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Progress in development and design aspects of advanced water cooled reactors

Proceedings of a Technical Committee Meeting held in Rome, 9–12 September 1991

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FOREWORD

The International Atomic Energy Agency has been providing an international forum for the exchange of information for many years. This has allowed Member States to prepare, taking into account their specific national requirements, an international consensus on subjects of mutual interest and thereby provide advisory standards and requirements in many areas. This effort should be continued for future nuclear reactors and should cover a wide range of topics including not only electricity generating plants but also plants for the supply of nuclear heat and for cogeneration, with sizes ranging from the smallest of the small and medium sized power reactors to the largest electricity generating plants. Member States with sizeable nuclear power programmes have also developed the necessary national infrastructures for design, development, safety assessment as well as licensing processes and stable regulatory regimes. Following the conclusions of the IAEA Conference on the Safety of Nuclear Power: Strategy for the Future, held in Vienna, 2-6 September 1991, there is a need for guidance/advice to Member States on a number of issues ranging from suitable regulatory requirements, preferably of an internationally acceptable nature, down to design objectives/targets and the assessment methodologies needed to show how these may be met by the various advanced reactor designs which are likely to be available as "off-the-shelf" plants in the future. This publication addresses the state of the art reached in the development of advanced reactor designs, their containments, safety systems and physics/thermohydraulic aspects.

EDITORIAL NOTE

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SUMMARY OF THE TECHNICAL COMMITTEE MEETING

INTRODUCTION

The objective of the Technical Committee Meeting (TCM) on Progress in Development and Design Aspects of Advanced Water Cooled Reactors held in Rome, from 9 to 12 September 1991, was to provide an international forum for technical specialists to review and discuss technology developments and design work for advanced water cooled reactors, safety approaches and features of current water cooled reactors and to identify, understand and describe advanced features for safety and operational improvements. The TCM was attended by 92 participants representing 18 countries and two international organizations and included 40 presentations by authors of 14 countries and one international organization.

Mr. J. Kupitz, Head of the IAEA's Nuclear Power Technology Development Section opened the TCM and presented an overview of the outcome of the IAEA Conference on the Safety of Nuclear Power: Strategy for the Future, held in Vienna, 2-6 September 1991 (the IAEA Safety Conference 1991). Mr. F. Velonà, Director of R&D Division of the Ente Nazionale per l'Energia Elettrica (ENEL), Italy, welcomed the participants. The TCM was chaired by Mr. L. Noviello, Technical Director of the Nuclear Department in the ENEL R&D Division. The meeting was divided into four sessions and Working Groups.

It was generally agreed by the participants that the topics covered at the meeting had met with great interest. The IAEA wishes to acknowledge the excellent organizational arrangements and the high level of hospitality provided by ENEL and especially by Mr. Noviello.

The IAEA-TECDOC-626 on "Safety Related Terms for Advanced Nuclear Plants" was presented by Mr. J. Kupitz. Mr. L. Kabanov, Head of the IAEA's Engineering Safety Section, presented information on the IAEA activities in the safety of advanced reactors.

The conclusions and recommendations agreed to at the TCM in Rome were reviewed at an IAEA Consultants Meeting in December 1991 in Vienna. A summary of the sessions, conclusions and recommendations was drawn up to provide a reduced survey of the main issues arising from the TCM.

As a result of the discussions and analysis of the papers, Working Group reports and the meeting Chairman's report, the following conclusions and recommendations, embodying the main issues raised, have been compiled.

SUMMARY OF PRESENTATIONS AND DISCUSSIONS

Summary by the Chairman of the TCM, Mr. L. Noviello

The meeting provided an international forum for the presentation and discussion of the Member States' approach to the development and design aspects of advanced water cooled reactors.

The summary of the IAEA Safety Conference 1991 presented by Mr. Kupitz has put this meeting in the proper perspective from the beginning. Actually we have been able to verify throughout our technical sessions the feasibility and the degree of maturity of the main improvements discussed at the Safety Conference in Vienna.

All the elements necessary to make a step forward in the technology of future reactors are becoming available, including, of course, the need to consider severe accidents from the inception of containment design. Probably this evolution is one of the key elements needed to reach the full maturity of nuclear power and a further expansion of the worldwide use of this important source of energy.

The meeting was divided into four sessions:

- Development programmes and conceptual design of ALWRs
- Containment systems
- Safety systems and analysis
- Physics and thermohydraulics

covering both the general concepts and the particular aspects of design.

It is possible to draw some general conclusions from all the valuable contributions that have been made by almost 100 participants from about 20 different countries and international organizations:

1. We have clearly seen three levels of design evolutions, although the timing of the expected commercial availability is somehow contradictory.

2. Caution has been asked in the adoption of new technologies, and only, in any case, after exhaustive testing of both components and systems.

3. Improved containments are the most readily and possibly the best way to meet the goal of a step forward in safety. In practice, this means explicit integration of severe accidents into the design criteria. There was a clear indication that the elimination of the high pressure core melt sequences can be achieved effectively by the adoption of a depressurization system. There were less unanimous indications on the approach to be followed to cope with the low pressure sequences; the containment can be designed for these events but, according to some authors the uncertainties must be further reduced in order to avoid the need for too large margins.

4. The depressurization systems being adopted for the PWRs, but also many other passive systems proposed for different reactors, present similar configurations and probably similar issues. It will be appropriate for the future IWGATWR activity at the right time, to provide a forum to discuss in detail these issues with such a basic interest.

5. Design tools enhancement has been widely discussed. Many of them referred to the natural circulation of the water in the primary circuit or of the air moisture and aerosols in the containment atmosphere. It was a proper focus since natural circulation is related to the adoption of passive systems and then to the simplification of the designs. Although present in the nuclear plants from the beginning, in many of the new designs natural circulation plays a much more important role.

6. The new designs have some other important goals beyond safety i.e. man-machine interface, construction techniques and schedule, licensing procedures, costs. These problems may be a good subject for the next meetings.

Recommendations

Some technical areas and specialists meetings which it is possible to recommend for the IWGATWR have been discussed and recommended.

- One is the need to clearly separate the technical discussions arena from the licensing discussions arena. Industry needs a forum to discuss openly, I would say with no licensing concerns, of its desires, trends, needs. This shall not prevent preliminary discussions with the licensing authorities only with mature, well founded industry positions in which, when appropriate, design improvements and particulary safety improvement will have been put in an appropriate cost-benefit balance.

- The second one refers to the subject of tests. This issue emerged from this meeting as particulary important for next generation power plants in the following fields:

  - severe accident phenomena
  - natural circulation thermohydraulics.

  Extensive testing is under way in both areas in several countries and it is appropriate to mention that an outstanding co-ordination activity is being performed by OECD. It is most important that the results of the experiments are brought to the attention of all designers.

SUMMARY OF SESSIONS

SUMMARY OF SESSION 1

DEVELOPMENT PROGRAMMES AND CONCEPTUAL DESIGN OF ALWRs
Chairmen: Mr. J. DeVine, Mr. T. Pedersen

The Session provided an overview of the ALWR development programmes in Member States and reviewed a number of conceptual designs that are being studied.

Fourteen technical papers were presented describing the current position in Czechoslovakia, Japan, the Republic of Korea*, USA, Switzerland, and the USSR.

The order of the presentations was changed from that in the Meeting Agenda to permit first discussion of technical requirements for ALWRs, followed by description of advanced water reactor designs, design concepts and their application for heat and electricity generation. Finally the future R&D needs for the next generation of power plants and some other design concepts were presented.

An up-to-date review of the US DOE ALWR programme, focussing on the EPRI Utility Requirement Documents (URDs) which embody the plant-wide needs of US utilities for the new ALWRs, was presented. The role, scope and top-tier requirements of the URDs were summarized and the overall safety objectives were given.

The requirements and conceptual design proposals for simplified small and medium sized light water reactor for the next generation of power plants were presented. In addition, a conceptual design for a pressure tube reactor, which avoids the need for emergency core cooling systems, was described.

Currently, advanced ALWRs have been sucessfully developed in Japan, and two advanced BWR units are now under construction. In addition, further developments in reactor protection are being advocated for PWRs, involving simplified active safety systems plus passive safety systems for severe accidents. Issues on the future labour force, etc., were discussed.

The results of a study of Swiss (nuclear) specialists views on the weight to be attached to future reactor issues was presented. To keep the nuclear option open, a call for effort to be focussed on stringent safety regulations was seen as necessary.

The US designs of ABWR and SBWR plants were described as examples of advanced light water reactors designed to address the needs for the next generation of power plants. Key features of the plants were discussed.

Commercializing of large evolutionary AWRs was outlined and a description of the design of CE System 80+ which is currently going through a regulatory review, was given. The rationale for this design approach which includes the addition of specific passive safety features, was presented.

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* Note the paper from the Republic of Korea was presented in Session 3.

The main features of the Westinghouse AP-600, a revolutionary PWR design, were described. The feasibility of using passive safety systems to mitigate the consequences of both design basis accidents and severe accidents was reviewed. The concept as proposed gives confidence that the plant could be expected in the mid-1990s.

The main development effort and the progress made in the USSR toward a future pressurized water cooled reactor design were described. Information included the use of horizontal steam generators and the operational experience gained with the current series of WWER type reactors.

The design status and perspective for advanced water cooled reactors in Czechoslovakia were described. The necessity for substantial improvements in safety and economics for future nuclear power plants was recognized previously and a brief description plus the main characteristics of a medium sized NPP (about 600 MWe) were presented.

CONCLUSIONS

A diversity of viewpoints and opinions was offered on this topic (both by presenters and the TCM participants). The numerous points made were widely supported by the meeting suggesting some level of consensus. In this Session the following points were made:

- There was widespread agreement on the need for clearly stated, technically sound design requirements for ALWRs and the papers presented were consistent in their presentations of the requirements needed. Greatest interest was shown in items like design and construction simplification, reliance upon proven technology safety aspects and the importance of the man-machine interface and human factors.

- There is a growing consensus on the safety objectives/targets for new plants (e.g., International Nuclear Safety Advisory Group (INSAG) and EPRI ALWR Utility Requirement Documents), but it is still an open issue as to how such targets can be met or how they are to be measured and regulated (i.e., deterministic and/or probabilistic design methodology)

- There was also agreement on the necessity of achieving public acceptance, as a precondition for building new plants. It was also noted that a nuclear power plant that would not require relocation of people under accident conditions would represent a significant and desirable improvement from the "public acceptance" point of view.

- Evolutionary plant designs were generally seen to represent the major option for near term development whereas the more innovative types will need more time before commercial availability.

- Concepts making use of passive systems and/or components have merit, but (in the view of many participants) only to the degree that they result in demonstrable improvement.

- Most participants felt that simplified evolutionary or passive plants employing proven technology will lead to real improvements (e.g. in safety and cost). Simplification, while supported in concept, needs better definition. Simplicity from the standpoint of the operator is considered very beneficial to safety.

- It was generally agreed that there may be merit in designs which employ active systems for common and high frequency events and passive systems for low frequency event prevention and/or mitigation.

- It was agreed that testing of ALWR features will be extremely important, particularly for innovative or passive safety systems. Requirements regarding the amount of testing of such systems, and associated test methods (scaling, modelling, etc.) before plant construction are not well established and warrant specific (further) attention.

RECOMMENDATIONS

1. It is recommended for the IWGATWR that an activity be initiated with the aim of defining a consistent and general set of criteria for the assessment of concepts with regard to degree of safety, as well as simplicity and concept maturity.

2. This TCM was considered by the participants to be a very effective forum for constructive, open dialogue. It is recommended that follow-up meetings be held, similar in structures that is relatively small group, international and diverse attendance, and stressing technical focus. Suggested scientific topics for future meetings include on-going testing and experimental programs and the concept assessment criteria of ALWRs.

SUMMARY OF SESSION 2

CONTAINMENT SYSTEMS
Chairmen: Mr. P.J. Meyer, Mr. Y. Dennielou

The papers presented in this Session dealt with the main issues associated with reactor containment.

The consequences of severe accidents to the population living in the vicinity of a nuclear power plant were outlined. Ideas on how these consequences may be minimized were given. This could be achieved by building in bigger safety margins through for instance a conservative design of containment and/or improved containment systems using passive safety features. The resulting longer grace periods possible would allow considered action to be taken if and when necessary.

Requirements and standards were discussed and proposals made for these to be internationally harmonized to provide uniform design standards with addtional guidance on how to meet them being important.

The philosophy of the defence in depth concept (barrier concept) was outlined and it would still remain valid for future systems, with accident prevention and mitigation continuing to be considered for all future containment designs.

In considering containment design it was noted that neither the release nor the probability of a release, of radioactivity as a consequence of a wide range of accidents should result in the need for evacuation of the public. However an evacuation plan may be established on request of the appropriate authorities.

Presentations were made on the wide range of tests which have been or are being planned for advanced passive containment systems; some of these have already been completed. Boundary as well as new design parameters have been specified for each different design proposal. Respective computer programmes are being both developed and improved and will be qualified by test results. Sensitivity analyses presented gave details of the influence of physical properties on containment configurations.

Future theoretical development programs as well as new tests were described which may lead to further optimization of containment and material configurations, of specifications for and simplification of, containment systems, leading to a reduction in the number of penetrations and higher reliability.

The need for developing new codes or for adapting existing codes calculating physical and thermohydraulic phemenoma was outlined along with the need for such codes to be verified against test results. Codes have been developed for specific purposes in containment design i.e. multicompartment system, and their adaptation to current and innovative containment systems was considered to be possible.

It was noted that the leak-tightness of the containment, under both normal and accident conditions was one of great importance. The different modes of plant operation each rely upon a certain number of operating systems which penetrate the containment (electrical, mechanical, valves, hatches, etc.), all of which need to be qualified as to their functionability and leak-tightness. Validation of the proposed solutions should be by either full-scale or reduced-scaled tests. For some advanced NPPs passive containments were described where the decay heat is transferred within and removed from the containment by internal natural circulation and by external natural convection respectively. New safety features will need to be qualified by test and/or calculation sufficient to meet the necessary licensing requirements.

The role of containments for existing plants, which are designed to withstand design basis accidents, defined by pipebreaks in the primary systems as well as specified steamline and feedwater line breaks, was discussed against possible future objectives. To minimize the risk of loss of containment integrity the existing safety margins may be increased by either increasing the containment volume or by the use of containment venting. It was noted that the latter is not contemplated for advanced NPPs. Accident management planning within future reactor safety margins would have to be analyzed very carefully and tests are necessary to justify such planning. In the construction of NPP the role of the site subsoil conditions in NPP construction was also reviewed. Calculations necessary to evaluate the effect of different soil conditions were discussed.

CONCLUSIONS

The design basis for current reactors containment is a double-ended break of primary piping based on already existing rules and regulations. This provides protection over a wide range of accident scenarios.

The next generation of reactor containment systems and structures must be designed to assure their design functions for the whole spectrum of accidents including severe accident consequences. For the containment layout and design respectively uncertainties and/or margins must be carefully investigated and included in the design.

Physical and thermo-mechanical behaviour and the source term assumptions, as used in designing the containment, should be considered for all accident conditions, including severe accidents. The input parameters for such computerized analyses must be carefully investigated and the respective codes qualified. International consensus should be reached.

The design of the plant should provide the technical bases to make public evacuation unnecessary. However, a simplified Emergency Planning could be adopted by decision of Administrative Authorities.

International consensus should be reached regarding design and layout requirements.

The containment should be designed as soon as possible with a passive safety feature. Credit can be given in the safety analysis for accident management actions after a certain grace period.

For the selection of the passive containment systems extensive studies are needed considering e.g. concrete containments with liner, prestressed concrete, freestanding cylinder or sphere, single, against double containment. These analyses have to consider carefully influences from inside as well as from outside.

The different components constituting the containment must be qualified under accidental conditions to ensure integrity and function under short and long-term conditions (such as penetrations and air locks). Proper diversification at sensor and logic level should be taken into account. Model tests of complete containment systems, as well as of single components, may be necessary to demonstrate their reliable functions (under accidental inner as well as outside events such as earthquake, missile protection).

For accident management procedures the possibility of monitoring the state of components during accident evolution should be considered as highly desirable.

In addition to the technical, safety-related and design requirements the economic viability needs to be evaluated, to establish the competitiveness of nuclear as opposed to conventional energy sources, a very important and crucial point for the future introduction of advanced reactors.

Licence criteria consensus for systems such as the containment by national authorities is essential in the frame of wide acceptance of the next generation of NPPs.

RECOMMENDATIONS

It is recommended for the IWGATWR:

- To consider setting down the requirements for new containment designs.
- To rationalize the containment design features and characteristics for future reactors.
- In a planned IWGATWR TCM meeting to consider design aspects for the advanced reactor containments and to identify to generally accepted technical solutions applicable to future reactors.
- TCM or CRP to consider code development needs for the different containment designs and for varying accident conditions. The targets should be internationally agreed and accepted code packages.
- TCM on economic comparison of containment systems with the goal of identifying realistic cost reduction proposals and internationally acceptable optimized containment concepts (design, assembling, modularization, testing, in-service inspections).

SUMMARY OF SESSION 3

SAFETY SYSTEM AND ANALYSIS
Chairmen: Mr. Y. Oka, Mr. Z. Mlady

The Session provided a complete picture of the status of the research, development and design activities, referring to innovative systems and components, currently underway in many Member States.

Nine papers were presented and discussed during the Session, from France, Germany, Italy* and Sweden.

The results of a systematic investigation of the role of passive systems in accident mitigation and severe accident mitigation for possible implementation in PWRs, as performed by Nuclear Power International (NPI), was presented. The merits of such features and their definition were investigated to assess their suitability for large sized PWRs. The results pointed to a positive safety gain from the use of a secondary side safety condenser (SACO), to replace the existing emergency feed water system.

A possible role for completely passive emergency shut-down and decay removal systems for PWRs was presented. Features developed for the MARS reactor concept were explained in some detail.

Studies on the design requirements of a steam ejector used as a passive component to supply high pressure water as a function of both water discharge pressure and steam pressure were described. Studies relating to an Isolation Condenser and a Passive Containment Cooling System aimed at providing a 72 hour grace period were also outlined, together with expected benefits.

The design of a proposed back-up protection system using ladder shaped module technology was presented. This aimed to avoid a common mode failure of the main protection system leading to an un-tripped faults.

The design philosophy of the PIUS plant, together with some of the experimental tests performed and test results obtained to support the design development were presented.

The need for a drastic design change "yielding a quantum jump" in perceived safety and reduced plant complexity was claimed as a prerequisite for a real revival of the nuclear option. The PIUS plant design has been proposed as an example meeting this claim.

The experimental activities presently under way in Italy to assess the thermal-hydraulic behaviour of innovative reactors were presented. These included the gravity driven reflood experiments performed in the PIPER-ONE facility and the tests performed at the SPES facility. Code predictions of the test results were reported.

CONCLUSIONS

The findings of Session 3 correlate closely with the results of the other sessions and only the most relevant findings are listed here.

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* Some of these papers were presented in Sessions 2 and 4.

Regarding the introduction of passive systems to replace active systems in nuclear power plants, further scientific investigations are needed. Some aspects (pros and cons) are listed below.

<u>benefits:</u>

1. allow for enhanced simplification
2. introducing higher degree of diversity
3. decreasing needs for safety-grade electrical power
4. increasing grace period
5. less dependence on human efforts in operation, and maintenance

<u>concerns:</u>

1. slow response
2. poor flexibility
3. functional uncertainty
4. testability
5. controllability
6. licensability.

SUMMARY OF SESSION 4

PHYSICS AND THERMOHYDRAULICS
Chairman: Mr. M. Cumo, Mr. J. Rixon

The Session discussed the way forward to more forgiving reactor core designs, through increasing the level of accident prevention rather than protection itself, and the behaviour of reactors with passive safety features for both reactor shutdown and decay heat removal.

Eight technical papers were presented covering the above topics, by organizations from Argentina, Brazil, France, India and Italy. In the following, a short synthesis of Session 4 presentations is given.

The ways available for the optimization of uranium consumption and economic benefits achievable by the use of different enrichments in an existing Heavy Water Reactor (1179 MW(th)) were presented and discussed.

The increased safety level achievable through the use of "cold fuels" and burnable poisons in advanced reactors was reviewed, along with reactivity accidents at zero/low power for the SBWR and the AP600. The possibility of preventing severe fuel damage by using operating procedures in the SBWR was proposed.

An analysis of the length of fuel cycles with changes in Plutonium enrichment was presented in association with thermal hydraulics for the ABWR.

The work carried out on the thermal-hydraulic analysis of core flow distribution and the possibility of maintaining the coolant flow through the core by natural circulation, in a heavy water moderated reactor, was presented and discussed.

Presentations were given on the application of the RELAP5/Mod2 code to the analysis of thermal hydraulic transients in the SBWR, AP600 nuclear reactors and the PIUS system. The RELAP5 capabilities and limits were discussed and the problems encountered in its use for natural circulation conditions outlined. The need for seperate effects tests was presented.

CONCLUSIONS

In general the core-physics and thermohydraulics of advanced reactors utilizing natural circulation and/or convection are at the stage where an understanding of a bigger range of parameters of the physical variables of already known phenomena is needed. New tests in this area are advisable.

Best estimate codes which coupled neutronics and thermohydraulics are now a possibility to be considered due to improved computational capabilities. Their development and application to advanced reactors is highly advisable and this should be an area for further considerations.

Some further consideration needs to be given toward a better understanding of accidents beyond the design basis for ALWRs. Some work in this area seems necessary.

RECOMMENDATIONS

1. Further work is needed in relation to natural circulation/convection to obtain a better understanding of a wider range of core physics and thermohydraulic parameters of already known phenomena.

2. Further considerations should be given to the development of best estimate coupled neutronic-thermohydraulic codes.

GENERAL CONCLUSIONS AND RECOMMENDATIONS

CONCLUSIONS

1. There is a need for clearly stated and technically sound internationally agreed design objectives/targets for advanced reactors.

2. While there is a growing consensus on the need for internationally agreed safety objectives/targets for new plants it is still an open issue as to how such targets can be met.

3. Advanced evolutionary reactor designs should utilize as far as possible, passive safety features instead of active systems, but only to the extent that safety is either maintained at the level attained in the best operating plants or it is improved.

4. For revolutionary designs of NPPs which mainly incorporate passive safety systems consideration needs to be given to the safety status and performance required of other supporting active systems which may contribute to meeting the overall safety objectives/targets.

5. Attention needs to be paid to the need for testing innovative and/or passive safety features.

6. In considering the design of future nuclear power plants, special attention should be given to the design of containments which should cover the whole spectrum of accidents, including severe accidents. The available margins to unacceptable performance, available grace periods and accident management measures should be included.

7. The design of the plant should provide the technical basis for making public evacuation measures unnecessary. However, simplified emergency planning could be adopted by decision of the appropriate authorities.

8. There is a need for international consensus on containment design and layout which may be met by an IAEA NUSS Series No. 50-SG-D12.

9. Extensive studies are needed to aid in the selection of passive containment concepts, to ensure their integrity under both short- and long-term conditions.

10. The economic viability of advanced reactors with the new design requirements for containment for systems and for equipment needs to be evaluated and demonstrated.

11. Discussion is needed to clarify and identify the main problem areas to allow useful discussions to be held at future IWGATWR TCMs.

12. In the area of core physics and thermohydraulics where natural circulation dominates there is a need for further R&D work to obtain a better understanding of a wider range of parameters as well as appropriate tests.

13. Development and application of best estimate codes in "coupled-neutronics and thermohydraulic" should continue to be aimed at improving computation for advanced reactors.

RECOMMENDATIONS

1. Special support should be given to the International activities under way aimed at harmonizing safety standards/goals and how they are to be implemented.

2. Requirements for new containment designs for advanced reactors should be documented.

3. There is a need for continued activities to review the on-going activities in the area of containment designs, safety objectives/targets and economics.

4. Further work should be persued to

   a) obtain a better understanding of a wide range of core physics and thermohydraulic parameters associated with natural circulation phenomena,

   b) develop the best estimating coupled neutronic thermohydraulic codes for advanced reactors,

   c) identify the testing needed to validate the claimed performance of innovative passive safety features.

DEVELOPMENT PROGRAMMES AND CONCEPTUAL DESIGN OF ALWRs

(Session 1)

Chairmen

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PROGRESS ON MEETING UTILITY REQUIREMENTS FOR ALWRs

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Abstract

The advanced light water reactor (ALWR) has been under development since 1981 in the United States. The paper briefly reviews the progress to date. The industry is working towards meeting the goal of the Nuclear Power Oversight Committee (NPOC) Strategic Plan for Building New Nuclear Power Plants which is to provide the U.S. utilities with a viable ALWR option by 2000. A key aspect of the Plan is the degree of commitment to standardization throughout the design, construction and operation of the ALWRs. Emphasis is placed upon the Utility Requirements Document (URD) and its role in this overall plan.

The URDs embody the plant-wide needs of the utilities for the new ALWRs. The role, scope and top tier requirements of the URDs are summarized. The overall safety objectives which exceed those required by regulation are presented. Containments for the new reactors have increased margin over current designs and numerous engineering features are specified to address key challenges potentially imparted by severe accidents. The status of NRC review of the URDs and current schedules are provided. Finally, future ALWR activities are discussed.

Introduction

The commitment of the U.S. utilities to reopening the nuclear option is summarized by the Nuclear Power Oversight Committee in the Strategic Plan for Building New Nuclear Power Plants.[1]

> "The extensive operating experience with today's light water reactors (LWRs), and the promise shown in the recent technical developments, leads the industry to the conclusion that the next nuclear plants ordered in the United States will be advanced light water reactors (ALWRs). Two types are under development: units of large output (1300 MWe) called "evolutionary" ALWRs and units of mid-size output (600 MWe) called "passive" ALWRs."

[1] "Strategic Plan for Building New Nuclear Power Plants," prepared by the Nuclear Power Oversight Committee, November 1990.

Table 1
NPOC Plan
Building Block Summary

Prerequisites from Ongoing Programs
(1)* Current Nuclear Plant Performance (Licensees)
(11) High Level Radioactive Waste (EEI-ACORD)
(12) Low-Level Radioactive Waste (EEI-ACORD)
(13) Adequate, economic Fuel Supply (EEI)

Generic Safety/Environmental Regulation & Industry Standards
(2) Predictable Licensing & Stable Regulation (NUMARC)
(3) ALWR Utility Requirements (EPRI-USC)

Project-Specific Activities
(4) NRC Design Certification (Plant Designers)
(5) Siting (EPRI-USC/NUMARC)
(6) First-of-a-Kind Engineering (EPRI-USC)
(7) Enhanced Standardization Beyond Design (NUMARC)

Institutional Steps
(8) Enhanced Public Acceptance (USCEA)
(9) Clarification of Ownership and Financing (EEI)
(10) State Economic Regulatory Issues (EEI)
(14) Enhanced Governmental Support (ANEC)

* Building Block Number

The NPOC Plan identifies 14 enabling conditions (or building blocks) necessary to reopen the nuclear option in the U.S. Those are listed in Table 1.

The Plan:

- identifies all the significant enabling conditions (technical/industrial, regulatory, environmental, financial, legislative/legal, organizational, political, and public acceptance) which must be met to achieve the goal;
- assigns lead and supporting responsibilities to the appropriate organizations or standing committees in the industry to develop and implement an action plan for achieving each condition;
- fosters joint and coordinated efforts between government and industry which would enhance implementation of the strategies and provide for shared resources.

Of particular technical interest to this paper are the building blocks 2, 3, 4, 6 and 7. The Plan was originally published in November 1990 and is to be updated in November 1991. The Plan includes a schedule which reflects the goal of having an ALWR which could be built and ready for operation on or about the turn of the century. This would require that an ALWR order be placed in the 1995 timeframe. Key dates from the Plan that support the 1995 order date include:

- Safety Evaluation Report (SER) on evolutionary plant Utility Requirements Document (URD) - 3/91
- SER on passive plant URD - 2/92
- Design Certification (DC) of the ABWR - 6/92
- DC for ABB-CE System 80+ - 12/92
- DC for AP600 - 12/94
- DC for SBWR - 2/95
- Complete first-of-a-kind (FOAK) engineering - 6/95 (depends upon design)

Some of the Plan dates have slipped and the situation with the review and approval schedule for the URDs will be discussed later in this paper.

Standardization

One of the central themes of the NPOC Plan is that of standardization. Utilities believe that the viability of ALWRs will be greatly enhanced by standardization in design and operation. In order to convey the importance of standardization, NPOC developed "The Position Paper on Standardization for Building New Power Plants."[2] The Chief Executive Officers (CEOs) of all US nuclear utilities approved this position paper before publishing.

The US Utility commitment to standardization is summarized in this paper as follows:

> "Nuclear power plant standardization is a life-cycle commitment to the uniformity in the design, construction and operation of a family of nuclear power plants. Rigorous implementation of standardization is expected to achieve the efficiency and economy typically associated with increases in scale or breakthroughs in technology."

The process of standardization includes four stages: the foundation level established by the ALWR Utility Requirements Documents; additional standardization through the NRC Design Certification Process; commercial standardization (engineering beyond DC); and enhanced standardization beyond design. The primary motivation for standardization is its economic benefit, and standardization's contribution to safety is also well recognized.

[2] "Position Paper on Standardization," prepared by the Nuclear Power Oversight Committee, April 1991.

For each family of ALWR plants, the following principles are proposed:

- Standardization will be maintained throughout the construction and operating life of the family of standardized plants. An owner/operator structure will be established with clear mechanisms for maintaining standardization including a formal process for the review of proposed modifications or other departures.
- Standardization within systems, structures and components needed for safety will be subject to regulatory acceptance. Standardization within systems required for reliable power generation will be maintained by all the owner/operators of a family of standardized plants or by the organizational entity established and charged with that responsibility by all the owner/operators of that family.
- The plant design will be transferable, without alteration, to any site within the design envelope for the family of plants.
- Layouts of major systems and components will be identical. Plant layout should preclude the use of any shared equipment between units.
- System functional requirements will be identical, with siting consideration as the only acceptable reason for differences.
- Major structural, mechanical, electrical, or I&C components (including installed spares) essential to nuclear safety or reliable power generation will be identical.
- Functional, physical, and interface requirements for bulk commodities and for other components will be identical. The specifications should identify critical design characteristics to allow selection of the component that best meets the requirements and allow qualified substitutions without modifying essential identical components.
- Each plant within a family will be built to construction drawings and specifications that are identical to the extent noted above. It is recognized that some differences will arise due to site-specific requirements and variations within acceptable construction tolerances.
- Permanent modifications to systems, structures, or components essential to nuclear safety or reliable power generation will be made only after review and approval of the organizational entity established and charged with that responsibility by all the owner/operators of a family of standardized plants. Such review and approval by the family of plants may be deferred in the case of an emergency modification. However, modifications to replace failed or obsolete components should maintain standardization, or if necessary, be planned so as to recover standardization as the same components are replaced in the other plants within the family.

- Standardization beyond hardware design will be implemented in such areas as training, maintenance and operating procedures, quality assurance, licensing, spare parts management and outage management.

From a technical standpoint, the development of Utility Requirements (Building Block 3 - NPOC Plan) is of key importance to the U.S. strategy.

UTILITY REQUIREMENTS

Since 1982, the U.S. utilities have been leading an industry-wide effort to establish a technical foundation for the design of the next generation of light water reactors in the United States. Since 1985, the utility initiative has been effected through a major technical program managed by the Electric Power Research Institute (EPRI); the US Advanced Light Water Reactor (ALWR) Program. In addition to the US utility leadership and sponsorship, the ALWR Program also has the participation and sponsorship of numerous international utility companies and close cooperation with the U.S. Department of Energy (DOE). One of the main goals of the ALWR Program has been to develop a comprehensive set of design requirements for the advanced LWR. The Utility Requirements Document[3] defines the technical basis for improved and standardized future LWR designs.

Although established nearly eight years before the NPOC initiative in setting building blocks for the revitalization of Nuclear Power, the ALWR Program, and particularly the Requirements Document effort fits naturally and fully into the broader NPOC program.

Scope

The Requirements Document covers the entire plant up to the grid interface. It therefore is the basis for an integrated plant design, i.e., nuclear steam supply system and balance of plant, and it emphasizes those areas which are most important to the objective of achieving an ALWR which is excellent with respect to safety, performance, constructibility, and economics, the document applies to both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs).

There are numerous basic design policies underlying the ALWR URD, established by the Utility sponsors. Among these:

Simplification — Simplification is fundamental to the ALWR success, is pursued with very high priority, and is assessed primarily from the standpoint of the operator.

[3] ALWR Utility Requirements Document

Design Margin — Like simplification, design margin is of fundamental importance and is pursued with very high priority. The ALWR is to be a rugged, forgiving design. Its design incorporates margins which go beyond regulatory requirements and these margins are not to be traded off or eroded for regulatory purposes.

Human Factors — Human factors considerations will be incorporated into every step of the ALWR design process but especially in the main control room design.

Safety — The ALWR design will achieve excellence in safety for protection of the public, on-site personnel safety, and investment protection. (This is discussed in more detail below.)

Design Basis Versus Safety Margin — The ALWR design will include both safety design and safety margin requirements (discussed later.)

Regulatory Stabilization — ALWR licensability is to be assured by resolving open licensing issues, appropriately updating regulatory requirements, establishing acceptable severe accident provisions, and achieving a design consistent with regulatory requirements.

Standardization — The ALWR Requirements Document sets the technical foundation for standardized designs, consistent with the NPOC Policy Paper. The ALWR design is intended to make extensive use of the extraordinary data base of information and lessons learned from 30 years of experience in operation of over 100 light water reactor power plants in the US and many more overseas.

Proven Technology — Proven technology will be employed (requiring no prototype) throughout the design.

Maintainability — The ALWR will be designed for ease of maintenance to reduce operations and maintenance costs, reduce occupational exposure, and to facilitate repair and replacement of equipment.

Constructibility — The ALWR construction schedule will be substantially improved compared to previous U.S. experience, due to use of more constructible plant configuration, better construction methodology and completion of most of the engineering prior to construction.

Economics — The ALWR is to be designed to be economically attractive in comparison to other central station alternatives on both a near-term (10 year) and life cycle (60 year) basis.

Good Neighbor — The ALWR plant will be a good neighbor by minimizing radioactive and chemical releases to its surrounding environment and chemical releases.

Of particular interest are the Requirements Document treatments of safety and containment performance.

Safety

It is ALWR Program policy to require excellence in safety and environmental performance. In this regard ALWR safety goals have been established to limit the core damage frequency to less than 10^{-5} per reactor year and to limit the site boundary whole body dose to less than 25 rem for those severe accidents with cumulative frequency which exceeds 10^{-6} per reactor year. These safety goals will be demonstrated by plant specific probabilistic risk assessment (PRA) as part of the design process.

In addition to the ALWR safety goals, containment performance requirements have been stipulated to provide high assurance of containment integrity even in the event of a severe accident. Further, the combination of safety goals and containment requirements provide a technical basis for substantial simplification in emergency planning, at the same time providing very high protection of public safety.

The ALWR safety design goal is achieved through an integrated approach which includes three overlapping levels of safety protection: accident resistance, core damage prevention, and accident mitigation. These levels of safety protection reflect the defense-in-depth philosophy. Accident resistance refers to minimizing the frequency of initiating events which could lead to a demand on engineered safety systems. This is accomplished through simplicity, increased design margin and other intrinsic characteristics to minimize frequency and severity of initiating events. Core damage prevention refers to engineered systems which prevent initiating events from progressing to core damage. Mitigation refers to systems and structures to contain fission products released from a damaged core. Table 2 lists examples of ALWR requirements established to address accident resistance and core damage prevention. It is important to note that requirements in the first two areas make any core damage event very low in probability for ALWRs.

ALWR safety requirements encompass Licensing Design Basis (LDB) and the Safety Margin Basis (SMB). The LDB is the set of ALWR design requirements which satisfy NRC regulations, including LDB transient and accident events, in the Code of Federal Regulations and associated regulatory guidance. Only safety-related equipment is assumed to be available for purposes of meeting regulatory limits (with the exception of a few multiple failure events such as ATWS and station blackout where limited credit for non-safety-related equipment is allowed.)

The SMB is the set of ALWR requirements which addresses severe accidents by providing margin beyond the LDB. The severe accident protection incorporates

Table 2

Examples of ALWR Requirements Addressing Accident Prevention

- Accident Resistance (i.e., minimizing frequency and severity of initiating events)
  - No recirculation piping in BWRs
  - No loop seal and minimal welds in PWR primary system piping
  - Greatly improved control room
  - Improved accessibility and design for maintenance
  - Improved resistance to embrittlement in reactor vessel
- Core Damage Prevention
  - High reliability decay heat removal
  - High reliability ECCS
  - Increased RCS inventory delaying core recovery
  - Increased time for operator action

NRC policy level guidance and provides increased assurance of containment integrity and low release of radioactivity during a severe accident. SMB analyses are typically best-estimate evaluations which meet utility specified limits. Reasonable credit is taken for non-safety-related equipment. While much has been learned in the last decade about severe accidents through experimental and analytical work, the SMB calculation methods and assumptions necessarily involve more engineering judgment than the LDB.

The accident resistance and core damage prevention requirements result in ALWR designs that are very safe from the following perspectives. First, the designs will result in fewer and less severe initiating transients. Second, given an initiating event, the ALWR design will have a much reduced likelihood of the transient leading to core melt. Furthermore, the very robust containment for ALWRs can tolerate "credible" severe accidents, those with probabilities of occurrence greater than 10^{-7}/year.

Containment Requirements

The Requirements Document specifies that containment systems and structures provide sufficient design margin to accommodate severe accident loads such that ASME Service Level C (steel), Factored Load 1.0 (reinforced concrete) limits are not exceeded. Those load levels are specified to insure leak-tightness of the containment, therefore the margin against gross rupture is considerably greater than that typically required by design codes. In addition, containment failures prior to or coincident with a severe accident, such as containment bypass, shall be precluded by design.

Related top tier requirements include for evolutionary plants:

- Electrical power supplies shall address all concerns about Station Blackout including alternate AC on-site generation

- Enhanced Man-Machine Interface Systems and Information Management Systems shall significantly improve the control room environment to simplify control and operations of the plant during transients, accidents and normal operation

For passive plants

- The plant shall satisfy the above severe accident limit for an indefinite time (72 hours without operator action and thereafter only simple operator actions).

- Leak-tightness shall be sufficient to meet applicable offsite dose limits
  - Containment system components for which a change of state is necessary to assure an intact containment (e.g., containment isolation valves) are not vulnerable to common cause failure because they are redundant and sufficiently independent from the systems whose failure could lead to damage.

In addition to these top-tier requirements, there are hundreds of requirements which address the various ways in which containment could be challenged. A comprehensive list of potential containment challenges is listed in Table 3. Requirements have been developed to limit the likelihood and effects of each of these challenges. For example, interfacing systems which could potentially experience reactor coolant system pressure (i.e., interfacing LOCAs) have double valve isolation and interlocks to limit the potential for the challenge as well as an increased pressure rating to accommodate the challenge should it ever occur.

The detailed requirements for each of the containment challenges are discussed in backup technical reports. These reports compile and evaluate the set of requirements for each containment challenge on an integrated basis.

The following safety-related observations are made for ALWRs.

- Accident prevention requirements make any core damage event very remote in probability. Notwithstanding this low probability, defense-in-depth containment performance is also provided.

- The ALWR margin approach is an appropriate means for addressing severe accidents outside of the conventional Licensing Design Basis.

- The systematic consideration of severe accidents with requirements to either preclude or accommodate the containment challenges provides a deterministic approach to containment performance consistent with the NRC position in

Table 3

Postulated Containment Challenges/Failure Modes

- *Challenges/Failure Modes that May Precede a Severe Accident*
  - Containment Isolation Failure
  - Interfacing System Loss of Coolant Accident (LOCA)
  - Blowdown Forces
  - Pipe Whip/Steam Jet Impingement
  - Multiple Steam Generator Tube Rupture
  - Containment Overpressure due to Anticipated Transient Without Scram (ATWS)
  - Containment Overpressure due to Catastrophic Reactor Pressure Vessel (RPV) failure
  - External Pressure Loading due to Partial Vacuum Conditions
  - Missiles from Internal (Plant) Sources
  - Tornado and Tornado Missiles
  - Site Proximity Hazards, Including External Missiles
  - Seismic Induced Failure

- *Challenges/Failure Modes Potentially Resulting from a Severe Accident*
  - High Pressure Core Melt Ejection
  - Hydrogen Related Issues (Deflagration/Detonation)
  - In-vessel Steam Explosion
  - Ex-vessel Steam Explosion
  - Noncondensable Gas Generation
  - Reactor Pressure Vessel Support Failure and Containment Basemat Penetration
  - Containment Sump Failure from Core Debris
  - Containment Overpressurization due to Core Debris Decay Heat Steam Generation
- Steam Generator Tube Rupture (SGTR) Failure from Natural Circulation of Hot Gases (PWR)

SECY-90-016[4]. The deterministic approach when coupled with the requirements for Plant Designer PRA-based evaluations, is consistent with the NRC Severe Accident Policy Statement.[5]

[4] SECY90-016

[5] NRC Severe Accident Policy Statement

Schedule

The schedule established for Building Block 3 in the NPOC Plan is being revised to reflect actual progress to date. The following table compares the original key milestone dates with the schedules recently published by the NRC in SECY 91-161[6]

Milestone Description	NPOC	NRC
NRC final Safety Evaluation Report (SER) on evolutionary ALWR Utility Requirements Document	3/91	8/92
NRC final SER on passive ALWR Utility Requirements Document	2/92	9/93

The present status of the Evolutionary Plant SER is that all but two (of fourteen) submittals have draft SERs prepared. There are several key issues that need resolution before the final SER can be approved.

For the Passive Plant, all requests for additional information (RAIs) are addressed on a schedule well ahead of the SECY-91-161 average of 90 days. Progress to date by NRC in review and evaluation of the Requirements Document has been disappointing. A major effort is being made to reach closure on key issues and recover some of the lost schedule.

THE NEXT STEPS

In order to meet the U.S. utilities needs for baseload capacity additions in the time period 2000-2010, the NPOC Plan schedule must be met. The following actions are necessary.

1. Close out issues and expedite the SERs for the evolutionary plant URD and the passive plant URD.
2. Expedite the DC process for the ABWR System 80+, AP600 and SBWR.
3. Initiate the first-of-a-kind engineering work as soon as practicable.
4. Complete the siting effort on its current schedule.
5. *Establish the infrastructure for the families of plants.*
6. Promote standardization to the extent possible through requirements, certification commercialization and beyond design.

[6] SECY91-161

REQUIREMENTS FOR NEXT GENERATION LIGHT WATER REACTORS FOR THE 21st CENTURY IN JAPAN

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Abstract

Although nuclear power generation with light water reactors has been successfully employed and developed in Japan, further improvements in economics and safety are desirable in order to achieve further development and maturity of nuclear power. At the present time advanced light water reactors have successfully been developed and construction of two units has started in Japan. Large LWRs of highly developed function and performance have been studied as a candidate for the next-generation power reactor adopting the newest technoligical concepts as for Advanced LWRs such as ABWR and APWR. On the other hand, innovative design ideas which differe from conventional concepts are being advocated abroad and in Japan.

Issues on the energy situation, labor force, etc. likely to arise in the 21st century in relation to the design of LWRs and of nuclear power development and requirements for Japan are discussed.

The anticipated social changes toward the future have been considered as the surrounding conditions for defining the required specifications of the new-concept reactors. The results of the discussion are summarized below for the items most likely to be encountered from the aspects of energy demand, labor issues and technical development.

1. Energy Consumption Outlook

1.1 World Energy Demand

(1) Population

The current world population (1990) is 5.2 billion of which 1.2 billion is in the developed countries and 4 billion in the developing countries (77% of the total).

According to a prediction by the United Nations, the population in the developed countries will increase slightly to 1.2 times the current level by 2025, but the population in the developing countries will increase explosively to 1.7 times the current level. Consequently, the world population will be 8.21 billions in 2025 with 84% in the developing nations. 2.8 billions of the 3 billion increase will be in the developing countries. The population problem in the 21st century will be attributed to the population explosion in the developing countries.

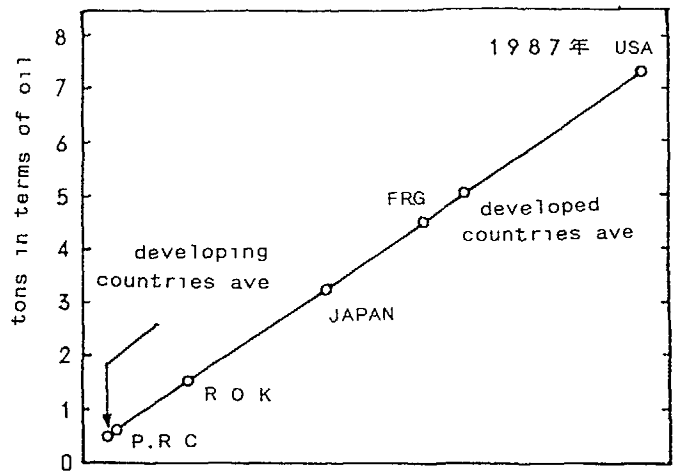

Fig 1 The per capita annual energy consumption

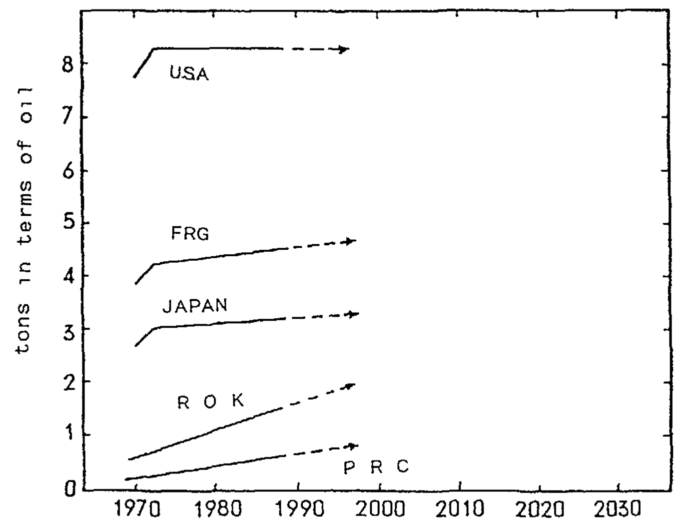

Fig 2 The changes in per capita annual energy consumption

(2) Energy Issues

(a) Per Capita Annual Energy Consumption

Figure 1 shows the per capita annual energy consumption for 1987 in various countries. The average consumption in the developed countries is approximately 5 tons in terms of oil and 0.5 ton in the developing countries, only 10% of that in the developed countries
The average values differ largely even among the developed countries (among the United States, West Germany and Japan). The consumption in Japan is less than half that of the United States and is approximately 60% of the average for the developed countries. Figure 2 shows the changes in per capita annual energy consumption in these countries. While the consumption is almost constant in U.S.A , in Germany and Japan, whose consumption levels are below the average for the developed countries, it grew steadily at 16% and 19% , respectively, from 1970 to 1987 On the other hand, per capita consumption in a newly industrialized country (REPUBLIC OF KOREA) and in a developing country (PEOPLE'S REPUBLIC OF CHINA) has remarkably increased by factor 2.5 during the same period.

(b) *Electricity to Primary Energy Ratio*

It can be deduced that the importance of electric energy will be further increasing after 2000 because of its convenience as an energy form. The portion of the primary energy used to generated electric power in the total consumed primary energy is currently 15% to 20% in the developed countries and 5% to 7% in the developing countries. This value must be increased to improve the living standards in the developing countries, by factor 1.5 to 2 under the assumption the whole world should reach the level of the developed countries by 2050.

(3) Uranium Resources in View of World Energy Situation

The energy consumption should continue to increase as mentioned above and part of it will be supplied by nuclear power generation. According to 1989 OECD report ('Uranium Resource,Production and Demand"), the confirmed deposit of uranium costing less than 130 $/kg (equivalent to 60 $/lb) is approximately 2,300,000 tons and estimated additional deposit will be approximately 1,300,000 tons.

According to the report, on the other hand, the cumulative uranium consumption by 2030 will be 2,400,000 to 3,090,000 tons if used only for LWRs, which is close to the amount of the reserves It may be not appropriate to judge on the basis of currently available data only since new mines have not been developed in these years, however consideratation should be made now for the possible problems arising when supply and demand becomes tight.

The keys to more effective use of uranium resources are,for LWRs, the development of high-burnup fuel, plutonium use in thermal reactors and an advanced flexible reactor core, and, for FBRs, the studies and development towards realization of the actual plants.

1.2 Energy Demand in Japan

(1) Long-term prospect for energy supply and demand in Japan

According to the long-term prospect for energy supply and demand published by the Advisory Committee for Energy (June 1990), higher growth rate of energy demand is anticipated in future despite the best efforts for energy-saving.

The nuclear energy could be the only non-fossil energy source reliable both quantitatively and economically to cope with the growing energy demand. 72,500 MW of nuclear power generation capacity is expected in the year 2010 with consideration to the feasibility of constructing new nuclear power stations (Figure 3). To achieve this target, about 40 1000-MWclass units must be newly built. It is anticipated that the construction of new plants with similar pace would be necessary also after 2010 .

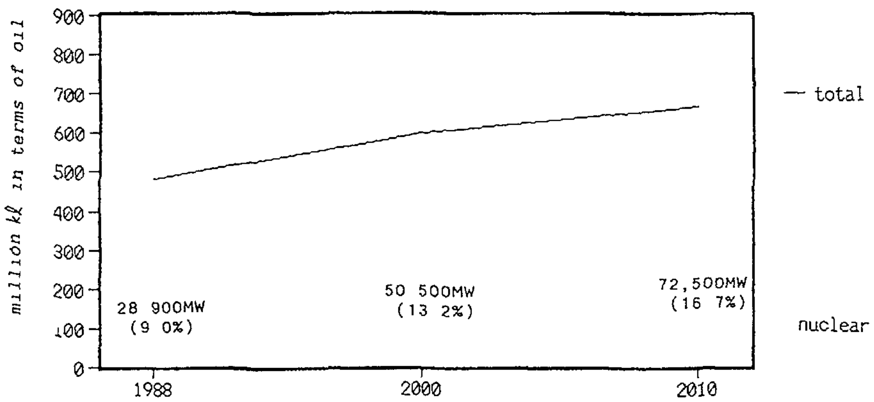

Fig 3 Long-term prospect for energy supply and demand in Japan

(2) Energy demand and Sitting Issues

On one hand the current situation of energy demand and the construction plan of nuclear power stations is as mentioned above, on the other hand, it is becoming more and more difficult to secure new sites for achieving the end.

Considerable effort must be made to develop the measures for solving the sitting issues. For this purpose it is necessary to increase the unit capacity and to develop nuclear reactors with fewer sitting restrictions thereby pushing forward the effective use of potential sites. It is also necessary to promote the PA activities to find new sites. There are various items to be discussed and studied for the promotion of the public acceptance The following items are based on the results of a public-opinion survey made by Prime MinisteruLs Office.

(a) It must be shown to the public that accidents cannot take place in nuclear power plants.

(b) The methods for management and disposal of radioactive waste must be clearly shown to the public.

(c) The fact that plant safety is properly secured and the possibility of human errors is minimized must be fully understood by the public.

Item (a) requires realization of highly-reliable plants. Item (b) requires promotion of PA activities based on the concrete and flexible R & D with particular emphasis on underground disposal. In addition, efforts must be made to reduce the radioactive waste generation amount by developing high-burnup fuel and highly-durable materials and also by reducing the activated materials. It is also necessary to examine the resume of activated materials. Item (c) requires R & D on highly-safe and easy-to-understand plants requiring less manual operations and incorporating redundant and passive safety-related systems.

2 Labor Problems

Figure 4 shows the changes in the working-age population in Japan. The available labor force is expected to decrease after reaching the peak in 1995, due to the falling birthrate.

A lifestyle survey targeted at new employees shows the general tendency to value private time and to put more importance on private life than on job, which is one of the reasons why annual working hours are becoming shorter in spite of decreasing working-age population.

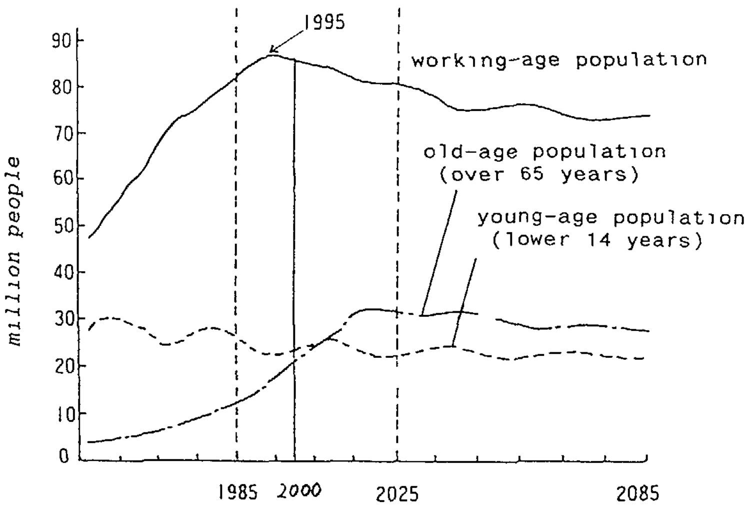

Fig 4 Changes in the working-age population in Japan

(Annual working hours in Japan are about 2100 hours while those in Europe and the United States are about 1700 hours. The working hours in Japan are expected to fall to the level in Europe and the United States in the near future.)

As demonstrated by the recent data on the employment-seeking of university graduates (engineering departments), there is a tendency for them to select companies which are not manufacturing industries. That results in difficulty for manufacturers to recruit appropriate human resources

Furthermore, young employees are feeling less resistance to changing their job and that makes it difficult for an employer retain them in a company.

The rapid rise of the average age of the working population will create larger working opportunities for older people and women workers will become more important.
Consequently, more and more older people and women are expected to join the working population to cope with the labor shortage. Foreign workers can be another remedy for the labor shortage.
Under such labor situation, the society may have to change so that women and older people can participate more easily in work. It is also necessary to make up for the labor shortage by applying labor-saving measures and automation. The number of nuclear power stations will increase in the 21st century under serious labor shortage More nuclear power stations will require larger number of nuclear power plant personnel (operators, maintenance personnel, designers, and construction workers). From the viewpoint of shortage of skilled laborers, it will be necessary to design the systems and components so that they can be operated by fewer people. For anticipated women and older workers, the facilities must be designed to be handled by people without significant technical knowledge. For this purpose, R & D must be conducted on personnel-saving plants incorporating automatic operation, inspection/maintenance robots and system simplification.

3. Required Specifications for New Reactor Concepts

3.1 Safety

(1) The defence-in-depth philosophy based on the current concept of safety functions (prevention of occurrence of abnormal events, prevention of their expansion, and prevention of abnormal radioactivity release) should be applied.

(2) Passive safety systems should be incorporated to make operators action unnecessary for an appropriate time, to give the operators sufficient time before initiating the counteractions, and to reduce the operator burden and consequently the possibility of human factors in accident cases.

(remark) Operators manual action should not be required for more than 72 hours. (EPRI specification)

3.2 Reliability

(1) Component failure frequencies should be reduced to assure plant reliability by applying passive and simplified features.

(2) Plants should be designed with larger design margin in order to be sufficiently trouble-torelant and to slow the trouble expansion.

3.3 Operability and Maintainability

(1) Higher operability and maintainability (easier operation and maintenance) should be achieved by applying expanded

automation, advanced man-machine interfaces, robotics and diagnosis technologies, and consequently the burden on operators and maintenance workers should be reduced.

(2) New technologies (materials and components) should be introduced to improve maintainability

(3) Plants should meet the requirements with respect to load-following capability.

3.4 Site Adaptability

(1) Plant design should be flexible enough to be adapted to various siting methods (underground,marine siting, etc.) and various site locations (e.g.,city suburbs) to expand the possible choice of new sites and to avoid over-concentration of sites.

(2) Flexibility/adaptability to various site conditions should be enhanced, e.g., by introducing seismic isolation technologies.

3.5 Constructibility

(1) Shorter construction period, consequently reduced construction man-power, should be pursued by expanding the application of prefabricated modules and large blocks as far as possible and by introducing large-scale construction machines.

(2) Learning effect should be taken into account more widely by standardizing the component design, etc.

3.6 Design Lifetime

(1) Plant lifetime should be extended to solve the siting difficulties and to reduce the amount of radioactive waste.

(2) Development should be made to establish the life-prediction/life-extension technologies for heavy components and systems and also to enable easier replacement of heavy components

(remark)The design lifetime should be about 60 years(EPRI specification).

3.7 Quality Assurance/Quality Control

(1) Extensive and thorough quality assurance should be implemented for entire plant life throughout design, operation and periodical inspection of the plant (establishment and documentation of necessary items such as design methods, verification and design modification).

(2) The experience of past accidents, troubles and plant operation should be duly reflected on the plant design.

(3) Easier quality assurance and control should be realized by introducing new technologies (improved welding techniques, advanced inspection methods, etc.), by reducing the safety system hardware as a result of application of passive and simplified features, and by reducing the site work with extended prefabricated modules.

3.8 Public Acceptance

(1) The plant should give an impression to the public that accidents/troubles cannot occur easily and sagety can secured easily if an accident does occur.

(2) The plant must coexist with the local community be in harmony with natural environment.

(3) The plant must not give a dangerous impression

(4) The plant design should minimize the troubles due to human factors.

3.9 Nuclear fuel

(1) The fuel and core should be handled easily including reprocessing with minimum necessity for reprocessing as the result of appropriate core design.

(2) The core and fuel design should make the best use of available uranium resources.

(3) The fuel should be high-performance fuel (PCI-free fuel) for rapid start/stop and load-following operation.

(4) The fuel design should incorporate passive safety technology.

3.10 Economics

(1) The systems and equipment must be simplified as much as possible and the physical number of systems, equipment and buildings must also be reduced to optimize the plant operating conditions (reduced capital expenditure).

(2) Equipment reliability and durability must be promoted and the plant must have high availability and high reliability factors as a result of extended continuous-operation periods

and reduced periodic inspection outages. The economics must also be enhanced by reducing the fuel cycle cost, number of personnel and exposure dosage. (reduced operation and maintenance costs).

(3) The economics must be enhanced by considering the plant service life.

(4) The goal must be to generate power at less cost than competiting power-generation methods.

3 11 Others

(1) No prototype reactor is required.

(2) The plant capacity should be large because new sites are hard to find.

(3) Consideration should be given to the power transmission systems because of the remote plant location and system instability resulting from the larger capacity.

(4) Promotion of thermal efficiency and effective use (multipurpose) of energy such as waste heat must be discussed and studied.

(5) The applicability of standard designs and equipment must be examined so as to simplify licensing procedures.

(6) The technologies (such as those related to water chemistry and reduced exposure dosage) gained from past operating experience must be utilized effectively.

CONCEPTUAL DESIGN AND DEVELOPMENT OF SIMPLIFIED LIGHT WATER REACTORS IN JAPAN

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Abstract

Japanese plant vendors have developed concepts of small and medium-sized light water reactors meeting international as well as domestic needs based on their reliable nuclear technologies in Japan. They are the Mitsubishi Simplified PWR (MSPWR) and the Hitachi Simplified BWR (HSBWR). The MSPWR is a new PWR concept. It has the innovative features of an optimum combination of active and passive systems and horizontal steam generators which enhanced its natural circulation capability. Active systems (charging/safety injection pumps and emergency diesel generators) are used to provide flexible operation and early termination of accident sequences. Passive safety features (quench pool, automatic depressurization systems and accumulators) are provided for long term cooling after a LOCA and make operator intervention unnecessary for 3 days. The HSBWR is at the design concept stage, has been conceptually designed. The major safety systems are an accummulated water injection systems as an ECCS and an outer pool which stands outside of the steel primary containment vessel, as a long term cooling systems after LOCAs. The safety systems have redumdancy in active and passive systems. The grace period is one day for core cooling and 3 days for the containment vessel heat removal.

I. INTRODUCTION

Presently, the global energy environment is falling into uncompromising situations. They are the dilemma between the energy supply and environmental pollution problems, such as annihilation of the tropical forests, the greenhouse effect caused by the increasing emission of carbon dioxide gas, consequently occurring sea water level rise and acid rain, and so on. They are accelerated by the increasing energy demand in the developed and developing countries. The dilemma between the energy supply and pollution problems can be solved by the world wide introduction of nuclear power as the energy source in the future. However some countries are reluctant to introduce nuclear power due to the spoiled public acceptance after the accident at Chernobyl.

The good records of nuclear plant operations greatly rely on human efforts which have been made continuously in the areas of design, manufacturing, operation, quality control etc. In other words, human efforts has been the key to excellent operational records. However, a big potential problem has emerged in such works related to construction, and maintenance in recent years. It is the chronic shortage of labor, which is generated by the changing society being composed largely of elderly people and the change of the workers' way of thinking to hate dirty and heavy works. These inclinations will accelerate in the future. Based on the background related to the above, nuclear power plants which do not heavily rely on personnel efforts or human sacrifice will be more desirable. Enhanced use of passive components in the future plants will be one of the way (but not only the way) to mitigate the problem.

Nuclear power is a major energy source in Japan and is promoted by Japanese energy policy. Emphasis has been put on large-sized light water reactors in the development, because of steady increase of power demand and pursuit of economy. As the 1,300 MWe class of advanced light water reactors, ABWR and APWR were developed. These advanced light water reactors were designed to increase both economy and reliability. They are expected to meet the Japanese electricity demand for the next 15 years. Small and medium-sized reactors, however, will complement the large sized reactors, since they will be able to comply with any power demand by making up the several rated electrical power outputs in series. In Japan, emphasis is put on economical improvement by simplifying the systems, and introduction of passive components and concepts of the simplified light water reactors is considered as on step for it. Further, the technologies and safety systems are considered also to help enhancement of the safety and economics of large-sized light water reactors in the future. In this paper, the research activities on the simplified nuclear power plants in Japan are described.

II. RESEARCH BY GOVERNMENTAL ORGANIZATIONS

In Japan, the LWR sophistication committee organized by MITI is responsible for developing the next generation LWRs. Reduction of human factor problems and research of the feasiblity of the passive (safety) systems are listed as one of the important research areas of the future LWR technology. It is pointed out the necessity for developing the "HUMAN FRIENDLY" plants in their operation and maintenance. The next generation LWR working group has been organized under the influence of the LWR Sophistication Committee.

RESEARCH AND DEVELOPMENT ACTIVITIES OF JAPANESE GOVERNMENT (MITI)

LWR TECHNOLOGY SOPHISTICATION COMMITTEE
Chairman: Dr. M. AKIYAMA
PROFESSOR
THE UNIVERSITY OF TOKYO

NEXT GENERATION LWR WORKING GROUP

FEASIBILITY STUDY OF ADVANCED SAFETY SYSTEM (NUPEC)	STUDY OF NEEDS AND ECONOMY OF THE NEXT GENERATION LWR REACTORS (IAE)	STUDY OF THE SIMPLIFIED LWRs WITH PASSIVE SAFETY FUNCTIONS (IRI)
Feasibility study of the simplified/passive safety systems applying to large sized LWRs (800-1350 MWe)	Review of the needs and utilization of the next generation reactors in terms of economy, site location and marketing	Review of the design of the simplified(small, medium-sized) LWRs

The following researches are being carried out.

(1) "Feasibility study of Advanced Safety Systems"
The applicability of advanced safety systems such as simplified systems and / or passive safety systems to large sized LWRs of 800 MWe to 1350 MWe is being investigated at the Nuclear Power Engineering Center (NUPEC)(1)
(2) "Study of the Needs and Economy of the Next Generation Reactors"
The utility needs of the next generation reactors is being surveyed by the Institute of Applied Energy (IAE). The design of the simplified LWRs will be evaluated in terms of the desired characteristics(2).
(3) "Study of the Simplified LWRs with Passive Safety Functions"
The design of the simplified LWRs with passive safety systems by the Japanese vendors is being developed and reviewed by the committee of the Institute of Research and Innovation (IRI). Those are MSPWR by Mitsubishi(3), SBWR by Toshiba / Hitachi / GE and HSBWR by Hitachi(4). The design of MSPWR and HSBWR are described in the chapter 4.

III RESEARCH BY JAPANESE ELECTRIC POWER UTILITIES

Japanese electric utilities are interested in large sized LWRs, more than 900 MWe. They are studying simplified LWRs as one of the approaches to improve the economy and the safety of the next generation LWRs. The emphasis is placed on the study of each sinplification passive technologies rather than the whole plant conceptual design. The studies are carried out by two groups. One is the group of PWR utilities , the other is the BWR s

The group of the PWR utilities are studying the simplified PWR (SPWR) by the help of Westinghouse and Mitsubishi. It is based on the AP-600 design. The increase in the power rating is pursued The feasibility of the 960 MWe reactor with three coolant loops is assessed, while the AP-600 is the two loop plants. The power density of the core is slightly higher than that of AP-600, but lower than the current ones. The pressure vessel is the same as that of the current 4 loop PWRs. The safety systems are similar to that of AP-600, for example the containment air cooling by the outside spray is adopted. The volume of the containment vessel will be 1 2 times larger than the current 3 loop PWRs. The canned motor pumps are used and the cross-over legs and their support structures are eliminated.

The group of BWR utilities are studying the technology of SBWR and HSBWR by the help of Toshiba and Hitachi and GE. They are the isolation condensers and the water pool cooling of the containment vessel. The experiments are being carried out at Toshiba and Hitachi The optimization study of the gravity driven ECCS system is also carried out using the TRAC code. The feasibility of the natural circulation core cooling of large sized plants is also being studied. The application of the technologies to the 1000 MWe class BWRs will be assessed.

IV. STUDY FOR SIMPLIFIED LWRs BY JAPANESE PLANT VENDORS

Japanese plant vendors are developing the simplified (small and medium-sized) LWRs meeting international as well as domestic requirements They are MSPWR (MS-300 and MS-600) by Mitsubishi[3] and HSBWR by Hitachi[4].

4 1 MS-300, MS-600, Mitsubishi's simplified PWR

Mitsubishi is developing a new PWR plant concept, Mitsubishi's simplified PWR, MS-300 and MS-600 It has the features of the hybrid safety systems (an optimum combination of the passive and active safety systems). In this plant concept, active safety systems are used for events of higher probabilities and the passive safety systems for the events of lower probabilities. These plants also adopt the unique horizontal steam generator systems. They are utilized for the core cooling after an accident

TABLE 1 PRINCIPAL PARAMETERS OF MS-300/600

PARAMETERS	MS 300	MS 600
Electrical Output (MWe)	~300	~630
NSSS Thermal Output (MWt)	854	1825
Reactor Type	PWR	
Reactor Core	Low Core Power Density	
Fuel Assemblies Type Number	 14 x 14 121	 15 x 15 157
Turbine	TC2F40	TC4F40
Containment Vessel Type	Steel Primary Containment With Concrete Filled Steel Secondary Containment	

PARAMETERS	MS 300	MS 600
<u>Reactor Coolant System</u> Number of Loops Operating Pressure (kg/cm²g)	 2 157	
Temperature Reactor Outlet (°C) Reactor Inlet (°C)	 325 0 302 5	 325 0 290 6
<u>Steam Generators</u> Number Type	 2 Horizontal U Tube Type	
Steam Pressure (kg/cm²a)	62	58
<u>Reactor Coolant Pumps</u> Number Type	 2 High Efficiency Type with Improved Seals	

The design objectives of the MS-PWR are to produce the plant that has improved safety, better economy, and higher reliability to satisfy the worldwide requirements. These objectives have been achieved by using the design philosophy of "simple is best". Both MS-300 (300 MWe) and MS-600 (600 MWe) are the two loop plants, using the same design concept. The larger versions are easily designed by using 3 or 4 loop arrangements. The characteristics of the MS-300 / 600 are given in table 1. The design features of the MS-600 are described in the following.

4.1.1 Reactor system

(1) Reactor core design 24 month cycle operation is possible in the MS-PWRs without the need for unusually high enrichments due to the use of low power density cores. Neutron economy and fuel costs are also improved by using radial neutron reflectors around the core.
(2) Top mounted incore instrumentation The incore instrumentation system has been changed to a top mounted system. As a result, the concrete structure in the lower part of containment has been reduced in size and the plant has become simpler and easier to maintain.
(3) Control rod drive mechanism (CRDM) coils A new technology has been adopted which uses high temperature windings for the CRDM coils and this eliminates the need for forced cooling of the CRDMs and simplifies the vessel head design.

(4) Steam generators The horizontal steam generators have been used and are part of the passive safety design. These steam generators have a number of advantages such as freedom from sludge build-up on the tube plates and a capability for withstanding any possible seismic events.

(5) Reactor coolant pumps The RCPs are of a high efficiency type. A monoblock ceramic No. 1 seal has been used and high temperature secondary seals are being developed to improve both the seal performance and the service life, as well as to give greater endurance during the loss-of-seal-cooling which occurs when all AC power is lost.

4.1.2 Safety system

The MS-PWR safety systems are based on the experience of existing plants but have been simplified to increase their reliability and safety. An optimized combination of active and passive systems (known as a hybrid safety system) has been adopted. It is shown in Fig. 1. Conventional active safety systems act effectively for many transients and accidents and have been improved by reflecting operating experience back into the design Since TMI accident there has been increasing concern about human errors. The ultimate protection should be provided by a simple and secure safety system We have selected a passive safety system for this purpose. Passive systems have many advantages such as simplicity, a reduced possibility of operator error, and to reject need for power However their weakness is that the post accident procedures are rigid and always involve flooding the containment vessel as a means of rejecting core decay heat to the ultimate heat sink. Consequently, the fundamental philosophy of the MS-PWRs is that the conventional active system will be used as the basic system but that it will be backed up by the secure passive safety system which needs no operator action and consequently is immune from operator's errors.

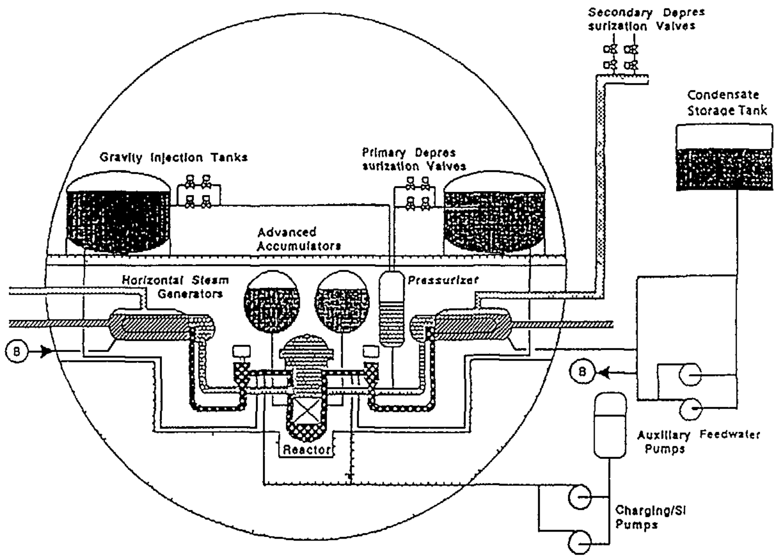

FIG. 1 MS-600 safety system

(1)Active systems

The active safety systems consist of combined charging / safety injection pumps and steam-turbine-driven and motor-driven auxiliary feedwater pumps together with small emergency diesel generators. They can handle very small break LOCAs, steam generator tube ruptures and non-LOCA transients. They function in the same way as they do on traditional PWRs.

(2)Passive systems

The MS-600 passive safety systems make maximum use of natural phenomena (natural circulation, gravity, gas pressure, etc.). These systems consists of the automatic depressurization system, the advanced accumulators, the gravity injection tanks and the horizontal steam generators.

To make sure that the reactor coolant system (RCS) is rapidly depressurized under any LOCA condition (other than a very small one), an automatic depressurization system has been provided For the MS-600, the steam generator cooling is used as part of the depressurization process. This system therefore consists of the primary and secondary depressurization valves and the horizontal steam generator. The primary valves blow down from the pressurizer steam volume into the gravity injection tanks. The secondary valves discharge to atmosphere through the relief valves on the main steam lines.

The advanced accumulators employ a fluidic flow control device When the accumulator is full, the water flow through the main flow standpipe and passes directly through the device to the outlet pipe When the level drops below the top of the standpipe, water enters the device from the side connection which set up a vortex that restricts the flow Thus high initial flow rate can be obtained followed by a prolonged iniection at a much lower rate. This matches the flow rates required in a large LOCA. The gravity injection tanks are adopted, which are used for core injection by gravity when RCS pressure is reduced to close to the containment pressure. Water spilling from the break will completely submerge the RCS.

