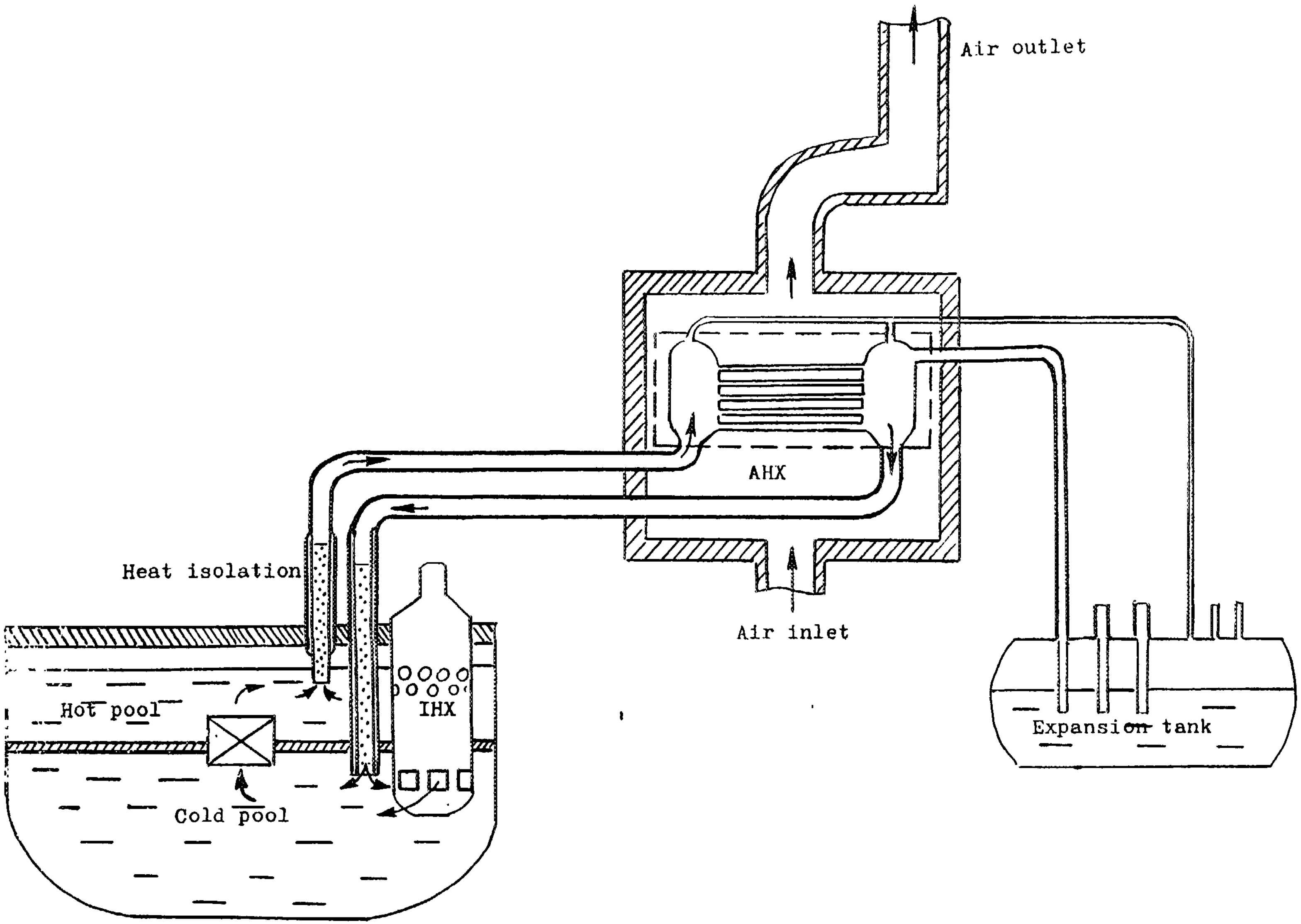


Fig. 10. Principle scheme of DRACS with passive switch-on.

They are sufficiently simple and independent of normal heat removal systems. They largely eliminate, the effect of transient conditions of normal heat removal active components upon the stability and efficiency of the operation of the DHRS.

There are feasible ways to enhance RVACS efficiency and to extend the limits of their applicability from the viewpoint of reactor power range. However, a rather high temperature level in the reactor vessel during their operation makes attractive the idea of combined use of RVACS and DRACS in LMFR projects. In this case DRACS would be used as a main heat removal system and RVACS would be used as an ultimate decay heat removal system if DRACS fails.

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AVAILABILITY ANALYSIS OF THE AP600 PASSIVE CORE COOLING SYSTEM

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Abstract

AVAILABILITY ANALYSIS OF THE AP600 PASSIVE CORE COOLING SYSTEM. The reliability analysis of the AP600 Passive Core Cooling System (PXS) has been done. The fault tree analysis method was used for the quantitative analysis. The PXS can be grouped to several sub-systems i.e: Reactor Coolant System (RCS) Injection Subsystem, Emergency Core Decay Heat Removal Subsystem, and Containment Sump pH Control Subsystem. The quantitative analysis results indicates that the system unavailability is highly dependent on the valves configuration of the Automatic Depressurization System (ADS). If the ADS valves is arranged in Option-1, the system unavailability is 2.347E-03, this means that the yearly contribution to plant down time can be estimated to be about 20.56 hours per year. Whereas, by using Option-2 of fourth stage ADS valves, the system unavailability is reduced to be 9.877E-04 or 8.65 hours per year and this value is consistent with the allocated goal value (8.0 hours per year). The ADS contributes 66.89% to the system unavailability if it is arranged in Option-1, and will reduced to be about 21.21% if its fourth stages are arranged in Option-2. If the ADS is not included as a subsystem of the PXS (relocate to RCS as a subsystem of RCS), then the PXS unavailability will reduced to be about 7.784E-04 or 6.82 hours per year and this is less then the allocated goal value. The major contributors to the system unavailability are mostly dominated by Stage-4 ADS valves (air piston operated valves and squib valves), inservice testing valves of ADS (solenoid operated valves), solenoid valves of Nitrogen Supply to Accumulator, and Passive Residual Heat Removal actuation valves (air operated valves). Therefore, it is recommended that those valves be analyzed more detail to gain the improvement in its reliability. It is also recommended that the fourth stage of ADS valves should be arranged according to Option-2, i.e. one 10-inch normally open motor operated gate valve in series with one 10-inch normally closed squib valve.

1. INTRODUCTION

The AP600 is a 600 MWe, two loop, advanced passive plant, developed by the Westinghouse Electric Corporation in cooperation with the U.S. Department of Energy (DOE) and Electric Power Research Institute (EPRI). Indonesia is one of the international participants in this program, and this paper is a part of that participation.

The AP600 Passive Core Cooling System has been allocated to contribute 8.0 hours per year to the total estimated plant yearly downtime. This paper presents an availability analysis based on current System Specification Document, by using the fault tree analysis method. The failure rate and unavailability data (data sources) for various components used for the fault tree quantification was mostly from: Nuclear Plant Reliability Data System (NPRDS) and Westinghouse Reliability Data Base.).

2. GENERAL SYSTEM DESCRIPTION

A detailed description of the Passive Core Cooling System (PXS) is provided in the System Specification Document, Reference 1. The PXS is a safety-related system designed to provide adequate core cooling for design basis events, this system consists of two Passive Residual Heat Removal systems (PRHR) Heat Exchangers, two Accumulators, two Core Makeup Tanks (CMTs), one In-containment Refueling Water Storage Tank (IRWST), two RCS depressurization spargers, associated valves, piping and instrumentation. The simplified diagram of the PXS is shown in Figure 1 and 2.

The PXS is designed to perform the following major safety-related functions:

- Emergency RCS Makeup and Boration, provide RCS makeup and boration during transients or accidents where the normal RCS makeup supply from the Chemical and Volume Control System (CVCS) is unavailable or is insufficient.
- Safety Injection, provide safety injection to the RCS to ensure adequate core cooling for the complete range of Loss Of Coolant Accidents (LOCAs) up to and including the double ended rupture of the largest RCS piping. The ADS supports the PXS in performing the safety injection function by depressurizing the RCS to allow lower pressure injection supplies to inject.
- Emergency Core Decay Heat Removal, provide core decay heat removal during transients, accidents, or whenever the normal heat removal paths are unavailable.
- Containment Sump pH Control, provide for chemical addition to the containment sump during post accident condition with high radioactivity in containment to establish floodup chemistry conditions that support radionuclide retention and prevent corrosion of containment equipment during loong-term floodup condition.

The PXS is also designed to perform the non safety related function i.e. to store the water required to flood the refueling cavity during normal plant refueling operations. The PXS can be grouped to several sub-systems i.e: RCS Injection Subsystem, Emergency Core Decay Heat Removal Subsystem, Containment Sump pH Control Subsystem, and Automatic Depressurization Subsystem. A detailed diagram of the PXS is provided in the system Piping and Instrumentation Diagram (P&ID), Reference 2.

2.1. RCS Injection Subsystem

The RCS injection subsystem consists of CMTs, Accumulators, an IRWST, and associated valves, piping and instrumentation. The CMTs provide RCS makeup and boration during events not involving loss of coolant when the normal makeup system is unavailable or insufficient. There are two CMTs located inside the containment at an elevation slightly above the reactor coolant loops, and the boration capability of these CMTs provides adequate core shutdown margin following a steam line break. The CMTs are connected to the RCS through a discharge injection line and two inlet pressure balance lines i.e. one connected to the pressurizer and one to a cold leg. The discharge line and the inlet pressure balance line from the cold leg are each blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. The pressure balance line from the pressurizer is normally open to maintain the CMTs at RCS pressure, which prevents water hammer upon initiation of CMT injection. This pressurizer line contains two check valves in series, to prevent the CMTs from depressurizing in the event of a pressurizer steam space break or a pressure balance line break. The pressurizer line and a portion of the cold leg pressure balance line between the isolation valves and the CMTs are normally maintained full of steam from the pressurizer. The small amount of condensate that forms in this insulated lines is collected by a condensate trap and returned to the cold leg channel head of one Steam Generator.

The outlet line from the bottom of each CMT is relatively large and provides an injection path to one of the two direct vessel injection (DVI) lines, which are connected to the reactor vessel downcomer annulus. Upon receipt of a safeguards actuation signal, the two parallel valves in each inlet and discharge line open to align the associated CMT to the RCS. At the end of CMT discharge line there are two tilt-disc check valves in series that are designed to remain open without flow in the line, these valves prevent reverse flow through this line from the accumulator, that would bypass the reactor vessel in the event of larger loss of coolant accident in the cold leg or the cold leg pressure balance line.

The two accumulator are located inside the containment contain borated water with a boron concentration of about 2500 ppm and compressed nitrogen cover gas (maintained at approximately 700 psig) to provide rapid injection. The discharge from each accumulator tank is connected to one of the DVI lines, which are connected to the reactor vessel downcomer. Each accumulator discharges through a normally-open motor operated isolation valves and two check valves in series, to isolate the accumulators from the RCS during normal plant operation.

The IRWST contains cold borated water, is located in the containment at an elevation slightly above the RCS loop piping. The IRWST is connected to the RCS DVI lines through two gravity injection lines. Each gravity injection line is connected to the bottom of the IRWST and the Containment Sump, and contains a normally open motor-operated isolation valves and four check valves (two series check valves in two paralel paths). Both Containment Sumps are connected to an associated gravity injection line via two paralel paths, one path contains two check valves in series to prevent backflow, and the other path contains two normally closed isolation valves which can be opened to dump the IRWST water into the Containment Sump in case of a core melt.

2.2. Emergency Core Decay Heat Removal Subsystem

The emergency core decay heat removal subsystem primarily consists of two PRHR heat exchangers and associated valves, piping and instrumentation. The heat exchangers are located in the IRWST, which provides the heat sink for the heat exchangers. Only one heat exchanger is required for core decay heat removal, the second heat exchanger provides additional heat removal capacity and to isolate a heat exchanger without a plant shutdown in the event of tube leakage. The PRHR heat exhangers are connected to the RCS through a common inlet line from one RCS hot leg through a tee from one of the fourth stage ADS lines, this inlet line is normally open and connects to the PRHR heat exchanger channel head. The common outlet line from the heat exchangers is connected to the associated steam generator cold leg plennum reactor coolant pump suction. This outlet line contains normally closed air-operated valves that open on loss of air pressure or on control signal actuation.

The alignment of the PRHR heat exchangers with normally open common inlet motor-operated valves and normally closed common outlet air-operated valves, maintains the the heat exchangers full of reactor coolant at RCS pressure and prevents water hammer upon initiation of PRHR heat exchanger operation. Both heat exchangers are elevated above the RCS loops to induce natural circulation flow through the heat exchangers when the reactor coolant pumps are not available. When the reactor coolant pumps are operating, they provide forced flow in the same direction as natural circulation flow through the heat exchangers. If the reactor coolant pumps are operating and subsequently trip, then natural circulation continues to provide the driving head for heat exchanger flow. A vertical pipe stub with the gas level detectors on the top of the inlet piping serves as a gas collection chamber. There are provisions to allow the operator to open shielded manual valves to locally vent these gases to the IRWST during power operation.

2.3. Containment Sump pH Control Subsystem

The Containment Sump pH Control Subsystem is located inside the containment, consists of the pH adjustment tank and associated valves, piping and instrumentation. This subsystem provides for chemical addition (a 30 weight percent sodium hydroxide) to the containment recirculation sump in certain severe accident floodup conditions where core damage has occured and core radioactivity has been released from the RCS into containment, this chemical addition is initiated upon receipt of a high containment radiation signal using a 2 out of 4 logic coincidence. The sodium hydroxide addition is designed to achieve a pH in the containment sump water between 7.0 to 9.5, which will significantly reduce radiolytic formation of elemental iodine in the containment sump, and ultimately will reduce the aqueous production of organic iodine and the total airborne iodine in containment.

The pH adjustment tank discharges through two paralel normally closed squib valves to two discharge sumps, each sump discharge line contain a flow tuning orifice that is used to provide a mechanism for the field adjustment to balance the discharge line flow rates. A temporary connection can be made up to demineralized water system to perform flow testing of the discharge piping. The used of squib valves (which do not open on loss of power) and packless metal diaphragm valves in this subsystem, minimizes the potential for leakage.

2.4. System Operation

2.4.1. Normal Plant Operation

During normal plant power operation, the two CMTs are full of cold borated water and maintained at RCS pressure. The pressurizer line and a portion of the cold leg line between the isolation valves and the CMT are normally maintained full of steam, with condensate removed through a steam condensate drain back to the cold leg channel head of one SG. The PRHR heat exchangers are maintained full of cold RCS coolant at full RCS pressure. The IRWST is normally maintained nearly full of water with an air space at the top of the tank that is sealed from the containment atmosphere, whereas the accumulators are normally maintained approximately 85 percent full of water with a nitrogen cover gas at about 700 psig.

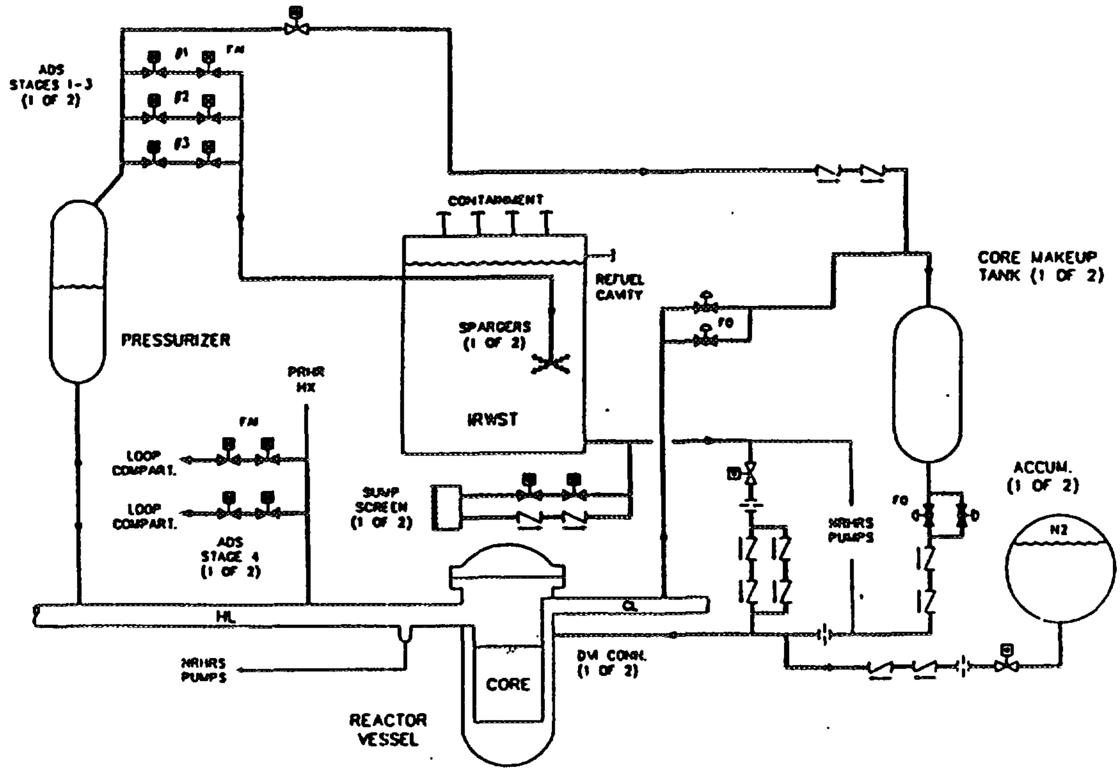

FIG. 1. AP600 passive safety injection.

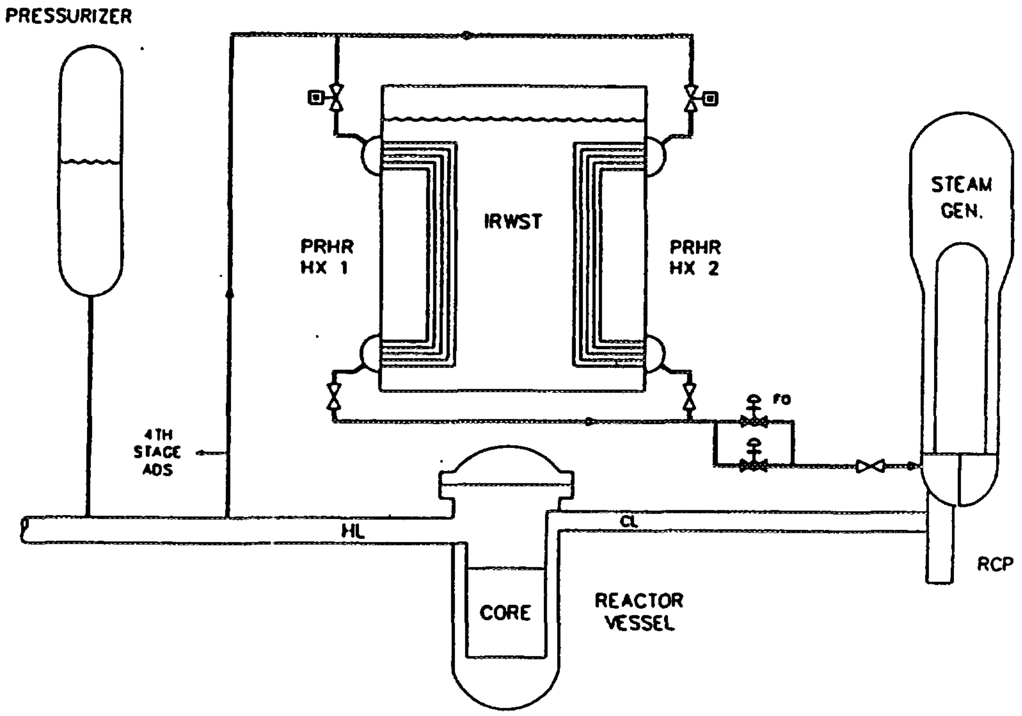

FIG. 2. AP600 passive residual heat removal.

Following a shutdown or trip, no PXS actuation occur as long as normal RCS heat removal from the Startup Feedwater System (SFW), and inventory control from the CVS, are maintained.

2.4.2. Plant Accident Operation

- Non LOCA Operation

The non-LOCA events can lead to significant increases or reductions in the capability of the secondary system to remove heat generated in the core, the two most limiting events are the loss of main feedwater and the feed line break, as well as the steam line break and inadvertent opening of a SG or safety valve.

Should the Main Feedwater System (MFW) and SFW be unavailable, the PRHR heat exchangers are actuated by a low narrow range SG level and coincident low SFW flow signal, to cool down the RCS. When the PRHR heat exchanger cooling sufficiently reduces pressurizer level or RCS temperature, the CMTs are initially actuated an inject borated water directly into the reactor vessel downcomer annulus. Once the CMTs are actuated the RCS pumps are tripped, and the PRHR heat exchangers begin to operate under natural circulation. The RCS does not depressurize sufficiently to permit the accumulators to deliver makeup water to the RCS. Subsequent to stabilizing plant conditions and satifying PXS termination criteria, the operator terminates PXS operation and initiates normal plant shutdown operations.

The feed line break are associated with a double-ended rupture of a feed line at full power. For this event, the PRHR heat exchangers and the CMTs are actuated. Since the RCS pumps are tripped on actuation of the CMTs, the PRHR heat exchangers operate under natural circulation. The RCS does not depressurize sufficiently to permit the accumulators to deliver makeup water to the RCS, and subsequently the operator terminates the PXS operation and initiates normal plant shutdown operations.

The steam line break are associated with a double-ended rupture of a main steam line, occurring at zero power. In this event the PRHR heat exchangers and the CMTs are actuated but not sufficient to prevent the reactor from returning to criticality during the transient. The injection flow is not sufficient to reduce CMT level and to actuate ADS. The RCS may depressurize sufficiently to permit the accumulators to deliver makeup water tank to the RCS, and subsequently the operator terminates the PXS operation and initiates normal plant shutdown operations.

Inadvertent opening of steam generator (SG) relief or safety valve, in this event the reactor is tripped, the CMTs and the PRHR heat exchangers are actuated and RCS pumps are tripped. The CMTs may drain down if the RCS cooldown is fast enough to reduce the pressurizer level to a low level. However, the safety analysis shows that the ADS is not actuated.

- LOCA Operation

A LOCA is a rupture of the RCS piping or branch piping that results in a decrease in RCS inventory that exeeds the flow capability of the normal makeup system. The postulated piping breaks in the RCS is divided into major pipe breaks or large breaks i.e. a rupture with a total cross sectional area equal to or greater than one square foot, and minor pipe breaks or small breaks (a rupture with a total cross sectional area less than one square foot). Following a postulated LOCA, the RCS pressure dcreases and initiates a reactor trip and safety injection. The safety injection signal trips the RCS pumps and opens the CMT inlet and outlet isolation air operated valves (AOVs). The CMTs provide high pressure injection and can operate via water recirculation or steam-compensated injection at full RCS pressure. For smaller breaks, the pressurizer level is sufficient to initially establish water recirculation, but for larger break sizes, the pressurizer level decreases more rapidly and steam-compensated injection occurs.

When the level in either of the CMT decreases to its Low-1 level setpoint, ADS is actuated. The depressurization of the RCS is staged to limit the depressurization rate and the maximum vent flow from the spargers to the depressurization spargers. At a volume of about 67%, the first stage valves actuate, these 4 inch MOVs are connected to the top of the pressurizer and discharge to the IRWST via the spargers. In 60 seconds after the first stage valves actuate, the second stage valves actuate, these are 8 inch MOVs that are connected with the same flow path as the first stage valves. In 120 seconds after the second stage valves actuate, the third stage valves actuate, these 8 inch valves are identical to the second stage valves. As the CMT drops to a low volume about 20%, the fourth stage valves are actuated , these are 10 inch MOVs that are connected to both hot legs and they discharge directly to the SG compartments.

After depressurization, the IRWST provides gravity injection flow and this flow continues until containment flood-up initiates containment sump recirculation. When the water level in the IRWST reaches a low-level, the water level in the containment sump has increased to a sufficient level to passively initiate recirculation flow. This permit continued cooling of the core by recirculation of accumulated water in the containment. When the IRWST level reaches a low-low level setpoint, two MOVs in the line between the continment sump and the gravity injection line open. This provides a redundant flow path in parallel with the containment sump check valves. In this long term cooling mode, the core is covered and steams to the containment via the break and/or ADS valves. The steam is condensed on the steel containment shell, which is cooled via the Passive Containment Cooling System (PCS) and water is returned to the reactor vessel via the IRWST and/or the recirculation lines.

The other LOCA operation are a Steam Generator Tube Rupture (SGTR) and PRHR Heat Exchanger Tube Rupture. Following a SGTR event, reactor coolant flows from the primary to the secondary side of the ruptured SG, the pressurizer level decreases due to the loss of inventory, and RCS pressure decreases, reactor trip and SI signals are generated due to low pressurizer pressure. The CMTs operate via water recirculation or steam-condensated injection to maintain RCS inventory. The PRHR heat exchangers serve to remove core decay heat, and since the reactor coolant pumps are automatically tripped on actuation of the CMTs, the PRHR heat exchangers operate under natural flow conditions. As the RCS cools, pressurizer level and pressure decrease, equalizing with SG pressure and automatically terminating break flow. In this events, the plant conditions are stabilized without actuating the ADS. Whereas in the PRHR heat exchanger tube rupture event, the operators can use available instrumentation to identify the faulted heat exchanger and action can then be taken to remotely isolate both heat exchangers by closing the motor-operated inlet isolation valves, which are normally open. The faulted heat exchanger can then be isolated, and the plant can operate indefinitely with one of the heat exchangers isolated.

3. METHODOLOGY

The methodology used in this analysis is a quantitative Fault Tree Analysis (FTA) method, that is to provide the quantitative estimate of the system's contribution to plant unavailability. The quantitative estimate of the system unavailability is then compared to its allocated or goal value as shown in Table 1 of Reference 4. If the estimated value is greater than the allocated value, then the recommendations will be presented for further consideration and evaluation.

The FTA methodology utilized for the PXS availability analysis is shown in the functional (reliability) block diagram in Figure 3. This diagram defines the logic flow for the functional success or failure of the system, the fault tree is then developed from this functional block diagram. In the fault tree quantification pocess; the failure rates, repair times, and unavailability values for each components are calculated based on the historic data for similar components. As results of quantifying the fault tree are : the quantitative estimate of the system's contribution to the plant unavailability, and a list that ranks the components in order of the relative contribution to the total system unavailability.

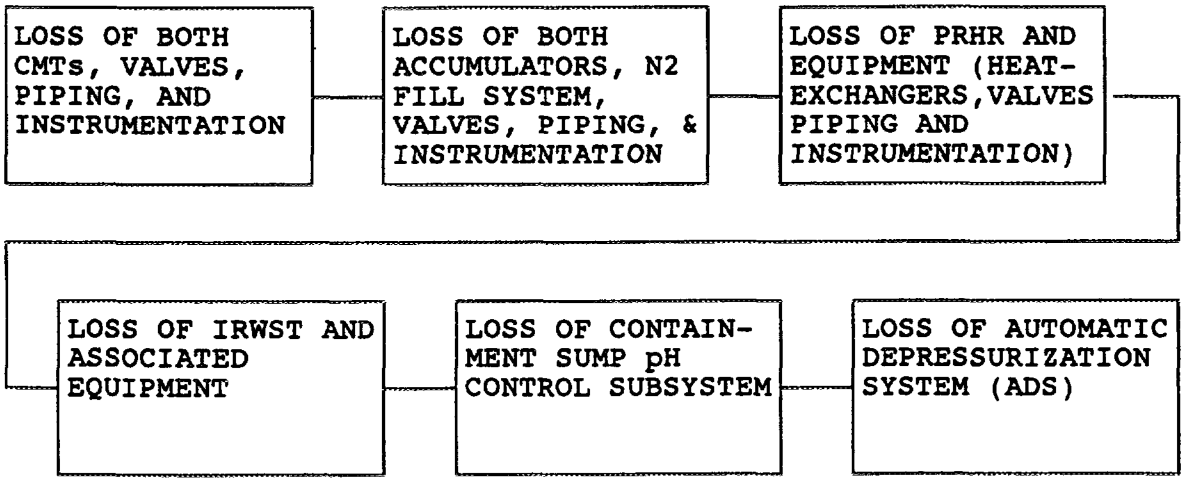

FIG. 3. Simlified reliability block diagram for the passive core cooling system.

Refer to the PXS System Specification Document (Reference 1), that the Automatic Depressurization System (ADS) has been relocated to RCS SSD, but because of successful operation of the PXS is dependent on automatic depressurization of RCS, therefore two scenarios is established in this analysis. The first scenario is by including the ADS as a subsystem of PXS, and the second is by excluding the ADS from the PXS subsystem.

There are two valve options in arrangement of ADS valves, for the second and third stage of the ADS valves, both options is considered to be similar from the reliability point of view, and for stage 4 both valve options are considered in the fault tree. The first PXS fault tree is constructed using the first option of ADS valves (in all stages), the second PXS fault tree is constructed with the stage 4 of ADS valves are arranged in Option 2, and the last PXS fault tree is constructed by excluding the ADS from the PXS subsystem.

The ADS valves are arranged in 4 different stages and the valve stages are configured into lines, each lines containing 2 valves in series. The configuration of ADS valves are as follows:

- Stage 1 : consists of one 4-inch motor operated normally closed (NC) isolation gate valve in series with, one 4-inch motor operated NC globe control valve.
- Stage 2 (is the same with Stage 3) : there are two valve options for stage 2 and stage 3;
  Option-1: two 8-inch motor operated NC gate valves in series.
  Option-2: one 8-inch motor operated NC gate valve (isolation) in series with one 8-inch motor opertaed NC globe valve (flow control).
- Stage 4 : there are also two valve options;
  Option-1: two 10-inch air piston operated NC gate valves in series.
  Option-2: one 10-inch motor operated normally closed gate valve (isolation) in series with one 10-inch normally closed squib valve (control).

The ADS RCS final valve configuration has not been selected, as such, the RCS P&ID (Reference 2) contains generic ADS valves.

4. RESULTS AND EVALUATION

In developing and quantifying the PXS fault tree, the following assumptions have been made:

- All component failure rates used are assumed to be constant with time, and age degradation is not modeled.
- The repair time (MTTR) for major PXS components with an external leak requiring maintenance is assumed to be 64 hours i.e. 10 hours to cooldown to 200 °F, 10 hours to repair valve, 40 hours to heatup to no-load temperature and 4 hours to synchronize to grid.
- For components requiring replacement, the MTTR is assumed to be 200 hours, and for components that are isolatable from RCS pressure or do not involve a pressure are assumed to have smaller MTTRs (24 hours or less).
- All repairs can be performed within technical specification requirement.

The GRAFTER code was used for the quantification process. The PXS data and fault tree are presented in Appendix A and Appendix B respectively. The quantitative analysis result is presented in Table 1, this quantitative results indicates that the PXS unavailability with all ADS valves arrangement in Option-1 is 2.347E-03, this means that the yearly contribution to plant down time can be estimated to be about 20.56 hours per year. Whereas, by using Option-2 of fourth stage ADS valves, the PXS unavailability is reduced to be 9.877E-04 or 8.65 hours per year. The major contributors to the yearly plant downtime are listed in Table 2 and Table 3. The major contributors are mostly dominated by Stage-4 ADS valves i.e.: air piston operated valves (if valves configuration of Option-1 is chosen) and squib valves (if configuration of Option-2 is chosen), inservice testing valves of ADS or solenoid operated valves, solenoid valves of Nitrogen Supply to Accumulator lines, and PRHR actuation valves.

It is shown in Table 1 that the ADS contributes 66.89% to the system unavailability if it is arranged in Option-1, and will reduced to be about 21.21% if it is arranged in Option-2 for fourth stage of ADS. If the ADS is not included as a subsystem of the PXS (relocate to RCS as a subsystem of RCS), then the PXS unavailability will reduced to be about 7.784E-04 or 6.82 hours per year.

Table 1. The PXS unavailability (Q) and its subsystem ontribution (using Option 1 and Option 2 of ADS valves and without ADS valves).

SUBSYSTEM	ADS OPTION 1		ADS OPTION 2		NO ADS	
	Q	%CONTR	Q	%CONTR	Q	%CONTR
RCS Inject.(CMTs, ACCs, and IRWST)	3.923E-04	16.71%	3.923E-04	39.72%	3.923E-04	50.40%
Emerg. Core Decay Heat Removal(PRHR)	3.123E-04	13.30%	3.123E-04	31.62%	3.123E-04	40.12%
Containment Sump pH Control.	7.386E-05	3.15%	7.386E-05	7.48%	7.386E-05	9.49%
Automatic Depre- surization (ADS)	1.398E-03	66.89%	2.095E-04	21.21%	-	-
Total (PXS)	2.347E-03	100%	9.877E-04	100%	7.35E-04	100%

Table 2. The PXS major contributors to the yearly plant downtime using ADS valves configuration in Option 1.

SYSTEM UNAVAILABILITY (Q) = 2.347E-03

	COMP. IDENTIFIER (BASIC EVENT)	COMPONENT DESCRIPTION	IMPORTANCE (%DECREASE)	# OF C.S.	DECREASE IN Q	BASIC EVENT PROBABILITY
1	PVV014AFTO	Piston op. vlv V014A failure	7.33	1	1.7200E-04	1.7200E-04
2	PVV004AFTO	Piston op. vlv V004A failure	7.33	1	1.7200E-04	1.7200E-04
3	PVV014CFTO	Piston op. vlv V014C failure	7.33	1	1.7200E-04	1.7200E-04
4	PVV004CFTO	Piston op. vlv V004C failure	7.33	1	1.7200E-04	1.7200E-04
5	PVV014BFTO	Piston op. vlv V014B failure	7.33	1	1.7200E-04	1.7200E-04
6	PVV004BFTO	Piston op. vlv V004B failure	7.33	1	1.7200E-04	1.7200E-04
7	PVV014DFTO	Piston op. vlv V014D failure	7.33	1	1.7200E-04	1.7200E-04
8	PVV004DFTO	Piston op. vlv V004D failure	7.33	1	1.7200E-04	1.7200E-04
9	AVV108AELRILSO	Air op. vlv V108A failure	5.11	1	1.2000E-04	1.2000E-04
10	AVV108BELRILSO	Air op. vlv V108B failure	5.11	1	1.2000E-04	1.2000E-04
11	SVV021AELR	Solenoid op. vlv V021A failure	.99	1	2.3200E-05	2.3200E-05
12	SVV021BELR	Solenoid op. vlv V021B failure	.99	1	2.3200E-05	2.3200E-05
13	SVV007AELR	Solenoid op. vlv V007A failure	.99	1	2.3200E-05	2.3200E-05
14	SVV007CELR	Solenoid op. vlv V007C failure	.99	1	2.3200E-05	2.3200E-05
15	SVV006AELR	Solenoid op. vlv V006A failure	.99	1	2.3200E-05	2.3200E-05
16	SVV006CELR	Solenoid op. vlv V006C failure	.99	1	2.3200E-05	2.3200E-05
17	QVV301AELR	Squib valve V301A failure	.41	1	9.6000E-06	9.6000E-06
18	QVV301BELR	Squib valve V301B failure	.41	1	9.6000E-06	9.6000E-06
19	RIRIA160ALL	Containment rad. sensor RIA160	.37	3	8.5800E-06	1.6900E-03

The AP600 system unavailability estimates and allocations for the PXS is allocated as 8.0 hours per year (Reference 4). Thus, the estimated PXS unavailability with the ADS valves configuration in Option-1 does not meet the system unavailability allocation (much greater than the allocated goal), and for the PXS with the fourth stage of ADS valves is arranged according to Option-2 (by using squib valves), the system unavailability is consistent with the allocated goal.

Table 3. The PXS major contributors to the yearly plant downtime using ADS valves configuration in Option 2.

SYSTEM UNAVAILABILITY (Q) = 9.877E-04

	COMP. IDENTIFIER (BASIC EVENT)	COMPONENT DESCRIPTION	IMPORTANCE (%DECREASE)	# OF C.S.	DECREASE IN Q	BASIC EVENT PROBABILITY
1	AVV108AELRILSO	Air op. vlv V108A failure	12.15	1	1.2000E-04	1.2000E-04
2	AVV108BELRILSO	Air op. vlv V108B failure	12.15	1	1.2000E-04	1.2000E-04
3	SVV021AELR	Solenoid op. vlv V021A failure	2.35	1	2.3200E-05	2.3200E-05
4	SVV021BELR	Solenoid op. vlv V021B failure	2.35	1	2.3200E-05	2.3200E-05
5	SVV007AELR	Solenoid op. vlv V007A failure	2.35	1	2.3200E-05	2.3200E-05
6	SVV007CELR	Solenoid op. vlv V007C failure	2.35	1	2.3200E-05	2.3200E-05
7	QVV004AFTO	Squib vlv V004A fail to open	1.21	1	1.2000E-05	1.2000E-05
8	QVV004CFTO	Squib vlv V004C fail to open	1.21	1	1.2000E-05	1.2000E-05
9	QVV004BFTO	Squib vlv V004B fail to open	1.21	1	1.2000E-05	1.2000E-05
10	QVV004DFTO	Squib vlv V004D fail to open	1.21	1	1.2000E-05	1.2000E-05
11	QVV004AELR	Squib vlv V004A ex. leak/rupt.	.97	1	9.6000E-06	9.6000E-06
12	QVV004CELR	Squib vlv V004C ex. leak/rupt.	.97	1	9.6000E-06	9.6000E-06
13	QVV004BELR	Squib vlv V004B ex. leak/rupt.	.97	1	9.6000E-06	9.6000E-06
14	QVV004DELR	Squib vlv V004D ex. leak/rupt.	.97	1	9.6000E-06	9.6000E-06
15	QVV301AELR	Squib vlv V301A ex. leak/rupt.	.97	1	9.6000E-06	9.6000E-06
16	QVV301BELR	Squib vlv V301B ex. leak/rupt.	.97	1	9.6000E-06	9.6000E-06
17	RIRIA160ALL	Containment rad. sensor RIA160	.87	3	8.5800E-06	1.6900E-03
18	RIRIA161ALL	Containment rad. sensor RIA161	.87	3	8.5800E-06	1.6900E-03
19	RIRIA162ALL	Containment rad. sensor RIA162	.87	3	8.5800E-06	1.6900E-03
20	RIRIA163ALL	Containment rad. sensor RIA163	.87	3	8.5800E-06	1.6900E-03
21	OVV306TELR	Press. ctrl op. vlv V306T fails	.80	1	7.8700E-06	7.8700E-06
22	AVV002AELR	Air op. vlv V002A failure	.73	1	7.2400E-06	7.2400E-06

5. CONCLUSIONS AND RECOMMENDATIONS

The analysis results show that the unavailability (Q) of AP600 PXS can be estimated as follows:

1. Q = 2.347E-03 or 20.56 hr/year, if the ADS is included as a subsystem of the PXS and the first option of ADS valves arrangement is used.
2. Q = 9.877E-04 or 8.65 hr/year, if the ADS is included as a subsystem of the PXS and the second option arrangement of ADS valves is used.
3. Q = 7.784E-04 or 6.82 hr/year, if the ADS is excluded from the PXS subsystem (relocated as a subsystem of RCS)

The system unavailability in conclusions 1 is greater than the allocated unavailability goal, the value in conclusion 2 is consistent with the allocated goal, whereas the system unavailability in conclusion 3 is better than the allocated goal of 8.0 hr/year.

The main contributors to the plant downtime attributed by this system are mostly dominated by air piston operated valves (ADS), solenoid operated valves (inservice testing of ADS), solenoid operated valves of Nitrogen Supply to Accumulator lines and PRHR actuation valves. Therefore, it is recommended that those valves be analyzed more detail to gain the improvement in its reliability. It is also recommended that the fourth stage of ADS valves should be arranged according to Option-2, i.e. one 10-inch normally open motor operated gate valve in series with one 10-inch normally closed squib valve, this configuration will lead the PXS unavailability much closer to the allocated goal.

ACKNOWLEDGEMENT

The assistance and support of the people from Reliability Engineering group of Westinghouse Electric Corporation are greatly appreciated.

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12. Nuclear Plant Reliability Data Systems, NPRDS.

13. IEEE STD-500, 1984.

APPENDIX A: FAILURE RATE DATE FOR RCS AND PXS

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'MASTER DATA FILE FOR AP600 RCS AND PXS SYSTEMS'