The horizontal steam generators are adopted for MS-600 in order to provide natural circulation cooling under accident conditions. The horizontal arrangement avoids the possibility of gas bubble forming in the U-tubes which could obstruct natural circulation. The secondary side of the steam generators is supplied by gravity from a condensate storage tank. No operator action is required for 3 days. After 3 days, operators refill the condensate storage tank for continuous decay heat removal and available active systems are put in service.

(3)Safety analysis

Many kinds of safety analyses have been performed to check the new features of the MS-600 including,

(a)Natural circulation. The analyses to determine the natural circulation behavior after the LOCA showed that natural circulation in the reactor system is established and is stable under the required conditions. If the break is in the hot leg at a point near the steam generator, the natural circulation flow in the damaged loop is initially small. However as the temperature of the water in the pool rises, the flow increases and in the later stages the heat removal rate becomes almost as high as that of the undamaged loop.

(b)Blowdown and coolant injection by gravity for small break LOCAs: A small break in the pressurizer is the most severe case for depressurization of the reactor coolant system. Initially, RCS pressure drops rapidly due to the break, and when the primary side automatic depressurization system is actuated at 90 kg / cm^2, continues to decrease. Then the secondary side automatic depressurization system starts and both primary and secondary pressures fall rapidly. The accumulators start to discharge at 50 kg / cm^2 and the gravity injection tanks start to inject water to the reactor coolant system after about 900 seconds. The result is shown in Fig. 2.

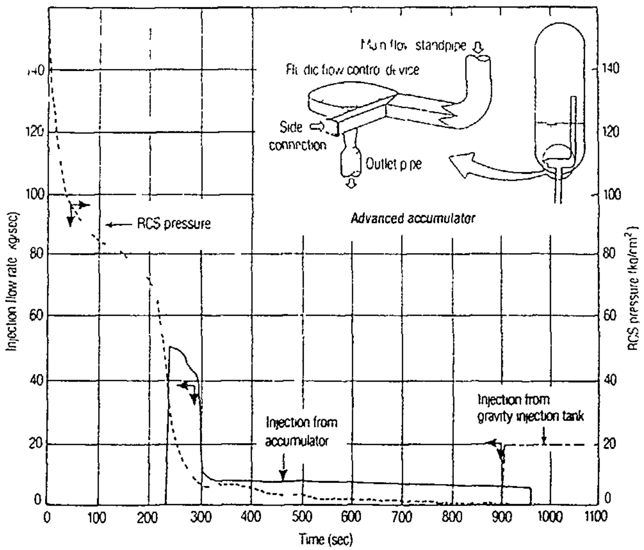

FIG. 2. Injection performance (small break LOCA).

(c)Containment vessel pressure transient In a large break LOCA, the peak occurs as a result of the blowdown and quenching of the core, just as for a conventional PWR. When the break is covered by water, steam release sops and pressure then falls as heat is absorbed into the building structure.

4.1.3 Containment vessel

The MS-600 uses a double containment consisting of a steel primary containment and a concrete filled steel secondary containment (Fig. 3). The primary containment is a 51 m steel sphere which gives a large operating floor space. The spent fuel pit and its cooling system are located inside containment together with the gravity injection tanks which are also used as refuelling water storage tanks.

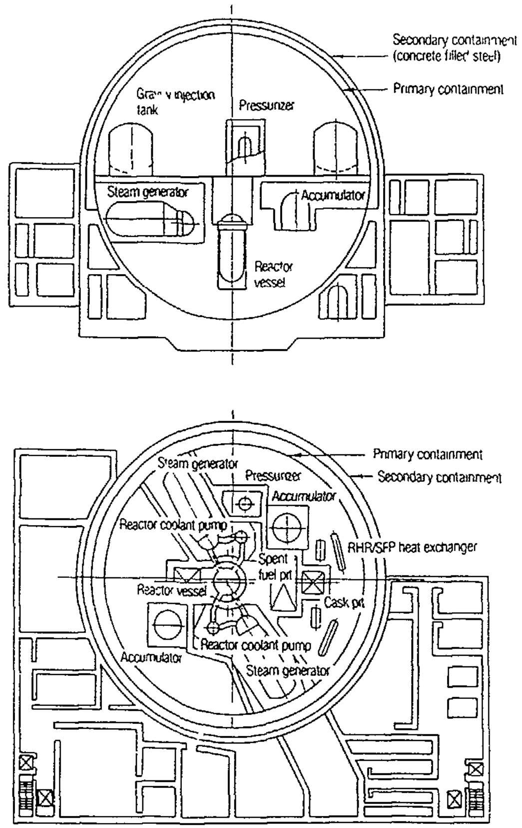

FIG. 3 MS-600. general arrangement

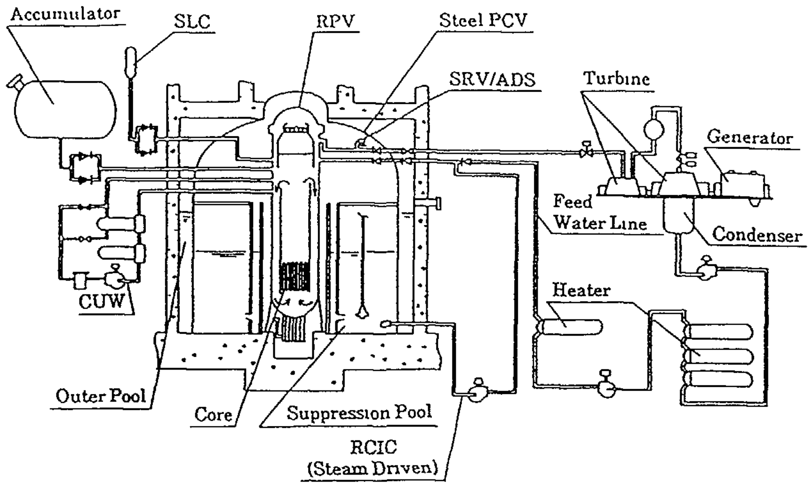

FIG 4 HSBWR system configuration

TABLE 2 HSBWR MAIN SPECIFICATIONS

Item	Value	Item	Value
Electric power	600 MW (540 MW)	Core height	3 1 m (3 7 m)
Thermal power	1800 MW (1593 MW)	Core diameter	4 7 m (3.3 m)
Operation pressure	7 0 MPa (7.0 MPa)	RPV height	25.0 m (20.0 m)
Core flow rate	17.8×10^6 kg/h (22.9×10^6 kg/h)	RPV diameter	6 3 m (4.7 m)
Volumetric power density	34.2 kW/ℓ (50.3 kW/ℓ)		

(   ) value in the current BWR

4 2 HSBWR, *simplified BWR*

The design target of HSBWR was to improve economy. As a basic principle, the maximum utilization of current BWR technologies has been considered Other design bases were to satisfy the current safety criteria and to improve safety margins, in consideration of simple and highly reliable safety systems, early termination of accident conditions, and an adequate grace period to mitigate severe accidents *(safer staying)*

4.2.1 Concept and features

The rated capacity of the HSBWR is 600 MW electricity. The main system configuration is shown is Fig. 4. And the main specifications are listed in Table 2. The features of the HSBWR are

(1) short fuel rod bundles of 3.7 m with 3 1 m active heated length to avoid seismic resonance between the core and the reactor building constructed on soft to firm ground, which make it possible to construct the standardized plant underground *with a high ability against terrorism*, if necessary,

(2) low volumetric power density of 34 2 KW/l, and a long continuous operation period of 23 months,

(3) natural circulation in the core and no steam separators, resulting in simple internals,

(4) no core uncovery in any loss of coolant accidents (LOCAs) during the given grace period without any recovery actions,

(5) decay heat *absorption in the suppression pool for one* day after accidents and natural heat removal from the primary containment vessel (PCV) by heat conduction through the steel fabricated PCV to the outer pool for three days,

(6) depressurization by the automatic depressurization system (ADS) and borated water injection by the accumulator to decrease reactivity and to shut down the reactor in an anticipated transient without scram (ATWS),

(7) standardized compact PCV, reactor building and turbine building and the same *plant layout at any reactor site*, and

(8) a short construction period of 32～36 months depending on the site conditions, including pre-operation and start-up tests.

4.2.2 Safety systems

The HSBWR has high safety configurations *for postulate LOCAs, because there* are no large diameter pipes below the top of the core The safety systems have redundancy in active and passive systems. The safety systems were designed to satisfy the current safety criteria, simplify the systems, terminate accident conditions early (quick quench), and provide an adequate grace period for safety action mitigating severe accidents.

(1) Core cooling

To simplify the engineered safety systems and keep them at a high standard, the steam driven reactor core isolation cooling (RCIC) system and low pressure accumulators are provided for short term core cooling (Fig. 4) instead of emergency diesel generators and pumped injection system in the current BWR designs. Elimination of these systems and equipments simplifies the emergency core cooling system (ECCS) and provides higher system reliability as there are fewer movable elements. The capacity of the accumulators is enough to supply emergency coolant for at least 24 hours after reactor scrams, which is the most severe case of an accumulator line break. The grace period is typically longer at other pipe breaks.

Core cooling after normal reactor shutdown and long-term core cooling after reactor scram are performed by the residual heat removal system (RHR) with injection pumps and heat exchangers (Fig. 4). The RHR has ability to cool down the reactor to 52℃ within 20 hours, and it is also enough for heat removal during accidents. The accumulators and the suppression pool can remove and sink decay heat for one day after reactor scram by themselves, giving the RHR one day to spare for start-up. Even if the RHR is not available because of pump failure, coolant can be fed into the RPV by manually refilling the accumulators with attachable pumps such as fire engines.

(2) Decay heat removal from PCV

Heat removal from the PCV can be achieved by natural circulation in the suppression pool and heat conduction through the steel PCV wall to the outer pool between the PCV and reactor building walls (Fig. 4). This heat removal from the PCV to the outer pool needs no movable components.

The decay heat is carried out to the drywell as steam through the break, and the steam is fed into the suppression pool through vent tubes. The fed steam condenses in the water near the vent tube outlet, and natural circulation is induced, resulting in the pool temperature rise. Heat is transferred to the outer pool through the steel PCV wall and long-term decay heat removal is achieved.

The temperature in the outer pool reaches saturation at 30 hours after accident initiation, and evaporation begins. At about 50 hours, the transferred heat to the outer pool exceeds the decay heat and the temperature in the suppression pool begins to decrease. The maximum temperature is 122℃, allowable considering the PCV toughness. After evaporation begins in the outer pool, its water level begins to decrease, resulting in decrease of effective heat transfer area in the outer pool. Further temperature increase in the suppression pool is, however, avoided by a water supply of about 12 m^3 / h into the outer pool, from 72 hours after accident initiation.

V. SUMMARY

(1) Nuclear power is essential to the solution of energy demand and pollution problems
(2) "Human friendliness" is the key word of the next generation LWRs
(3) Simplification / passive technologies are being studied as one of the way (but not only the way) to the "human friendly" plants.

The following studies and developments are being carried out.
(1) Applicability of simplification / passive technologies to large sized LWRs
(2) SPWR (960 MWe version of AP-600)
(3) Experiments of isolation condensers and containment cooling by water pool
(4) MSPWR Mitsubishi's simplified PWR with hybrid safety systems
 ·flexible operation and early termination by active systems
 ·long term cooling and 3 days grace period by passive systems
(5) HSBWR, Hitachi simplified BWR with natural circulation core cooling
 ·containment cooling by water pool and accumulators for ECCS.
 ·short construction period 32-36 months, simplified structures and buildings

REFERENCES

1. M.Aritomi et al., "Study on applicability of advanced safety systems and con cepts to large size light water reactors" to be presented at ICONE-I, The fist JSME-ASME Joint International Conference on Nuclear Engineering, Nov. 4-7, 1991, Tokyo.
2. K.Matsui and K.Kuriyama, presented at this meeting.
3. T.Matsuoka et al., "A simplified Japanese PWR", Nucl. International, p.47-51, June 1991.
4. Y.Kataoka et al., Nucl, Technol, 82 p.147-156 (1988) and 91, p.16-27 (1990).

SWISS RESEARCH AND DEVELOPMENT ACTIVITIES IN THE DOMAIN OF ALWRs

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Abstract

Following the 10 year moratorium on new nuclear installations in Switzerland, reactor development activities have been drastically reduced. However, in order to keep the nuclear option open, to satisfy the needs of current nuclear facilities and to build-up the knowledge necessary for an eventual deployment of advanced reactors in Switzerland, basic research focused mainly on reactor safety and waste disposal is further carried out with undiminished efforts.

In a study carried out jointly by the Paul Scherrer Institute (PSI) and the Swiss Federal Institute of Technology in Lausanne (EPFL) under the sponsorship of the Swiss Federal Office for Energy, general criteria for futur nuclear plants to be built in Switzerland have been established. The time horizon considered addresses the second decade of the next century. The results indicate that primary weight is given to the safety of future nuclear installations; this comprises more stringent safety regulations as well as the wish to completely avoid the necessity for evacuation measures.

A large experimental programme for the investigation of passive decay heat removal and fission product retention in Advanced LWRs (ALWRs) has been started at PSI in close cooperation with the EPRI ALWR research programme and with the financial support of the Swiss utilities. First investigations will concern isolation condensers of GE's SBWR and mixing phenomena in ABB's PIUS.

1. Actual situation of nuclear energy in Switzerland

Five NPPs ranging from 320 to 990 Mwe are actually connected to the grid in Switzerland (in chronological order of their start of commercial operation): Beznau-I and Beznau-II (KKB), Mühleberg (KKM), Gösgen-Däniken (KKG) and Leibstadt (KKL). The following table summarizes the main characteristics of these plants:

Table 1: Characteristics of Swiss nuclear power plants

	KKB	KKM	KKG	KKL
Reactor type	PWR	BWR	PWR	BWR
Commercial operation	1969/71	1972	1979	1984
Net nominal capacity (MWe)	2 × 350	320	920	990
Number of turbines	2 × 2	2	1	1
System pressure (bar)	155.0	71.7	157.0	70.0
Cooling system	river	river	tower	tower
Fuel rods/assembly	14 × 14-17	8 × 8-1	15 × 15-20	8 × 8-1
Number of fuel assemblies	2 × 121	240	177	616
Gross load factor 1990 (%)	83.9/86.2	87.1	89.0	87.6

The total installed capacity of 2.93 GWe covered in 1990 41% of the domestic electricity production. The rest of electricity demand in Switzerland has been covered by domestic hydroelectricity (57%) and oil-fired plants (2%). During winter a significant part of electricity is imported while peak power is exported during summer. The demand for electric energy is continuously increasing. The growth rate of 2.4% in 1990, although below the average of the last decade (3%), is still important; it is not expected to decrease unless drastic legal measures are taken to encourage electricity economies.

The opposition against nuclear energy in Switzerland resulted into a series of federal votes[1] on its future use: two "popular initiatives" aiming at a restriction of nuclear energy in Switzerland have been rejected by 51.2% and 55.0% in February 1979 and September 1984 respectively. More recently, two new "initiatives" have been submitted to votes in September 1990. The first of them postulating a phase-out of nuclear energy has been rejected by 52.9% of the votes. The second one postulating a so-called "Moratorium" of 10 years has been accepted by 54.6% of the votes. According to this law "... for a period of ten years (...) no frame-, construction-, commissioning- or operation licences according to the Federal Law on new installations for generation of atomic energy (atomic power plants or atomic heating reactors) will be granted."

At the same votes a new "Energy Article" has been adopted by a clear majority of 71.0%. It postulates that central and local authorities in Switzerland should care for a sufficient, diversified, safe, competitive and environmentally acceptable energy supply and for a efficient and rational energy use. Short after passing of this article, the Federal Office of Energy presented under the name "Energy 2000" a list of measures aiming at a stabilization of the energy consumption in Switzerland around the turn of the century by means of information and public awareness campaigns along with coercive legal measures (e.g. energy or CO_2 taxes). On the other hand this list of measures postulates a power increase of 10% in the existing NPPs and a continuation of the R&D efforts on advanced reactor concepts in order to maintain know-how and to provide a capability for concept assessment if a new plant would have to be built in Switzerland after the end of the moratorium period.

[1] Federal votes on "popular initiatives" for new legislations or on referenda against laws proposed by the government are the most commonly used instrument of "direct democracy" in Switzerland; they are guaranteed by the constitution, take place 3-4 times yearly on different subjects and thus belong to the usual Swiss political environment.

It is against this background that the future of nuclear energy in Switzerland has to be considered. It becomes clear, that given the time constraints, if new NPPs would be built in this country, they would base on a future generation of reactor concepts. As the nuclear capacity in Switzerland is exclusively based on LWR technology, it can reasonably be expected that these future reactor concepts will most probably belong to the ALWR family.

It thus becomes clear, why, in this period of reduced R&D activities, the main efforts are focussed on two objectives

- a timely definition of the required characteristics of future NPPs;
- the familiarization with the new technologies and especially their novel features, in particular passive safety systems and inherent safety mechanisms. This last goal can be most efficiently achieved by specific contributions to selected international research activities and by keeping alive international contacts and the information flow.

2. Swiss user's requirements for next generation reactors

In spring 1990 the Laboratory for Energy systems of the Swiss Federal Institute of Technology in Lausanne (EPFL) and the Laboratory for Reactor Physics and Systems Engineering of the Paul Scherrer Institute (PSI) in Würenlingen, Switzerland initiated a study ("Réacteurs 2000") in order to provide the Swiss decision makers with a broader and more concrete view on the conditions to be fulfilled by nuclear installations, which could eventually be built in Switzerland in the first decades of the next century. The study was sponsored by the Swiss Federal Office of Energy and was completed by end of February 1991 [1].

The raw material was collected by means of two questionnaires sent to 72 Swiss specialists knowledgeable of nuclear energy. These included representatives from the Swiss utilities, from economic, industrial, scientific and political circles as well as from safety authorities and some concerned public associations. They represented therefore a fairly broad sample of the opinion of people who in some way have a direct influence on the definition of the Swiss policy in matters of nuclear energy development. It is however fair to say that a majority of the consulted specialists are more or less convinced supporters of nuclear energy.

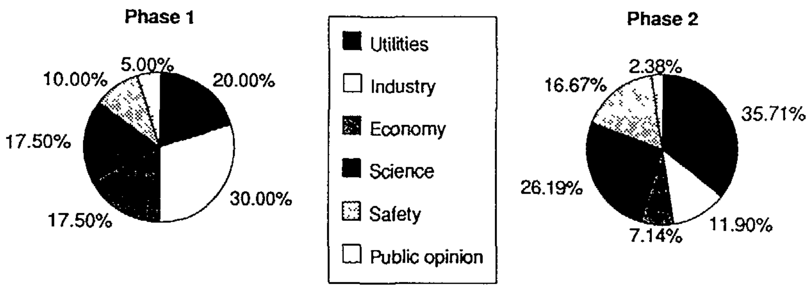

Figure 1: Distribution of consulted specialists among professional groups

The first questionnaire included 24 questions with multiple choice answers and space for more detailed comments; it was structured around following topics: general design requirements, safety, performance, and economic criteria. Whenever this was relevant, a discrimination was made between the answers regarding reactors for power generation or co-generation, and those specifically designed for district-heating applications. In the following however only the results concerning NPPs will be discussed. This questionnaire was answered by 49 persons

After the results of the first inquiry round were evaluated and items of consensus as well as unclear points have been identified, a second questionnaire was constructed and sent to the same persons. Its purpose was to confirm strong statements and to clarify weak points, and also to assess the effects of the outcome of the federal votes of September 1990; it was accompanied by a survey of the results obtained in the first phase and has been answered by 42 persons. The distribution of the answers received in the two rounds is shown on Fig. 1.

2.1 General design requirements

- Relative importance of factors to be considered and of design criteria

  In the eyes of the consulted specialists safety is clearly the most important factor to consider in the choice of the type of the future nuclear installations to be built in Switzerland; it is followed by economy. The same tendency is also reflected in the criteria cited at first place as being important for future reactor concepts: Safety margins and level of passive safety systems have been placed at the first places, just after availability (Fig. 2).

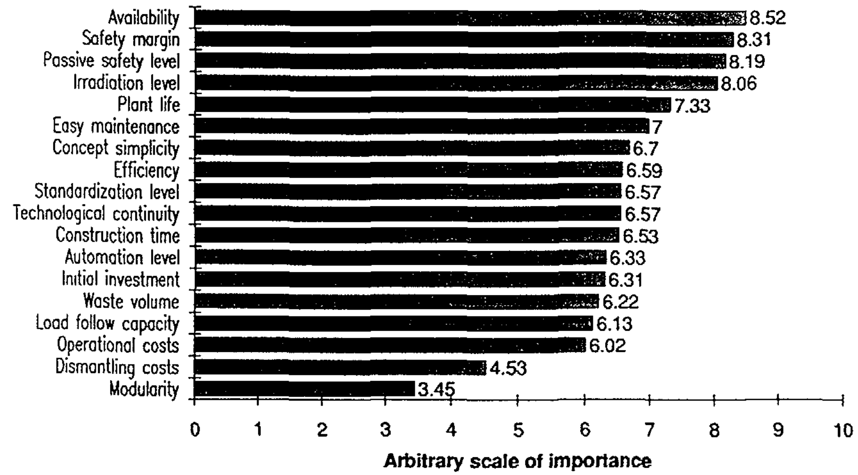

Figure 2: Classification of design criteria on a scale of 10

- Design philosophy

  With regard to the philosophy which should be adopted for the design of new NPPs, the people consulted show a slight preference for the "evolutionary line" over the "classical line"; the "revolutionary" philosophy is clearly rejected. In any case, Switzerland cannot afford to play a pioneering role in this matter, and a successful cumulative operating experience of five to six reactor-years abroad is generally considered as a prerequisite to the possible selection of a new type of reactor in the country. This tendency is confirmed by the results of the second phase of the study: 85.7% of the specialists agree fully or partially to follow an "evolutionary" line for future NPP development.

- Nuclear application

  The experience gained with the 50-MW REFUNA district heating network fed by the Beznau nuclear power plant has demonstrated that the combined production of electricity and heat can be quite attractive in Switzerland both from an economic and an energetic point of view. It is therefore not surprising that cogeneration takes the first place among potential applications. Power plants for dedicated electricity generation come in second position whereas dedicated heating reactors follow third. Due to the structure of the Swiss industry, the supply of nuclear process heat is not considered as important as the other applications.

- Plant size

  The specialists consulted in the first phase of the study were practically equally divided between supporters of reactors of the present 1000 MWe line (including most of the representatives of the utilities) and supporters of smaller reactor units. This stalemate situation is further illustrated by the fact that some 25% of the interviewees declare to have no clear-cut opinion on this question. The evaluation of the results of the second phase has shown a slight advantage for SMRs (40.5% vs. 38.1% of the votes) but this difference is still not significant.

  The main argument in favour of larger units is the limited number of sites available for the construction of nuclear installations in a small country with as high a population density as Switzerland. Moreover, the present opposition to the nuclear energy by a large fraction of the population makes it unlikely that in the near future a plant could be constructed on a new site; the few existing sites should therefore be used at their full capacity. Further arguments for large plants include the good operating experience of the existing Swiss plants and the well developed electrical network in Europe. The supporters of smaller reactors put the potential advantages of these concepts with respect to safety, ease of operation and shorter construction time in the foreground.

- Plant underground siting

  The question has been raised to know if a totally or partially underground construction could have a positive impact on the acceptance of new nuclear installations by the Swiss public opinion: The opinion of the consulted specialists on this question is evenly divided (44.9% yes vs. 46.9% no).

2.2 Safety

- General safety requirements

  Many of the consulted specialists consider that the safety requirements imposed in Switzerland upon the nuclear installations are already among the most severe in the world. Nevertheless, the question *"Do you think that there are objective reasons for more stringent safety measures (as compared to those valid for Swiss nuclear plants actually in operation) to be imposed upon future nuclear installations, which could possibly be build in Switzerland?"* was answered with *"yes"* by some 60% of the interviewees in the first phase of the study; this opinion has been confirmed by 70% of the specialists in the second phase. Among the most often mentioned new requirements are the installation of a filtered containment venting system in all Swiss nuclear power plants and a more systematic recourse to inherent safety mechanisms in possible future installations. 59.5 % of the interviewees in the second phase of the study have expressed the opinion, that more stringent safety requirements could be satisfied also with new reactor concepts emerging from an "evolutionary" design philosophy.

  It is worth mentioning that an important fraction of the consulted specialists do not believe that more stringent safety measures could significantly alleviate the fears of the people opposed to the nuclear energy. Nevertheless, three quarters of the persons consulted in the first phase have the opinion that, for any kind of application, only such future reactors should be built that "no population evacuation would ever be needed". Moreover, a negative answer to this question does often not mean that this requirement is considered as not crucial but rather that an absolute requirement like "no evacuation ever" cannot be backed up scientifically. The large consensus on this question is due to the fact that with the population density of Switzerland and its limited territory, a permanent relocation of civil populations is indeed absolutely inconceivable, because it would lead to unbearable economical and social consequences.

  In the second phase of the study this statement has been confirmed by 73.8% of the votes (additional 21.4% agree partially). For a majority of the specialists the requirement of "no evacuation" has to be interpreted as a deterministic demonstration, that no significant radioactive releases outside the plant fence can occur for any accident scenario; 32.6% of the interviewees are still in favour of a probabilistic approach. A slight majority of the specialists considers this requirement as being only a development goal (47.6%) while 45.2% would prefer to establish it as an absolute condition (no future reactor concept should be taken into account, which does not satisfy this requirement).

- Required safety level

  Three different safety levels were proposed:

  Level 1: safety is assured without having to rely on any active system, even in case of the failure of one of the key components of the installation; the installation is "self-protected" against the consequences of any severe structural failure or serious mistake of an operator.

  Level 2: passive safety systems should be able to maintain the integrity of all critical parts of the installation for a sufficient lapse of time in case of the failure of one of its key components; the installation is partially "self-protected" against the consequences of a severe structural failure or serious mistake of an operator.

  Level 3: (present safety level) safety is primarily assured by the action of various active systems in case of the failure of one of the key components of the installation or serious mistake of an operator; safety is based on the "defense-in-depth" philosophy; the installation is not "self-protected" against the consequences of a severe structural failure.

  61.9% of the interviewees think that new nuclear power plants constructed in Switzerland should at least reach the safety level 2; only a small minority (12%) would be satisfied with the present safety level. This position is absolutely coherent with the "evolutionary" development philosophy and the enhanced safety requirement already taken up above for this kind of installations.

- Core damage frequency

  According to the sample survey, the upper limit for the core damage frequency should be contained in a bracket of 10^{-5}-10^{-6} per reactor-year (IAEA/INSAG recommendation: 10^{-5} per reactor-year).

2.3 Performance

- Plant design life

  On the average, the persons consulted evaluate about 45 years as the desirable plant design life for new nuclear installations, but lifetimes as high as 60 years or as low as 30 years have also been proposed by a few interviewees. The proposed plant design life seems a wise compromise between longer lifetimes which would prevent the construction of up-to-date units, and shorter lifetimes which would increase the dismantling frequency and create site availability problems.

- Load follow capability

  Continuous operation between 50% and 100% of the nominal power is the average maneuvering margin required by the consulted specialists for the future Swiss nuclear power plants. This is rather unexpected, because until now nuclear power plants in Switzerland have been exclusively used to satisfy the base load demand; it is however conceivable that nuclear power could take in the future a higher share in the national electricity production than the present 40% value and thus justify such a maneuvering flexibility.

  The above maneuvering margins should nevertheless be considered with some reserve, because it seems that significant fraction of the interviewees have not a very definite opinion on this question; this is demonstrated in the sample survey by the great dispersion of the proposed values, and the 40% of "no opinion" answers.

- Occupational radiation exposure

  On a scale stretching between zero manrem per year and 400 manrem per year (present limit for the collective radiation exposure of the personnel in the Swiss nuclear power plants), the questioned specialists set on average the limit of the collective radiation exposure for the personnel in the new nuclear power plants which could be constructed in Switzerland at 300 manrem per year.

2.4 Economics

The results of the first phase of the study, indicated that total generation costs for future Swiss NPPs could be some 10% higher than the corresponding costs from fossil power plants of comparable size. 42.8% of the interviewees in the second phase agree with slightly higher costs (10-20% higher than the concurrence) and 31.0% would even postulate an economic penalty of more than 20%.

The reason generally mentioned to justify this economic penalty is the environmental benefit resulting from the use of a type of power plant not contributing to the air pollution and global warming. It is also often stated that this is an acceptable price to pay for the saving of irreplaceable alternative natural resources and for a better energy autonomy of the country, considering that the nuclear energy can practically be regarded as a domestic energy resource (natural energy resources being very scarce in Switzerland).

3. The ALWR experimental programme at PSI

Over the past 5 to 10 years work has been initiated by various reactor vendors around the world on the development of a new generation of reactors with increased use of passive safety features. Typically in the 600 MWe power range, these smaller units could offer enhanced safety, improved economics and eventually provide increased public acceptance of nuclear power (with beneficial consequences regarding utilisation of energy resources and environmental damage). Two such reactor concepts are the PIUS reactor of ABB-Atom and the SBWR, proposed and being developed within the framework of the DOE/EPRI ALWR-Programme by the General Electric Company (GE).

Though safety cases for decay heat removal in existing light water reactor (LWR) systems rely on forced convection cooling and emergency stand-by power units, licencing trends for future LWR's indicate that much greater reliance will be placed on passive safety systems, such as natural circulation, and "walk-away" safety concepts. Consequently, large international programmes are underway, in several key industrial areas, which aim to convert basic understanding of natural circulation phenomena into reliable, engineering technology, for use in advanced reactors. PSI and the Nuclear Energy Subcommittee of the Swiss Utilities (Unterausschuss Kernenergie der Überlandwerke, UAK) have, in collaboration with EPRI and GE, jointly sponsored the research project ALPHA which aims to demonstrate the viability of passive cooling mechanisms for ALWR.

Advanced LWRs are likely to dominate the nuclear power industry in the intermediate-term future. In general, the Swiss utilities, the machinery industry (ABB, Sulzer, etc.), and the regulatory authorities (HSK) will acquire by association, interaction, and familiarization, in the framework of the ALPHA project know-how and specific knowledge about the passive ALWRs. Engineers and specialists will be trained and exposed to the new passive reactor concepts so that relevant experience will be available when, and if, such systems are also implemented in Switzerland. Moreover, Swiss industries could become involved in building the experimental facilities and, possibly, in backfitting existing NPPs worldwide with the newly developed passive containment cooling systems.

The programme is structured around three experimental facilities [2]:

3.1 The PANDA facility

Following discussions with EPRI the experiment chosen for investigation is related to the long term decay heat removal and cooling of the SBWR reactor containment following a loss-of-coolant accident, that may (or may-not) develop into a severe accident following the loss of all emergency core cooling systems. Following a loss-of-coolant accident the long term decay heat is removed from the SBWR by condensing the steam produced in the reactor core in a series of condensers interconnected to the reactor/containment. However the performance of these condensers will be degradated by the flow of air from the containment, and if the accident develops into a severe accident then the flow of hydrogen

The purpose of the experments is to simulate on a full height and 1/20 to 1/24 volumetric scale, the decay heat removal capability of the condensers in the presence of the non condesibles air and hydrogen (simulated by He). The non-condensible gases arise from:

1. The mixing of the steam from the reactor vessel with the air in the containment and
2. the addition of hydrogen following the oxidation of the reactor core materials if the accident progresses into a severe accident.

It is well known that condensation heat transfer is dramatically reduced in the presence of non-condensibles because of the build-up of a non-condensible boundary layer adjacent to the condensing surface. One of the most important generic questions to investigate therefore, is the mixing of the steam and air/hydrogen in the heat sink. In particular following phenomena should be investigated:

- mixing of steam and air/hydrogen;
- transport of these mixtures within the reactor/reactor containments;
- condensation heat transfer in the presence of non-condensibles (air/hydrogen).

The experiment will provide experimental data from scaled facilities and requires therefore analytical support both for its definition (i.e. to specify the initial conditions) and for the extrapolation to full scale reactor systems. This analysis will partially be performed with codes which have been used at PSI for several years, so that there is a considerable body of expertise within PSI's Thermal Hydraulics Laboratory on the methods and application of these codes.

In the longer term following the completion of the collaboration with EPRI the facility is expected to provide for PSI a centre of expertise for a large range of related activities. For example a direct follow-up to the EPRI program would be to examine the design limits of the SBWR containment cooling system. This could, for example involve an investigation of

- what is the maximum concentration of non-condensibles, such that the heat removed is less than the reactor decay heat;
- also what concentration of fission products degrades the isolation condenser, either by tube plugging or heat resistance due to plate-out, to the point where the heat removed is less than the reactor decay heat.

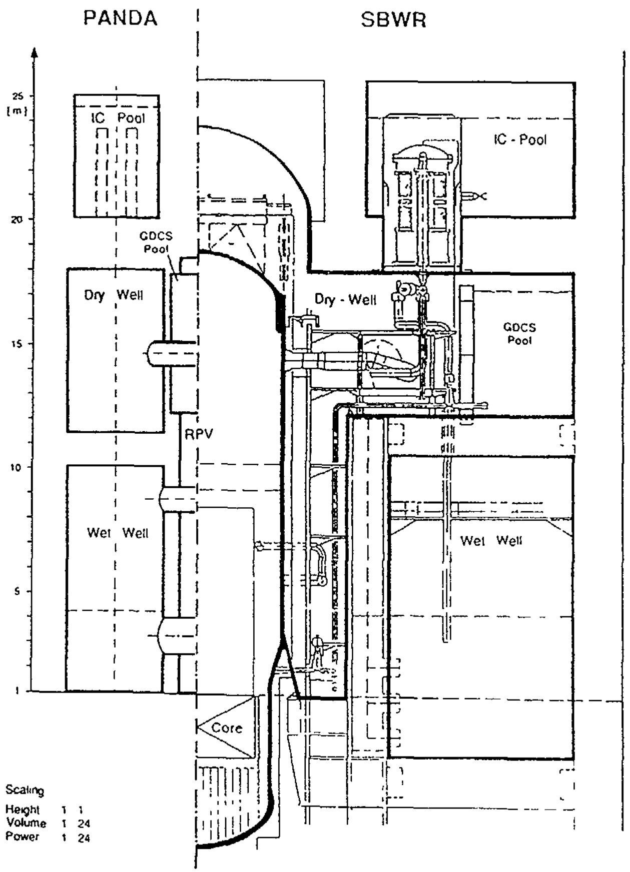

Figure 3: Arrangement of the PANDA facility and comparison to the SBWR plant

The test programme involves the building of (Fig. 3)

- 2 full height scaled low-pressure condenser units and one high-pressure unit;
- 1/20 to 1/24 volumetric scale simulation of the containment, dry-well and wet-well compartments with various steam/air/water mixtures;
- a steam supply (approx. 1.5 MW) and storage vessel.

The above components and associated piping are designed to operate at pressures up to 8.6 bar. Instrumentation will be available to measure pressure, wall and fluid temperatures, non-condensible to steam mass ratio and flows of steam/air from the Wet-well/Dry-well regions through the reactor vessel to the isolation condenser.

The parameters to be investigated include

- Collection of non-condensibles as a function of time within the condenser units; for various distributions within the containment compartments and for different pipe breaks (ie. connections to the reactor vessel)
- The purging of the non-condensibles from the condenser for various removal schemes.
- The different behaviour of air (N_2) and hydrogen (simulated by helium).
- The long term pressure build-up decay within the reactor vessel/containment.
- For severe accidents in addition to the transport of hydrogen, aerosol transport would be simulated.

3.2 The LINX facility

Complex 3-dimensional (3-D) mixing of single and two-phase flows occurs in a number of situations of interest to reactor technology. Examples of immediate applications of such technological developments and objects of experimental investigations, are the passive decay heat removal concepts that are considered for future ALWRs. Complex mixing of single- and two-phase flows will occur in pools surrounding the primary system, in the pressure suppression pools, and/or in containment volumes where gases, steam and water can mix.

There are very few systematic investigations of such phenomena in large-scale geometries and the numerical methods (and codes) available for the calculation of such mixing processes need considerable further development. Thus research in these areas is indicated.

Two concrete applications where thermal mixing in pools is important are (a) for the PIUS reactor concept: the transient flow of hot water from the primary loop into the main pool through the upper density lock during a forced shut down, and (b) for the SBWR concept: the flow of steam and air into the suppression pool or "Wet-Well" following a loss of coolant incident. In the first application, turbulent and convective mixing of a rising plume of hot water in a cold water pool having a complex geometry is investigated. In the second application, the the convective mixing of a flow of steam and air under the surface of a cold water pool is studied.

Both experiments are of great importance for the verification of the safe operation and/or handling of accidental situations for these reactors. Indeed, the passive safety features of these systems could be invalidated if problems related to the mixing phenomena above were discovered. The research described here will allow investigation of the basic phenomena at an adequate scale, and most importantly building up of the analytical capability to calculate such effects.

The investigation/work program applicable to the family of problems cited above will incorporate four tasks:

(i) Experimental investigations of the relevant mixing phenomena in scaled facilities. The basic apparatus, i.e. tank, instrumentation, data collection system, etc., will be common to both aforementioned applications. The scales will be typically 1/5 to 1/10 (linear) for the PIUS pool and 1/200 to 1/400 (volumetric) for the SBWR Wet-Well (see Fig. 4).

(ii) Detailed modeling of the mixing phenomena. A 3-D fluid dynamics code will be chosen, further developed with the addition or modification of mixing models as necessary, and validated using our own or any other available experimental data. The experimental observations will be compared to code calculations in detail to reveal and further correct model and code deficiencies. Analytical work will be required relating to turbulence modelling for the single-phase, PIUS problem, and to provide interfacial relations for the SBWR two-phase, mixing problem.

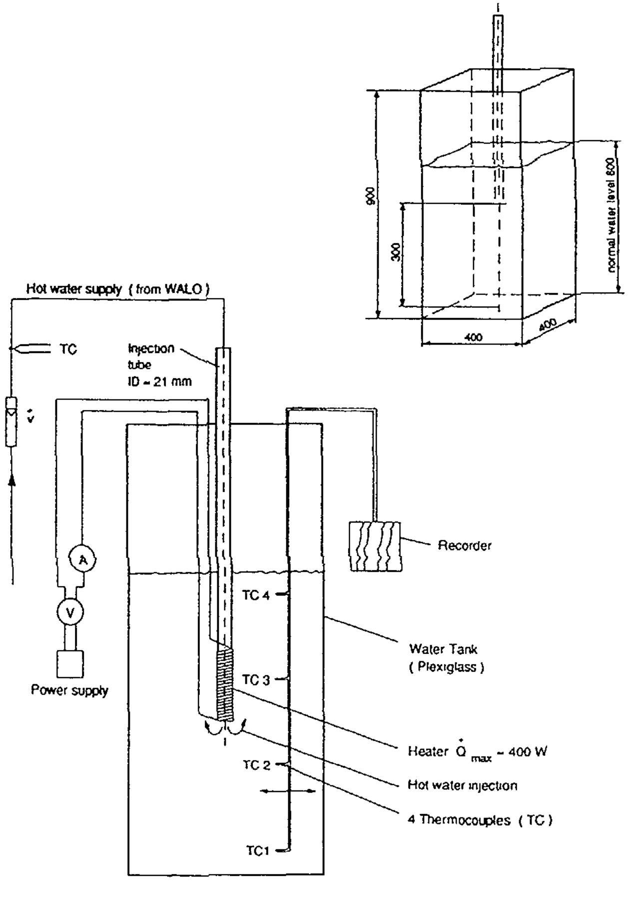

Figure 4: Mixing visualization experiment

(iii) Numerical investigation of actual mixing problems. The numerical tools developed will be used to investigate the turbulent and convective mixing in the given geometries, for both the scaled experiments and the full reactor (PIUS, SBWR) cases. The same code can be used for both tasks.

(iv) In the last phase of the work, the information obtained from the experiments and the detailed numerical investigations will be used to develop simple models and/or correlations. For example, for the PIUS problem, ABB has developed, for design purposes, a semi-analytical, lumped-parameter mixing model which requires as input an empirical mixing parameter, E. In the context of the current proposal, correlations for E would be derived based upon the scaled experiments and detailed numerical analyses, at both small and full scale, for the specific geometry and scale of the PIUS reactor cold water pool.For the SBWR investigation, a simple mixing model or energy distribution correlation is required for use in a reactor system code such as TRAC, which will then be used to investigate the overall SBWR containment system.

3.3 The AIDA facility

Under severe accident conditions fission products in form of aerosols can escape from the reactor pressure vessel and reach various rooms of the reactor building. It is therefore possible, that aerosols reach the decay heat removal condensers and influence their performance. The goals of this part of the programme are

- the demonstration of the fission product removal characteristics in the passive decay heat removal system and
- the development of a model/correlation for fission product transport and removal in the isolation condensers.

The activities proposed include

- source term analysis to provide the required boundary conditions and
- experimental investigation of the fission product retention and transport in the condensers.

The experimental part of the investigations includes

- aerosol deposition and retention in the upper drum of the condensers, also in order to define the input boundary conditions for the single condenser tubes;
- aerosol deposition and retention in a single full scale tube;
- visualization of the process in a full height, scaled radius gals model under realistic operation conditions.

References

[1] K. Foskolos et al. "Swiss User's Requirements for Next Generation Reactors", 3rd Int. Post-SMiRT Seminar on Small and Medium-sized Nuclear Reactors, New Delhi, Aug. 1991

[2] P. Coddington et al. "ALPHA - The Long Term Passive Decay Heat Removal Program at the Paul Scherrer Institute, Switzerland", to be presented at the ANS Winter Meeting, Nov. 1991

ABWR — THE FIRST OF THE NEW GENERATION OF REACTORS FOR THE 1990s

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Abstract

The advanced BWR (ABWR) is an advanced light water reactor designed by an international team of engineers and designers to address the utility and public needs for the next generation of power plants It incorporates major innovations and includes the best from BWR designs in Europe, Japan and the USA

The major emphasis in the design of the plant has been on improved operability and maintainability This has led to an overall plant design that is simpler to build, maintain, and operate and at the same time has significantly enhanced safety features - including several features that are inherent to BWR designs and ensure 'passive" responses to transients and accidents The ABWR design has several additional features that ensure that offsite doses would be extremely low following a severe accident

This paper also discusses the other key features of the ABWR - internal recirculation pumps, fine-motion control rod drives, digital control and instrumentation, multiplexed fiber optic cabling network, pressure suppression containment, structural integration of the containment and reactor building and advanced turbine/generator with 52" last stage buckets

The 1356 MWe ABWR design is being applied as a two unit project by the Tokyo Electric Power Co , Inc at its Kashiwazaki-Kariwa site in Japan On May 15, 1991, Japan's Ministry of International Trade and Industry (MITI) formally announced the granting of the "Establishment Permit to Tokyo Electric Power Company for constructing two ABWRs at the Kashiwazaki site This licensing milestone culminates the successful safety review in Japan and clears the way for construction of the two ABWR units Construction will begin in 1991 and commercial operation is scheduled for 1996, with the second unit one year later.

INTRODUCTION

The ABWR design is based on design, construction and operating experience in Japan, USA, and Europe, and was jointly developed by the BWR suppliers, General Electric Company, Hitachi, Ltd , and Toshiba Corporation, as the next generation BWR for Japan. The Tokyo Electric Power Co (TEPCO), has provided leadership and guidance in the establishment of the ABWR and has combined with five other Japanese utilities (Chubu Electric Power Co , Chugoku Electric Power Co , Hokuriku Electric Power Co , Tohoku Electric Power Co , and Japan Atomic Power Co) in participating and providng support for the test and development programs

The ABWR development program was initiated in 1978, with subsequent design and test and development programs started in 1981 Most of the development and verification tests of the new features have been completed Conceptual designs followed by detailed design engineering of the ABWR has progressed to the point where the Tokyo Electric Power Company, Inc announced the selection of GE Nuclear Energy, Hitachi Ltd , and Toshiba Corporation to design and construct the two lead ABWRs as Units 6 and 7 at the Kashiwazaki-Kariwa Nuclear Power Station The three companies will form an international joint venture to design the plant and supply equipment

BWR EVOLUTION

The evolution of the BWR has occurred in two major areas -- the reactor system and containment design This evolution resulted from design enhancements and experience gained from operating reactors, abnormal occurrences and testing programs The ABWR is the result of this progressive design simplification of the BWR and its containment structure

Throughout the BWR evolution, there has been an overridding trend toward simplification and optimization Systematic review of both the technical merits and cost have played a key role

TABLE 1 ABWR MAJOR PLANT SPECIFICATION

Item	ABWR	Current Japan BWR
Plant Output		
Thermal Output	3926 MW	3293 MW
Electric Output	1356 MW	1100 MW
Plant Cycle	(Direct Cycle)	(Direct Cycle)
Vessel Dome Pressure	73 1 kg/cm^2	71 77 kg/cm^2
Main Steam Flow	7640 tons/hr	6410 tons/hr
Feedwater Temperature	215 5°C	215 5°C
Turbine	TC6F-52 in	TC6F-41 in
Reheat	Two-stage Reheat	No Reheat
Nuclear Boiler		
Reactor Vessel		
Inner Diameter	7 1 m	6 4 m
Height	21 m	22 1 m
Power Density	50 6 kw/liter	50 0 kw/liter
Fuel Assemblies	872 (8 x 8 barrier)	764 (8 x 8)
Control Rods	205-B_4C	185-B_4C
Control Rod Drive		
Normal Operation	Fine Motion Electric Motor	Hydraulic locking Piston
Scram Insertion	Hydraulic piston with electric motor backup run-in	Hydraulic
Recirculation System	10 Internal pumps	2 external pumps with 20 internal jet pumps
Core Flow (100% rated)	52200 tons/hr	48300 tons/hr
Emergency Core Cooling System*		
	3 Divisions	3 Divisions
System Configuration	Div 1 RCIC+LPFL Div 2 HPCF+LPFL Div 3 HPCF+LPFL	Div 1 LPCS+LPCI Div 2 LPCI+LPCI Div 2 HPCS
Spray Sparger Type	N/A	Peripheral Ring
Residual Heat Removal	3 Divisions	2 Divisions
Primary Containment		
Type	Pressure Suppression	Pressure Suppression
Configuration	Cylindrical reinforced concrete vessel integrated with Reactor Building	Conical free-standing steel vessel
Controls and Instrumentation		
Type	Micro-processor based digital controls	Relay-analog circuit
Data Communication	Intelligent multiplexed fiber optics	Hard-wired cable

*HPCF High Pressure Core Flooder
RCIC Reactor Core Isolation Cooling
LPCI Low Pressure Coolant Injection
LPCS Low Pressure Core Spray
LPFL Low Pressure Flooder

in achieving designs that meet all objectives This through evaluation by the designers and subsequent review by the utility sponsor, TEPCO, has helped keep the effort focused and achieve excellent results

PLANT DESIGN OBJECTIVES

The major ABWR plant objectives were defined very early in the ABWR development in close cooperation with TPECO These overall plant objectives were selected to mainly improve the performance and safety and reduce costs.

The major plant objectives which guided the selection of new technologies and features of the design are:

(1) Enhance plant operability, maneuverability and daily load following capability.
(2) Increase plant safety and operating margins.
(3) Improve plant availability and capacity factor.
(4) Reduce occupational radiation exposure.
(5) Reduce plant capital and operating costs
(6) Reduce plant construction schedule

ABWR FEATURES

The ABWR design represents the integration of eight years of conceptual development and design along with the extensive confirmatory test program. Table 1 summarizes the major plant specifications and compares them to the current BWR in Japan.

Increased Plant Output and Turbine Design

The ABWR plant is designed for a rated thermal output of 3926 MWth which provides for an electrical output in excess of 1350 MWe In order to improve plant efficiency, performance and economy, the turbine design incorporates a 52-inch last stage bucket design. Combined moisture separator/reheaters remove moisture and reheats the steam in two stages. Also to help increase plant output and reduce cost, the design has incorporated the concept of both high pressure and low pressure pumped-up drains. Rather than cascading the heater drains back to the condenser, the pumped-up drain system takes advantage of this waste heat and injects it back into the feedwater ahead of the heaters. This concept has increased the generator output nearly 5 MWe, has helped to reduce the capacity of the condensate polishers, and has also reduced the size requirements for both the high and low pressure heater areas The overall design has made optimal use of the design improvements to maximize plant output and reduce cost.

Improved Core and Fuel Design

The ABWR core and fuel design was aimed at improving the operating efficiency, operability, and fuel economy of the plant. This was accomplished primarily by utilizing PCI-resistant (barrier) fuel, axially zoned enrichment of the fuel, control cell core design, and increased core flow capability. The use of minimum shuffle fuel loading schemes reduced refueling times, while fuel burnup increased to higher values allows for longer continuous operating cycles and has improved fuel cycle costs.

The axially zoned fuel, with higher enrichment and less gadolinia (Gd-absorber) content in the upper half of the fuel rods, allows the axial power distribution to be kept uniform throughout the operating cycle This feature assures a higher thermal margin that together with the other core design features results in improved fuel integrity, plant capacity factor, and operational flexibility The axially zoned fuel eliminates shallow control rods which control the axial power shape and the control rods are only used to control reactivity.

The core design employs the control cell core concept successfully applied to many of the latest operating plants. In this design all of the control blades are fully withdrawn throughout an operating cycle except for those in the control cells. Each control cell consists of four depleted fuel bundles surrounding a control blade. Only these control cell rods move to compensate reactivity. This minimizes the operator's tasks of manipulating control rods during the cycle to compensate for reactivity or power distribution shaping. This design also improves capacity factor since the control cell eliminates the need for rod sequence exchanges. The flat hot excess reactivity minimizes rod adjustments during the cycle.

The capability for excess core flow above rated of greater than 11% provides for several benefits. Daily load following from 100% to 70% to 100% power (in a 14-1-8-1 hour cycle) is easy using core flow rate adjustment and no control blade movement. For both maximum use of the excess flow and slight control blade adjustment, load following of 100%-50%-100% is easily attainable. In addition, the excess flow capacity allows for spectral shift operation to provide additional burnup with all rods out to increase operational flexibility, extend operation, and reduce fuel cycle costs

Reactor Vessel Incorporating Internal Pumps

The most dramatic change in the ABWR from previous BWR designs is the elimination of the external loops and the incorporation of internal pumps for reactor coolant recirculation The reactor pressure vessel (RPV), and core internals have been optimized for the internal pump concept. As shown in Figure 1, all large pipe nozzles to the vessel below the top of the active fuel are eliminated. This alone improves the safety performance during a postulated Loss of Coolant Accident (LOCA) and allows for decreased ECCS capacity.

The RPV is 7.1 m in diameter and 21 m in height. The reactor vessel height and total volume have been minimized, which has resulted in reduced volume requirements for the containment and reactor building (R/B). In-service inspection (ISI) has been reduced due to the incorporation of internal pumps and elimination of the recirculation pipe nozzles and reduced amount of vessel welding during vessel fabrication. The RPV has a single forged ring for the internal pump mounting nozzles and the conical support skirt. Forged rings are also utilized for the core and upper regions of the vessel shell sections The elimination of the external recirculation piping and the use of vessel forged rings has resulted in over a 50% reduction in the weld requirement for the primary system pressure boundary.

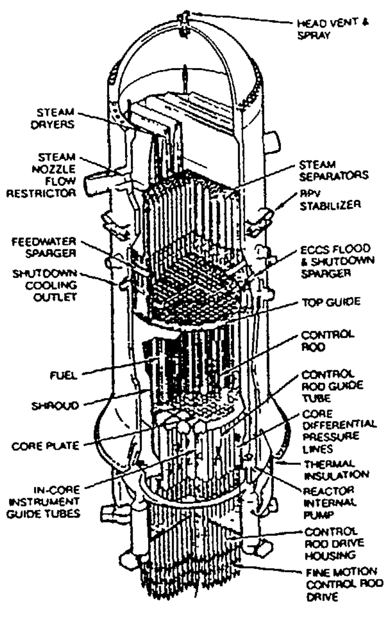


Figure 1. RPV and Internals

The reactor vessel has been designed to permit maximum ISI of welds with automatic equipment This will help minimize manpower and reduce radiation exposure. Other features of the ABWR RPV design include main steam outlet nozzles containing a reduced diameter throat and diffuser which is used to measure steam flow This also acts to restrict flow and reduce loads on the reactor internals and reduces the containment loads during postulated LOCA. The steam dryers and separators are of the improved lower pressure drop design developed for BWR/6 This lower pressure drop contributes to increased stability margins and lower pump power costs.

The ABWR incorporates ten internal recirculation pumps (see Figure 2) located at the bottom inside of the RPV. This simplifies the nuclear boiler system and allows for compact space requirements in the RCCV and R/B Elimination of the external recirculation loops has had many advantages Key advantages have been the reduction in containment radiation level by over 50% compared to current plants, lower pumping power requirements, and the excess flow provided by the pump design has enhanced plant operation and allows for full power operation with one pump out of service

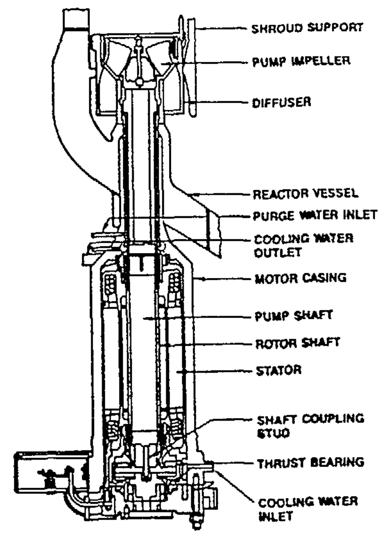


Figure 2. Reactor Internal Pump (RIP)

The internal pump is a wet motor design with no shaft seals. This provides increased reliability and reduced maintenance requirements and hence, reduced occupational radiation exposure. These internal pumps have a smaller rotating intertia, and coupled with the solid-state variable frequency power supply, can respond quickly to grid load transients and operator demands. These pumps are now accumulating plant operating experience in several European BWRs. Improved designs have also been tested in Japan in testing programs now underway.

Fine Motion Control Rod Drives (FMCRD)

The ABWR incorporates the electric-hydraulic FMCRD, which provides electric fine rod motion during normal operation and hydraulic pressure for scram insertion. Reduced maintenance with reduced radiation exposure is a feature of the new drive Integral shootout steel built into the FMCRD replaces the external beam supports of the current BWRs and improves maintainability and reduces radiation exposure

The drive mechanism operates to allow fine motion (18 mm step size) provided by the ball screw nut and shaft driven by the electric motor during normal operation. The electric motor also provides increased reliability through diverse rod motion to the hydraulic scram and acts as a backup with motor run-in following scram. The fine motion capability allows for small power changes and easier rod movement for burnup reactivity compensation at rated power. It also reduces the stress on the fuel and enhances fuel rod integrity. Ganged rod motion (simultaneous driving of a group of up to 26 rods) and automated control significantly improves startup time and power maneuvering capabilities for load following.

The ABWR FMCRD design has been improved from the European design by reducing the length and diameter, and by adding the fast scram function. Other refinements in the ball-screw assembly and seal designs have led to less maintenance requirements. Other design features include: (1) a two-section design for the CRD housing with the wearing parts concentrated in the shaft seal housing section in order to provide ease of maintenance and reduce both maintenance time and radiation exposure. (2) The FMCRD scram water is discharged directly into the reactor vessel eliminating the scram discharge volume and associated valves and piping. This reduces radiation exposure and eliminates a potential source of common mode failure for the scram function. (3) The drive maintains a continuous purge into the reactor thus eliminating the accumulation of radioactive crud in the drive and reduces exposure. (4) Continuous full-in indication. (5) Dual safety grade separation switches to detect rod uncoupling and a new bayonet coupling to help eliminate the control rod drop accident. (6) Ganged hydraulic control units (HCU) with two CRDs per HCU. (7) A brake mechanism to prevent rapid wind down of the screw prevents rod ejection. Figure 3 illustrates the key components of the FMCRD.

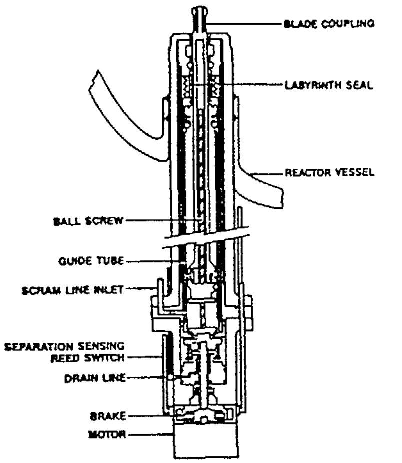


Figure 3 Fine Motion Control Rod Drive (FMCRD)

Safety and Auxiliary Systems

The ECCS and Residual Heat Removal System (RHR) along with the other auxiliary systems were reviewed to simplify and optimize their design. The design incorporates three redundant and independent divisions of ECCS and containment heat removal. The RPV with its deletion of the external loops and no large pipe nozzles in the core region allows for a reduced capacity ECCS and yet the fuel remains covered for the full spectrum of postulated LOCAs including a single failure. The ECCS network has each of the three divisions having one high pressure and one low pressure inventory makeup system. The high pressure configuration consists of two motor driven High Pressure Core Flooders each with its own independent sparger discharging inside the shroud over the core and the Reactor Core Isolation Cooling System (RCIC) which has been upgraded to a safety system. The RCIC has the dual function of providing high pressure ECCS flow following a postulated LOCA, and also provides reactor cooling inventory control for reactor isolation transients. The RCIC, with its steam turbine driven power, also provides a diverse makeup source during loss of all A-C power events. The lower pressure ECCS for the ABWR utilizes the three RHR pumps in the post-LOCA core cooling mode. These pumps provide Low Pressure Flooding and are labeled LPFL. The ECCS pumps provide core makeup over the full pressure range. For small LOCAs that do not depressurize the vessel when high pressure makeup is unavailable, an Automatic Depressurization System (ADS) actuates to vent steam through the safety relief valves to the suppression pool, depressurizes the vessel to allow the LPFL pumps to provide core coolant.

The RHR system has a dual role of providing cooling for normal shutdown and also provide core and containment cooling during LOCA. The ABWR RHR systems have been improved such that core and suppression pool cooling are achieved simultaneously since, in the core cooling mode, the flow from the suppression pool passes through the heat exchanger in each of the three divisions of the RHR.

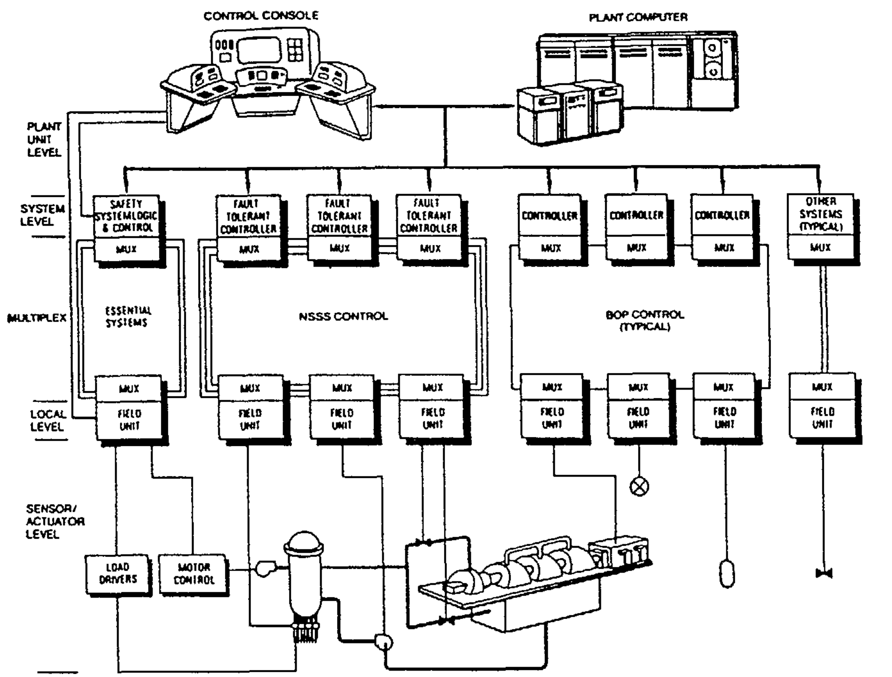
As a result of these enhancements in the ECCS network and RHR, there is a substantial increase in the safety performance margin of the ABWR over earlier BWRs. This has been confirmed by the preliminary probabilistic risk assessment (PRA) for the ABWR which shows that the ABWR is a factor of at least 10 better than BWR/5 and 6 in avoiding possible core damage from degraded events.

Rationalizations have also been made in other auxiliary systems. For example, the main steam leakage control system has been deleted. The Combustible Gas Control System utilizes an inerting system to reduce oxygen concentration in the primary containment and also uses portable recombiner units to be shared at the site for combustible gas control following postulated degraded core accidents. The Standby Gas Treatment System flowrate was reduced due to recent favorable leak rate data for containments and the number of passive filter trains was reduced from two to one.

Advanced Control and Instrumentation Systems

Digital technology and multiplexed fiber optic signal transmission technology have been combined in the ABWR design to integrate control and data acquisition of both the reactor and turbine plants. For the feedwater, recirculation, and pressure control systems, triplicated fault tolerant digital control is utilized. A redundant design for the rod control and information system is used. These systems feature automatic self-test and built-in calibration and trip test to improve reliability. Microprocessor-based digital monitoring and control (DMC) has been implemented for most principal NSSS and Balance of Plant monitoring and control functions. ABWR DMC technical advantages include self-test, automatic calibration, user interactive front panels, full multiplex system compatibility, standardization of the man-machine interface, and where possible, use of common circuit cards. Plant startup and shutdown operations, TIP operation and nuclear instrumentation gain adjustments have been automated to reduce operator error and reduce plant startup time. Technological advances in the ABWR nuclear instrumentation areas include fixed wide range in-core monitors with a period based trip design to replace the current source and intermediate range monitors and eliminate range switching during startup.

The use of multiplexed fiber optic data transmission in the ABWR is another new feature applying advanced technology. The use of fiber optic multiplexing reduces the amount of cabling and cable pulling time during construction. This also reduces the overall cost of the control and instrumentation area. Multiplexing easily provides a high degree of redundancy in the control system, and also improves maintainability. The ABWR combines multiplexing communications capability with intelligence to perform logic and control functions for interfacing systems by employing the DMC in the multiplexing system. This multiplexing system has been applied to the plant protection and engineered safeguard systems as well as nuclear steam supply and balance of plant control systems. The conceptual structure for application of this digital and optical transmission technology is shown in Figure 4.



* REPRESENTS ONE OF FOUR SAFETY DIVISIONS

Figure 4 Multiplexing Concept

Containment and Reactor Building Design

For the ABWR the cylindrical RCCV, with its pressure suppression concept, was selected as the reference design of the primary containment vessel The concrete walls of the RCCV are integrated with the reactor building and form a major structural part of this building The annular top slab of the drywell is also integrated with the upper pool girders that run across the building and have direct connection with the building's outer walls The pressure retaining concrete wall of the RCCV is lined with leak-tight steel plate. The cylindrical design, a simple shaped concept same as the Mark III drywell design, allows for easier and faster construction

The upper drywell encloses the reactor, and the process lines and valves of the reactor coolant system The lower drywell, located under the reactor vessel, is the space for installation and maintenance of the internal pump motors and their heat exchangers, and the control rod drives Piping and cables are arranged inside and lead out of this space Personnel and equipment access are provided for by hatches in the upper drywell, and through the tunnels in the lower drywell.

The wetwell provides an air space and a pool to suppress the steam from the postulated LOCA. Multiple horizontal vents derived from the Mark III containment design, discharge the vessel blowdown steam water mixture and the air from the drywell to the wetwell pool. The steam is condensed, and the fission products are scrubbed and retained in the pool.

The ABWR design represents a very significant reactor building volume and cost reduction The reactor building volume has been reduced to approximately 167,000 cubic meters which has both reduced construction cost and provided a schedule saving of 2-1/2 months.

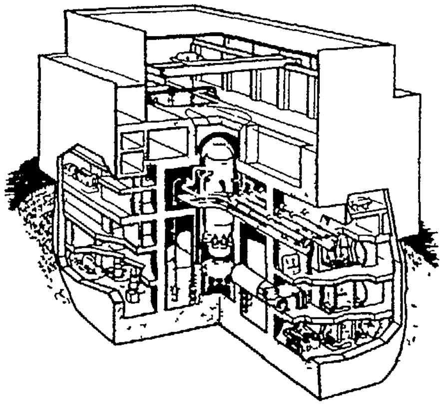


Figure 5. RCCV and Reactor Building

The structural integration scheme discussed earlier takes advantage of both the RCCV and reactor building to carry dynamic and shear loads, and hence, reduce overall size and thickness of the supporting walls The reactor building has been separated into three quadrants to provide separation for the three division configuration of the safety systems. The reactor pedestal has also been revised to support the drywell diaphragm floor, connect the access tunnels to the undervessel area, contain the horizontal vent system, and provide connecting vents between the lower and upper drywell. The design also allows for fabrication in the shop applying modular construction techniques Figure 5 represents the RCCV and reactor building design concept.

TECHNICAL EVALUATION

The technical evaluation of the ABWR design demonstrates that the integrated plant characteristics and performance were improved over the existing BWRs. This has resulted from the careful consideration of the new features, application of advanced technologies, and incorporation of proven technologies. Several of the key evaluation results are summarized below

Safety

The internal pump concept, which eliminated the internal loops and large nozzles in the core region, along with the improved ECCS network has improved the safety performance relative to a postulated LOCA. For a spectrum of pipe breaks including the Design Basis Accident, the fuel remains covered and there is little core heatup. The three division redundant containment heat removal system (RHR) has also improved the long term heat sink reliability. Other features also contribute to the overall safety improvement. These other important ABWR safety features include: the diversified electric-hydraulic control rod insertion capability of the FMCRDs to enhance scram reliability, and the improved structural characteristics of the integrated RCCV and reactor building for increased seismic and LOCA-generated load capability.

The PRA study was completed to evaluate the relative safety of the optimized design The ABWR PRA result was estimated to be at less a factor of 10 lower for core damage probability than for current BWRs.

Operability

The ABWR has improved operability over present plants through incorporation of reactor internal pumps and their excess core flow above rated flow, FMCRDs, triplicated fault-tolerant digital controls, improved core designs, automated reactor plant operation, and advanced control room with improved operator interface.

ABWR can load follow easily in the range of 100%-50%-100% power due to the excess core flow. It has fast response to positive load demands due to ganged control rod operation of the FMCRDs, and fast response to power maneuvers due to rapid flow control with the low inertia RIP and solid state variable speed control power source.

Annual Refueling and Maintenance Outage

For the ABWR plant even though the number of fuel assemblies increased approximately 14%, the actual duration of fuel handling is less because shuffling is reduced by core design improvements and reduction in fuel movement for CRD inspection since there is less CRD removal due to FMCRD adoption. The FMCRDs substantially reduce the maintenance because a complete CRD need not be removed when checking and exchanging motor or spool piece or replacing the easy to repair shaft seal due to the two piece housing design In addition, the graphite parts have been eliminated, thus reducing the number of drives to be maintained. Use of automated handling and maintenance equipment for the fuel, RIP and CRD also have helped reduce the outage The current evaluation indicates that the annual inspection can be completed in 55 days, assuming 12 months of continuous operation Additional reduction to 45 days was also studied and considered feasible with additional improvements and additional shifts.

Capacity Factor

The mature ABWR is projected to have a capacity factor of 86% based on past experience at US and Japan BWRs with major equipment problems deleted and appropriate Japanese factors applied to the data. This was based on 12 months continuous operation and a 55 day refueling outage. This excellent capacity factor is made possible through the application of the ABWR features such as improved barrier fuel, improved core designs that eliminate rod pattern exchange during operation, automated plant startup, the self-test and fault tolerant characteristics of the digital solid state controls, and others. With further reduced refueling outages and longer operating cycles, even greater operating factors can be achieved

Occupational Radiation Exposure

The elimination of the large recirculation loops in the drywell has eliminated one of the largest radiation sources in the drywell and reduced the maintenance and inspection requirements as well. The reduced FMCRD removal and maintenance as well as the FMCRD clean water purge reduces exposure during maintenance Use of automated maintenance and handling equipment also reduces radiation exposure. Improved corrosion control throughout the plant and adoption of cobalt free materials all contribute to reduce radiation levels throughout the plant.

The ABWR occupational radiation exposure has been estimated to be 49 man-Rem/year This was estimated based on the latest technology to limit the radiation buildup in the plant and was also based on the latest plant exposure data in Japan.

Construction Schedule and Economics

The schedule from rock inspection to commercial operation is projected to be 48 months. This schedule accounts for site unique differences and interference from neighboring units during construction. Adoption of new construction practices and modular fabrication, use of a tower crane for early construction use, all contribute to shorter construction schedule and hence reduced overall cost Reduced volume of the Reactor and Turbine Buildings are important to reduced schedules, decreased material quantities and reduced cost.

The increased plant efficiency and higher plant output both help cost reduction. Optimization and effective use of components and systems have reduced capacities and eliminated equipment and piping. Operating costs and fuel cycle costs were reduced as discussed earlier. The ABWR plant has proven itself a very competitive product.

SUMMARY

The ABWR development objective focused on an optimized selection of advanced technologies and proven BWR technologies for an improved BWR. The description of the key ABWR features and their advantages and performance improvements demonstrate the success of the integrated plant design The overall technical evaluation shows the superiority in terms of performance characteristics and economics that the ABWR design has achieved The many features were seen to provide improvement in schedules, reduced maintenance requirements, contributed to less radiation exposure, and still provided plant operation improvements and increased safety. Significant cost reduction in both capital and operating costs were achieved by systematically optimizing and simplifying the design while retaining the existing benefits and performance improvements.

The confirmatory test and development program has been highly successful in confirming the technical feasibility and reliability of the new components and technology. This has provided considerable support for the design in terms of test data and licensing

The ABWR design in Japan is currently in the licensing review process for application to the Kashiwazaki-Kariwa Nuclear Power site. Construction for the first unit (K-6) will begin in 1991 with commercial operation in 1996. The second unit (K-7) will follow two years later.

The ABWR program in the US has been initiated to obtain regulatory certification of the ABWR as a world-class standard plant.

EVOLUTIONARY ADVANCEMENTS TO PROVEN TECHNOLOGY ARE THE KEY TO SUCCESS

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Abstract

ABB Combustion Engineering Nuclear Power (ABB-CENP) has been concentrating on commercializing its large evolutionary Advanced Light Water Reactor (ALWR) - System 80+. The System 80+ Standard Plant Design is an advanced version of the three reactors currently in operation at Palo Verde, Arizona, and the four reactors under construction at Yonggwang and Ulchin in the Republic of Korea. It incorporates the advanced features recommended by the U.S. utility industry as specified in the EPRI's ALWR Utility Requirements Document. It has been under development since 1986 and is currently going through regulatory review as part of the new NRC design certification process which aims at pre-licensing standard designs before actual construction projects are undertaken. System 80+ will be one of the first new generation reactors to receive Final Design Approval and Design Certification by the NRC under this new process.

As a natural extension of the evolutionary process, ABB-CENP will evaluate the cost-benefit of adding certain passive safety features to the System 80+ design. The principal area of concentration will be on those features that enhance containment integrity and thereby further strengthen the defense-in-depth philosophy to which the System 80+ design has held fast. If cost-beneficial, such passive features when fully designed and tested, will be considered for a future revision of the System 80+ design.

The rationale for the evolutionary approach, a description of System 80+ and the specific passive design features that are to be evaluated are presented in this paper.

INTRODUCTION

In order to meet world wide utility needs for electrical power generation in the 1990s, ABB-Combustion Engineering Nuclear Power (ABB-CENP) is developing its next generation nuclear plant offering. ABB-CENP has decided that the offering will be an enhanced, standardized version of the successful System 80 design. The new design, called the System 80+ Standard Plant Design, addresses new licensing requirements and meets utility needs for increased public safety, investment protection, lower cost, and ease of operation and maintenance.

Nuclear suppliers throughout the world are developing five basic types of advanced reactors for electric power generation:

- o Evolutionary Advanced Light Water Reactor (ALWR);
- o Passive ALWR;
- o Advanced Heavy Water Reactor (HWR);
- o High Temperature Gas-cooled Reactor (HTGR); and,
- o Liquid Metal Reactor (LMR).