 98   15
  1 3.13E-04 0.00E+00 2 'AP6 '   'AV'     'RCS'     'PRESSURIZER BALL VALVE SPUR. OPER.(FO)/LEAK/RUPT      '
  2 1.20E-04 0.00E+00 2 'AP6 '   'AV'     'RCS'     'PRESSURIZER BALL VALVE FAILS TO CLOSE                 '
  3 4.16E-05 0.00E+00 2 'AP6 '   'AV'     'RCS'     'PRESSURIZER BALL VALVE FAILS TO OPEN                  '
  4 3.12E-07 0.00E+00 2 'AP6 '   'FE'     'RCS'     'FLOW ELEMENT (ORIFICE) EXTERNAL LEAK/RUPTURES/PLUGS   '
  5 5.55E-05 0.00E+00 2 'AP6 '   'FT'   . 'RCS'     'FLOW INDICATOR FAILS ANY MODE                         '
  6 2.60E-04 0.00E+00 2 'AP6 '   'FV'     'RCS'     'FLEXI-DISC CODE SAFETY VALVE FAILS SPUR. OP/LK/RUPT   '
  7 5.76E-07 0.00E+00 2 'AP6 '   'LT'     'RCS'     'LEVEL TRANSMITTER LEAKS/RUPTURES                      '
  8 6.12E-06 0.00E+00 2 'AP6 '   'LT'     'RCS'     'LEVEL TRANSMITTER FAILS ALL MODES                     '
  9 1.29E-07 0.00E+00 2 'AP6 '   'RD'     'RCS'     'CODE SAFETY RUPTURE DISC FAILS                        '
 10 1.60E-04 0.00E+00 2 'AP6 '   'RV'     'RCS'     'MISCELLANEOUS VESSEL PROBLEMS                         '
 11 6.84E-07 0.00E+00 2 'AP6 '   'PI'     'RCS'     'PIPING CRACKS/RUPTURES  (10 METER LENGTH)             '
 12 3.00E-04 0.00E+00 2 'AP6 '   'PM'     'RCS'     'REACTOR COOLANT PUMP FAILS ANY MODE                   '
 13 5.76E-07 0.00E+00 2 'AP6 '   'PT'     'RCS'     'PRESSURE TRANSMITTER FAILS ALL MODES                  '
 14 2.70E-03 0.00E+00 2 'AP6 '   'SG'     'RCS'     'STEAM GENERATOR FAILS ANY MODE                        '
 15 8.11E-07 0.00E+00 2 'AP6 '   'TE'     'RCS'     'TEMPERATURE ELEMENT EXTERNAL LEAK/RUPTURE             '
 16 4.17E-05 0.00E+00 2 'AP6 '   'TE'     'RCS'     'TEMPERATURE ELEMENT FAILS HIGH/LOW OR DRIFTS          '
 17 3.07E-06 0.00E+00 2 'AP6 '   'XV'     'RCS'     'MANUAL VALVE INTERNAL LEAK/FAILS OPEN                 '
 18 4.85E-06 0.00E+00 2 'AP6 '   'XV'     'RCS'     'ANGLE VALVE RUPTURES/LEAKS/FAILS CLOSED               '
 19 2.94E-06 0.00E+00 2 'AP6 '   'XV'     'RCS'     'MANUAL VALVE FAILS CLOSED OR FAILS TO OPEN            '
 20 1.91E-06 0.00E+00 2 'AP6 '   'XV'     'RCS'     'MANUAL VALVE EXTERNAL LEAK/RUPTURE                    '
 21 4.98E-06 0.00E+00 2 'AP6 '   'XV'     'RCS'     'MANUAL VALVE RUPTURES/LEAKS/FAILS OPEN                '
 22 2.47E-04 0.00E+00 2 'AP6 '   'PZ'     'RCS'     'PRESSURIZER FAILS ALL MODES                           '
 23 7.78E-06 0.00E+00 2 'AP6 '   'MV'     'RCS'     'MOV FAILS CLOSED/EXTERNAL LEAK/RUPTURE                '
 24 1.18E-04 0.00E+00 2 'AP6 '   'MV'     'RCS'     'MOV FAILS TO CLOSE/INT. LEAK/EXT. LEAK/RUPTURE        '
 25 7.24E-06 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP GLOBE EXTERNAL LEAK/RUPTURE                    '
 26 7.20E-05 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP GLOBE SPURIOUSLY OPENS                         '
 27 3.28E-04 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP GLOBE/GATE VALVE FTO/FTC, MTTR=40 hrs.         '
 28 4.82E-05 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP GLOBE EXTERNAL LEAK/RUPTURE/INTERNAL LEAK      '
 29 1.20E-04 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP GLOBE EXT. LEAK/RUPTURE/INT. LEAK/SPUR. OPEN   '
 30 1.99E-04 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP BALL VALVE EXT. LEAK/RUPTURE/SPURIOUSLY OPENS'
 31 6.02E-06 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP BALL VALVE FAILS OPEN/FAILS CLOSED             '
 32 3.28E-04 0.00E+00 2 'AP6 '   'AV'     'PXS'     'AIR-OP BALL VALVE FAILS TO CLOSE/FAILS TO OPEN        '
 33 2.63E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE EXTERNAL LEAK/RUPTURE             '
 34 4.81E-06 0.00E+00 2 'AP6 '   'XV'     'PXS'     'MANUAL GLOBE/GATE EXTERNAL LEAK/RUPTURE               '
 35 8.40E-07 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE RUPTURES                          '
 36 9.97E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GLOBE EXT. LEAK/RUPTURE/SPURIOUSLY OPENS     '
 37 1.54E-05 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GLOBE INTERNAL LEAK/SPURIOUSLY OPENS         '
 38 7.13E-04 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GLOBE EXT. LEAK/RUPTURE/INT LEAK/FAILS OPEN'
 39 3.66E-06 0.00E+00 2 'AP6 '   'CV'     'PXS'     'CHECK VALVE EXTERNAL LEAK/RUPTURE                     '
 40 3.84E-05 0.00E+00 2 'AP6 '   'CV'     'PXS'     'CHECK VALVE INTERNAL LEAK                             '
 41 1.26E-06 0.00E+00 2 'AP6 '   'CV'     'PXS'     'CHECK VALVE EXTERNAL LEAK/RUPTURE                     '
 42 1.20E-04 0.00E+00 2 'AP6 '   'CV'     'PXS'     'CHECK VALVE INTERNAL LEAK                             '
 43 1.84E-06 0.00E+00 2 'AP6 '   'CV'     'PXS'     'CHECK VALVE RUPTURES                                  '
 44 4.62E-06 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (68 m), MTTR=200 hrs                   '
 45 2.18E-08 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE, BLIND FLANGE (1 m), MTTR=64 hrs       '
 46 6.80E-07 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (10 m), MTTR=200 hrs                   '
 47 1.63E-06 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (100 m), MTTR=48 hrs                   '
 48 8.16E-07 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (100 m), MTTR=24 hrs                   '
 49 1.56E-06 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (23 m), MTTR=200 hrs                   '
 50 4.22E-06 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (62 m), MTTR=200 hrs                   '
 51 3.72E-06 0.00E+00 2 'AP6 '   'PI'     'PXS'     'PIPING RUPTURE (62 m), MTTR=200 hrs                   '
 52 3.07E-06 0.00E+00 2 'AP6 '   'XV'     'PXS'     'MANUAL GLOBE VALVE INTERNAL LEAK                      '
 53 1.69E-06 0.00E+00 2 'AP6 '   'XV'     'PXS'     'MANUAL GLOBE EXTERNAL LEAK/RUPTURE                    '
 54 2.30E-06 0.00E+00 2 'AP6 '   'XV'     'PXS'     'MANUAL GLOBE INTERNAL LEAK                            '
 55 8.40E-07 0.00E+00 2 'AP6 '   'XV'     'PXS'     'MANUAL PLUG VALVE RUPTURES                            '
 56 1.76E-06 0.00E+00 2 'AP6 '   'XV'     'PXS'     'MANUAL GATE VALVE RUPTURES/EXTERNAL LEAK              '
 57 3.20E-04 0.00E+00 2 'AP6 '   'LT'     'PXS'     'LEVEL TRANSMITTER FAILS - ALL MODES, MTTR=64 hrs      '
 58 1.20E-04 0.00E+00 2 'AP6 '   'LT'     'PXS'     'LEVEL TRANSMITTER FAILS - ALL MODES, MTTR=24 hrs      '
 59 1.20E-04 0.00E+00 2 'AP6 '   'PT'     'PXS'     'PRESSURE TRANSMITTER FAILS - ALL MODES                '
 60 1.20E-04 0.00E+00 2 'AP6 '   'FT'     'PXS'     'FLOW TRANSMITTER EXTERNAL LEAK                        '
 61 3.60E-06 0.00E+00 2 'AP6 '   'TK'     'PXS'     'TANK RUPTURES, MTTR=200 hrs                           '
 62 2.32E-05 0.00E+00 2 'AP6 '   'SV'     'PXS'     'SOLENOID-OP GLOBE EXTERNAL LEAK/RUPTURE               '
 63 5.28E-05 0.00E+00 2 'AP6 '   'SV'     'PXS'     'SOLENOID-OP GLOBE INTERNAL LEAK/SPURIOUSLY OPENS      '
 64 3.00E-05 0.00E+00 2 'AP6 '   'SV'     'PXS'     'SOLENOID-OP GLOBE EXT LEAK/RUPT/INT LEAK/SPUR. OPENS'
 65 1.26E-06 0.00E+00 2 'AP6 '   'SV'     'PXS'     'SOLENOID-OP STOP CHECK VALVE EXTERNAL LEAK/RUPTURE    '
 66 6.27E-06 0.00E+00 2 'AP6 '   'RV'     'PXS'     'RELIEF VALVE EXT LEAK/RUPTURE/INT LEAK/SPUR. OPENS    '
 67 3.80E-06 0.00E+00 2 'AP6 '   'FE'     'PXS'     'FLOW ELEMENT, VENTURI FAILS                           '
 68 4.45E-05 0.00E+00 2 'AP6 '   'HX'     'PXS'     'HEAT EXCHANGER TUBE LEAK IN HX 2, HX 1 UNAVAILABLE    '
 69 2.17E-05 0.00E+00 2 'AP6 '   'HX'     'PXS'     'HEAT EXCHANGER TUBE RUPTURE                           '
 70 1.63E-08 0.00E+00 2 'AP6 '   'BF'     'PXS'     'BLIND FLANGE, 2 INCH PIPE EXTERNAL LEAK/RUPTURE       '
 71 1.22E-04 0.00E+00 2 'AP6 '   'CV'     'PXS'     'CHECK VALVE INTERNAL LEAK/RUPTURE                     '
 72 6.02E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE FAILS OPEN/FAILS CLOSED           '
 73 6.14E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE FAILS OPEN/INTERNAL LEAK          '
 74 6.98E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE FAILS OPEN/INT. LEAK/RUPTURE      '
 75 7.81E-05 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE INT. LEAK/SPUR. OPEN/FAILS OPEN   '
 76 3.07E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP GATE VALVE INTERNAL LEAK                     '
 77 4.47E-06 0.00E+00 2 'AP6 '   'PV'     'PXS'     'PISTON-OP GATE VALVE EXTERNAL LEAK/RUPTURE            '
 78 1.54E-05 0.00E+00 2 'AP6 '   'PV'     'PXS'     'PISTON-OP GATE VALVE INTERNAL LEAK/SPURIOUSLY OPENS   '
 79 1.98E-05 0.00E+00 2 'AP6 '   'PV'     'PXS'     'PISTON-OP GATE VALVE EXT LEAK/RUPT/INT LEAK/FAIL OP   '
 80 2.94E-06 0.00E+00 2 'AP6 '   'MV'     'PXS'     'MOTOR-OP VALVE FAILS TO OPEN DUE TO MECH. FAILURE     '
 81 9.60E-06 0.00E+00 2 'NPRD'   'QV'     'PXS'     'SQUIB VALVE EXTERNAL LEAK/RUPTURE, MTTR=24 hrs.       '
 82 1.28E-04 0.00E+00 2 'AP6 '   'SV'     'PXS'     'SOLENOID-OP GLOBE VALVE FAILS TO OPEN, MTTR=64 hrs.   '
 83 2.56E-04 0.00E+00 2 'AP6 '   'SV'     'PXS'     'SOLENOID-OP GLOBE VALVE FTO/FTC, MTTR=64 hrs.         '
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84 2.94E-06 0.00E+00 2 'AP6 '   'XV'    'PXS'   'MANUAL GLOBE VALVE FAILS CLOSED                           '
85 4.89E-04 0.00E+00 2 'AP6 '   'ST'    'PXS'   'STEAM TRAP FAILS CLOSED                                   '
86 1.76E-06 0.00E+00 2 'AP6 '   'ST'    'PXS'   'STEAM TRAP EXTERNAL LEAK/RUPTURE -same as manual vlv'
87 2.29E-06 0.00E+00 2 'AP6 '   'MV'    'PXS'   'MOTOR-OP GLOBE VALVE EXT. LEAK/RUPTURE                    '
88 5.88E-06 0.00E+00 2 'AP6 '   'MV'    'PXS'   'MOV GATE/GLOBE VALVE FTO/FTC DUE TO MECH. FAILURE         '
89 2.02E-07 0.00E+00 2 'AP6 '   'SV'    'PXS'   'SOLENOID-OP GLOBE VALVE RUPTURES                          '
90 1.30E-06 0.00E+00 2 'AP6 '   'FT'    'PXS'   'FLOW TRANSMITTER RUPTURES, MTTR=100 hrs.                  '
91 1.87E-07 0.00E+00 2 'ALWR'   'MV'    'PXS'   'INADVERTENT ACTUATION OF ADS VALVES.                      '
92 1.72E-04 0.00E+00 2 'NPRD'   'PV'    'PXS'   'PISTON OPERATED GATE VALVE FTO, MTTR=24 hrs.              '
93 2.15E-03 0.00E+00 2 'IEEE'   'RI'    'PXS'   'RADIATION SENSOR (GM) & TRANSMITTER FAILS ALL MODES.'
94 7.87E-06 0.00E+00 2 'IEEE'   'OV'    'PXS'   'PRESSURE CONTROL /SELF OPERATED EXT. PRESSURE, EL,R.'
95 3.12E-05 0.00E+00 2 'NPRD'   'QV'    'PXS'   'SQUIB VALVE FAILS ALL MODES, MTTR=24 hrs.                 '
96 1.20E-05 0.00E+00 2 'NPRD'   'QV'    'PXS'   'SQUIB VALVE FAIL TO OPEN, MTTR=24 hrs.                    '
97 4.80E-06 0.00E+00 2 'NPRD'   'QV'    'PXS'   'SQUIB VALVE INTERNAL LEAK, MTTR=24 hrs.                   '
98 2.74E-03 0.00E+00 2 'ENGJ'   'TF'    'PXS'   'ADS TESTING & MAINT. UNAVAILABILITY (every 4 months)'
'         '
'NOTES : The MTTR for major RCS components with external leak requiring maintenance is assumed to be :  '
'        64 hours (10 hrs to cooldown to 200 deg F, 10 hrs to repair valve, 40 hrs to heatup to no load '
'        temp and 4 hrs to synchronize to grid), and for components requiring replacement the MTTR is   '
'        assumed to be 200 hours. Components that are isolatable from RCS pressure or do not involve a  '
'        pressure boundary (e.g. valve operators) are assumed to have smaller MTTRs.                     '
'        '
'        '
'REFFERENCES :'
'            '
'            '
'AP6  : AP600 RAM PROGRAM, Calc. Note : RE-352, November 1990. And AP600 Doc. #: PXS-GOA-001, Oct. 1992 '
'ALWR : SPWR RAM PROGRAM, Inadvertent Actuation Anal. of the ADS, J.T. Kitzmiller & J.M. Lynde, RE, NATD'
'IEEE : IEEE STD 500-IEEE, 1984.'
'NPRD : NUCLEAR PLANT RELIABILITY DATA SYSTEM'
'ENGJ : ENGINEERING JUDGEMENT'
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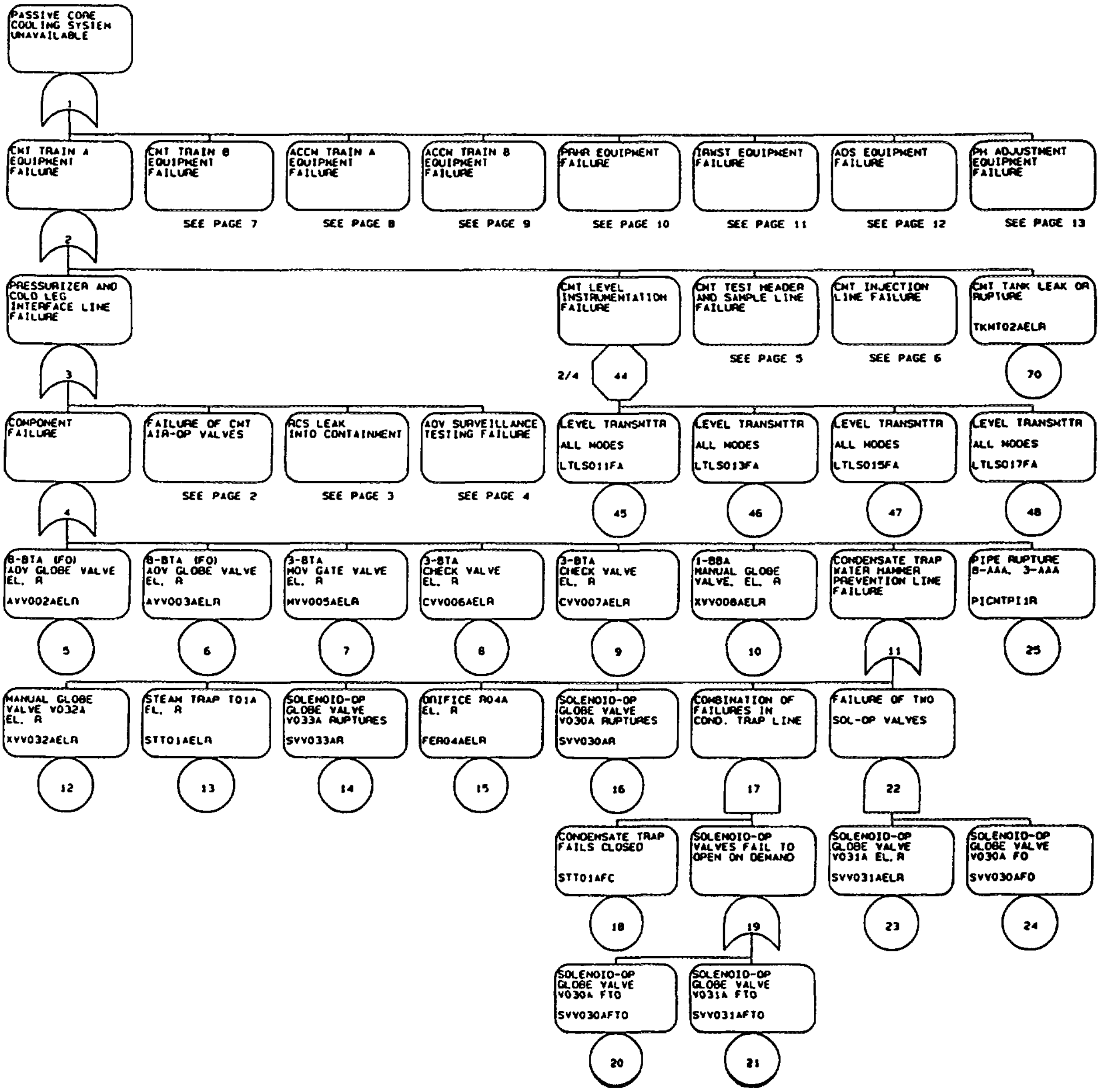
APPENDIX B: AP600 PASSIVE CORE COOLING SYSTEM FAULT TREE



PASSIVE HEAT REMOVAL SYSTEM WITH INJECTOR-CONDENSER

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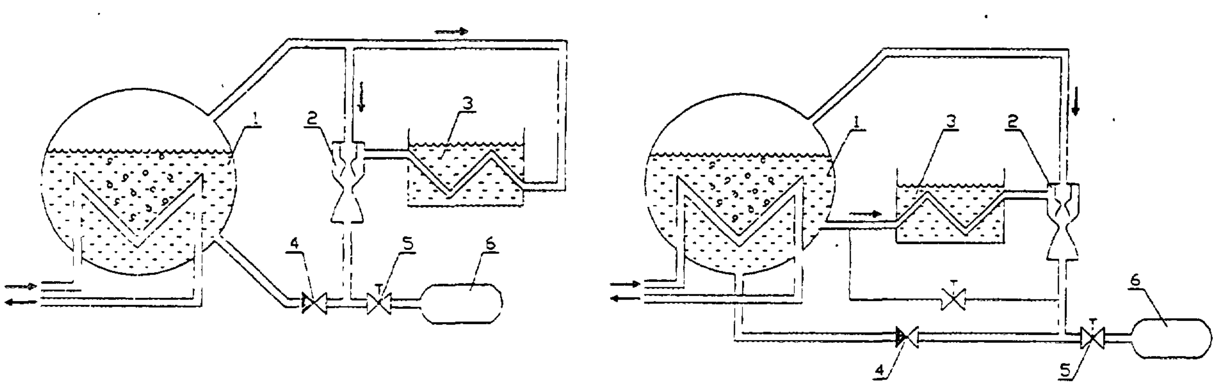
Abstract

The system described in this paper is a passive system for decay heat removal from VVERs. It operates off the secondary side of the steam generators (SG). Steam is taken from the SG to operate a passive injector pump which causes secondary fluid to be pumped through a heat exchanger. Variants pass either water or steam from the SG through the heat exchanger. There is a passive initiation scheme. The programme for experimental and theoretical validation of the system is described.

Description of PHRS-IC

The All-Russian Scientific Research Institute for Nuclear Power Plant (NPP) operation (VNIIAES) has developed a System for Passive Heat Removal using an Injector-Condenser (PHRS-IC). The principle PHRS-IC scheme for NPP steam generator (SG) heat removal are shown in fig.1. The main components of the PHRS-IC are: 1 - the steam generator to be cooled (SG); 2 -injector-condenser (IC); 3 - heat exchanger of evaporating type; 4 - check valve; 5 - start-up valve; 6 - start-up tank.

Scheme "steam-steam" (s-s)   Scheme "steam-liquid" (s-l)



1- Steam Generator   2- Injecter-Condenser   3- Heat Exchanger
4- Check Valve   5- Start-up Valve   6- Start-up Tank

Fig. 1.

The arrangement "Steam-steam" (S-S) is different from the scheme "Steam-liquid" - (S-L) in the way the coolant is supplied into the heat exchanger (3).

The PHRS-IC works in the following way:coolant from the SG (1) enters the IC (2) and the evaporating heat exchanger (3). The cooled condensate from heat exchanger (3) is directed to the IC mixing chamber (2). After the diffuser, the water (the pressure of which is higher than that in SG) is directed into the SG (1) through the check valve (4).

The PHRS-IC start-up. Initially the start-up valve (5) is closed, and the pressure in the start-up tank is much less than that in the circuit. If an emergency situation occurs the start-up valve (5) is opened in a passive way and the coolant begins to enter the start-up tank coming through the IC. The pressure in the start-up tank increases and at some moment in time becomes higher than that in the SG (1). The check valve (4) opens and the PHRS-IC begins to operate.

Experimental facility

The PHRS-IC intended for heat removal from the heat generating source was developed and tested on the experimental facility of the Electrogorsk Research and Engineering Centre of Nuclear Plant Safety (EREC). The general view of the facility and relative elevations of the main components are presented in fig. 2:

1) Supply tank.

The total height of the Supply tank is 5.3 m, its volume is 0.5 m^3. There is a system for steam supply and removal of liquid and steam (not shown in fig. 2) which allows simulation of the different emergency situations (for example, in the SG). The pressure, temperature and water level are measured in the Supply tank. The maximum design pressure and temperature are up to 10 MPa and 315°C respectively.

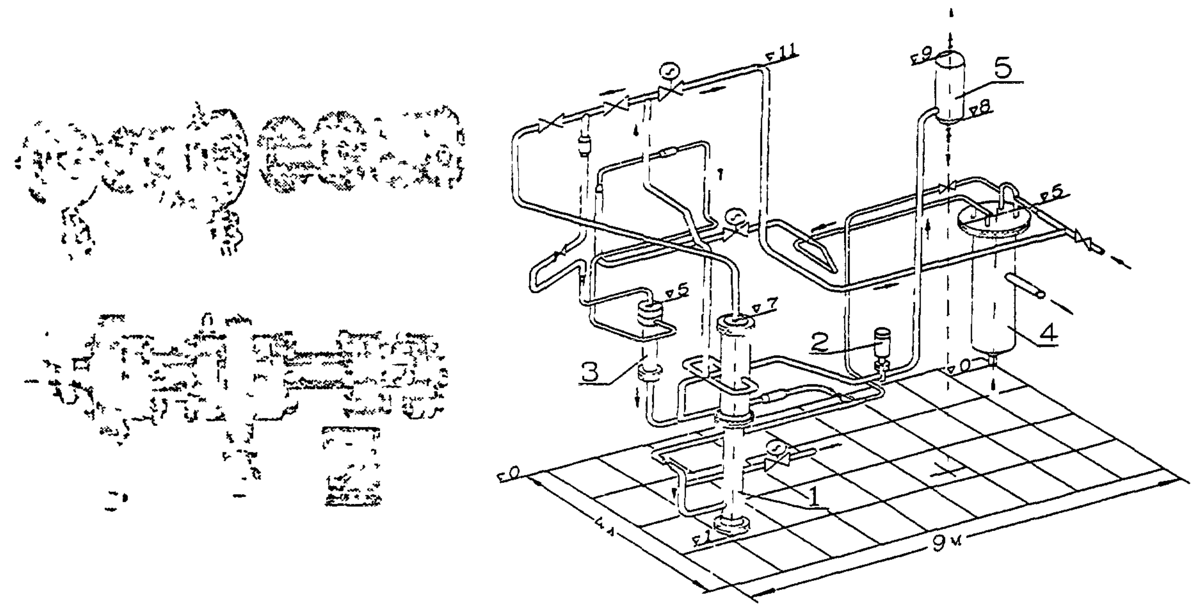


Fig. 2.

2) Injector condenser.

The typical IC for PHRS-IC and thermal-physical processes experiment is presented in Fig.2. The injector allows removal of 3.5 - 4 MWt of power from the heat source.

3) Heat exchanger.

The heat exchanger is: 5 m - height and 0.53 m - diameter. There are two pipe coils enclosed in the vessel. The length of each piping is 80 m. The total heat exchange area is 21 m^2. If the forced water supply is directed to the secondary cooling loop then it is possible to remove 4 - 6 MWt of heat power.

4) Check valve

It is a typical check valve, located between the IC and the supply tank.

5) Start-up valve.

An air-driven valve is installed on the start-up line. Its actuation time is 0.7 - 0.9 s.

6) Start-up tank.

The start up tank is 1.5 m height and 0.426 m in diameter. The allowed pressure is 10 MPa, and the maximum temperature is 315°C.

Measurement system.

The facility allows measurement of pressures, temperatures, pressure differences and water levels. The measurement in non-stationary conditions is made using the conventional primary transducers typical for thermal-physical experiments. The pressure transducers are installed in the facility to measure the fast processes in the IC and the start-up tank during start-up.

The measurement and data collection system is based on the bases of a personal computer IBM PC/AT. Standard "CAMAC" hardware is used as means of communication with the experimental set-up. The system provides for the collection, storing and displaying of the data from pressure and temperature probes. The frequency of channel measurement is 2 Hz and for fast processes is 2 kHz.

Experimental results

The "steam-liquid" scheme was tested on the experimental facility (fig.2) with the injector, a photo of which is presented in the same figure. Results are presented in figures 3-5. The following designations have been used:

P_{sg} - pressure in SG;
P_c - pressure in the cold leg upstream of the IC;
dP_p - pressure drop on the IC;
P_p - pressure downstream from the IC; P_n - pressure at nozzle outlet;
P_{mch} - pressure in the IC mixing chamber;
T_{sg} - temperature in the IC;
T_c - temperature of the "cold" water upstream of the IC;

T_p - temperature of the coolant downstream from the SG;
G_o - mass flow rate of the steam passing through the nozzle;
G_c - mass flow rate of the "cold" water;
G_p - mass flow rate of the water downstream from the IC.

The experimental results of the PHRS-IC start-up and definition of the maximum pressure drops on the IC are presented in fig.3. After start-up valve 5 opens - (fig.2) there is a drop of pressure, P_p, P_n, and P_{mch} (fig.3) and mass flow of the steam and "cold" water are established through the IC. It can be considered that the IC is put into operation if the pressures P_n and P_{mch} correspond to the saturation pressure at the temperature in the mixing chamber. It can also be seen that the injector start-up time is very short (3-5 s). The start-up of the system ($dP_p > 0$) occurs later and depends on the volume of the start-up tank 6 (fig.2). After the filling of the tank 6, the pressure downstream from the IC becomes higher than that in the SG and this leads to the opening of the valve 4. This is one way of starting up the PHRS-IC. It should be noted once more that heat removal from the SG begins just after the valve 5 opens and it is strongest at that time.

The hydraulic resistance downstream from the IC suddenly increases during the experiment when the time is equal ≈ 200s, ≈ 400s, and ≈ 600s (fig.3). The system works in a stable way and the pressure in the mixing chamber does not change. The maximum pressure drop in the IC was slightly higher than 2 MPa.

Behaviour of the PHRS-IC parameters is shown in fig.4 for the conditions of step-by-step pressure reduction in the SG. It could be seen that the system works in a stable way in the range of pressures 1-7 MPa.

In fig.5, where during the first 400s the pressure in the SG was increased from 4.6 MPa up to 6.8 MPa and then it was reduced in a monotone way to 2 MPa, the system operated in a stable way. The changing of the pressures P_n and P_{mch} is caused by the fact that temperature of the "cold" water T_c was changed during the experiment.

The series of experiments have showed us that PHRS-IC has the following characteristics:

- Passive operations. Steam energy resulting from residual heat generation is used for coolant circulation in the circuit;
- Simplicity of passive start-up and possibility of renewed manual start-ups;
- Minimal time interval between the accident and beginning of heat removal;
- Stable functioning in non-stationary conditions within the wide range of SG changing parameters;
- Stability of the system to strong external disturbances (e.g., safety relief valve operation);
- the possibility of automatic or manual PHRS-IC power control in order to -ensure heat removal.

Mathematical modeling of PHRS-IC

Mathematical models at different levels of detail have been developed for the description of the dynamic processes in the PHRS-IC.