At ABB-CENP, it is believed that the evolutionary ALWR (in this case, the System 80+ Standard Plant Design) will be the dominant technology of choice throughout the world, for at least the next few decades. It is the only LWR technology that is truly "proven" and will provide the most economical electricity. It will offer a level of safety that rivals, or exceeds, that of other advanced designs, including the passive ALWR. With Final Design Approval expected in 1993 by the U.S. Nuclear Regulatory Commission (NRC), the design will be marketed well into the 21st century. Naturally, the design may be modified at selected internals to incorporate new improvements, as technological advancements are developed.

CHARACTERISTICS OF AN EVOLUTIONARY DESIGN

The System 80+ Standard Plant Design has been developed to be a state-of-the-art pressurized water reactor (PWR) available to meet both domestic and international needs. The design has been driven by three specific sets of requirements and policies: (1) the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD), which specifies features desired by utilities in future plant designs; (2) the U. S. Nuclear Regulatory Commission (NRC) Severe Accident Policy, which identifies new standards to be applied to future nuclear plant designs; and (3) the NRC Standardization Policy, which provides the framework for licensing of new standardized designs.

While the latter two policies listed above provide general guidance for all types of designs, the EPRI ALWR URD provides separate detailed technical requirements for two specific classifications of Advanced Light Water Reactors (ALWR), the large evolutionary ALWR and the small passive ALWR. System 80+ has been designed to meet the requirements established for the former. As an evolutionary ALWR, System 80+ meets the EPRI goals of simplicity, improved reliability, improved accident prevention and mitigation, improved economics, and better man-machine interfaces, while maintaining two fundamental attributes. The attributes that characterize the evolutionary ALWR are:

1. A well proven and successfully operating reference design with incremental technological changes that are also well proven, thereby assuring the immediate availability of the design without the need to perform prototype testing or the need to re-define licensing criteria.

2. The economy of scale afforded by today's large plants is preserved, thereby assuring a competitive low power generation cost on a per kWe basis.

ABB-CENP feels that these attributes of the evolutionary ALWR will best meet the current and immediate future needs of utilities worldwide.

Why did ABB-CENP choose this approach rather than the "clean sheet of paper" approach afforded by other advanced designs? The nuclear "recession" in the United States resulted from institutional factors, rather than any inherent technological deficiencies. For this reason, a "call to arms" for a completely new reactor technology is unwarranted -- unless that technology creatively and effectively addresses the real institutional factors which resulted in the decline of nuclear power in the U.S. ABB's position, therefore, is to press for a demonstration of the standard plant licensing process in the near term, with a growing emphasis on technological advances for the long term. A more effective, higher quality regulatory infrastructure, and advocacy by government are needed if there is to be a market in the U.S. for new orders in the coming decade. More advanced technologies may fit into new market niches after the turn of the century. Building on the experience of more than 111 operating light water reactor plants in the U.S. and 4,200 reactor years of successful LWR operation worldwide is a rational and unemotional basis for a resurgence of nuclear orders.

In order to crystallize ABB-CENP's approach, direct contact has been made with potential utility customers by the formation of the System 80+ Executive Advisory Committee composed of senior utility executives from some of the most prominent nuclear utilities. This committee has consistently recommended evolutionary improvement of the most successful of the current plant designs. Additionally, it has called for these new designs to be significantly pre-engineered as well as pre-licensed by the U.S. Nuclear Regulatory Commission. The bases for this consensus is the fundamental tenet that none of these utilities will order a new nuclear power plant unless there is extremely high confidence in the cost. That confidence in cost can only be achieved with similarly high confidence in the construction and startup schedules. These utilities see two separate but related threats to the necessary schedular confidence: (1) Licensing Risk - that the plant will be delayed by NRC required design changes and (2) Technical Risk - that the plant will be delayed by design changes made necessary by the discovery of technical inconsistencies. System 80+ has been developed to eliminate the licensing risk via certification in accordance with the U.S. NRC's new one step licensing process

Table 1
System 80+ Evolutionary Changes

Design Area	Design Objectives	Design Changes
Reactor	Maintain Proven Design	Very Few Changes
	Meet Utility Performance Needs	Part-Strength Rods for Load Follow
Reactor Coolant System	Improve Plant Margins	Lower Operating Temperatures
		Increased System Volumes
		Improved Materials
Engineered Safety Systems	Reduce Core Melt Frequency	Redesign in Conformance with EPRI ALWR Requirements
		Added Safety Depressurization System
Auxiliary Systems	Simplify Design	Non-Safety CVCS
Containment and Nuclear Annex	Address Severe Accidents and Meet Utility Maintenence Needs	Use Dual, Spherical Steel Design
Instrumentation and Control	Provide State-of-the-Art, Human Factors Engineered Control Complex	NUPLEX 80+
Electrical and Support Systems	Improve Reliability of Engineered Safety Systems	Greater Redundancy and Diversity

(10 CFR 52) and to eliminate the technical risk by the evolutionary experience-based approach to design improvements.

SYSTEM 80+ SUMMARY DESCRIPTION

The System 80+ design has been developed by making specific changes to an established reference design. To accomplish this in an organized manner, specific design objectives were established for each type of system and structure (i.e., design area) to improve the overall performance of the plant. Table 1 summarizes the design areas, the improvement objectives and the types of changes which were made to meet these objectives.

The design enhancements of the System 80+ Standard Plant are both numerous and significant. They meet virtually all of the design requirements of the EPRI ALWR Program and address all new U.S. NRC licensing criteria for future plants. The key enhancements for the plant are summarized below in terms of advanced design features. Although they are presented as individual features, they were developed as part of an integrated design process which considers many important aspects, including safety, cost, operability, maintainability, and human factors. Many of these features cannot be separated from one another without losing a significant portion of the desired benefit or without substantial redesign.

1. Reactor

The main emphasis in the reactor area has been to maintain the proven design of System 80; therefore, changes are very few. The core operating margin has been increased by reducing the normal operating hot leg temperature and revising core parameter monitoring methods. The ability to change operating power level (i.e., maneuver) using control rods only (without adjusting boron concentration in the reactor coolant system) has been provided, simplifying reactivity control during plant load changes and reducing liquid waste processing requirements.

The reactor pressure vessel is ring-forged with material specifications that result in a sixty year end-of-life RT_{NDT}; well below the current NRC screening criterion. This results in a significant reduction in the number of welds (with resulting reduction in in-service inspection) and eliminates concern for pressurized thermal shock. These improvements are summarized in Figure 1.

2. Reactor Coolant System

The principal design objective in the Reactor Coolant System has been to improve plant margins. This has been accomplished by increasing system volumes, using improved materials, and lowering operating temperatures. The pressurizer volume is increased by thirty-three percent to enhance transient response and reduce unnecessary challenges to safety systems.

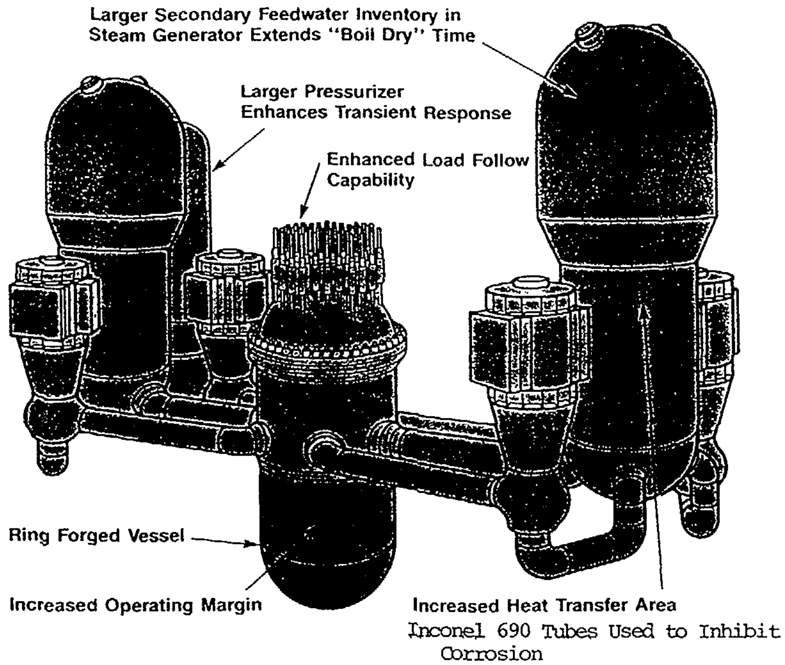

Figure 1: Principal Reactor Coolant System Improvements

The System 80+ steam generators include Inconel 690 tubes, improved steam dryers, and a seventeen percent increase in overall heat transfer area, including a ten percent margin for potential tube plugging. The steam generators have a twenty-five percent larger secondary feedwater inventory to extend the "boil dry" time and improve response to upset conditions. Steam generator improvements also have been added to facilitate maintenance and long term integrity. These include larger and repositioned manways, a standby recirculation nozzle, and a redesigned flow distribution for the economizer (See Figure 2).

3. Safety Systems

The safety systems have been reconfigured to reduce the calculated core melt probability. This has entailed the use of increased redundancy of mechanical trains, higher ratings of systems, and the addition of a new system.

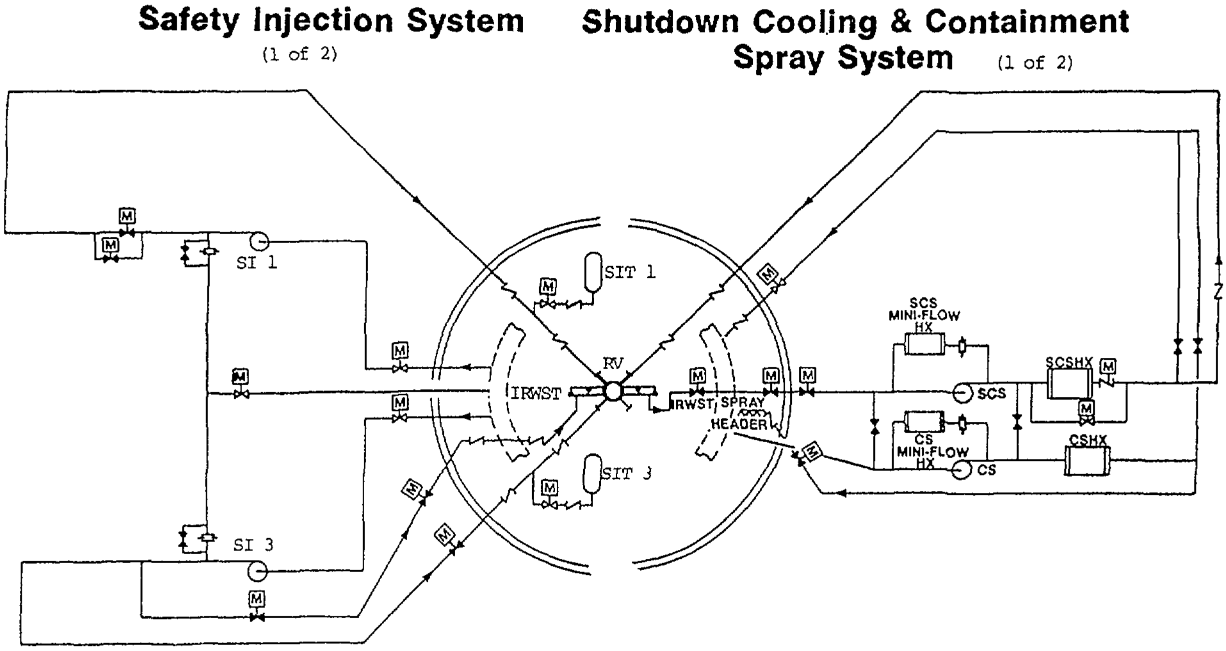

Figure 3: Integrated Engineered Safety Features System

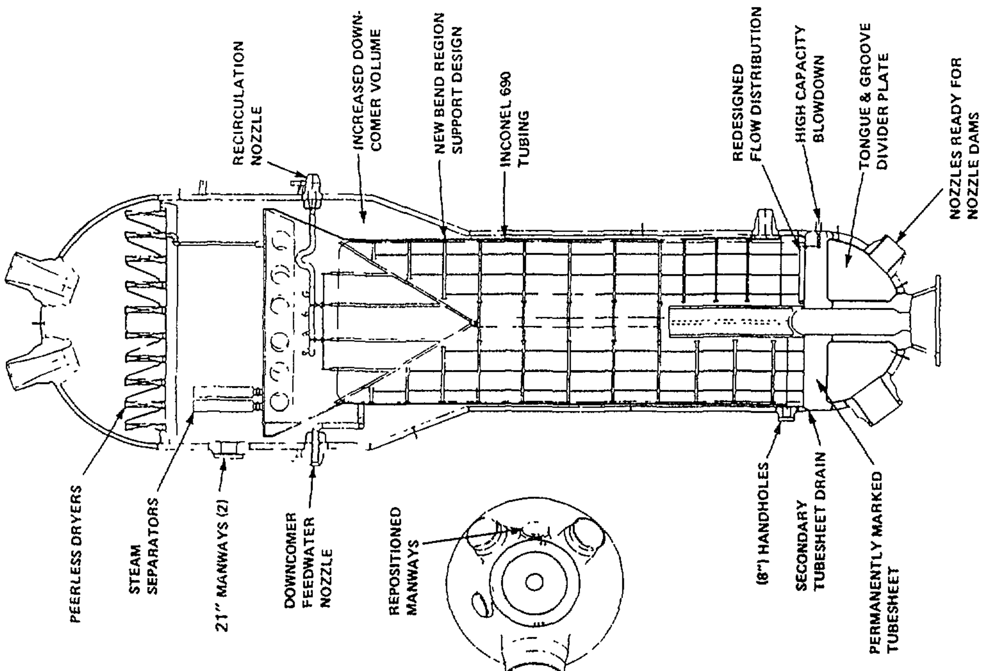

Figure 2: System 80+ steam generator enhancements.

The Safety Injection System design has been improved to provide a simpler and more reliable system with increased redundancy. It has four mechanical trains for safety injection, direct-to-vessel injection connections, and an in-containment refueling water storage tank, as shown in Figure 3. The same size pumps

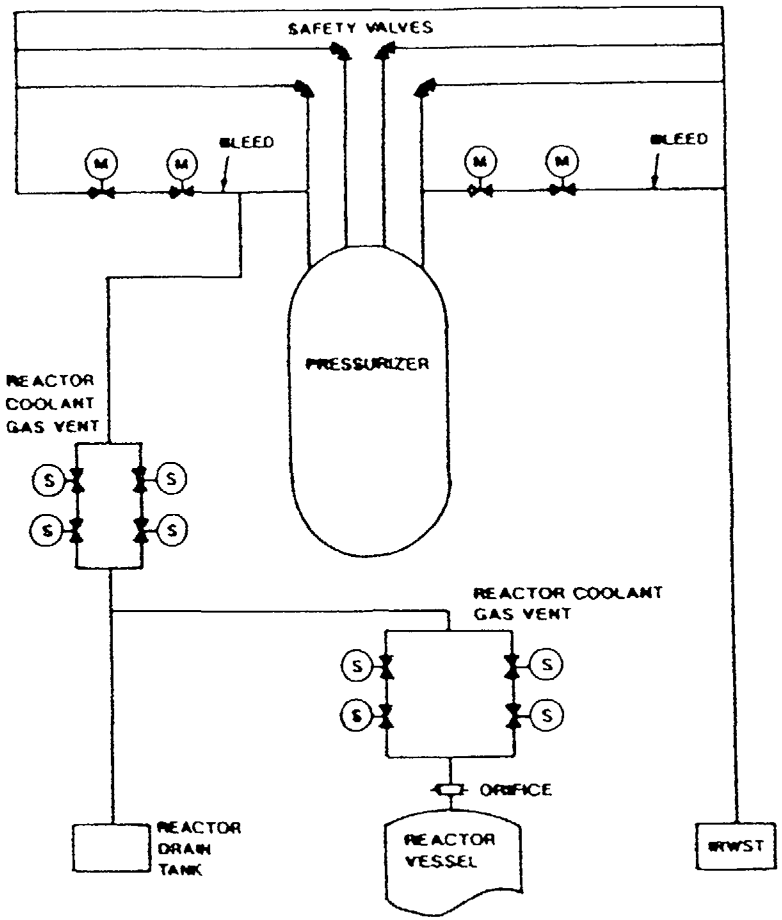

Figure 4: Safety Depressurization System

and valves used in the original System 80 two train design are now used in all four trains. The trains are not interconnected by common headers and include provision for full flow, on-line testing to eliminate the need to extrapolate bypass-flow test results to demonstrate compliance to with Technical Specifications.

The Shutdown Cooling System design pressure has been increased to 900 psig. This higher pressure provides greater operational flexibility and results in an ultimate rupture pressure that eliminates concern for system over-pressurization. The Shutdown Cooling System is interconnected with the Containment Spray System, which uses identical pumps. The reliability of both systems is therefore increased, and each set of pumps can serve as a backup for the other.

The In-Containment Refueling Water Storage Tank (IRWST) has been located in the containment building, in a torus-like configuration around the reactor vessel cavity. Containment water collection points empty into the IRWST. This means that the safety injection and containment spray pumps always take water from this tank, eliminating the need to switch from an external tank to the containment sump following a loss of coolant accident.

The Emergency Feedwater System (EFWS) is a dedicated safety system intended for emergency use only. The EFWS has two separate trains. Each consists of one emergency feedwater storage tank, one full capacity motor-driven pump, one full capacity non-condensing turbine-driven pump, and one cavitating venturi. The cavitating venturi minimizes excessive emergency feedwater flow to a steam generator with a ruptured feed or steam line. The EFWS therefore requires no provision for automatic isolation of emergency feedwater flow to a steam generator having a ruptured steam or feed line.

The Safety Depressurization System (SDS) is a new, dedicated system designed to permit depressurization of the Reactor Coolant System (RCS) when normal processes are not available. The SDS provides the capability to rapidly depressurize the RCS so that an operator can initiate primary system feed and bleed (using the safety injection pumps) to remove decay heat following a total loss of feedwater event. Motor operated valves connected to the pressurizer discharge to the IRWST as shown in Figure 4.

4. Reactor Auxiliary Systems

The main objective for the reactor auxiliary systems was to simplify their design so that cost reductions could be realized and their operation could be more straight forward. This is best typified by the changes to the Chemical and Volume Control System (CVCS). The CVCS incorporates numerous improvements which include centrifugal charging pumps, a high pressure letdown heat exchanger, and simplified charging and auxiliary spray piping. Many simplifications have been made because the CVCS has been designed as a non-safety system, i.e., no credit is taken in the safety analysis for CVCS operation during design basis accidents or safe shutdown.

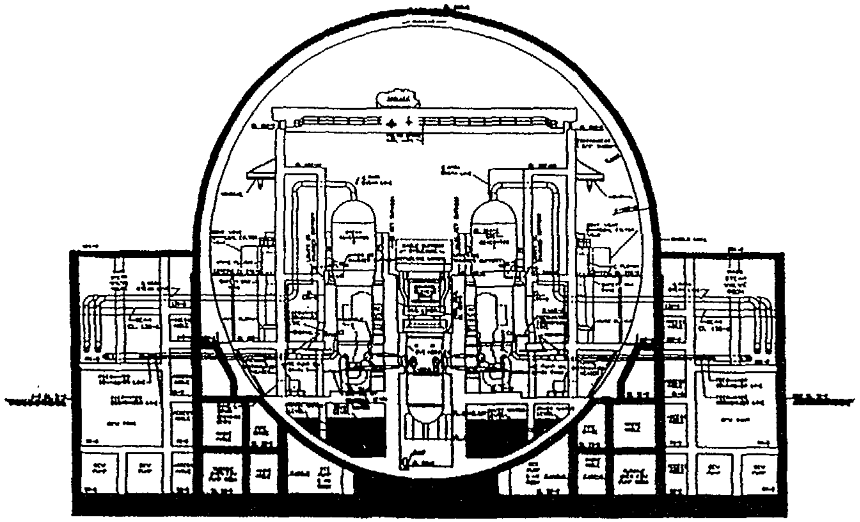

Figure 5: System 80+ Spherical Steel Containment

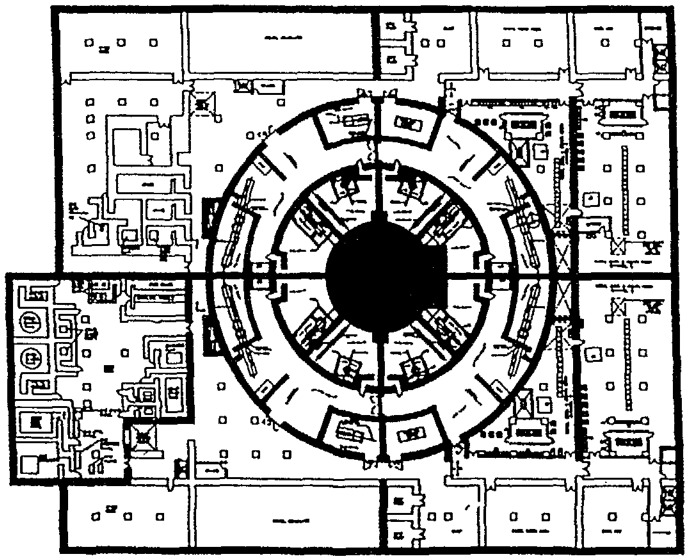

Figure 6: General Arrangement of Containment and Nuclear Annex (Base Mat Level)

5. Containment and Auxiliary Building

The containment and auxiliary building (called the Nuclear Annex) has been designed to address severe accident issues and to meet utility needs in the operations and maintenance areas. The containment for System 80+ is a 200-foot diameter steel sphere (See Figure 5) which maximizes space for equipment and maintenance while minimizing unusable volume in the upper part of the containment. The operating floor has seventy-five percent more usable area than a cylindrical containment of equal volume.

Enhancements in the System 80+ containment design include the In-Containment Refueling Water Storage Tank (IRWST) which is located low in containment to minimize interference with operation and maintenance activities and to make use of existing containment structural features to form the tank boundary. Natural boundaries have also been used in the containment ventilation system, allowing substantial reduction in the amount of ventilation ducting required. A concrete shield building surrounding the steel containment provides a dual containment function which reduces the potential radiation release during design basis events. The large free internal volume (3.4 million cubic feet) provides increased capacity for absorbing energy and diluting hydrogen concentration following a severe accident. The steel shell acts as a natural heat sink and has the potential for passive heat removal, using external cooling. Features for mitigating the consequences of postulated severe accidents include a reactor vessel cavity designed to improve the ability to resolidify molten core material on the cavity floor by cooling and retaining the molten core debris.

The spherical containment provides an area under the sphere (See Figure 6) which replaces a conventional safety-grade auxiliary building, and is an ideal location for safety systems. Placing of safeguards equipment in the sub-sphere area is an economically attractive approach to addressing numerous regulations associated with this equipment. Separation for internal flood mitigation, fire protection, security, and sabotage concerns is easily addressed without adverse effect on accessibility. A high degree of symmetry is provided in the layout of the systems, and piping and cabling complexity are reduced, resulting in lower costs.

The Nuclear Annex completely surrounds the containment and houses the remainder of the nuclear island, including the fuel pool, control room, CVCS, and electrical equipment. The Nuclear Annex and Containment are on a common base mat to enhance the response of the nuclear island to seismic events. Careful attention has been paid to providing adequate aisles and clearances so that all equipment can be maintained and/or replaced during its 60-year life. ALARA has been specifically addressed in the placement of radioactive equipment and in the design of ventilation system and maintenance access.

6. Electrical Distribution and Support Systems

The electrical distribution and support systems have been reconfigured to improve their reliability to be consistent with the greater redundancy and diversity of the safeguard systems. For example, a diverse onsite AC power supply (combustion turbine) is added as a backup to the two safety-grade emergency diesels. A turbine-generator output breaker is incorporated to allow use of the main transformer as a redundant supply of offsite power. The electrical distribution system now has three tiers: safety, non-safety, and permanent non-safety. The last tier is provided to power loads that have previously been added to the safety buses because of their importance, but were not credited in the safety analysis. Taking these loads off the safety buses allows the diesel generators to be smaller and more reliable.

7. Instrumentation and Control Systems

The main emphasis in the Instrumentation & Control (I&C) area is to provide reliable, state-of-the-art, human-factors-engineered systems. In no other area has technology change been so rapid as in this area. These changes, coupled with social, industrial and regulatory pressures to improve the safety, availability and cost competitiveness of I&C systems have collectively had a profound influence on the design of the modern power plant control complex.

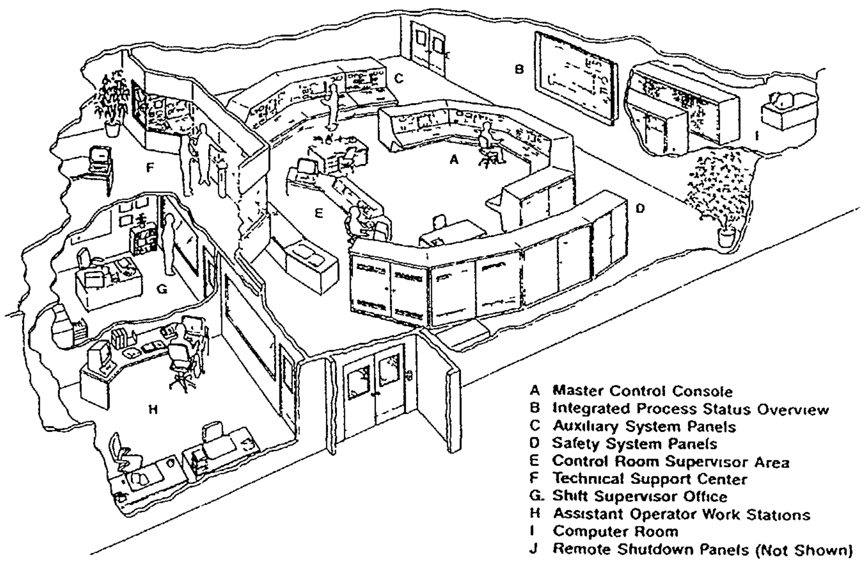

Figure 7: Nuplex 80+ Advanced Control Complex

Balancing this drive for I&C modernization and advancement has been the realization that advanced design must avoid unproven technologies or unique "one-of-a-kind" hardware, which may hinder licensing, training, operations and maintenance.

Thus, the challenge faced by today's I&C designer is to embrace the new technology advances and methodologies while adhering to the sound principles of utilization of field proven design.

The ABB-CENP Nuplex 80+ Advanced Control Complex utilizes a unique combination of currently available commercial I&C technologies making possible an orderly, natural transition to an all-digital computer-based control room. The evolutionary design of the Nuplex 80+ control room (shown in Figure 7) explicitly takes advantage of previous control room design experience and lessons learned. Despite large scale incorporation of digital technology, it maintains a high degree of conservatism in its design approach and fully addresses all U.S. NRC design requirements.

PASSIVE FEATURES IN SYSTEM 80+

The current features of System 80+ as presented to the U.S. NRC for design certification do not represent any major change in the overall design philosophy from current plants. In particular, the reliance on active safety features that forms the basis for current safety system design has been maintained. It must be noted however that the nature of the large evolutionary plant does not automatically exclude consideration of features that characteristically are more passive. In fact, System 80+ has already included some design features that are passive in nature. For example, cavitating venturies have replaced the active steam generator isolation logic in the Emergency Feedwater System. Increases in primary and secondary coolant volumes in the pressurizer and steam generator respectively are also passive improvements. As a natural extension of the evolutionary process, ABB-CENP will be evaluating the cost-benefit of adding certain additional passive safety features to the System 80+ design. The principal area of concentration will be on those features that enhance containment integrity and thereby further strengthen the defense-in-depth philosophy to which the System 80+ design has held fast. This evaluation will not stress passive features for their own sake, but rather will look for those that might enhance safety in a cost beneficial way, e.g., by displacing active components and thereby reducing overall plant cost. If cost-beneficial, such passive features, when fully designed and tested, will be considered for a future revision of the System 80+ design. The specific passive design features that are highest on the list for evaluation are discussed below:

1. External Containment Spray

The System 80+ PRA shows, as have most PWR PRAs, that long term containment heat removal is a critical function dependent on many of the same systems that are relied on for core cooling and, therefore, a dominant contributor to the large release frequency. The benefit of a diverse means of containment cooling would then potentially provide significant benefit. The current EPRI passive plant requirements call for the use of a passive containment cooling system based on the spray of water over the outside of the steel containment with the collection of run-off and the venting of steam. In the long term, this scheme gives way to natural air cooling through conduction and convection. In principle, these same methods can be applied to the large plant in either passive or quasi-passive configurations. For example, fire trucks can be used to provide the pumped cooling water to spray the outside of the steel containment shell.

2. Containment Overpressure Protection

An alternative to additional diverse containment cooling would be to provide overpressure protection for loss of cooling scenarios. While generally not necessary to meet U.S. requirements, consideration is being given to the installation of a filtered vent to provide containment overpressure protection in severe accident sequences. To mitigate the potential consequences (i.e., release of radioactivity) of such an overpressurization event, the discharge of this system can be filtered. With the use of a rupture disc for initiation it is essentially a passive system.

3. Passive Initiation of Cavity Flooding

The current cavity flooding scheme for System 80+ employs active valves for manual initiation. The potential failure of these valves to open or the operator to initiate their operation is a critical factor in the assessment of severe accident mitigation. By using a more passive mechanism for initiating cavity flooding, these active components may be eliminated.

4. Catalytic Hydrogen Control

System 80+ currently relies on battery-backed, powered ignitors for control of hydrogen under severe accident conditions. The availability of catalytic "ignitors" which work in principle as recombiners could replace the current ignitor system and eliminate the DC dependence on electrical power.

5. Emergency Boration System

Resolution of the ATWS issue in the U.S. is now predicated on an alternate electronic scram and Engineering Safety Features Actuation System. In order to alleviate concerns with single point failures in the actual mechanics of dropping the control rods, consideration is being given to an alternative boration system. Motive power would be provided by pressurized tanks or the pressure difference across the reactor coolant pumps during flow coast down.

The above features are by no means the only possible passive improvements that could be considered for a large PWR. Other features will be identified, screened and then evaluated along with the features described above.

SUMMARY

Because the great preponderance of worldwide nuclear experience is with the light water reactor (LWR), it is natural for this technology to be at the forefront of nuclear power's resurgence. The basis for this reactor technology is well established, as is the supporting infrastructure to design, license, construct and operate new LWRs. Light water reactors have had an excellent safety record and in most countries an excellent operating record. It is clear that the technology itself has been sound and thus a revolutionary departure from this technology is not desirable or necessary.

In every country where nuclear power has been successful, there is a continual striving for improved reactor designs. This striving attempts to address the public's desire for safer plants and the owner-operator's need for plants that can generate electric power more economically. The path to more advanced reactor designs will naturally evolve through the existing LWR technology. Each generation of LWR to date has been built on the experiences of its predecessors by adopting features that make it safer, more reliable, and easier to operate. The hard lessons learned on operating reactors provide the keys to success for future designs. This process must be evolutionary to be successful; otherwise, there is no historical basis for predicting future performance.

DEVELOPMENT OF PROJECTS OF ADVANCED WATER COOLED REACTOR PLANTS

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Abstract

The report deals with the main trends and the progress made in the field of the development of the pressurized water-cooled reactor designs:

- advanced reactor WWER-1000 (V-392);
- reactor plants of new generation: WWER-110 NG and WWER-500/600 MWe

Quantitative and qualitative comparison was made for the specified developed and operating pressurized water-cooled reactors concerning main feasibility characteristics of safety, reliability and ecological acceptability.

For the developed designs of reactor plants the passive principle is more widely used for safety systems in order to reduce the probability of the core melting under beyond-design accidents to the value of 10^{-6} - 10^{-7} per reactor year.

Much attention is given to optimization of design and layout decisions with regard for economical factors and to decrease of dose committment during maintenance of power unit. In the report are described the issues requiring further development to power unit. In the report are described the issues requiring further development to make the final decision, for example, overcoming the beyond-design accident with primary leaks and loss of emergency power supply for 24 hours at the expense of joint operation of residual heat removal passive system and additional hydrotanks.

Information is presented on modernization of horizontal steam generator design with regard to operational experience of serial WWER-type reactors and the recent calculation-and-experimental studies.

Introduction

The following projects of reactor plants of the PWR-type are now under development:

- high power (1000-1100 MWe) - V-392 and V-410,
- medium average power (~ 600 MWe) - V-407.

Table 1 shows the comparison of a number of design performances of the reactor plants: V-320 (commercial design, 17 units are in operation), V-392 (evolutionary updating of commercial design V-320) and V-410 - reactor plant for a new generation NPP. The corresponding characteristics are also given for the APWR (USA) and N4 (France). Special emphasis is placed on safety improvements in advancing the commercial reactor plant V-320 and in changing over to V-392 and V-410 designs. For the design of a new medium power reactor plant, the V-407 plant, in addition to safety improvements, it is possible to achieve a marked improvement in economical efficiency (see Table 2) as well as simplification of the design and safety systems (Fig. 1).

Information on the structure and the description of the safety systems for V-392 and V-407 designs have been presented previously [1].

1. Principle changes being made in V-392 design (as compared to V-320) are,

1.1 A broader spectrum of design basis and beyond-design-basis accidents is analysed and, in addition, the design shall ensure that the probability of severe core damage or core meltdown must (should) not be higher than 10^{-5}I per reactor year.

A range of accidents, although classified as beyond-design-basis accidents provides the design-bases for the following additional safety systems:

- a system for rapid boron injection;
- a system for residual heat removal - PHRS;
- a system of additional hydro-accumulators.

For these accidents the objective was to prevent severe fuel damage by including the specified systems.

1.2 The inherent safety of the reactor is also greatly improved.

Thus, CPS control (rod) worth was increased to 10.2% as compared to 6% in V-320 design. This improvement was achieved at the expense of increasing the number of control members to 121 (~ 2 times more) which will maintain the reactor in the sub-critical state during cooldown to 100-120°C.

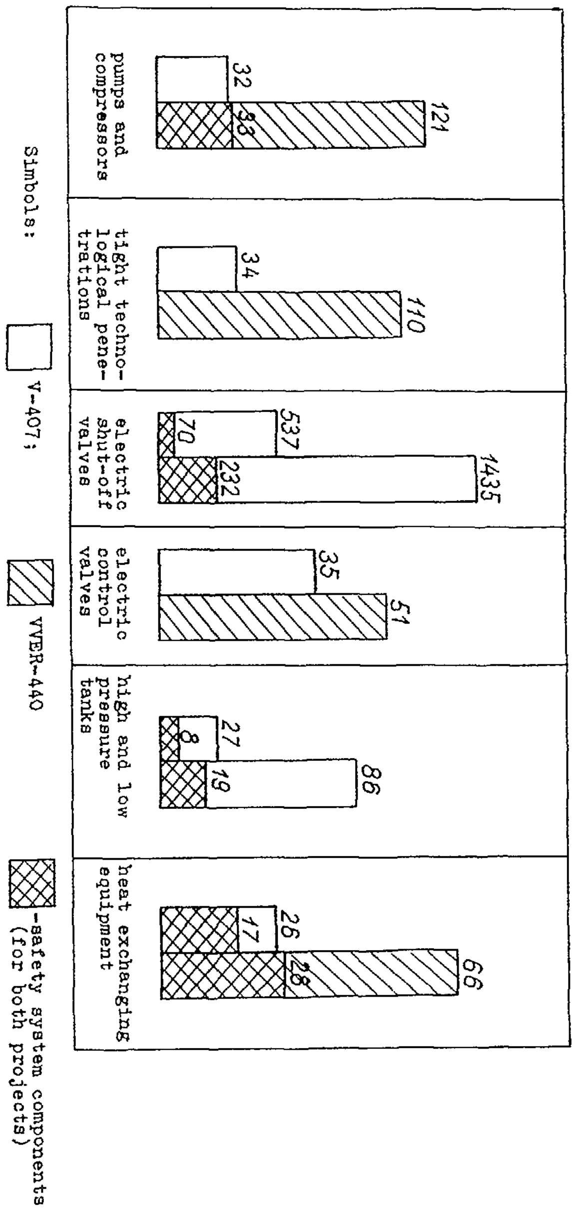
TABLE 1 COMPARISON OF DESIGN PERFORMANCES OF WATER-COOLED REACTOR PLANTS

Parameter	Reactor plant				
	V-320 (commercial USSR)	V-392 (USSR)	V-410 (APWR, USSR)	APWR (USA)	No 4 (France)
Thermal capacity, MW	3000	3000	3000-3300	3823	4270
Electrical generating capacity,MWe	1000	1000	1000-1100	1350	1400
Number of loops	4	4	4	4	4
Primary pressure,MPa	15,7	15,7	15,7	15,5	15,6
Coolant flow rate through the reactor, m^3/h	84800	84800	84800-110740	106500	94115
Coolant heating-up in the core,°C	30,3	30,3	34,0	31	37,3
Number of fuel assemblies	163	163	163-253	193	205
Number of fuel elements in fuel assembly	312	311	311	289	
Fuel element diameter,mm	9,1	9,1	9,1	9,7	9,5
Number of control rod assemblies	61	121	121	89	73
Core power density, kW/l	108,1	108,1	108,1-118	96,2	105,2
Peak heat flux, W/cm	448	448	400-448	417	448
Reactor vessel: inner diameter,mm	4135	4135	4135-5400	4623	4500
height,mm	10897	10897	11700	15280	12595
neutron fluence,n/cm^2	$5,7.10^{-19}$	$5,7.10^{-19}$	1.10^{-19}		
life duration,years	40	40	60	60	60
Steam generator type	horizont.	horizon.	vertical	vertical	vertical
Steam capacity,t/h	1470	1470	1600		2165

TABLE 2 COMPARISON OF ECONOMIC EFFICIENCY INDICES OF TWO SEISMIC NPPs (EACH HAVING TWO UNITS) OF AVERAGE POWER WITH A REACTOR PLANT NEW GENERATION V-407 AND UNIFIED VVER-440

No	Parameter	V-407	VVER-440	Ratio, %
1	Reactor maximum electric capacity,MW	635	460	138
2	Unit cost of installed capacity, rbl/kW	642	1012	62,5
3	Production cost of *) installed capacity, cop/kWh	1,033	-	-
4	Area of building erection,m^2/MW	64	94,5	67,7
5	Civil engineering volume, m^3/MW	1077	1788	60,2
6	Reinforced concrete, m^3/MW	194	395	49,1
7	Rolled products consumption, t/MW	13,3	28,1	47,3

*) Prices of 1990.



Notes: Values are given in units of equipment

FIG. 1. Simplification of the design and safety systems of a nuclear power plant with a V-407 reactor, compared with a VVER-440 reactor.

1.3 Reactivity feedback was improved by ensuring a negative coolant temperature coefficient of reactivity during the entire life of the fuel.

Initially feedback improvements were made by including non-removable burnable poison rods that contain boron. Subsequently, at the stage of core design updating, a burnable poison within the fuel is to be used, i.e. fuel element with gadolinium poison. The employment of a core with "low neutron escape" will also allow for a change to be made in the refuelling strategy. This measure provides an opportunity to obtain econominal benefits due to either increased burnup or to the extension of the life by 5 or 7 % and thereby to reduce the fluence to the reactor vessel.

1.4 Other changes that were made to the reactor design include:

- a control rod guiding device that allows the insertion of CPS control rods in the event of core barrel failure;
- the elimination of one joint on the CPS nozzle ;
- the installation of a reactor core water level monitor for use under accidents conditions.

Some other changes were also introduced in other equipment of the reactor plant, such as a modified design of steam generator header.

The reactor coolant pump seal design was such as to prevent coolant leakage on loss of power supplies for 24 hours and on the interruption of seal cooling water supplies. Water lubrication of the RCP GTzN-1391 main bearing is employed.

Other important variations are:

- use of an extended reactor vessel shell that excludes welds in the core region;
- use of inconel protective jackets in the hottest nozzles to extend the life;
- changeover to use fuel movements for operational manoeuvres of the NPP;

Diagnostics systems for revealing faults during operation and in a shut-down reactor are being developed.

2. The principal features of the V-410 design

In the proposed design of the reactor plant for a new generation of NPPs, there are major changes in the concept of the design. First and foremost, is the wider use of passivity principles in the design of safety systems and in selecting the optimum design and layout decisions with regard to economical factors.

The goal was set which aimed to improve the level of safety, resulting in a probability of core meltdown during beyond-design-basis of accidents 10^{-6} - 10^{-7} per reactor year.

TABLE 3. STEAM GENERATOR MAIN PERFORMANCES

Parameter	V-320	V-392	V-410
1. Steam generator type	horizont.	horizont.	vertical
2. Steam pressure at SG outlet,MPa	6,27	6,27	7,35
3. Coolant temperature at SG inlet,°C	320	320	330
4. Coolant temperature at SG outlet,°C	289,7	289,7	296
5. Steam capacity, t/h	1470	1470	1600
6. Feedwater temperature,°C	220-164	220-164	230-120
7. Steam humidity at SG outlet,%	0,2	0,2	0,2
8. SG vessel diameter,mm	4290	4290	5000
9. Vessel length,mm	13800	13800	13100
10. SG vessel material	10GN2MFA	10GN2MFA	10GN2MFA
11. Header material(perforation area)	10GN2MFA	08Kh18N10T	08Kh18N10T-VD
12. Design pressure, MPa	7,84	7,84	12,5

Simultaneously, it is intended to improve the plant reliability giving a life for the main plant and equipment of 50-60 years.

Another objective was to reduce the dose committment for power unit maintenance to 1 man-Sy per year.

Higher secondary-side parameters have been set to increase the efficiency of the power unit, see Table 3.

A number of parameters of V-410 reactor plant have not yet been determined and these will be selected when results are available as the project develops.

Alternate options, wtih different numbers of fuel assemblies and different diameters of the reactor vessel, are to be analysed.

These sim of the alternate analysis is to allow an optimization of the core design to be made and to obtain a reduction of the fluence to the reactor vessel to 1.10^{19}n/cm as compared to $5.7.10^{19}$ in V-392 and V-320 projects.

The option of a core and vessel designed for V-392 project but with a 300 mm longer fuel column is included as one of the alternate designs under consideration.

For all design options the application of uranium-gadolinium and uranium-plutonium fuel that meets the requirements for load following of NPP is envisaged. All parts made of stainless steel in fuel assemblies are to be replaced by zirconium parts.

In the design a concept of "leak-before-break" will be realized. Water seals on primary pipework will be excluded and to meet this requirement the RCP design will be changed. Two alternates design options for the RCP are available, one a gland-type RCP and the other a glandless RCP.

A more extensive application of passive principles coupled with a combination of functions of normally operating active systems and safety systems require the execution of a large R&D programme.

In the V-410 design project (as compared to V-392) PHRS capacity will be sufficient, not only for afterheat removing decay heat following a complete blackout but also for ensuring safe cooldown in the event of primary pipeline breaks. In such a case the possibility of eliminating the containment spray system and relying upon the removal of containment heat via the PHRS will be considered.

3. V-407 reactor plant

The medium power (~ 600 MWe) design allows, at this stage of development, a merging of nuclear science and engineering to give the best in the most optimum way the high requirements for safety, reliability and economy. It is planned to achieving the following main objectives:

INTRODUCTION

In response to initiatives established by the U S Department of Energy (DOE) and Electrical Power Research Institute (EPRI) for a new generation of smaller, safer, and more economical nuclear reactors, Westinghouse has designed the AP600, an advanced passive pressurized-water reactor (PWR) that generates 600 MWe (see Figure 1) This advanced design satisfies the requirements established by DOE and the Advanced Light Water Reactor Utility Requirements Document, and can be available to meet the need for a safe, secure, and affordable power supply as early as the mid-1990s.

The AP600 is an elegant combination of innovative natural safety systems and proven technologies based on Westinghouse PWRs which have consistently achieved availability factors 10% above the U S national average during more than 20 years of operation Because the reactor vessel and internals are of essentially proven conventional design, no new manufacturing development is necessary

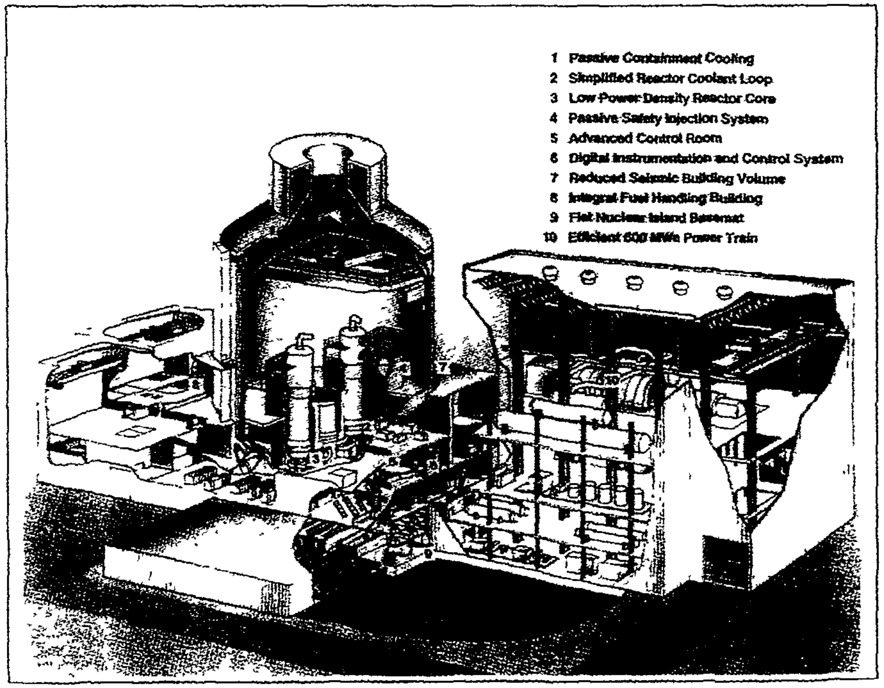


Figure 1 Cutaway of AP600

DESIGN SIMPLIFICATION

The essential technical concept underlying the AP600 design is simplification, the main features being a simplified reactor coolant system, simplified plant systems, and a simplified plant arrangement including a modular construction approach

The AP600 natural safety systems rely predominantly on the engineered use of natural forces such as gravity, convection, and natural circulation – a combination that provides enhanced reliability and public safety This reliance on natural forces also allows a significant reduction in equipment and components, particularly in the nuclear island, which uses 60% fewer valves, 75% less pipe, 80% less control cable, 35% fewer pumps, and 50% less seismic building volume than conventional reactors, resulting in major reductions in capital costs and a shortened construction schedule (see Figure 2)

The AP600 also features a low-power density core with a greater safety margin, proven high-reliability components, and an 18- or 24-month refueling cycle. The plant arrangement allows access to all buildings and systems that are important to safety, and enables all safety-related systems to be located in a single, compact area. All of these factors translate into a reactor that is less expensive to operate and maintain, and easier and less expensive to build

The AP600 can be constructed and operating within 5 years of a placed order A modular approach to construction accelerates the process and allows close supervision of the assembly of each module at the factory. Complete, prefabricated sections of the reactor can be shipped to the plant site and assembled The modular construction approach also facilitates standardization, a key issue with DOE, EPRI, and utility companies, and an important factor because assembly-line manufacturing allows greater quality control, fewer construction delays, and lower costs. It would also enable any nuclear engineer or operator to transfer more easily from one AP600 nuclear plant to any other Module development is currently underway at Avondale Industries, a leading manufacturer in the shipbuilding industry, and a member of the Westinghouse development team

Advanced microprocessor-based instrumentation and control (I&C) systems also simplify and improve plant operation and maintenance A digital, multiplexed control system takes the place of hard-wired analog controls and cable-spreading rooms, accounting for a significant decrease in control cable (80%

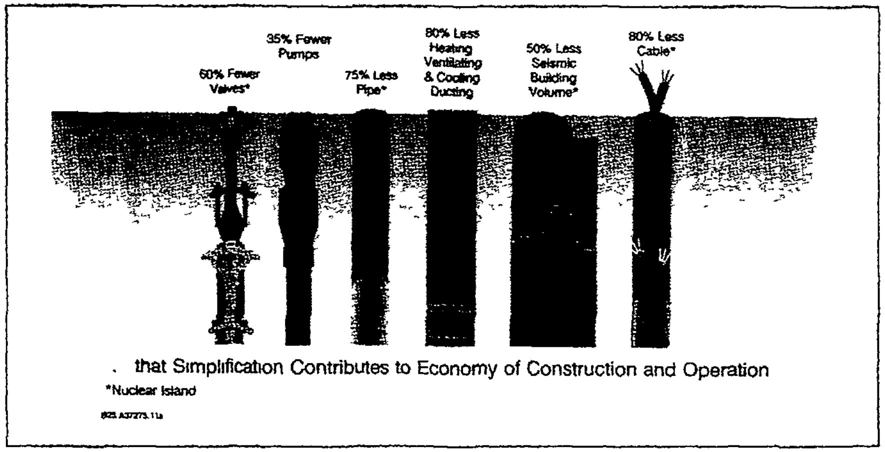


Figure 2 AP600 Simplification Contributes to Economy of Construction and Operation

less control cable than current nuclear plants). Data derived from extensive human factors studies are used throughout the I&C and control room design to enhance operability and decrease the probability of operator error. The computer-driven graphics and safety-qualified displays simplify the operators' tasks in assimilating information. Other AP600 I&C features that enhance the safety and reliability of the plant are: four-way redundancy, train separation, self diagnostics, and equipment monitoring.

DESIGN TEST STATUS

In order to achieve design availability for commercialization, the AP600 development program is timed to receive certification from the U S. NRC by 1995 Westinghouse is, therefore, currently preparing the Standard Safety Analysis Report (SSAR) for submittal to the NRC in 1992, and is performing a series of natural safety system tests.

Westinghouse has completed the AP600 conceptual design phase which included feasibility and design studies, and tests of major components unique to the AP600. The evaporative heat transfer tests were one of many tests conducted under this design phase. The results verified the feasibility of the passive containment cooling system (PCCS). The natural residual heat removal test was also performed to determine the feasibility of using a natural circulation heat exchanger. The reactor coolant pump high inertia bearing test verified the feasibility of using a high inertia bearing to increase flow during coastdown of the canned motor reactor coolant pump (RCP) All of these conceptual design test results verified the feasibility of the features unique to the AP600.

The second phase of the AP600 test program is in various stages of completion, and will verify the safety analysis computer models and confirm the feasibility of using AP600 natural safety systems to mitigate the consequences of design basis accidents and severe accident scenarios. Analyses and scaling tests are being performed at Westinghouse and with the cooperation of various national and international test organizations

The PCCS tests consist of four separate but concurrent testing programs. the PCCS integral heat transfer test, the large-scale PCCS test, the PCCS wind tunnel test, and the PCCS water distribution test. The data derived from the first completed part of the PCCS integral heat transfer test is being evaluated, and construction is underway for the large-scale PCCS test to begin in the Fall of 1992. These tests will examine the natural convection and steam condensation on the interior of the containment combined with the exterior water film evaporation, air cooling heat removal, and water film behavior. The test results will be compared to a detailed analysis model to verify the computer code used to predict the containment performance.

In the PCCS wind tunnel test, Westinghouse engineers will examine the operation of the PCCS for various wind conditions. The data derived from this test will complement the PCCS heat transfer test results. The PCCS wind tunnel test will establish a suitable location for PCCS air inlets in the shield building and examine the possibility of any natural draft air cooling interference, as well as determine the pressure loads across the air baffle within the shield building. The first part of this test was completed in July 1991, and data evaluation is currently being done. The second part, the detailed model and final verification, will be based on the outcome of the first part's results, and is expected to be completed in December 1991.

The first part of the PCCS water distribution test has been completed and the data is currently being evaluated. The second part, involving a full-scale, 1/8 section of the containment vessel, is under construction, with an expected completion date of October 1991. The purpose of this test is to demonstrate the capability to distribute water on the steel containment dome outer surface as part of the overall PCCS function.

Loss-of-coolant (LOCA) mitigation tests include the core makeup tank test (test organizations are currently under consideration, to be chosen November 1991), the long-term core cooling test (in the design phase now), the natural safety injection check valve test, and the automatic depressurization system (ADS) test.

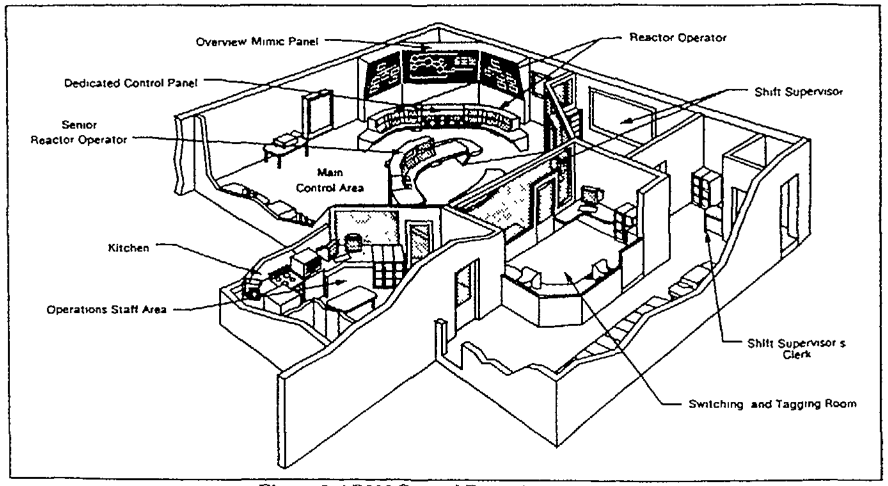


Figure 3 AP600 Control Room Arrangement


Figure 4 AP600 ... A New Concept in Safety

Preliminary hydraulic tests for 4-inch and 6-inch check valves, a series of tests in the natural safety injection check valve test program, were completed in June 1991, with the qualification testing scheduled for completion in December 1993 Additional test requirements will be established after reviewing the results of these preliminary tests The current plan is to test prototypic valves at operating temperature and pressure, and test for degradation after extended operation

The facility in Italy for the first ADS tests, the sparger performance, and quench tank loads have been completed, and instrumentation installation is currently underway with testing to begin in September 1991 Valve vendor selection is also underway for the valve performance testing, part two of the ADS tests It is scheduled for completion in the summer of 1992

The non-LOCA transient tests include the natural residual heat removal (RHR) heat exchanger test, and the departure from nucleate boiling (DNB) test The natural RHR heat exchanger test was completed in October 1990, and its results determined the heat transfer characteristics of the natural RHR heat exchanger, and the effect of mixing in the in-containment refueling water storage tank (IRWST) The test specifications for the DNB test have been drafted, with an expected completion date of December 1992 The DNB test will extend the lower flow limit of the NRC-approved DNB correlation as required for the design certification of the AP600 low-power density core design

The component design tests (RCP/steam generator channel head air flow, RCP water flow, RCP journal bearing, in-core instrumentation system, reactor vessel internals, instrumentation and controls cabinet cooling, and concrete/steel module structural tests) are all in various stages of completion or planning

The results of these test programs will provide fundamental engineering data not only to ensure NRC design certification, but also to provide utilities with added confidence in, and familiarity with all aspects of AP600 systems operations In parallel with these testing programs, Westinghouse is also preparing and verifying nuclear safety computer codes which are used to perform the safety analysis of the AP600

SUMMARY

As the need for power increases and the consequences of burning fossil fuels and dependence on imported oil become better understood, the U S is realizing the necessity of reviving the nuclear power option The simplicity, proven safety, and modular construction techniques make the AP600 the leading technology for the next generation of nuclear power plants in the 1990s and beyond

SBWR — SIMPLIFICATIONS IN PLANT DESIGN FOR THE 1990s

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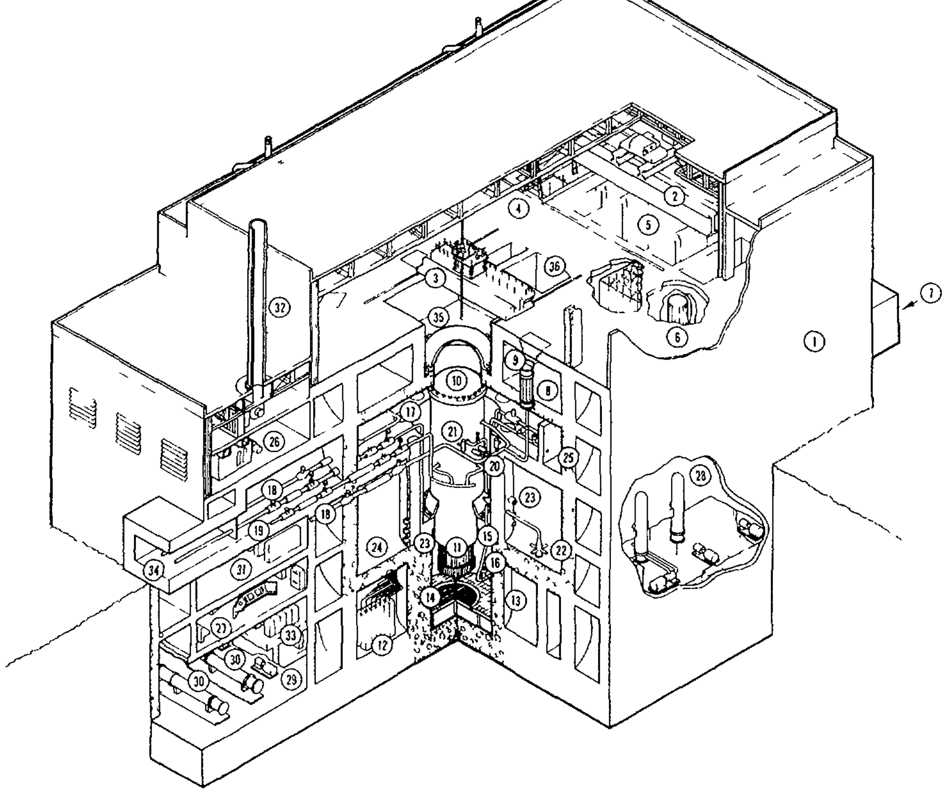
Abstract

An international co-operative effort has been ongoing for several years on the design and development of the next generation of Boiling Water Reactors (BWR) This effort utilizes the experience gained from the design, development and operation of nuclear reactors on three continents over the last thirty years The initial efforts led to the design of BWRs with the latest technologies in the 1300 MWe size range The advances in technology from that effort have been applied to the smaller (600 MWe) reactor sizes The smaller reactor is called the Simplified Boiling Water Reactor (SBWR) The ongoing development of the smaller reactors is resulting in a reactor design that either meets the requirements of each specific country or is easily adapted to that country's specific requirements

The design for the SBWR nuclear plant is a complete design for a 600 MWe power plant This paper provides a description of the SBWR including the design features and the analysis of their performance The SBWR is a plant that is significantly simpler to build, operate and maintain compared to currently operating plants The reason for this simplification is inherent in some of the key features – elimination of the forced recirculation system, use of simplified and enhanced safety features and the use of a simplified direct cycle system The simplified safety systems have resulted in the elimination of all safety-grade pumps and diesel generators and has enabled significant simplification of the plant design The safety systems utilize only natural forces or stored energy to cool the core and remove decay heat The key safety systems include a depressurization system and a gravity-driven cooling system to provide core cooling An isolation condenser is used to remove decay heat for all transients Decay heat is removed for accidents using a variation of the isolation condenser that functions automatically without valves or other moving parts

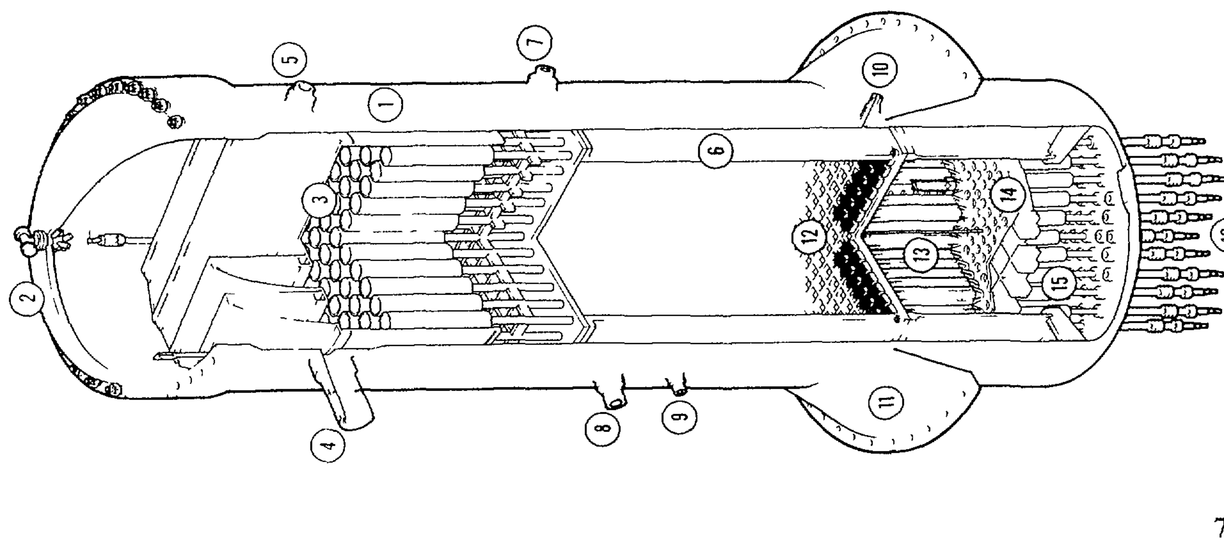
A. INTRODUCTION

Recently there has been increasing utility interest in potential future nuclear units combining the characteristics of smaller size, greater simplicity and more passive safety features This interest is driven by a worldwide slowdown in electrical growth rates which provides an incentive for smaller capacity additions, short construction schedules and by a recognition that smaller nuclear units offer potential simplifications with attendant economic benefits In response to such interest GE initiated studies in 1982



1 Reactor Building
2 Reactor Building Crane
3 Refueling Machine
4 Fuel Handling Machine
5 Spent Fuel Storage Pool
6 Spent Fuel Shipping Cask & Pool
7 Equipment Main Entry Hatch
8 Isolation Condenser Pool
9 Isolation Condenser
10 Reactor
11 Fine Motion Control Rod Drives
12 FMCRD Hydraulic Units
13 Reactor Pedestal
14 Under Vessel Servicing Platform
15 Lower Drywell
16 Shutdown Cooling Line
17 Upper Drywell
18 Main Steam Lines
19 Feedwater Lines
20 Depressurization Valves
21 Safety Relief Valves
22 SRV Quenchers
23 Horizontal Vents
24 Suppression Pool
25 Gravity Driven Cooling Pool
26 Building HVAC
27 Control Room
28 Residual Heat Removal System Heat Exchangers
29 Reactor Component Cooling Water System Pump
30 Reactor Service Water System Heat Exchangers
31 DC Batteries
32 Plant Stack
33 FMCRD Electric Panel
34 Steam Tunnel
35 Drywell Head
36 Steam Separator Storage Pool

FIG 2 Simplified boiling water reactor island



1 Reactor Pressure Vessel
2 RPV Top Head
3 Integral Dryer-Separator Assembly
4 Main Steam Line Nozzle
5 Depressurization Valve Nozzle
6 Chimney
7 Feedwater Inlet Nozzle
8 Reactor Water Cleanup/Shutdown Cooling Suction Nozzle
9 Isolation Condenser Return Nozzle
10 Gravity-Driven Cooling System Inlet Nozzle
11 RPV Support Skirt
12 Core Top Guide Plate
13 Fuel Assemblies
14 Core Plate
15 Control Rod Guide Tubes
16 Fine Motion Control Rod Drives

FIG 1 Simplified boiling water reactor assembly

- Reactor Isolated
- Isolation Condenser Removes Decay Heat to Atmosphere
- No Containment Heatup

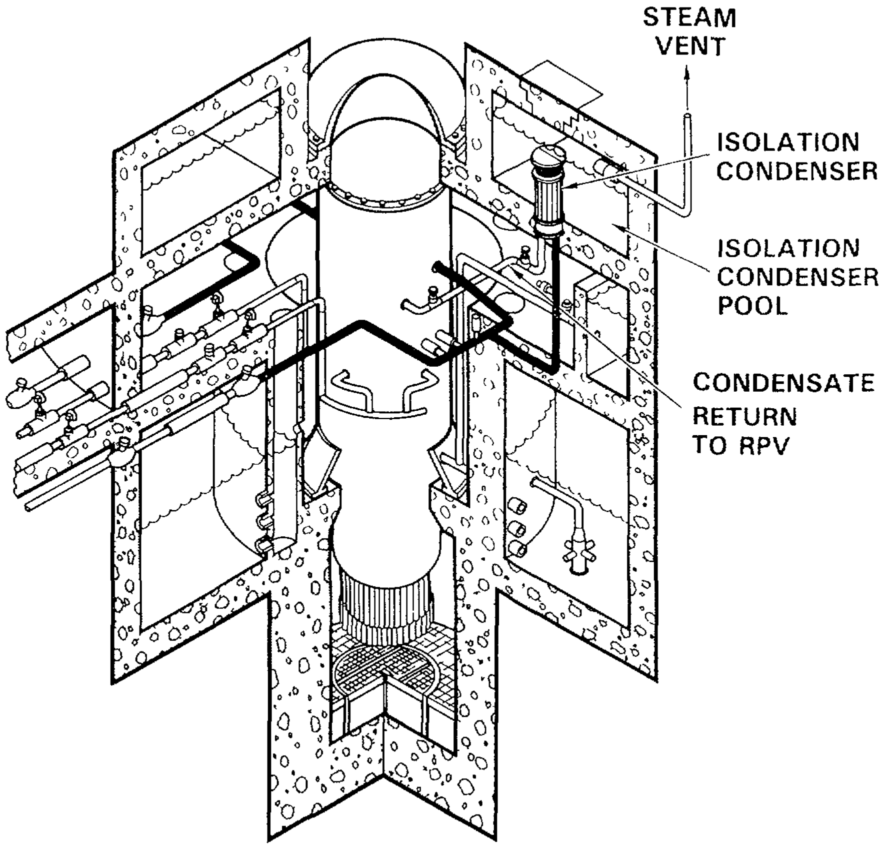


FIG 3 Isolation response

- Suppression Pool Absorbs Blowdown Energy
- RPV Depressurized by DPVs
- GDCS Floods RPV
- Isolation Condenser Removes Decay Heat
- No Containment Flooding for Most Breaks

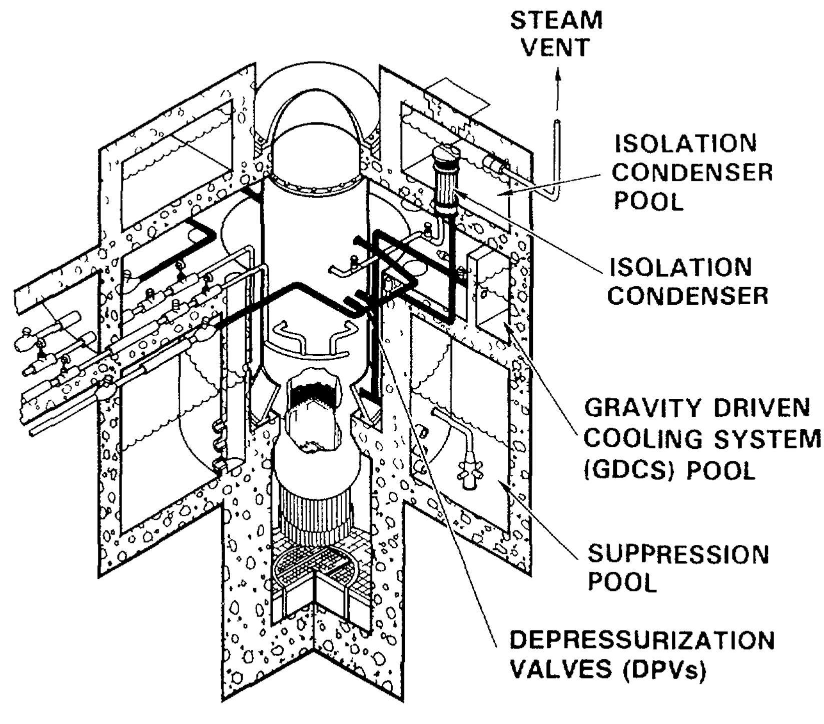


FIG 4 LOCA response

of a 600 MWe(e) BWR with simplified power generation, safety and heat removal systems The following basic objectives for the new design were established

- Power generation costs superior to coal
- Plant safety systems simpler than those employed in current designs
- Design based on existing technology
- Design shorter construction schedules
- Electrical rating in the 600 MW(e) range

The design described in the following section meets these objectives

B. DESCRIPTION

A brief description of GE's SBWR follows Additional information is provided in *References 1, 2 and 3*

The SBWR reactor assembly is shown in Figure 1 The SBWR Reactor Island is shown in Figure 2 The Turbine Island (not shown) is mounted on a common basemat with the reactor and service buildings

For passive emergency response the SBWR employs a suppression pool, a gravity driven cooling pool and an isolation condenser pool For isolation response the isolation condenser passively removes decay heat to the atmosphere with no containment heatup as shown in Figure 3 For loss of coolant events, the suppression pool absorbs the blowdown energy, the reactor pressure vessel is depressurized with depressurization valves, the gravity driven cooling system floods the reactor and the isolation condenser removes decay heat as shown in Figure 4 There is no major containment flooding for most LOCA events due to the compartmentalization of the containment upper and lower drywells The SBWR response to low elevation LOCAs and severe accidents provides for lower drywell flooding For all accident events, the isolation condenser removes decay heat, fission products are retained in the suppression pool and multiple barriers contain fission products inside the plant For SBWR accident response, passive safety features produce the result that no operator action is required for a period of at least three days Use of these and other passive systems allows the elimination of emergency diesel generators, core cooling pumps and heat removal pumps which simplifies plant design and licensing and will reduce plant costs Descriptions of specific simplifications are provided in the following paragraphs

Low Power Density Core and Natural Circulation

The selection of natural circulation as the means for providing coolant flow through the reactor, coupled with a 42 kW/liter core power density, provides a number of benefits to help satisfy SBWR objectives Compared to existing forced circulation plants, the natural circulation SBWR reactor offers low fuel cycle costs, a reduced number of operational transients and increased thermal margin for the transients which are expected to occur In addition, elimination of the recirculation loops, pumps and controls needed for forced circulation substantially simplifies the design

Isolation Condenser

An isolation condenser is placed in the isolation condenser pool An isolation condenser is employed in many earlier BWR designs When the reactor vessel is isolated from the turbine condenser, the isolation condenser controls reactor pressure automatically without the need to remove fluid from the reactor vessel Thus, the conventional BWR safety/relief valves, which open and close to discharge reactor vessel steam to the suppression pool, are not needed in the SBWR concept

Gravity Driven Core Cooling System

The gravity driven core cooling system provides a simple approach to emergency core cooling, eliminating the need for pumps or diesels It requires more water in the reactor vessel above the core and additional depressurization capacity so the reactor can be depressurized to very low pressures and gravity flow from the elevated gravity driven cooling pool can keep the core covered However, because there are no large pipes attached to the vessel near or below the core elevation, the design insures full core coverage for all design basis events The additional water provided also has other benefits such as reduced pressure rates for transients and substantially more time *before core uncovering in multiple failure scenarios*

Passive Containment Cooling System

The passive containment cooling system provides a three-day passive cooling capability for the containment using natural convection processes Passive containment cooling is provided by the isolation condenser Decay heat is rejected to the isolation condenser pool and then to atmosphere No active pumps or diesels are needed to *provide for heat removal Beyond three days, water makeup with simple operator action* is all that is needed to continue the passive cooling function This feature, together with drywell flooding for design basis and severe accident events, offers the potential of the site radiological consequences for design basis events or severe accidents being a very small fraction of licensing limits

Simplified Control and Electrical System

A passive natural circulation air system is used to provide habitability control for control room operators This feature, combined with the passive gravity driven core cooling system and the passive containment cooling system allows safety grade emergency diesel generators to be eliminated from the SBWR concept The building space needed for the control complex is less than half of that required on conventional designs because of the advanced man-machine interface incorporated into the design as well as the use of a plantwide, multiplexing system and extensive use of standard microprocessor based control and instrumentation modules

Simplified Power Generation Systems

A tandem, double-flow turbine with 52-inch last stage buckets reduces building size and simplifies the condenser and piping arrangement A single string of feedwater heaters is employed to reduce costs and simplify the feedwater and condensate system arrangement Variable speed motor-driven feed pumps provide reduced cost and simpler controls The pumps used to pump forward the high pressure drains have been eliminated by regulating the feed pump suction pressure to allow the drains to be pressure driven into the feedwater cycle to reduce capital and operating costs The separate steam seal system used in earlier applications is eliminated based on evaluations which indicate that the contribution of this system to radiation exposure reduction (particularly with modern BWR fuel) is insignificant The main condenser is located under and to the side of the turbine, allowing the turbine pedestal to be lowered, thus reducing capital costs

Development Status

The SBWR Conceptual Design was completed in 1989 Development and Testing work is 90% completed in terms of funds expended and funds are committed for the remaining tests Refer to the companion paper, *Reference 4, presented at this* conference for further information

GE was selected by DOE and EPRI in 1989 to carry out a $140 million program to design develop and certify the SBWR by 1995 This program is being carried out by an international team that currently includes the following participants

SBWR *Team Members*

United States

- Bechtel North American Power Corporation (Bechtel)
- Burns and Roe Company (B&R)
- The Cleveland Electric Illuminating Company (CEICO)
- Commonwealth Edison Company (CECO)
- Foster Wheeler Energy Applications, Inc (FWEA)
- General Electric Co (GE)
- Massachusetts Institute of Technology (MIT)
- Philadelphia Electric Company (PECO)
- Public Service Electric and Gas (PSE&G)
- Southern Company Services, Inc (SCS)
- University of California—Berkeley (UC–Berkeley)
- Yankee Atomic Electric Company (YAECO)

Italy

- Ansaldo Spa (Ansaldo)
- Comitato Nazionale per ricerca e per la sviluppo dell Energia—Nucleare e delle Energie Alternative (ENEA)
- Direzione sicurezza Nucleare e Protezione Sanitaria (ENEA-DISP)
- Ente Nazionale per l'Energia Elettrica (ELEL)

The Netherlands

- N V Gemeenschappelijke Kernenergiecentrale (GKN)
- N V Tot Keuring Van Electrotechniche Materialen (KEMA)
- Stichting Energieonderzoek Centrum Nederland (ECN)
- NUCON Engineering and Constructing B V (NUCON)

Japan

- Hitachi, Limited (Hitachi)
- Toshiba Corporation (Toshiba)

Indonesia

- Indonesia National Atomic Energy Agency (BATAN)

Summary

In summary, international cooperative programs are in place to bring into being a new generation of BWR plants These plants will incorporate the best features and technology from the current generation of worldwide BWRs and will represent "world class" standard plants to serve utility needs in the 1990's and beyond

References

1 "Simplicity The Key to Improved Safety Performance and Economics," R J McCandless and J R Redding, Nuclear Engineering International, November 1989

2 "Status of Advanced Boiling Water Reactors," D R Wilkins, J F Quirk, and R J McCandless, 7th Pacific Basin Nuclear Conference, San Diego, California, March 4-8, 1990

3 "GE Advanced Boiling Water Reactors," D R Wilkins, American Power Conference, Chicago, Illinois, April 23-25, 1990

4 "SBWR *Technology and Development,"* A S Rao et al , *Presented at the* IAEA Technical Committee Meeting, Rome, Italy, September 9-12, 1991

ENHANCED SAFETY REACTORS FOR NUCLEAR DISTRICT HEATING AND CO-GENERATION PLANTS

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Abstract

A safety concept of AST-500 reactor designed for nuclear district heating plants, which is based on the self-protection principle, is briefly reviewed. In addition to the reactor major engineering decisions, made include; the use of an integral arrangement of the reactor equipment; coolant natural circulation; a guard vessel; intermediate heat transfer circuit and double isolation valves in auxiliary system pipelines forming an additional closed localizing system; passive safety systems which self-actuate when the plant parameters deviate beyond the normal operation conditions. The reactor is maintained in a safe condition without engineered safety systems actuation or operator intervention. The main engineering decisions of the AST-500 reactor design were successfully carried over to the design of the ATETS-200 power plant intended as a combined cycle power plant producing electricity, steam and hot water. The attained level of reactor safety ensures limited radiological consequences under the most severe accident conditions in both the AST-500 and the ATETS-200 which enables them to be sited in the immediate vicinity of large cities.

Introduction

The concept of prospective enhanced safety reactor plants is based on the combination of engineered features preventing radioactive products being released beyond the protective and localizing barriers, not only under normal operation and in design-basis accidents, but in severe beyond design basis accidents as well.

The passive principles underlying the safety concept are:

- inherent safety properties,
- passive protective and localizing safety systems and devices,
- natural "defence-in-depth" physical barriers on the way of radioactivity propagation.

The passive safety features maintain independence from power sources and operator actions and since they are self-acturated, operate on the basis of natural processes.

These principles were mostly implemented in AST-500 reactor for nuclear district heating plants (NDHP) and in its follow-up modifications.

In the case of the NDHP, siting led to the need to develop additional requirements on safety as compared to NPP which met existing national regulations. These plants incorporate the requirements for prevention of fuel melt under all reactor pressure vessel leakage conditions, and the need to take into account external events like aircraft crash and blast waves. The requirements on radiological release effects were also made tighter.

In this connection the AST-500 reactor plant safety assurance was based on the maximum use of natural physical processes and passive systems, on the application of direct-acting features and on the "fail-safe" design principle.

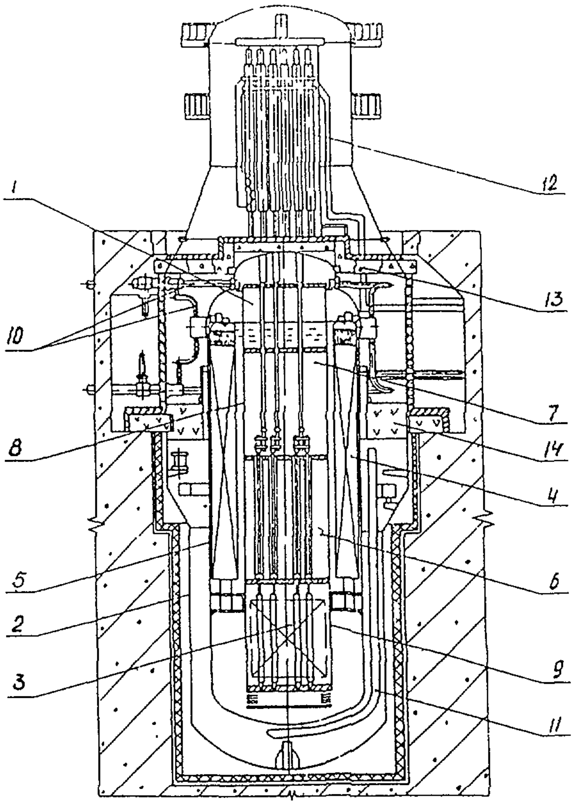
The main engineering decisions applied to the AST-500 were then successfully used in the design of the ATETS-200 nuclear power plant, intended for combined production of electricity, steam and hot water. The attained levels of guaranteed safety ensures limited consequences from the most severe accidents in both the AST-500 and ATETS-200 nuclear plants, and the possibility of locating them in the immediate vicinity of large cities.

Principal Decisions on AST-500 Safety Provision

The AST-500 reactor is one of the well studied and proven type of pressurized water reactors and is characterized by the enhanced property of power self-control due to strong negative temperature, power and void reactivity (Fig. 1, 2).

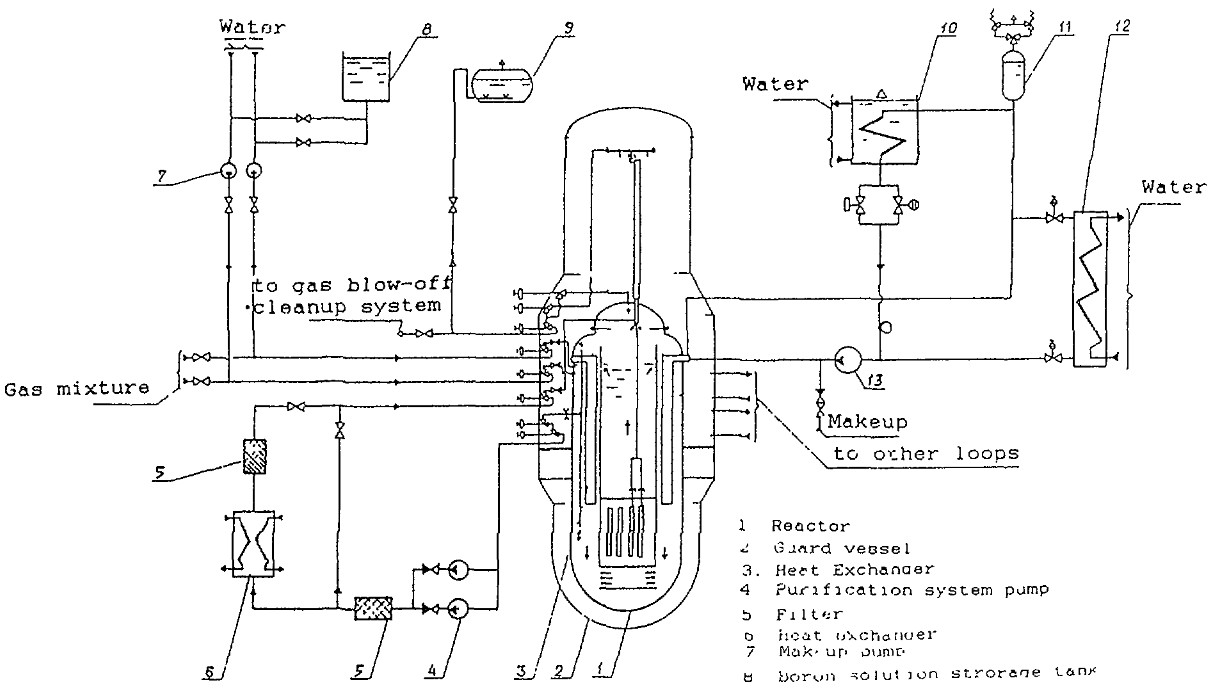
Coolant natural circulation is used under all operational conditions along with an integral layout of primary circuit equipment, i.e. the core, heat exchangers and pressurizer arranged within the single reactor pressure vessel. As a result this has led to a simplified reactor structure, omission of the circulation pumps, a reduced length of primary circuit pipework and the elimination of the large diameter pipe potentially hazardous in respect to loss of coolant accidents.

The effect of increased coolant flowrate with a reactor thermal power rise and flowrate self-shaping over the fuel assemblies, results in considerable increase in DNB margins at normal and emergency operating conditions. That effect is associated with the natural circulation of the coolant in the reactor. A thick layer of water surrounding the core maintains a low neutron exposure to the reactor pressure vessel. By including a large heat exchanges surface directly inside the reactor, the availability of the highly reliable passive emergency residual heat removal systems (ERHRS) enabled safety valves in the primary circuit to be eliminated, avoiding the hazardous condition arising from their failure to close and to make use of the heat removal principle for the reactor overpressure protection.



1 - reactor
2 - guard vessel
3 - core
4 - primary/secondary heat exchanger
5 - reactor pressure vessel
6 - flow chimney
7 - common draught section
8 - guide tubes-connecting devices assembly
9 - core support barrel
10 - secondary circuit pipelines
11 - rotary device
12 - CRDM assembly
13 - biological shielding blocks
14 - support belt of a guard vessel

FIG 1 AST-500 reactor unit.



1 Reactor
2 Guard vessel
3. Heet Exchanger
4 Purification system pump
5 Filter
6 heat exchanger
7 Make-up pump
8 Boron solution strorage tank
9 Bubbler
10 Decay heat removal system heat [illegible]
11 [illegible]
12 [illegible]
13 2-[illegible]

FIG. 2. AST-500 RP flow diagram.

The following properties of AST-500 reactor are of importance to safety:

- low power density
- low operating parameters
- large water inventory in the reactor causing its inertia and relatively slow development of emergency transients.

The duration of transients amounts to tens of minutes or even hours. The slow reactor response time is an important property giving ample time to make decisions particularly for correcting operator errors.

For the protection of heat consumers (hot water customers), a three-circuit heat transfer scheme with an intermediate pressure barrier is used.

The reactor pressure vessel (RPV) arrangement with a second RPV as a guard vessel (GV) - is intended to guarantee keeping the core covered and radioactivity confinement in accidents with primary circuit breaks; this is principally a new decision used in designing the AST-500 plant. The guard vessel is an additional safety barrier which eliminates core uncovery and fuel element overheating even in the most adverse case of loss of integrity of the reactor pressure vessel. The GV makes it unnecessary to use engineered features for emergency core cooling NPP reactors. Double isolation valves, which are closed by precompressed springs on receipt of an emergency signal, are installed in all primary circuit auxiliary systems pipelines leaving the GV. For localizing leakage in the case of primary/secondary heat exchangers leakage, the isolation valves in the secondary circuit loops are used which are located on the GV.

The primary circuit auxiliary system pipework is arranged in a leaktight containment or in leaktight rooms equipped with features for radioactivity releases mitigation (filters, bubblers).

To achieve reactor shutdown and to maintain it in a safe condition the following are envisaged:

- absorber rods inserted into the core under CRDM electric motion condition in response to the signals of different physical nature;
- absorber rods insertion under gravity in response to the signals of different physical nature;
- absorber rods insertion resulting from actuation of direct-acting electric breakers in CRDM immediately on a pressure rise in the reactor;
- forced injection of boric acid;
- passive injection of boric acid;

The ERHS is a three-channel system which operates by coolant natural circulation in all the circuits, without external power sources. The water inventory in the ERHRS tanks provides sufficient water for system operation over several days. One ERHRS channel is sufficient for the reactor residual heat removal. Residual heat removal through the secondary circuit loops is effected by:

- active connection of the ERHRS,
- passive connection of the ERHRS by direct-acting devices,
- constantly operating channel of ERHRS.

Passive heat removal is also envisaged via the ERHRS channel arranged immediately on the reactor which actuated by direct-acting devices. Evaporation of the secondary circuit water through pilot operated relief valves serves as a backup emergency heat removal channel.

Main results of AST-500 Safety Analysis

At normal operation the radiological effect of the AST-500 is substantially lower (by 10^4 times) than that of natural radiation background as a result of the elimination of radioactive leaks and discharges, the leak-tightness of the circuits and the low neutron fluxes beyond the reactor pressure vessel. In design basis accidents reactor safety is ensured over long periods of time (more than a day) without the need for supplies power and operator intervention, by the passive systems and devices for reactor shutdown, residual heat removal and radioactive products confinement/containment.

In beyond-design-basis-accidents coupled with redundant safety system failures, the core damage is prevented by the reactor's intrinsic safety properties (self-protection), by the actuation of backup passive direct-acting devices, and by operator personnel corrective actions involving backup safety systems actuation (there is an ample time margin for that). A postulated accidental loss of heat removal without scram, is considered to illustrate the inherent plant properties, while decision on the reactor design resulted in the accident being technically non-relizable (Fig. 3). Such an accident is associated with primary circuit over-heating and over-pressure conditions in the reactor in spite of ERHRS channels' actuation. Reactor power decreases due to the negative reactivity coefficients. Plant parameters stabilize at a reactor pressure of 3.8 MPa.

The loss of integrity of the lower part of the rector vessel of diameter equivalent to 45 mm pipework is accompanied by coolant outflow into the lower part of the guard vessel, a decrease in the reactor pressure and a pressure rise in the guard vessel (Fig. 4). In response to a pressure rise signal from the GV the emergency protection system actuates and the plant is brought into the cooling down mode. Would the water level fall in the draught section below the overflow windowns coolant circulation in the main flow path terminates. In this case steam-condensate circulation regime develops due to steam generation in the core and steam condensation on the uncovered portion of heat exchangers. Equalizing of pressures in the reactor and the GV would be reached in approximately one hour into the accident. The core remains covered by water during the whole accident. Reactor makeup is not required.

Beyond-design-basis-accident, with a cleanup system pipeline rupture (of maximum possible cross section) is considered to occur with the non-closure of double isolation valves (Fig. 5). It is additionally assumed that residual heat is removed from the reactor through incomplete number of the ERHRS channels. This accident is characterized by maximum loss of the reactor coolant. The time available to undertake corrective actions (the reactor makeup to keep the core covered) is not less than 24 hours.

Therefore, in the beyond design basis accidents with postulating a large number of safety system failures the core cooling is ensured during long period of time. There is no fuel claddings overheating and loss of tightness. Radiological consequences of accidents are substantially lower than the limits specified by regulations and are close to the natural radiation background.

Owing to the engineering decisions adopted for ensuring AST-500 safety and to the defence in depth attained in the beyond design basis accidents the AST-500 reactor plant may be claused as the system of ultimate attainable safety.

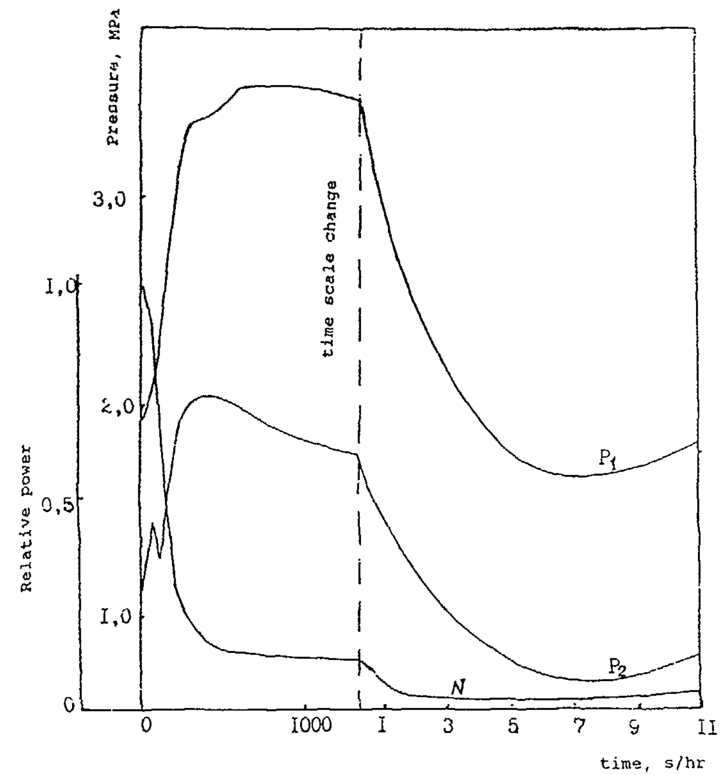


FIG. 3. 'Stop grid' without scram accident.

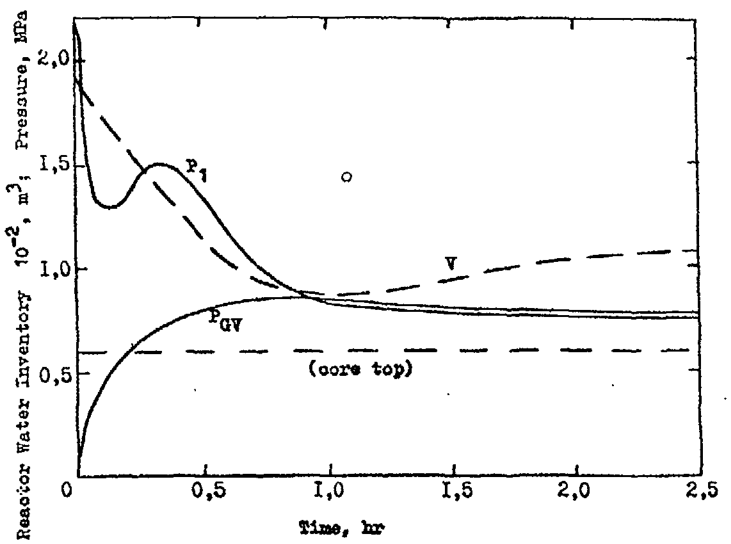


FIG. 4. Reactor pressure vessel loss of integrity in bottom part of 45 mm diameter equivalent.

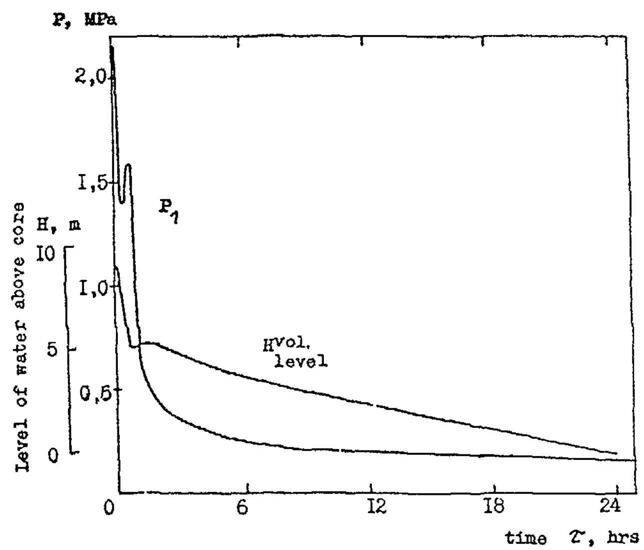


FIG. 5. Clean-up system pipeline rupture beyond a guard vessel boundary with failures of isolation valves and of two ERHRS channels.

AST-500 Main Design Characteristics

Parameter	Value
REACTOR	
Type	Integral PWR without boiling
Thermal power , MW(th)	500
Coolant circulation	Natural convection
Diameter / Heigt , mm	5320 / 25290
Water volume , m^3	190
Neutron fluence on RPV , n/cm^2	10^{16}
CORE	
Fuel	UO_2
Fuel lifetime, full power years	up to 10
Fuel enrichment , %	up to 2.0
Power density , MW/m^3	up to 30
Number of CRDM	36
(boric acid is not used for reactivity control)	
PRIMARY CIRCUIT	
Pressure , MPa	2.0
Temperature , °C	
inlet	131
outlet	208
SECONDARY CIRCUIT	
Pressure, MPa	1.2
Temperature, °C	
inlet	87
outlet	160
Number of loops	3
DISTRICT HEATING CIRCUIT	
Pressure , MPa	2.0
Water temperature , °C	
direct	144
return	64
GUARD VESSEL	
Diameter / Height , mm	8150 / 27950

ATETS-200 Reactor For Nuclear Co-generation Power Plant

ATETS-200 reactor has been developed on the basis of AST-500 pilot plant design and construction experience using technically proven design decisions and technologies of VVER reactors for NPP and those for nuclear icebreakers whose operability has been verified by the long-term successful operation.

The ATETS-200 safety concept is based on the design decisions of AST-500 reactor plant whose qualitatively new level of safety has been recognized by the IAEA (PRE-OSART mission on the Gorky NDHP). The ATETS-200 power unit may be used for:

- electricity generation;
- combined production of electricity and hot water (co-generation);
- electricity and process steam co-generation;
- sea and salted water desalination with independent auxiliary power supply for the desalination plant.

The economic efficiency of ATETS-200 power unit rises substantially owing to electricity and heat combined production with reduction in irretrievable losses of heat.

Enhanced reliability and safety of ATETS-200 is provided by:

- reactor self-protection and its power self-control due to negative reactivity coefficients (on power, void, fuel and coolant temperature) over the whole range of operating parameters variation;

- natural coolant circulation in the primary circuit in all modes of the reactor plant operation;

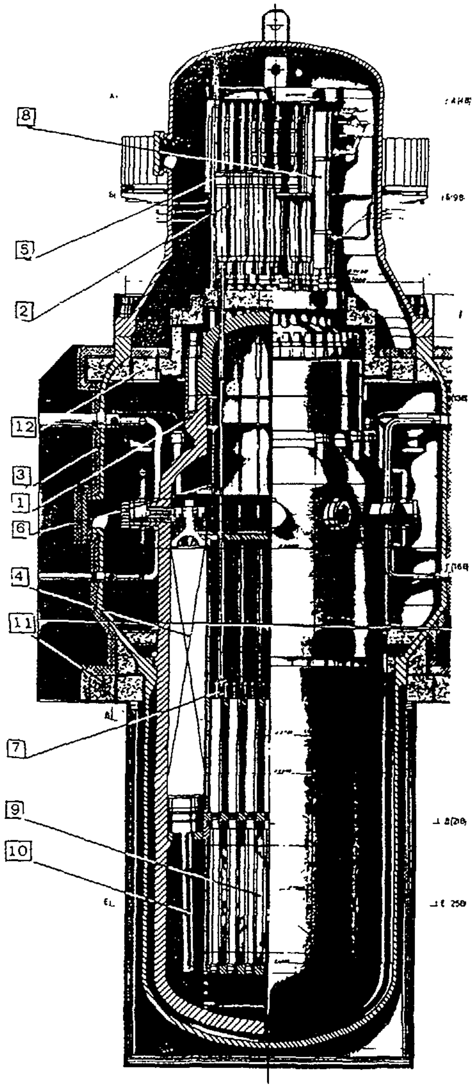
- passive principle of emergency residual heat removal system actuation and operation with natural coolant circulation in all the circuits from the reactor to the ultimate heat sink large time interval (not less than three days) withing which the reactor plant is maintained in a safe state without operator intervention, even without personnel presence in a plant;

- redundancy of safety systems and components, use of diverse safety systems;

- extensive use of self-actuated devices to initiate safety systems actuation (for a reactor shutdown as well) would the most important operating parameters of the reactor plant deviate beyond the present limits;

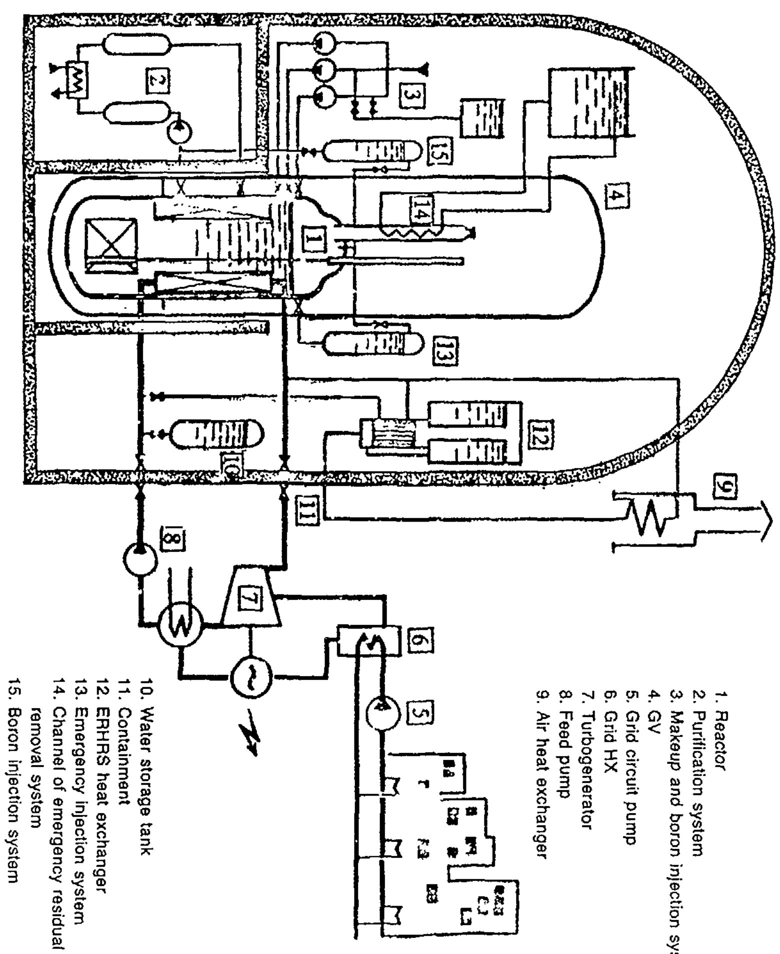
- low power density of the core, increased value of DNB ratio self-regulation of coolant flow through fuel assemblies that provides high fuel operation performance under normal and emergency conditions;

- large heat accumulating capacity of the reactor, slow progression of transients and emergencies, large time available for operator



1 Reactor vessel
2 Upper block
3 Guard vessel
4 Steam generator
5 CPS control rod drive
6 In-vessel barrel
7 Guide tubes - connecting devices unit
8 Condenser
9 Core
10 Suspension of ionization chamber
11 Biological shielding of the first level
12 Biological shielding of the secondary level

FIG. 6. ATETS-200 reactor unit.



1. Reactor
2. Purification system
3. Makeup and boron injection system
4. GV
5. Grid circuit pump
6. Grid HX
7. Turbogenerator
8. Feed pump
9. Air heat exchanger
10. Water storage tank
11. Containment
12. ERHRS heat exchanger
13. Emergency injection system
14. Channel of emergency residual heat removal system
15. Boron injection system

FIG. 7. Flow diagram of ATETS-200 co-generation power plant.

to analyse emergencies and, if necessary, to organize the accident management;

- the reactor integral layout, minimization of penetrations and their location in upper part of the reactor vessel, large water inventory over the core, negligible neutron irradiation of the reactor pressure vessel;

- use of a guard vessel that allows a substantial reduction in the amount of local radioactivity and keeps the core covered during the accidents involving a loss of reactor pressure vessel integrity. Simplicity of circulation circuits and residual heat removal channels, clearly understood and well studied thermal-physical processes, ensure confidence to the design.

ATETS-200 Main Design Characteristics

Parameters	Value
Reactor thermal power , MW(th)	up to 620
Plant electric power , MW(e)	up to 210
CORE	
Height , m	2.5
Max. fuel rod linear hea rating , W/cm	260
Fuel enrichment , %	3.0
Number of fuel assemblies	109
Fuel burn-up , MW.day/t	32000
Fuel lifetime , full power years	8
Refuelling interval , years	2
Number of CRDM	36
(boric acid is no used for reactivity control)	
Neutron fluence on RPV , n/cm^2	10^{17}
PRIMARY CIRCUIT	
Flow path height , m	9.4
Core outlet temperature , °C	340
Pressure , MPa	15.5
SECONDARY CIRCUIT	
Steam pressure , MPa	4.5
Steam temperature , °C	295

Residual heat is removed through:

- steam generators when the feed water is available and rejection of heat into the steam turbine condenser;

- steam generators when feed water is available and heat is rejected into the grid heaters (heating grid);

- steam generators by natural circulation of steam-water mixture on actuation of emergency heat removal channels connected to the secondary circuit loops (into service water and atmosphere);

- condensers of ERHRS channel which is located on the reactor and operates at natural coolant circulation in all the circuits;

- heat exchangers of primary coolant clean-up system.

Safety analysis is based on a combination of deterministic and probabilistic approaches. Safety analysis involved consideration of a wide spectrum of design basis and beyond design basis reactivity and heat removal accidents with primary coolant looses including reactor vessel loss of integrity, external effects such as earthquake, aircraft crash, shock wave and operator errors, including inaction.

Engineering decisions and safety systems adopted in ATETS-200 virtually exclude core melt accidents and the result severe radiological consequences.

Core damage probability is substantially reduced compared with that for NPP existing in the world and does not exceed 10^{-8} per reactor-year. Such accidents do not make a significant contribution to the risk to the population and completely eliminates the necessity for evacuation of the population.

DESIGN STATUS AND PERSPECTIVES OF ADVANCED WATER COOLED REACTOR APPLICATION FOR PRODUCTION OF ELECTRICITY AND HEAT IN CZECHOSLOVAKIA

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Abstract

At present there are 8 units with VVER-440 reactors in operation in the CSFR. Moreover, further 4 units with VVER-440 reactors and 2 units with VVER-1000 reactors are now under construction. It is also assumed to bring into operation 2 additional units with APWR after the end of this decade.

SKODA as a main equipment manufacturer and supplier of power plants in the CSFR has already about 5 years ago recognized the necessity of substantional improvement of technological, safety, and cost features of all nuclear power plant (NPP) units which will be in the future constructed in the CSFR, and delivered by SKODA. For this reason, at the end of 1985 year, the start of preliminary design study was initiated with the aim to develop an APWR type unit of medium power range (500-600 MWe) that should be appropriate for electricity and district heat cogeneration. The initial design stage was finished in the first half of 1990 year with the result that it is reasonable to proceed the completion of final design and to supply such a NPP with a predominant participation of domestic industry. An information is given in the submitted paper with regard to results of the above mentioned preliminary design study.

1. STATUS AND PERSPECTIVES OF ENERGY PRODUCTION IN THE C S F R

At present the total installed electrical capacity of all power stations in the CSFR is nearly 15 GW and the share of NPPs in the total electric power generation amounts about 27 %. After completion of all NPPs which are now under construction this share will reach the value of about 42 % in the second half of the current decade.

The future development of energy production in the CSFR is supposed to be very strongly affected in several next years by the transition to free market economy and also by the necessity to solve extremely unfavourable ecological situation which has been mainly caused by the extensive and moreover inefficient combustion of domestic inferior coal with high sulphur content in coal-burning power plants. The present significant increase of energy prices in the CSFR will necessarily cause the successive reduction of energy consumption along with the effort to produce energy in a more effective way. From above described state of affairs follows that in the near future capital investments needed for upgrading of already operated power plants will tend to dominate those associated with the construction of new energy sources. It is therefore reasonable to assume that an increase of total power generation in the CSFR can be expected mainly after the year 2000. In spite of the above mentioned situation, the role of a nuclear power generation should remain important. That is why both main energy production companies in the CSFR have jointly initiated a bid invitation process for delivery of two units with APWR reactors (rated power 1000-1400 MWe per unit) or for delivery of two mid-size passive units (rated power about 600 MWe per unit). The beginning of commercial operation of both units is demanded to be in the time interval between 2000 and 2005 year.

2. OBJECTIVES OF THE PRELIMINARY DESIGN STUDY RELATED TO THE N P P WITH MID-SIZE A P W R UNITS

Objectives of the preliminary design study performed by SKODA were following:

(1) to identify and analyse main problems associated with the development and construction of mid-size passive units,
(2) to identify the scope of all design and other activities which are of particular concern with regard to delivery of mid-size units,
(3) to assess the capability of SKODA, and of domestic industry to carry out all necessary activities in the field of engineering, manufacturing, and construction of mid-size units, and to determine the scope of appropriate technology transfer,
(4) to perform a preliminary assessment of important economic parameters associated with the design and construction of mid-size units,
(5) to establish the future business strategy of SKODA in the field of nuclear power generation.

3. REQUIREMENTS REGARDING THE DESIGN, OPERATING CONDITIONS AND PERFORMANCE OF THE N P P

Before the beginning of design activities the following design objectives and requirements had been set down.

(1) The NPP must enable simultaneous cogeneration of electricity and heat with the capability to extract heat in the range up to 600 MW per unit to district heat systems. With regard to necessary redundancy of heat supply there must be situated at least two units on the site.

(2) From the point of view of safety its level must enable to locate the NPP relativly close to areas with high population density (up to 15-20 km). For this reason it is necessary to meet following safety goals:
- core melt frequency: 10^{-5} - 10^{-6} per year,
- severe release frequency: 10^{-7} - 10^{-8} per year.

(3) The units of the NPP must be solved with a view to apply the standardization of reactor coolant system main components for following power range groups:
- 500-600 MWe: 2 or 4/2-loop arrangement,
- 900-1000 MWe: 3-loop arrangement,
- 1200-1300 MWe: 4-loop arrangement.

(4) The solution of the NPP must enable to provide low specific capital investment costs, nuclear fuel cycle costs, and operation costs which will be competitive with those attained in contemporary large-power NPPs. From this follows that it is necessary to aim at: simplified plant arrangement, short construction schedule, high burnup of fuel, refueling cycle of 18-24 months, simplified maintenance and repair procedures, etc.

(5) The design of the NPP must enable to reach:
- station lifetime: 50-60 years,
- availability: greater than 0.85.

4. BRIEF DESCRIPTION OF N P P

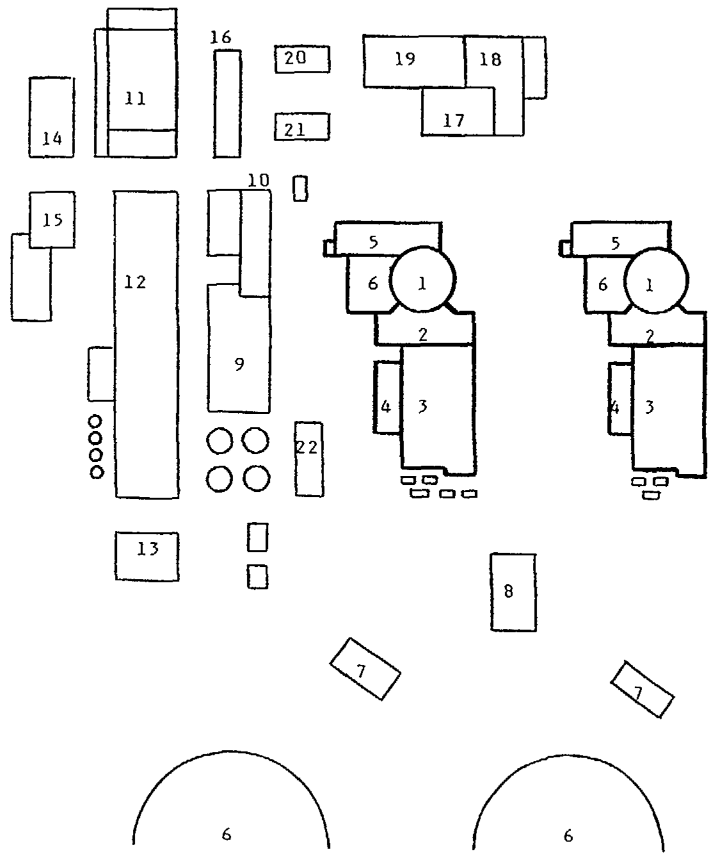
4.1 S i t e l a y o u t

The site layout is arranged so that it is possible to operate individual units separately and independently. The simplified site layout is shown in Figure 1.

4.2 N u c l e a r i s l a n d

The reactor coolant system (RCS) consists of: reactor (dimensions of the pressure vessel are nearly the same as in the case of VVER-1000 reactor), 2 vertical steam generators, pressurizer, RCS main loop piping (2 hot and 4 cold legs) and 4 main reactor coolant pumps. An arrangement with 2 VVER-type horizontal steam generators was studied as well. The square configuration of fuel in the active core was selected. The design of pressure vessel and internal parts of reactor is very similar to solutions applied in VVER-1000 reactor with the exception of active core and fuel assemblies. Main parameters of RCS and turbine system are presented in Table 1. Schematic representation of RCS is then shown in Figure 2.

The reactor coolant system is enclosed in cylindrical double-wall containment of conventional design (inner shell made of steel) with the vented space between both shells. The outer concrete shell is designed to resist all design basis loads caused by severe external events (earthquake, blast wave, aircraft crash, etc.).

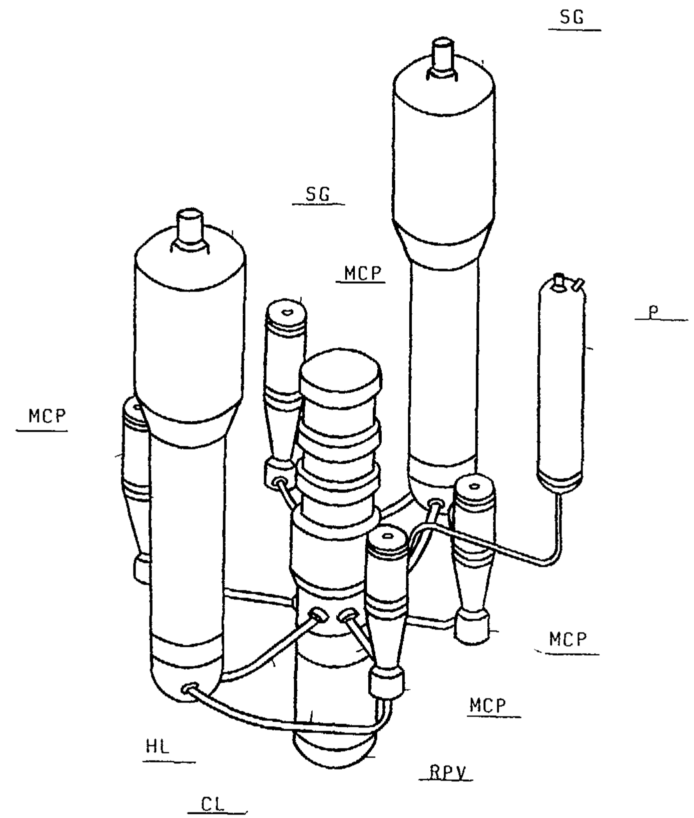


1 - reactor building
2 - electrical equipment building
3 - turbine building
4 - heat extraction station
5 - reactor auxiliary building,microtowers, dieselgenerator
6 - service building
7 - cooling water pump house
8 - district heat supply station
9 - administrative building
10 - main control building, simulator
11 - workshops,stores
12 - water treatment building
13 - auxiliary boiler building
14 - medical service center
15-main entrance building
16-firehouse
17-labs
18-active workshops
19-radioactive waste treatment building
20-flammable substances storage
21-gas storage
22-compressor and refrigerator station

FIGURE1 : Station layout - plan

TABLE 1. Main parameters of the NPP

PARAMETER	VALUE
A. UNIT	
1. - number of units on site	> 2
2. - gross electrical output, MW	610
3. - net electrical output, MW	573
4. - net efficiency	0.32
B. REACTOR	
1. - thermal power, MW	1790
2. - number of fuel assemblies	121
3. - fuel assambly array	18 x 18
4. - active core height/diameter, m	3.4/2.855
5. - fuel enrichment, w/o U-235	3.6
6. - average linear power, kW/m	14.5
7. - average power density, kW/l	82.2
8. - average heat flux, kW/m^2	486
9. - coolant outlet pressure, MPa	15.9
10. - coolant inlet temperature, ^oC	295
11. - coolant outlet temperature, ^oC	330
12. - active core flow rate, m^3/hr	42.45×10^3
13. - pressure vessel i.d., m	4.131
C. TURBINE	
1. - steam pressure, MPa	6.9
2. - steam temperature, ^oC	284.8
3. - steam flow, kg/sec	1.003×10^3
4. - feedwater temperature (s.g. inlet), ^oC	230
5. - exhaust pressure, kPa	7 (max)



RPV - reactor pressure vessel (o.d.4535 mm)
SG - steam generator
P - pressurizer
MCP - main coolant pump
HL - hot leg (i.d. 850 mm)
CL - cold leg (i.d. 650 mm)

FIGURE 2 : Reactor coolant system configuration

All equipment and structures of the nuclear island are designed to withstand the effects of the maximum design basis earthquake (in this case 7^o MSK-64), and to maintain their operating functions. When a greater seismic resistance is required the spring-supported concept is assumed to apply in the design of the containment building ground slab.

4.3 Turbine system

The turbine SKODA is a 3000 rpm machine and it consists of a double-flow, high pressure cylinder and 2 double-flow, low pressure cylinders which exhaust to condensers. The cylinders are arranged in a tandem configuration. In the design of the turbine base slab a spring-supported concept is applied.

The three-phase generator SKODA has a direct water cooling of the stator winding and as cooling agent of rotor and stator-core stampings is supposed hydrogen.

There are 2 horizontal moisture separator reheaters located between HP cylinder exhaust and LP cylinder inlet on both sides of turboset for extracting moisture from the steam and for steam reheating

The plant uses a six-stage feedwater heater cycle with 4 horizontal LP feedwater heaters, deaerator and 2 in parallel connected vertical HP feedwater heaters. Two main feedwater pumps are driven by small steam turbines.

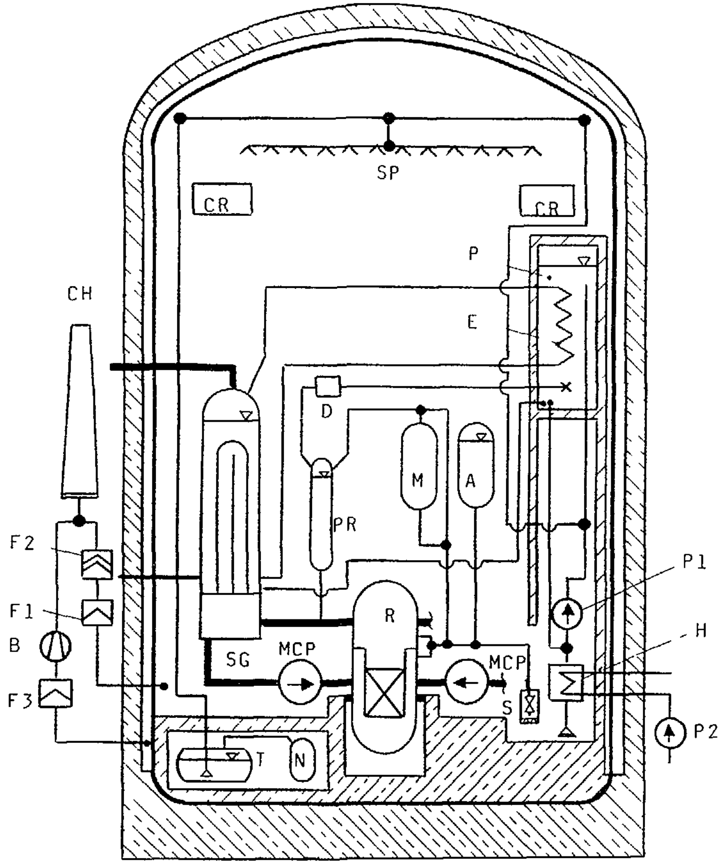
The heat extraction station produces hot water for the district heat supply system and its capability can range up to 600 MW of delivered heat. The temperature of the heated water is rised in a three-stage heating system from 60 °C up to 160 °C. The heating steam is transferred from turbine extraction lines to 3 horizontal heat exchangers.

Each NPP unit is eqipped with one wet cooling tower (height about 150 m, bottom diameter about 120 m).

5. SAFETY AND THE ENGINEERED SAFETY FEATURES (E S F)

The approach to operating safety of the NPP and to determination of the ESF concept is, in general, based on the postulated requirement that all critical safety functions must be maintained even in the case of total blackout (total loss of all external and internal a.c. power sources) for sufficient long time (grace period) in which the full restoration of power supply sources, namely those of active containment heat removal and spray systems, has to be accomplished. In the preliminary design study 8 hours were set down as the above mentioned time interval. This value is about two times greater than the duration of electrical network disintegration historically experienced in the CSFR It is however possible to increase this grace period approximately two times, by corresponding sizing of ESF systems, without change of their concept. Within this period it is supposed to maintain the critical safety functions only by means of passive safety systems without need of operating staff actions or active safety system operation.

The above described safety approach enables to use a tight conventional containment design without need to apply passive-type heat removal from containment to surrounding environment by natural circulation of air. The effectiveness



R - reactor
MCP - main coolant pump
P - water pool
M - make-up tank
H - heat exchanger
N - nitrogen tank
CR - hydrogen removal system
SG - steam generator
D - depressurization system
E - heat exchanger
A - hydroaccumulator
P1 - pump
SP - spray nozzles
F1,F2,F3 - filters
PR - pressurizer
T - water storage tank
CH - chimney
S - valve
P2 - pump
B - blower

FIGURE 3 : Engineered safety feature systems of NPP

TABLE 2. Parameters of ESF equipment (supplement to Fig.3)

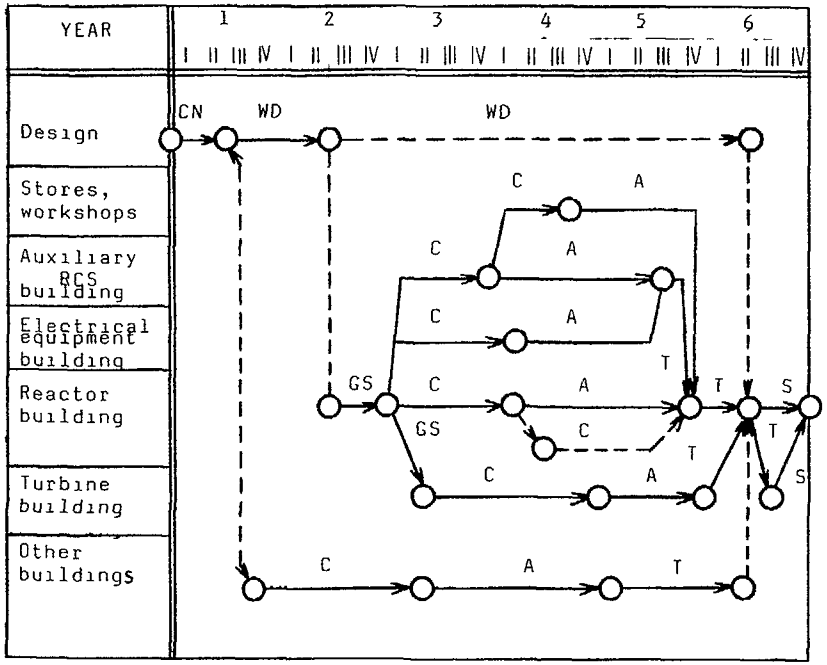
	Equipment	Number per unit	Volume per equipment, m^3
M	Make-up tank	3 (3 x 100%)	30
A	Hydroaccumulator	3 (3 x 100%)	60 (5 MPa)
P	Water pool	2 (2 x 50%)	750
E	Heat exchanger	2 (2 x 50%)	-
H	Heat exchanger	3 (3 x 100%)	-
P1	Pump	3 (3 x 100%)	-
P2	Pump	3 (3 x 100%)	-
T	Storage water tank	3 (3 x 50%)	60
N	Nitrogen tank	3 (3 x 50%)	-

and reliability of passive heat removal, in fact, strongly depends on changeable meteorological conditions and under some unfavourable circumstances not enough heat could be removed.

The schematic representation of basic ESF systems is shown in Figure 3 and some characteristics of their equipment are given in Table 2. The most important passive ESF are: reactor heat removal system, RCS depressurization system, passive containment spray system, emergency feedwater system (it acts after depressurization of steam generator secondary side) and isolation systems. The containment heat removal system and the active containment spray system belong to the active part of ESF. The main purpose of passive and active containment spray systems is to prevent the containment internal pressure from exceeding the value of about 0.3 MPa. As the main heat sink, two water pools are supposed to accumulate the residual heat generated in reactor. Partial evaporation of their water is, in principle, acceptable.

6. CONSTRUCTION SCHEDULE AND PROJECT IMPLEMENTATION

In the framework of the preliminary design study a time schedule for final design, erection, assembly, testing and start-up was evaluated as well. From the analysis follows that a simplification of plant arrangement, reduction in equipment and bulk quantities, application of advanced construction procedures (monolithic concrete structures, modular assembly, etc.) and time optimization of construction activities enables to attain relatively short construction schedule of about 55-60 months (from the start of concreting up to commercial operation). In addition to this, the peak number of construction personnel on site is supposed to be less than 4200 people. As an illustration the simplified construction schedule is shown in Figure 4.



CN - contract negotiations
GS - ground slab concreting
C - construction (structures)
S - start-up
WD - working (final) design and safety documentation
A - equipment assembly
T - testing

FIGURE 4 : Construction schedule of NPP

As for the project implementation, the turnkey contract or split package contract approach is assumed to apply. From the results of evaluation follows that the delivery share of a domestic industry could reach the value of about 80 %. The remaining part must be covered by use of licence transfer (e.g. vertical steam generators) or by delivery of some equipment from abroad (e.g. some instrumentation and control subsystems).

TABLE 3. Economic evaluation of power plants

NPP	Net electrical output/unit, MW	Supposed load factor	Normalized specific capital investment cost	Normalized specific electricity generating cost
NPP Temelín CSFR	1000	0.85	1(=100%)	1(=100%)
ABB 80+ NPP	1345	0.85	1.14	1.06
Advanced fossil-fired power plant	300	0.68	0.98	1.78
Combined gas-steam cycle power plant	340	0.42	0.51	1.82
NPP with mid-size APWR units	600	0.85	0.86	0.88

7. ECONOMIC EVALUATION

A comparision of main economic parameters of the mid-size NPP with those of some advanced nuclear and fossil-fired power plants has been made with the aim of economic evaluation. The results are given in Table 3 in which as a basis for comparision the NPP Temelin was selected. The comparision by means of normalized values was preferred, because of absolute values are changing at present as a result of the transition to free market economy (cost rise, inflation pressure, currency exchange rate changes, etc.). From Table 3 follows that economic parameters of the NPP with APWR mid-size units are sufficiently favourable.

In the economic analysis, the "modified capital investment cost" methodology was applied with the use of "utility functions". This method enables to compare power plants of various types built in various periods of time. Moreover the approach of the investment economic evaluation which is common in west european countries was taken into account as well.

8. OTHER ACTIVITIES OF S K O D A IN THE FIELD OF A L W R

The project of a nuclear district heating plant with 200 MW reactor is now in final stage in SKODA. Negotiations are, at present, proceeding with municipal authorities of Plzen city concerning the construction, and consultations are in progress with authorities of some other regions of the CSFR.

9. CONCLUSIONS

The results of the preliminary design study have shown that it is reasonable to proceed in the project activities with the aim to offer such a NPP in near future as a new power source of energy system in the CSFR. The scope of SKODA activities in this field will, however, depend on the way of supposed association of SKODA with some strong foreign company.

REFERENCES

/1/ - "Preliminary design study of the NPP with APWR mid-size units", Volume 1 - "Initial conditions and main objectives", SKODA, 1990

/2/ - "Preliminary design study of the NPP with APWR mid-size units", Volumes 2-11 - "Description of the NPP", SKODA, 1990

/3/ - "Preliminary design study of the NPP with APWR mid-size units", Volumes 12-18 - "Evaluation of the NPP", SKODA, 1990

CURRENT ACTIVITIES ON ADVANCED LIGHT WATER REACTOR DESIGN AND TECHNOLOGY AT THE JAPAN ATOMIC ENERGY RESEARCH INSTITUTE

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Abstract

An overview of the research activities on advanced light water reactors (LWRs) in the Japan Energy Research Institute (JAERI) is presented. The advanced LWR concepts proposed are divided into three research areas which feature, respectively, simplified and passive safety systems, high performance reactor core and fuel designs and the transmutation/incineration of transuranium elements. Major research and development activities and test facilities for exploring the advanced LWR technologies are described.

1. INTRODUCTION

The Japanese basic strategy for the reactor development is "from LWRs to FBRs". However, it is presumed that LWRs will continue to be the mainstream of nuclear power generation for a considerably long time before FBRs become practical. The evolutionary improvement of the conventional LWR system can be carried out by the private sector. In view of the forecast prolongation of the LWR-mainstream era, JAERI (Japan Atomic Energy Research Institute) has been making efforts in the pursuit of new possibilities for LWR technology innovation. In this regard JAERI is carrying out research on advanced LWR concepts in order to further improve safety, reliability/maintainability, economy, and efficient utilization of uranium resources with a wide and long-range view.

There are three different research areas for advanced LWR concepts development in JAERI. In the first area, a small and medium size reactor system having advanced safety features is pursued. From an innovative approach the SPWR (system-integrated pressurized water reactor) concept with simplified and passive safety systems has been proposed. In the second area, research works on high performance reactor core and fuel are promoted to aim at a high conversion ratio up to around unity and a high burnup of more than 100GWd/t. The purpose of the third area is to explore a reactor core and fuel cycle system concept in view of the transmutation/incineration of transuranium elements (TRUs excluding plutonium) and the TRU-recycle. The core characteristics of the reactor concepts proposed for the latter two research areas will be described.

From the viewpoint of implementing basic research and underlying technology development for the advanced LWR concepts a study of test facilities required is now in progress in JAERI.

Table 1 Major design parameters of SPWR

Reactor :	
Thermal power output(MW)	1,100
Electric power output(MW)	350
Coolant inlet/outlet temp.(°C)	290/320
Coolant flow rate(ton/h)	23,000
Total pressure drop(MPa)	0.14
Core outlet pressure(MPa)	13
Core/Fuel Assembly :	
Equivalent core diameter(m)	2.99
height(m)	2.4
^{235}U enrichment(%)	4.5
Core average power density(kw/l)	65
Number of fuel assemblies	121
Average fuel burnup(GWd/t)	45
Fuel assembly pitch(triangle)(mm)	259
Number of fuel rods	325
Fuel rod diameter(mm)	9.5
Lattice pitch(triangle)(mm)	14.0
Average linear heat rating(W/cm)	141
Steam Generator :	
(once-through helical coil type)	
Steam temperature(°C)	285
pressure(MPa)	5.3
Feedwater temperature(°C)	210
pressure(MPa)	5.9
Steam flow rate(ton/h)	2,000
Tube material	Incolloy 800
Tube inner/outer diameter(mm)	15/19
Heat transfer area(inner total)(m^2)	8,840

2. SYSTEM-INTEGRATED PRESSURIZED WATER REACTOR(SPWR)[1 ~ 3]

A design study of a passive safe integral reactor SPWR has been carried out in the direction of developing a new concept of small and medium size power reactors. The main design goals of SPWR are aiming at enhanced passive safety features, and easy operation/maintainability and improved economy through the simplification of plant systems. The major design parameters are listed in Table 1.

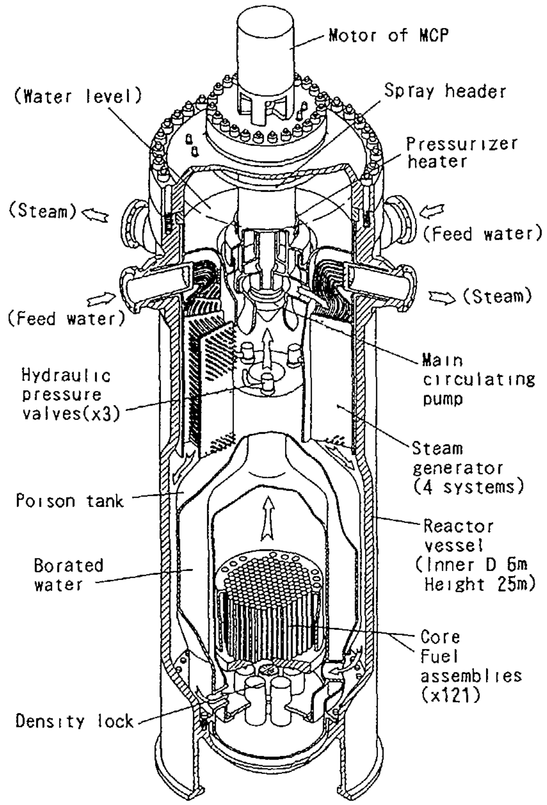


Fig. 1 Reactor vessel and internals of SPWR(350 MWe)

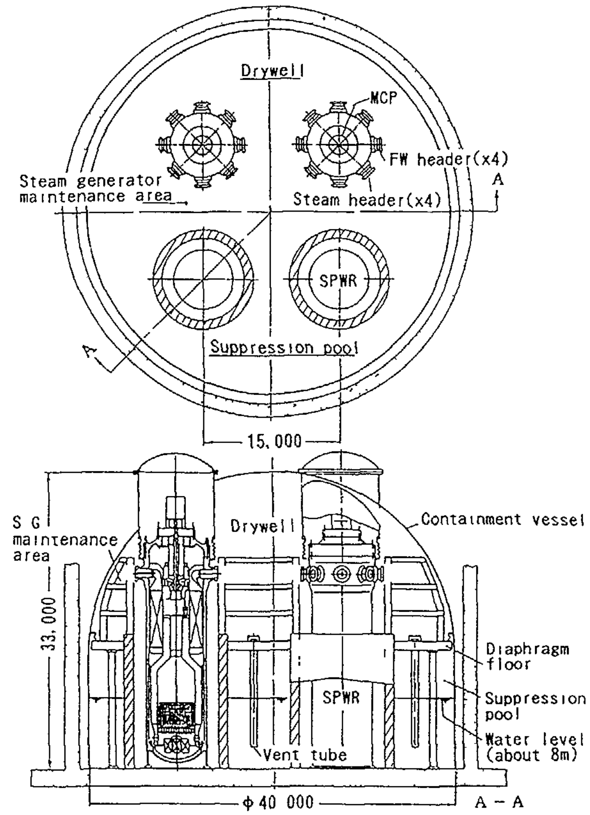


Fig. 2 Quadruplet unit installation of SPWR(1400 MWe)

(1) Basic Design Features

The SPWR structure concept is shown in Fig. 1. The reactor core, the steam generator, the main circulation pump, the pressurizer and the borated water tank are all contained within a single pressure vessel Control rods are not installed, which provides the advantage of a great simplicity in the reactor structure design. The use of control rod drive system for safe shutdown of the reactor is excluded on the basis of the design philosophy placing reliance on natural laws of thermohydraulics and gravity in accident situations without any dependence on active mechanical devices. For emergency shutdown of the reactor highly borated water is influxed into the core through passive mechanisms The normal reactor operation including startup, scheduled shutdown and load following is controlled by means of the boron concentration in the primary coolant and negative reactivity coefficients of temperature and coolant density

The reactor core is immersed in water of a relatively low boron concentration and is surrounded by a tank of highly borated (neutron poison) water. The poison tank is connected to the bottom of the reactor core through a lower core/tank interface zone (honeycomb structures) similar to that used in the PIUS concept. At the top of the tank, the passive hydraulic pressure valves are installed as the upper interface between the primary coolant and the tank water. The SPWR containment design adopts the pressure suppression containment vessel for utilizing passive safety feedwater and suppression pool (heat sink) systems

The integral reactor housing the steam generator within a reactor vessel has a smaller power output due to the current fabrication capacity of the reactor vessel, compared with conventional large LWRs. However, the compact and simplified SPWR design is easy of modularization, which enhances the economy. Figure 2 exemplifies a plant layout concept for a quadruplet unit SPWR (1400 MWe) installation

(2) Safety Features

The safety design philosophy of the SPWR concept is based on simplified systems and passive performance systems. The reactor core is designed to provide the intrinsic safety characteristics similar to those of existing PWRs which possess negative reactivity coefficients of temperature and coolant density. In the latest SPWR design a relatively low power density core (65 kw/l) is employed, which increases safety margins. The enhanced safety features are as follows.

(a) There are no possibilities of rapid reactivity insertion accidents because of the elimination of control rods.

(b) Because the integral design does not include any large-bore piping, there cannot be large-break pipe failures (LOCA). Besides all of the primary piping are positioned at an elevation above the reactor core. Hence, these can keep the core fully covered for a long time period (several hours) even without cooling system.

(c) The emergency shutdown system is made with use of hydraulic pressure valves and highly borated water tank.

(d) Decay heat in accidents is removed by passive pressure-balanced feedwater (supply by natural circulation) systems with passive hydraulic pressure valves and the suppression pool housed within the containment vessel.

(e) For long-time cooling of decay heat the suppression pool water is designed to be passively cooled with use of a heat pipe system which transports the decay heat to the ultimate heat sink (atmosphere or seawater).

(3) Functions of Passive Safety Systems

Transients in accident situations are very slow due to the elimination of rapid reactivity insertion accidents and large-bore primary piping failures, and passive safety systems containing large water inventory provide a long grace period. The design features of the passive safety systems are as follows.

1) Emergency reactor shutdown

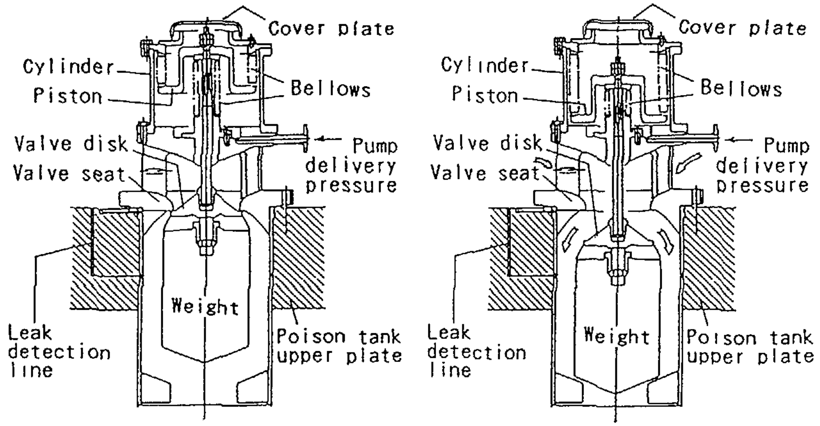
Three hydraulic pressure valves are installed at the top of the neutron poison tank. The valve is controlled by a hydraulic piston normally closed by a delivery pressure from the main circulation pump. In the design, the valve shown in Fig. 3 is closed at the flow rate of higher than 40% of rated flow of the pump. If the pump delivery pressure falls below the minimum required to overcome the weight, the weight pull opens the valve, allowing the reactor coolant to flow into the top of the tank and the highly borated water in the tank into the bottom of the reactor core. As a consequence the reactor is shutdown within several seconds after the pump trip and is cooled by natural circulation.

This passive system can be used as an active reactor shutdown system by reducing/stopping the pump rotation or by closing a valve installed between the pump and the hydraulic pressure valve, in order to open the hydraulic pressure valve.

2) Decay heat removal systems

During normal reactor shutdown the residual heat in the core is removed through the secondary cooling system via the steam generator, and during reactor maintenance an auxiliary residual heat removal system is provided.

In accident situations the passive emergency reactor shutdown system is capable of maintaining the core covered for long time cooling. To further enhance the safety



Normal operation – valve closed **Emergency operation – valve open**

Fig. 3 Hydraulic pressure valve structure

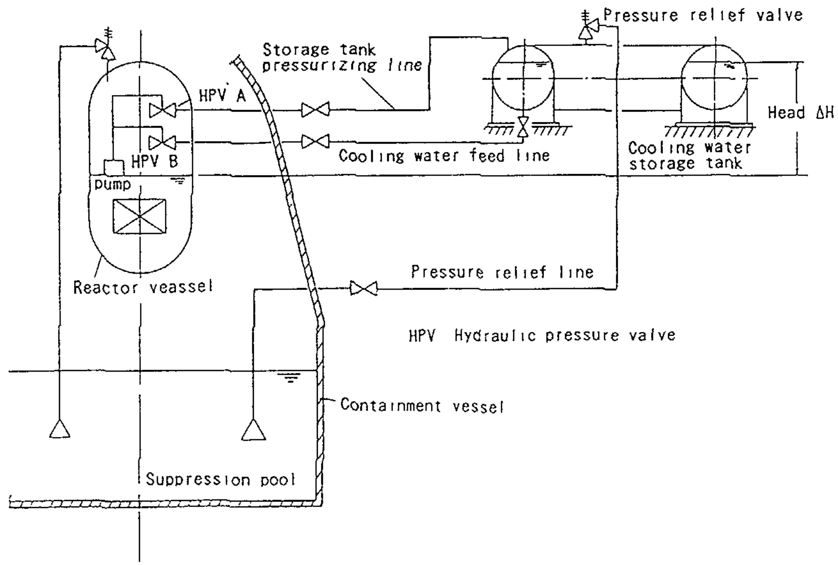


Fig. 4 Passive core cooling systems

features for preventing an uncovered core the following passive safety systems are installed for long-time decay heat removal.

(a) Pressure-balanced feedwater system

Figure 4 illustrates the pressure-balanced feedwater system which consists of borated feedwater tanks, a suppression pool, two types of hydraulic pressure valves and safety relief valves. The feedwater tanks are outside of the containment vessel and elevated above the reactor. The hydraulic pressure valves are normally held closed by the delivery pressure of exclusive-use pumps and installed at pipe lines connecting the reactor core and the feedwater tanks.

In case the reactor water level falls below a designated level, the hydraulic pressure valve A shown in Fig. 4 opens, which causes pressure reduction in the reactor vessel and pressure buildup in the tanks. If the pressure in the tanks exceeds a safety design level during an accident, the relief valves discharge the steam to the suppression pool inside the containment vessel and its heat is released to the pool. When the pressures in the reactor vessel and tanks are balanced, the valve B opens and then the borated feedwater enters the reactor core by natural circulation. This circulation water flow prevents the core from approaching dryout conditions. The hydraulic pressure valves are normally held closed by the delivery pressure provided from small pumps placed at the designated level in the reactor vessel. For safety redundancy in accident situations, two pumps are in parallel installed and two valves of type A and B each are also in parallel provided. The feedwater capacity of the tanks is designed to provide a three-days grace period.

(b) Heat removal system of the suppression pool

The decay heat released into the suppression pool is passively removed to the ultimate heat sink (atmosphere or seawater) by a heat pipe system. A few design options of heat pipe system are under investigation.

3. HIGH CONVERSION LIGHT WATER REACTORS(HCLWRs)

For the HCLWR concept development the design efforts aim at both high conversion and high burnup. On the other hand, the use of core with a tighter pitch lattice and MOX(mixed uranium and plutonium oxide) fuel with highly enriched plutonium to attain these goals leads to serious problems with respect to positive void reactivity and large plutonium inventory. In reactor physics aspect the design requirement to secure negative void reactivity effects degrades the conversion and burnup performances. Moreover, thermohydraulic feasibility should be demonstrated concerning safe heat removal for the tight lattice fuel assembly during normal operation and accidents. Accordingly, an optimum design concept development is required upon consideration of these aspects.

(1) Design Features[4, 5]

A new HCLWR concept was proposed at JAERI, to achieve both high conversion and high burnup with a negative void coefficient. The reactor core concept is based on an axially heterogeneous configuration of flat pancake core and blanket. The void coefficient problem can be overcome by increasing the neutron leakage in the axial direction from the core. The leakage neutrons are utilized in the axial blankets to enhance the conversion ratio.

Because of the short core height, an output power of single flat core may be restricted by the current fabrication capacity of the reactor vessel from the viewpoint of its diameter size. One of the most direct way to apply the flat core concept is its use in small and medium sized reactors (≤ 400 MWe). To provide an output power as large as that of current LWRs a double-flat-core HCLWR concept has been employed. In this concept two such flat cores with axial blankets and an inner blanket are stacked to construct a reactor as shown in Fig. 5.