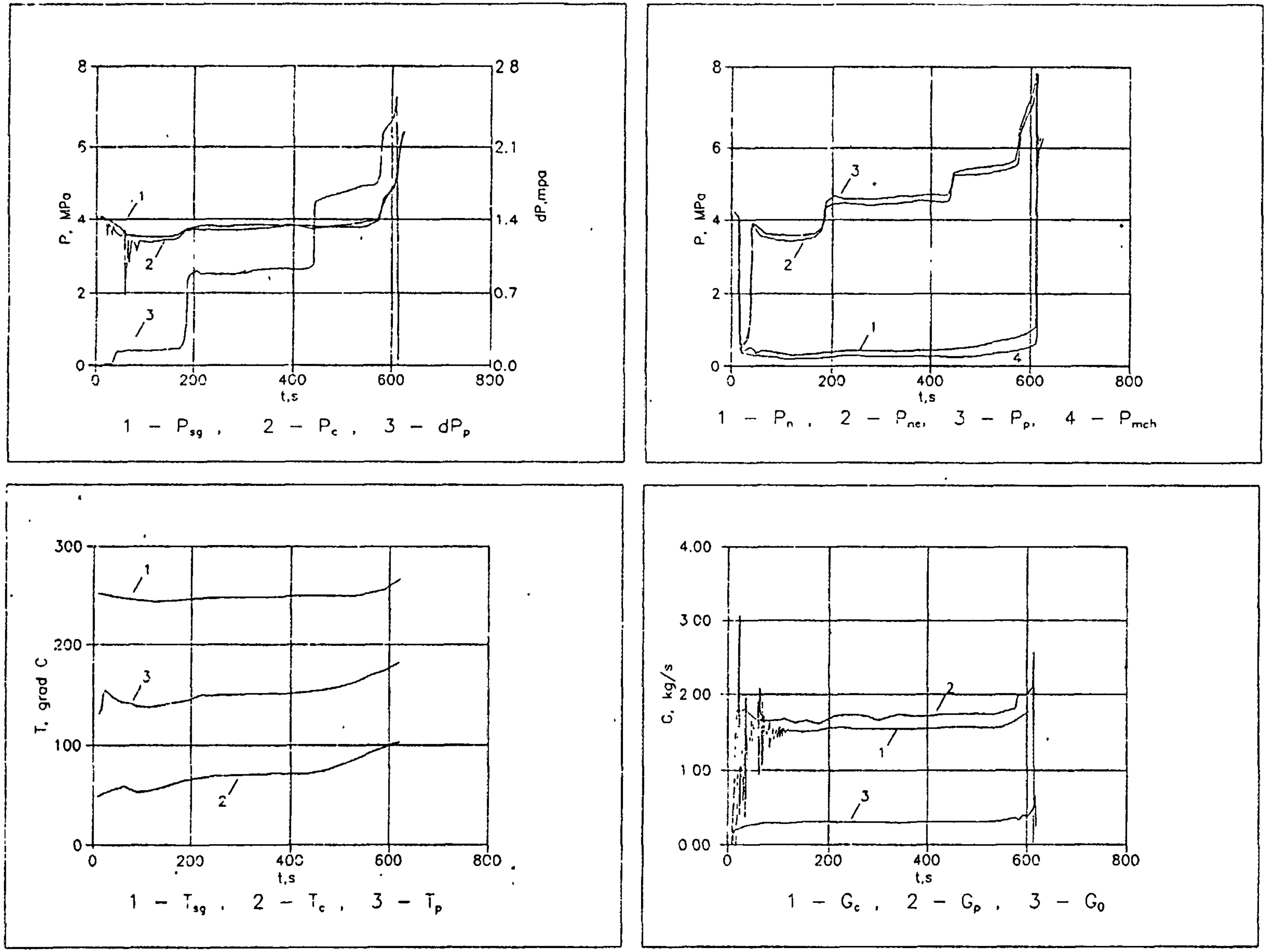
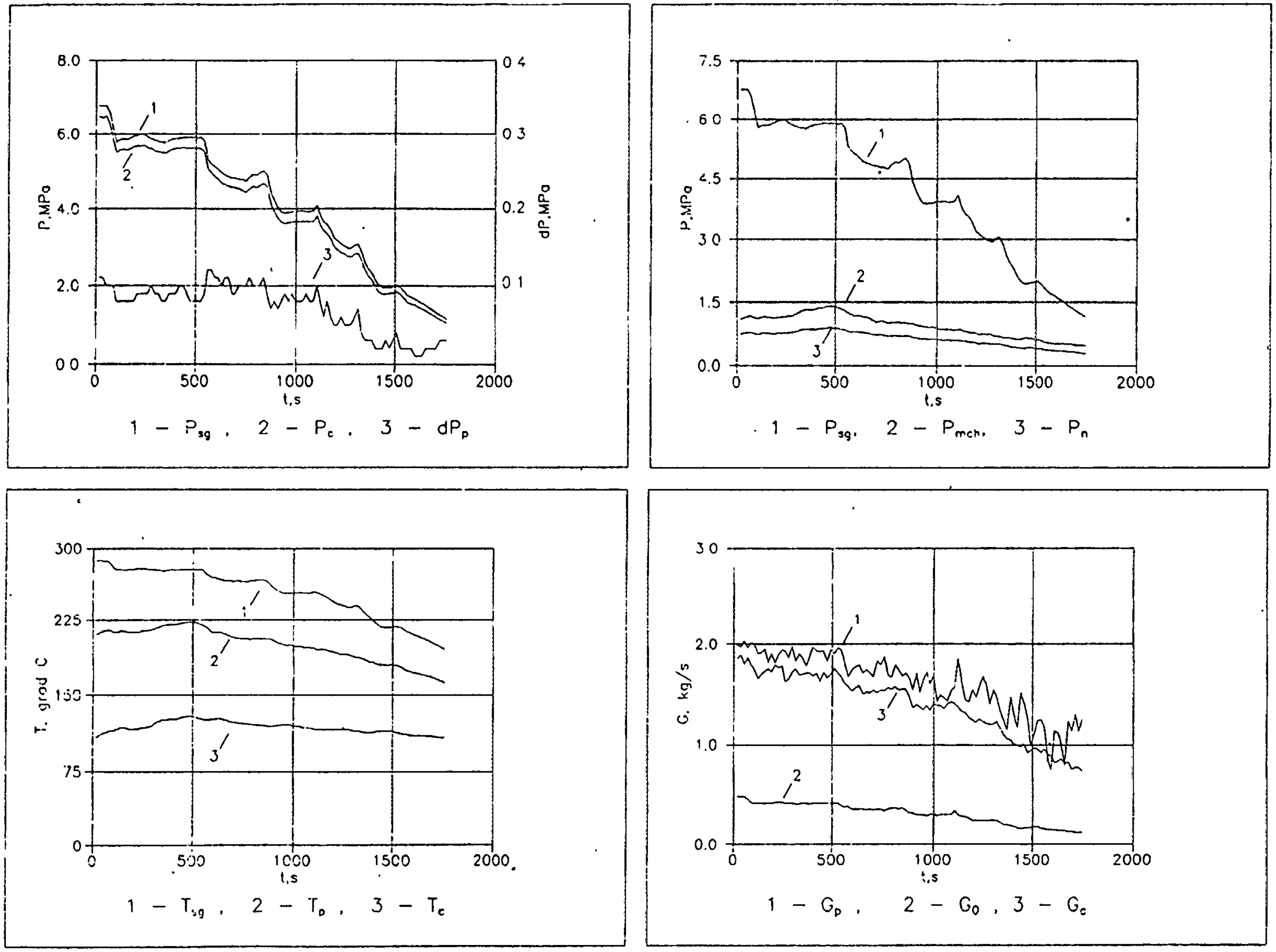
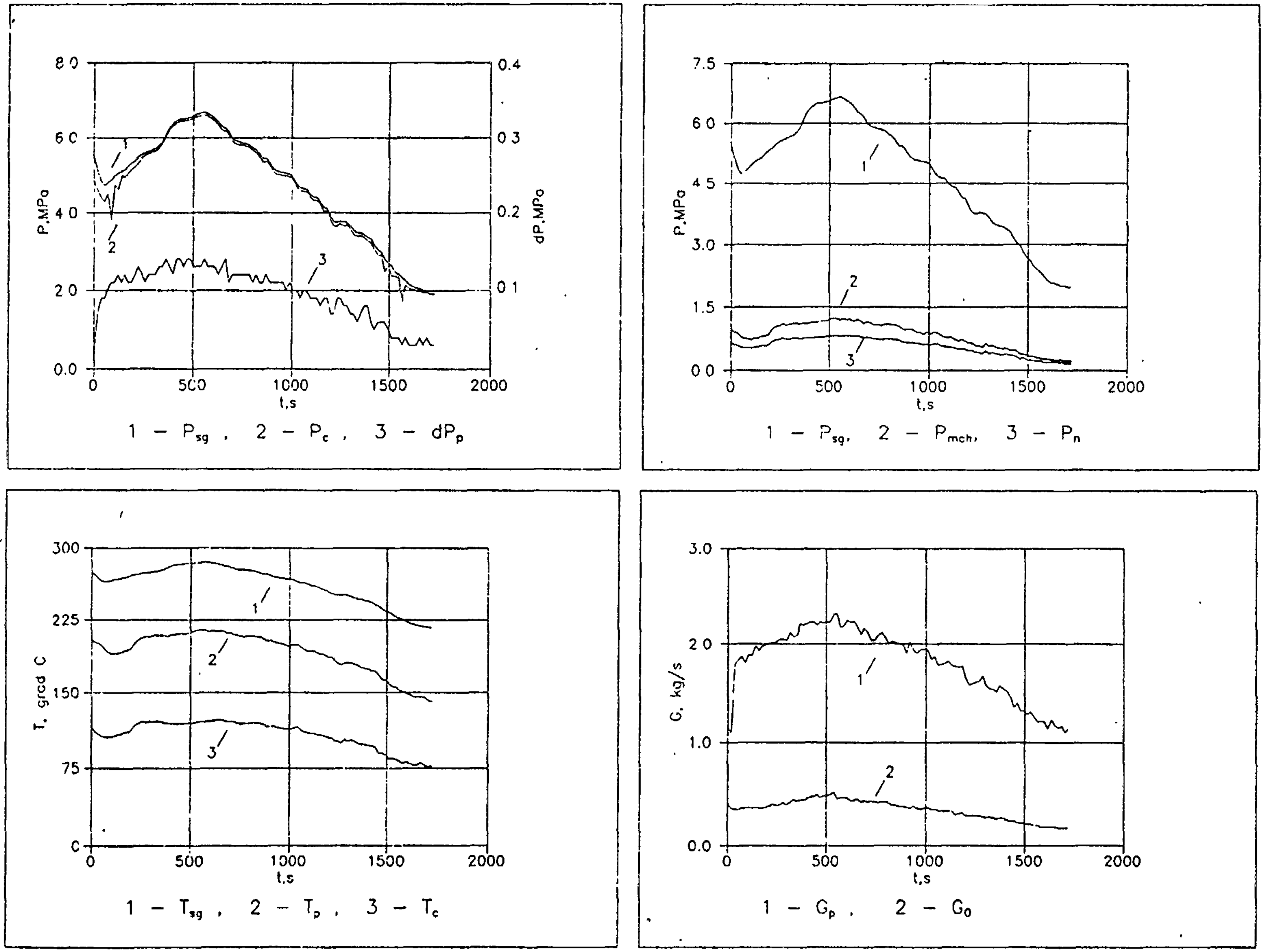

P, MPa
dP,mpa
t,s
1 – P_sg , 2 – P_c , 3 – dP_p
P, MPa
t,s
1 – P_n , 2 – P_ne, 3 – P_p, 4 – P_mch
T, grad C
t,s
1 – T_sg , 2 – T_c , 3 – T_p
G, kg/s
t,s
1 – G_c , 2 – G_p , 3 – G_0

Fig. 3.



P,MPa
dP,MPa
t,s
1 — P_{sg} , 2 — P_c , 3 — dP_p
P,MPa
t,s
1 — P_{sg}, 2 — P_{mch}, 3 — P_n
T, grad C
t,s
1 — T_{sg} , 2 — T_p , 3 — T_c
G, kg/s
t,s
1 — G_p , 2 — G_0 , 3 — G_c

Fig. 4.



P,MPa
dP,MPa
t,s
1 — P_sg , 2 — P_c , 3 — dP_p
1 — P_sg, 2 — P_mch, 3 — P_n
T, grad C
1 — T_sg , 2 — T_p , 3 — T_c
G, kg/s
1 — G_p , 2 — G_0

Fig. 5.

Integral model ("black-box").

It was determined that flow through PHRS, heat removed and other parameters depend mainly upon the pressure in the SG under condition that the water temperature in the pool is about 100°C. The model is valid under the conditions that $t > t_{ts}$, t_{ts} - transport time in the loop.

Model with distributed parameters.

This model carries out the usual calculations of the loop in which the new element IC is introduced. The IC is described in the steady state conditions on the basis of the parameters upstream of the IC, the model determines the parameters downstream of the IC.

Modeling of "rapid" processes in the PHRS loop is intended to analyze transient phase of filling tank, etc.

The first two models are used in the code TRAC-PF1/MOD2 for the simulation of accident situations at the NPP with the PHRS-IC.

Utilization of PHRS-IC for the VVER-440 NPP

The VNIIAES, EREC, GIDROPRESS and AEP Institutes have developed the technical requirements for PHRS IC the VVER-440 NPP, where it was shown, that:

- the system is compact and it could be easily incorporated into existing equipment;
- one can use the existing technology of the HE, piping, valves etc;
- the cost of the PHRS-IC is much lower than other safety systems having the same functions.

Calculations of the "station blackout" accidents for the VVER-440 have been conducted using the Code TRAC-PFl/Mod2. They showed that it is enough to have one PHRS-IC with power of 10 MWt in order to remove heat from one SG.

The behaviour of parameters in the steam generators of the VVER-440 NPP is shown in fig.6 when 4 PHRS-IC are put into operation. The pressure in the primary circuit dropped to 4 MPa after 4 x 10^4 s (11.1 hours).

The behavior of parameters when 2 PHrS-IC are put into operation is shown in fig.7. After 10^4s (2.8 hours) the system achieves a quasi-stationary regime when residual heat generation is equal to heat removal by the PHRS-IC.

The special project has been developed in EREC for testing the full-scale PHRS-IC for the VVER-440 NPP with a power of 10 MWt. A general view of this facility is presented in fig.8.

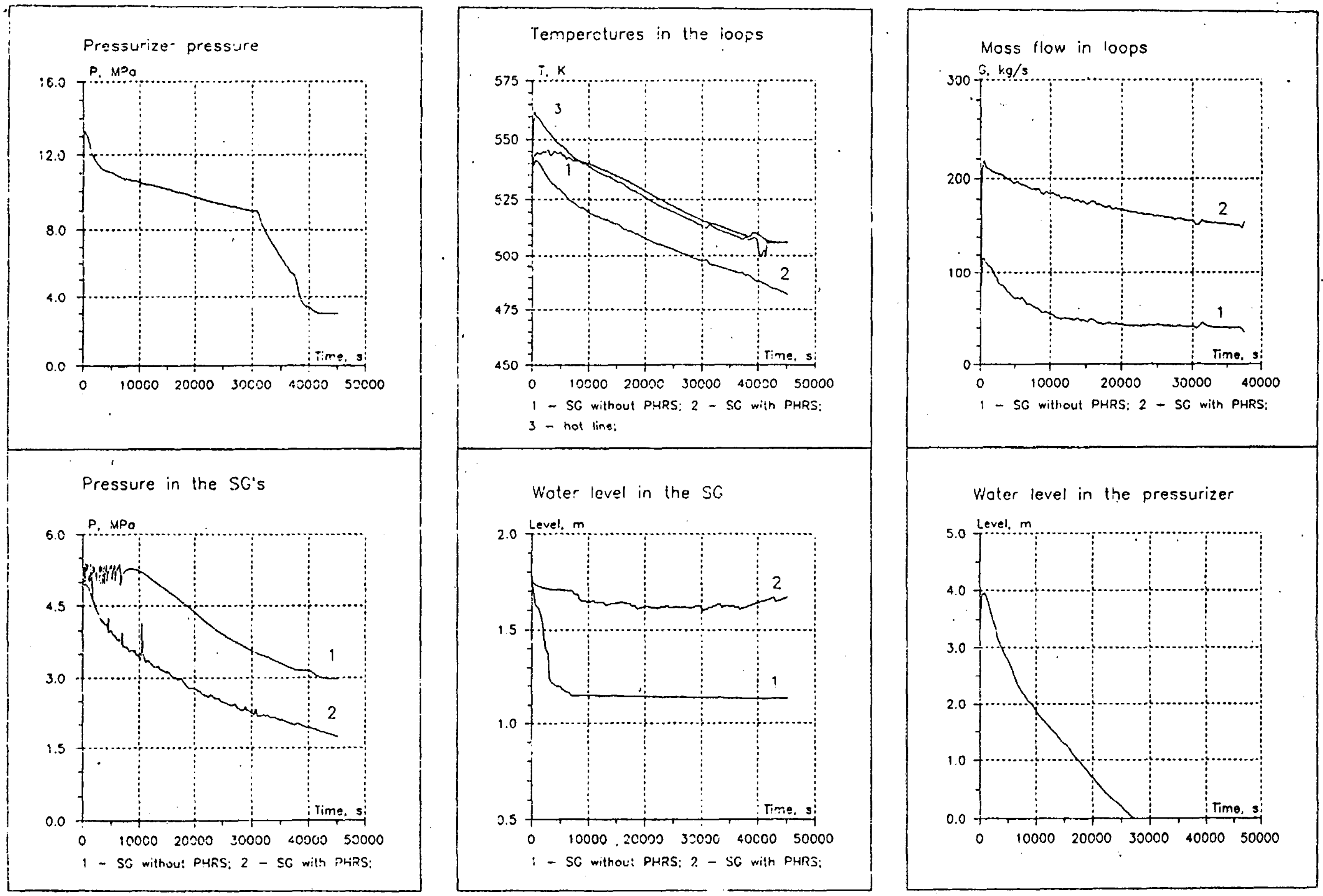


Fig. 6. VVER-440 BLACK OUT, 2 PHRS-IC-10 MWt

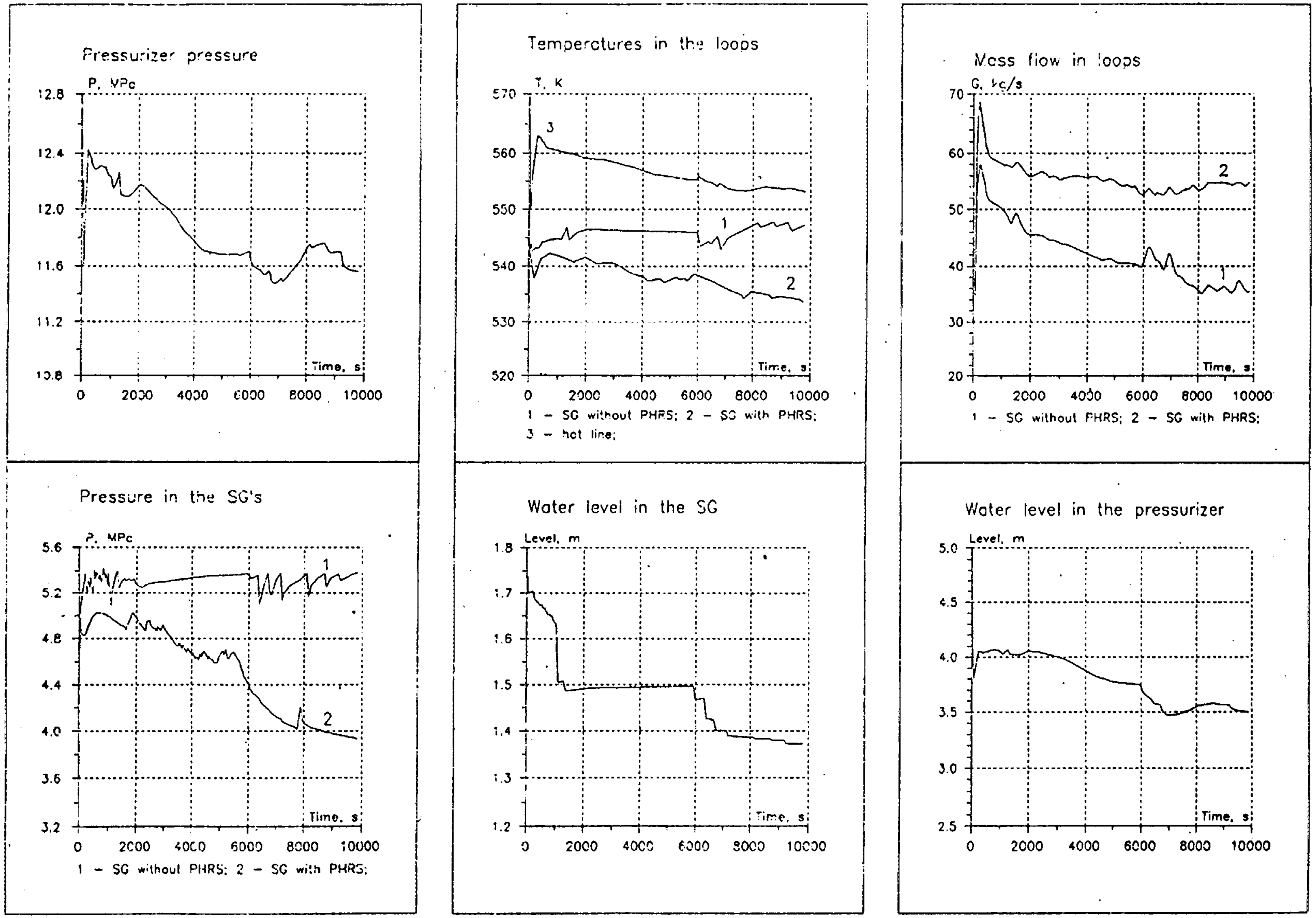


Fig. 7. VVER BLACK OUT 4 PHRS-IC-10 MWt

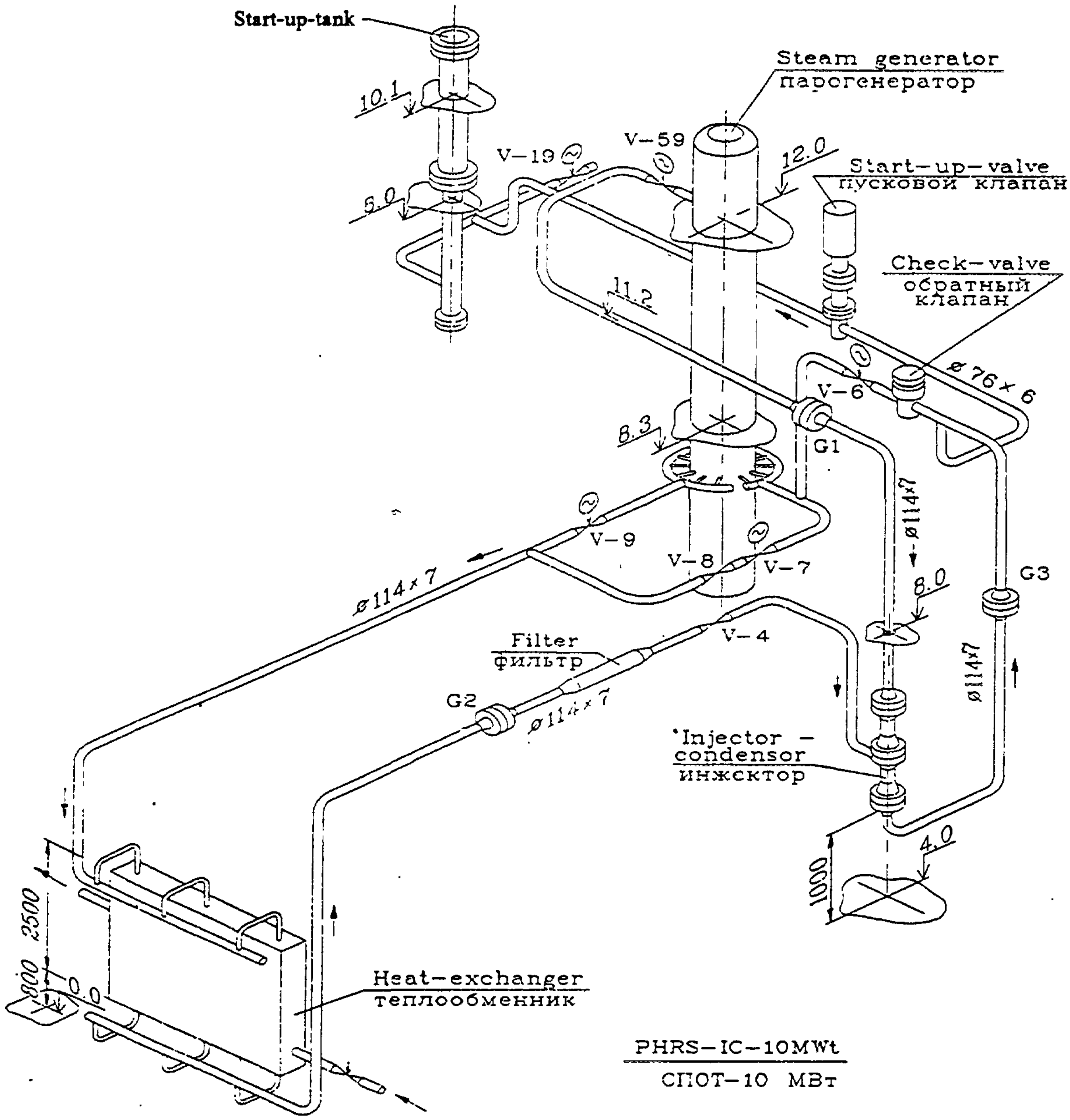


Fig. 8.