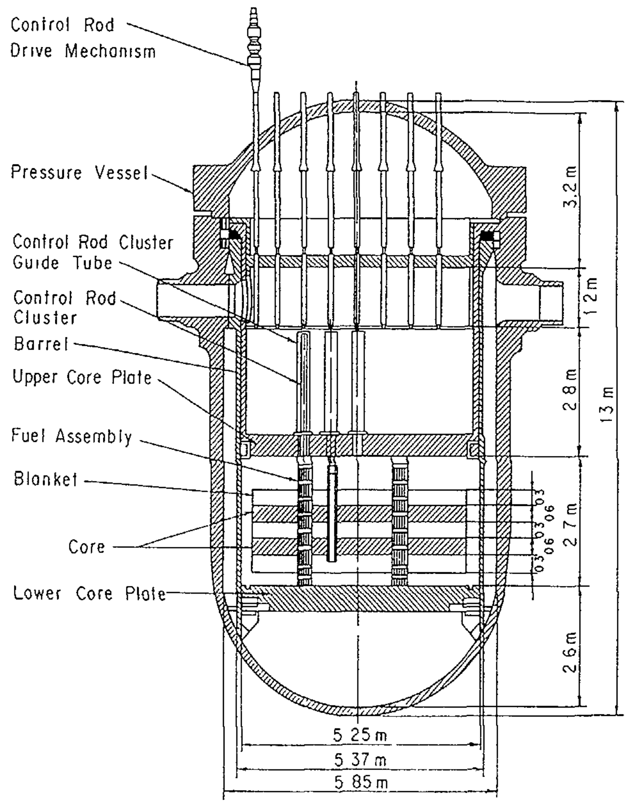


Fig. 5 Double-flat-core type HCPWR(HCPWR-JFD1)

The axially heterogeneous reactor concept has been investigated for both pressurized water reactor (HCPWR) and boiling water reactor (HCBWR) In the HCPWR design study, one-dimensional survey calculations of reactor physics characteristics were performed changing fundamental design parameters such as moderator-to-fuel volume ratio, core height, inner blanket thickness and so on Based on the survey results on reactor physics characteristics as well as thermohydraulic and safety analyses, reference design parameters of the double-flat-core HCPWR-JDF1 with a semitight pitch lattice were determined, as shown in Table 2 Fulfilling a negative void coefficient, an optimization among allowable linear heat rating, obtainable burnup and conversion ratio provides the dimensional parameter of core height, blanket thickness and moderator-to-fuel volume ratio For the reference HCPWR-JDF1 an effective burnup of 56 GWd/t and an average conversion ratio of 0 83 were obtained from three-dimensional burnup calculations

Table 2 Reference parameters of HCPWR-JDF1

Reactor Core	
Thermal power output(MW)	2432
Electric power output(MW)	810
Equivalent core diameter(m)	4 37
Equivalent blanket diameter(m)	4 80
Unit core height(m)	0 6
Top and bottom blankets thickness(m)	0 3
Inner blanket thickness(m)	0 3
Blanket material	Depleted UO_2
Number of fuel assemblies(FA)	313
Number of control rod clusters	85
Average linear heat rating at BOC/EOC(W/cm)	114/109
Radial power peaking factor	1 48
Axial power peaking factor	1 22
Average conversion ratio	0 83
Equilibrium Puf enrichment(%)	10 0
Initial Puf loading(ton)	4 9
Fuel Assemblies	
Number of fuel rods	372
Fuel assembly pitch(mm)	235 4
Fuel rod pitch(mm)	11 7
Fuel rod outer diameter(mm)	9 5
Fuel cladding material	Zircaloy
Cladding thickness(mm)	0 57
Spacer type	grid
Number of control rods per FA	24
Moderator-to-fuel volume ratio	1 06

The HCBWR design effort is aiming at a higher conversion ratio of around unity by taking full advantage of BWR core features where a low effective moderator-to-fuel volume ratio can be attained due to the moderator voidage resulting from boiling The concept development work has been carried out jointly with Energy Research Laboratory, Hitachi, Ltd The main objectives of the neutronic core design are to reach a negative void coefficient and to reduce a large plutonium inventory in the previous plutonium generating BWR (PGBR) design concept[6] To achieve a conversion ratio close to unity, the effective moderator-to-fuel ratio must be less than 0 3, which can be realized by the closely packed hexagonal lattice with the rod-to-rod clearance of 1 3 ~ 1 5 mm and the core averaged void fraction more than 50 % A preliminary design study based on one-dimensional parametric survey calculations indicates that the void coefficient is reduced by increasing the inner blanket thickness A further detailed investigation is in progress for developing the design concept and the methodology including three-dimensional neutronic and thermohydraulic analyses

(2) Safety Features[7]

Thermohydraulic feasibility studies have been performed for the double-flat-core HCPWR design concept Evaluation methods and models for the design concept were developed based on experiments in critical heat flux, pressure drop and reflood cooling Under the steady-state operational conditions in the present design, the estimated minimum DNBR (departure from nucleate boiling ratio) assures an enough safety margin, and the core pressure drop is estimated to be much smaller than that in current PWRs

Safety analyses for large break loss-of-coolant accident (LOCA), locked rotor accident of one out of three primary coolant pumps, and anticipated transient without scram (ATWS) triggered by station blackout were performed with a best-estimate code J-TRAC developed for primary system transient calculations The results show that the HCPWR design meets the current safety criteria for licensing applied to LWRs The thermohydraulic feasibility of the double-flat core HCPWR is mainly attributed to the design features such as the large water inventory in the reactor vessel, short core length, low axial peaking factor, and negative void and coolant temperature reactivity coefficients

4 VERY HIGH BURNUP PRESSURIZED WATER REACTOR[8, 9]

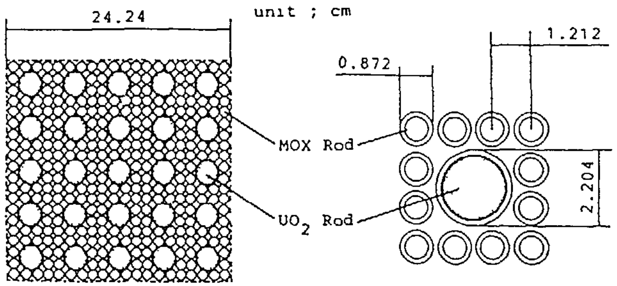
A very high burnup pressurized water reactor (VHBPWR) core concept aiming at a high burnup of more than 100 GWd/t was proposed to improve fuel utilization and to reduce the fuel cycle cost of MOX fuel In the VHBPWR concept fuel assemblies consist of highly enriched plutonium MOX fuel rods with a lattice pitch close to that of current PWRs and fertile natural/depleted uranium oxide rods used for neutron spectral shift

Larger sized fertile rods in diameter than the MOX fuel rods are used to provide a spectral shift effect required for a burnup reactivity increment They are inserted into the core at the beginning of the burn cycle As the fuels burn up, fertile rods are withdrawn to gain a further burnup reactivity by varying the moderator-to-fuel volume ratio from the initial value of 1 5 to 3 0 (Table 3)

Table 3 Parameters of the very high burnup fuel assembly

	MOX rod	UO_2 rod
Fuel rod diameter(mm)	8.72	22.04
Pellet diameter(mm)	7.60	19.22
Cladding thickness(mm)	0.48	1.32
Gap thickness(mm)	0.16	0.18
Number of fuel rods	300	25
Fuel rod pitch(mm)	12 12	48.48
Pu (Puf) enrichment(%)	15.0(10.7)	(nat. U)
Pellet density(% T.D.)	85.0	95.0
Initial fuel loading(kg)	449.1	365.0

Moderator-to-fuel volume ratio :
1.5 (UO_2 rod inserted)
3.0 (UO_2 rod withdrawn)



A concept of the fuel assembly

Safety related subjects in the neutronic core design for the MOX fuel with high plutonium content are to lower an initial excess reactivity of loaded MOX fuels and to secure a negative void coefficient throughout the whole burnup period. The fertile rods play a role of control rods to suppress the initial reactivity as well as to provide a negative void coefficient even at an early burnup stage, because of their resonance absorption for epithermal neutrons. Results of a parameter study show that a burnup of about 120 GWd/t can be obtained for the MOX fuel of 15 % plutonium with a burnable poison of 0.4 % Gd_2O_3 added for decreasing the initial reactivity.

In the VHBPWR core concept fairly large spatial changes in thermal neutron flux and power distributions in the fuel assembly arise by withdrawing the fertile rods. As for this problem a further detailed investigation is necessary

5. TRU TRANSMUTATION LIGHT WATER REACTOR[10]

One of the challenging problems for the future nuclear technology is to confine in a fuel cycle system high level radwaste including TRU from the spent fuel processing and to transmute long-lived radioactive nuclides to shorter or stable ones by the exposure of neutron flux. In Japan a research program called OMEGA Project is being promoted by joint efforts of JAERI, PNC and others. JAERI is going to develop nuclide partitioning technology, TRU handling technology as well as TRU transmutation technology. The transmutation is to be performed either by a fast reactor or a high-powered proton accelerator of GeV energy level.

Apart from the OMEGA a new approach for the development of a light water reactor with TRU transmutation/incineration features has been proposed at JAERI. Reactor core characteristics have been investigated in terms of coolant void reactivity coefficient, conversion ratio, burnup reactivity swing, recycling effect and transmutation efficiency.

In the preliminary study, TRU nuclides are mixed homogeneously to MOX or UO_2 fuel. The TRU-mixed fuel serves to suppress remarkably initial excess reactivity and burnup reactivity swing, and provides a high conversion ratio. Long-lived TRUs such as ^{237}Np, Am and Cm behave like a burnable poison at the beginning of a burn cycle, and then are gradually transmuted to fissile nuclides and incinerated with burnup. In this sense the TRUs are considered to be a useful fuel with burnup reactivity controllability. It should be noted, however, that the TRU content in fuel is restricted to secure a negative void coefficient.

In order to achieve a high TRU content in fuel and a high burnup in terms of efficient TRU transmutation with a negative void coefficient, a well-moderated core of the moderator-to-fuel ratio of 3.0 is adopted for the reference design concept. The amount of TRU generated in a current UO_2-fuel PWR(3410 MWt) with a burnup of 33 GWd/t is estimated to be 23 kg per 300 days, and the ^{237}Np amounts to 60 % of the TRU. Transmutation efficiency was investigated for several types of PWR containing TRU or ^{237}Np of 3%, respectively, in the fuels. A MOX-fuel PWR can transmute the TRU from 6 units of the current LWR or the ^{237}Np from 10 units, while a UO_2-fuel PWR can do the TRU from 13 units or ^{237}Np from 20 units.

Existing fuel reprocessing systems are designed to extract highly purified uranium and plutonium. The TRU transmutation reactor concept, however, would allow a low decontamination reprocessing process. A simplified and economical low-decontamination fuel cycle scenario is under investigation from the viewpoint of fuel reprocessing, refabrication and transportation.

6. RESEARCH AND DEVELOPMENT ACTIVITIES

A number of research and development requirements have been listed up for the respective advanced reactor concepts. Among them the following are current activities on research and development which have been implemented and are planned in JAERI with respect to advanced reactor technologies.

(1) Activities on the SPWR

Major safety components in the SPWR design are the hydraulic pressure valves and the lower interface. Design studies for these components and a fundamental test for the half-scale hydraulic pressure valve fabricated based on the design concept have been carried out. The results of the test show satisfactory characteristics of the valve in terms of piston motion stability, hydraulics, allowable water leakage through the valve seat and so on.

In the SPWR design study analyses on reactor dynamics for various conditions of startup, shutdown, load following and accidents have been in detail carried out. To confirm experimentally these dynamic characteristics at thermohydraulic mockup test facility is under investigation, which simulates the passive safety system features and the reactor core by using an electric heater of about 10 MW.

As for the SPWR safety, basic experiments on the DNBR at an emergency reactor shutdown are planned and an application of probabilistic safety assessment (PSA) is in progress.

(2) Activities on HCLWRs

The JAERI has conducted a research and development program for HCLWRs which consists of two phases ; Phase I running from 1985 to 1990, and Phase II from 1991 to 1994. The major technical fields in the program are reactor physics and thermo-hydraulics.

The main purpose of Phase I is to evaluate the technical feasibility of the HCPWR concept and to identify the key problems to be solved, while deepening the understanding of the relevant basic phenomena. In theoretical reactor physics studies on neutronic characteristics, nuclear data and calculation methods were extensively made to improve the data and methods, and to design the reactor core with a negative void coefficient. In order to verify the data and methods experiments were conducted on two types of critical assemblies, FCA (Fast Critical Assembly) and TCA (Tank-type critical assembly). As for thermohydraulics small scale experiments have been made on hydraulic characteristics such as pressure drop and thermal diffusion, critical heat flux, reflood cooling and so on, for analysis models assessment and understanding the basic phenomena. Computer codes for safety analysis (J-TRAC) and subchannel analysis (COBRA-IV-I) have been also developed.

The purpose of Phase II is to choose the optimal design parameters of the HCPWR core and to evaluate the technical feasibility of the HCBWR concept with a conversion ratio of about unity and a negative void coefficient. Large scale experiments on neutronic and thermohydraulic characteristics are planned. Hydraulic test facilities (multi-dimensional hydraulic loop and multi-purpose hybrid loop) planned in Phase II are now at a preparatory stage to install.

(3) A Study of Integrated Test Facilities

From a long-term research and development standpoint in the prolonged LWR-mainstream era, we are now performing a preliminary study of integrated test facilities which aim at exploring the innovation of LWR technologies to be deployed in the 21st century. A plan of the test facilities is investigated to implement the research and development of advanced LWR technologies in an effective and comprehensive way. The emphasis on the research and development using the facilities is currently put on testing the characteristics of the passive safety system and the HCLWR core, and developing the very high burnup fuel, the TRU-mixed fuel and the TRU-recycling technology including the fuel processing.

The facilities complex is classified into three groups ;

(a) a test reactor for fuel irradiation, passive safety system performance, and so on
(b) fuel engineering facilities for test-fuel fabrication, post-irradiation examination, fuel reprocessing process competent to advanced fuel concepts, and so on
(c) reactor engineering facilities for criticality experiment, thermohydraulics, and so on

Among these facilities the test reactor plays a central role of the advanced LWR technology development. Key design features and parameters are as follows ; PWR core with semitight pitch lattice (moderator-to-fuel ratio of 0.7 for UO_2 and 1.4 for MOX) as a standard core, and thermal output power of 300 MW with an average power density of 100 kW/*l*. Fuel assemblies with a semitight pitch lattice are adopted to provide a relatively hard neutron spectrum. A design study of BWR-type test reactor is also underway. Installation of a large-scale irradiation loop is under consideration. The test of the passive safety system performance is now planned to be implemented by installing the necessary components (valves and poison tank) and loops outside the reactor vessel to simulate the hydraulic characteristics of the SPWR concept.

REFERENCES

[1] SAKO, K., "Conceptual Design of SPWR (System-integrated Pressurized Water Reactor)", ANS Topical Meeting on Safety of Next Generation Power Reactors, Seattle, USA, May, 1988.

[2] SAKO, K., et al., "Feasibility Study of SPWR as the Next Generation Power Plant", Proc. for the Seminar on Small and Medium-sized Reactors (Post Conf. SMiRT), III.5.21, San Diego, USA, Aug. 1989.

[3] SAKO, K., et al., "Conceptual Design of SPWR, a PWR with Enhanced Passive Safety", JAERI-M 89-208(in Japanese), 1989.

[4] ISHIGURO, Y., et al., "The Concept of Axially Heterogeneous High Conversion Light Water Reactor", IAEA-TCM on Technical and Economic Aspects of High Converters, Nuremberg, FRG, March 1990.

[5] OKUMURA, K., et al., "Conceptual Design Study of High Conversion Light Water Reactor", JAERI-M 90-096(in Japanese), 1990.

[6] TAKEDA, R., et al., "Plutonium Generation Boiling Water Reactor Concept", Proc. of Int. Conf. on the Physics of Reactors(Marseille, France, March 1990), Vol. 4, P.III-64.

[7] IWAMURA, T., et al., "Thermal Hydraulic Feasibility Study of a Double-Flat-Core Type High Conversion Light Water Reactor", Thermal Hydraulics of Advanced Nuclear Reactors, HTD-Vol. 150, ASME, 1990.

[8] ICHIKAWA, H , et al , "A Very High Burnup Pressurized Water Reactor with Highly Enriched Plutonium Fuel Assemblies Using a Spectral Shift Concept", Proc of Int Reactor Physics Conf , Vol IV, pp 21, Jackson Hole, USA, Sep 1988
[9] NAITO, Y , et al , "Design Study on a Very Long Life Light Water Power Reactor Core", JAERI-M 91-028 (in Japanese), 1991
[10] TAKANO, H , et al , "Higher Actinide Confinement/Transmutation Fuel Cycles in Fission Reactors", Ref [6], Vol 4, P III-145, and private communication

SBWR TECHNOLOGY AND DEVELOPMENT

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Abstract

The simplified Boiling Water Reactor (SBWR) is a 600 MWe total plant design that is based on proven and state of-the art Light Water Reactor (LWR) technology It includes technology extensions that are based on extensive testing This paper provides the technology basis and a description of specific testing done to support the SBWR design The primary focus of the testing has been on safety systems that include innovations beyond the operating plant technology base The overall technology program provides high confidence in the overall plant reliability and safety

1 INTRODUCTION

The development of the next generation of GE nuclear power plants is a cooperative effort involving plant designers, constructors and owner-operators The joint effort results in a plant design that has the overall characteristics of being significantly simpler to build, operate and maintain To achieve these characteristics, the plant design must focus on simplification of systems, processes, controls, and components as well as simplification of the structural configuration and arrangement The design must utilize and incorporate advances in existing proven technology that have been developed over the past thirty years of commercial nuclear plant operation The development of the advanced and simplified next generation plant must consider and involve the entire power plant and not only the nuclear steam supply or safety systems Considering the entire plant in the design and development process ensures that the final plant design will be highly reliable, simpler to construct, and easier to operate and maintain The resulting enhancement of plant safety systems ensures a greater degree of acceptance of the design by regulatory bodies and the general public

As part of this effort to develop the next generation of the Boiling Water Reactor, General Electric Company started the development of the Advanced Boiling Water Reactor (ABWR), a 1300 MWe reactor design that utilizes all of the advantages of the economy of scale but represents the utilization of all the advances in technology Table I lists the major technology enhancements in the design of the ABWR and shows that major improvements have been made in the design of the plant to make it simpler to operate and maintain in addition to ensuring a high reliability of the systems Table I also lists the features that are common to the smaller advanced BWR, called the SBWR The remainder of this paper describes the features and the technology basis for the features that are unique to SBWR Additional details on the design of the ABWR are given in Reference [1] and on the design of the SBWR in Reference [2]

Table I Comparison of Advanced BWR Features to Operating BWR's

Feature	GE Operating BWRs	ABWR & SBWR
Materials	IGSCC sensitive	Fully qualified, tested
Water treatment	Filter demineralizers and/or deep bed	Hollow fiber plus deep bed
Operator Procedures	Complex, written	Optimized displayed
Emergency Procedures	Event based (retrofit sympton based)	Symptom based
Environmental Qualification	Retrofit	Fully Qualified
Separation	Mixed	Complete
Fire Protection	Distance or barriers	3 hour barriers
Structural Loads	Retrofit	Fully defined, tested
Plant Operation	Manual and Demanding	Semi automated and Pre engineered
Fuel	GE standard	GE standard
Reactor Vessel	Welded plate	Forged rings
Control Rod Drives	Hydraulic	Electro hydraulic
Neutron Monitoring	3 stage	2 stage
Safety/Relief Valves	Pilot operated	Air operated
Control & Instrumentation	Analog	Digital
Cabling	Hardwired	Multiplexed
Control Room	System based	Operator task based
Offgas	Complex	Simplified
Feedwater	Turbine/motor	Variable speed motor
Turbine Last Stage Bucket	43 inches	52 inches
Severe Accident Mitigation	Active	Passive
Core Flow/Recirculation	External pumps	Internal pumps (ABWR)
		Natural circulation (SBWR)
Safety Systems	Active	Simplified active (ABWR)
		Passive (SBWR)

Table II - SBWR Technical Data Summary

Plant output	
Net electrical output	600 MWe
Plant cycle	Direct
Vessel dome pressure	7 17 MPa
Main steam flow	3490 tons/hour
Turbine	TC2F-52 inches
Reheat stages	None
Nuclear boiler	
Reactor vessel	
Inner diameter	6 m
Height	24 6 m
Primary coolant recirculation	Natural circulation system
Recirculation flow	23,700 tons/hour
Core and fuel	
Active fuel length	2 743m
Equivalent core diameter	4 73 m
Power density	41 5 kw/1
Number of assemblies	732
Fuel material	UO_2
Cladding material	Zircaloy 2
Fuel lattice type	8x8 barrier
Reactivity control	
Number of control rods	177
Neutron absorber	B_4C
Control rod form	Cruciform
Control rod drive	Electro-hydraulic, fine-motion
Other control	Burnable poison (Gd203)
Containment	
Type	Pressure suppression
Configuration	Cylindrical reinforced concrete, steel lined
Construction Schedule	
First concrete to fuel load	30 months

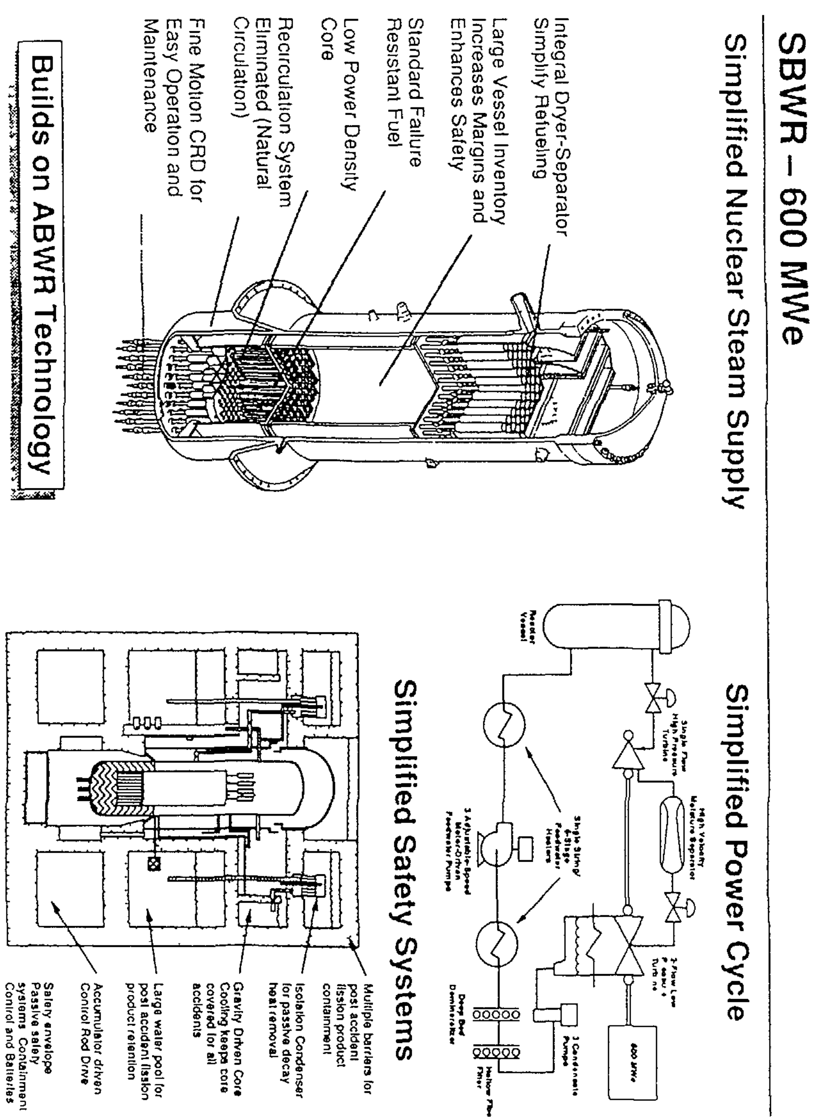

Figure 1. SBWR Design Features

2. DESCRIPTION OF THE SBWR

The SBWR power production systems utilize the simplicity inherent in a direct cycle nuclear plant The nuclear fuel directly heats the water that is converted to steam in a single pressure vessel This direct cycle has the inherent advantage of being easy to startup, operate and shutdown during power production and following transients and accidents The SBWR incorporates an additional major simplification -- elimination of any pumping to recirculate the water through the core by using natural circulation The use of natural circulation results in an extremely reliable and simple system to produce the steam needed to drive the turbine and generator The turbine island includes several simplifications This simplification is achieved by the use of advanced GE turbine technology and the use of a turbine design with tandem compound two flow (TC2F) 52" last stage buckets Table II summarizes the key technical parameters of the plant Figure 1 shows the SBWR reactor vessel and a schematic of the SBWR plant and safety systems

The power cycle is dependent on auxiliary systems during normal operation and startup and shutdown The SBWR has a design that has improved the reliability of these systems by the appropriate use of margin, redundancy and diversity in these auxiliary systems For example, shutdown cooling has been considerably simplified by incorporating the ability to remove decay heat over the full reactor pressure range by modifying the reactor water cleanup system and eliminating a separate shutdown cooling system The entire man machine interface aspects of the design have been substantially improved using state-of-the-art technology, including use of digital controls, fiber optics and multiplexing

Major strides have been made in the SBWR for the handling of operational transients and accidents The basic philosophy has been to first build in inherent margin into the design to eliminate system challenges The second line of defense is to enhance the normally operating systems to handle transients and accidents And as a final line of defense, passive safety-grade systems (see Figure 1) have been included in the design to provide confidence in the plant's ability to handle transients and accidents

The system simplifications discussed in the above paragraphs have resulted in a building design that is easy to construct and have also permitted a significant reduction in safety-grade equipment and structures The use of passive safety systems requires only a relatively small containment and a small area around it to house this equipment, resulting in major plant simplifications by reducing the safety envelope (i e , building volumes) and thus simplifying construction The basic configuration of the structures results in a building design that has multiple compartments and barriers to leakage of fission products following accidents

3 DEVELOPMENT TESTING AND VERIFICATION

The overall approach to the design of the SBWR has been to use proven state-of-the-art LWR technology with modest technology extensions Where innovations or technology extensions were made there was a plan to perform early confirmatory testing The major sources of technology have been

a) The 30 year international investment in the worldwide complement of operating BWRs

b) Adoption of proven and state-of-the-art technology from the ABWR program

c) Completed SBWR specific technology development programs designed to show the feasibility and performance of SBWR unique features

d) Ongoing development programs that will demonstrate the margin built into some of the key safety features - such as decay heat removal and inventory control

The emphasis of new testing and analyses falls into two categories a) component testing to ensure performance and reliability of the design during the lifetime of the plant and b) testing analysis to provide a high confidence that safety systems will perform as expected and as analyzed in safety analyses The component testing focused on major new features whereas the safety system testing focused on passive safety systems

Table III summarizes the technology base for the key SBWR features It shows that the majority of the new features are based on applications from operating plants or the technology developed for the ABWR In a few cases there is a modest extrapolation from operating plants experience These extrapolations do not result in operating conditions for the system components that are beyond the range of operating experience, ensuring high confidence in the system operation

The four basic, safety functions decay heat removal and control of coolant inventory, reactivity and fission products are being performed using simplified safety systems that rely primarily on natural forces like gravity, to operate This approach is different in some respects to that for operating plants, hence an extensive test and development program has been focused on these areas The following paragraphs discuss this technology development in the high and low pressure inventory control, high and low pressure decay heat removal, reactivity control and fission product retention and control

High pressure decay heat removal and inventory control for the SBWR is performed using three isolation condensers - heat exchangers that condense the steam generated by decay heat and return the condensate by gravity back to the vessel The basis for this technology is the experience gained over the last 20 years with isolation condensers on operating plants Since the basic concept is the same no feasibility issues were raised However because of the use of vertical tubes instead of horizontal tubes on operating plants, a full scale confirmation test is planned for completion in 1993 This test program will verify the margin in the thermal hydraulic performance and mechanical design of this key component

Inventory control following loss of coolant accidents is handled similar to operating plants by depressurizing the reactor to enable low pressure inventory makeup The depressurization system has been made more reliable by providing diversity using the standard safety/relief valves and a squib actuated

Table III Technology Base for SBWR Features

	SBWR Feature	Experience/Technology Base
1	Natural Circulation	- Natural circulation operating plants e g Dodewaard, Humboldt Bay
		- Forced circulation operating plants operating without recirculation pumps
2	Fine Motion Control Rod Drive	- Operating plants in Europe
		- ABWR test/development program
		- Demonstration test at the Lasalle plant
3	Turbine Island Simplifications	
	- 52" Last stage buckets	- Operating plants
	- No reheat and single feedwater heater train	- Operating plants
	- High velocity moisture separator	- Operating plants
	- Adjustable speed drives for feedwater pumps	- Operating fossil fuel plants
4	Materials	- Operating plants technology
5	Digital Control and Instrumentation	- Retrofits to operating plants
		- Experience in other industries e g airline
6	Multiplexing and fiber optics	- Operating plants and airline industry
		- GE ERIS/Omnibus products
7	Man Machine Interface	ABWR program
8	Reinforced concrete technology and pressure suppression containment with horizontal discharges	- ABWR development testing
9	Equipment and structural modularization techniques	- ABWR Program
10	System simplifications/ enhancements e g CRD hydraulic flow increase, full pressure shutdown cooling	- ABWR design and operating plants
11	Refueling system	- Extrapolation of operating plant remote technology

depressurization valve The latter being a new feature, it was tested extensively to choose a reliable propellant and confirm the reliability of the valve design under various operating conditions Unlike operating plants, the inventory makeup relies on gravity instead of a pumped system To demonstrate that the low driving heads and slower mixing did not result in new phenomena, a full height, volume scaled system test was performed of the gravity makeup system The tests were completed in 1989 and were used to qualify the analytical codes described in reference [3] These tests supplemented the already existing extensive data base used to qualify the model in reference [3]

Since the major safety innovation in the SBWR is the use of the isolation condenser system to remove decay heat for all transient and accident conditions, a major effort went into demonstrating the feasibility of the system and to develop the additional technological base The testing program involved testing to develop heat transfer correlations for full length individual tubes Then a full height volume scaled integral system performance test was performed to demonstrate the feasibility of the concept This test was completed in 1990 and showed that the noncondensable gas is purged from the isolation condenser to the wetwell via the vent line (Figure 1), as a result of the drywell/wetwell pressure difference Additional testing is planned to demonstrate the margin in the overall plant design These tests include a full scale heat exchanger mechanical and thermal hydraulic performance test to be completed in 1993 An additional full height integral test is planned for completion in 1995 to demonstrate the impact of 3-dimensional effects on the system response This facility is full scale in height and about 1/25 by volume of the drywell and wetwell volumes

The SBWR has a diverse reactivity control system consisting of an accumulator driven boron injection system Confidence in the ability to get the boron well mixed in the vessel is based on extensive testing done as part of the ABWR program, where testing for a number of injection locations and configurations showed that the boron is well mixed

4. SUMMARY AND CONCLUSIONS

The SBWR design is based on the extensive experience gained from the 30 years of LWR operating experience This experience has been supplemented by the utilization of technology developed over the last 15 years for the ABWR, which is a reactor that utilizes state-of-the-art technology SBWR specific testing was undertaken under sponsorship of the U S Department of Energy, with an emphasis on the innovations in the safety systems Testing to demonstrate feasibility has been completed Ongoing or additional future testing is planned to better quantify the margins in the overall safety systems designs

REFERENCES

[1] Hucik, S A , Rao, A S 'ABWR-First of the New Generation of Reactors for the 90 s" IAEA meeting on "Progress in Development and Design Aspects of Advanced Water Cooled Reactors" Rome, Italy, Sept 9 12, 1991

[2] Rao, A S , Sawyer, C D and McCandless, R J , "Simplified Boiling Water Reactor Design" JSME/ASME International Conference on Nuclear Engineering, Tokyo, Japan Nov 3-8, 1991

[3] Andersen, J G M , Chu, K H and Shaug, J C , "BWR Refill-Reflood Program Task 4 7-Model Development Basic Model for the BWR versssion of TRAC, August 1983", GEAP-22051

CONCEPTUAL DESIGN OF AN INHERENTLY SAFE AND SIMPLE TUBE REACTOR USING WATER MODERATOR AND COOLANT

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Abstract

This paper describes conceptual design features of an inherently safe and simple tube reactor (ISSTR) using light or heavy water moderator and coolant The ISSTR is a loop-type pressurized water reactor which does not require emergency core cooling system against loss of-coolant accident (LOCA)

In ISSTR, fuel channels are arranged in the water moderator pool within a large Calandria vessel Each fuel channel consists of fuel matrix and a surrounding channel tube Fuel matrix is designed so that fuel elements and coolant holes are distributed in the Zircaloy matrix The radial gap between the channel tube and fuel matrix plays a role as a thermal switch between normal operation and LOCA

In normal operation, the fission heat generated in fuel elements is transferred to coolant and then used to generate steam in steam generators as in the present PWRs or PHWRs At complete LOCA, however, the decay heat is transferred to the moderator by conduction through Zircaloy matrix, radiation between the matrix and the channel tube, and natural convective boiling between the channel tube and moderator Moderator is cooled by natural circulation through a condenser located in the large in-containment water storage tank above the reactor Analysis shows that the maximum Zircaloy temperature can be maintained well below the permissible value during a postulated complete LOCA

The ISSTR presented in this paper is expected to maximize the utilization of the proven technology as well as to revolutionarily improve the safety, since design features of present PWRs or PHWRs can be directly applied except the reactor It is also expected to be easy to test, simple to analyze accidents and easy to explain the safety to the public

1. INTRODUCTION

There is a world-wide effort to design next-generation nuclear reactors which are expected to revolutionarily improve the safety and economy of the nuclear power plant These efforts are pursued mainly in three directions light water reactors (LWRs), liquid metal reactors (LMRs), and high temperature gas-cooled reactors (HTGRs), among which the LWR design concept is the most actively studied primarily due to its much greater operating experience than the other types

Lately conceptual designs of four small-sized passive safety LWRs have been developed AP600 (Advanced Passive 600MWe Reactor), SBWR (Simplified BWR), PIUS (Process Inherent Ultimate Safety) and SIR (Safety Integral Reactor) All of them are designed with a primary objective to minimize the probability of core uncovery since severe accident cannot occur as far as the reactor is properly tripped and core is submerged in coolant Among them, the PIUS and SIR have basically the pool-type NSSS and the design efforts for them seem to be somewhat depressed now AP600 and SBWR, which adopt the loop-type NSSS, are sponsored by the US government and are expected to acquire design certification by the mid 1990s But they require the emergency core cooling system (ECCS) to assure the safety during LOCA Though the ECCS operates using the natural phenomena such as the gravitational force or natural circulation, it requires an active depressurization system for proper injection of emergency core cooling water

There is a general consensus that the loop-type passive safety LWR is the most promising candidate for the next generation reactors before commercialization of liquid metal fast breeder reactors In this regard, a inherently safe loop-type LWR not requiring the ECCS would be the most attractive solution to overcome problems related to safety, licensibility and public acceptance

This paper describes conceptual design features of the ECCS free Inherently Safe and Simple Tube Reactor (ISSTR) under study at Center for Advanced Reactor Research (CARR) in Korea Advanced Institute of Science and Technology (KAIST) The ISSTR is a loop-type pressurized light or heavy water reactor which reflects above considerations Although more detailed design and analysis are being performed now, preliminary design calculation shows that the excessive temperature increase can be avoided even for the complete core uncovery following LOCA

2. CONCEPTUAL DESIGN FEATURES

Basic configuration of the ISSTR NSSS is shown in Figure 1 with more detailed reactor configurations in Figures 2 and 3 *Figures show that ISSTR is similar to CANDU in the arrangement of NSSS equipments* However ISSTR adopts much different core design including matrix type fuel channels The primary design objective is to achieve inherent decay heat removal capability by moderator cooling at complete loss-of-coolant accident

The unique design feature of ISSTR is the use of matrix-type fuel in a tube as illustrated in Figure 3 Each fuel matrix is composed of 8 fuel elements, 12 coolant holes, 1 instrumentation/control hole and Zircaloy matrix in a manner that fuel elements and coolant holes are distributed in Zircaloy matrix Design of fuel elements and coolant holes may have several different options As the first option, we adopt the CANDU-600MWe fuel pellets of 0 012154m diameter stacked in 0 012243m holes The diameter of coolant holes is adopted as 0 008859m which gives the same flow area per fuel element as CANDU-600MWe In this design, the pressure boundary of the primary system would be the Zircaloy matrix But the fuel channel tubes are designed *to the reactor coolant condition so that the failure of Zircaloy matrix can be properly* accommodated

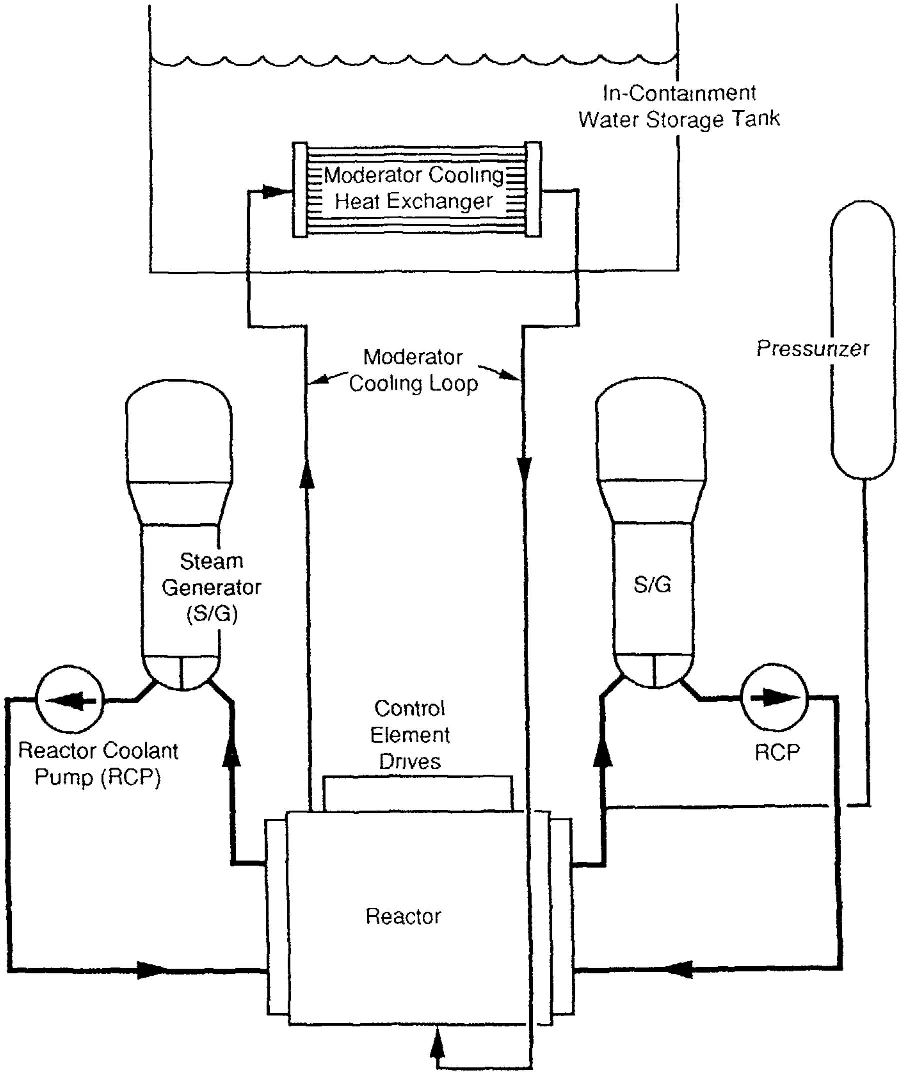

Figure 1 Schematic of the ISSTR NSSS

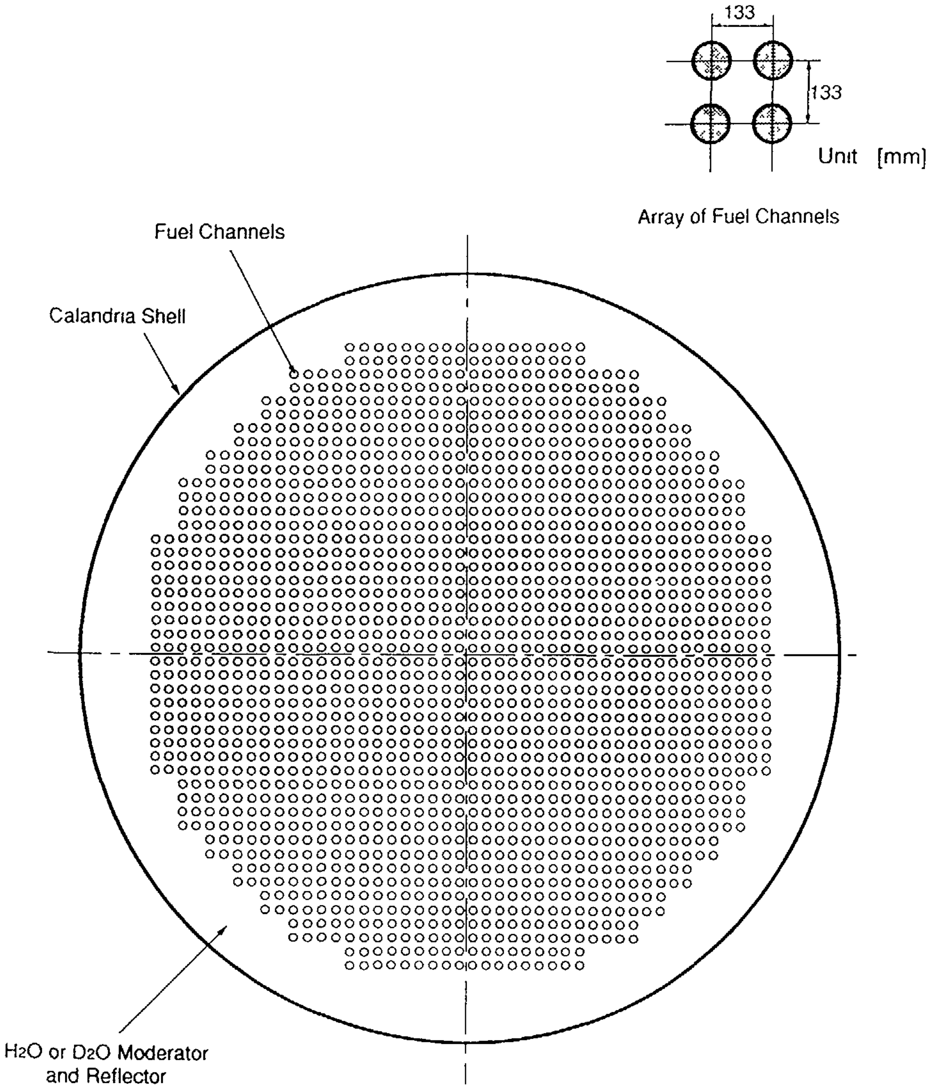

Figure 2 Array of Fuel Channels in Calandria

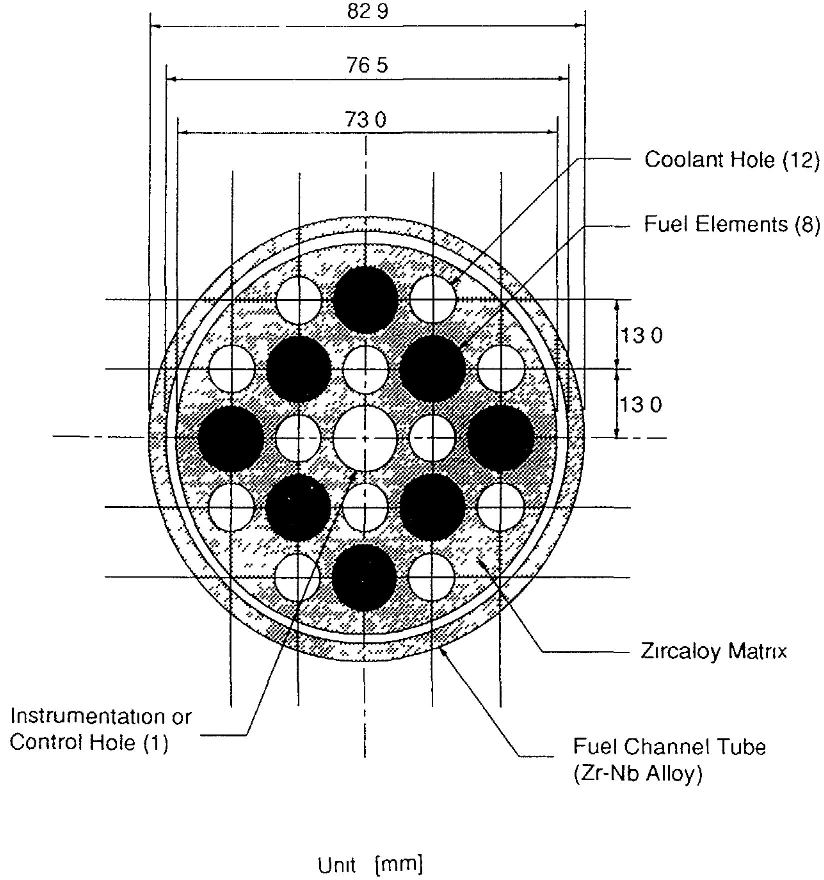

Figure 3 Detail of Fuel Channels

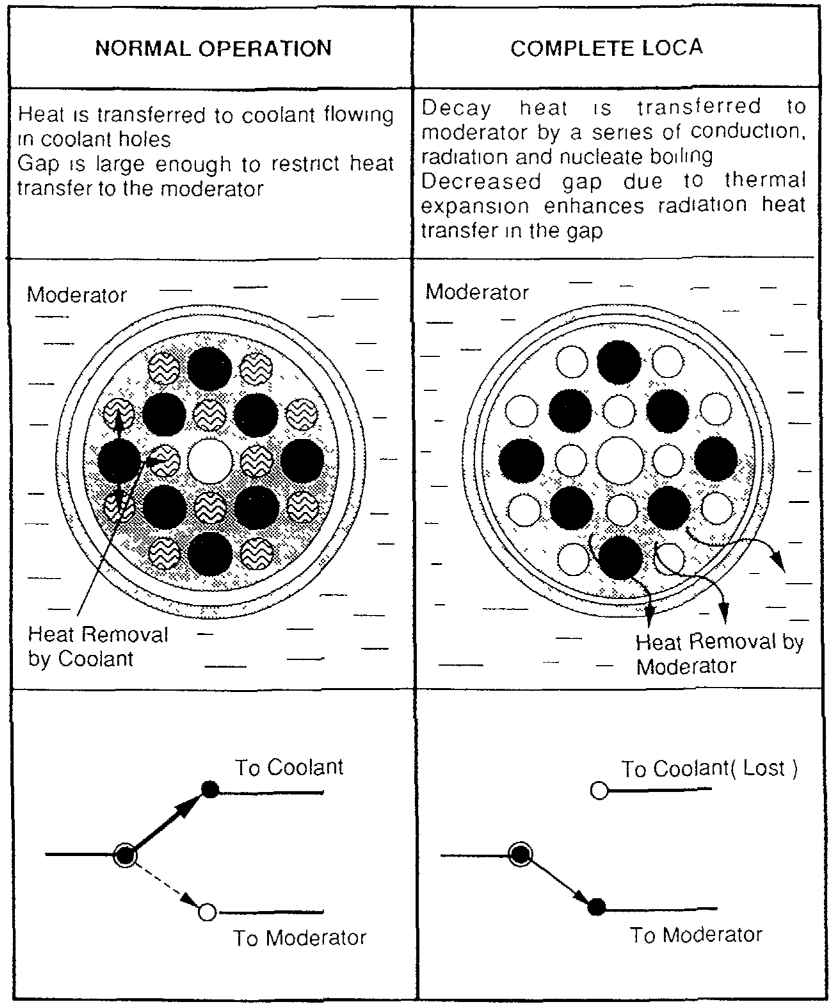

Figure 4. Concept of the Thermal Switch

In normal operation, the fission heat generated in fuel elements is transferred to coolant via conduction through the Zircaloy matrix Heat loss to moderator is minimized by the gap between the channel tube and fuel matrix. The heat transferred to coolant is used to generate steam in steam generators as in the present PWRs or PHWRs

At complete LOCA, however, the decay heat cannot be cooled by coolant. Instead, it is transferred to the moderator by conduction through Zircaloy matrix, radiation between the matrix and the channel tube, conduction through the channel tube wall, and natural convective boiling between the channel tube and moderator. Moderator is cooled by natural circulation through a condenser located in the large in-containment water storage tank (IWST). The Zircaloy matrix behaves as a heat absorber during short period immediately after reactor trip as well as a conductor of the decay heat. Analysis shows that the maximum Zircaloy temperature can be maintained below 1200°C in any location during a postulated complete LOCA. Therefore ECCS is not required in ISSTR. The non-safety grade chemical and volume control system can supply cooling water if electrical power is available.

The concept of thermal switch between normal operation and LOCA is illustrated in Figure 4. The difference in thermal expansion between the Zircaloy matrix and the channel tube due to different temperature rise results in smaller gap in LOCA condition compared with that in normal operation. This effect facilitates the decay heat transfer to moderator in the complete LOCA.

Fuel channels are arranged in the water moderator pool within a large horizontal Calandria vessel as in CANDU. The pitch-to-diameter ratio of the fuel channel array is about 1.54. Reactivity control is basically achieved by the control elements inserted from the top in the moderator pool. Since moderator pool is under near atmospheric pressure, automatic drop of the control elements by gravity is assured at loss of driving power. Furthermore any excessive heat generation to generate the void in moderator gives negative reactivity insertion.

Table 1 Comparison of ISSTR-600 Design Parameters with CANDU-600

Parameters		ISSTR 600MWe (LWR Option)	CANDU 600MWe
Core thermal power [MWt]		2060*	2060
Coolant	- material	H_2O	D_2O
	- pressure [MPa]	11 1*	11 1
	- flow rate [kg/h]	2 68x10^{7}*	2 68x10^{7}
	- temperature [°C]	266 6 / 312*	266 6 / 312
Core	- radius [m]	3 143*	3 143
	- length [m]	5 944*	5 944
Fuel channels	- total number	1748	380
	- fuel elements per channel	8	37
	- pitch [m]	0 133	0 28575
	- outside diameter [m]	0 0829	0 13035 (Calandria tube)
Fuel elements	- total number	13984	14060
	- pellet diameter [m]	0 012154*	0 012154
	- material	UO_2	UO_2
	enrichment	2 - 3 %**	natural
Moderator	material	H_2O	D_2O
	- pressure [MPa]	~ 0 1***	0 1

Notes * The same data as CANDU-600 are used

** 1 - 2% for the HWR-option

*** Subject to change according to the cooling loop design

A loop is provided for moderator cooling in both normal operation and accident condition A centrifugal circulating pump is located in the loop to achieve forced circulation in normal operation In case of the pump trip at accident condition, the two phase natural circulation mode is established The heat exchanger (condenser) in IWST is designed so that single-phase and two phase circulation is established in the moderator cooling loop in normal operation and complete LOCA, respectively

Many options are possible for the design parameters of the reactor coolant system Particularly PWR or PHWR(CANDU) design parameters are the most attractive options because they allow us to use the similar NSSS equipments as in the present plants Considering the configuration of ISSTR similar to CANDU, the CANDU design parameters are selected as the first option In this case, three different moderator and coolant combination is possible, H_2O-moderator / H_2O-coolant (LWR option), D_2O-moderator / H_2O-coolant, and D_2O moderator / D_2O-coolant (HWR option) Passive containment concepts proposed in AP600 et al can be directly applied for ISSTR

The more detailed design study are now being performed at CARR with consideration of various options for design parameters and arrangement of NSSS Basic design parameters of ISSTR 600MWe (LWR Option) are compared with CANDU 600 in Table 1, admitting that they are only typical values Here the same design parameters are used for core and Calandria dimensions, coolant pressure and flow rate, etc

3. DISCUSSION

The ISSTR presented in this paper has several advantages over existing LWRs and suggested passive LWRs as follows

a) The safety of reactor is remarkably improved since no active system is required to cool the core after LOCA Core damage could occur only at the simultaneous loss of coolant and moderator which constitute separate loops

b) Selection of loop type NSSS maximizes the use of proven technology and existing design of NSSS components

c) Elimination of the ECCS makes the plant be simple, economical, easy to design and analyze, and easy to understand

d) Design verification test is remarkably simple and requires short time Two principal subjects to be tested are the heat removal capability of a separate uncovered fuel channel and natural circulation of the moderator cooling loop

e) Change of reactor power is easily achievable by changing the number of fuel channels and re sizing of the coolant loop and the moderator cooling loop

Though more detailed analysis is still needed to optimize the basic design parameters of the ISSTR, it has been clearly shown that inherently safe decay heat removal can be achieved without the ECCS and low enrichment fuel can be used even for the LWR concept The ISSTR concept suggests us several specific research topics in both thermal hydraulics (T/H) and material engineering

In the T/H area, the radiation heat transfer in the gap between the Zircaloy matrix and the fuel channel tube is very important Extensive experimental study is required to determine the radiational heat transfer properties of the materials and to determine the optimum gap size Low pressure two-phase flow and boiling is another important subject in relation to the moderator cooling Here low pressure natural circulation, nucleate boiling and condensation would be the typical subjects

In the materials area, manufacturing of the Zircaloy matrix would be the most important subject As the manufacturing process is improved with improvement of mechanical properties, the smaller and more ecomomical fuel channel can be achieved

As described in the previous section, ISSTR may have various options in the aspects of loop design and thermal power ISSTR concept can be easily applied for the reactor of various purposes including the small sized district heating reactors or research reactors

4. CONCLUSION

The concept of the Inherently Safe and Simple Tube Reactor (ISSTR) is briefly introduced in this paper It is shown that an ECCS free and economical loop type water reactor is feasible with maximizing the use of existing technology It is also expected to be easy to test simple to analyze accidents and easy to explain the safety to the public Many options are possible in the aspects of power and loop design Comparative studies for the different possible options and more detailed design study are now being performed

Acknowledgement

The authors express great appreciation to many colleagues, especially to N E Todreas, MIT, whose advice helped us to choose the direction of the reactor development

References

1 K Hannerz, Making progress on PIUS design and verification, *Nucl Eng Int l* 29 31 (Nov 1989)

2 S N Tower et al , Passive and simplified system features for the Westinghouse 600MWe PWR, *Nucl Eng Des* **109**, 147 154 (1988)

3 R Bradbury et al , The design goals and significant features of the Safe Integral Reactor, Presented at ANS 1989 Annual Meeting, Atlanta (June 1989)

4 R J McCandless and J R Redding, Simplicity the key to improved safety, performance and economics, *Nucl Eng Int l*, 20-24 (Nov 1988)

5 W Kwant et al , PRISM reactor design and development, *Proc Int Topical Mtg on Safety of the Next Generation Power Reactors*, Seattle, 130 135 (May 1988)

6 J Varley, Interest grows in the Modular HTGR, *Nucl Eng Int l*, 25-28 (Nov 1989)

7 P Hejzlar, N E Todreas and M J Driscoll, Passive Decay Heat Removal in Advanced Reactor Concepts, MIT Report No MIT-ANP-TR-003 (May 1991)

8 W M Rohsenow and J P Hartnett, *Handbook of Heat Transfer*, McGraw-Hill, New York (1973)

9 F P Incropera and D P DeWitt *Fundamentals of Heat Transfer* (3rd ed), John Wiley & Sons, New York (1990)

CONTAINMENT SYSTEMS

(Session 2)

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TRENDS IN THE REQUIRED PERFORMANCES OF CONTAINMENTS FOR THE NEXT GENERATION OF NUCLEAR POWER PLANTS

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Abstract

There is a widespread consensus that the containment system will continue to play a key role in the safety philosophy of next generation plants as it did in most designs of commercial nuclear power plants in the past. But, at the same time, it seems clear that the design bases should be revisited and extended to provide a clear technical answer to the events that operational experience and analyses showed plausible.
Required performances will not be limited to assure the gross structural integrity of the containment structure but also its leaktightness, in order to provide technical bases for simplified Emergency Planning requirements and for the reduction of the evacuation zone to (or closed to) the site boundary, as it was before the TMI-2 accident.
The paper outlines some criteria currently being discussed in Italy, describes some common trends that can be identified among several countries, underlines the need for additional design guidance and standards, and, finally, points out the need for an international consensus on such themes, which could be achieved also with the help of international organizations such as the IAEA.

Introduction

In the followup of the Chernobyl accident public debates on the use of nuclear energy in Italy lead to the cancellation of the existing projects and the shutdown of the operating plants, but, at the same time, the Governement requested studies and evaluations of a new generation of nuclear power plants with enhanced and transparent safety characteristics more acceptable to the public.
ENEL has complied with these general guidelines and is producing a number of detailed requirements to be discussed with the Industrial and Regulatory parties in Italy. These requirements, based on well proven technologies and operating experience, will be the benchmark to assess the compliance of the existing projects with the Italian objectives; at the same time ENEL is pursuing the largest international consensus on these requirements, since it is our belief that the acceptability of any future projects will be played in an International arena and single country approaches will be more and more difficult to support for socio-political, industrial and economic reasons.

Safety objectives for a new generation of nuclear reactors in Italy

The step forward in safety requested in Italy has been interpreted by ENEL in a number of general safety objectives [1]:

- the impact on the population and on the environment shall be virtually negligeable for any credible situation and, in particular, evacuation of the population shall not be needed in any conceivable circumstances,

- safety levels shall be achieved as indipendently as possible from operator actions; operator role, however, shall be facilitated and optimized to provide even larger safety margins,

- passive safety systems and inherently safety characteristics shall be included in the design to the extent possible in order to achieve the safety levels by natural laws, reducing the potential failure modes,

- reactor design shall provide slow and benign responses to all transients and accidents to facilitate operator actions, to reduce the stresses to the components and to leave the longest time to recovery actions.

It is worth mentioning that the means to achieve the above objectives shall mantain compatibility with plant productivity and economics goals.

Containment role and required performances evolution

The control of public exposure after an accident was first attained by the use of large exclusion areas surrounding the first non-commercial reactor installations [2]. Thus in 1943 the Clinton pile in U.S. was located in a portion of 240 km^2 restricted area in Tennessee, designated Oak Ridge; the graphite production reactors were located in an even larger restricted area, designated Hanford, in the Washington desert near the Columbia river, and the National Reactor Testing Station was established in the desert of Idaho. In each of these istances the distances from the reactors to the site boundary was intended to permit adequate control (i.e. reduction) of any activity that might be accidentally released.
However, the need to site the commercial reactors in populated areas, where restricted areas could be much smaller, required the reduction of external releases by an outer vessel or containment, which was very simple at the beginning and became more complex later, including several active systems to control containment atmoshere and valves to isolate the penetrating lines.
Consequence calculations were requested in the USA to be carried out with a severe accident type source term, and an intact containment with the performances required for a LOCA. The distance over which emergency planning was required was generally limited to an area around the plant site referred to as the Low Population Zone, which typically was about two to three miles in radius. As a result of this relatively small area, emergency planning only indirectly involved offsite authorities.
The containment performance requirements were, at a certain extent, decoupled from the presence of core damage preventing systems, in line with the defense in depth philosophy; therefore a containment has been always required in the commercial LWRs of the western world.
The problem at that time was to select the appropriate design bases for such a containment.

The risk of severe accidents was recognized since the dawn of the commercial reactors era as real, but remote and largely undefined. At that point a decision was made to use a surrogate accident as a design basis for the containment and to move forward with the development of nuclear power. That surrogate accident, a sudden large-break LOCA, has been the basis for a conservative LWR containment design in the western world setting the design thermohydraulic conditions.
At the same time in the USA fission products assumed to be released into the containment were selected in order to represent a severe core damage event (at least for noble gases and iodine), rather than the design basis LOCA (3).
In many European countries the approach in general was similar with regard of considering the LOCA as the reference accident, but differed in terms of the associated source term, which was more realistic, or mechanistic, while at the same time the allowed population doses were lower than 10CFR100 (4).
While in the early '70s the ECCS rule in the US was published and a net improvement was achieved on the prevention side, the situation remained essentially stabilized with regard to the containment design and assessments until WASH-1400 was published; as a result of the study a probability was calculated for each containment failure mode and a systematic list of containment failure modes was presented; the conclusions were strictly valid for the plants included in the study, but in general were considered as typical of the American type containments.
The conclusions were that, in the average, plant safety levels were acceptable (no immediate regulatory action followed the publication of the results) and that in particular no generic action was warranted for the containment.
But, at the same time, actions were initiated in the USA to increase the EPZ size, in line with the potential large releases calculated by WASH-1400. In December 1978 (NUREG 0396) an NRC/EPA task force reccommended a plume exposure planning zone of about 10 miles and an ingestion pathway zone of about 50 miles in radius.
During the 1979 accident at TMI-2, a containment designed to the DBA requirements functioned effectively to protect the public in case of a large core degradation and melting providing a confirmation of the capabilities of the containment design basis, but the emergency planning organization, although not actually needed, was considered poor. Therefore both the Kemeny and the Rogovin reports on the accident pointed out the need for better Emergency Planning and better management; so in the aftermath of the accident, regulations were promulgated essentially in agreement with the above mentioned task force reccommendations assigning a leading role on Emergency Planning to FEMA.
Severe accident researches and risk assessments, increased in number and scope since 1979 both in USA and in Europe; they confirmed the conclusions of WASH-1400 that a broad range of high-energy loads caused by accidents more severe than at TMI-2, might threaten containment integrity and source term reduction systems, but provided more realistic and supportable data to quantify their effects.
Mainly in Europe backfitting actions were initiated on the containment and on the Emergency procedures.
While these actions were evolving in 1986 the Chernobyl accident occurred and the lack of an efficient containment system was blaimed as a major cause of the effects on the environment. The accident gave an additional push to the evaluation of stronger containments on one hand and of severe accidents on the other.
Therefore PSA results and, much more, the profound impact of these two accidents seemed to show at the end of the '80s with striking evidence to most designers, utilities, Regulatory Bodies, scientists and general public that the dividing line between design basis events and incredible accidents has not been drawn at the right place. The consequence was in many cases the adoption of backfitting measures on the operating plants and a few regulatory requirements for future reactors.
In summary the situation of containments of current operating plants is generally the following. Current plants may be proved to survive many severe accident sequences giving credit to the margins in the design, a proper Accident Management and taking advantage of a filtered containment venting to prevent containment failure by late overpressurization. However, since certain highly unlikely severe accidents phenomena could cause containment loads well beyond the DBA loads, and since the backfitting actions could not change some characteristics of the existing containments, there is still the possibility, although very remote, of a release of health-threatening quantities of fission products. In order to cover this residual risk, Emergency Plans enforced for all commercial reactors; their size and complexity are generally in relationship to the extent of the implemented backfitting actions.
It could be argued from this historical outline that the approach so far has not been fully consistent, albeit it resulted in strong and effective containments. We believe that the industry has the possibility now to rivisit the entire approach in the future plants and, on the bases of the results from the research, the operating experience and the insights gained from PSAs, explicitly include severe accidents and other credible "beyond current design bases" into the containment design process. Of course while expanding the design bases the "rules of the game" for the assessement of these extreme events should be changed and many conservative assumptions used for DBA shall be reconsidered. On the other side these improvements should be reflected in a more rational approach to the EPZ requirements.
However, once established that a revisiting of the design bases is needed, a number of questions should be answered. For most of them ENEL is currently developing a position in concurrence with the original designers, the Italian industry and Safety Authority, and is mantaining a technical dialog with foreign utilities and international organizations.
One of the most important issues is the selection of the reference scenarios to be taken into account for the containment design.
It is not in the scope of the present paper a detailed discussion of the selection process, considering also that the details are different for different reactors. The general phylosophy, however, is that this process cannot be limited to the use of a probabilistic approach but it should be supplemented by a well-balanced engineering judgement and, where applicable, by the operating experience analysis. The combination of the selection approaches should assure that even the all plausible events are properly assessed in the design process.
In the following some specific requirements are presented.

Structural requirements

Containment structure shall withstand all internal and external loads associated with internal accidents (including credible severe accidents) and external events (including in our case a severe earthquake and an aircraft crash).
In order to mantain the required leaktightness even large deformations, not involving pressure boundary collapse, shall be avoided.
Severe accident loads may vary depending on the specific sequence. All potential loads shall be evaluated and included in a realistic and mechanistic load combination, if they are not prevented by design, so that they are impossible.

The phenomena associated with severe accidents and subject to the above screening process are:

- In-vessel and ex-vessel steam explosions,
- Direct containment heating,
- Hydrogen detonations and combustions,
- Molten core-concrete interactions,

In addition the extreme seismic loads will be identified with a similar process and factored into the design.
The ultimate capacity of the structure shall also be investigated with the aim of verifying the margins in the design with respect to the expected loads; it is required that, indipendently from the fact that collapse pressures are not expected in any circumstances, as a good engineering practice, the containment structure be designed to have a leak-before-burst behaviour and a preferential rupture point be included. As good engineering rupture point shall be located so that surrounding buildings can still provide a reduction in the radioactive release to the environment.
As already said an important question is what design margins should be used for severe accidents. Traditional engineering procedures include the use of large safety margins. Part of the reasons of the safety margins of the containment was the some kind of protection against events beyond the design bases. However in this new approach, where all events are considered in the design, we consider acceptable a realistic assessement of severe accident consequences, provided that uncertanties in the load calculation and design methods are quantified and taken into account. It is clear that national and international rules and regulations shall be interpreted and even updated to formalize this approach.

Leaktightness requirements

The main charter of the containment is, of course, to contain the fission products released in case of an accident and it is clear that its performances in terms of leak rate are strictly related to the required limitation of external consequences, once an accident happened.
We consider that a design goal of 0.1 % per day in weight of the internal atmosphere would be feasible without any major step forward in the current technology, taking into account the operating experience and the simplifications in the isolation systems in terms of number of valves. This value should represent a significant margin with respect to the value to be verified by periodic testing and to be credited in the safety analysis, which could be higher. Specific features such as penetration pressurizing systems could be added, if necessary, particularly on the lines directly connected to the environment.
In addition the contribution of the primary containment to the dose reduction shall be supplemented by a passive secondary containment, which shall be capable of holding up the leaks from the primary containment. Since it may be expected that current construction technologies may well limit the leak from a reinforced concrete shell with a metallic liner or from a metallic containment shell, this concept may be implemented by enclosing the penetrations into a sealed or low-leakage structure. Particular care shall be devoted to the potential bypass paths of this structure.
New containments should not need a filtered containment venting, since they are designed to withstand all credible sequences, even without operator actions for a long period. Moreover the compatibility of a containment venting to the environment with the low doses to the public without credit to evacuation is highly questionable. A small vent may be added to finally depressurize the containment in the long term and terminate any release, but its use shall not be dictated by the risk of a containment failure.
An interesting issue is the possibility of a so-called primary containment bypass, which is, as a matter of fact, speaking of a primary containment not really a bypass, but a leak occurring at a location not considered in the usual leak paths. A major example is a single or multiple Steam Generator Tube Rupture producing a direct path from the primary system to the environment. Such "bypasses" shall be eliminated since their effects are not easily quantifiable and may exceed the external dose limits, even without a substantial core damage. In the case of a SGTR a possible solution is to increase the design pressure of the steam generators secondary side, in order to prevent the opening of the safety and relief valves on the steam lines.
The isolation system shall be passive by at least one of the IAEA definition of passivity (5). This implies that e.g. Motor Operated Valves cannot be closed by ac motors supplied by large distribution networks and rotating machineries.
A continuous monitoring system for the containment integrity, e.g. similar to the one include in the EDF plants, is required to further reduce the probability of a large preexisting opening.
A potential issue is the Integral Leak Rate Test pressure; if the containment shell is not designed for normal operation at severe accident pressure it is not possible to test the leak rate periodically at such a pressure. This problem may be, however, overcome by some special design features and testing provisions: most isolation valves, for istance, may be tested individually at severe accident pressure and other penetrations as well, while special provisions may assure that the leaks through the shell (both steel or concrete) are negligeable with respect to the others.

Heat Removal Capability Requirements

Since the containment is expected to provide a sealed boundary with very high leaktighness performances, it shall provide the Ultimate Heat Sink function directly, or it should provide at least a passive path to remove the decay heat. In general, however, contaminated fluids (air/steam and water) shall remain in the containment boundary or in its close proximity, in order to minimize the possibility of radioactive spreading in the plant and the environment.
The required performances shall be carried out by passive systems (mainly natural circulation and large heat transfer surfaces).
The heat removal function shall prevent containment overpressurization, in the short and long term, and, if possible, shall not require any actuation signal or valve movement; at the same time the performances shall be assured without credit to operator actions for at least 72 hours. A design capable of preventing containment failure also in the absence of any actions in the very long term (at least 30 days) is in our opinion feasible and would add margins to the assurance of the containment integrity.
In addition, since releases should be terminated as soon as possible, the internal pressure should be reduced and practically equalized with the atmosphere in the short time after the 72 hours limit, taking into account also the use of qualified non-safety, ac powered systems.

The role of the International organizations

There is presently a number of activities in all international organizations related to the development of containments for future designs, including design criteria.

For example the WG1 of the CEC, which has as a main charter the harmonization of the rules and regulations in the European Market member countries, has included severe accidents and containment as priority targets. A Reinforced Concerted Action will be probably launched in the next future on the themes of the containment with specific reference to the future reactors. The need for an harmonization in the member countries ig going to be more and more stringent with the increased integration of the economies, the industrial structures and the social attitudes.
OECD has a number of Principal working groups, expert groups, etc. on the subjects of severe accidents and containment, which are working hard to compare different national positions, to produce documents on the state-of-the-art and to promote benchmark exercises.
The IAEA has a unique position, in that it is the only worldwide organization in the nuclear energy field. It may provide a unique forum for discussions and information exchanges for utilities, industries, regulatory bodies separately and not.
Its role is very important since safety criteria and, more generally, the safety culture shall be harmonized worldwide and should not be better or worse in relationship with the specific situation of each country (public attitude, social and political environment, economic situation).
One specific subject on which the international organizations could produce an extremely important position is the definition of a dose limit, under which by any practical means no effect on the population may be expected and the calculation of collective doses are meaningless. This is the standard approach for all toxic chemicals and the same approach should applied to radioactive materials; in practice this has been adopted for radioactive waste with the introduction of Below Regulatory Concern definition, but a similar approach, reasonably developped for practical purposes, should be applied to the public doses also. This limit, together with a proper selection of plausible severe accidents to be considered, could provide the needed boundary conditions to optimize the containment design.

Conclusions

Next generation plants should provide, in general, "better" safety than existing plants. This may not imply that public protection will be better than the best existing plants, but that protection levels shall be achieved in a more transparent way(e.g. by explicitly including severe accidents in the design process), not relying on protective actions disrupting people life and perceived by the public as "Damocle's sword", which may spring out every single day. These are the requirements for the next generation plants to be adopted in Italy. We are confindent that the new designs presently under scrutiny may comply with the Italian requirements.
With specific reference to the containment there are clear indications that there are converging international trends to similar design approaches for future plants, but this trend shall be emphasized with the help of the interested international organizations.

REFERENCES

(1) L. Noviello, I Tripputi - Italian criteria for a new generation of nuclear power plants - Paper presented at the SMS Nuclear Reactors Seminar, New Delhi - August 1991

(2) Wm.B. Cottrell, A.W. Savolainen - U.S. Reactor Containment Technology - ORNL-NSIC 5 - August 1965

(3) J.J. DiNunno et al., Calculation of distance factors for power and test reactor sites, U.S. Atomic Energy Commission, TID-14844 (March 1962)

(4) H. Karwat - Commission of the European Communities, Practices and rules applied for the design of large dry PWR-containments within EC countries - EUR 12251 EN - 1989

(5) Safety related terms of advenced nuclear plants - IAEA - draft TECDOC report - December 1990

PIUS, ASPECTS OF CONTAINMENT: PHILOSOPHY AND DESIGN

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Abstract

A key word in safety philosophy and reactor design is defence in depth. Defence in depth can be expressed as a hierarchically ordered set of different, independent levels of protection. These different levels include both accident prevention and accident mitigation. It is the combined efforts of prevention and mitigation together with accident management and off-site intervention that should be assessed to establish the environmental consequences.

The containment is one of the most important mitigative measures, after Chernobyl probably the most important. When considering the mitigative measures one has, especially as a reactor vendor, also to consider what can be done and/or what has been done regarding the preventing measures.

The weaker the first line of defence (preventive measures) the more efforts have to be spent on the following lines of defence (mitigation, accident management and off site intervention). A very strong first line of defence will on the other side ease the burden of the following lines of defence. Consequently, more efficient and reliable preventive design efforts should make it possible to accept simplifications in the containment design. The trade off between preventive and mitigative measures should be carefully considered for the next generation of reactors.

The current trend is to employ more inherent and/or passive safety features (preventive measures) in the reactor design. The goal is to decrease the risk of core melt by simplifying the plant, ease the burden of the operators and eliminating the need of safety-grade AC-power. The containment should preferrably be passive and above all for a passive plant.

Double containment requires safety grade AC-power and will consequently not be compatible with the design goal for a passive plant.

1. Introduction

Two nuclear plant accidents, the TMI and the Chernobyl accidents, have affected many people's view on reactor safety.

After TMI a number of safety measures were undertaken to lower the probability of such events and to decrease the environmental impacts should they occur.

After the Chernobyl accident it was obvious that public aversion of sudden short timeframe catastrophic events, especially in the nuclear industry, is much higher than the opposition to continuous releases, regardless the overall risk involved. During the last decade it has become evident to the nuclear community that in order to make the nuclear option more attractive a rebuilding of public confidence is necessary. The nuclear technology must be based on a higher degree of safety and based on principles clearly understandable by laymen.

A key word in safety philosophy and reactor design is defence in depth. Defence in depth can be expressed as a hierarchically ordered set of different, independent levels of protection. These different levels include both accident prevention and accident mitigation. It is the combined efforts of prevention and mitigation together with accident management and off site intervention that should be assessed to establish the environmental consequences.

The containment is one of the most important mitigative measures, after Chernobyl probably the most important. When considering the mitigative measures one has, especially as a reactor vendor, also to consider what can be done and/or what has been done regarding the preventing measures. The trade off between preventive and mitigative measures should be carefully considered for the next generation of reactors.

The PIUS reactor belongs to the next generation of LWR plants and has been under development by ABB Atom during the last decade [1, 2]. This paper will present aspects on the containment issue related to PIUS.

2. Safety Guidelines for the next generation of nuclear power plants

The current generation of reactors represents a proven, mature technology. Most of the plants are light water reactor plants (LWRs) and consequently the LWR technology is the most established and wellknown.

It is natural that the next generation should be based upon and utilize the vast experience from the about 4,500 reactor years of LWR experience cumulated so far.

Basic elements of safety policy, which can respond to the demands after TMI and Chernobyl can be expressed as follows [3]:

a. Enhanced safety margins in order to slow down the accident sequences progression, to increase the potential for recovery actions (grace period) and the inherent stability and power self-limitation

b. Delay the radioactive release, in order to avoid early evacuation, and related provisions in the Emergency Plans as periodic drills.

c. *Limit the integrated radioactivity release during an accident so* that even temporary relocation needs beyond a short distance from the plant can be avoided and land uses can be maintained to the maximum extent.

d. Reduce the role of operating personnel, the most critical link in the chain

e. Simplify the plant design, in particular through the adoption of passive system

f. Strengthen the containment

3. Design aspects

Since the public has a deep-rooted fear of radiation the prime goal should be efforts that directly or indirectly lead to a limitation of the environmental impact. To reach this goal the design has to provide a balance between core protection and core-damage mitigation. This is in line with the defence in depth principle, i.e. the use of various layers of multiple, diverse and complimentary means that help to ensure the safety. Defence in depth can be characterized as a safety and design logic where the layers are categorized as prevention, protection, mitigation and emergency planning.

The PIUS design maintains the defence in depth by addressing all these four categories.

For a new reactor design it comes natural to investigate if a trade-off between prevention and mitigation would be a better way to fulfil the safety goal listed above. The same reasoning has been expressed by U.S. NRC [4], where it is stated:

"... The advanced reactor concepts shall maintain the "defence in depth" concept, consideration may be given to the unique safety characteristics of the advanced plants. Some trade-off between prevention and mitigation is acceptable. ..."

The PIUS design is based on current LWR technology but makes a shift in emphasis from mitigation features to highly reliable prevention and protection features. Inherent and passive features are employed in a novel design to prevent core degradation in all credible events.

Mitigation is also provided, but, considering the unique safety characteristics in PIUS, in a different way (trade-off) than in current LWRs.

4. Containment

4.1 General

Mitigation contains a number of different components and the containment is the most important.

The containment is traditionally an important part of the defence in depth. The TMI accident, causing limited environmental consequences, and the Chernobyl accident, causing major environmental consequences, confirmed the great value of a containment. To restore the public confidence in nuclear power after Chernobyl we consider that a containment is a must. As a consequence the trade-off between preventive and mitigative action cannot be taken too far, i.e. the enhanced prevention in the PIUS design cannot be fully credited in the mitigation efforts.

The containment philosophy applied in PIUS is based on multiple physical barriers:

1. The UO_2-fuel pellets. Ceramic material with high melting point, lowered linear heat load and high retention capacity for radioactive materials

2. The fuel cladding

3. The pressure retaining parts of the primary system, i.e. Concrete Pressure Vessel, primary piping and Steam Generators

4. The Containment

5. The Containment Isolation Valves (in pipes penetrating the Containment)

The strength of the third barrier is enhanced in PIUS by the employment of the PCRV prestressed concrete pressure vessel. The PCRV makes it possible to provide a sufficiently large quantity of water directly available to the core. The decay heat is removed via the reactor pool and a fully passive system to the environment [1, 5]. This arrangement of the PIUS will reduce the risk of a core melt to an extremely unlikely event. In addition to this the low rated core, the large amount of water, and the PCRV will result in enhanced performance of the following components of mitigation:

- Physical phenomena
  - Fission product hold-up
  - Fission product plate-out
  - Fission product decay
- Long response time

4.2 Design philosophy for the PIUS containment

The general safety philosophy for PIUS is that, considering all defence in depth provisions as a whole, the Committed Effective Dose Equivalent (EDE) shall, for all credible events, be below the level where no health effect to the surrounding population is expected without the need for evacuation.

The basic design philosophy for the PIUS containment is the same as for current LWRs. There are, due to the unique characteristics of PIUS, some essential differences which have to be regarded in the containment design:

- Source term

  The amount of radioactive matter, which can be released to the containment after a pipe break, LOCA, is significantly lower for PIUS than for a current LWR, as the core integrity in PIUS is protected by passive, self-protective means.

- Decay heat removal, DHR

  Following the initial blow-down of the primary system after a LOCA, there will be no significant release of steam or water and hence no long-term pressurization of the containment. The long-term DHR is always ensured by the fully passive system, containing pool coolers, piping and the dry cooling towers. The initial release of steam is condensed on the walls, structures and components inside the containment and in the condensation pools.

- Core coverage

  Both for a LOCA inside as well as outside the containment, the core will be covered with water and the decay heat will be removed by the passive DHR system.

These three special design features of PIUS will ease the requirements on the containment.

The cladding temperature will be very low after a LOCA and the amount of hydrogen generated by metal-water reaction thus insignificant.

During the long-term phase after a LOCA there will be no release of steam or water to the containment as the reactor pool will be kept below 100°C Hence, no dedicated containment spray and venting is required.

The inherent/passive safety features make PIUS more "forgiving", i.e. independent of operator action. A "grace period" of one week was arbitrary chosen as a design criteria.

All these design features comply with the overall safety goal: enhanced safety by inherent/passive means.

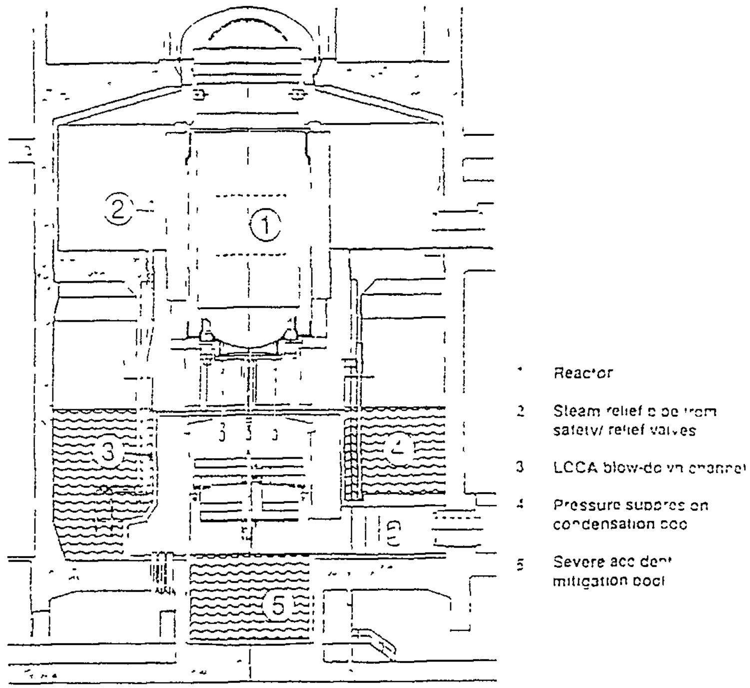

Figure 1 BWR 90 Reactor Containment

4.3 Design basis for the PIUS containment

ABB Atom has a long experience from design and construction of containments for the eleven BWRs built by ABB Atom. These containments are all of the pressure suppression type, using prestressed concrete and an embedded carbon steel liner, see figure 1.

All the Swedish reactors, BWRs and PWRs, have during the 80ies been equipped with filtered-venting systems of ABB Atom's design [6], see figure 2.

Utilizing this vast experience the PIUS containment will employ the pressure suppression principle. The containment contains the primary system and the condensation pool. The containment is a safety-grade system. The containment and the PIUS safety-grade structure is shown in figure 3.

This type of containment has two advantages:

- The condensation pool is a very efficient scrubber, i.e. aerosols and gaseous iodine are separated and kept in the pool water
- The containment pressure will be rapidly reduced to "near atmospheric pressure".

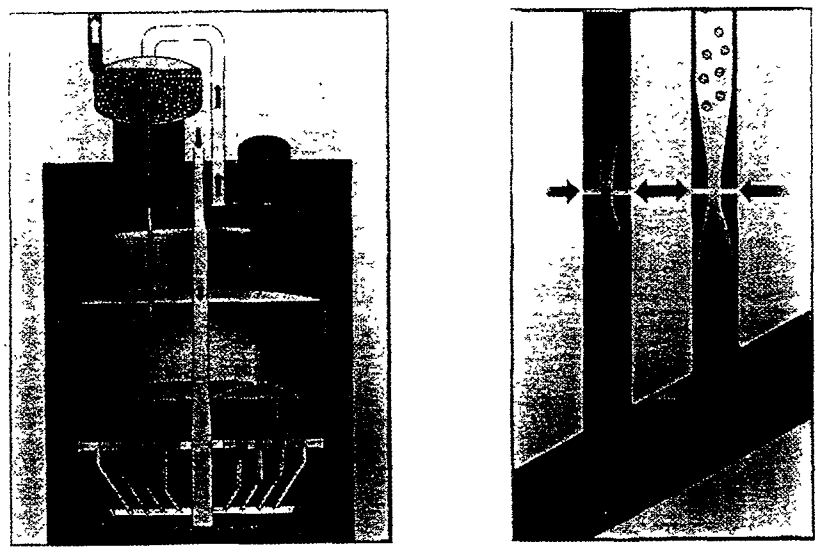

Figure 2 FILTRA MVSS, The Multiventuri Scrubber System

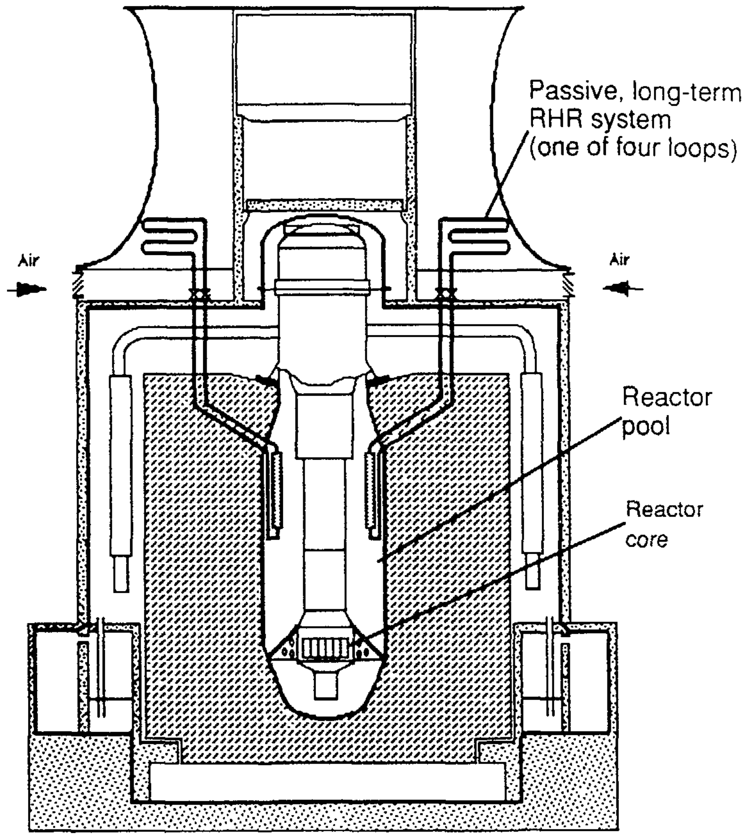

Figure 3 PIUS Safety-grade structures

The basis for the containment design is that the containment shall be able to withstand the design pressure, without exceeding a design leakage rate.

The condensation pool is also utilized as heat sink for the blow-down from the primary system, as well as the secondary side of the steam generators.

The relatively large containment volume and the pressure suppression principle result in a low design pressure, 2.8 bars absolute (after a pipe rupture).

The objective for the design leak rate is 0 5 % per day in weight of the containment free volume at the peak pressure of the DBA (the worst accident)

There are a number of external events to be considered such as

- Earthquakes
- Extreme winds and water levels
- External fire and explosion
- Transport accident
- Lightning
- Aircraft hazard
- Sabotage

The external events and the specific design requirements, to be taken into account for the containment designs, are to a large extent site related

An external event will not cause a pipe break within the containment and these are generally assumed not to be combined

The two external events that, beside the pipe break, create an envelope for the containment design are earthquakes and aircraft hazards A containment, designed to resist 0 3 g SSE for earthquakes and a crash of a jet liner, will be a very rugged structure, that can comply with other external events

The PIUS containment is made of reinforced concrete provided with an internal steel liner, see figure 4 and 5

A general basis for the structural design is ASME Code, Section III, Div 2, Article CC-3000 for the concrete structures and ASME Code, Section III, Div 1, Article NE-3000 for the pressure retaining steel structures

The main load-bearing parts of the containments are

- the base mat
- two cylindrical walls
- the roof structure (including the wall and slab structure of the reactor service room)
- the containment steel dome
- internal concrete and steel structures

The primary leak-tight barrier is an inside surface liner attached to the concrete wall and slab structures The liner is made of stainless steel in the pool compartments, otherwise of carbon steel

The PCRV is anchored to the base mat by the outermost vertical row of prestressing cables

The low elevation of the core and the sturdy PCRV facilitate the aseismic design

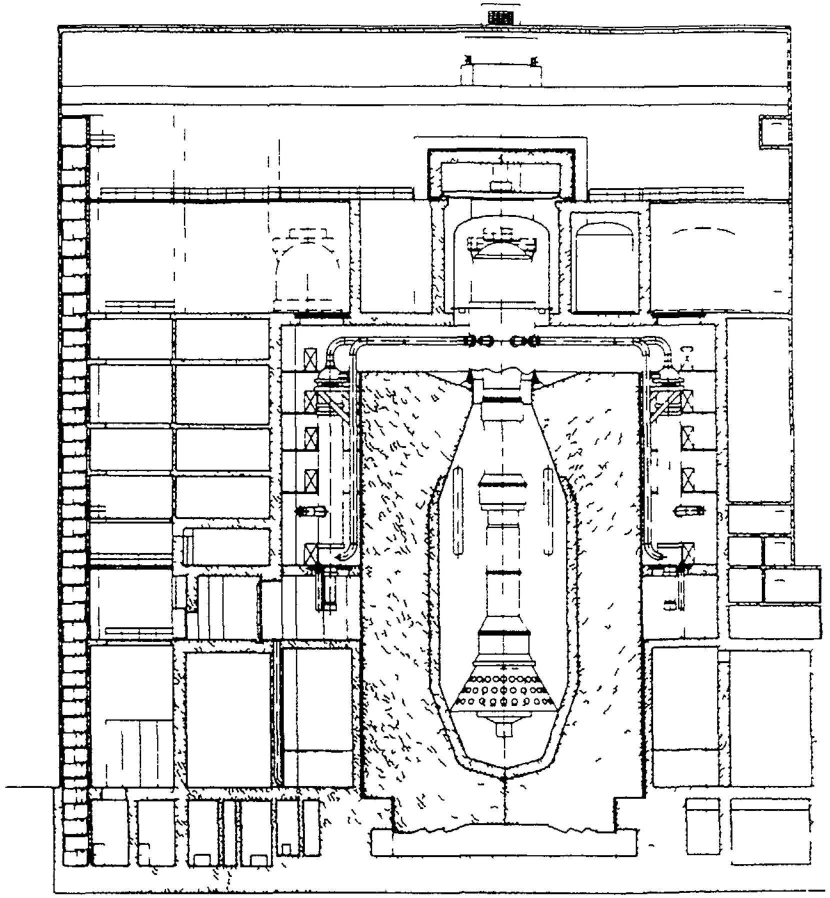

Figure 4 PIUS Containment

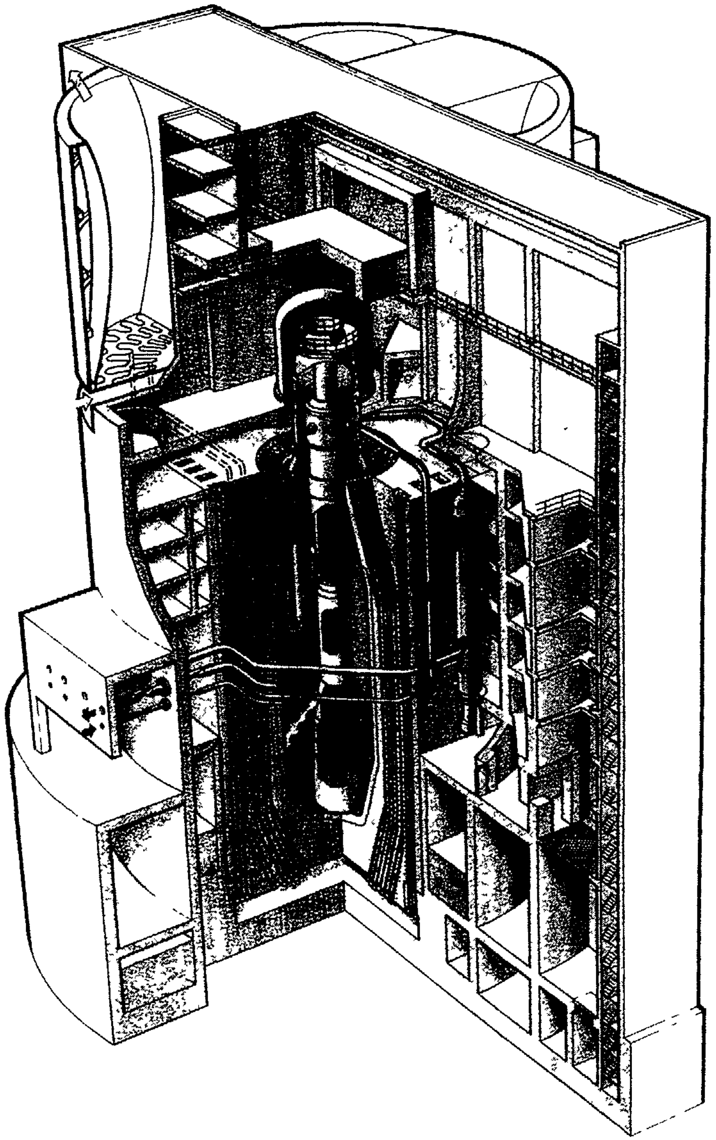

Figure 5 PIUS

The general arrangement of containment isolation valves in PIUS is applied in a similar manner as in current LWR plants. The most important difference is that even if a pipe rupture occurs in a PIUS plant and the break is not isolated by means of the isolation valve, the core integrity will still be protected by the inherent passive, self-protective features.

The Leak-Before-Break concept is adapted for the design of pipe-whip restraints inside the containment.

Leak testing of the containment will be carried out in accordance with LWR practice (Appendix J to US 10CFR Part 50).

Conclusions

Defence in depth, a long standing fundamental principle of reactor safety, results in the concept that multiple barrier should be provided to ensure against any significant release of radioactivity. This calls for a design which provides a balance between prevention and mitigation. The PIUS design maintains the defence in depth principle. Inherent and passive safety features are employed to prevent core damage in all credible events. Although the rugged PCRV is a strong and very reliable physical barrier, PIUS is, in line with the defence in depth principle, equipped with a containment.

The unique inherent/passive features will ease the requirements on the containment. The containment is, however, designed to cope with pipe ruptures and external events, such as earthquake, flooding, fire, sabotage.

The PIUS containment is made of reinforced concrete and utilize the pressure supression principle.

The design basis for the PIUS containment is that, considering all defence in depth provisions as a whole, the Committed Effective Dose Equivalent (EDE) shall, for all credible events, be below the level where no evacuation planning is needed.

REFERENCES

1. L Nilsson, T Pedersen
   Advanced Reactor Design Philosophy and Application - ways and means to prevent core melt
   Progress in Development and Design Aspects of Advanced Water-Cooled Reactors, Rome, 9-12 September, 1991

2. C Sundqvist, K Hannerz, L Nilsson, T Pedersen
   Concept Status and Marketing Strategy of ABB PIUS and SECURE Reactors
   Jahrestagung on Nuclear Technology '91, Bonn, May 1991

3. L Noviello
Evolution of the Design Criteria presently considered in Italy
Energia Nucleare, Anno 7/N2/Maggio-Settembre 1990

4. Key Licensing Issues associated with DOE sponsored Advanced Reactor Designs
SECY-88-203, July 15, 1988

5. C Pind
Some Natural Convection Phenomena in the PIUS Reactor Design
Jahrestagung on Nuclear Technology '91, Bonn, May 1991

6. L T Tirén
Filtered Venting of Swedish Reactor Containments,
U.N. Conference on the Promotion of Uses of Nuclear Energy,
Geneva, March 23-April 10, 1987 (A/Conf.108/C.2/Inf.1)

NEW RESEARCH TRENDS FOR THE STRUCTURAL ASSESSMENT OF CONCRETE CONTAINMENT STRUCTURES UNDER EXTREME LOAD CONDITIONS WITH EMPHASIS ON CONSTITUTIVE LAWS OF CONCRETE

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Abstract

The new National Energy Plan approved on August 1988 by the Italian Government, has defined the general guide-lines for the next generation reactor designs considered for a potential future use in Italy. In terms of safety the new design criteria require a step forward focused to reduce the consequences of an accident, rather than the probabilities of occurrence. In case of core damage and release of fission products into the containment and the environment, the environmental impact has to be minimized and reduced below a level so low that no effects to the health of the surrounding population are expected, without the need of evacuation.
Based on the requirements of this new safety policy, the concrete containment structures become of extreme importance as integrity and leaktightness must be assured for the entire duration of all conceivable accidents. To design and assess the level of technical performance of concrete containment structures new advanced numerical modelling techniques are required. The computational methods have to account for the cracking of concrete, in the areas of high stress concentration, as well as the deformation limits at the interface between concrete and steel components.

A research program is being developed jointly by ENEL and ISMES to define, test and consolidate a complete analysis procedure capable of supporting the selection of the reference containment and the related structural optimization. The preliminary results of the research, primarily oriented to the methodical aspects, are presented and discussed in the paper. The future developments of

the research are also shortly considered, with special emphasis to the open problems that still affect the analysis of the long term structural behaviour of concrete for thick walled structures.

INTRODUCTION

On August 1988, a new National Energy Plan was approved by the Italian Government. The Plan, according to the December 1987 request of the Italian Parliament, for studies and research activities on a new generation of nuclear power plants, points out the general guide-lines for design of nuclear reactors with the potential of being built in Italy. In line with the most advanced trends at the international level, the Plan calls for a step forward in safety, to be achieved by avoiding any radical departure from the existing well proven technologies. As a result of a great number of technical, political, economic and social issues, emerged in Italy in the last few years, the basis of the new safety policy is focused to reduce the consequences of an accident rather than the probabilities of occurrence. To comply with the above policy in the Italian context, the Plan requires the possible environmental impact from all conceivable accidents be minimized and reduced below any significant effect, with reference to :

- population short and long term doses;
- emergency actions, and in particular the evacuation of the population;
- relocation of population groups.

The technical performance of the new advanced nuclear plants must be such that, should a core damage occur and fission products be released into the containment and the environment, the Committed Effective Dose Equivalent (EDE) is limited to levels so low that no effects to the health of the surrounding population are expected, without the need of evacuation.

In the development of advanced nuclear plants, based on simplified systems and components having inherent safety characteristics, the containment plays then a very important role. The emphasis is to design simpler and safer containment structures, whose integrity and leaktightness are assured for the entire duration of all conceivable accidents. To pursue this design requirement, the plant Designer can nowadays take advantage of the new available numerical modelling techniques, capable of providing reliable results in terms of global and local behaviour of the structure under investigation.

For concrete containment structures, the level of technical performance is significantly affected by the cracking of material, in the areas of high stress concentration, as well as by the deformation pattern at the interface between concrete and steel components, such as liners and hatch reinforcements. To evaluate the leaktightness of the containment for each Plant Condition and the ultimate structural capability of the proposed design configurations, the calculation methods have to take into account the rheology of materials for each state of stress and strain, and rely to an experimental characterization and qualification which is not trivial. The need of meeting the new safety design requirements for containment structures by means of state of the art calculation methods has oriented the Italian National Power Board (ENEL) to initiate in the 1990 a new research program specifically addressed to the structural analysis issues of reinforced concrete with or without pre-stressing. The final scope of the work is to define, test and consolidate a complete analysis procedure capable of supporting the initial phase of definition of the best preferable containment configuration, as well as of assisting the plant Designer in the final optimization process which necessarily includes the reference to the sequence of construction and pre-loading.

The research activity has started by recovering the previous experience ENEL and ISMES collected in the middle seventies when numerical and experimental analyses were carried out for the sake of providing the required support for the possible use of Pre-stressed Concrete Pressure Vessels (PCPV). After that, what is now in progress is a careful definition of the analysis targets in terms of solution strategies and comprehension and qualification of the existing tools that seem best suited to support the design and verification of the actual concrete containment structures. As later on discussed and presented, most of the activity is up to now dedicated to cover the methodical aspects of the simulation problems, with special emphasis to the definition and application of solution procedures to be used for the analysis of short term loadings with moderate temperature effects. Further developments of the research program are planned to overtake the current limits of analysis and being able to face the specific problems that may affect the design of thick-walled containment structures, never built up to now.

BACKGROUND FROM THE SEVENTIES

The most recent studies and research activities carried out in the field of structural assessment of concrete containments have a common key reference in the international round robin pretext analyses executed on the 1·6 scale reinforced concrete containment tested by SANDIA [1]. The results of the benchmark, attended by many renown organizations operating in the design of nuclear structures, provide a well comprehensive overview of the numerical methods available to tackle the problem of interest and has certainly to be considered as a starting point of any new attempt to define and qualify the target solution procedure. However, a parallel analysis of the previous research activity executed by ENEL and ISMES in the early seventies has been also considered of great interest to highlight the changes needed in the research strategies and purposes. The previous research activity was carried out by ENEL and ISMES to support the conceptual design of PCPV structures, for High Temperature (HTR) and Boiling Water (BWR) reactors [2,3]. At that time, the capabilities of numerical simulation were at the beginning of their evolution, with tremendous constraints in terms of available computational resources and programs. Computers having less memory and speed than a small PC currently distributed at the supermarket, were accessed by tens of engineers that were, at the same time, programmers of the in house developed software and structural analysts.

The research program was consequently based and carried out assuming that:

- numerical simulation is used to assist in the initial conceptual design which assumes the structure remains within the linear elastic range for the applied loading conditions;

- structural assessment makes use of a strength criterion which points out the unsafe regions (safety factor < 1) and provides a distribution of the safety factor for the safe regions;

- reduced scale tests are used to validate the results of computations and evaluate the ultimate structural capability of the pressure vessels.

An example of the PCPV containments considered in the previous research is shown in Figure 1. Figure 2 provides a close-up of the steel reinforcement distribution and Finite Elements (FE) model used for computations. The structural response of the same PCPV containment, derived experimentally from a hydro-test, is reported in Figure 3. The response of the structure is clearly

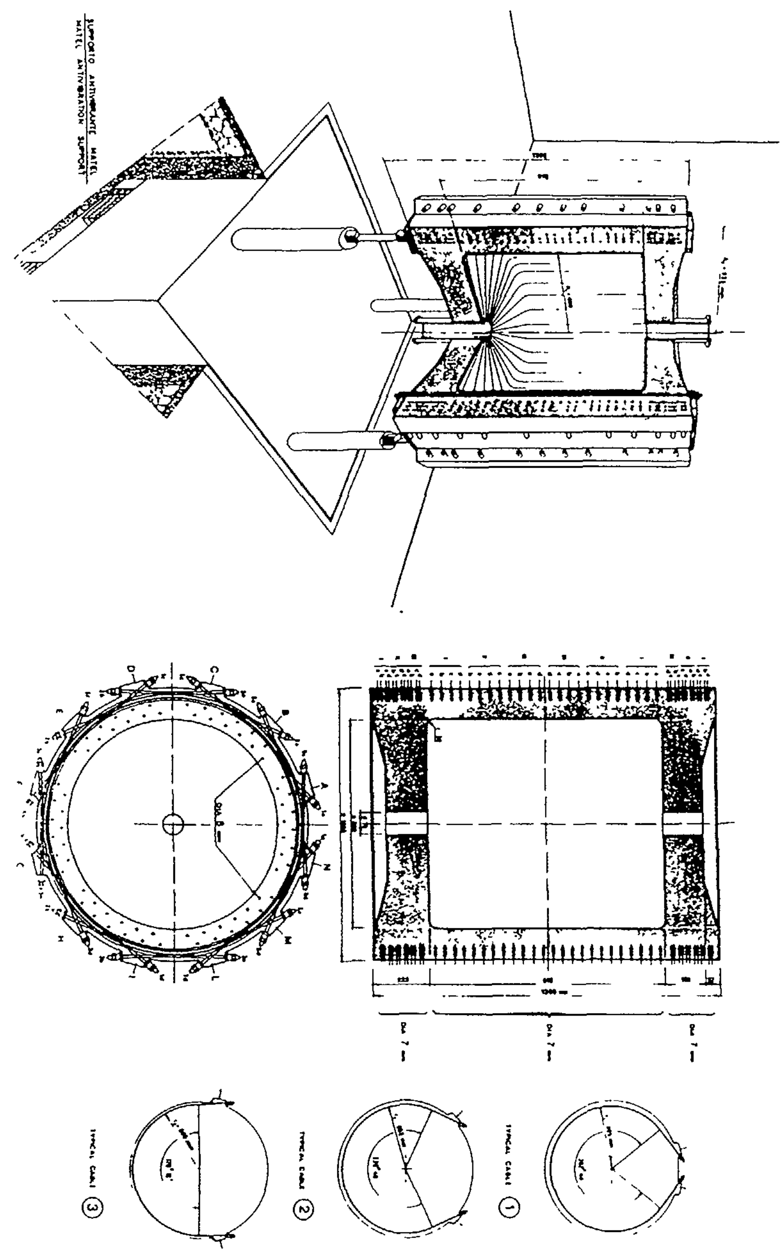

Fig 1 Model of a PCPV structure tested in the previous research activity [3]

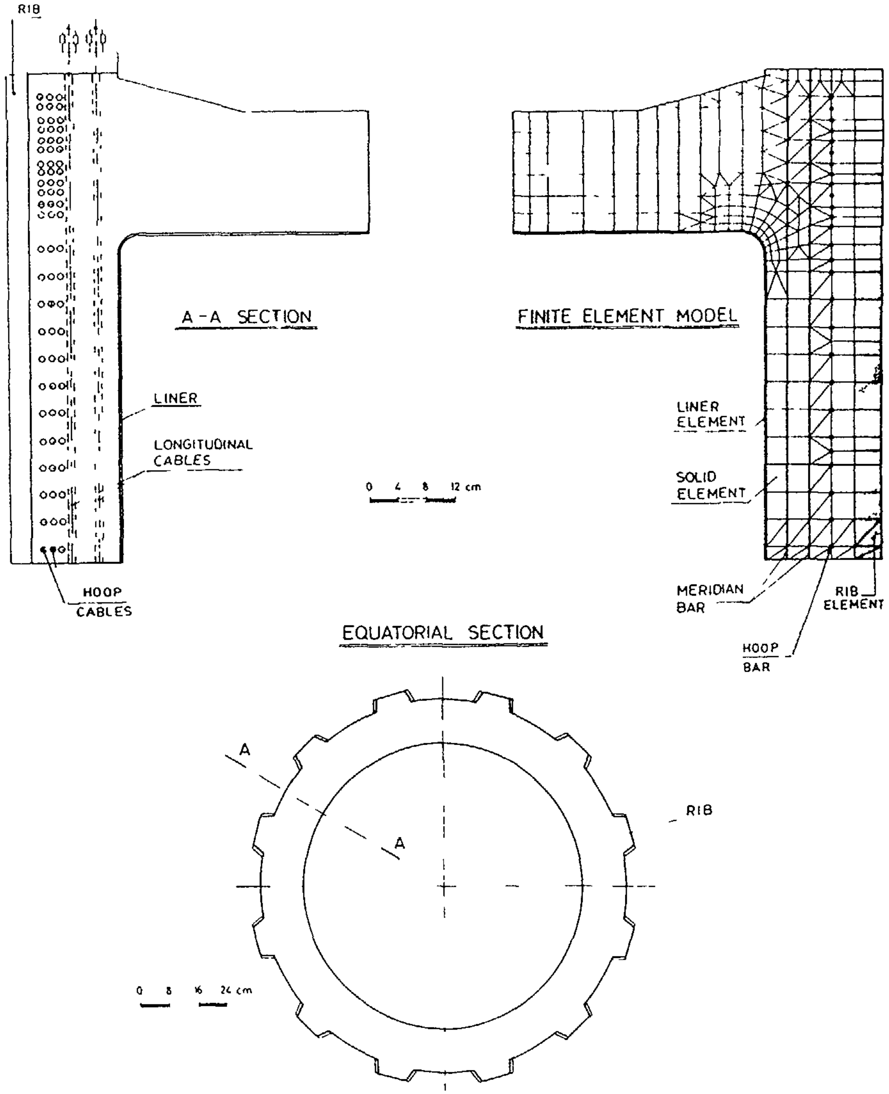

Fig.2 Reinforcement detail and FE model [3]

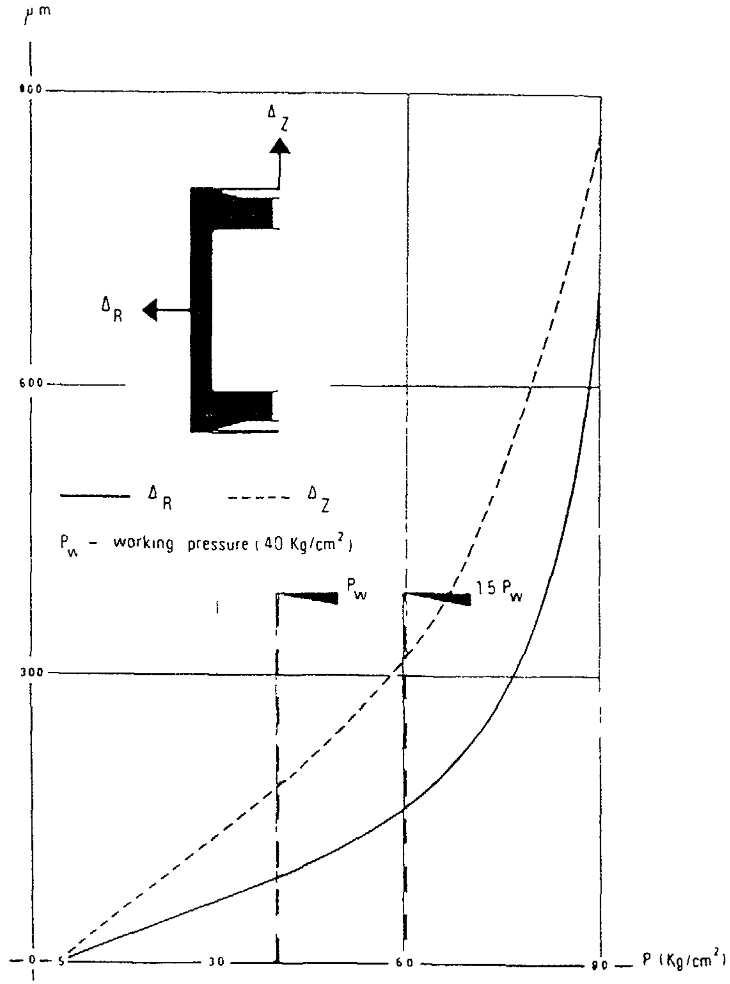

Fig 3 Pressure vs radial and axial deformation obtained from testing [3]

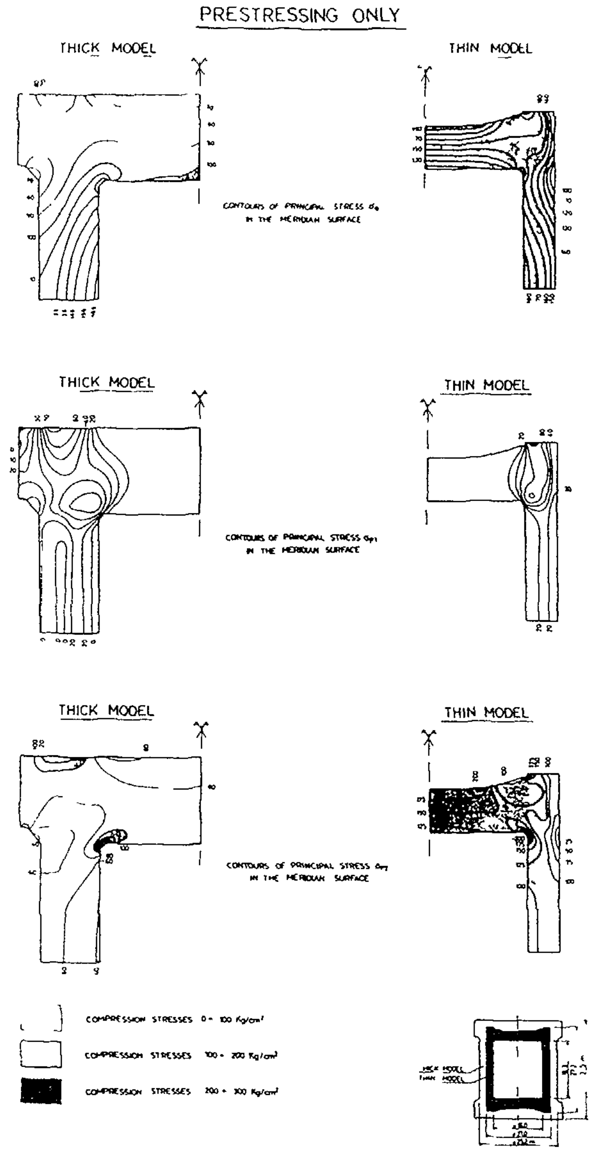

Fig 4 Comparison between thick and thin models based an stress results form FE elastic analyses [2]

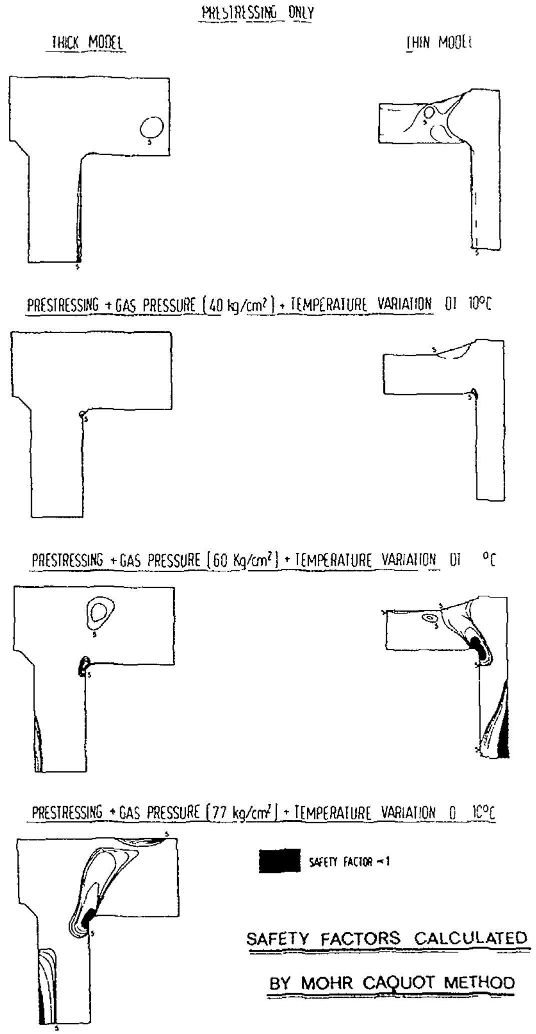

Fig 5 Safety factor evaluation based on the results of an elastic stress analysis [2]

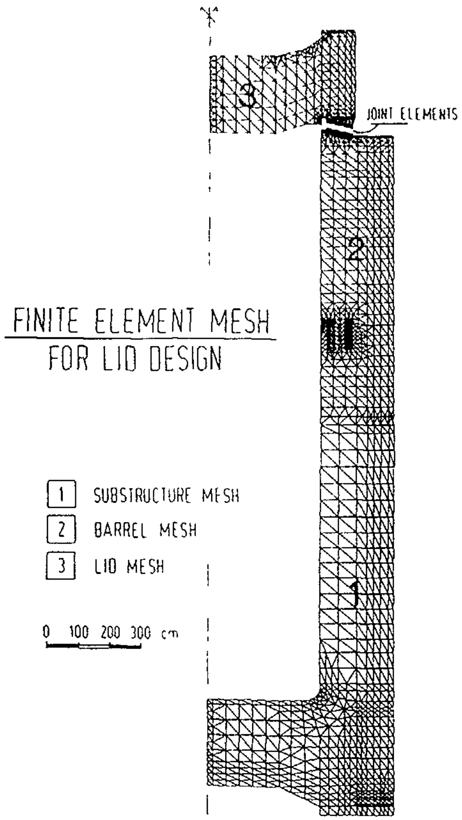

Fig.6 Global FE model of a PCPV for BWR with removable lid [4]

only linear up to values of the applied pressure that slightly exceed the working pressure. Beyond these values the structure shows a highly non-linear behaviour, which includes the effect of all irreversible phenomena that occur in and between the various materials.

The type of computations feasible in the seventies provided important parameters to support the designer when assessing the relative level of technical performance of alternate design configurations, as in the case of the thick and thin walled models of the HTR containment shown in Figure 4. However, the

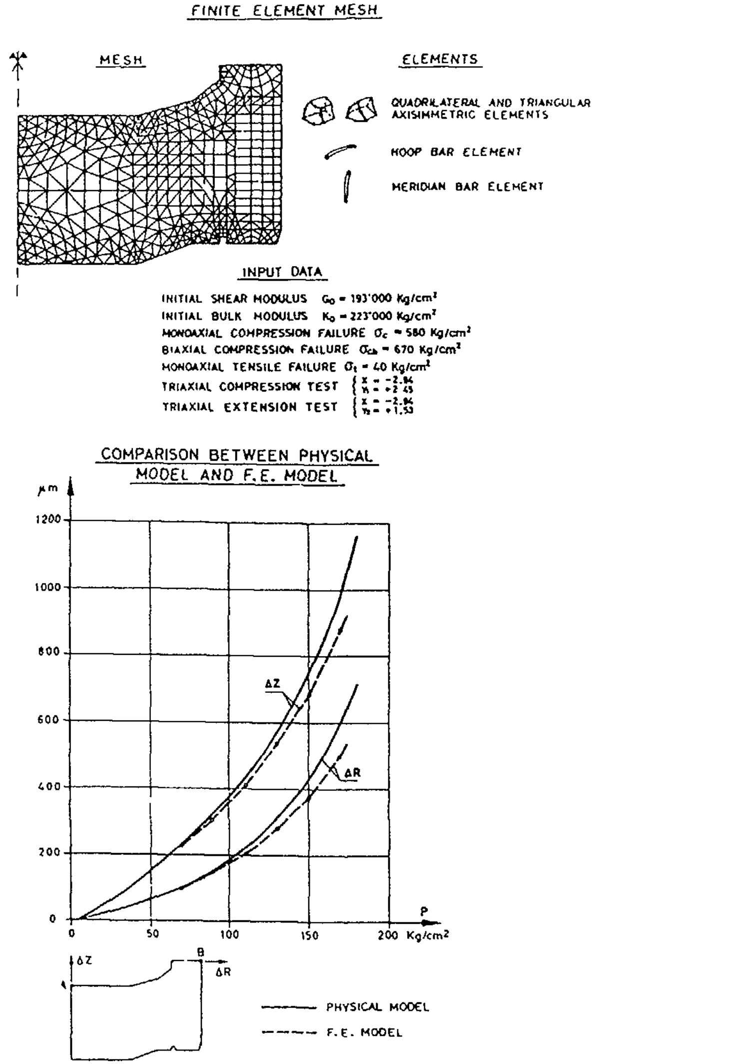

Fig.7 Detailed FE model of the lid and comparison between experimental and numerical results computed using non linear elastic costitutive behaviour for concrete [4]

available results neither could provide any measure of the actual damage produced in the unsafe regions, for example shown in Figure 5, nor could take into account the effect of stress and strain redistribution which follows each local failure To overcome the above mentioned problems an attempt was made in the previous research activity, by implementing a non-linear elastic constitutive model for concrete, capable of considering the basic effects of cracking and crushing post failure [4] The model was successfully applied to foresee the global behaviour of the lid of the BWR shown in Figures 6 and 7 Although containing the same basic concepts nowadays available in the most sophisticated elasto-plastic models, the model was still unsatisfactory to provide a measure of the local damage in the areas of failure and furthermore was implemented in a solution procedure hard to handle up to the required convergence levels

In spite of that, the whole work done in the seventies is certainly to be considered of great interest and success, expecially if evaluated considering the pioneering scenario of development Starting from this point the new research strategies can be now well defined, taking into account that there are

- wide experiences available in the constitutive modelling of complicated materials,
- no more costraints in computer's power,
- additional design requirements to be satisfied

From the above considerations the new research activities have been first of all pointed at the definition and qualification of a reliable computational procedure capable of providing accurate results for each normal or extreme load condition in terms of global and local behaviour

NEW TREND OF THE RESEARCH

In addition to code assessment based on safety factors, the standards, as in ASME code [5] recommend the analysis of the ultimate capacity of the structures, for precise probabilistic risk assessment, but also as a global confirmation of the conventional assumptions in terms of failure modes

During the last five years, many research programs, see for example [6], both numerical and experimental, have tried to define a global strategy to fix some uncertainties pertaining to the highly non linear behaviour of the typical containments The first consequence of the new safety criteria points to leakage as the dominant failure mode for concrete containments, partially modifying previous theories stressing the importance of the stability of the overall structure This is particularly true for pre-stressed containments owing to the reduced ductility of the overall structure The containment leakage is essentially tied to the steel liner tearing, occuring usually near penetrations or stiffness discontinuities, and requires refined local effects analyses Some data from one of the samples studied in [6] highlight the problem, showing that

a) in a cylindrical reinforced containment the design pressure is 40 Psi,

b) the concrete hoop cracking initiates at 30 Psi,

c) the liner yelds at 70 Psi

d) the first leakage occurs at 112 Psi

e) the global failure of the structure occurs between 120 Psi and 135 Psi, when liner exhibits hoop strain of 7 8%,

The above results indicate that leakage occurs for values of pressure which are almost 2 times the design pressure The local failure, that has nowadays become one of the major concerns, is then attained without reaching the structural collapse that occurs at three times the design pressure The possibility of assessing the leakage failure mode, in the design process, requires the analysis of the local aspects related to concrete - steel interaction, mainly governed by punch shear effects induced by the anchoring studs The liner is in fact subjected to a displacement controlled process which induces local perturbation to the global stress field due to the opening of the various cracks even below the design pressure Furthermore this phenomenon is often complicated by the complexity of the geometry in the most critical areas The difficulties in modelling this local problems have suggested simplified criteria for predicting leakage, based on local strain values calculated as a function of pressure and geometry, using global 2D analysis According to the results of the most recent researches, some well known procedures for the evaluation of leakage limit state have been developed and assessed The approach proposed in [6,7] uses a relationship between global and local results which is expressed by means of three magnification factors related to stress intensifications, triaxiality of stress field, and strain localization next to stiffness discontinuities All these factors have to be initially computed for each specific stress concentration area, following a combined numerical and experimental analysis on

3D local models. Based on the above described phenomena and results of most recent researches, the development of a strategy for the assessment of a new type of containment structures requires the availability of:

- reliable 2D numerical models capable to provide global stress fields and to determine critical areas. Such models have to consider the liner-concrete interaction, as well as the actual rebar configuration. Concrete constitutive law has to reproduce fully developed cracking fields even corresponding to stress redistributions;

- results from either numerical or experimental analysis leading to the required estimate of local strain intensifications, strictly tied to geometry and loading configurations.

The above considerations furtherly substantiate the need of completing the qualification of numerical modelling techniques, as well as of supporting the results with an adequate experimental activity which is essential to estimate the accuracy of the whole sequence of analysis. As later on discussed, the need for experimental assessment is also of great importance for a confident characterization of material data.

NUMERICAL QUALIFICATION IN PROGRESS

The principal phenomenological characteristics of concrete mechanical behaviour that are believed should be described by a comprehensive concrete constitutive model are:

- departure from linearity in uniaxial and biaxial tests occurring long before the maximum stress, with occurrence of irreversible strain;

- degradation of the elastic stiffness with progressing inelastic deformation;

- strain dependency of post peak behaviour on the confinement stress;

- generation of damage, characterized locally by anisotropy and by a unilateral constraint on displacements across the crack;

- localized character of post-failure deformation and non-uniqueness of its description.

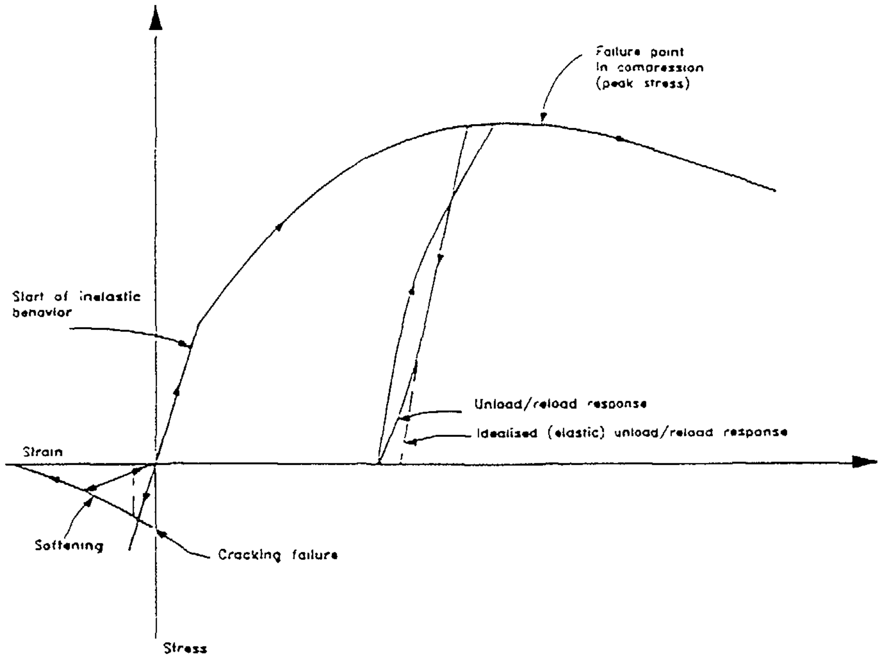

Fig.8 Uniaxial behaviour of plain concrete predicted by ABAQUS 4.9 model

Formerly the approach used as a first attempt to model the concrete behaviour was within the theory of elasticity [4]. However, it is only with the development of plasticity, and strain hardening plasticity models, that it was possible to start taking into account many of the characteristics listed above.

Todays constitutive models for concrete implemented in codes as ABAQUS [8] and DIANA [9] may be considered as a reasonable compromise between the sophistication of the most advanced models proposed in the scientific literature [10, 11], which take into account many of the above requirements for a comprehensive model for concrete, and the design oriented approaches.

These codes have been selected as the most advanced in the field of geomechanical analysis in the non linear domain, respect to element library, constitutive models implemented and robustness of the integration procedures.

The qualification activity is based on the evaluation of the predicted response under static isothermal conditions with respect to some well documented medium and large scale tests on reinforced concrete structures with emphasis on crack prediction [12, 13]

A typical uniaxial stress strain response that can be modelled by ABAQUS and DIANA is presented in Figure 8 It may be observed how features like the difference in strength values in compression and in tension of concrete is reproduced as well as the non linear behaviour in the compression range, the softening response beyond the crack limit, the degradation of the elastic stiffness under unloading after a crack has been generated

For reinforced concrete, bars, layers and prestressed cables can be modelled under the hypothesis of full bonding, using an elasto-plastic constitutive model for reinforcement steel

As an example of the combined effects the model can reproduce, in Figure 9 the radial pressure-displacement response computed on a plane strain section of the SANDIA model is presented It may be noticed that the result, here obtained with the DIANA code, is able to account for the decrease in stiffness of the model due to concrete cracking Further evidenced changes in the structural stiffness are liner yielding and hoop rebar yielding

For the purpose of assessing the leaktightness of the containment, one of the major problem is the correct modelling of cracks initiation and propagation To that scope, both codes identify crack direction as that perpendicular to the maximum tensile stress and smeare over the finite element the crack opening effects on the global deformation of the structure

This approach to crack modelling seems satisfactory in predictive evaluation of the global response of structural components performance, as it allows the analysis of the evolution of the stress-strain response from the uncracked state to the cracked one

Prediction of crack distribution based on the smeared crack approach seems instead only sufficient as an indicator of the actual crack pattern in the structures to finalize more refined local evaluations of the deformation patterns, nearby the liner or other steel components

In Figure 10 the crack distribution obtained with DIANA on a large scale test performed by Kong [12] on a deep beam of reinforced concrete with web opening is

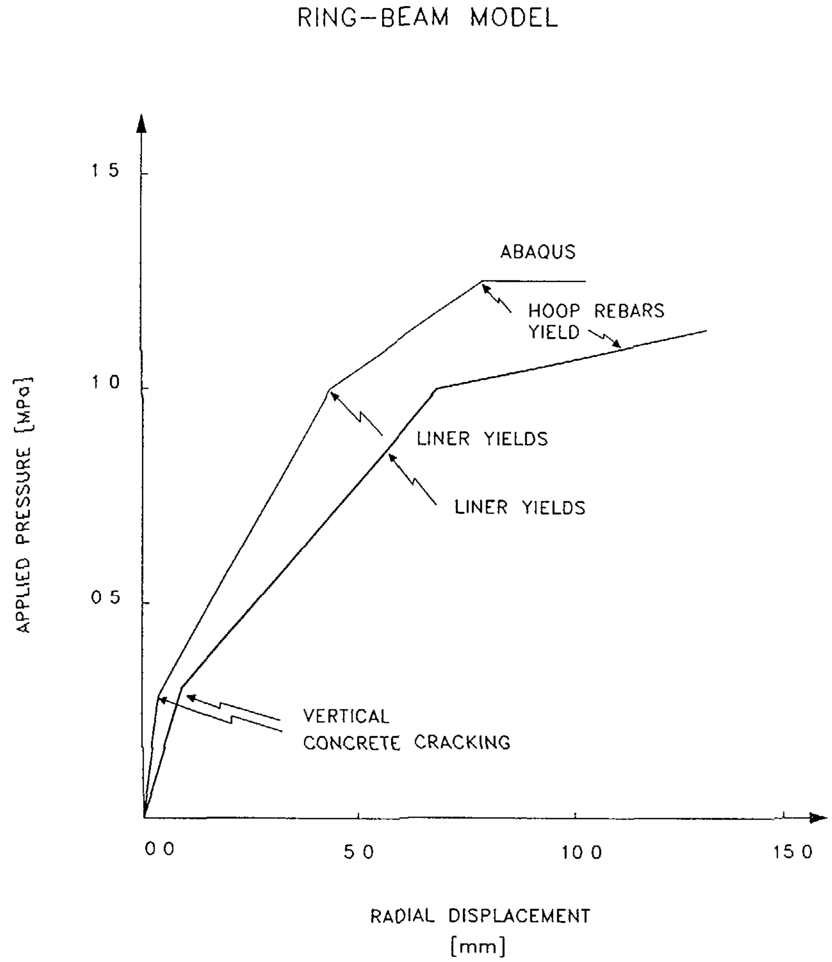

Fig 9 Radial displacement versus applied pressure from the ring beam model [1] as predicted by DIANA (bold line) and ABQUS 4 5 171 [1]

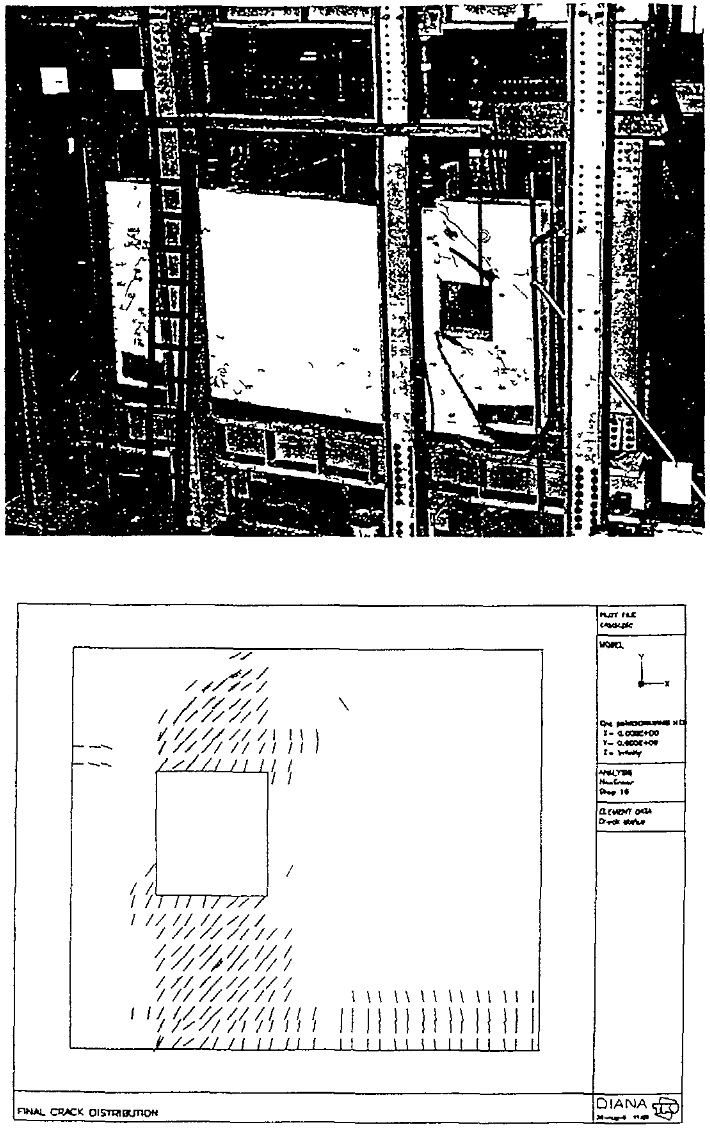

Fig 10 Predicted (DIANA 4 0) and experimental crack pattern of the reinforced deep beam with web opening tested by Kong [12]

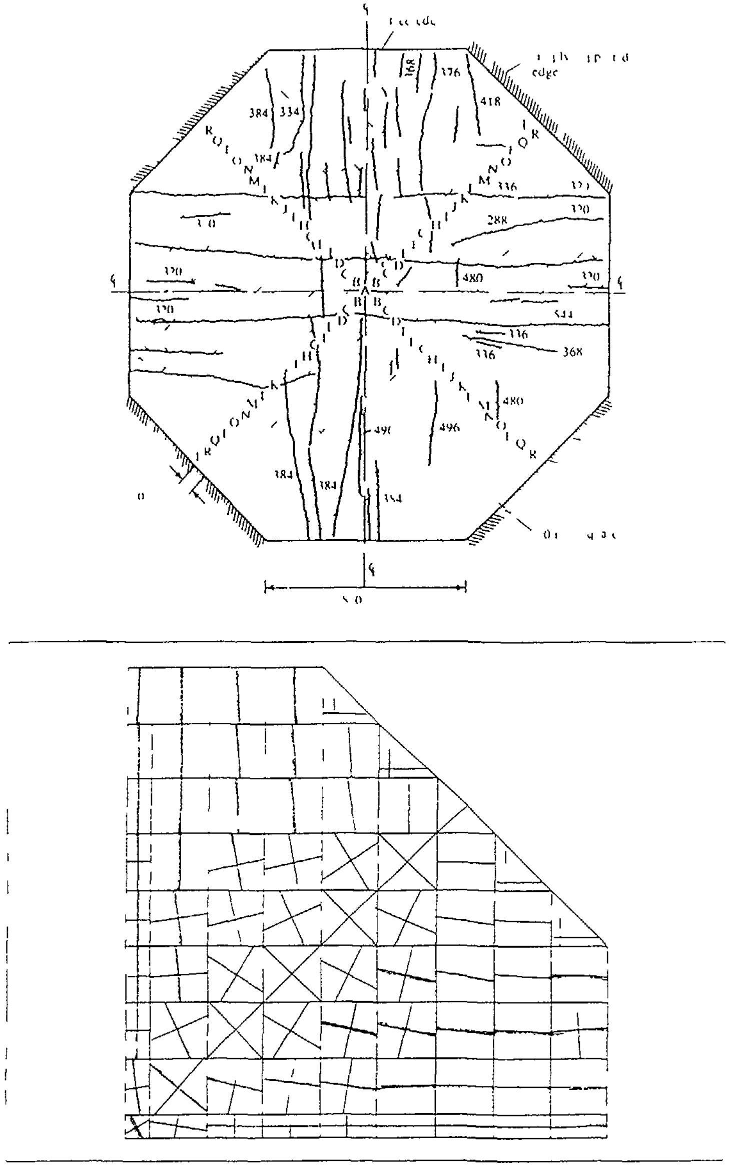

Fig 11 Predicted crack pattern at ultimate load for the prestressed slab tested by Bangash [13] Results obtained with ABAQUS 4 9

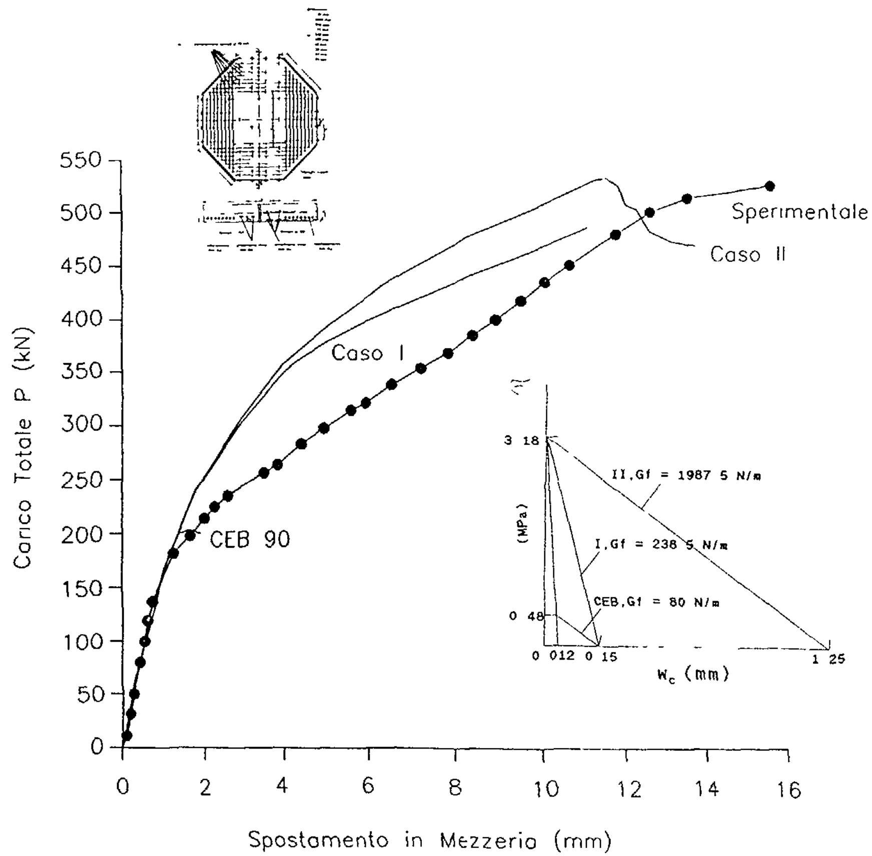

Fig 12 Predicted and experimental response of the center of the prestressed slab tested by Bangash [13] under the applied pressure Results obtained with ABAQUS 4 9

presented. The crack zone compares well with the experimental results, also pointing out the two major cracks that are evidenced displaying the cracks that are actually open at the final level of loading.

In the case of the reinforced prestressed concrete slab reproducing the top cap of a pressure vessel tested by Bangash [13], the crack pattern obtained at the

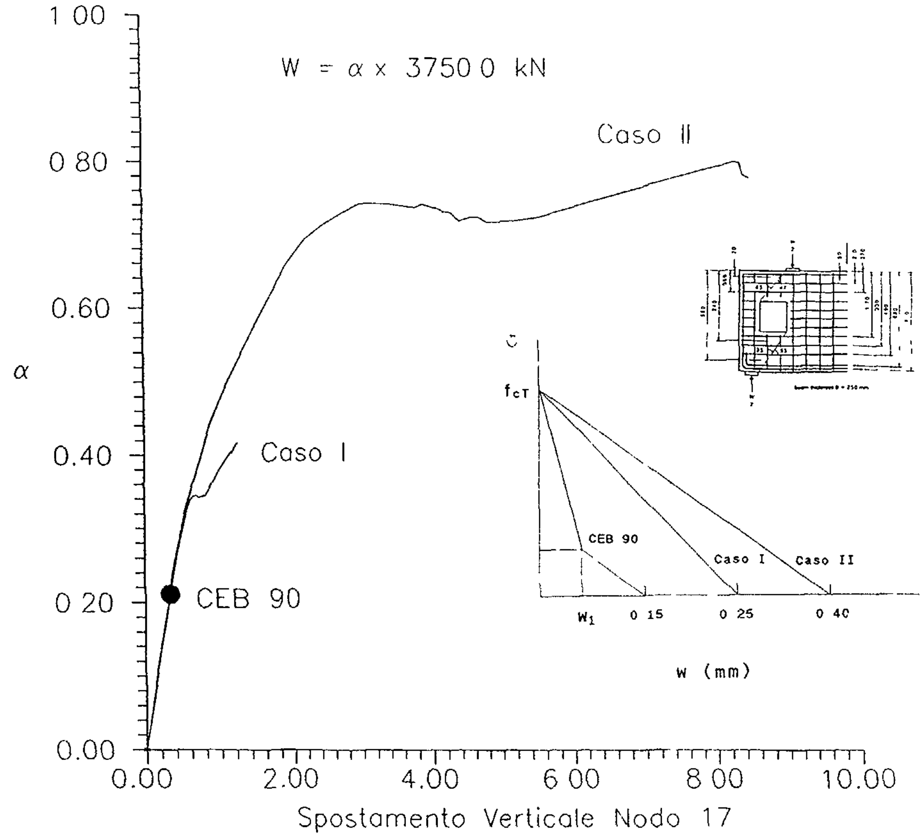

Fig.13 Predicted load deflection at the midspan of the deep beam tested by Kong [12], for different fracture energy values of concrete Results obtained with ABAQUS 4 9

ultimate load on the lower face of the slab with ABAQUS (Figure 11) shows the presence of two major crack families, which were observed also in the test.

The most important consideration arising from the first phase of qualification is that the selection of the properties influencing the post crack behaviour, such as fracture energy and crack band width, is the key issue for finalizing a procedure of evaluation of reinforced concrete structures with these models.

There is in fact an agreement about fracture energy values for a given concrete to be a material property and selection of the relevant values may be done for most structures with some confidence.

The meaning of fracture energy is instead far less established for reinforced structures, where, if full bonding is considered between reinforcement and concrete, fracture energy should be considered a structural property rather than a material property. In fact, due to the hypothesis of full bonding between the two materials implemented in the codes under qualification, fracture energy of concrete should include the steel-concrete interaction (tension stiffening effect) to correctly reproduce the structural deformability.

In the case of the slab shown before, the load deflection curve is depicted in Figure 12 obtained with different assumptions on fracture energy values, starting from that of plain concrete, which follows CEB-FIP model code 90 recommendations [14]. The sensitivity of the result is evident. However lower what is to be is the possible scattering of the results that, depending on the type of application, can be also quite large as in the case of the results of the deep beam with a web opening shown in Figure 13.

For these reasons, the qualification of concrete models is planned in combination with a set of experimental tests on representatives samples that should allow the final setting up of the analysis procedure.

FUTURE RESEARCH DEVELOPMENT

The research activity in progress is expected to lead to the definition and qualification of a complete solution procedure to be used to assess the technical performance of concrete containment structures according to the new safety requirements.

At the current level of development of the next generation reactor designs considered for a potential future use in Italy, the procedure will be initially applied to carry out the sensitivity analysis needed to help in establishing a preference priority among the various alternate configurations that are currently proposed. For the development of the sensitivity analysis on thin-walled concrete containments the main reference loading will be the inside pressure.