Usage of PHRS-IC

The PHRS-IC is a perspective safety system for the new generation of NPPs, since it solves the heat removal problem for various accident scenarios.

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APPENDIX I

ADDITIONAL PAPERS ON PASSIVE SAFETY

PASSIVE SAFETY SYSTEMS FOR INTEGRAL REACTORS

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Abstract

In this paper, a wide range of passive safety systems intended for use on integral reactors is considered. The operation of these systems relies on natural processes and does not require external power supplies. Using these systems, there is the possibility of preventing serious consequences for all classes of accidents including reactivity, loss-of-coolant and loss of heat sink as well as severe accidents.

Enhancement of safety system reliability has been achieved through the use of self-actuating devices, capable of providing passive initiation of protective and isolation systems, which respond immediately to variations in the physical parameters of the fluid in the reactor or in a guard vessel. For beyond design base accidents accompanied by complete loss of heat removal capability, autonomous self-actuated ERHR trains have been proposed. These trains are completely independent of the secondary loops and need no action to isolate them from the steam turbine plant.

Passive safety principles have been consistently implemented in AST-500, ATETS-200 and VPBER 600 which are new generation NPPs developed by OKBM. Their main characteristic is enhanced stability over a wide range of internal and external emergency initiators.

1. INTRODUCTION

The design of reactor plants with enhanced safety for the new generation of NPPs is one of the most important problems for the nuclear industry.

The solution of this problem is based principally on the design of reactors with inherent safety properties and deployment of passive safety systems. The basis of improved reactor plant designs developed in OKBM is the integral PWR characterized by simplicity and compactness of the primary circuit. Retaining all positive self-protection properties of PWRs, the integral reactor allows provision of further improvement in safety. This includes protection against the class of accidents most critical for PWRs, namely primary circuit loss-of-integrity. This protection has been established by exclusion of accidents with large and medium leaks. The large water inventory above the core provides a long time margin before core uncovery in accidents with primary pipeline breakage.

Decrease of the neutron fluence to the reactor vessel extends the useful life-time of the Reactor Pressure Vessel (RPV) to 60 years of operation. The integral design also eliminates the damaging influence of cold coolant on the reactor vessel during operation of powerful ECCS. Together with the inherent safety characteristics of the integral reactor, safety enhancement is achieved by passive safety systems, operating on the basis of natural processes without external power supply, and utilization of self-actuated devices to initiate them. Passive safety is applied for all kinds of accidents including accidents with positive reactivity insertion, accidents with loss of heat removal from the reactor, and accidents with primary circuit loss-of-integrity. A plant design with enhanced, perhaps maximised, safety was successfully implemented in the AST-500 nuclear district heating plant. Subsequently, the main critical design solutions for AST-500 safety, such as the integral design of the reactor and the various passive safety systems became the basis for the design of several other plants developed with natural and forced coolant circulation (e.g. ATETS-200, VPBER-600, etc.).

2. SYSTEMS FOR REACTOR EMERGENCY SHUTDOWN

2.1. A characteristic feature of the reactors developed by OBKM is the application of a highly effective mechanical system for reactivity control. This has been achieved by installation of control rod/assemblies in almost all the fuel assemblies of the core (Fig. 1). Reactor shutdown is performed by emergency protection signals leading to the dropping of control assemblies into the core following drive deenergization. In this case the reactor is transferred to a subcritical state with a reserve to allow for some control rod drive assemblies remaining stuck in a withdrawn position without the need for injection of boron solution. Residual Heat Removal Systems (RHRS) are also provided to bring the reactor system to a cold shutdown condition.

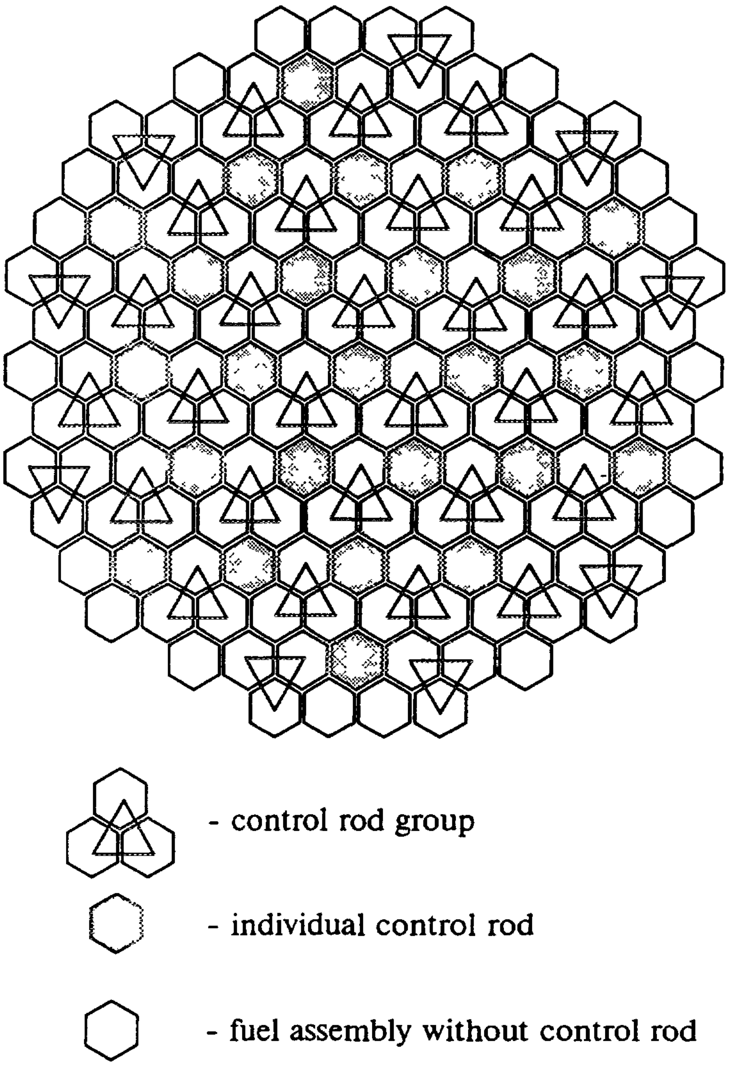


Fig. 1.
VPBER-600 Control Rods Layout

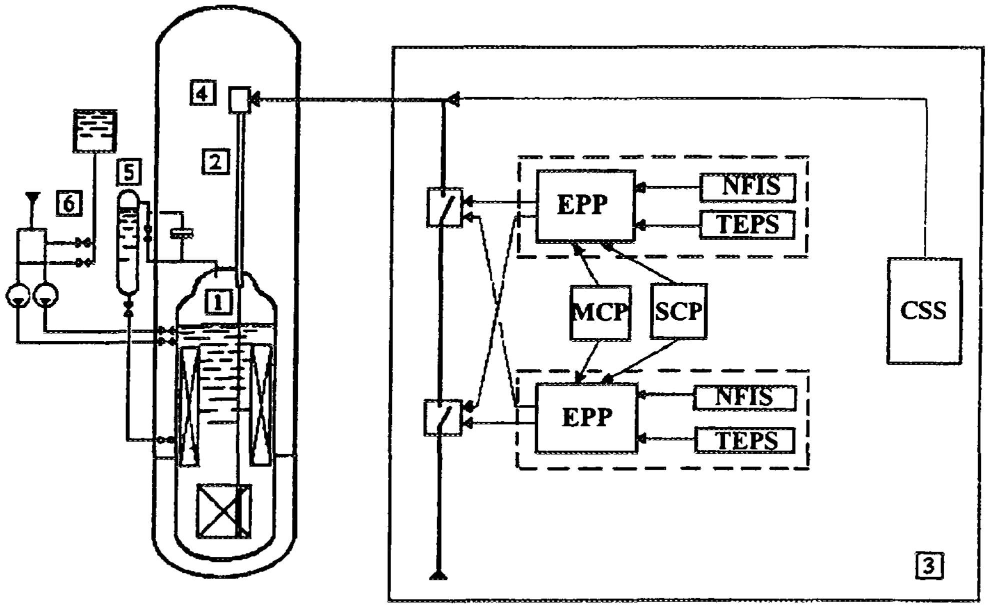
2.2. Together with automatic systems, self-actuated devices are used for deenergization of a sufficient number of Control Rod Drive Mechanisms (CRDM) to provide emergency protection. This actuation system is independent of the automatic system circuits; it responds to a pressure rise in the reactor or guard vessel (Fig.2).

Two types of pressure actuated power breakers (PAPB) are being developed: a PAPB built in as an integral part of the CRDM or a remote PAPB built into the reactor cover. The latter may deenergize a single CRDM or a group of CRDMs.

2.3. A passive system of boron emergency injection is intended for complete scram of the reactor core and maintaining it in a subcritical state in the case of the mechanical system malfunctioning (Fig.2). System actuation is performed by opening pneumatically operated valves on the pipelines connecting the boron solution tank and the reactor pressure vessel, or by a rupture disk actuated directly by a rise in the primary circuit pressure. Boron solution is supplied to the reactor by gravity due to the elevation of the boron solution tank above the reactor, once the pressures in the reactor and in the tank have been equalised.(Fig.2).

3. EMERGENCY HEAT REMOVAL SYSTEM

3.1. The AST-500 (AST-500M) emergency heat removal system will use the main heat exchanger loops for heat removal. Heat is removed in a three circuit scheme by natural coolant circulation and evaporation of water from designated tanks (Fig.3).



1 - Reactor
2 - CSS CRDM
3 - Control and safety system (CSS)
4 - Direct-acting device
5 - Passive channel for liquid boron injection
6 - Active channel for liquid boron injection

CSS - control rod controling system
NFIS - neutron flux instrumentation set
TEPS - technological equipment protection set
EPP - emergency protection panel
MCP - main control panel
SCP - standby control panel

Fig. 2.
Reactivity Control Means

An air heat exchanger is provided on one of the channels side by side with the water tanks. This ensures an unlimited period for residual heat removal without the need for power supply or water make-up (Fig.4).

When an accident occurs, the system is initiated by valves actuated by signals from the automatic control system or by direct reactor parameter effects (pressure, level).

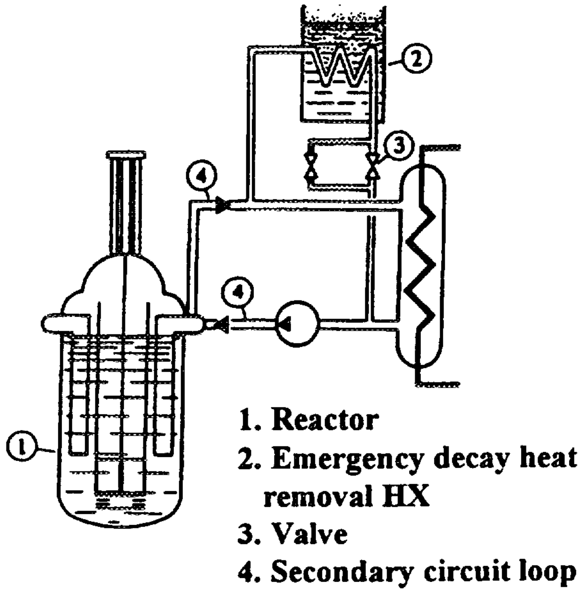


Fig.3

Emergency decay heat removal channel switched to the secondary circuit valves

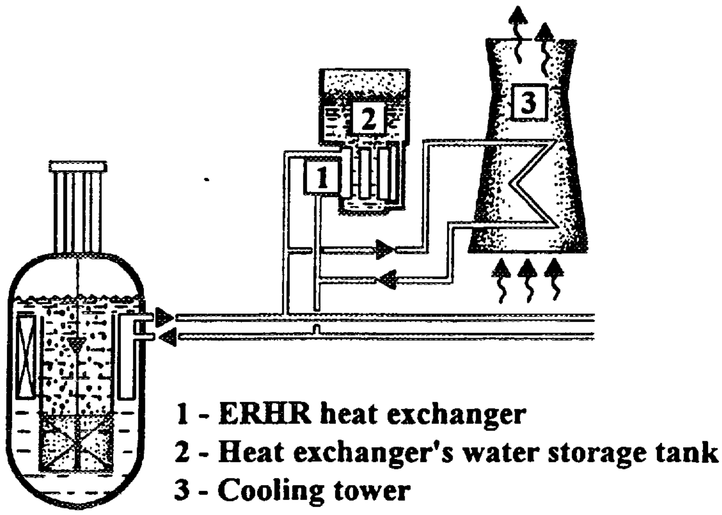


Fig.4

Emergency decay heat removal channel connected to the secondary circuit

3.2. In the ATETS-200 power reactor, heat is removed through the steam generator by natural circulation of coolant following secondary loop isolation from the steam-turbine plant. Secondary water circulates through the SGs and the RHR heat exchanger located in a water storage tank.

3.3. To exclude failure of heat removal capability in AST-500, a continuously operating heat removal system with an auxiliary heat exchanger situated in parallel with the main cooling heat exchanger is proposed. In this case the heat drawn by the auxiliary heat exchanger is used for the district heating network (Fig. 5).

3.4. Owing to the considerable water inventory in the AST-500 plant intermediate circuit, evaporation of intermediate circuit water through a relief valve (Fig.6) can be considered as an auxiliary system of emergency heat removal during beyond design base accidents.

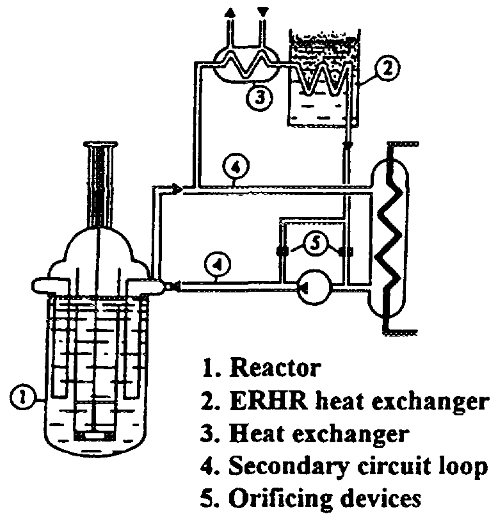


Fig.5

Emergency decay heat removal channel permanently connected to the secondary circuit

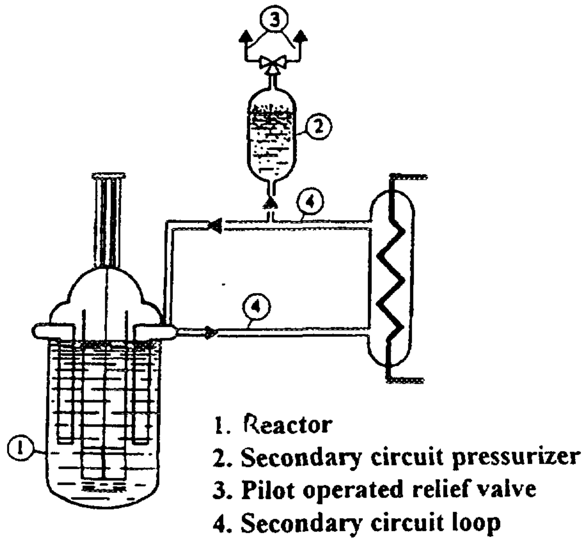
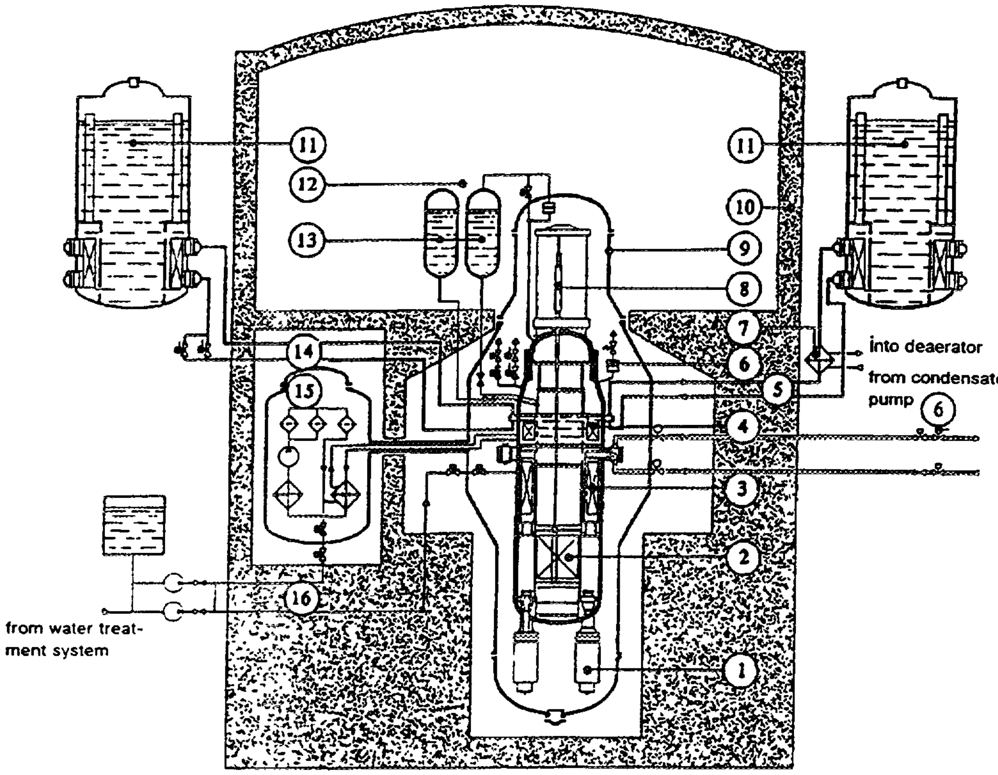


Fig.6

Emergency decay heat removal channel via the pilot operated relief valve in the secondary circuit

3.5. A passive emergency heat removal system considered for installation in VPBER-600, is independence of the secondary circuit and requires no isolation from the steam turbine plant. For this purpose, special emergency cooling heat exchangers are arranged above the SG level but below the primary water level (Fig.7). Heat is removed through an intermediate circuit by natural coolant circulation (Fig.8). Another continuous passive heat removal system is also being considered. In this case the heat drawn from the primary circuit by the emergency heat removal channel is used for heating the secondary circuit feed water (Fig.8).

3.6. An independent passive cooling system intended for beyond design accidents with complete loss of normal reactor heat removal capability is being considered for some integral reactors designs. A condenser-heat exchanger is arranged above the RPV head. Primary circuit heat is transferred through a double wall condenser-heat exchanger by natural circulation to a water inventory tank and removed to the atmosphere (Fig.9). The channel is self-actuated in response to primary pressure increase via a rupture disk.



1 Main circulating pump
2. Reactor
3. Steam generator
4. Heat exchanger - condenser
5. Continuous heat removal system
6. Self-actuating devices (direct action)
7. Intermediate heat exchanger
8. CRDM
9. Guard vessel
10. Containment
11. Heat exchangers unit
12. Emergency boron injection system
13. Tank with boron solution
14. Passive heat removal system
15. Coolant clean-up and boron reactivity control system
16. Primary circuit makeup system

Fig.7

VPBER-600 reactor plant flow diagram

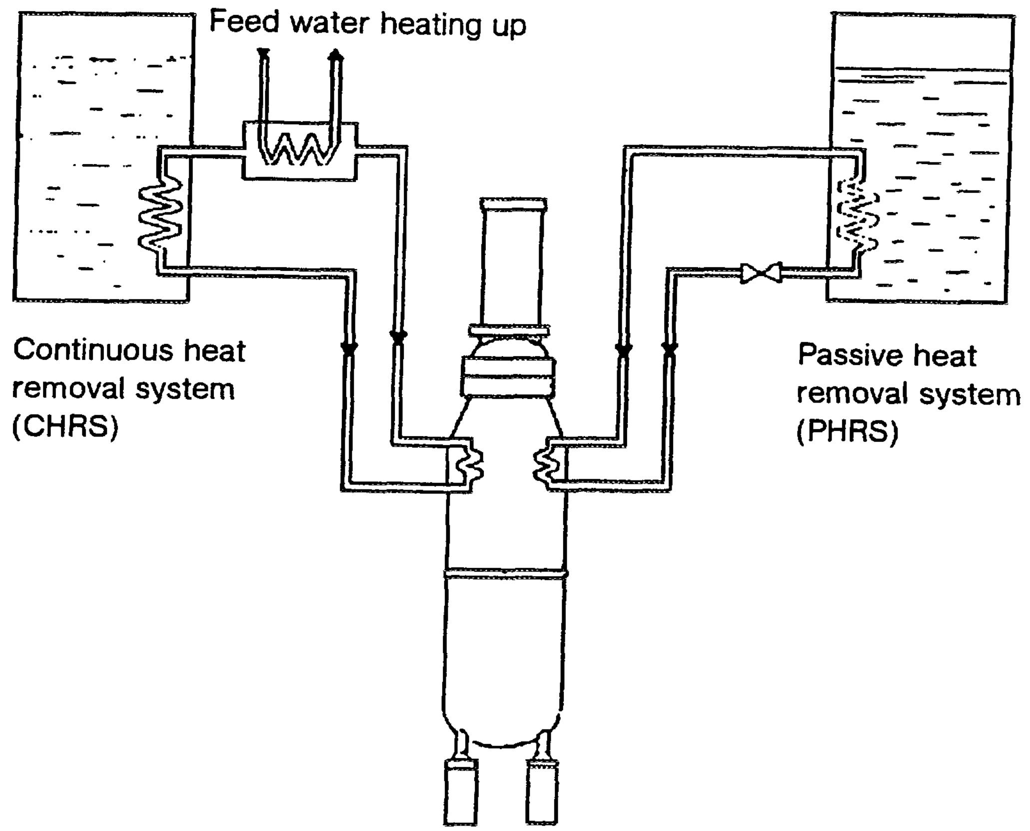


Fig. 8

Emergency residual heat removal system

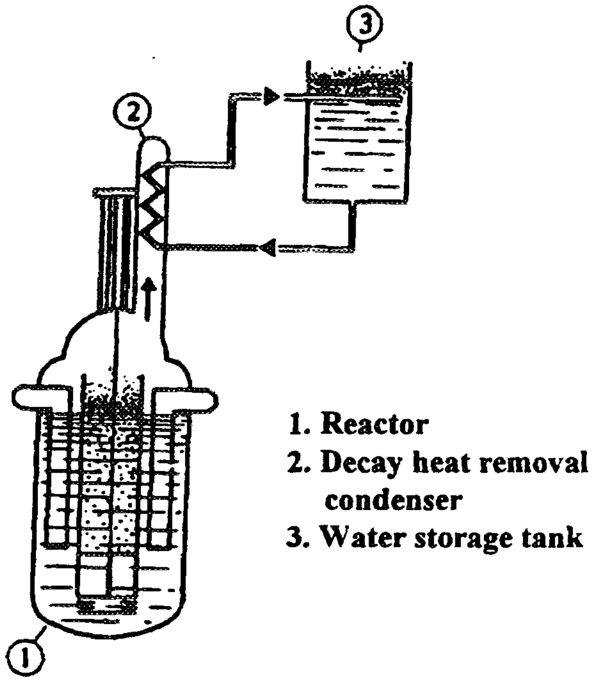


Fig. 9

Self-actuated ERHR channel on the reactor

4. ISOLATION SYSTEMS

4.1. Integral reactor compactness allows to locate the reactor in a strong-leaktight guard vessel (GV). The guard vessel is a passive protective and isolating device, ensuring safety in the case of primary pipeline rupture or reactor vessel loss-of-integrity (fig.10). Its design pressure is that expected to occur following a primary circuit loss-of-integrity. The GV prevents core uncovery and provides core cooling. It also provides for confinement of radioactive products. There is no need for active water injection systems for core cooling in emergency situations when a guard vessel is used, because the reactor core is kept covered by water.

The guard vessel performs an important function during severe accidents. When postulating complete core melting, corium confinement in the reactor vessel is provided by the guard vessel. Along with the integral reactor feature of decreased thermal load on the vessel, the presence of water in the guard vessel during primary circuit loss-of-integrity accidents ensures, from the very outset of the accident, external cooling of the reactor vessel. High efficiency of heat removal due to the relatively high pressure in a guard vessel is an important factor in mitigating the consequences of the accident.