After the first screening and the definition of the reference containment configuration, the solution procedure will be applied to study the local effect

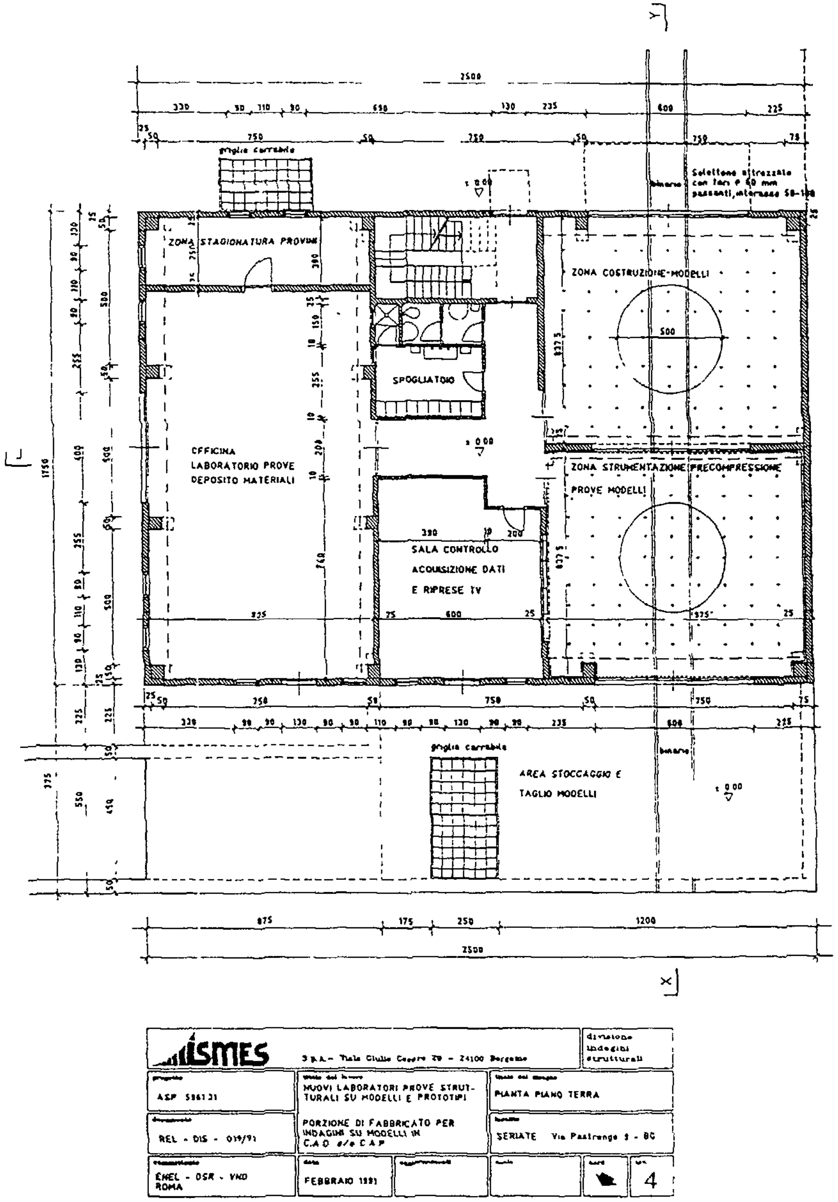

Fig 14 Layout of the testing section on reinforced and prestressed concrete structures in the structural laboratories at ISMES.

of main penetrations and geometry discontinuities For carrying out these analyses the global model is expected to be replaced by local refined models that allow a more detailed simulation of the interface between concrete and steel components The structural analyses of the penetrations will be developed to assist in the definition of the best layout which simplifies the isolation system and reduces the potential leak paths

Future developments of the research program are also planned beyond the current analysis requirements of thin-walled concrete containments In view of the possible future interest toward the use of heavier massive structures both numerical and experimental analysis capabilities are required to account for additional phenomena which are intrinsic characteristics of concrete, such as shrinkage and dehydratation heat diffusion after casting, creep in the long period under constant loading, decrease of strength and toughness with time, effects of the thermal loads on the long term behaviour

The development and qualification of numerical procedures to be used from these simulations have already started and are based on the use of state of the art achievements in constitutive modelling However, the results of the research activity on computational methods will be additionally supported by an experimental work to be executed on reduced scale models of the actual structures of interest The experimental activity will be developed in a new laboratory which is now in preparation at ISMES (Figure 14) The laboratory's layout is foreseen to house the construction and testing of models having an outside diameter and a height up to 6 and 8 meters, respectively

With the coupling of numerical and experimental testing capabilities the research program is expected to lead to the complete solution for all global and local problems of concern, also for heavy thick-walled structures, including the simulation of construction phases and loading sequences

The research activity under development will then make available an integrated design and testing environment where to assess and optimize huge structures never built up to now, and with a potential field of application that can be of great interest also outside the nuclear engineering constructions

REFERENCES

[1] Clauss D B (1987) Round Robin Pretext Analysis of a 1 16 Scale Reinforced Concrete Containment Model Subject to Static Internal pressurization NUREG/CR 4913, SAND87-0891

[2] Scotto F L , (1974) Triaxial State of Stress in "Tiny Walled PCPV for HTR Comparison with a conventional "Thick Solution", IABSE Seminar on Concrete Structure Subjected to Triaxial Stresses, ISMES, Bergamo

[3] Fanelli M , Riccioni R , Robutti G (1974) Finite Element Analysis of Prestressed Concrete Pressure Vessels", IABSE Seminar on Concrete Structure Subjected to Triaxial Stresses, ISMES, Bergamo

[4] Riccioni R , Robutti G , Dal Bo C , Scotto F L , (1977) Finite Element and Physical Model Analysis of a Removable Lid of a PCPV for BWR, SMIRT Paper H6/S

[5] ASME, Section 3rd, Division 2, App B, General requirements for structural evaluation of concrete containments

[6] EPRI NP-6260-M Criteria and guide-lines for ι redicting concrete containment leakage, April 1989

[7] SAND90-0019, Insights into the behaviour of nuclear power plant containments during severe accidents

[8] Hibbitt Karlsson and Sorensen ABAQUS Manuals Version 4-9

[9] DIANA Analysis B U DIANA Manuals Version 4 0

[10] Ortiz M (1985) A constitutive theory for the inelastic behaviour of concrete, Mech of Materials, 4 67

[11] Kachanov M C (1982) On micro crack model of rock inelasticity Mech of Materials, 1,19

[12] Kong F K , et al (1978) Structural idealization for deep beams with web opening, further evidence Magazine of Concrete Research Vol 30, No 103

[13] Bangash N Y M (1989) Concrete and Concrete Structures, Numerical Modelling and Applications, Elsevier

[14] CEB-FIP Model Code (1990) Material Properties, Chapter 2

DEVELOPMENT AND QUALIFICATION OF THE FUMO CODE FOR THE CONTAINMENT SYSTEM SIMULATION OF ADVANCED LWRs

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Abstract

In the framework of a cooperation between ENEL-CRTN and the Dipartimento di Costruzioni Meccaniche e Nucleari (DCMN) of Pisa University, a best estimate thermal hydraulic computer code, named FUMO, for multi compartment containment systems has been developed and assessed It is a lumped parameter code and contains all the models necessary to realistically describe the physical phenomena occurring in a passive containment system, following a DBA or a severe accident

A number of different containment tests have been analyzed to assess the code response and to evaluate the influence of the different models implemented in the code on the calculation results The tests were selected on the basis of their focus on the most relevant physical phenomena for the new containment systems

Preliminary analyses of passive LWR containment systems were also carried out In particular, a Large LOCA sequence in the AP 600 nuclear reactor was studied, taking, as reference data, a published sequence calculated by Westinghouse

1 - INTRODUCTION

Thermal hydraulic codes used for evaluating safety margins in nuclear reactor containment systems need to be carefully assessed against experimental tests in order to establish their ability to describe the main physical phenomena In particular, in order to realistically evaluate the hydrogen distribution and the behaviour of the fission products inside the containment system a correct modelling of natural convection processes is required, these phenomena are potentially very important with respect to deposition rates, transport and leakage of aerosols, i e to the source term definition [1]

In the framework of a cooperation between ENEL CRTN and DCMN, an improved version of the FUMO code [2] [3] [4] for containment analysis was developed to achieve a better simulation of a wide range of phenomena, including natural circulation, and to evaluate their influence on the distribution of gases and aerosols during severe accidents

The code was submitted to a systematic assessment process, based on the validation of some models on separate effect tests, followed by post calculation of integral containment tests This work has recently been completed by performing best estimate (BE) analysis to simulate the HDR E11 2 and T31 5 tests [5] and the BMC FIPLOC F2 experiment [6] These tests were selected because they are representative of the thermal-hydraulic conditions in the containment during SA and they allow the models implemented in the code for natural circulation and hydrogen distribution to be accurately assessed

2 - FUMO CODE DESCRIPTION

FUMO is a BE thermal hydraulic code for multicompartment systems It is a lumped parameter code and contains the necessary models to realistically describe physical phenomena occurring in a containment system following a DBA or an SA

The containment can be described by a set of up to 100 volumes, 200 junctions, 30 of which can have time dependent areas, and 100 heat transfer structures The thermodynamic conditions in each control volume are calculated according to the following input options

1 a stagnant homogeneous mixture of steam, liquid water and air
2 a two regions model, consisting of a homogeneous gas/steam mixture region and a liquid pool region The pool region may or may not be in thermal equilibrium with the atmosphere one In the gaseous region it is possible to solve the mass balance for seven gases Different deentrainment models are available for evaluating the water mass condensed in the atmosphere region and its falling velocity They allow the drop diameter and the atmosphere turbulence and composition to be taken into account

The control volumes are connected together through junctions The flow area can be time dependent It is also possible to model pressure dependent junctions which are opened by a trip on the pressure difference between the two joined volumes Junctions with the inlet or/and the outlet submersed by the water, present in the source or/and in the receiver node, can also be simulated This model can be used to analyse the pressure suppression containment systems or the water management in the LWRs with gravity driven safety systems

The junction flow rate is calculated as follows

1 The Bernouilli equation is solved for the evaluation of water flow rate
2 The momentum balance equation is solved for transients characterized by rapid thermofluid dynamic variations The flow rate is evaluated using one of the following models

Moody critical flow,

homogeneous inertial flow,

orifice polytropic flow

The method requires small time steps and therefore should not be used in a long term transient because of the considerable CPU time needed

3 In order to study the long term phase of an accident a model is used which assumes the pressure differences between control volumes to be negligible, the flow rates are evaluated in such a way that the pressure in the connected compartments is balanced This option turns FUMO into a fast running code and a large number of control volumes even if they are connected in complex ways, can be used

The natural circulation model [3] is able to predict the buoyancy driven flow, both in serial volumes and in complex loops Furthermore a simplified model that simulates the hydrogen vertical distribution in a control node is implemented in the code The model is utilized to modify the hydrogen mass percentage of junction flow rates, depending on the junction heights in the source volumes

Particular care is given in the code to the flexibility in modelling the heat sinks and the heat transfer coefficients The temperature distribution inside a structure is evaluated by solving the Fourier equation by a finite difference method or with a coarse-mesh method, specifically developed by DCMN [7]

Several models, covering a wide spectrum of physical processes that may occur within the containment, were introduced in the code These models include processes like hydrogen combustion heat and mass transfer (convection and/or condensation on cold structures), corium water interaction, as well as BWR specific features (suppression pool and safety relief valve discharge) A detailed model for the description of thermal transient of a metallic structure cooled by a passive external spray was also developed [8] (briefly described in section 4) As the code does not consider in vessel processes, it must rely upon separate analysis performed with a primary system code, to provide sources of mass and energy to the containment To overcome this limit, a simplified model, which can simulate a transient in the primary system, is under development This model will allow the simultaneous analysis of the relevant processes both in the primary and in containment systems

3 - FUMO CODE ASSESSMENT AND QUALIFICATION

The FUMO code was tested to verify its reliability in simulating the abnormal transients that can take place in a multicompartment containment system The code validation was based on the comparison between the values of the most important thermal hydraulic variables measured during experimental tests and the values of the same variables calculated by the code

3.1 Review of the past assessment work

First assessment activities of FUMO include some numerical benchmark problems, in order to evaluate the degree of approximation of its models A series of separate effect tests were carried out in a facility installed at the DCMN, in order to obtain a better understanding of the influence on the containment behaviour of some important phenomena, such as the deentrainment of droplets and the effect of the high turbulence, induced by the blowdown jet expansion, on the heat transfer to the thermal structures

In order to obtain an in depth validation of the code the qualification program continued with participation in some international exercises based on tests carried out in apparatuses with various volumetric scales Particular attention was paid to the validation of the models which describe the long term phase of an incidental sequence During this phase the following tests were simulated

- the OECD CSNI Benchmark Problem 2 [2] proposed in the framework of the PWG 2 Task group on Ex Vessel Thermal hydraulics, in particular to quantify the pressurization and the combustible gas distribution in a containment system during a core melt down accident in a LWR
- the OECD CSNI ISP 16 [2], carried out in the HDR Facility at the end of 1982, for the analysis of the first 1200 s of a large LOCA sequence, and the second phase of the same test, named HDR V44, for the study of the temperature stratification,
- the pre and post test calculations [9] of the LACE tests 4 and 6, performed at the CSTF (Hanford W USA)

3 2 - Recent assessment work

FUMO assessment has recently focused on some complex phenomena that may occur in the containment of new simplified plants, such as buoyancy driven natural convection flows, condensation of steam on structures and long term cooling of the plant by passive features This was achieved again by comparing experimental data (from HDR and BMC facilities) and code calculation results

3 2.1 - The HDR facility

The HDR containment (11300 m^3) consists of a steel shell 60 m in height and 20 m in diameter [10] This steel shell is surrounded by a secondary concrete shell which is separated from the former by an annular gap The containment is subdivided into 69 subcompartments of different shapes and sizes by concrete partition walls Two spray systems internal and external to the steel liner can be activated during the transient

Two HDR tests [5] [11] were analyzed, simulating the containment behaviour after the rupture of a large diameter steam pipe (HDR T31 5) and a small break LOCA (Test E11 2) respectively Their timing characteristics allow a good assessment of the code particularly for the long term analysis

Analysis of the HDR T31.5 test

A seven volume nodalization was used for the analysis with FUMO of the T31 5 test [5] This test is characterized by a relevant stratification over the height of the containment, due to the heat exchange processes and to the evolution of the convective flows A stable temperature distribution can be identified, with a measured temperature differences of roughly 50 °C between the upper and lower parts of the containment Stratification decreases gradually during the last phase of the test, with the exception of a localized region around the blowdown node The good evaluation of temperature trends, obtained with FUMO, is a necessary condition for a correct simulation of the hydrogen distribution

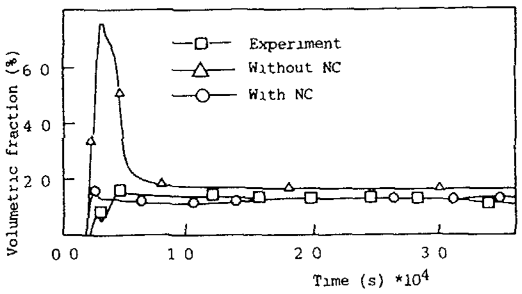

Fig 1 HDR T31 5 - Hydrogen concentration in the blow-down volume

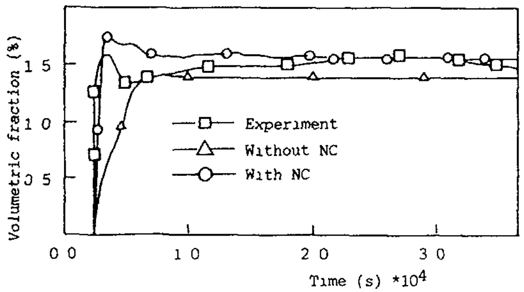

Fig 2 HDR T31 5 - Hydrogen concentration in the central volume

Hydrogen volumetric concentrations are compared with the experimental ones in Figs 1 and 2 The hydrogen injection starts at 35 min into the transient and lasts for 12 min Most of the released hydrogen leaves the blowdown volume, along the staircases, and reaches the upper dome region The zones below the injection position are also affected, due to diffusion and convective flows Figure 1 shows the results for the blowdown volume, the agreement between experimental and calculated values is quite good, especially in the long term The natural circulation and the hydrogen distribution models are fundamental for this goal In the same plots also the results from a calculation performed without activating the model for the natural circulation are shown In this case FUMO calculates an unrealistic hydrogen peak, due to the lack of convective flows

For the other control volumes the influence of the natural circulation is not so relevant In a volume located at the same height of the blow-down node (Fig 2) there is a very good agreement in the results, with the exception of the first few minutes after the hydrogen injection The differences are probably due to the technique used for hydrogen concentration measurement, which requires some time to initiate the catalytic process in the hydrogen transducers

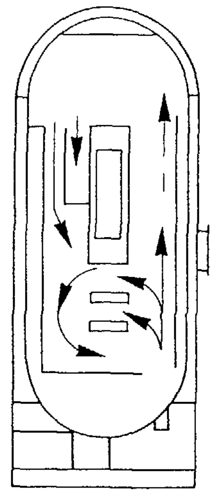

Fig 3 Natural circulation flows inside HDR containment

Analysis of the HDR E11.2 test

The experimental E11 2 conditions simulate, in the first phase (740 min), a SB-LOCA (Fig 3) During the following phase a steam and hydrogen/helium mixture is injected at an elevation of +23 5 m, near the staircase This mixture, being lighter than the air present in the containment rises up to the dome The air is thus pushed through the spiral stair into the lower region of the HDR containment, like in a plug flow A very large amount of air is displaced in the first 100 min, so that 50% of the air has been expelled from the dome at this time The injected steam strongly accumulates in volumes above the break location, but it does not flow down in a substantial quantity The temperature transients make this impressively clear, with a peak temperature of 130 °C in the region above +25 m, whilst only 65 °C was measured at +15 m At the bottom of the containment, below 10 m, the temperatures are almost constant between 22-26 °C

A detailed nodalization was set up for FUMO, based on 15 control volumes, 85 heat structures and 64 junctions The compartments inside the steel shell were nodalized using 12 control nodes two nodes were used for the gap and one for simulating the external back-up volume of the gas sampling sensors

Figure 4 shows the comparison between the measured and the predicted total pressure in the containment up to 1300 min The very good prediction of the long term phase shows the capability of FUMO to simulate the overall thermal-hydraulic transient In the figure the data obtained without using the natural circulation model are also presented In this case, the steam concentration in the dome is underestimated this leads to too low a condensation and therefore to too a high pressure trend This result highlights the effect of natural circulation flows on the thermal hydraulic transient and in particular on the steam concentration and distribution over the containment volumes Furthermore without the simulation of convective flows, the hydrogen volumetric concentration is overestimated in the nodes located near the staircase (Fig 5), while it is underestimated in other compartments Using the natural circulation model a better prediction of this fundamental variable is obtained

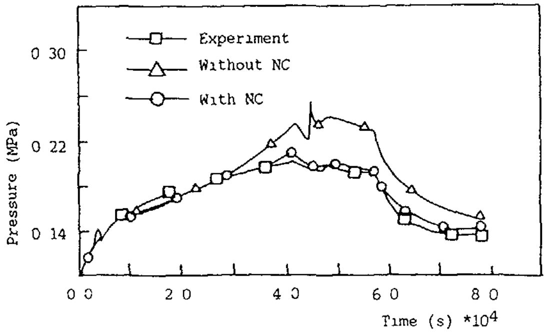

Fig 4 HDR E11 2 - Total pressure

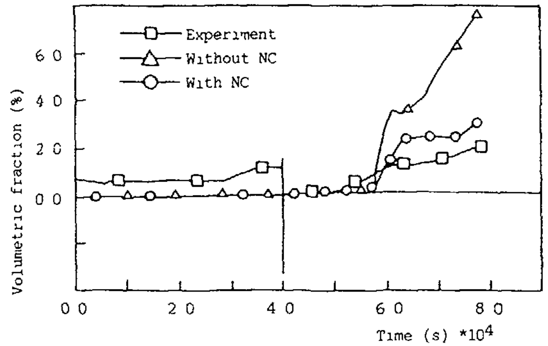

Fig 5 HDR E11 2 - Hydrogen concentration in central volumes

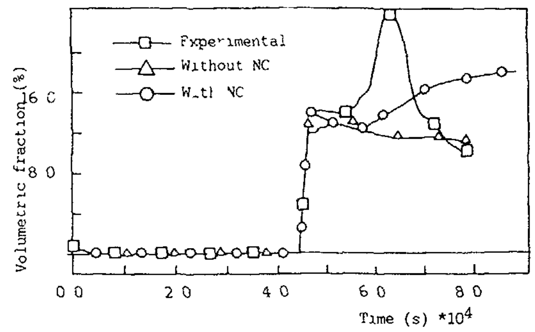

Fig 6 HDR E11 2 Hydrogen concentration in upper dome

At the end of the E11 2 test, the external spray system for cooling the spherical region of the steel liner is activated As a consequence, a sudden increase in the steam condensation rate takes place inside the dome, followed by an increase in the local hydrogen volumetric concentrations in the same volume, due to the "suction" effect from the lower nodes of the facility Both FUMO (Fig 6) and other participants in the international exercise [12] show a general underestimation of this local phenomenon

3.2.2 - The Battelle Frankfurt Model Containment facility (BMC)

The BMC, with a total volume of 640 m^3, was originally designed for DBA analysis and later was also used for hydrogen distribution experiments and for the DEMONA aerosols depletion tests [6] The facility is made of reinforced concrete and its internal volume is subdivided into several compartments, sizes and locations of the vent openings between the individual compartments can be adjusted to the specific test conditions The height of the facility is reduced by a factor 4/5 compared with a real containment, which might have some influence on the natural circulation behaviour

Analysis of the FIPLOC F2 test

The FIPLOC-F2 test is focused on the analysis of natural circulation phenomena in a multicompartment containment system During the first phase of the test (48 h) the pressure and the temperature inside the facility are step wise increased by steam injection, until a partial pressure of the vapour of about 2 7 bar is reached In the following phases steam, air and sensible heat are injected into different containment subcompartments, developing convection flows which have a direction and a velocity depending on the injection locations (Fig 7) A detailed nodalization was set up for FUMO analysis 9 control volumes, 45 heat structures and 16 junctions were used

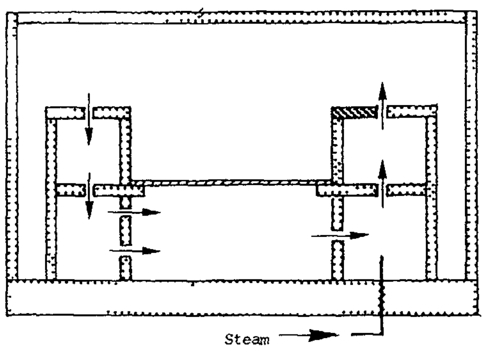

Fig 7 Natural circulation flows in BMC facility

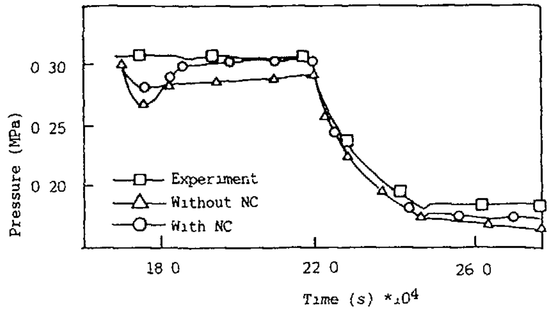

Fig 8 FIPLOC F2 - Total pressure

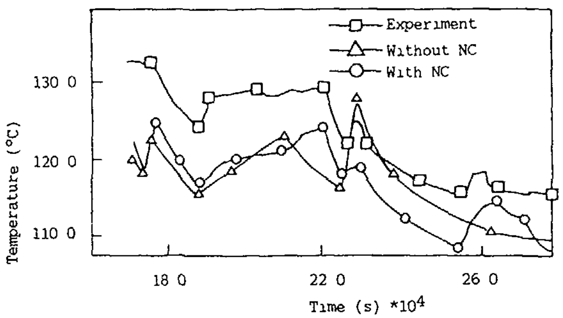

Fig 9 FIPLOC F2 - Temperature

In Figs 8 and 9 the measured and calculated global pressure and temperature with and without the use of natural circulation model are compared The underestimation of pressure data (Fig 8), calculated without using the natural circulation model, can be explained through the prediction of a higher steam concentration in the dome of the BMC facility this induces an overestimation of the condensation on thermal structure

More insight into the problem of a correct prediction of the long term containment condition is obtained from the analysis of temperature trends (Fig 9) The difference between measured and calculated values, of about 8 °C during all the transient, is due to uncertainties of initial value The calculation performed without the use of natural circulation model shows an oscillatory behaviour due to an unrealistic evaluation of heat exchange coefficients

4 - APPLICATION OF FUMO CODE TO INNOVATIVE LWRs

FUMO is being used at present for analysing the containment systems of SBWR and AP600 reactors The nodalization has been completed and tested for both reactors (Figs 10 and 11) For AP600 some results are already available Below they are compared with data available in literature [13], obtained by Westinghouse using an independent methodology

4.1 External spray model

The AP-600 containment design is similar to a dry containment arrangement in terms of size and design pressure The safety functions are devoted to a Passive Containment Cooling System, which is capable of transmitting heat directly from the reactor containment structure to the environment [14]

Containment cooling is performed by a falling liquid film and a counter current natural circulation air flow A monodimensional model is used in FUMO Mass and energy balance equations are solved both for the liquid film and for the air/steam mixture The energy balance takes into account terms due to heat conduction through the film, convective heat transfer between film and air and latent heat carried by the evaporating flow The evaporation rate is evaluated using a correlation derived from the analogy between heat and mass transfer in turbulent forced convection, this is acceptable because of the high Reynolds number which characterizes the mixture flow The solution of the mass balance equation along the height of the containment structure means that the film thickness can be calculated this is necessary for the evaluation of the heat transfer phenomena and also for the verification of its hydro dynamic stability

A minimum flow-rate is required to guarantee the stability of the liquid film, but there is a large spread in the experimental values of this limit, ranging from 0 015 Kg/ms to 6 0 kg/ms for water at room temperature [15] The exact value depends upon the geometry and the nature of the vertical surface the initial distribution of the liquid and the mass transfer between the liquid and the surrounding gas Specific tests in order to investigate this phenomenon will be carried out at the DCMN In FUMO the value of 0 015 Kg/ms is used at present When the liquid film flow rate drops

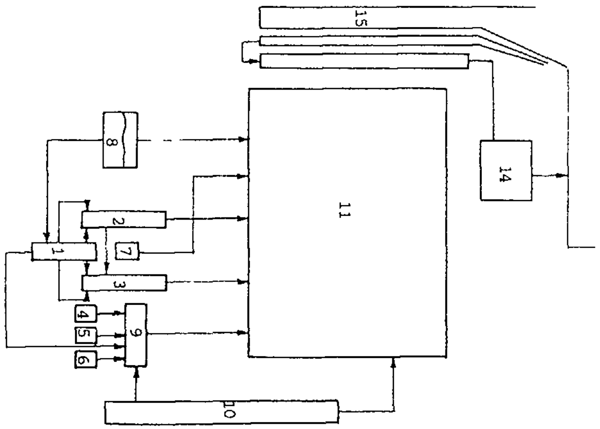

Fig. 10: AP600 nodalization for FUMO code.

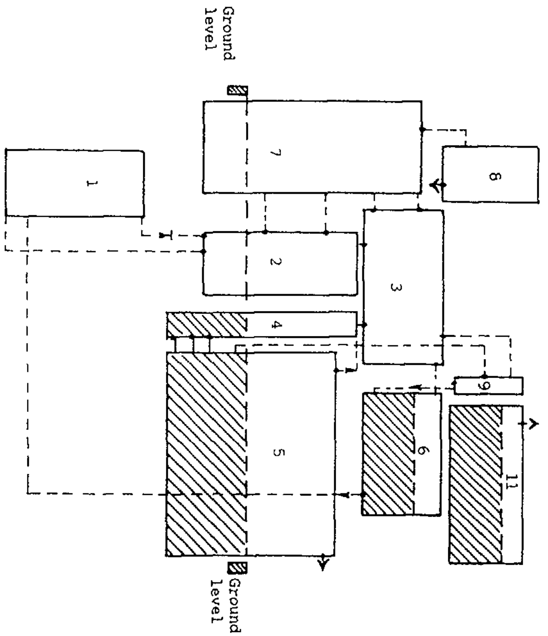

Fig. 11: SBWR nodalization for FUMO code.

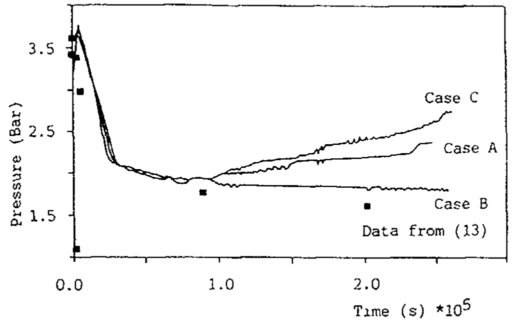

Fig. 12: Sensitivity analysis for Large LOCA in AP600. Pressure inside containment.

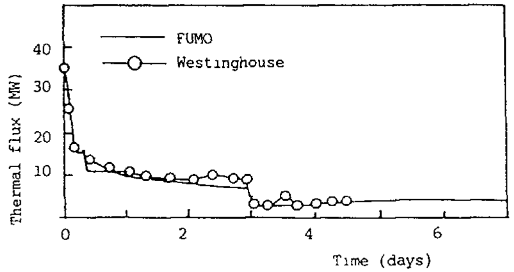

Fig. 13: Thermal flux through liner internal surface.

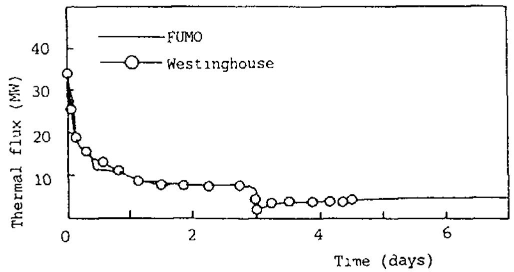

Fig. 14: Thermal flux through liner external surface.

below this value the lower portion of the containment is assumed to be dry and the convective heat transfer coefficient is evaluated on the basis of the air/steam flow rate induced by natural circulation

4.2 AP600 Large LOCA analysis

A preliminary analysis of AP600 containment response to a Large LOCA was performed using FUMO The containment was nodalized with 15 control nodes, 11 inside the steel liner and 4 outside

A reference calculation (case A) was performed without modelling the IRWST pool or injecting steam into the containment during the whole transient with a total enthalpy flow equivalent to the decay power of the reactor

A sensitivity study was also performed to evaluate the effects of the limit for film instability and of the way of providing the blow down table In the second calculation (case B) the instability test described in the previous paragraph was by passed Figure 12 shows that the assumed minimum flow rate value strongly influences the pressure transient inside the containment In fact, the instability condition is reached at about 80000 s, and, starting from this time, the pressure rises significantly

In the third calculation (case C) the IRWST was simulated, and decay power, given by the ANS curve + 20%, was directly added to the pool in the reactor cavity This approach needs a more complex calculation, but it matches the actual situation better, in fact it allows a correct simulation of the water balance inside the containment and of the steam generation in the long term phase, when a pool of saturated water is present The importance of a correct evaluation of the blow-down is evident from Fig 12 The higher steam generation present in case C induces a significantly higher pressurization of the containment

In Figs 13 and 14 the heat flux trends, internal and external to the containment steel wall, are given The results are compared with those obtained by Westinghouse in a similar analysis [13] The good agreement between the results obtained in the two calculations is highlighted, in particular, a good quantitative agreement is present also in the initial phase of the transient, when the heat transfer process is dominated by the external spray action

5 - CONCLUSIONS

The presented results show FUMO's ability to analyze thermal hydraulic transients in the containment system of the current and the new generation of LWRs during a DBA or a SA sequence In particular the results obtained in the assessment process show the validity of the natural circulation model and its importance for a BE analysis of a containment system

Also the new models, related to the ESFs present in the new LWRs, have been tested with satisfactory results A further assessment may be useful to improve the external spray film model especially as far as the stability condition is concerned This work will be done both with calculations for separated effects tests and for integral tests when available

The first application of FUMO, performed for the AP600 reactor also showed the importance of a correct simulation of the interaction between the primary system and containment A solution to this problem will be achieved through the integration of the FUMO code with a fast running primary system program that will also allow to improve the evaluation of the feedback effect of the containment pressure on the primary system

REFERENCES

[1] H Karwat, *The Importance of Natural Circulation inside Nuclear Containment Systems* , EUROTHERM Seminar No 16, Pisa, October 1990

[2] M Mazzini, F Oriolo, S Paci, *Realistic Simulation of Thermal Hydraulic Transient in Full Pressure Containment System during Severe Accident* , IAEA/OECD International Symposium on Severe Accidents in Nuclear Power Plants, Sorrento (Italy), March 1988

[3] A Manfredini, F Oriolo, S Paci, *A Model for Simulating Natural Circulation Phenomena in LWRs Containment Systems* , ENC 90 Conference, Lyon (France), September 1990

[4] A Manfredini, M Mazzini, F Oriolo, S Paci, *Il codice FUMO descrizione e manuale d uso* DCMN, Università di Pisa, RL 416 (89)

[5] L Valencia, L Wolf, *Overview of First Results on H_2 Distribution Test at the Large Scale HDR Facility* Second International Conference on Containment Design and Operation Toronto (Canada), October 1990

[6] K Fischer, *CEC benchmark Exercise on FIPLOC verification experiment F2 in BMC Specifications for Phase II* , Battelle Institute e V , Frankfurt (FRG), September 1989

[7] W Ambrosini, B Montagnini, F Oriolo *Un metodo coarse mesh per il calcolo del calore scambiato per conduzione nelle strutture in un impianto nucleare ad acqua leggera in seguito ad un incidente* IV Congresso Nazionale sulla Trasmissione del Calore UIT, Genova (I), Giugno 1986

[8] A Manfredini M Mazzini, F Oriolo, S Paci, *Valutazione dell efficacia dello spruzzamento esterno del contenimento in impianti nucleari a maggiore sicurezza passiva* VIII Congresso Nazionale sulla Trasmissione del Calore UIT Ancona (I), Giugno 1990

[9] M Mazzini S Lanza F Oriolo S Paci, C Zaffiro *Thermal hydraulic Analisys of the LACE Experiments and its fall out on the Safety Analisys of LWRs Containment Systems* , II International Conference on Containment Design and Operation, Toronto [Canada], October 1990

[10] M Schall, *Design Report for the HDR Containment Experiments V21 1 to V21 3 and V42 to V44 with specifications for the pre test computations* , PHDR Report No 3280/82, January 1982

[11] L Valencia et alii, *Hydrogen Distribution Experiment E11 1 E11 5 Preliminary Design Report* , PHDR Work Report No 10 000/89 March 1989

[12] Minutes of the *"HDR Workshop on E11 2 pre-test calculations"*, KfK Karlsruhe (FRG), November 29 30, 1990
[13] A F Gagnon, K S Howe *"Containment integrity analysis for the (W) Advanced AP-600"*, ANS Winter Meeting, S Francisco (CA), November 1989
[14] L K Conway, *"The Westinghouse AP600 Passive Safety System-Key to a Safer, Simplified PWR "*, ANS Winter Meeting, S Francisco (CA), November 1989
[15] R H Perry, C H Chilton, *"Chemical Engineers' Handbook "*, Fifth Edition, Mc Graw Hill Book, pp 5-57

STUDIES ON ALWR CONTAINMENT SYSTEM PENETRATIONS

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Abstract

Public acceptance of the next generation of nuclear power plants could be achieved if the "Containment System" is able to withstand Severe Accident (SA) conditions so that the radioactive release to the environment can be minimized. The reduction of the radioactive release during an accident may be achieved improving some precise plant requirements, among which the leaktightness of primary containment.

One of the critical points of this matter is reppresented by the containment penetrations that should be carefully investigated to highlight weak points.

Penetrations may be divided into :

1. Penetrations which see the containment atmosphere and release directly into the environment,
2. Penetrations which do not see the containment atmosphere and release directly into the environment,
3 Penetrations which see the containment atmosphere and do not release directly into the environment,
4. Penetrations which do not see the containment atmosphere and do not release directly into the environment.

With reference to external dose reduction target, the most critical kind of penetrations is the first one.

The study program is therefore focalized on the following typologies:

- electric penetrations
- mechanical penetrations
- personnel and equipment hatches

The updated quality (particularly the pressurized one) allows to solve satisfactory the issue of the electric penetrations.

About the mechanical penetrations, the isolation valves of the ventilation system can, presently, represent a possible source of release. Studies are in progress to find a different kind of valve having a double airtight, the possibility to be tested under accident pressure value, a long life and the possibility to apply a positive leakage control system

Experience done on personnel and hatches have shown a good behaviour of whole components with respect to pressure (untill 2.0 MPa even at 425° C). Tightness of both personnel airlocks and hatches depents on the degradation temperature of the elastomer in place The studies are aimed at exploring some critical areas such as heat dissipation, temperature of the seal gaskets, deformation under accident pressure and applicability of a back-up system. A finite element analysis shall be done on this equipments to better estimate their behaviour under accident conditions and to verify the efficiency of the improvements.
The program also foresees the final tests of all technical solutions at the accident conditions.

INTRODUCTION

A significant discussion has taken place in the scientific community over the last several years on the subject of containment performance for new generation of nuclear plants.
It is growing opinion that the new containment performance shall be based upon definite target of environmental impact limitation. The Italian target is to eliminate the necessity of evacuation planning and to limit extensive land contamination for any conceivable accident including severe accidents and related phenomenologies.
To pursue this target it is necessary to improve the current plant design in order to cover the technical reasons and the "what if" beyond the design criteria which are the causes of emergency planning
To do this in a real way it seems suitable to proceed through three steps

a. Consider, in the plant design, events beyond the current licensing design basis, this item needs a deep examination of Severe Accident (SA) phenomenologies in order to avoid unrealistic situations causing design requirements to become too heavy.

b. Provide a containment design able to assure the design leak rate for any conceivable accident. This target needs:

   b 1 To solve containment bypass issues (interfacing system LOCA, multiple SG tube rupture, isolation function improvements, etc.),

   b.2 To provide isolation devices able to assure the design leak rate for the new environmental condition, accident duration included.

c Improve the plant system design in order to be sure about the containment leak rate in any moment in which accident may occur (e.g. by means of a continuos leakage monitoring system).

As the ALWR design philosophy allows a considerably reduction of mechanical and electrical penetrations and the current technology provides zero-leakage electrical penetrations the most critical issue regarding the isolation device involves the personnel airlock, the equipment hatch and the isolation valves of Heating Ventilation and Air Conditioning HVAC systems (vent and purge included). These are critical componentes because they are:

- potential main contributors to total containment leak rate in SA conditions,
- a potential direct path between the containment atmosphere and the environment,
- a weak point, if their tightness is trusted to organic materials that are susceptible of aging and thermal damage.

The purpose of ENEL/CISE studies is to develop the criteria and conceptual design for personnel airlocks, equipment hatchs and isolation valves of HVAC systems in order to solve the issue b.2 according to the new target for future nuclear plants.

CONTAINMENT ISOLATION VALVES FOR HVAC SYSTEMS

The containment HVAC systems have the largest isolation valves (up to 42") in the nuclear plant and the system arrangements are such that the leakage through the two isolation valves of each line can flow directly to the environment.

Currently these isolation valves are of the butterfly type because the governing requirements are size, weight, simple and relatively fast acting, cost ect. They also present a good behaviour about leaktightness at uniform accident design temperature.

Also taking into account the system arrangement, the problems related to the butterfly valve isolation function are:

- high closing torque under accidental flow condition,
- in the soft seal application, tightness is trusted to organic material susceptible of aging and thermal damage,as above mentioned,
- in the metal-to-metal seal application, the tightness is threatened by particulate deposition (9),
- deformation of the valve body, with consequent loss of tightness, can occur when temperature reaches high values in a not uniform way (9),
- containment wall displacement can generate loads on the valves, this may prevent the valve from isolating the containment during an accident.

Many tests were performed to investigate these aspects The most meaning results are:

- the closing torque under accidental flow condition is high (about 1000 kg-m for a 24" valve) (1)
- the leakage (1) at pressure of 0 9 MPa and 180° C is quite low but increases remarcably after the valve has been cooled down·

valve size	leakage
8"	1' 5 Nm^3/h (0 65% volume/day)*
24"	4C 5 Nm^3/h (2% volume/day)* [1]

- the currently extimated SA temperature is not far from elastomer property limits.

To overcome these concerns it seems advisable, first of all, to define some general requirements such as.

- Pursue improvements in fuel design and in primary loop leaktightness in order to reduce containment cleaning and purge need The target is to reduce containment penetration number and size,
- Provide containment Air Conditioning system with fan coolers inside the containment. The target is to eliminate large normally open isolation valves.
- Especially for small Pressure Suppression Pool (PSP) type containment, provide the design for containment interfacing systems in order to avoid the pressure build-up inside containment itself. The target is to avoid containment venting needs during normal operation and then to allow blind flanges installation on vent lines, removable during plant shutdown,

and then some precise requirements like:

- elastomer performance characteristics can be optimized by valve design Valves with smaller cross section seals seem to be somewhat more resistant to the compression set of the seals than other valve designes,
- study case by case the possibility to use different type of isolation valves like the gate one and in particular the split wedge (fig.1) or the parallel slide double disc gate valve (fig.2),
- provide leak test on each valve at SA condition and in the expected direction of the leakage flow.

[1] conversion to percent volume per day are based on 50 000 m^3 containment building

About this subject the ENEL/CISE activity in progress is to define the valve type able to optimize the above issues and to include a potential simple and reliable positive leakage control system in order to claim for zero leakage for this kind of containment isolation valves without stressing weight and cost
Furthermore these valves and their interfaces have to be designed and qualified for conservative estimated SA condition. For this reason a finite element analysis shall be done on whole mechanical penetrations related to these equipment to better evaluate their behaviour under accident conditions and to verify the rightness of the improvements.

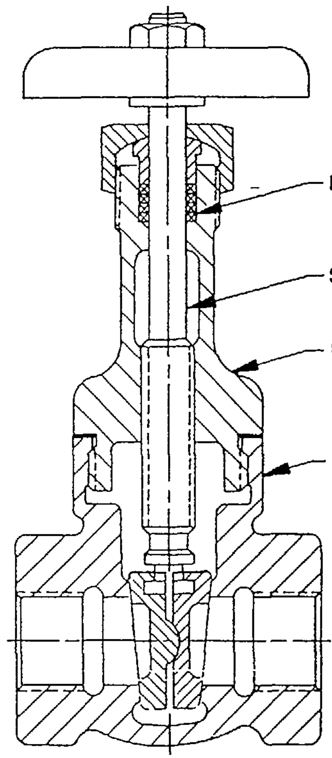

Fig. 1 - Split wedge gate valve

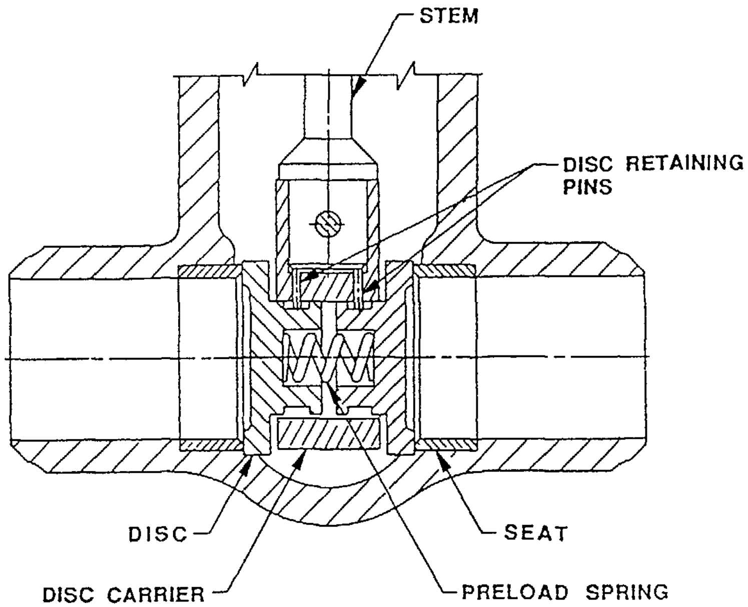

Fig. 2 - Parallel slide double disc gate valve.

PERSONNEL AIRLOCKS

Personnel airlocks provide access into and out of the containment building for maintenance and inspection crews and for the trasport of light equipment.

As part of the U.S. Nuclear Regulatory Commission's Containment Integrity Program, tests (6) were performed on a full-size personnel airlock.
The purpose of the test was to characterize the performance of the whole airlock equipment when subjected to conditions that simulate currently estimated SA conditions

The gaskets tested had a cross-section known as a "double dog-ear" configuration and were made from EPDM E603. The seal were aged at an accelerated rate to simulate aging that might occur during 40 years of continuous service and a loss of coolant accident (LOCA). A total of nine tests were performed on the airlock. In the most rigorous of these tests, the airlock inner door was subjected to pressures and temperature largely beyond the design basis. The airlock was originally designed for a pressure of 410 kPa and 170oC.

The conclusions of this test program were:

- although the gasket was degraded by an accelerated aging process, simulating both heat and radiation damages, no leakage of the inner airlock door occurred untill the gasket was subjected to the temperature lower than degradation temperature of EPDM E603 (approximately 330oC),
- failure of gasket seal was related to temperature beyond the above mentioned value,
- the gasket expanded while increasing temperature causing significant upward deflection of the door and resulting in larger gaps between the inner door and bulkhead,
- the personnel airlock survived 2.07 MPa internal pressurization at 427° C. The behaviour of the structure remained in the elastic range,
- the outer door at 2.07 MPa did not leak due to its low temperature (below 100oC).

The examination of the above results leads to these following thoughts:

- the failure of the inner door gasket must not cause temperature increase on outer door due to the metal-to-metal contact between door and bulkhead. Although the gasket could be completely destroyed the metal-to-metal contact accomplishes a strong resistence to the gas circulation,

  heat transfer conditions of the structure have an important effect on the temperature distribution. The outer door temperature and consequently the airlock isolation function depend on this heat transfer.

The objective of ENEL/CISE ongoing activity on this subject is to develop a design approach able to solve the above issues and it can be summarized in the following steps (based on the current airlock designes):

1. Get SA containment environmental condition input from safety analysis currently in progress, if possible.
2. define, based on aging test of gaskets, the maximum temperature at which the outer door can survive for a very long time (e.g. some months).
3. Define the thermal conditions of the structure (see Fig. 3) in order to keep the temperature of the outer door at the above mentioned value while the inner door temperature is at the SA conditions or at a value beyond the gasket degradation ones (about 330°C),

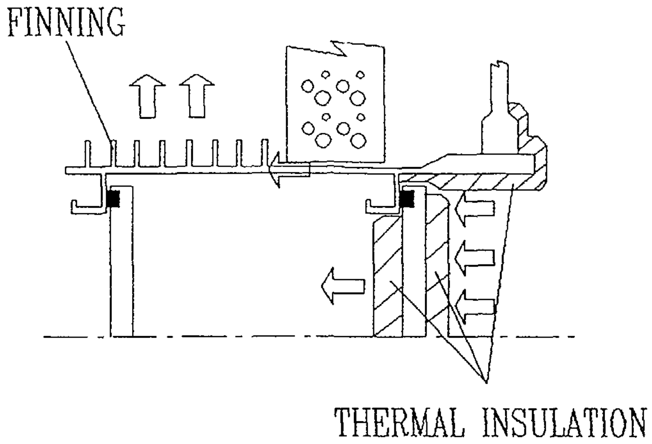

Fig. 3 - Improved current personnel airlock in order to get lower temperature at the outer door.

4. Optimize airlock design and material.

5. Perform a finite element analysis on the whole device to verify its behaviour under the foreseen conditions.

6. Test the rightness of the new approach on an actual component, if possible

EQUIPMENT HATCHES

Equipment hatches provide access into and out of the containment building for large items.
Because of the large size of these hatches the critical point are the structural deformations of the assembly and the seal behavior.

Three seal designes commonly used for nuclear power plant containment were tested at the Idaho National Engineering Laboratory (5)

These seal design were:

a) double tongue-and-groove (DTG) with silicone rubber seals

b) double O-ring (DOR) with neoprene and EPDM seals.

c) double gumdrop (DGD) with neoprene and EPDM seals.

The leakage behaviour for the three seal design has been assessed
The temperature range at which failure of elastomer seals can be expected is as follow

seal design	seal material	failure temperature
DTG	Silicon rubber	370°C
DGD	EPDM	315- 350°C
DGD	Neoprene	240-260°C
DOR	EPDM	300-330°C
DOR	Neoprene	260°C

The leak rate varied from 3.2E-04 to 1.3E01 Nm^3/h per meter of seal before the failure of the seals and from 1.2E+00 to 4.2E+01 after failure of the seals.
As the above mentioned case the leak rate was not significantly influenced by thermal aging of the seals.
Degradation and subsequent failure of the gasket seal was related to temperatures exceeding the temperature at which organic materials are stable. In this condition metal-to-metal contact at the sealing surface can reduce leakage (7).

The conclusions are the same as those personnel airlock concerning temperature while a very important problem is the deformation of the assembly essentially due to pressure although the general arrangement of the seal area allows to have a satisfactory behaviour about tightness.

Therefore the ENEL/CISE activity about hatches starts from the following assumptions:

- door design shall be such that a metal-to-metal contact is provided as back-up of elastomer seal according to the most of current designes,

- improvements of the current hatch geometry are required in order to get a better defence in depth level. A radical improvement could be a double door arrangement.

Then the activity steps are the following:

1. Get SA containment environmental condition input from safety analysis currently in progress, if possible.

2. Study improvements of a current hatch design (with a single door) in order to get:

   - a sort of back-up system for tightness function

   - a passive heat dissipation of the improved structure in order to assure better thermal conditions of the sealing elastomer

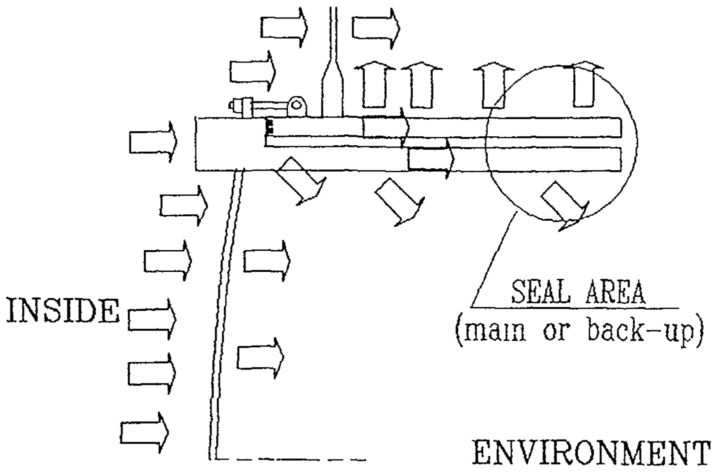

Fig. 4 - Improved current hatch in order to get lower temperature at the seal area.

3. Define, on the basis of the gaskets aging tests, the maximum temperature at which the outer sealing area can survive for a very long time (e.g. some months).

4. Define the thermal conditions of the structure (see Fig. 4) in order to keep the temperature of the sealing area at the above mentioned value while inner gasket temperature is at the SA conditions or at a value beyond the gasket degradation ones (about 330° C),

5 Optimize hatch design and material.

6. Perform a finite element analysis on the whole device to verify its behaviour under the foreseen conditions.

7. Test the rightness of the new approach on scaled models if possible.

If the above approach leads to unsatisfactory results, a double door arrangement would be studied, designed and analitically verified at the worst known SA conditions

REFERENCES

1) NUREG/CR-4648 - A Study of Typical Nuclear Containment Purge Valves in an Accident Environment - EG&G Idaho, August 1986

2) Nuclear Engineering and Design 115 (1989) - Containment Penetration System (CPS) Valve Tests under Accident Loads

3) NRC Generic Letter 89-10, Supplement 3 - Consideration of the results of NRC-Sponsored Tests of Motor-Operated Valves - October 1990

4) NRC Information Notice Nº 90-40 - Results of NRC-Sponsored Testing of Motor-Operated Valves - June 1990.

5) NUREG/CR-4944 - Containment Penetration Elastomer Seal Leak Rate Tests - July 1987.

6) NUREG/CR-5118 - Leak and Structural Test of Personnel Airlock for LWR Containments Subjected to Pressures and Temperatures *Beyond Design Limits - May 1989.*

7) Nuclear Engineering and Design 100 (1987) - Leakage Potential of LWR Containment Penetrations under Severe Accident Conditions

8) NUREG/CP-0056 - Determination of Containment Large Opening Penetration Leakage During Severe Accident Conditions - November 1984

9) EPRI NP-6516 - Guide for the Application and Use of Valves in Power Plant Systems. August 1990

DESIGN OF THE AP 600 PASSIVE CONTAINMENT COOLING SYSTEM STRUCTURES

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Abstract

The AP 600 is a greatly simplified 600 MWe Pressurized Water Reactor, for the 1990's and beyond.

Among its major features, a passive containment cooling system (PCCS) is designed to remove the residual heat, directly from the containment steel vessel, and transmits it to the environment.

The design and analysis of the PCCS structures is being performed jointly by ANSALDO and WESTINGHOUSE, as part of the AP 600 program under WESTINGHOUSE's overall leadership, and will be presented in this paper.

The PCCS uses the steel containment wall as a heat transfer surface. The surrounding concrete shield building is used along with a baffle to direct air from the top located air inlets down to a lower elevation of the containment and back up along the containment vessel. In addition a water storage tank is housed in the shield building roof at an elevation sufficient to allow gravity drain of the water on top of the steel containment.

The air and the evaporated water exhaust through a chimney in the roof of the shield building.

For the roof, a reinforced concrete conical configuration, with a cylindrical stainless steel lined water tank, has been selected and a comprehensive feasibility study has been performed including seismic behaviour, constructability, prefabrication and modularization, adaptability to cope with the External Events strength requirements. For the PCCS air baffles general functional and structural design criteria have been established by including requirements for the air flow path configuration, leakage through the baffles, inspection and maintenance; a series of alternate designs have analyzed, leading to the selection of a reference configuration that will be briefly described.

1. INTRODUCTION

The AP 600 is a greatly simplified 600 MWe Pressurized Water Reactor PWR) plant with major improvements in safety, operability, maintainability, life cycle cost and construction schedule, compared to existing nuclear plants.
Major features of this new plant are a simplified reactor coolant system, employing sealed canned motor pumps mounted on the steam generator channel head, simplified plant systems including passive safety systems and an integrated plant arrangement including a modular construction approach.
The passive containment cooling system (PCCS), more specifically, is provided in case the normal containment fan coolers are not available or an accident has occurred that requires containment heat removal at elevated pressures and temperatures.
The design and analysis of the PCCS structures is being performed jontly by ANSALDO and WESTINGHOUSE, as part of the AP 600 program under WESTINGHOUSE's overall leadership, and will be presented in this paper.

2. PCCS FUNCTIONS

Scope of the PCCS is to keep the containment internal pressure and temperature within the design values, following any postulated design basis event and to reduce the internal pressure in the long term.
The PCCS is required to perform its containment heat removal function only following a postulated design basis (e.g., a LOCA or steam line break) event which results in a large energy release into the containment or following the loss of all alternate ultimate heat sinks for an extended period of time. Heat removal by the PCCS is initiated automatically in response to a high containment pressure signal, and, with the exception of the initial actuation, requires no active components or electrical power to perform its safety function. Also manual actuation can be accomplished by the operator from the main control panel or the safety grade shutdown control panel.
As shown in Figure 1, actuation of the PCCS initiates water flow by gravity from a tank contained in the shield building structure above the containment onto the containment dome outer surface forming a water film over the structure. The path for the natural circulation of air upward along the outside walls of the containment structure is always open. Heat is then removed from the containment, utilizing the steel containment structure as the heat transfer surface combining conductive heat transfer to the water film, convective heat transfer from the water film to the air, radiative heat transfer from the film to the air baffle, and mass transfer (evaporation) of the water film into the air. Connections are provided to an alternate water source (s) such as the fire protection system to provide longer term (> 3 days) capability of containment external spray.
As the containment shell heats up in response to high containment temperature, heat is transferred to the water on the containment surface and to the air in the annulus space surrounding the containment. As the air heats up and water evaporates into the air, it becomes less dense than the cold air in the air inlet annulus.

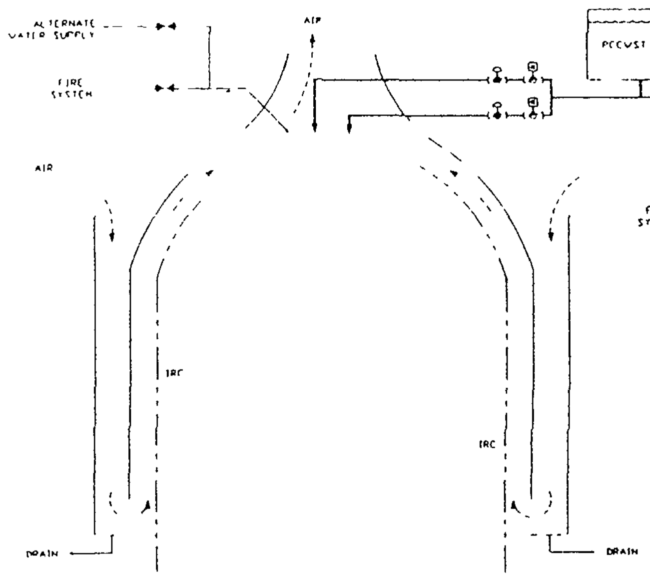


FIG 1 AP 600 passive containment cooling system

This causes an increase in the natural circulation of the air upward along the containment surface, with heated air/steam exiting the top/center of the shield building.
The elevated PCCS water storage tank will provide water for wetting the containment surface for three days following PCCS actuation. Operator action can be taken to replenish this water supply through installed piping connections and water sources. If no operator action can be taken after 3 days when the tank has drained, natural convective air heat removal will continue to function.
This provides sufficient heat removal to ensure that containment integrity is maintained although containment pressure would increase to near the design pressure.
The arrangement of the "cold" air inlet and "hot" air exhaust in the shield building structure has been selected to ensure air natural circulation during all site meteorological conditions. The air inlets are placed at the top/outside of the shield building providing a symmetrical air inlet which will not be affected by wind speed/direction or adjacent structures. The air/water vapor exhaust structure is elevated above the"cold" air inlet to eliminate the possibility of "hot" air being drawn into the air inlet.

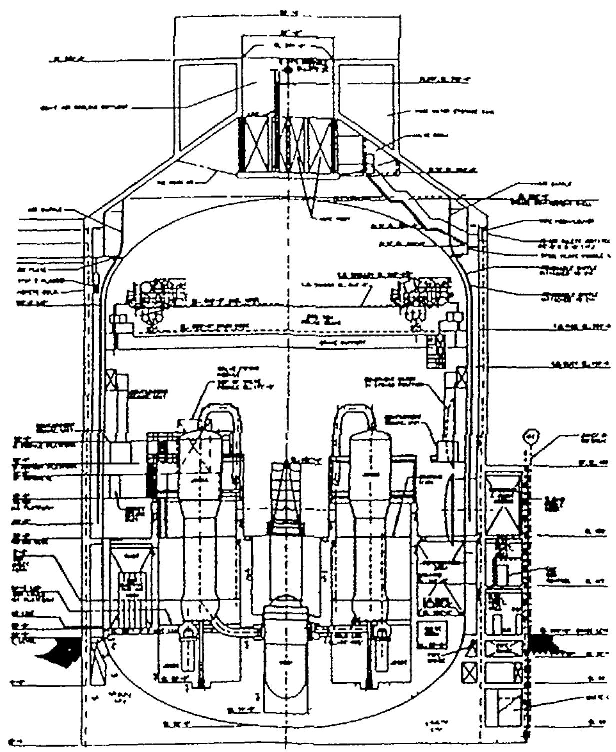


FIG 2 AP 600 general layout

3. PCCS STRUCTURE

The PCCS structures are schematically represented in the containment general layout of fig. 2.
The 130 ft diameter steel containment, hausing the primary equipment and safeguard systems, includes a cylindrical shell with elliptical heads for both the upper and lower head of the vessel.
The surrounding concrete shield building has top air inlet openings and houses a baffle to create the required air path.
A reinforced concrete conical roof ends with the air exhaust chimney and houses a 450.000 U.S. gallons water storage tank at an elevation sufficient to allow gravity drain of the water on top of the steel containment.
The roof, 2 ft thick, rests on the cylindric shield building wall with 16 reinforced concrete columns, 5 ft high and spaced circumferentially to leave 16 openings, 16 ft wide. A flat shield plate is suspended from the chimney lower end.

4 SHIELD BUILDING ROOF AND PCCS WATER TANK DESIGN

4.1 Optimum configuration selection.

Different roof and tank configurations have been investigated, prior to the final selection.

- A flat roof with deep carbon steel girders and integral water tank was first considered but significant recoating effort was expected, to assure long term tank durability.

- Preliminary cost and material comparisons showed that there are benefits in having a stainless steel boundary that would not be expected to require maintenance during the life of the plant. The second option therefore attempted to retain the original configuration and build the tank using stainless steel and clad plates. This configuration results in a combination of carbon steel, clad plate and stainless steel members in a single structure. It was recognized that this would be a difficult construction and could lead to quality and licensing questions.

- These considerations led to the selection of the reinforced concrete conical roof with a separate stainless steel lined concrete tank. Material quantities and costs were estimated for the cylindrical stainless steel lined concrete tank and the conical reinforced concrete roof. The cost was estimated to be only slightly higher (7%) than that of the original integral carbon steel option, with significantly lower maintenance cost. The conical roof with the integral stainless steel tank also has other advantages, such as larger stiffness and strength reserve margins, better thermal insulation, better tornado missile impact resistance. Its main disadvantage is that it is more difficult to construct in comparison with the flat roof where all of the steel structure would be installed as a single large module.

4.2 Structural feasibility

For the reinforced concrete roof a comprehensive feasibility study has been performed. An axisymmetric model of the roof has been developed and coupled with a sufficiently long portion of of the Shield Building cylinder with fixed botton end boundary conditions. Orthotropic properties have been included for the air inlet region. Deflections and member forces in the main structural elements (cylinder, cone, inner and outer pool walls, pool roof) were analysed for each of the main design loads (dead weight, pool hydrostatic, seismic). The deformed shape under dead weight and hydrostatic loads is shown, as an example, in fig. 3. Preliminary reinforcement sizes were developed complying with ACI 349 Code requirements, including typical connection details at the main structural discontinuities. (see figures 4 and 5).

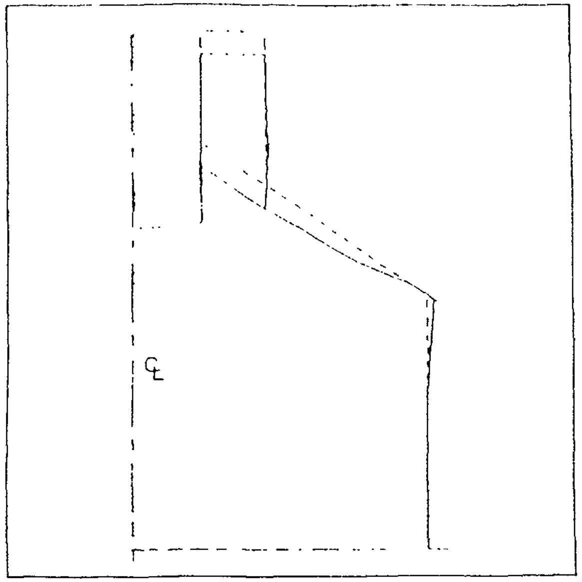


FIG. 3. AP 600 R.B. roof: Deformed shape under dead weight and hydrostatic loads.

4.3 Seismic Behavior

The axisymmetric model of the shield building roof has also been used to evaluate its dynamic characteristics. The pool water mass has been rigidly lumped to the tank pool boundary, based on results of a preliminary analysis of the seismic induced hydrodynamic effects. The model was used to size the suspension columns to avoid vibration resonances of the suspended shield plate with the supporting shield building.
The horizontal and vertical fundamental modes have frequencies of 5.2 and 7.5 Hertz respectively and are shown in figure 6. The vertical frequency is significantly higher than that of the flat steel roof (4.5 Hertz).
An equivalent lumped mass stick model of the concrete roof and supporting columns has been developed, as shown in figure 7, with inertia and stiffness properties determined to match the dynamic characteristics of the axisymmetric model. This simplified model has been included in the overall seismic model of the Nuclear Island.

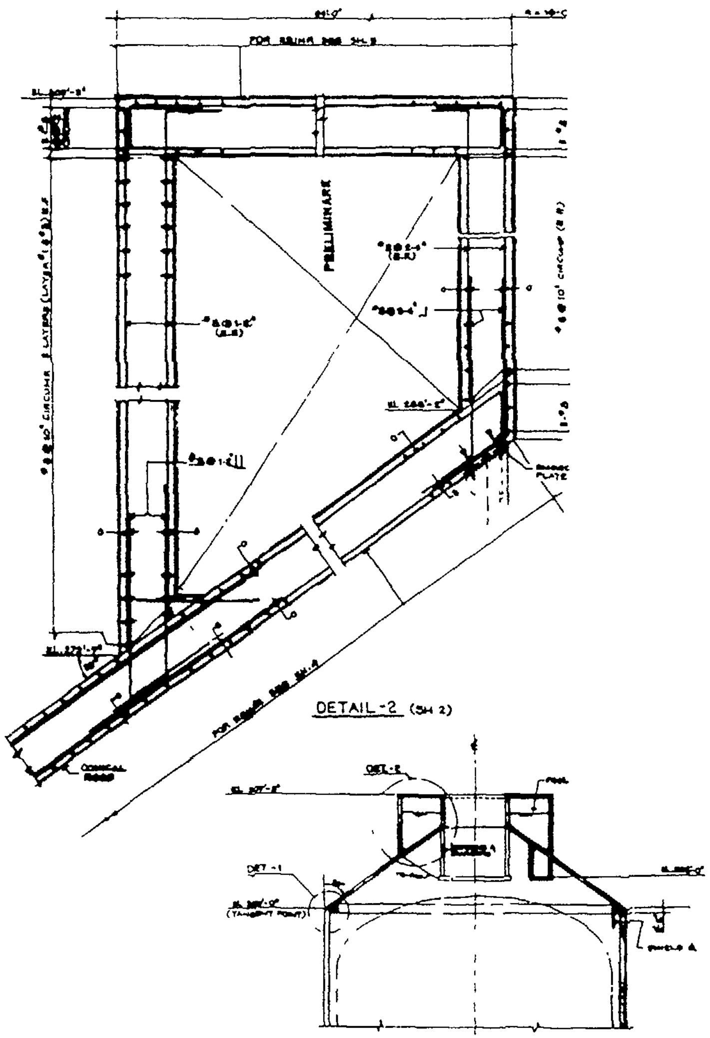


FIG. 4. PCCS tank rebar pattern

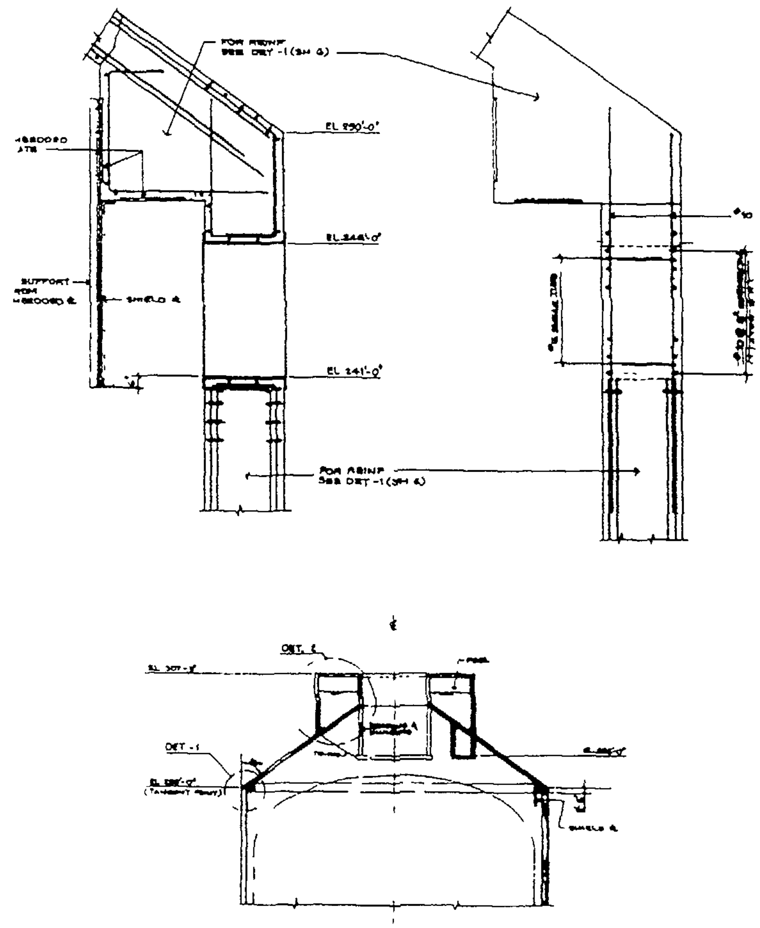


FIG. 5. Roof-cylinder connection

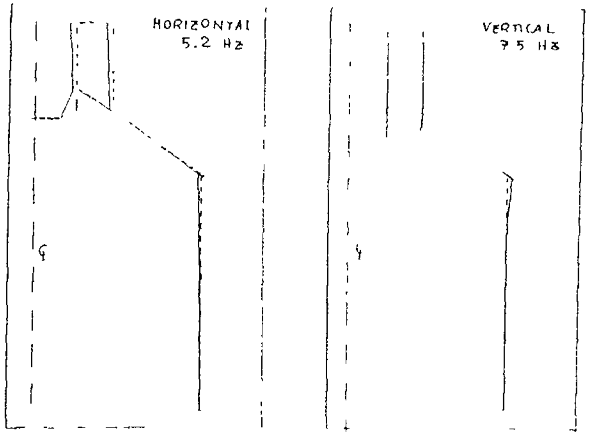


FIG. 6. AP 600 R B. concrete roof: fundamental mode shapes

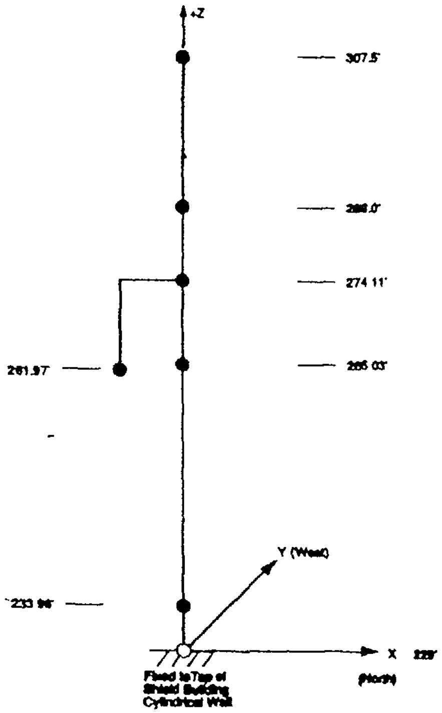


FIG. 7. Seismic analysis model: shield building roof structure.

4.4 Constructability

Constructability of the roof has been extensively investigated, together with significant advantages achievable with prefabrication; the following basic criteria have been used as a guide for such construction study.

- Wide use of removable formwork; although this would imply some time for formwork removal, it will allow that the finished structure be all reinforced concrete, without steel exposed to corrosion potential.

- Extensive use of prefabrication of reinforcing rebar moduli, including liner or embedded plates, leading to significant construction time savings.

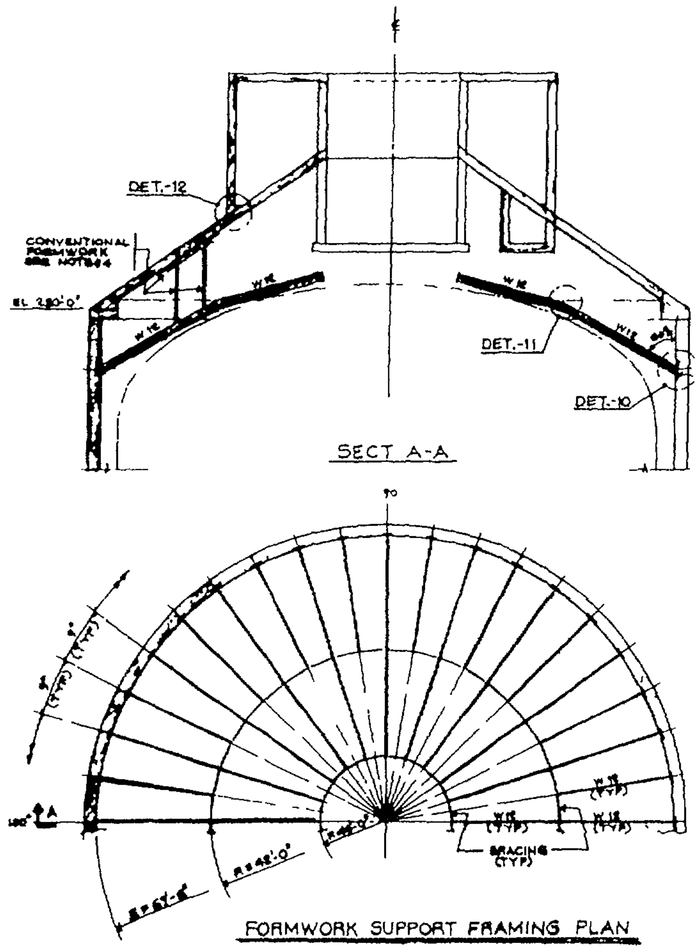
- Use of mechanical rebar splices at selected locations, instead of conventional longer lap splices, to reduce the weight of some prefabricated moduli.

- All above mentioned techniques have been already satisfactorily used for more recent nuclear power plant constructions.

- Typical construction sequences, identified for the conical roof and the upper radiation shield slab are shown in figures 8 and 9 respectively.

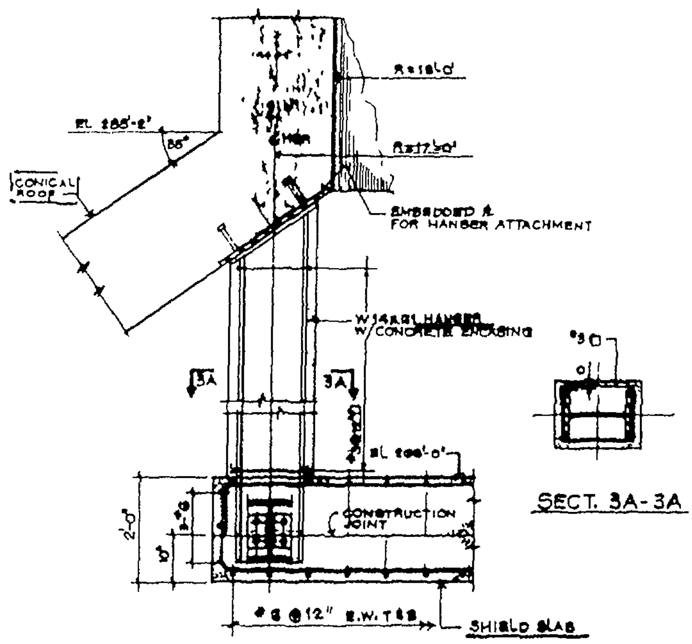
5. PCCS AIR BAFFLE

The function of the air baffle is to provide a pathway for natural circulation in the event of a postulated design basis event resulting in a large energy release into containment. In this event the outer surface of the containment vessel above the operating floor would transfer heat to the air flowing upwards between the baffle and the containment shell. Air is drawn in through the air intakes at the top of the shield building cylinder, as shown in figures 1 and 2, flows down between the shield building wall and the baffle and then up the 12 inch wide gap between the baffle and the shell to the diffuser and discharge stack. Heat from the vessel heats the air adjacent to the vessel and the lower density air creates flow.
General air baffle functional and structural design criteria have been established and used in evaluation of alternate designs. These include requirements for the air flow path configuration, leakage through the baffle, inspection and maintenance of the baffle and the containment vessel, and extreme environmental conditions for which the baffle must function.
A series of alternate designs were prepared in order to develop an optimum design to meet these functional requirements. These designs include alternates attached to the containment vessel and others attached to the shield building, as well as various sizes of panels to permit their removal if maintenance of the containment vessel coating is required. When the baffle is attached to the containment,



- FORMWORK SUPPORTED ON THE STEEL DOME
  - TEMPORARY CONSTRUCTION LOAD VERIFICATION REQUIRED
- TEMPORARY STEEL BEAMS SUPPORTED AT ONE END ON STEEL PLATES EMBEDDED IN THE SHIELD BUILDING WALL
- HALF THE BEAMS STOPPED AT ABOUT HALF RADIUS
- FORMWORK AND REMAINING FORMWORK SUPPORT CONVENTIONAL AND SUPPORTED ON RADIALLY ARRANGED STEEL SECTIONS
- THE 32 FEET DIAMETER CHIMNEY OPENING AMPLE ENOUGH TO REMOVE FORMWORK AND SUPPORT BEAMS AFTER CONCRETE HARDENING

FIG 8 Concrete conical roof construction scheme



1 INSTALL FORMWORK SUPPORTED BY THE STEEL CONTAINMENT DOME
2 PREFABRICATE BEAMS & HANGER SECTIONS
3 INSTALL BOTTOM REINFORCEMENT OF SLAB
4 INSTALL BEAMS & HANGERS IN THE FINAL POSITION.
5 INSTALL SHEAR TIES
6 POUR THE LOWER 10" OF SLAB
7 AFTER THE CONCRETE IS MATURED, REMOVE FORMS FROM UNDERNEATH
8 INSTALL TOP REINFORCEMENT
9 POUR THE REST OF THE THICKNESS OF SLAB.
10 REMOVE FORMS FROM THE SIDE OF THE SLAB
11 INSTALL TIE REINFORCEMENT AROUND STEEL HANGERS.
12 INSTALL FORMS AROUND HANGERS AND POUR CONCRETE PROTECTION AROUND HANGERS
13 REMOVE FORMS

FIG 9 Construction sequence for shield slab

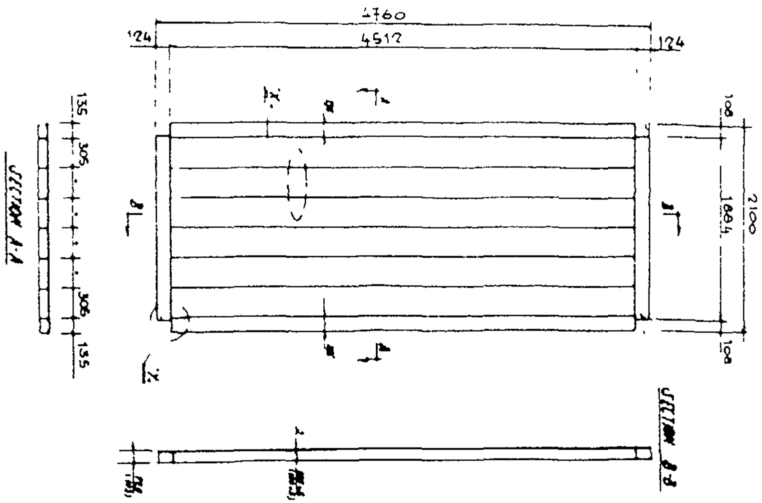


FIG 10. Vertical panel.