4.2. For the three-circuit heat transfer scheme in AST-500, the intermediate circuit functions as a passive protection and isolation system ensuring retention of primary circuit radioactive product.

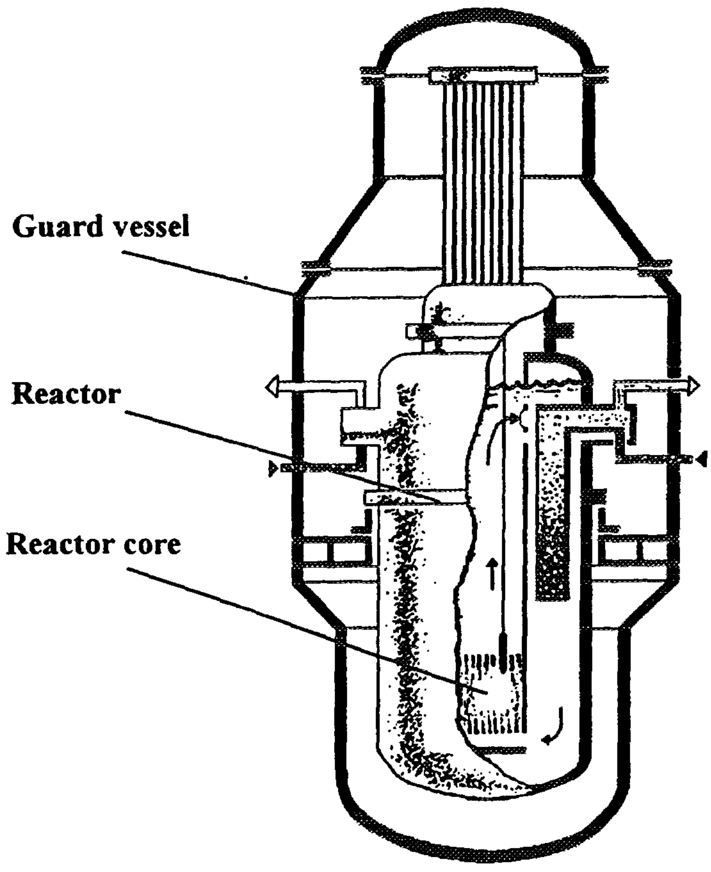


Fig.10

AST-500 Reactor

4.3. In the VPBER-600 reactor plant double isolation valves designed for primary circuit pressure are built in to each loop. One group of valves is activated by a signal from the automatic control system, the other one is actuated either by a signal from the automatic control system or passively as a result of low coolant level in the RPV.

5. MAIN FEATURES OF PASSIVE SAFETY SYSTEMS

The wide spectrum of passive safety systems presented in this paper allows their common main features and advantages to be identified.

- The operation of passive safety systems relays on natural processes and does not require supply of external energy. This ensures the reliability of the safety system in the condition of a station blackout for a long or unlimited time;
- Failure-free operation of passive systems relaxes the need for system redundancy, provides for simplification, and enhances the economics;
- An important benefit in the use of some passive systems is the possibility of checking during operation their efficiency and conformity to design performance requirements;
- Enhancement of system reliability is achieved by use of self-actuating devices leading to reliable operation of protective and isolation safety systems following any change in the reactor physical parameters;
- Use of passive safety systems gives effective protection against erroneous actions or personnel non-action and creates an additional protective barrier against sabotage;
- Specially designed passive systems or devices for prevention of failure of emergency reactor shutdown, prevention of over-pressurization of the reactor in the event of loss of all means for heat removal, and the ensured cooling of the reactor vessel from the outside practically excludes damaging severe accidents.

6. CONCLUSION

The safety concept of integral PWRs using natural or forced circulation of coolant (e.g. AST-500, ATETS-200 and VPBER-600 type) developed by OKBM for power plants of the new generation is based on the wide use of multi-purpose passive safety systems. This concept ensures, in principle, a higher level of safety, enhanced techno-economic indices and stability of nuclear power units in the case of severe accidents.

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CANDU SAFETY UNDER SEVERE ACCIDENTS

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Abstract

The characteristics of the CANDU reactor relevant to severe accidents are set first by the inherent properties of the design, and second by the Canadian safety/licensing approach.

The pressure–tube concept allows the separate, low–pressure, heavy–water moderator to act as a backup heat sink even if there is no water in the fuel channels. Should this also fail, the calandria shell itself can contain the debris, with heat being transferred to the water–filled shield tank around the core. Should the severe core damage sequence progress further, the shield tank and the concrete reactor vault significantly delay the challenge to containment. Furthermore, should core melt lead to containment overpressure, the containment behaviour is such that leaks through the concrete containment wall reduce the possibility of catastrophic structural failure.

The Canadian licensing philosophy requires that each accident, **together** with failure of each safety system in turn, be assessed (and specified dose limits met) as part of the design and licensing basis. In response, designers have provided CANDUs with two independent dedicated shutdown systems, and the likelihood of Anticipated Transients Without Scram is negligible.

Probabilistic safety assessment studies have been performed on operating CANDU plants, and on the 4 x 880 MW(e) Darlington station now under construction; furthermore a scoping risk assessment has been done for a CANDU 600 plant. They indicate that the summed severe core damage frequency is of the order of 5 x 10^{-6}/year.

CANDU nuclear plant designers and owner/operators share information and operational experience nationally and internationally through the CANDU Owners' Group

(COG). The research program generally emphasizes the unique aspects of the CANDU concept, such as heat removal through the moderator, but it has also contributed significantly to areas generic to most power reactors such as hydrogen combustion, containment failure modes, fission product chemistry, and high temperature fuel behaviour.

Abnormal plant operating procedures are aimed at first using event–specific emergency operating procedures, in cases where the event can be diagnosed. If this is not possible, generic procedures are followed to control Critical Safety Parameters and manage the accident. Similarly, the on–site contingency plans include a generic plan covering overall plant response strategy, and a specific plan covering each category of contingency.

1. NATIONAL CONTEXT

1.1 STRUCTURE OF THE CANADIAN NUCLEAR INDUSTRY

The CANDU nuclear power concept was realized in 1962 with the first operation of the Nuclear Power Demonstration (NPD) prototype, a co–operative venture between the designer, Atomic Energy of Canada Limited (a federal government Crown Corporation), and the operator, Ontario Hydro (a provincial government utility). Now there are twenty seven operating CANDU reactors worldwide, with eleven more under construction in Canada, Romania, and India. Ontario Hydro owns and operates 18 Canadian reactors (with two under construction), and now performs both plant design and safety assessment. Two more Canadian provincial utilities own and operate CANDUs – Hydro Québec and the New Brunswick Electric Power Commission. Atomic Energy of Canada Limited (AECL) has evolved into a number of related companies, including a design company, AECL-CANDU, and a Research Company (AECL–Research) which performs both fundamental and applied research in support of the CANDU concept.

All of these organizations, plus owners of CANDU plants in Korea and Argentina, coordinate their needs and exchange information through the CANDU Owners' Group, or COG. The safety analysis to support the CANDU is performed both at the owner/utility and at AECL–CANDU.

Nuclear power is regulated in Canada by a federal government agency, the Atomic Energy Control Board (AECB). Each provincial government has the responsibility for offsite emergency planning within its borders.

1.2 DEFINITION OF A SEVERE ACCIDENT

A severe accident is defined as one in which the fuel heat is **not** removed by the coolant in the primary heat transport system (PHTS). In most other reactor designs, this is equivalent to a core melt, and indeed severe core damage (defined as loss of core structural integrity) is one end of the spectrum in CANDU. However the inherent characteristics of CANDU provide a broad spectrum of scenarios where, even if primary and emergency cooling are lost, the fuel does not melt.

1.3 REPORT OUTLINE

The characteristics of the CANDU reactor relevant to severe accidents are set first, by the inherent properties of the design, and second, by the Canadian safety/licensing approach. For a basic introduction to the safety approach, see [Snell, 1985].

In Section 2, the evolution of the licensing approach is described, and in Section 3, the design aspects related to severe accidents are summarized. The characteristics of severe accidents themselves, in terms of frequencies and consequences, are summarized in Section 4.

The Canadian research programme which supports the conclusions reached in severe accident analysis is presented in Section 5. The operating philosophy and procedures relevant to arresting and mitigating severe accidents are described in Section 6.

Hereafter the features of a "typical" CANDU reactor are described. Most of the conclusions reached are generic.

This report is a more detailed version of an invited paper presented at "The International Symposium on Severe Accidents in Nuclear Power Plants", in Sorrento, Italy, March 21–25, 1988 [Snell, 1988].

2. CANADIAN SAFETY AND LICENSING APPROACH

2.1 EARLY PROBABILISTIC DEVELOPMENTS

The safety/licensing approach in Canada has had a probabilistic component going as far back as the 1950s (see [Snell, 1986] for a review). The accident which damaged the NRX research reactor at Chalk River, Ontario, in 1952, was caused by a failure of a normal process system (a process system is defined as any system required for the normal operation of the plant) and a partial failure of the protective system (shutdown) designed to protect against the fault ([Lewis, 1953], [Hurst, 1953]). This spurred an early interest in the frequency of accidents, and in the reliability and independence of the protective systems. These ideas led [Siddall, 1959] to definition of frequency targets for failures of the process systems, and reliability targets for the protective systems, which had to be demonstrated in practice.

A severe accident could only result if a process system failed **and** the appropriate protective system was simultaneously unavailable. With the systems sufficiently independent, the frequency of a severe accident could be made acceptably low. This philosophy of **separation** of process and safety systems, and **verifiable reliability** targets for each, became the hallmark of Canadian safety and licensing [Laurence, 1961].

These ideas were expanded with Canada's first power reactors – the 25 MW(e) Nuclear Power Demonstration reactor and the 208 MW(e) Douglas Point prototype. A safety goal was defined for risk of death to any member of the public. This goal was used to establish the reliability requirements for the process systems, the shutdown systems, the emergency coolant injection system, and containment (the latter three were later called the special safety systems). The possible combinations of process failures with or without safety system action formed an accident **matrix** which included a number of severe accidents; these were analyzed with the tools of the day as part of the licensing submission.

2.2 SINGLE/DUAL FAILURE MATRIX

With the construction of the four–unit Pickering–A nuclear generating station near Toronto, the AECB licensing guidelines, while retaining a probabilistic component, became more deterministic. All currently operating plants up to Darlington have been since licensed under these rules [Hurst, 1972]. The spectrum of possible accidents is divided into two broad categories – **single failures**, or the failure of a process system which requires the intervention of one or more of the special safety systems; and **dual failures**, a much less likely event defined as a single failure coupled with the assumed unavailability of one of the special safety systems. (The single failure is an assumed system failure and is not related to the same term used elsewhere to describe a random component failure additional to the initiating event.) For each class, designers had to demonstrate that specified frequency and consequence targets were met (Table 2.1). For example, the spectrum of dual failures included loss–of–coolant plus failure of the emergency coolant injection system, and loss–of–coolant plus failure of containment ventilation to isolate. For each of these, the maximum individual whole–body dose to the critical individual at the site boundary had to be demonstrated to be less than 0.25 Sv, and the maximum collective dose in the surrounding population had to be shown to be less than 10^4 person–Sv, under pessimistic atmospheric dispersion conditions. There were additional limitations on thyroid dose, as shown in Table 2.1. The frequency targets are not **expected** frequencies – they were chosen large enough that compliance could be demonstrated (from direct observation of single failure frequency, and from safety system reliability in periodic testing) in a few years of actual station operation.

TABLE 2.1 - DOSE/FREQUENCY GUIDELINES

ACCIDENT	MAXIMUM FREQUENCY	INDIVIDUAL DOSE LIMIT	POPULATION DOSE LIMIT
Single	1 per 3 years	0.005 Sv wb	10^2 person–Sv
Failure		0.03 Sv thy	10^2 thy–Sv
Dual	1 per 3000 years	0.25 Sv wb	10^4 person–Sv
Failure		2.5 Sv thy	10^4 thy–Sv

Note: wb=whole body; thy=thyroid.

The special safety systems are designed and operated to:

1) demonstrate during operation, by test, a dormant unavailability no greater than 10^{-3}, or about eight hours per year;
2) be physically and functionally separated from the normal process systems and from one another. For example each shutdown system has its own detectors, amplifiers, logic relays, and actuating mechanisms, which cannot be shared with either the other shutdown system, the other special safety systems, or the reactor control system; which are of generally diverse design and manufacture; and which where practical are located in physically separated areas of the plant. During the design a stringent peer review, design review and QA programme is applied to ensure, among other things, that separation standards are met.

2.3 THE C–6 FIVE–CLASS APPROACH

The latest CANDU in Canada, the 4 x 880 MW(e) Darlington Nuclear Generating Station, was licensed using the approach described in AECB document C–6 [AECB, 1980]. C–6 retains the deterministic elements of its predecessor, the single/dual failure matrix, but recognizes that describing the spectrum of accidents by only two classes is too limiting: the number has been expanded to five, with consequence targets set for each class, and accidents assigned to classes implicitly according to frequency.

C–6 represents another step in the continuous tradition, in the licensing approach in Canada, of designing for a number of classes of severe accidents, and limiting their predicted consequences.

2.4 2.4 TWO–GROUP PHILOSOPHY

For common–cause events such as earthquakes, fires, and missiles, Canada has developed the two–group approach. All important systems in the plant are divided into two spatially separated and independent groups, either of which can, by itself, shut the plant down, remove decay heat, and monitor the plant safety status. Physical protection or environmental qualification is provided so that at least one group will be available when required – e.g., a protected secondary control area is provided in case the main control area becomes uninhabitable due to an earthquake or fire. This approach reduces the chance of severe core damage due to common–cause initiators.

3. DESIGN ASPECTS RELATED TO SEVERE CORE DAMAGE

3.1 THE CANDU CONCEPT

The CANDU is a natural uranium fuelled, heavy water moderated, heavy water cooled reactor. The pressure–boundary in the core consists of several hundred 10 cm. diameter pressure tubes, each containing twelve or thirteen short (0.5m) fuel bundles. Surrounding each 0.44 cm. thick pressure tube (Fig. 3.1) is a 0.14 cm. thick calandria tube; between the calandria tube and the pressure tube is an insulating gas–filled gap, which reduces normal heat loss to the moderator.

3.2 PRESSURE TUBES AS PRESSURE BOUNDARY

All the reactivity devices (control rods, shutoff rods) penetrate the moderator but not the coolant pressure boundary, and so, with one exception, are not subject to hydraulic forces from a loss–of–coolant accident (LOCA). The exception is a channel failure, which is a small break LOCA. The calandria tube may or may not contain the pressure tube rupture, but no credit is taken for this in the design, and relief pipes, sized for channel failure, are provided to protect the calandria vessel (which contains the moderator) from overpressure. It has also been shown ([Ross–Ross, 1963], [Muzumdar, 1987]) that a channel failure will not propagate to other pressure tubes, nor will it damage more than the neighbouring few shutoff rod guide tubes. The shutoff rod shutdown system is therefore provided with enough redundant mechanisms (typically 28) to remain effective without credit for the rods with possibly damaged guide tubes, nor for the most effective rod among those with undamaged guide tubes.

3.3 DECAY HEAT REMOVAL

There are a number of emergency heat sinks for decay heat from the fuel, which can prevent or mitigate a severe accident.

3.3.1 Shutdown Cooling System

A shutdown cooling system is provided in all CANDUs for normal decay heat removal (as an alternative to the boilers) and for cooldown below 100C. It can operate at full heat transport system temperature and pressure, and can therefore be used as an emergency heat sink from hot, shutdown, full–pressure conditions, should the boilers be unavailable.

3.3.2 Emergency Water System

Most CANDUs also have an Emergency Water System (EWS), seismically qualified to remove heat after a Design Basis Earthquake. It provides a supply of water to the boilers independent of the normal and auxiliary feedwater. Since it is sized to remove decay heat shortly after shutdown, it can also be used as an emergency heat sink should normal and auxiliary feedwater, and the shutdown cooling system, all be unavailable.

3.3.3 Emergency Coolant Injection

As in all water–cooled reactors, an emergency coolant injection system refills the fuel channels following a loss–of–coolant accident. To ensure that injection for small breaks is not blocked by high Primary Heat Transport System pressure, the main steam safety valves on the main steam lines are opened on a loss–of–coolant signal. These are sized to cool down the boiler secondary side, thereby depressurizing the PHTS. Emergency coolant injection thereafter is conventional recovery of water from the reactor building sump, and re–injection.

3.3.4 Moderator

In normal operation, about 5% of the thermal energy produced by the fuel is deposited into the moderator, by radiation and direct nuclear heating and, to a much smaller extent, by conduction through the insulating gas gap of the pressure–tube/calandria–tube assembly. This heat is removed by a moderator cooling system, consisting of pumps and heat exchangers. In a severe loss–of–coolant accident, the same system will remove decay heat from the fuel channels, even if they contain no coolant at all. Fuel would be severely damaged, but would not melt, and the channel would remain intact and contain the debris. This capability has been verified by full–diameter channel tests at the Whiteshell Research Laboratory, as described in Section 5. The moderator is thus a distributed low–pressure emergency heat sink surrounding each fuel channel.

3.3.5 Calandria and End–Shield Cooling System

In normal operation, heat is generated in the calandria shell and in the end shields which support the fuel channels and which provide radiation shielding for the reactor vault in front of the reactor faces. This heat, amounting to about 0.3% of the full–power heat generation, is removed by a dedicated shield cooling system. In addition, the cylindrical calandria shell is located inside either a metal shield tank, or a concrete vault, either of which is filled with water to provide both cooling and radiation shielding (Fig. 3.2). The vault floor itself is typically 2 1/2 m. thick. Should the emergency coolant injection system, and the moderator heat sink be lost after a loss–of–coolant accident, the shield cooling system can, depending on the failure sequence, prevent melt–through of the calandria vault or shield tank, or delay it for many hours as the shield water is boiled away. The analysis which supports this conclusion is discussed in Section 4.3.

3.4 3.4 CONTAINMENT

CANDU reactors use two types of containment (Fig. 3.3): single–unit containment, at the CANDU 6 nuclear generating stations, and multi–unit vacuum containment, at the Ontario Hydro nuclear generating stations. Both use a high–rate water spray, called dousing, which condenses steam released in an accident and reduces the containment pressure. In the single–unit containment, the reactor and the dousing system are all located in the same building. In the vacuum system, four or eight reactors, each with its own local containment, are connected by large ducting to a separate common vacuum building kept, as its name implies, at near–zero absolute pressure. Should steam be released from a pipe break in the reactor building, it, and any radioactivity, is sucked along the duct and condensed by dousing in the vacuum building. Long–term pressure control is by local containment air coolers, and by a filtered air discharge system.

The dual failure "loss of coolant plus loss of emergency coolant injection", while it does not lead to a loss of core geometry, nevertheless permits the fuel to reach high temperatures. The Zircaloy fuel sheaths can be highly oxidized, and the hydrogen gas which evolves will make its way through the break to containment. The control of hydrogen following severe accidents depends on the station design. In the vacuum containments, a network of hydrogen ignitors, powered by the most reliable source of electricity (Class I batteries), is engineered to reduce local flammable hydrogen concentrations before they can reach explosive conditions, and to ensure that the energy from combustion is released gradually. Alternatively, the natural circulation through the reactor vault can be accounted for. In particular, for the single unit containment at Point Lepreau, the accident analysis for loss of coolant plus loss of emergency coolant injection (LOCA/LOECI) predicts that flammable concentrations (in excess of 4% hydrogen) of the mixed containment atmosphere are not reached. Should a flammable concentration nevertheless occur, the analysis also shows that the containment would not be damaged by the pressures generated by the burn. The reason for the low hydrogen concentration, even for LOCA/LOECI, is that when the moderator acts as an emergency heat sink, it limits the pressure tube temperatures below the level at which significant oxidation can occur. Thus the pressure tube metal does not contribute significantly to the hydrogen source term. This is discussed further in Section 4.

3.5 REACTIVITY CONTROL AND SHUTDOWN SYSTEMS

The reactivity control devices penetrate the low–pressure moderator but not the coolant pressure boundary, as noted earlier, so they are not subject to pressure–assisted ejection (a channel failure is too small a break to develop enough pressure within the calandria to delay the rods significantly). The maximum rate of reactivity addition from the control devices is set by their inherent mechanical or hydraulic operation – normally this is 0.1 mk/sec (1 mk or "milli–k" is about 1/6 beta or 16 cents), and at most it is about 1 mk/sec during shutoff rod withdrawal from a shutdown state. The total reactivity holdup in the movable control devices is about 15 mk. This low value is set not by the need to compensate excess reactivity, but by operational requirements on decision and action time after a reactor trip. The pressure–tube, natural uranium concept permits on–power refuelling as the longer–term means of reactivity control.

CANDU stations control reactor power automatically over the entire range from 6 or 7 decades below full power up to full power. At low powers, up to about 10% full power, power measurement is based on ion chambers, while at high powers, in–core flux detectors are used. Both types of measurement are sufficiently prompt for all practical purposes.

Reactivity control at all power levels, both for bulk and for spatial purposes (spatial control is needed only above 25% power), is based on water–filled zone controllers. If their worth is inadequate, mechanical control absorber and adjuster rods are available for both positive and negative reactivity addition, again under totally automatic control. There is also poison addition to, and removal from, the D_2O moderator, both of which are very slow and relatively rarely required.

Protection against reactivity insertion accidents is provided both by the control system itself, via power stepbacks on high rate log and high flux, and also by powerful, rapid shutdown – see [Snell, 1986 December], [Snell, 1987], and [Howieson, 1987] for more detail. In the CANDU 6 for example, shutdown system 1 consists of 28 gravity–operated, spring–assisted absorber (shutoff) rods, and shutdown system 2 consists of 6 nozzles which inject liquid gadolinium nitrate, at high pressure, into the moderator. Each system is, independently, fully capable of shutting down the reactor for all accidents. Each system has its own detectors, amplifiers, relays, logic, and actuating mechanisms, and is independent of the control system and of the other shutdown system. Because the shutoff units are inserted into the low–pressure liquid moderator, they can respond very quickly to an accident.

In particular, the trip parameters on each system are chosen to provide redundant coverage, where practical, for every accident in the design/licensing set. These trips have been studied extensively in terms of their trip coverage (i.e., the range of initial power level and process conditions for which the trips are effective), and the combined reactor trip coverage is found to be fully comprehensive.

The fastest means of reactivity insertion is through the large loss of coolant accident. Thermohydraulic effects limit the rate to less than 4 mk a second for the worst break. For a large LOCA, the initial rate of rise of reactor power is 50–100%/second. These rates determine the speed of the shutdown safety systems, which must therefore act within about two seconds for the most severe LOCA, a rate achievable with mechanical or hydraulic devices. For hypothetical reactivity insertions even **above** prompt critical, the rate of power increase is set by the longer prompt neutron lifetime of CANDU (at least ten times that of light water power reactors). Thus the rate of rise of power is not very sensitive to going beyond prompt criticality in CANDU.

All recent CANDU reactors have two fully independent shutdown systems, either of which can, by itself, terminate any reactivity insertion accident or LOCA. The provision of dual, fast, independent shutdown systems means that, for these reactors, Anticipated Transients Without Scram, including LOCA, are low enough in probability that they can be ignored for design purposes, as they are a negligible contribution to total risk.

4. SEVERE ACCIDENTS

4.1 PROBABILISTIC SAFETY ASSESSMENTS

In Canada, the nuclear industry (designers and plant owners/operators) has taken the lead in performing probabilistic safety assessments (PSAs). From the early use of risk to define design requirements, the PSAs evolved in the 1970s to fault tree/event tree analyses, which were used to confirm the reliability and separation goals specified for the design.

The first such study was performed by AECL in 1975 on the service water system at Ontario Hydro's Bruce–A Generating Station. The benefits from this study were:

a. a comprehensive identification of crosslinks, since service water has interfaces with many systems;

b. identification of which support functions needed backup cooling water; and

c. definition of the necessary operator actions to mitigate the loss of service water.

PSA studies (called Safety Design Matrices at the time) were performed from 1978 to 1983 on the CANDU plants in the design and construction phase during that period ([Gumley, 1985], [Shapiro, 1986]).

The regulatory agency also recognized the usefulness of the studies for the Bruce–A Nuclear Generating Station, and the PSA studies became part of the licensing process for all CANDU stations constructed in Canada after Bruce–A.

These PSAs went beyond the assessment of support systems, and were used to systematically identify and quantify all major sequences that could release radiation from the plant at credible event frequencies [Rennick, 1987].

4.1.1 Darlington Probabilistic Safety Evaluation

The Darlington Probabilistic Safety Evaluation (DPSE) [Ontario Hydro, 1987] was carried out by Ontario Hydro as part of the design verification process for the 4 x 880 MW(e) Darlington station, during the final stages of station design and construction. The study is essentially equivalent to a Level 3 Probabilistic Risk Assessment (PRA); however, no detailed off–site consequence analysis was performed for certain low probability severe accidents. The scope of the study in addition did not include initiating events external to the station nor those arising from internal fires and flooding, which were judged unlikely to contribute significantly to risk because of rigorous, deterministic design criteria, as discussed in Section 2.4.

Important features of the DPSE include a very comprehensive set of initiating events internal to the plant and a high degree of detail in the fault tree analysis of all major process, safety and support systems, including specific modelling of potential instrumentation and control failures, e.g., [Raina, 1986]. Human reliability modelling was likewise comprehensive, including detailed identification of human error opportunities both prior to and after the initiating event. Preliminary human reliability quantification was performed using prepared data tables [Iwasa–Madge, 1985]. Final quantification of important human error probabilities was obtained by using a structured expert judgment procedure involving plant operations staff.

The spectrum of potential core damage was divided into ten fuel damage categories (FDCs), labelled FDC0 to FDC9. Of these, FDC0 to FDC3 cover the range of events considered to meet the severe accident definition in this report. FDC1 to FDC3 deal largely with loss–of–coolant initiating events accompanied by failure of emergency core cooling, either on demand or during the mission time, in which the moderator is called upon to act as the heat sink. FDC0 contains all events with the potential to cause a loss of core structural integrity. This can occur due to the failure of the moderator to act as a heat sink when required, failure to shut down (if such failure would result in fuel damage), or severe over–stressing of the calandria structure.

The magnitude of fuel damage associated with FDC3 is quite small and largely represents an economic, rather than public health, risk. FDC2 results in significant fuel damage, and FDC1 is conservatively estimated to result in 15–30% of the core equilibrium fission product inventory being released from the fuel. FDC0 could result in a greater or smaller release, depending on the nature of the mechanism causing loss of core structural integrity.

The frequency estimates are the result of complete computer–assisted integration of the event trees and fault trees, fully accounting for system crosslinks. Due to the level of detail, the development of special methods and procedures was needed in order to simplify the fault trees and structure the integration process to make it computationally feasible ([Chan, 1987], [King, 1987]). Table 4.1 contains the DPSE mean frequency estimates for the severe accident categories.

TABLE 4.1: DARLINGTON SEVERE ACCIDENT FREQUENCY ESTIMATES

Category	Description	Mean Frequency (/reactor–yr)
FDC3	Moderator required as a heat sink more than 1 hour after reactor trip	6×10^{-4}
FDC2	Moderator required as a heat sink from 200s to 1 hour after reactor trip	8×10^{-5}
FDC1	Moderator required as a heat sink less than 200s after reactor trip	2×10^{-6}
FDC0	Potential for loss of core structural integrity	4×10^{-6}

The severe core damage frequency is bounded by the frequency for FDC0, and at 4×10^{-6}/reactor–year, it is very low indeed.

The complex multi–unit CANDU containment includes, as described earlier, a negative–pressure vacuum system and an emergency filtered air discharge system. The containment event trees include failures of: overpressure suppression, envelope integrity, long–term pressure control and filtration [Dinnie, 1986]. Fuel damage and containment failure logic were fully integrated to search for potential crosslinks. Consequences were estimated representative of a wide range of fuel damage category and containment subsystem failure combinations. The results include an estimate of the frequency of a large release from the core accompanied by the potential for loss of the containment function, leading to the possibility of a large, offsite release. The mean frequency was estimated to be 8 x 10^{-7}/reactor–year.

The overall conclusion is that the calculated health risk is very low.

4.1.2 Probabilistic Risk Assessment of CANDU 6

In 1986 and 1987, AECL performed a CANDU Level 2 Probabilistic Risk Assessment study of CANDU 6, because some countries have expressed a need to be able to compare the overall safety of reactors as a factor in making a choice of a reactor option. This involved a probabilistic evaluation of events of a frequency of less than 10^{-7}/year, and a consequence analysis of severe core damage events and related releases [Howieson, 1988].

In performing the probabilistic evaluation, the following groundrules were used:

1. The reference plant was an existing Canadian CANDU 6 unit, licensed for operation in 1983, with the addition of automatic cooldown of the heat transport system on high end–shield temperature. The required licensing, operating, and design information was readily available.
2. As assumed in water reactor probabilistic risk assessments, the initial plant state was 100% full power operation. External events such as earthquakes and fires were not assessed in this study, although they of course are covered deterministically in the design, as described in Section 2.4.
3. The reactor core contains equilibrium fuel.

Previous probabilistic safety assessments of CANDU proved to be valuable input to the CANDU 6 PRA.

Fault trees were used to determine the frequency of the initiating events, and the failure probability of the mitigating systems. Event trees were used to assess the plant response following the initiating event, and incorporated the possibility of failures of the required mitigating systems.

In the preparation of the event trees, crosslinks were identified between systems, up to the level of major components and electrical power supplies; however, crosslinks between systems via control components (e.g. contacts, relays, and fuses) were not examined.

The operator model used was the same as was used in the previous CANDU 6 Probabilistic Safety Assessment studies. The PSA operator model is a post–initiating–event model, in which operator actions are shown explicitly in the PSA event sequence diagrams. This approach was carried over, and the operator actions were explicitly shown in the event trees. A brief comparison was made, where appropriate, to other operator models.

This preliminary CANDU 6 study was sufficient to identify the major risk contributions with a high degree of confidence because:

1. The detailed design is "known" and has already been subject to probabilistic safety analyses; in addition, the reference plant has been running successfully for the last five years.
2. The CANDU design philosophy calls for independence of safety systems and process systems. This approach minimizes the potential for crosslinks between the two types of systems. As discussed in Section 2.1, separation has always been a key issue in design and licensing of CANDU, and is verified on each plant by an exhaustive review during the design and construction, by the designer, the utility, and the regulatory board.

The study analyzed a total of thirty–two "internal" initiating events, with detailed event tree analysis to estimate frequencies of release categories. As shown in Fig. 4.1, the severe core damage frequency for the reference CANDU 6 plant is of the order of 5 x 10^{-6} per year. This low frequency, comparable to that found for the Darlington plant as discussed above, reflects the presence of the moderator as an emergency heat sink. The major contributor to severe core damage is loss of service water, as this affects cooling water for systems such as the moderator, calandria vault, secondary heat transport system and emergency coolant injection system heat exchangers. The study also found that the initiating event leading to a power excursion coupled with a failure to shutdown is very unlikely: 3 x 10^{-8}/year. This low value results from the use of two independent, rigorously tested shutdown systems in CANDU.

The analysis of severe accident releases drew upon existing Canadian safety analysis for predictions of the containment behaviour for many events. For severe core damage sequences, the significant CANDU design features are the moderator water surrounding the reactor fuel channels, and the shielding/cooling water surrounding the calandria. This water (even without cooling) provides an inherent heat sink for many hours, limiting the progression of a severe core damage sequence. Finally, if the sequence should progress further, the CANDU 6 prestressed concrete containment may crack (but is unlikely to fail) due to overpressure, reducing potential releases from a severe accident.

Further work is underway to refine the operator and consequence models, as described in [Howieson, 1988].

4.2 SEVERE ACCIDENTS SETTING DESIGN REQUIREMENTS

4.2.1 LOCA with Coincident LOECI

As discussed in Section 2, one of the classes of postulated events considered in CANDU licensing analysis is a loss of coolant accident (LOCA) coincident with the failure of the Emergency Coolant Injection (ECI) system to operate on demand. Given the high reliability (99.9 percent) of the ECI system, such a combination of events is extremely unlikely to occur.

Analyses of LOCA/LOECI sequences focus on demonstrating that the regulatory dose limit is met and on verifying that the safety design target of maintaining the integrity of all the fuel channels is also met. The maintenance of channel integrity provides assurance that the fuel bundles remain within their respective channels throughout the accident. Thus, the geometry of the reactor core is well defined and can be analyzed on a channel–by–channel basis, to provide estimates of the timing and extent of fission product release and hydrogen generation.

Over the past few years, considerable effort has been devoted by the Canadian utilities and AECL to further characterize the phenomena relevant to this class of accidents. Conservative analytical models, supported by a large body of experimental data (Section 5), are now in place and are used to assess the consequences of such accidents. Highlights of the recent developments in these models are now discussed, along with the mechanisms and factors that impact on channel integrity, fission product release, and hydrogen evolution during these accidents. Analytical models and methods vary somewhat throughout the Canadian nuclear industry; the models and methods presented here are typical, but the specifics are those currently used by Ontario Hydro.

4.2.1.1 Large LOCA With Coincident LOECI

Postulated large LOCAs are characterized by rapid coolant voiding in the fuel channels which induces an overpower transient. Reactor shutdown systems are activated by one of several redundant trip signals and consequently reactor power is reduced to decay power levels within seconds. The PHTS depressurizes at a rate determined by the break size. For most large breaks there is sufficient convective cooling throughout the transient to avoid severe fuel temperature excursions and consequent large fission product releases, unless the ECI system is assumed unavailable. Severely degraded fuel cooling and significant activity release result if it is assumed that the ECI system fails to operate on demand.

The overall methodology used to assess CANDU core behaviour during large LOCAs with ECI unavailable is shown schematically in Fig. 4.2. System thermohydraulic codes are used to determine the pressure transient and to provide an estimate of the range of transient heat removal conditions in the core. Variations in individual fuel channel characteristics such as initial power, elevation and feeder hydraulic characteristics, in conjunction with possible variable conditions in the reactor headers (e.g. coolant phase separation), produce a wide range of channel conditions which may co-exist in the reactor core during a degraded cooling accident. In the assessment of the thermal and mechanical behaviour of the fuel and fuel channels, channel flow transients are derived from a conservative estimate of the system behaviour. Various conservative channel flow transients are applied to groups of channels with similar characteristics. The results of these single channel analyses are then combined to provide a bounding assessment of the core behaviour under severely degraded cooling conditions.

Under such conditions, fuel heatup leads to fuel deformation and may cause pressure tube yielding ([Howieson, 1986], [Brown, 1984], [Muzumdar, 1983 January], [Muzumdar, 1983 May]). The coolant pressure at the time of overheating determines the mode of pressure tube deformation. Higher pressures (greater than approximately 1 MPa) lead to pressure tube ballooning and, at 16% pressure tube strain, to complete circumferential contact between the pressure tube and the calandria tube in the overheated region (Fig. 4.3). At lower pressures (near atmospheric), pressure tube sag is more prevalent and leads to more localized contact between the pressure tube and the calandria tube. At intermediate pressures (between atmospheric and 1 MPa) a combination of sag and ballooning can result in localized contact followed by complete circumferential contact between the pressure tube and the calandria tube. In all cases – sag, strain, or no contact (for regions of lower power) – a heat removal path to the moderator is established which is effective in limiting the fuel temperature excursions, and consequently limiting fission product release and hydrogen production ([Gordon, 1982], [Lau, 1981]). The detailed assessment shows that:

Over the entire range of large break LOCAs with ECI unavailable, the fuel temperatures do not reach the melting point of UO_2.

An assessment is performed to ensure that channel integrity is maintained throughout large break LOCAs with coincident loss of ECI. The potential for pressure tube failure prior to uniform pressure tube contact with the calandria tube is examined. The potential failure mechanism under these conditions is local overheating of the pressure tube followed by rapid local strain to failure. It is found that for the expected range of contact conditions between the fuel and a pressure tube, local pressure tube overheating is **not** severe enough to cause localized over-strain and failure prior to pressure tube/calandria tube contact [Gulshani, 1987a].

The prevention of sustained, calandria tube dryout following pressure tube contact is a sufficient condition for pressure tube integrity, since it ensures that the fuel channel will not strain further. The factors which affect the potential for calandria tube dryout are the pressure tube contact temperature (i.e. the stored energy available), the contact heat conductance between the pressure tube and calandria tube (i.e. the ease with which heat can be transferred from the pressure tube to the calandria tube), and the subcooling of the surrounding moderator (since this determines the magnitude of the critical heat flux). The experimentally-determined relationship between these factors is presented in Section 5.2 (see Fig. 5.1, [Gillespie, 1981] and [Gillespie, 1982]) and its application is discussed in [Archinoff, 1984], [Brown, 1984], and [Muzumdar, 1982 March]. An assessment is performed of the transient, local moderator subcooling required to prevent calandria tube dryout anywhere in the core. This required moderator subcooling is then compared to the predictions of the available moderator subcooling during the accident. The assessment of the transient spatial variation of the moderator subcooling throughout the accident includes the effect of the additional heat load due to pressure tube/calandria tube contact in a number of channels in the core.

The thermal and mechanical behaviour of a fuel channel under degraded convective cooling conditions is assessed ([Lau, 1986], [Akalin, 1982], [Reeves, 1985], [Reeves, 1982]), including the feedback effect of pressure tube and fuel deformation on the distribution of steam flow in a channel, and consequently on the fuel and pressure tube thermal behaviour (Fig. 4.2). Pressure tube ballooning and/or fuel bundle slumping promotes the bypass of steam flow around the interior of the fuel bundles in a channel ([Reeves, 1983], [Akalin, 1983 February], [Akalin, 1983 September]). Thus, the extent of both the exothermic Zircaloy/steam reaction and the convective heat removal may be reduced in the central region of the fuel bundles due to a limited steam supply. Depending on the rate of fuel heatup, the extent of the exothermic Zircaloy/steam reaction may be further reduced due to relocation of the molten Zircaloy–4 sheath material [Lau, 1987], as now discussed.

If the fuel sheath is not completely oxidized when the Zircaloy–4 melting temperature is attained, then there is potential for the molten Zircaloy to react with the UO_2 fuel, form a low melting point eutectic, and relocate ([Akalin, 1985], [Rosinger, 1985 June], [Rosinger, 1985 September]). If the oxygen content of the molten Zircaloy is high, then the melt wets the UO_2 easily and tends to relocate into pellet cracks and dishes. If the oxygen content of the molten Zircaloy is low, the melt does not easily wet the UO_2 and tends to relocate along the outer surface of the element (Fig. 4.4). The results of experiments on CANDU fuel bundle behaviour at temperatures in excess of 1900C demonstrate this type of melt relocation behaviour ([Wadsworth, 1986], [Hadaller, 1984 September], [Kohn, 1985]). Contact of this eutectic with the pressure–tube has been shown not to threaten pressure–tube integrity – an experimental programme to confirm the models is underway.

Flow bypass due to pressure tube and fuel bundle deformation, and molten Zircaloy relocation, are both mechanisms which effectively reduce the overall rate of the Zircaloy/steam reaction in a channel. This exothermic reaction is an important source of heat to increase fuel temperatures under severely degraded cooling conditions, and also determines the timing and extent of hydrogen evolution from a channel. An example of the effect of fuel and pressure tube deformation on the cumulative hydrogen production is shown in Fig. 4.5, for a large LOCA/LOECI wherein a constant steam flow of 10 g/s is assumed to occur in **all** channels after 40 seconds [Blahnik, 1984]. Such a steam flow is chosen because it maximizes hydrogen production and heat addition due to the exothermic Zircaloy/steam reaction, while providing a minimum amount of convective heat removal.

The fuel temperature transients generated are used in the assessment of fission product release (Fig. 4.2). The distribution of active fission products within the fuel under normal operating conditions [Muzumdar, 1982], the timing and extent of sheath failure, and the transient release of fission products from the fuel are assessed [Archinoff, 1983]. The transient release mechanisms considered include pressure–driven release of the free inventory, rewet and/or high temperature release of the grain boundary inventory, temperature–driven diffusion release from the fuel grains, steam–enhanced grain growth and consequent grain boundary sweeping release, release from the fuel grains due to the reaction with molten Zircaloy, and long–term leaching release from the failed fuel in water [Lau, 1985].

At present it is conservatively assumed that any fission products released from the fuel are transported out of the channels, through the long, relatively cool feeders to the break, with no retention. Future research (see Section 5.5) will address the extent to which fission product retention in the PHTS delays and/or precludes the release to containment of various fission product species.

4.2.1.2 Small LOCA With Coincident LOECI

Small LOCAs are characterized by continued forward flow through the reactor core during most of the transient and relatively slow system depressurization. For these breaks, the reactor power regulating system can compensate for most, if not all, of the void–induced reactivity. Therefore, reactor power is maintained in the operating range until a reactor trip occurs.

In small LOCAs with an assumed coincident failure of emergency coolant injection, the fuel channels would receive adequate single–phase liquid or two–phase coolant (Fig. 4.6a) until well after reactor trip. Eventually, due to the unabated loss of coolant inventory from the PHTS, feeder connections at the supply header would be uncovered. The uncovered inlet feeders still contain low quality coolant, which must drain into the channel before single–phase steam cooling commences (Fig. 4.6b). There is also a substantial liquid level in the channel which contributes to effective cooling. Eventually, as the liquid in the channel boils off (Fig. 4.6c), the fuel is cooled by a decreasing flow of steam and fuel temperature excursions commence. The thermohydraulic response of a fuel channel under these conditions has been assessed using models that are well–verified against experiments ([Luxat, 1987], [Archinoff, 1986], [Hussein, 1985], [Gulshani, 1986]).

Slow boil–off in a fuel channel may result in temperature variations around the circumference of the pressure tube. If the pressure tube is locally hot enough to deform, then these temperature variations could result in localized over–strain and pressure tube failure prior to uniform pressure tube/calandria tube contact. Transient thermohydraulic information is used to assess the transient fuel and pressure tube temperature distributions at any axial location in a channel ([Locke, 1987 March], [Locke, 1987 June], [Locke, 1987 April], [Gulshani, 1987b], [Locke, 1985 June], [Lowe, 1986]). The results of analyses indicate that localized over–strain failure, due to thermohydraulic–induced circumferential temperature gradients on the pressure tube, is not expected for the range of conditions of interest in this accident.

Fission product release and hydrogen generation are bounded by the results for large break LOCA/LOECI scenarios. As in that case, there is **no fuel melting** in any small LOCA/LOECI.

4.2.2 Containment Impairments

The containment system is comprised of several active and passive subsystems. Active subsystems are those which are not normally operating prior to an assumed process system or equipment failure and are then required to operate. Passive subsystems are those which are in operation prior to a failure and continue to operate. The requirement to design for dual failures means that combinations of process system and containment failures must be analyzed; however, the independence which is designed into containment subsystems permits impairments of containment subsystems to be considered rather than total containment system failure.

The active subsystems which could be impaired are containment isolation and dousing. Isolation impairments include failure of the ventilation inlet or outlet dampers to close, and failure of isolation logic which implies that <u>both</u> inlet and outlet dampers fail to close. Due to separation of dousing into two subsystems in some CANDUs, a failure of both dousing subsystems is highly unlikely but has been examined in certain cases. The personnel and airlock door seals are normally inflated when the doors are closed. However, to account for the possibility that the airlocks have been recently opened and closed and the seals have not been re–inflated, deflated door seals are considered.

Therefore, the containment impairments considered for the dual failure analyses are:

1. failure of an isolation damper,
2. failure of containment isolation logic,
3. failure of dousing, and
4. deflated airlock door seals.

The containment response to an accident is determined by: the containment design (either multi–unit/vacuum building containment or single unit containment), the magnitude of break discharge, containment heat sources and sinks, and the particular containment impairment being considered.

In addition to the phenomenological response, discussed below, having to meet licensing requirements for dual failures has led to a significant emphasis on the reliability of containment isolation. A programme of tests during plant operation is established, so that the reliability of isolation on demand can be established at greater than 99.9%. Provision of test logic and the arrangement of components in order to demonstrate the availability of active containment subsystems are requirements of the conceptual design, and form an integrated part of containment.

4.2.2.1 Multi–Unit Containment

Following a postulated large LOCA, operation of the multi–unit negative pressure containment system would be as follows [Morison, 1984]. There is a large release of steam into the Reactor Building which results in a rapid increase in containment pressure. The self–actuated Pressure Relief Valves open automatically at 7 kPa(g). The steam enters the Vacuum Building where the pressure increase actuates the dousing system. The resultant spray condenses the steam, thereby reducing the pressure. A typical pressure transient is shown in Fig. 4.7. For the largest break size, the pressure peaks within the Reactor Building accident vault in 3 seconds, and returns to subatmospheric in about 60 seconds.

Assuming no containment impairments, the pressure slowly returns to atmospheric pressure as a result of the small inleakage of air from outside, and because of the consumption of instrument air by equipment located within the containment.

When containment pressure returns to atmospheric, the Emergency Filtered Air Discharge System (EFADS) is brought into operation to control the pressure at slightly sub–atmospheric. EFADS is equipped with particulate filters and charcoal filters for iodine removal, and radiation monitoring equipment to measure any radioactive releases from the station.

Envelope impairments reduce the time scale of repressurization, but the phenomena are basically similar. The vacuum containment is powerful enough to hold the reactor building subatmospheric for significant periods of time even **with** an impairment.

4.2.2.2 Single Unit Containment

A typical single unit CANDU containment pressure response with an impaired containment is shown in Fig. 4.8. The containment pressure increases due to the accident, then falls due to the combination of:

- reduced break discharge rates as the coolant pressure falls;
- dousing in containment which acts to condense the steam; and
- leakage through the impairment itself.

4.2.3 Anticipated Transients Without Scram

As discussed in Section 3.5, a hypothetical unterminated accident in a CANDU 6 would be an extremely improbable event, because many independent systems would have to simultaneously fail, namely:

* failure of a normal control system,
* **plus** failure or incapability of stepback,
* **plus** failure of shutdown system #1
* **plus** failure of shutdown system #2.

Such an accident has an estimated frequency of less than 1 in 10 million years per reactor in CANDU 6 [Howieson, 1988], and, in common with world practice on very low frequency events, requires no further design provision. Nevertheless the containment response to such a sequence is discussed in Section 4.3.2.

4.3 SEVERE ACCIDENTS WHICH CHALLENGE DESIGN

4.3.1 Loss of the Moderator Emergency Heat Sink

4.3.1.1 Background

In Section 4.2, it was shown that the moderator can provide an emergency heat sink for the fuel in the absence of normal coolant and emergency coolant injection. Studies for the Atomic Energy Control Board have examined the more severe consequences that would follow if even this emergency heat sink were to fail – namely, the effects of a loss of moderator heat sink capability in a Bruce–A NGS unit occurring **simultaneously** with the dual–failure accident of a large LOCA accompanied by complete unavailability of emergency coolant injection (LOECI) ([Rogers, 1984 June], [Rogers, 1984 August], [Rogers, 1984 September]). These studies were deterministic in nature in that they did not select the failure mode from a probabilistic analysis. In addition, they concentrated on the calandria thermal/mechanical response, rather than on the full spectrum of events accompanying such a severe accident – e.g., hydrogen production and transport. They nevertheless show instructive trends, as they reveal a further inherent heat sink beyond the moderator.

4.3.1.2 Hypothetical Accident Sequence

The study focussed on a hypothetical accident sequence following a large LOCA/LOECI in which the moderator cooling system fails, with the result that the moderator heats up, begins boiling and is eventually expelled from the calandria. As fuel channels are uncovered by moderator expulsion, they overheat and fall to the bottom of the calandria where they are quenched by the remaining moderator. Eventually, all moderator is lost from the calandria and the core debris heats up and melts. The study showed that the molten core material would be contained in the calandria which, because of the separately cooled water–shield heat sink, maintains its integrity throughout the accident sequence. This sequence is now described in detail.

4.3.1.3 Analytical Approach

The behavior of core components under the extreme conditions of this accident sequence is, in general, quite speculative since little experimental information is available. Thus, the approach used required that the essential features of the behavior of core components and the moderator be modelled on a physically sound basis; that sensitivity studies be undertaken; and that bounding analyses be used in certain cases.

4.3.1.4 Results

Typical thermal behavior of the fuel in different rows of fuel channels in this accident sequence is shown in Fig. 4.9, for a case where the mode of pressure tube deformation is assumed to be by sagging onto the calandria tube. The first temperature peak for any fuel channel row occurs while the channel is still covered by liquid moderator, while the second peak is predicted after channel uncovering. Actually, the channels would be expected to fail before the second peak is reached, when the pressure tube and calandria tube temperatures reach about 1750C. Fig. 4.9 shows that fuel in uncovered channels would be well below the UO_2 melting point up to the times of channel failure.

The amount of moderator remaining in the calandria, as a function of time, for the reference conditions assumed, is shown in Fig. 4.10 [Rogers et al, 1984 August]. As bulk boiling initiates and propagates downward through the calandria, at about 16 minutes, it is predicted that more than half the moderator is expelled. The moderator would be expelled in surges in this period but pressure peaks in the calandria are not severe (<220 kPa abs.) because of the low steam qualities (<4%) in the relief ducts. Subsequent rapid expulsions of moderator seen in Fig. 4.10 are caused by groups of fuel channels failing and dropping into the remaining moderator. Clearly the simplification of the model is responsible for the fine structure – in reality the pressure transient would be smoother. Pressure peaks during these subsequent flow surges are again low (<470 kPa abs.) and the integrity of the calandria is not threatened. For the reference conditions, Fig. 4.10 indicates that all the moderator is expelled from the calandria in about 50 minutes.

The study also showed that just before the last of the moderator is expelled from the calandria, almost all of the core debris in the bottom of the calandria is in the solid state and has been quenched to quite low temperatures (about 150C).

Typical results of the analysis of the subsequent heat–up of the solid core debris in the bottom of the calandria are given in Fig. 4.11. For a wide range of bed porosity, Fig. 4.11 shows the maximum and upper and lower surface temperatures of the bed as a function of time after reheating starts, following the loss of moderator from the calandria. The maximum temperature in the bed reaches the melting point for oxidized core material, about 2700C, about 80 minutes after the start of reheating, or about 130 minutes after the start of the accident. The upper and lower temperatures are well below the bed melting point at this time. The lower surface temperature is also well below the melting temperature of the stainless steel calandria wall and the lower surface heat flux into the shield–tank water, 15 W/cm^2, is well below the critical heat flux, about 280 W/cm^2, at the time that melting begins within the bed.

Fig. 4.11 shows that the thermal behavior of the debris bed is very insensitive to bed porosity. Other analysis showed that the thermal behavior was also very insensitive to pore size, material thermal conductivities and contact conductance between pieces of debris. It was concluded that the integrity of the calandria would be maintained during this stage of the accident sequence.

Once melting begins in the debris bed, some time will be required for the transition to a completely molten pool. No attempt was made to develop an analytical model of the debris bed for this transition stage. Instead, the time required for this period and the accompanying heat source decay were ignored, so that the analysis predictions are conservative.

Results for the analysis of a molten pool were inconclusive as to whether the pool would boil or not, depending on the property values used. Nevertheless, the maximum predicted heat flux into the shield tank water for all conditions (about 20 W/cm^2) is well below the critical heat flux. The interaction of the molten pool with the calandria wall, as illustrated in Fig. 4.12, indicates that there will be no melting of the calandria wall and that a solidified crust, over 2 cm. thick, would form on the wall, thus providing a protective shield. Analysis also showed that the heat flux into the wall would have to be about 100 W/cm^2, about five times the maximum predicted, before melting of the calandria wall would begin. Similarly, for conditions that would result in boiling of the molten pool, analysis showed that the calandria wall would be well–removed from melting conditions and the heat flux into the shield–tank water would be well below critical heat flux under both the boiling pool and the condensing vapor film above the pool.

These analyses indicate that the core material debris, whether solidified or molten, will not jeopardize the integrity of the calandria vessel, irrespective of whether the molten pool boils or not. Further calculations show that the molten pool would solidify at a time between 10 and 50 hours, depending on property values. Thus, it is concluded that the entire mass of core material would be retained within the calandria, as long as the shield–tank cooling water system continues to function. The consequences of failure of heat removal from the shield tank have also been assessed, as described in detail in [Howieson, 1988].

4.3.1.5 Conclusions

a. The moderator would be completely expelled from the calandria in about an hour.
b. No gross fuel melting would occur even when fuel channels are uncovered, and the core debris would not begin to melt until more than 2 hours after the accident begins.
c. The calandria would retain its integrity provided that the shield–tank water cooling system remains operational.
d. Core debris would be contained within the calandria and would begin to re–solidify in the period of 10 to 50 hours after accident initiation.
e. The shield tank system provides an additional heat sink to stop the progression of a severe core damage sequence.

4.3.2 Containment Response to Severe Accidents

The containment is designed for the challenges that could result from rupture of the largest main primary cooling pipe. For the typical example of the CANDU 6 reactor, the maximum pressure inside containment for this accident is predicted to be less than 70 kPa(g), well below the design pressure.

Consequential failures in the containment structure could occur due to severe core damage accidents which lead to addition of a large amount of energy to the containment atmosphere. A key feature of the CANDU containment structure is that it is a pre–stressed concrete building. Experiments at the University of Alberta ([Asmis, 1983], [MacGregor, 1980]) have demonstrated that at 330 kPa(g) internal pressure (2.3 times the proof test pressure), cracks would penetrate through the wall. Leakage through cracks is negligible at pressures below 345 kPa(g), and increases exponentially as the pressure is increased beyond that. At pressures approaching but still below the predicted failure load of around 530 kPa(g), the experiments suggest a leakage rate sufficiently high that the internal pressure is relieved; so it is difficult to have a condition in which the containment fails due to internal pressure loading.

This has a significant advantage – the structure would be unlikely to fail in a catastrophic way, and hence fission products would be largely retained inside a containment structure. The "wet" atmosphere therein will immobilize them further.

5. SAFETY RESEARCH

5.1 PROGRAMME GOALS

The aim of the safety research programme is to ensure a firm technical understanding of the various phenomena that could occur during an accident. The work of the safety program has covered both the loss–of–coolant accident, by now well understood, and, more recently, the lower probability severe accident conditions.

There exists a large amount of interaction with the technical community, both nationally and internationally. Much of the research in Canada is supported and coordinated by the CANDU Owners Group (COG); most of the rest is performed by AECL–Research. Internationally there is interaction through participation with formal agencies such as the Organization for Economic Cooperation and Development Principal Working Groups and the International Atomic Energy Agency, and also through various technical associations.

The main focus of the research is on aspects that are unique to the CANDU system. However, sufficient generic and underlying research is also performed to ensure contribution to, and an ongoing interaction with, international programmes.

5.2 FUEL BEHAVIOUR

An ongoing research programme on high temperature fuel behaviour has provided a solid database and verified codes to describe fuel behaviour under LOCA conditions. This has been achieved by separate effects experiments to evaluate properties of the UO_2 and cladding, development of computer models to describe sheath deformation and gas release processes, and in–reactor tests to provide all–effects verification of the behaviour of fuel and fission products. Using this methodology, current work is extending the database and models to the high temperatures associated with severe accidents. An advantage of the short (50 cm.) CANDU fuel bundles is that full–scale tests are relatively easy to do.

Work on fuel oxidation and consequent fission product release is being performed with both fresh and irradiated fuel. The oxidation kinetics of unirradiated UO_2 in air and steam has been studied at temperatures up to 1650C in steam and up to 1200C in air ([Cox, 1986 September], [Cox, 1986 October]) and is being extended to higher temperatures. To date, an extensive database [Iglesias, 1987] for oxidative release of fission products has been built from 60 experiments performed in the temperature range 400 to 1700C.

Severe fuel damage phenomena such as molten Zircaloy relocation, collapse of individual fuel elements, bundle deformation and UO_2–Zircaloy interaction are also being studied in the laboratory. Much of the work is performed with horizontal fuel element simulators.

The laboratory work to date has shown that the fuel cladding, oxidized by steam, has sufficient structural strength to maintain a coolable horizontal geometry, even during very high temperature transients. Other high temperature transient experiments with entire fuel bundles heated in steam to temperatures in excess of 1900C [Wadsworth, 1986] have shown that the relocation of molten un–oxidized Zircaloy to inter–element spaces, as discussed in Section 4.2.1.1, can lessen the effective oxidation rate by reducing the exposed surface area. Small–scale experiments are in progress to quantify this relocation phenomenon [Wren, 1986].

The interaction between the Zircaloy fuel cladding and the UO_2 fuel may be affected by the presence of a very thin layer of a graphite–based lubricant (CANLUB^{tm}) applied to CANDU fuel elements during manufacture. Experiments have shown that CANLUB^{tm} reduces the interaction at temperatures below 1500C (which is relatively slow), but above this temperature it has no effect [Lim, 1986].

5.3 FUEL CHANNEL BEHAVIOUR

The horizontal pressure tube/calandria tube fuel channel has led to an extensive research programme, which has been underway for a number of years, on fuel channel behaviour during postulated loss of coolant accidents involving coincident loss of emergency coolant injection. The work has established the conditions under which the residual heat in the fuel channel can be transferred to the moderator radially while maintaining fuel channel integrity. This heat removal may be accompanied by deformation of the pressure tube at high temperature, as it can contact the calandria tube either by ballooning at a high coolant pressure, or by sagging under its own weight when the coolant pressure is low.

Understanding the phenomena involved in the integrated tests on moderator heat sink effectiveness requires understanding of various single effects and the capability of modelling them. Such tests include studies of high temperature pressure tube sag and ballooning, development of high temperature creep deformation models, studies of

Zircaloy–steam reactions, measurements of critical heat flux in horizontal tube banks, and measurements of contact heat conductance and of high temperature Zircaloy emissivities.

Fig. 5.1 shows the results of a large number of integrated tests where the pressure tubes ballooned into contact with the calandria tube ([Gillespie, 1981], [Gillespie, 1982]). Shown are the boiling regimes on the outside surface of the calandria tube after contact with the hot pressure tube. If the surface of the calandria tube can be maintained in nucleate boiling or patchy dryout, it will be sufficiently cool that significant deformation will not occur. These data are used in accident analyses to assess fuel channel integrity, as discussed in Section 4.

Current experiments and model development are focused on the effect of temperature gradients on the deformation of pressure tubes, as discussed in Section 4.2.1.2. These experiments measure the temperature gradients and pressure tube deformation under various power, coolant pressure and coolant flow rate conditions.

5.4 BLOWDOWN TEST FACILITY

Single– and three–element in–reactor high–temperature fuel tests have been performed at AECL–RC for many years. In–reactor blowdown tests on CANDU fuel, at temperatures around 1000C, were performed in the U.S. Power Burst Facility reactor, and confirmed models of fuel behaviour in LOCA.

Now, a series of in–reactor severe fuel damage tests are planned. These will be performed in the new Blowdown Test Facility (BTF) ([Fehrenbach, 1987], [Wood, 1986]) in the NRU reactor. The purpose is to confirm fission product release fractions and chemical behaviour for overheated fuel, under depressurizing conditions. The focus will be on the release of **active species** fission products from fuel operating at temperatures in the range of 1500 to 2500C, and the subsequent transport and deposition of fission products in the primary heat transport system. The specific test objectives of this programme are:

- to measure the amount and timing of fission product activity release to the coolant during depressurizing conditions, during high–temperature post–depressurization, and during subsequent rewet, and to correlate the measured releases with the stages of fuel element behaviour;
- to measure the rate of fission product transport and deposition in carbon steel and stainless steel pipes, and determine the partition of fission product isotopes between liquid, solid, and vapour phases, and the chemical form of fission product species in the blowdown tank;
- to demonstrate techniques and procedures for decontamination of system components experiencing extensive fission product deposition and transport of irradiated fuel debris.

This programme will provide information on fission product behaviour that will be used to assess and refine the predictive ability of accident analysis codes.

The Blowdown Test Facility is shown schematically in Fig. 5.2 and its main design parameters are included in Table 5.1. The in–reactor test section of BTF is a vertical, re–entrant, pressure tube arrangement which will accommodate assemblies of three fuel elements plus a thermal shroud up to 70 mm in diameter and up to 3 m in length.

5.5 AEROSOL TRANSPORT

In contrast with other water reactors where there are substantial sources of aerosol material due to the presence of borated water, stainless steel structural materials and silver–cadmium control rods, CANDU aerosol source materials are limited to the Zircaloy fuel cladding, the UO_2 fuel and the fission products themselves. The smaller amount of low–melting–point material results in much lower aerosol densities for severe accidents in a CANDU. Currently the attenuation of these aerosol–borne fission products in the Primary Heat Transport System is not credited in CANDU safety analyses.

TABLE 5.1

NRU Blowdown Test Facility (BTF) Design Parameters

	Reactor and Test Section
Reactor power	130 MW
Cosine flux length	3.6 m
Mid–core (maximum) flux – thermal at cell boundary – fast	1.7×10^{18} $n.m^{-2}.s^{-1}$ 0.3×10^{15} $n.m^{-2}.s^{-1}$
Maximum fission heat in BTF	2 MW with pressurized water cooling 200 kW with superheated steam cooling
	Normal Operation **Coolant Conditions**
Coolant type	Recirculating pressurized water or superheated steam
Pressure (maximum)	10.5 MPa
Temperature (maximum)	water 300°C, steam 350°C
Flow (maximum)	water 10 kg/s; steam 1 kg/s
	Blowdown Conditions
Delay from loop isolation to reactor trip	0.1 – 60 s
Blowdown time to 1 MPa 0.3 MPa	10 – 300 s 30 – 500s
	(Post–Blowdown Stagnation)
Coolant type	Saturated steam or helium
Flow (steam) (inert gas)	2 – 20 g/s variable
	(Rewet)
Coolant	25°C water
Rewet flow	0.04 – 4.8 kg/s
	(Post–Rewet)
Coolant	Once–through de–ionized water
Pressure	atmospheric
Temperature (inlet)	25°C
Flow	0.01 to 0.05 kg/s

The current focus of the research programme is to model the production and transport of aerosols which may be created in the Primary Heat Transport System [Mulpuru, 1987]. The laboratory experiments will be augmented by results from in–reactor BTF experiments. Development of a model coupling aerosol transport to thermohydraulics is also underway. This code development effort is supported by theoretical analysis of the validity of key assumptions which are used in aerosol physics models [McDonald, 1987 September a,b].

Key work on containment aerosols focuses on the two–phase jet at a break, in order to characterize the water droplet aerosols and their size distribution, and to determine the extent of fission product washout by the droplets.

5.6 FISSION PRODUCT CHEMISTRY

In parallel with the aerosol programme, there is a research programme to develop a fundamental understanding of fission product chemistry. The main focus is the chemistry of iodine in the containment building, although the chemistry of iodine and other fission products (Cs, Ru, Te) in the Primary Heat Transport System is also being addressed ([Wren, 1983], [Garisto, 1986]). The behaviour of iodine in the containment building depends on parameters such as pH, Eh, temperature, radiation fields, impurities in the sump water, and the presence of chemical additives that could be added to suppress iodine volatility. The work shows that, at equilibrium and at the low iodine concentrations ($<10^{-5}$ mol.dm^{-3}) of an accident, the dominant forms of iodine would be I^-(aq) and IO_3^-(aq) [Lemire, 1981].

The chemical kinetics data has been used in the formulation of the code LIRIC (Long–term Iodine Release Integrated Code) which predicts the iodine chemical forms and distribution between the aqueous and gas phase in containment [Wren, 1985].

Integrated tests are starting in the Radioiodine Test Facility (RTF) to validate the LIRIC model. The RTF is an intermediate–scale facility designed to provide radiation fields and chemical conditions appropriate to the reactor containment building conditions after an accident [Kupferschmidt, 1986]. Shown in Fig. 5.3, the main reaction vessel is a 400 dm^3 cylindrical vessel capable of containing a Co–60 radiation source of variable strength (0–10 kGy.h^{-1}) to simulate radiation fields encountered in an accident. The inner surface lining of the vessel can be altered to include painted and bare steel and concrete, and the temperature can be varied up to 80C. It includes extensive on–line instrumentation to monitor the various process variables and sampling systems for off–line chemical analyses.

5.7 HYDROGEN COMBUSTION

A study of hydrogen combustion has been underway for a number of years. The research addresses the hydrogen produced both by the Zircaloy/steam reaction and by the radiolysis of water. The experimental programs use both bench–scale tests as well as tests on intermediate scale vessels housed in the Containment Test Facility (CTF) [Kumar, 1984]. Important combustion phenomena which have been studied include flammability limits [Kumar, 1985], laminar burning velocities [Liu, 1983], and flame acceleration by venting [Kumar, 1987]. Also, the effectiveness of hot surface ignitors, such as glow plugs, has been examined for both lean and rich hydrogen–air–steam mixtures ([Tamm, 1985], [Tamm, 1987]) in a cooperative program between AECL, COG and EPRI (Electric Power Research Institute).

Current programmes focus on hot surface ignition limits, flame acceleration mechanisms, turbulent burning velocities, detonation limits, and transition from deflagration to detonation, to provide the basic information needed to develop and verify models for localized hydrogen combustion.

6. OPERATIONAL ASPECTS

Under the Canadian regulatory process, the licensee of a nuclear power plant is responsible to ensure that the plant staff and the general public are adequately protected from the consequences of plant accidents. Comprehensive studies are undertaken to ensure that, following accidents, essential features of safe plant operation are maintained. These include:

* habitability of control room(s),
* means to ensure reactor subcriticality after shutdown
* containment integrity,
* assured heat sink, and
* monitoring of plant safety status.

As part of this, operating procedures are developed, and staff trained, with the focus on stabilizing the plant Critical Safety Parameters (CSPs). Contingency response procedures are also developed to mitigate the consequences of an emergency and to provide assurance that all reasonable measures are undertaken to ensure human safety and to minimize property damage.

In the following, a fairly typical approach by a Canadian utility is described, which ensures post–accident operational safety. The specific example is the Lepreau I CANDU 6 plant in New Brunswick, owned and operated by the New Brunswick Electric Power Commission.

6.1 POST–ACCIDENT OPERATIONS REVIEW

All nuclear power plants are required to perform a plant–wide review of operation following a worst case loss of coolant accident. A detailed review of the relevant safety assessment documents, emergency plant operating procedures, and contingency plans is conducted in order to identify operator actions required to maintain essential plant safety functions. Each operator action is assessed for feasibility of execution based on its location, duration, access and expected radiation field.

The methodology for performing a radiation field study is described in [Natalizio, 1983] and is based on an improbable accident sequence which involves a break in the primary coolant circuit and a failure of a safety system (ECI or containment). The estimated worst case source term for a case of LOCA with impaired ECI is typically 10% halogens and noble gases and 3% particulates (of total core inventory).

The CANDU 6 moderator and primary heat transport system components are located inside containment, so the potentially hazardous high radiation fields following a LOCA are limited to: parts of the Emergency Coolant Injection System located outside containment; the reactor building ventilation; piping penetrations through the containment wall; and the airlocks.

To date, several post–LOCA operations reviews have been performed, covering short–term procedures and long–term equipment reliability. Although these studies generally confirmed the adequacy of the original design and operating procedures, a detailed assessment led to several recommendations to improve access to specific locations, to address the need for remote operability, and to confine contamination.

Examples of recommended changes included:

- Installation of additional shielding to improve control room habitability. At the Gentilly–2 CANDU 6 sister plant, a shielding wall was installed alongside the main airlock; and at Point Lepreau, a shielding wall was installed beside the reactor building exhaust filters.
- Provision of remotely operated isolating valves on instrument air lines to the reactor building at Gentilly–2 and Point Lepreau.
- Relocation of the ECI pump switchgear from the vicinity of the ECI pump pit to an accessible area, to permit operability and maintainability.
- Re–routing of the ECI system leakage to the reactor building to avoid excessive contamination of active drainage.

6.2 EMERGENCY OPERATING PROCEDURES

The day–to–day operation of a Canadian nuclear generating station is governed by the AECB–approved Station Operating Policies and Principles which define the envelope within which that station must be operated.

Besides the system–specific operating manuals covering normal operation, Emergency Operating Procedures (EOPs) are produced to cover abnormal or accident situations.

Where the cause of the upset can be recognized, an event specific EOP is produced. The ability to predict the anticipated plant response is perceived as a major advantage of these procedures, as they permit optimization of the corrective action. Typical event–specific EOPs include: dual computer failure; loss of services such as instruments or cooling water supplies; loss of main electrical power; loss of feedwater; LOCAs; and boiler tube failures. A generic EOP is also produced to cater to situations where the upset cannot be clearly diagnosed or identified; or the initial response by the operator proves inadequate; or the status of a Critical Safety Parameter is unsatisfactory.

An unsatisfactory status of a Critical Safety Parameter (CSP) indicates a threat to the integrity of the fuel sheath, or the Primary Heat Transport System, or the containment. Typical CSPs include the primary coolant sub–cooling margin, primary coolant inventory, reactor power, boiler pressure and level, containment pressure and radioactivity levels. Fig. 6.1a ([Kelly, 1986], [Kelly, 1987]) shows a typical response strategy to a plant upset, and Fig. 6.1b shows a guideline for controlling the CSPs.

Each EOP is developed, or is being developed, to meet the requirements prepared by a joint utilities task force [Kelly, 1987]. This document covers the complete life cycle of the EOP program which include its generation, verification, validation, issuance, training requirements and revision.

The operating staff are provided with comprehensive training to develop the necessary knowledge and skills to identify and respond to a plant upset. Training methods are normally a combination of classroom and field sessions, with the former providing the technical and procedural understanding and the latter developing the operating skills. This training may include control room training, plant walks–through or simulator training.

6.3 CONTINGENCY PLANNING

The starting point for all emergency plans that cope with severe radiation contingencies is the assumption that an accident causing highly radioactive releases to the environment can occur even if its occurrence is extremely unlikely. The emergency planning and preparedness systems must define the necessary countermeasures required to mitigate the consequences of severe accidents to protect public health and minimize damage to property. The Atomic Energy Control Board [AECB, 1984] requires the licensee to:

- develop on–site contingency plans for coping with emergencies within the facility boundary, and
- participate with federal, provincial and municipal governments to develop an off–site contingency plan to deal with those events which result in release of radioactivity beyond the station boundary.

Although detailed procedures differ in scope and methodology, the general approach followed at all CANDU stations is to define two sets of complementary plans: a general plan defining overall plant response strategy and a specific plan covering each category of contingency, namely: Radiation, Medical, Chemical, Security and Fire contingencies [Weeks, 1987].

The severity and extent of any particular contingency will dictate the degree of response required. Normally incidents are classified as Alerts or Emergencies. An Alert is declared for localized events which can be controlled by station staff and on–site resources, while an Emergency is declared when the contingency threatens more severe harm to site personnel, public, or the continued safe operation of the plant.

The key to the success of any Contingency Plan is of course the organizational framework, the efficiency of communications and level of expertise and training of the response groups involved. Fig. 6.2 shows a typical organization chart for the purpose of contingency planning. The Shift Supervisor on duty retains the overall responsibility for all response duties as head of the Command Unit. The Response Team consists of a group of designated and specially trained shift staff as part of the normal shift complement. The Assistance group is assembled from the non–shift staff to provide senior–level technical and operational advice to the Shift Supervisor in case of an emergency.

A comprehensive training program is developed to provide staff with the necessary knowledge and skills to support the contingency–related activities, the extent of training being commensurate with the individual's role in the overall plan. Thus the Response Team members, because of their key function, receive extensive advanced training in a variety of topics which include fire–fighting, first aid, chemical protection, and all the specific contingency plans.

7. CONCLUSIONS

Inherent CANDU properties, namely:

- a moderator which acts as a dispersed emergency heat sink for fuel heat;
- the presence of a water–filled shield tank which can prevent melt–through of the calandria; and
- a containment which exhibits forgiving behaviour under hypothetical overpressure conditions;

all contribute to a design for which the probability of severe core damage is low, of the order of 5 x 10^{-7} per reactor–year, and the consequences of core damage are limited.

The licensing philosophy of examining dual failures as part of the design basis set, has led to redundancy of shutdown which makes an unterminated accident a negligible contribution to total risk, and to a design which will accommodate impairments in the containment and emergency coolant injection systems.

Furthermore, these same characteristics mean that the plant response to increasingly severe accidents is gradual – there is no sudden change in behaviour.

The design characteristics and the licensing approach have also resulted in:

- a research programme which supports models for the predicted behaviour of CANDU for both loss of coolant **and** severe accidents, and
- a flexible approach to severe accident management on site.

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