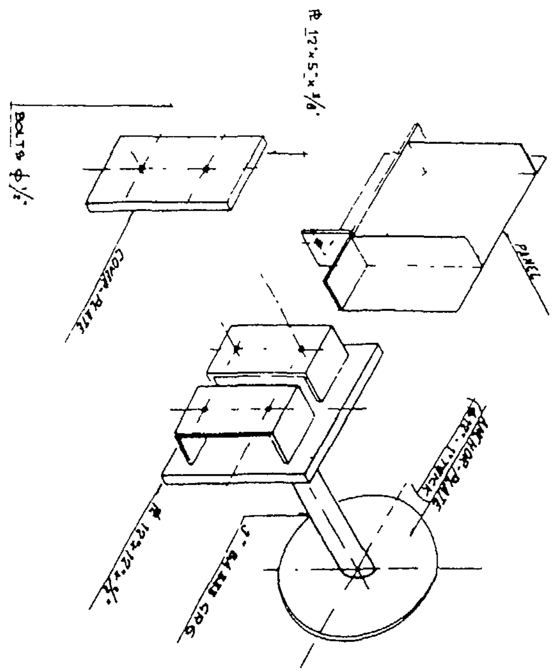


FIG. 11. Air baffle panel anchor

provisions must be made to accomodate the containment growth under pressure and temperature conditions. The critical structural loads are the results of extreme wind or seismic conditions. Evaluation of these alternatives led to the selection of the reference design which is described below for the cylindrical portion and shown in figure 10.
The cylindrical portion of the air baffle comprises 64 panels in each of 5 rows (320 total), with each panel approximately 210 cm wide and 467 cm high. Each panel is fabricated from either stiffened steel plate or corrugated plate with the final selection based on vendor availability and cost. The panels are attached to the containment vessel at the corners of each panel.
The panels are guided at each attachment point so that they may slip relative to the containment vessel growth. (see fig.11).
Cover plates are provided at these joints to minimize leakage flow through the baffle. For minor inspections, a series of windows are provided in the baffle or a TV camera may be lowered into the annulus from the top of the cylindrical portion.
The cylindrical portion terminates with an upper row of dome panels interfacing the conical roof.

6. ADAPTABILITY TO EXTERNAL EVENTS REQUIREMENTS

Feasibility studies have been performed to investigate the adaptability of the present Reactor Building layout to cope, with minor modifications with special external events, strength requirements frequently specified in European countries.
For the aircraft impact excitation, thickening of the conical roof and additional protection at the air intake and exhaust openings of the Reactor Building have been characterized and their acceptability from the air flow path requirement view point has been investigated with model wind tunnel testing.
For the external pressure wave excitation, originated by assumed gas cloud explosions, the wave propagation through the air inlet and outlet openings into the annulus between the containment and the Reactor Building has been preliminarily evaluated and the resulting transient external pressure acting on the steel containment vessel wall has been determined.
As a conseguence of the relatively small buckling resistence capability to external pressure of the cylindrical steel wall, the need of some stiffening rings has been identified, to be welded on the inside face of the containment, in order to avoid obstacles to the external flow of cooling water and air.
Less problems are anticipated for the steel dome because its buckling capability is larger and an increased thickness can be used with respect to the one needed for the internal pressure load.

ENHANCEMENT OF ADVANCED PWR SAFETY MARGINS THROUGH RELAXATION OF PCS AND CONTAINMENT DBA ASSUMPTIONS

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Abstract

Referring to advanced Pressurized Water Reactors (PWRs) there is the tendency to conceive them with larger and deeper pressure vessel to mitigate core thermal response during anticpated accidents or abnormal events This is however resisted by present Design Basis Accident (DBA) assumptions which would prescribe the reduction of Primary Cooling System (PCS) volume in order to minimize mass and energy release and thus pressure and temperature build up in the containment following a large break Loss of Coolant Accident (LOCA) caused by a complete severance within the primary and secondary systems pipework

The relaxation of the current DBA assumptions which on the basis of the acquired experience and probabilistic risk assessment studies have been shown to be overly conservative could have a significant impact on the conceptual development of advanced PWRs Specifically, the volumes of both the primary and secondary cooling systems could be optimized without close linkage to containment performance providing enhanced safety margins with respect to primary thermal excursions as partially confirmed by LOCA experiments conducted in the LOBI Test Facility

INTRODUCTION

The evolutionary approach in the conceptual development of advanced reactors while still relying on engineered safety features is also trying to exploit the potential advantages of any conceivable inherent safety characteristic This approach, however, has to comply with current safety practices based on the "defense-in-depth" concept which is ultimately aimed at preserving containment integrity in the aftermath of postulated accidents

In PWRs of current design, the primary as well as the secondary fluid volumes are minimized in order to mitigate mass and energy releases into the containment following a DBA occurrence In a typical 4-loop PWR, this event is associated with the release into the containment of the entire inventory of the primary cooling system and of the secondary cooling system of one steam generator This is in turn used to determine eventual containment loads and related design requirements

The acquired reactor operational experience and probabilistic risk assessment studies, however, have shown that the ensuing prescriptive requirements to be complied with in the safety evaluation of water cooled reactors are generally overly conservative [1] The adoption of more realistic DBA assumptions such as the Leak Before-Break (LBB) criterion, on the other hand, could lead to enhanced safety margins by providing the basis for the optimization of primary and secondary cooling system volumes without stringent constraints strictly coupled to containment response

Within this context, experiments conducted in the LOBI installation, an integral system test facility operated in the Ispra Site of the Commission of the European Communities Joint Research Centre, have shown that significant safety improvements could be achieved through the increase of primary system volume, generally, core thermal response was considerably mitigated when the experimental installation was configured with a large Reactor Pressure Vessel (RPV) volume Although the reported experimental results cannot be directly extrapolated to full-size prototypes, they are, however, indicative for the development of advanced PWR concepts which, among others, are being conceived with larger and deeper pressure vessel, larger pressurizer and secondary system volumes

PERSPECTIVE ON DBA ASSUMPTIONS

In the safety evaluation of current PWRs a complete severance of the primary loop pipework and of the steam line of one steam generator is taken as the reference case for the assessment of the emergency safety systems performance and containment response The inherent safety margins embedded in this assumption have lead to the adoption of the diversity and redundancy criteria in the configuration of the safety systems and to rugged design constraints for the containment

Referring to the containment vessel, the pressure build up resulting from flashing of the inventory released by the primary and secondary cooling system under DBA assumptions and added safety allowance have lead to typical maximum containment pressures of c 6 3 bar Regardless of the safety allowances which notionally included [2] decay heat (ANS standard + 20 %), free volume of containment (reduced by 2 %), primary system and steam generator volumes (increased by 2 %) and a 15 % increased for excess pressure, present DBA assumptions call for a general reduction of liquid volumes This approach, however, while minimizes the safety challenges to the containment, neglects the potential benefits resulting from an adequate optimization of both primary and secondary system volumes

It is now generally recognized that the reconsideration of present DBA assumptions could be justified by the acquired reactor operational experience and by the fact that probabilistic risk assessment studies have shown an expected frequency of occurrence far below 10^{-6}/y, on the other hand, there is emerging a general consensus of opinions that the adoption of the LBB criterion is a more realistic one Also, there are indications that the safety features of advanced containments should be rather matched to cope with the consequences of severe accidents whereby eventual overpressures could be limited by venting possibilities

All in all, it appears that compliance with more realistic criteria could lead to advanced reactor configurations which while satisfying generally accepted safety criteria would provide sufficient margins for the exploitation of the safety features inherent to the availability of a larger fluid inventory

THE LOBI RESEARCH PROGRAMME

Programme Objectives

The LOBI (LWR Off normal Behaviour Investigations) Research Programme which represents a significant contribution to the overall Reactor Safety Research Programme of the Commission of the European Communities, has been mainly devoted to the generation of an experimental data base relevant to PWR postulated accident conditions Within this context experiments have been conducted in a scaled, integral system test facility for the

- identification and/or verification of basic phenomenologies governing overall system response and components interaction during the evolution of
  - large break LOCAs
  - small break LOCAs
  - anticipated and abnormal transients
  - recovery procedures
  - accident management strategies
- assessment of the predictive capabilities of system codes used in reactor safety analysis

The research programme originated from a reactor safety R&D contract with the Bundesminister fur Forschung und Technologie (BMFT) of the Federal Republic of Germany which early in the 70's decided on the need of an integral system test facility for thermal hydraulic investigations relevant to PWRs of German design On the basis of its tender the Commission of the European Communities was charged by the BMFT with the execution of this project under a contractual agreement The BMFT programme was then complemented with a Community programme of general interest and freely accessible to all EC member states

The experimental programme is strictly linked to comprehensive code application and assessment activities RELAP5/MOD1 EUR and RELAP5/MOD2 in their IBM versions as well as CATHARE, ATHLET/DRUFAN, TRAC and RETRAN have been used either within the JRC or by outside organisations for test design and test prediction calculations

The LOBI Test Facility

The LOBI test facility is a full-power, full pressure integral system test facility representing an approximately 1 700 scale model of a 4-loop, 1300 MWe PWR of German design The test facility which incorporates the essential features of the reference reactor primary and secondary cooling systems, was commissioned in December 1979 and operated until June 1982 in the MOD1 configuration for the investigation of large break LOCA phenomenologies, it was then extensively modified in the MOD2 configuration which was recommissioned in April 1984 and operated until June 1991 for the investigation of phenomenologies relevant to small break LOCAs and Special Transients

The test facility comprises two primary loops, the intact and the broken loop which represent respectively three loops and one loop of the reference PWR Each primary loop contains a main coolant circulation pump and a steam generator The simulated core consists of an electrically heated 64 rod bundle arranged in an 8 x 8 square matrix inside the pressure vessel model, nominal heating power is 5 3 MW Lower plenum, upper plenum, an annular downcomer and an externally mounted upper head simulator are additional major components of the reactor model assembly The primary cooling system which is shown schematically in Fig 1 operates at nominal PWR conditions approximately 158 bar and 294 326° C pressure and temperature, respectively Heat is removed from the primary loops by the secondary cooling system which contains a condenser and a cooler, the main feedwater pump, and the auxiliary feedwater system Normal operating conditions of the secondary cooling system are 210° C feedwater temperature and 64 5 bar pressure

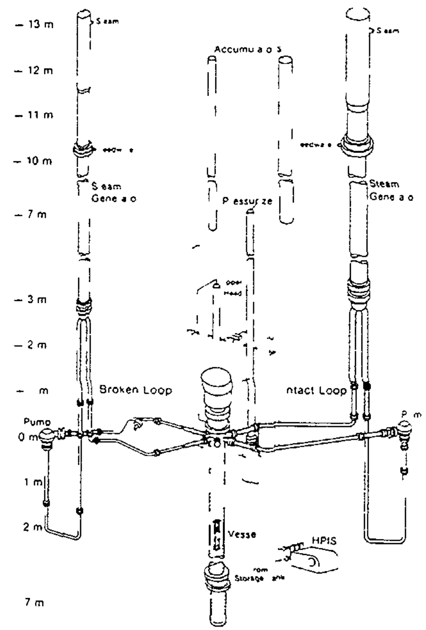

Fig 1 The LOBI Test Facility

The measurement system consists of a total of about 470 measurement channels. It allows the measurement of all relevant thermohydraulic quantities at the boundaries (inlet and outlet) of each individual loop component and within the reactor pressure vessel model and steam generators. The whole LOBI-MOD2 test facility and individual components were scaled to preserve, insofar as possible or practical, similarity of thermal-hydraulic behaviour with respect to the reference plant during normal and off-normal conditions. A process control system allows the simulation of both reactor main coolant pump hydraulic behaviour and core decay heat.

The Experimental Programmes

The overall LOBI Experimental Programme comprises two parts defined A and B:

- the Experimental Programme A has been performed in the framework of the contractual agreement between the CEC and the German BMFT
- the Experimental Programme B has instead been performed in the framework of the CEC Reactor Safety Research Programme with independent contribution from several industrial and institutional organisations of EC Member Countries.

Generally, the methodology used in the definition of each test case and in the establishment of the corresponding test profile, was to reproduce governing physical phenomena rather than reference plant expected specific behaviour. While the test cases of the A matrix were defined to reproduce phenomenologies of specific interest to PWRs of Siemens-KWU design, the test cases of the B matrix were instead specified to represent conditions of general interest in reactor safety analysis. Also, a Partner Country was allocated to each test of the B matrix; that is, an EC member country organisation having the task to collaborate, on behalf of all participating countries, with the LOBI staff in the detailed test specification as well as in the pre- and post-test analysis of the results with large system codes. A list of all tests executed in the context of the LOBI Experimental Programme is given in Table 1.

LOBI RESULTS RELEVANT TO ADVANCED PWRs

In the context of the LOBI-MOD1 Experimental Programme which was mainly devoted to the evaluation of the DBA scenarios, two test series were performed differing in the volume of the simulated reactor pressure vessel model; specifically, the gap width of the annular downcomer was 50 mm in the first test series and 12 mm in the second series (Fig. 2). This lead to a RPV volume which differed by a factor of c. 3.6 in the two configurations. The original underlining rationales for the two configurations are detailed in [3]; here, we are interested on the enhancement of safety margins deriving from increased RPV volume with particular reference to prospective conceptual developments of advanced PWRs.

The relevance of increased RPV volume for the potential enhancement of safety margins during the blowdown-refill period of a large break LOCA can be inferred from the following comparative analysis of tests having similar transient assumptions.

Table 1: LOBI-MOD1 and -MOD2 Experimental Programmes

LOBI-MOD1 Experimental Programme

(Dec 1979 - June 1982)

Test	Partner Country	Date	Description
A1-04	D	12 12 79	200 % CL Break LOCA, CL ECC, DC 50 mm
A1-01	D	29 01 80	200 % CL Break LOCA, CM ECC, DC 50 mm
A1-02	D	14 02 80	200 % CL Break LOCA, CM ECC, DC 50 mm
A1-03	D	19 03 80	200 % CL Break LOCA, CM ECC, DC 50 mm
A1-04R	D	17 04 80	200 % CL Break LOCA, CL ECC, DC 50 mm
A1-05	D	06 05 80	200 % CL Break LOCA, CM ECC, DC 50 mm
SD-SL-01	D	04 06 80	10 % CL Break LOCA, DC 50 mm
SD-SL-02	D	18 06 80	1 % CL Break LOCA, DC 50 mm
SD-SL-03	D	24 09 80	0 4 % CL Break LOCA, DC 50 mm
A2-59	D	27 10 80	100 % CL Break LOCA, CL ECC, DC 50 mm
B-101	F	26 11 80	2 x 50 % CL Break LOCA, CL ECC, DC 50 mm
A2-55	D	19 01 81	50 % CL Break LOCA, CL ECC, DC 50 mm
A2 59R	D	11 02 81	100 % CL Break LOCA, CL ECC, DC 50 mm
B-R1M	D	17 03 81	25 % CL Break LOCA, CL ECC, DC 50 mm
A1-66	D	03 07 81	200 % CL Break LOCA, CL ECC, DC 12 mm
A1-07	D	09 07 81	200 % CL Break LOCA, no ECC, DC 12 mm
A1-06	D	21 07 81	200 % CL Break LOCA, CM ECC, DC 12 mm
A1-67	D	30 09 81	25 % CL Break LOCA, CM ECC, DC 12 mm
A1-68	D	28 10 81	50 % CL Break LOCA, CM ECC, DC 12 mm
A1-10A	D	25 11 81	200 % HL Break LOCA, CM ECC, DC 12 mm
A1-10B	D	10 12 81	200 % HL Break LOCA, CM, DC 12 mm
A1-70	D	13 01 82	200 % PS Break LOCA, CM ECC, DC 12 mm
A1-73	D	04 02 82	25 % HL Break LOCA, CM ECC, DC 12 mm
A1-72	D	24 03 82	200 % CL Break LOCA, CM ECC, DC 12 mm
A1-69	D	06 04 82	100 % CL Break LOCA, CM ECC, DC 12 mm
A1-74	D	21 04 82	200 % CL Break LOCA, CM ECC, DC 12 mm
B-222	F	05 05 82	2 x 50 % CL Break LOCA, CL ECC, DC 12 mm
B-302	I	16 06 82	2 x 50 % HL Break LOCA, CL ECC, DC 12 mm

HL	Hot Leg
CL	Cold Leg
CM	Combined Hot and Cold Leg Injection
DC	Downcomer
PS	Pump Suction
LOCA	Loss-of-Coolant Accident
SG	Steam Generator
HPIS	High Pressure Injection System
ECC	Emergency Core Cooling
LONOP	Loss of Offsite and Normal Onsite Power
ATWS	Anticipated Transient without SCRAM
LOAF	Loss of All Feed Water
LOFW	Loss of Main Feed Water
ACCU	Accumulator
SGTR	Steam Generator Tube Rupture
PCS	Primary Cooling System
CPT	Counter Part Test
MCP	Main Coolant Pump

LOBI-MOD2 Experimental Programme

(April 1984 - June 1991)

Test	Partner Country	Date	Description
A1-76	D	12 04 84	SG Performance
A2-81	D	27 09 84	1% CL Break LOCA, 2/4 HPIS in CL
A1-82	D	28 09 84	1% CL Break LOCA, 2/4 HPIS in HL
A1-78	D	24 10 84	2% CL Break LOCA, ECC in CM
A2-77A	D	28 11 84	Natural Circulation, 90 and 75 bar
A1-83	D	19 12 84	10% CL Break LOCA, ECC in CM
A2 90	D	27 03 85	LONOP ATWS "Station Blackout"
A1-85	D	07 05 85	0 4% Pressurizer Break, 2/4 HPIS in HL
BL-00	F	03 07 85	0 4% CL Break LOCA, 1/3 HPIS in CL
A1-84	D	14 10 85	10 % HL Break LOCA, ECC in CM
BT-00	UK	30 11 85	LOFW + Bleed and Feed
BT-01	B	24 01 86	Small (10%) Steam Line Break + PTS
BL-02	UK	22 03 86	3% CL Break LOCA, 2/4 HPIS in CL
A1-79	D	15 05 86	1% CL Break LOCA, 4/4 HPIS in HL
A1-88	D	11 06 86	0 4% CL Break LOCA, as Cooldown
BL-01	D	20 09 86	5% CL Break LOCA, HPIS + ACCU in CM
BC 01	WG-B	18 10 86	SG Secondary Inventory
BC 02	WG B	26 11 86	SG Heat Losses
BL-21	I	24 01 87	0 4% SGTR + 'SSN' Recovery
BL-12	F	19 02 87	1% CL Break LOCA, no HPIS, no Cooldown
BT-02	F	09 05 87	LOAF + Feed and Bleed
BT-12	UK	17 06 87	Large (100%) Steam Line Break
A1-91	D	26 09 87	1% CL Break LOCA, 1/4 HPIS in HL
BT-03	I	24 10 87	LOFW ATWS + 'SSN Recovery
A1-92	D	30 11 87	Natural Circulation, 40 bar
BL-16	D	19 03 88	0 4% CL Break LOCA, as Cooldown
BC-03	WG B	15 04 88	SG Heat Losses
A1 93	D	30 04 88	2% CL Break LOCA, no HPIS
A1-94	D	27 05 88	4% CL Break LOCA, 40 bar
BC-04	WG-B	07 02 89	Core Bypass Measurement
BL-30	WG B	15 04 89	5% CL Break LOCA, HPIS + ACCU in CL
BL-22	B	17 06 89	0 4% SGTR + Cooldown
A1-87	D	11 11 89	PCS Cooldown, MCP off
BT-04	F	10 02 90	PCS Cooldown, MCP on, 1 SG Isolated
BL-34	WG-B	22 03 90	6 % CL Break LOCA at Low Power, BETHSY CPT
BL 44	JRC	26 04 90	6 % CL Break LOCA at Full Power, no HPIS
BT-56	UK	03 07 90	Multiple Failures
BT-15/16	UK	22 11 90	LOFW with SG Boiloff and Refill, MCPs on/off
BT-17	D	07 02 91	LOFW with SG Bleed and Feed
BT-06	F	21 03 91	Small (10 %) Feed Line Break
BL-40	E	16 05 91	SGTR in 1-Loop PWR
BL-06	F-UK	21 06 91	1 % CL Break LOCA, HPIS off, MCP on

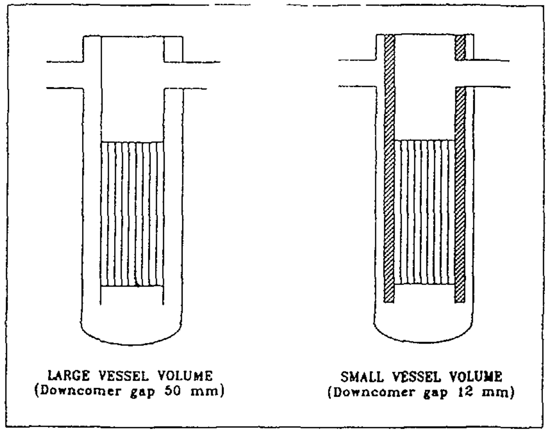

Fig 2 LOBI-MOD1 RPV Configuration with Large (50 mm) and Small (12 mm) Downcomer Gap Width

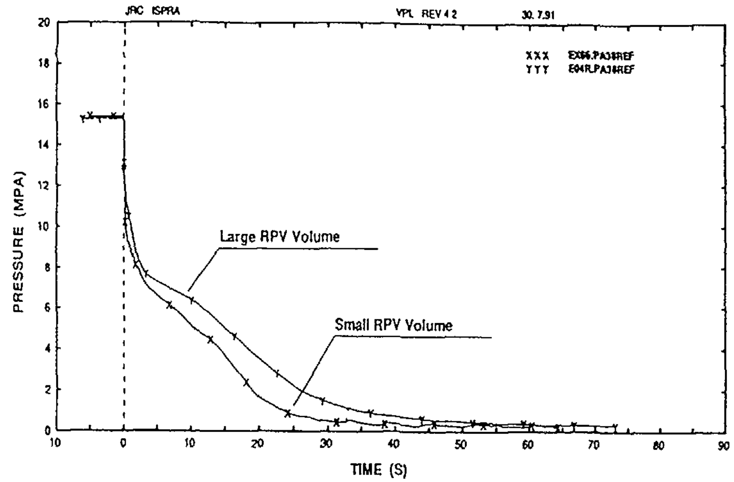

Fig 3
LOBI MOD1 Pressure Response during a Large(200%) CL Break LOCA for Large and Small RPV Volume

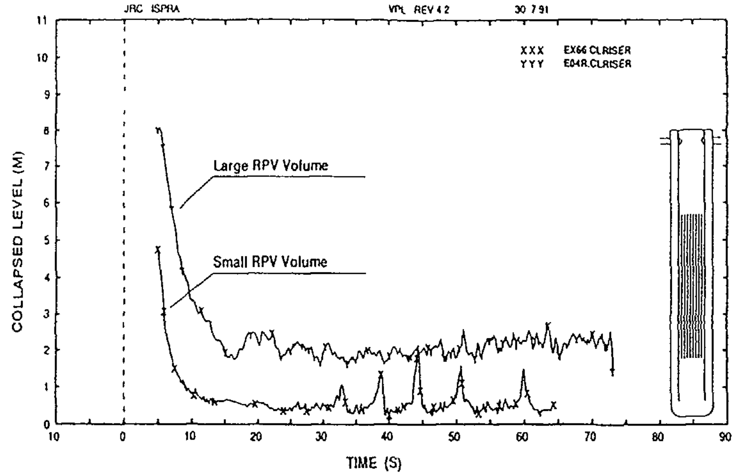

Fig 4
LOBI-MOD1 Core Collapsed Liquid Level Response during a Large(200%) CL Break LOCA for Large and Small RPV Volume

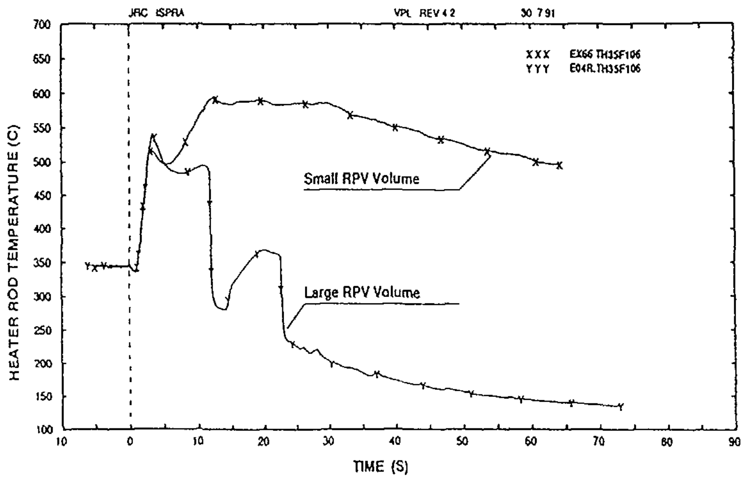

Fig 5
LOBI MOD1 Peak Heater Rod Temperature Response during a Large(200%) CL Break LOCA for Large and Small RPV Volume

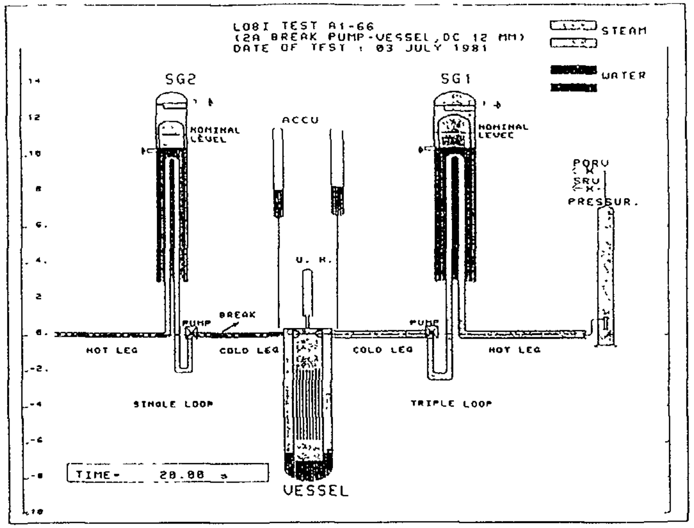

Fig. 6.
Fluid Distribution within the LOBI Test Facility with Small RPV Volume during a Large (200%) CL Break LOCA

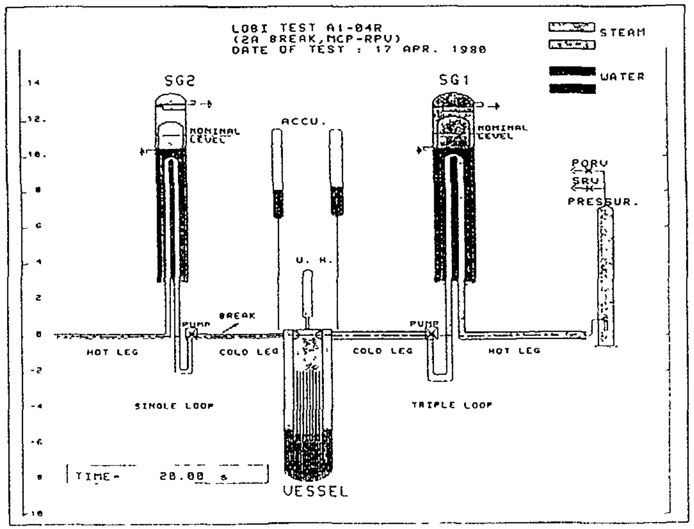

Fig. 7:
Fluid Distribution within the LOBI Test Facility with Large RPV Volume during a Large (200%) CL Break LOCA

Large 200 % Cold Leg Break LOCA

Both tests referred to hereafter, simulated a large 200 % cold leg break LOCA with accumulator emergency core cooling safety injection. Overall initial and boundary conditions were generally similar or directly comparable being the RPV volume the discriminating parameter.

No perceptive influence of RPV volume on the system thermal-hydraulic response occurred during the very first blowdown period when subcooled fluid conditions persisted upstream of the break (Fig. 3 to Fig. 5). However, the course of the transient was strongly affected *during the subsequent saturated blowdown and refill periods.* After fluid evaporation had also started in the cold regions of the system as expected the PCS depressurized at a reduced rate in the case of large RPV volume (Fig. 3); however, persistance of higher liquid level in the simulated core region (Fig. 4) and the re-establishment of core flow were much more pronounced in the test with large RPV volume leading to enhanced cooling of the heater rod bundle (Fig. 5). The respective fluid distributions within the primary and secondary cooling systems at a significant time corresponding to minimum primary fluid inventory are given in Fig. 6 and Fig 7 for the two test cases.

Intermediate 50 % Cold Leg Break LOCA

The influence of RPV volume on overall system response was much more pronounced in the intermediate break size range (Fig. 8 to Fig. 10). As in the large break LOCA, regardless of RPV volume, the depressurization rate was quite similar until subcooled conditions existed upstream of the

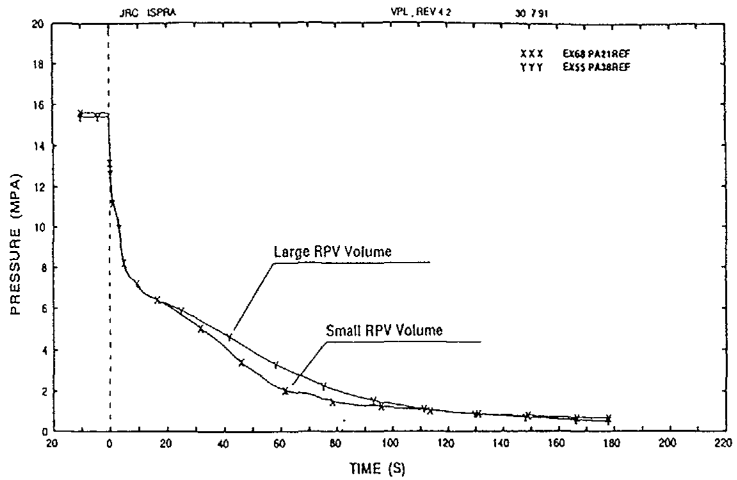

Fig 8.
LOBI-MOD1 Pressure Response during an Intermediate(50%) CL Break LOCA for Large and Small RPV Volume

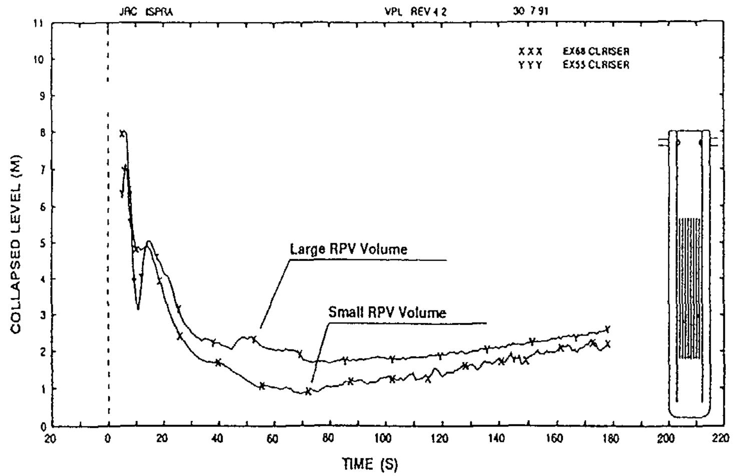

Fig 9
LOBI MOD1 Core Collapsed Liquid Level Response during an Intermediate (50%) CL Break LOCA for Large and Small RPV Volume

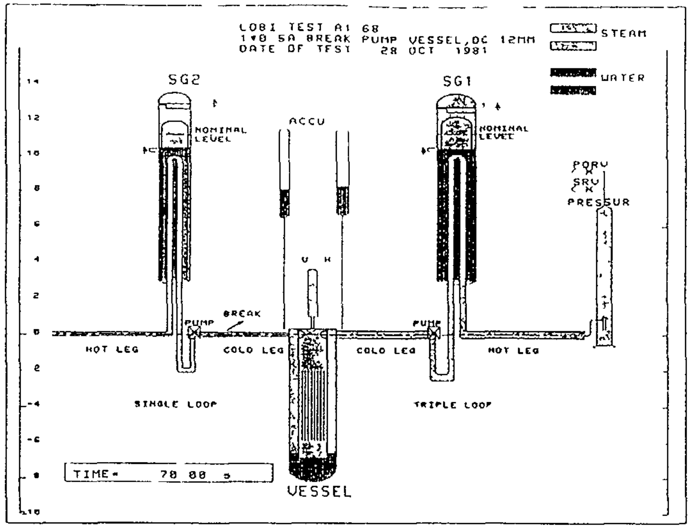

Fig 11
Fluid Distribution within the LOBI Test Facility with Small RPV Volume during an Intermediate (50%) CL Break LOCA

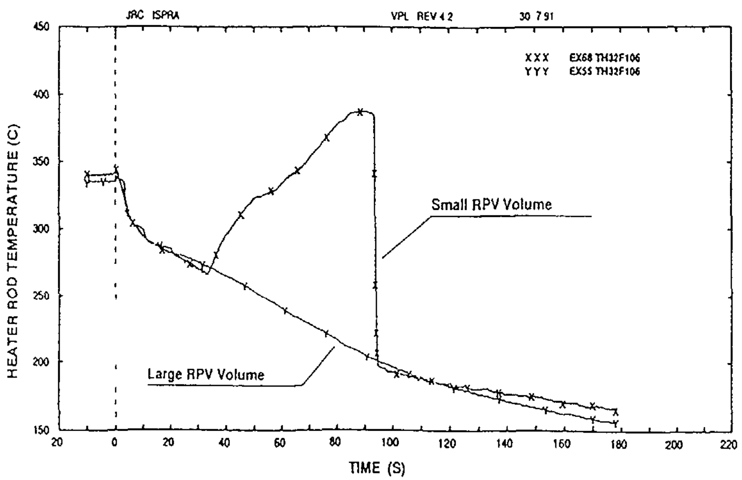

Fig 10
LOBI MOD1 Peak Heater Rod Temperature Response during an Intermediate (50%) CL Break LOCA for Large and Small RPV Volume

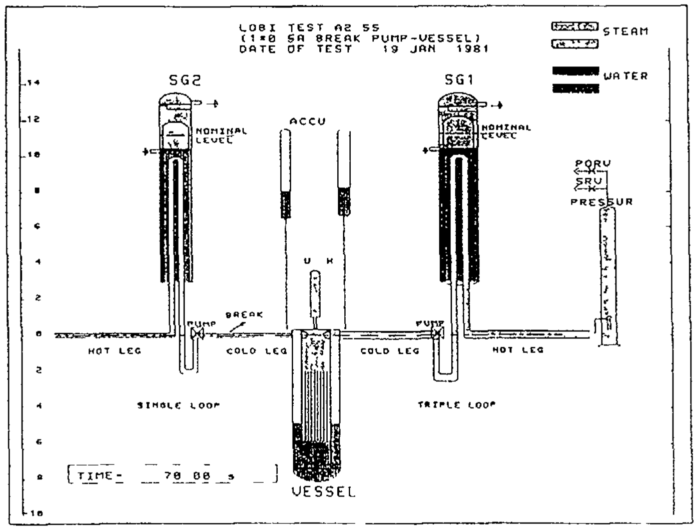

Fig 12
Fluid Distribution within the LOBI Test Facility with Large RPV Volume during an Intermediate (50%) CL Break LOCA

break and the pressurizer surge line had not uncovered Thereafter, the test case with small RPV volume experienced a faster depressurization (Fig 8) with early core uncovery (Fig 9) and dryout (Fig 10) Core coolability was always ensured in the test case with larger RPV volume The relative fluid distributions within the experimental installation at 70 s into the transient are given in Fig 11 and Fig 12

As a general consideration, the reported experimental results confirm the potential benefit resulting from increased PCS volume in general and RPV volume in particular, with respect to the general evolution of LOCA events and related safety systems performance as well as operator requirements

CONCLUDING REMARKS

The adoption of larger primary and/or secondary system volumes in the conceptual development of advanced PWRs could certainly provide enhanced safety margins This is however in apparent contrast with the need to comply with present DBA assumptions which try to minimize mass and energy release into the containment following postulated accidents On the other hand, it is also generally recognized that present DBA assumptions are overly conservative and unnecessarily hinder potential safety improvements In addition, there are indications that the safety features of advanced reactors should be rather matched to cope with the consequences of severe accidents whereby eventual containment overpressures could be limited by venting possibilities This could pre-empty the need to strictly couple primary liquid volume to containment performance and design requirements

REFERENCES

[1] G A Davis, R A Matzie, M D Green, H D Brewer and W A Fox
Design Considerations for Severe Accident Containment Performance
Nuclear Engineering and Design, 120 (1990), 115 - 121

[2] J Czeck, H Fabian, P Gast and O Gremm
Mitigation of Severe Accident Consequences by the Containment Design of KWU-LWR
Nuclear Engineering and Design, 117 (1989), 11 - 17

[3] H Stadtke, D Carey, W L Riebold
Influence of Downcomer Volume and Gap Width on Blowdown
Proceedings of the "International Meeting on Thermal Nuclear Reactor Safety", Chicago, USA, 1982

FOUNDATION BEARING CAPACITY ON SOFT SOILS

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Abstract

In Pressurized Water Reactors (PWRs) of current design, the primary as well as the secondary volumes are minimized in order to mitigate mass and energy releases into the containment following a postulated Design Basis Accident (DBA). In a typical 4-loop PWR, this event is associated with the release into the containment of the entire inventory of the primary system and of the secondary of one steam generator. This is in turn used to determine eventual containment loads.

Present DBA assumptions consider a simultaneous large break LOCA within the primary pipework and within the secondary steam line. The relaxation of these assumptions which on the basis of the acquired experience have been shown to be overly conservative could have a significant impact on the conceptual development of advanced PWR concepts. Specifically, the volumes of both the primary and secondary systems could be optimized providing enhanced safety margins.

Within this context, experiments conducted in the LOBI installation, an integral system test facility operated in the Joint Research Centre, Ispra Site, have shown the potential benefit resulting from increased primary system volume with respect to the general evolution of LOCA events and related safety as well as operator requirements; generally, core thermal response was considerably mitigated when the experimental installation was configured with a larger primary volume.

Although the experimental results acquired in the LOBI installation and proposed in the paper cannot be directly extrapolated to full-size plants, they are however indicative for the development of advanced reactors which, among others, are being conceived with larger and deeper pressure vessel, larger pressurizer and low care power density.

1 INTRODUCTION

Present work is part of the studies started by ENEL-DSR/VDN on ALWR systems

Among these studies, soil structure interaction in dynamic stress conditions determining the forces transmitted to the soil and consequent bearing capacity is of fundamental importance from a geotechnical point of view

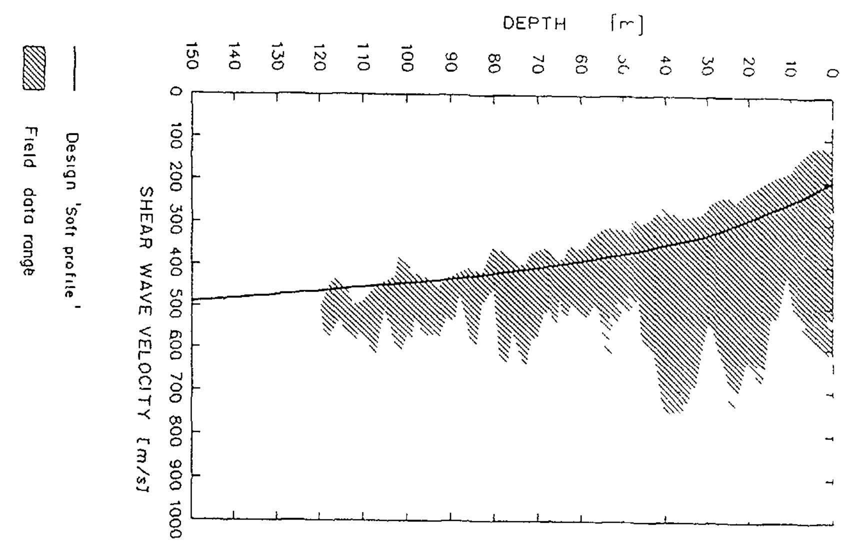


FIG 1 Italian incoherent soil profiles

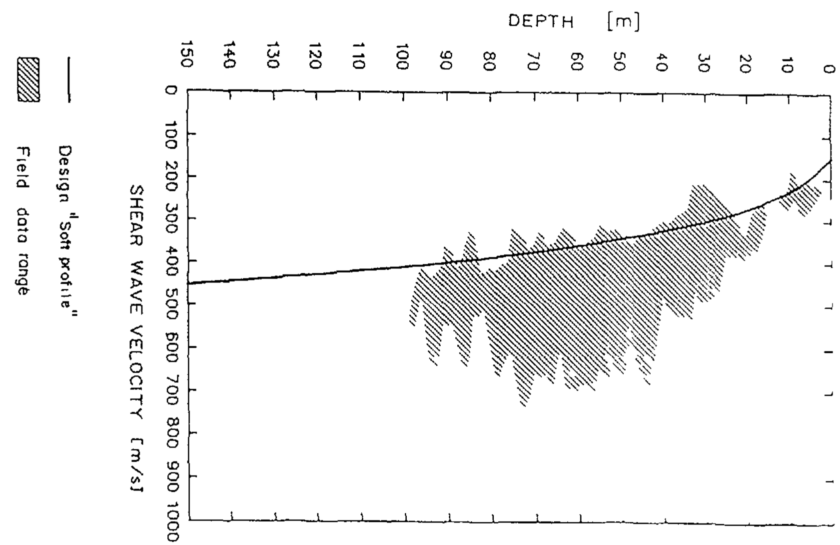


FIG 2 Italian cohesive soil profiles

Infact the characteristics of possible Italian sites generally alluvial sites are such that the soil bearing capacity is of particular interest to develop the project especially for dynamic conditions

In the following are discussed the results of the preliminary parametric evaluations carried out illustrating the study methods applied that are divided into

- definition of geotechnical reference profiles
- resolution of soil structure interaction determining the forces transmitted to the soil starting from a model of the structures and definition of seismic design parameters
- evaluation of bearing capacity

The evaluations carried out up to now, allow some comments to be made on the main factors that influence the bearing capacity with reference to some codes of pratices of interest to Italy, to the safety factor concept and to the possible use of other calculation methods in addition to classical ones

2 GEOTECHNICAL REFERENCE PROFILES FOR ANALYSIS

The following geotechnical profiles have been defined to carry out soil structure interaction analyses from a dynamic point of view, using elastic equivalent soil behaviour and to evaluate bearing capacity with pseudostatic methods (plastic rigid soil behaviour)

In determining the representative geotechnical profiles of the real Italian sites reference was made to data collected during previous studies and extensive investigations carried out on sites where possibly ENEL Plants would be set up

The data related to the various sites were examined firstly recognising homogeneous stratigraphic profiles and initially subdividing them between mainly incoherent profiles and mainly cohesive profiles

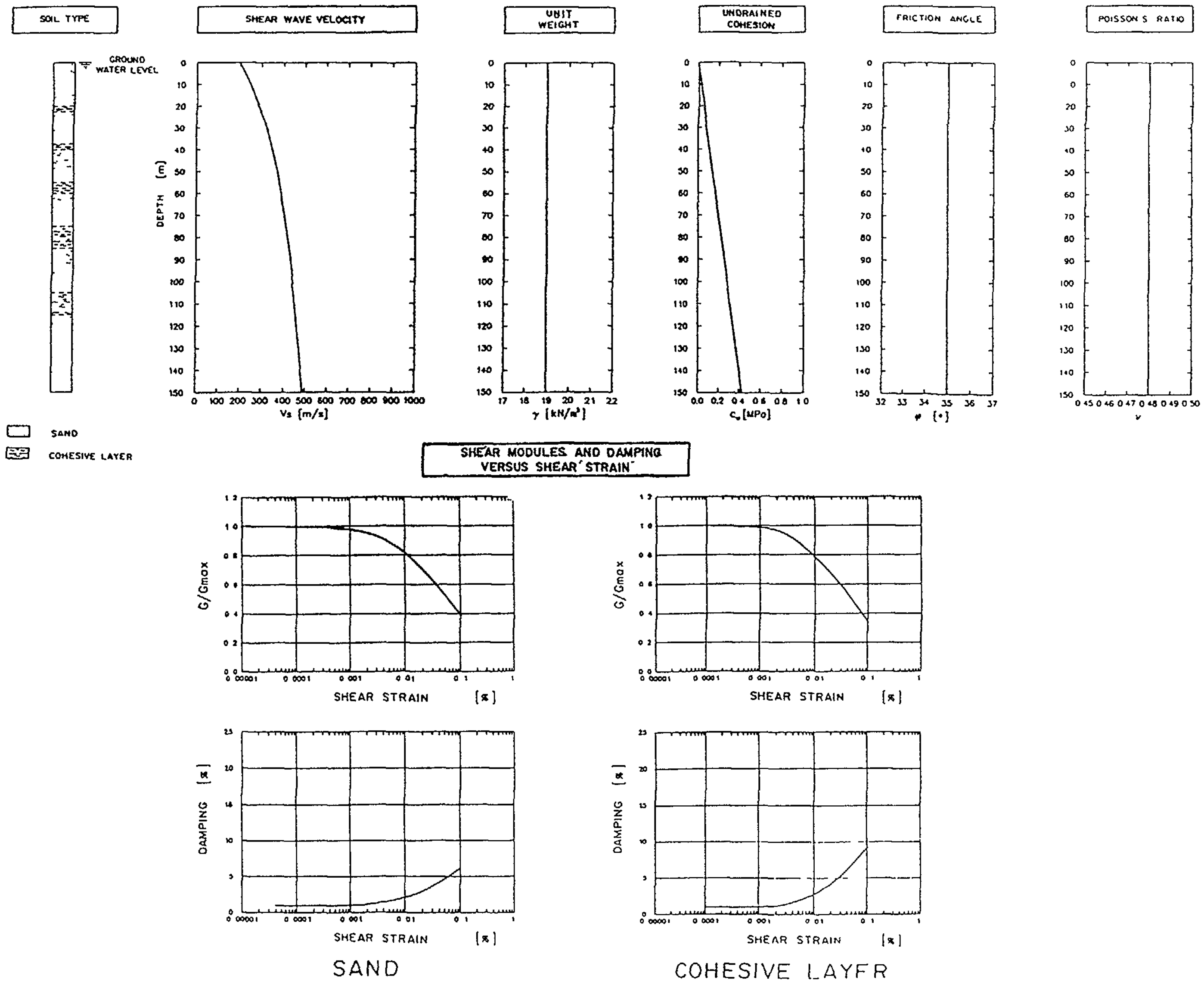


FIG 3 Incoherent soft profile

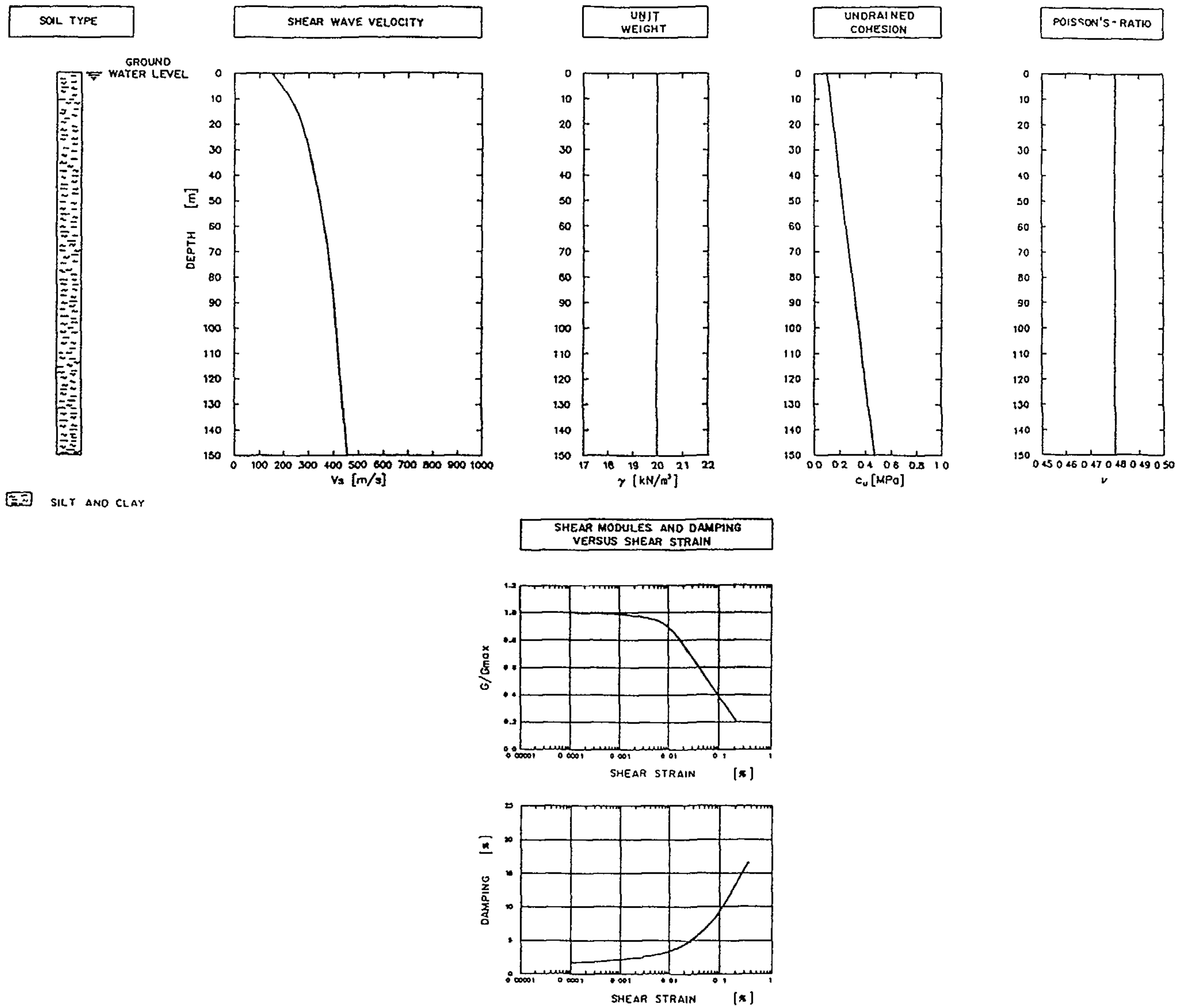


FIG. 4. Cohesive soft profile.

Subsequently shear wave propagation velocity values in relation to depth were examined. Shear wave velocity can be correlated to the initial shear modulus values (Gmax). As well-known, those two parameters are the most widely used to characterize the soil behaviour in dynamic stress conditions.

Shear wave velocity (Vs) design curves were determined starting from Vs field data (fig. 1, 2), relating both to mainly incoherent profiles and mainly cohesive profiles. These two curves can be considered representative of so - called "soft profiles" among incoherent and cohesive profiles.

It should be noted that the "soft profile" definition does not therefore corresponds to an absolute criteria but to a relative criteria, that arises from the variability of the field data examined and from the effective stratigraphic conditions found in Italian alluvial areas.

The Vs values related to the sites defined as soft show an increasing trend with depth, starting from 150÷200 m/s at ground level up to 450÷500 m/s at 150 m depth. As shown in figures 1 and 2, these values represent the lower bound of the variability range of the sites used as references. It is to point out that the sites considered do not include rocky sites, characterized as well known by much higher shear waves velocity values (Vs=1000÷1500 m/s).

Considering the preliminary and parametric character of the studies to be carried out, the soil conditions have been modelled with stratigraphically homogeneous geotechnical profiles, respectively incoherent and cohesive, to analyse soil-structure interaction.

For the bearing capacity evaluations referred to the incoherent profile, the presence of cohesive layers, as found in some incoherent sites, cannot be ignored. Therefore for such evaluations the presence of cohesive levels was taken in account at average depths of 20; 40; 55; 80; 110 m from ground level, with thickness of 2.5; 2.5; 5; 10; 10 meters.

The geotechnical parameter values related to the two soft incoherent and cohesive profiles were defined according to the experimental data collected at those sites where shear wave velocity values are closer to the Vs curves selected.

The position of the ground water level was assumed to coincide with ground level considering also the real site conditions.

The geotechnical reference profiles selected for present work are shown in figures 3 and 4, where the main soil parameters are summarized.

3. SOIL-STRUCTURE INTERACTION

Soil-structure interaction assessment requires the following activities to be carried out:

- modelling of structure;

- seismic input definition.

Soil-structure interaction was analysed on the base of these elements and of the soil data referred to in the previous paragraph, assuming a linear-equivalent soil behaviour, taking in account the stratigraphic effect and foundation embedment.

3.1 Structure Modelling

The structures examined relate to the SBWR and PIUS system buildings; in the following, we do not go into the modelling problems, summarising only the main features of the models necessary to carry out the analyses.

The semplified models adopted for the analysis are shown in figures 5, 6, 7, 8; the PIUS system is characterized by its circular foundation shape with mass and rigidity axial symmetry, while the SBWR system has a rectangular foundation shape with a mass and rigidity eccentricity on the plane XY.

Another essential element of the models is determined by the embedment, that for PIUS is assumed equal to 8 meters, whilst for the SBWR system it is equal to 20 meters. This depth appears to be of particular importance from a geotechnical point of view, taking in account construction methods especially in order to control the water level. Embedment value therefore could require a further study in the future, that at present time goes beyond the aim of soil-structure interaction studies.

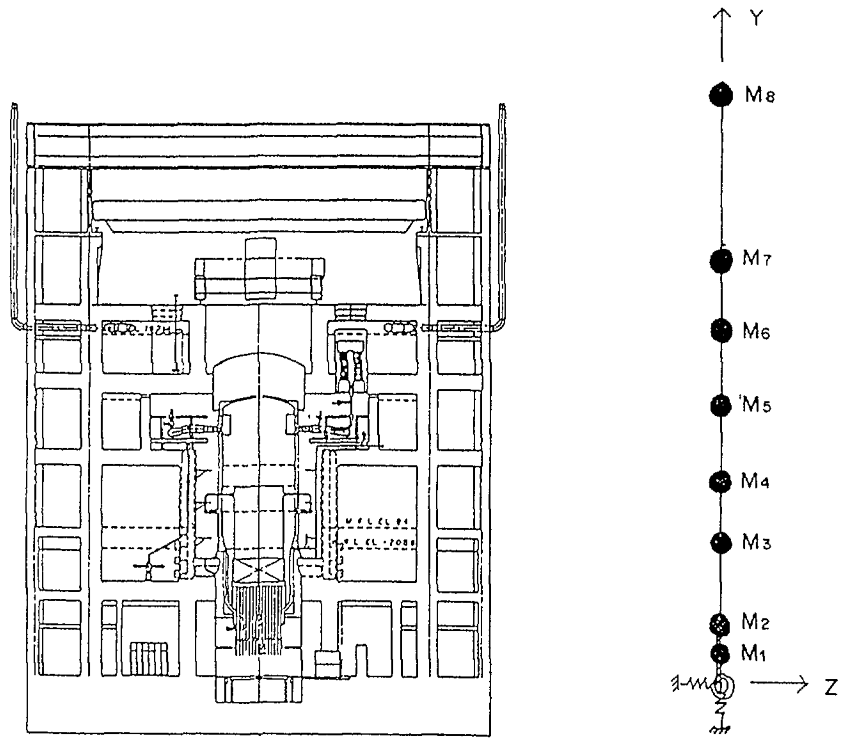


FIG. 5. SBWR: Y-Z section.

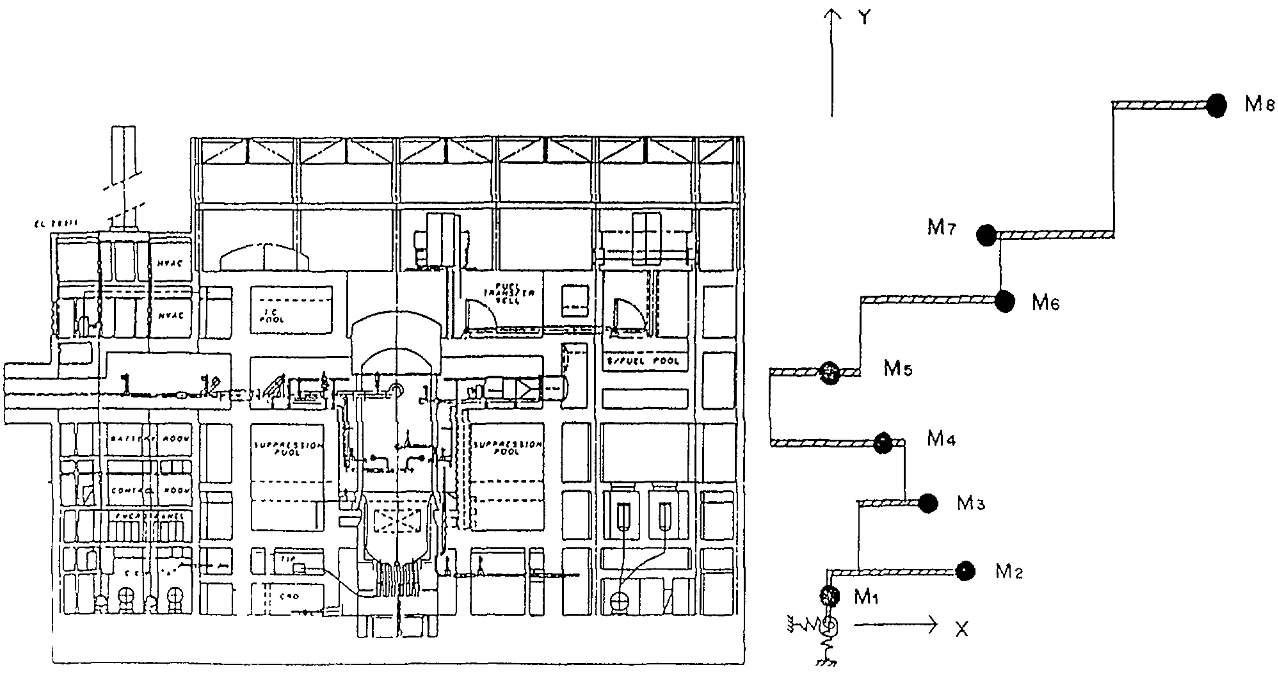
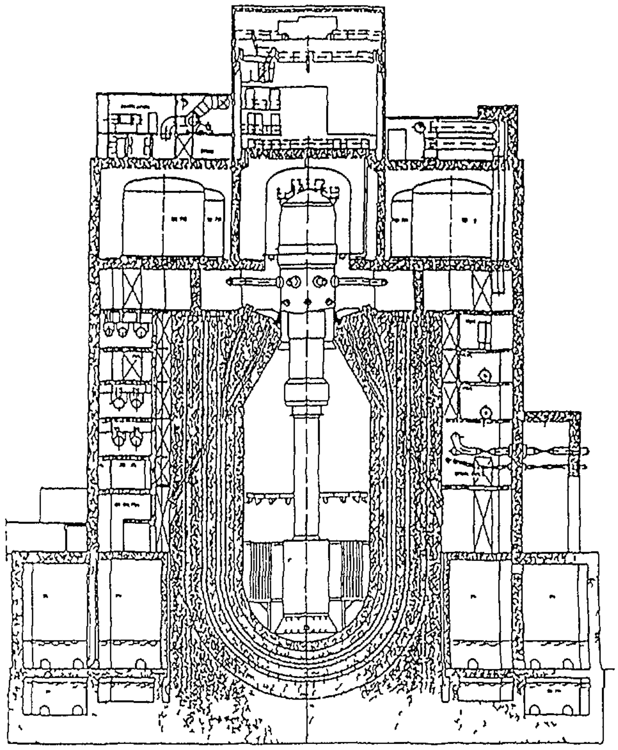


FIG. 6. SBWR: Y-X section.



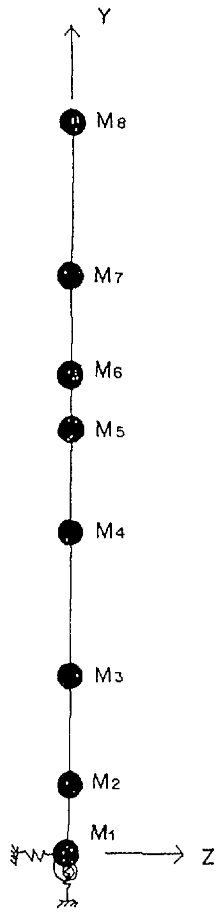
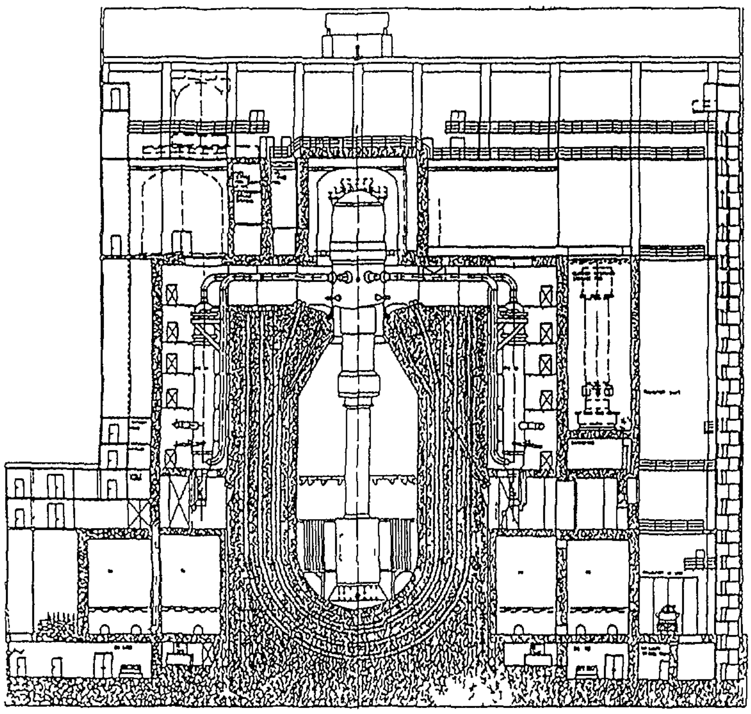


FIG 7 PIUS Y-Z section



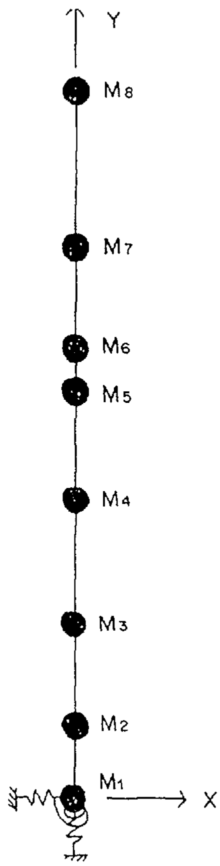


FIG 8 PIUS Y-X section

3.2 Seismic Input

With reference to the criteria followed in Italy, the analyses were carried out taking as seismic input synthetic time histories obtained from a wide band *response spectrum*

In particular the Newmark spectrum, modified for soft soils according to ENEL criteria (ENEL 1986) was used scaled to 0 3g and 0 18g (hereinafter S3 and S18)

The 0.3g value was assumed in accordance with the preliminary value proposed by ENEL (1991) for SSE (Safe Shutdown Earthquake) The 0 18g value represent the minimum value in accordance with the design criteria requested in Italy for Nuclear plants

In Italy is available also a large amount of data related to historical seismicity (with a catalogue very good from a temporal and spatial point of view, covering at least 1000 years); neotectonic seismicity, instrumental strong motion records (Friuli, Irpinia etc) Therefore it was possible also to choose some real time-histories (hereinafter R1, R2, R3), characterised by a maximum acceleration ranging from 0 08g to 0 14g and selected from a group of Italian records related to site, magnitude and epicentral conditions reasonably representative for typical Italian "soft sites"

3 3 Foundation Loads

Foundation loads were calculated with an iterative procedure (SOILFLEX and STICK/NASTRAN Computer Codes), that involves calculating representative concentrated soil parameters and the dynamic response of the soil-structure system The iterative process ends when convergence on soil deformation values is reached

First, the equivalent soil damping and rigidity values were calculated for the "soft incoherent and cohesive profiles", to better estimate the geotechnical parameters (Newmark, 1980).

TABLE I LOAD VALUES OF SOIL PROFILE

SBWR

Seismic Input	S18	S3	R1
Seismic load			
N (t)	32120	52130	10800
TX (t)	42865	61140	10230
TZ (t)	40200	67770	17000
MX (t*m)	864680	1445360	358200
MZ (t*m)	889890	1246580	230900

PIUS

Seismic Input	S18	S3	R1	R2	R3
Seismic load					
N (t)	25280	41335	6990	9090	5180
TX (t)	40940	57690	15140	10190	6630
TZ (t)	40100	65860	17250	18980	6590
MX (t*m)	1227300	1855830	521500	555060	210260
MZ (t*m)	1298700	1951150	479780	334030	246690

Then, the seismic loads acting on the foundation were calculated Load values shown in table 1 represent the worst condition, corresponding to the incoherent soil profile

To these dynamic loads are to be added the vertical dead loads that as order of magnitude are equal to 1.800 000 kN (185.000t) for both sistems buildings

4. BEARING CAPACITY ANALYSIS

Bearing capacity evaluations were carried out starting from the foundation loads and soil resistance parameters using traditional pseudostatic approach by limit equilibrium methods

The bearing capacity evaluations were carried out combining the whole loads relating to a direction with the 40% of the homologous loads arising from the excitation acting along the others directions.

The calculations were performed referring to the following conditions for soil parameters

- cohesive homogeneous soil
- incoherent homogeneous soil,
- incoherent soil with interbedded cohesive layers

The geotechnical parameter values related to these three situations were assumed according to geotechnical reference profiles discussed above

Calculations were carried according to the Terzaghi Brinch Hansen formula, strictly valid for homogeneous soil, both for incoherent and cohesive profile To examine the presence of cohesive layers at different depths within the incoherent profile, reference was made to the slice method, also of limit equilibrium kind, assuming sliding surfaces passing through the cohesive layers

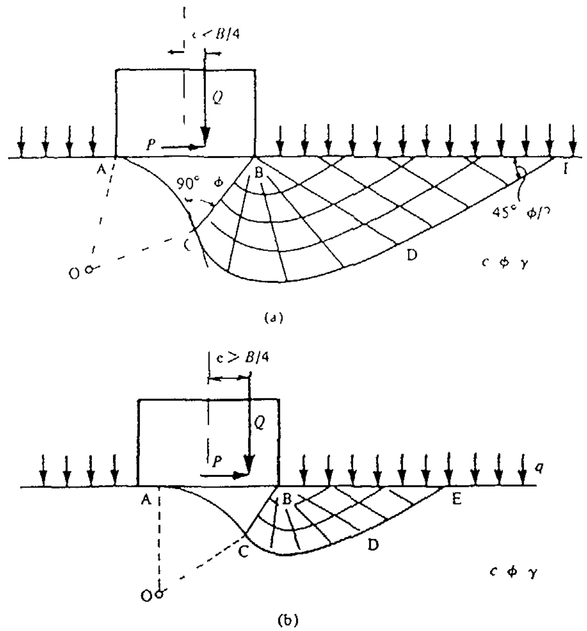
As well-known, with the Terzaghi Brinch Hansen method, the safety factor value is defined in global terms of forces, as ratio between the resistant vertical force (calculated taking in account the horizontal and eccentric loads) and the acting vertical force However, in the slice method the safety factor is defined in terms of stresses, as ratio between the resistant shear stress and the acting one along the sliding surface, assuming a constant safety factor along this one

In order to compare the results obtained with these two methods, the values obtained with the slice method were expressed in terms of forces, increasing the acting vertical load until the safety factor value (in terms of stresses) reaches the unit value along the sliding surface The ratio of vertical resistant forces determined in this way versus the acting loads, represents the safety factor value in terms of forces that can be compared with that deduced with the Terzaghi Brinch-Hansen formula (Fig. 9).

The qualitative results of the numeric analyses obtained in this way expressed as safety factors in terms of forces, are shown in table 2 for the two systems SBWR and PIUS

Examining these results, the preliminary and parametric character of the analyses performed shall be taken in account

TERZAGHI BRINCH HANSEN



$$q_0 - Q_0/B'L = c\, N_c\, \zeta_c\, \zeta_{ci} + q\, N_q\, \zeta_q\, \zeta_{qi} + \tfrac{1}{2}\, \gamma\, B\, N_\gamma\, \zeta_\gamma\, \zeta_{\gamma i}$$

$B = B \quad 2e_B$
$L' = L \quad 2e_L$

N_c, N_q, N_γ } Bearing Capacity Factors

ζ_c, ζ_q, ζ_γ } Shape Factors

ζ_{ci}, ζ_q, $\zeta_{\gamma i}$ } Inclination Factors

(Winterkorn Fang Foundation Engineering Handbook)

FIG 9 Theoretical slip patterns under eccentric and inclined loads

TABLE 2. SAFETY FACTORS IN TERMS OF FORCES

SAFETY FACTOR

PIUS			
	Soil type		
Seismic input	Cohesive Homogeneous Soil	Incoherent Homogeneous Soil	Incoherent Soil with Cohesive Layers
S3	*	**	* - **
S18	*	***	* - ***
R1	**	***	** - ***
R2	**	***	** - ***
R3	**	***	** - ***

SBWR			
	Soil type		
Seismic input	Cohesive Homogeneous Soil	Incoherent Homogeneous Soil	Incoherent Soil with Cohesive Layers
S3	*	*	** - ***
S18	**	***	** - ***
R1	**	***	*** - ***

Safety Factor

* - would be satisfied with further analysis
** - will be satisfied with further analysis
*** - is satisfied

with reference to existing codes of practice

S3 - 0.30 g
S18 - 0.18 g

R1, R2, R3 Real Time Histories

For cohesive homogeneous soil conditions, referring to the Italian sites, the analyses performed suggest the need of performing further studies that conventionals involving seismic input level also, to better assess the foundation bearing capacity.

For incoherent homogeneous soil conditions, at this step of studies do not arise problems in order to bearing capacity. Some more considerations shall be developed about the maximum seismic input level

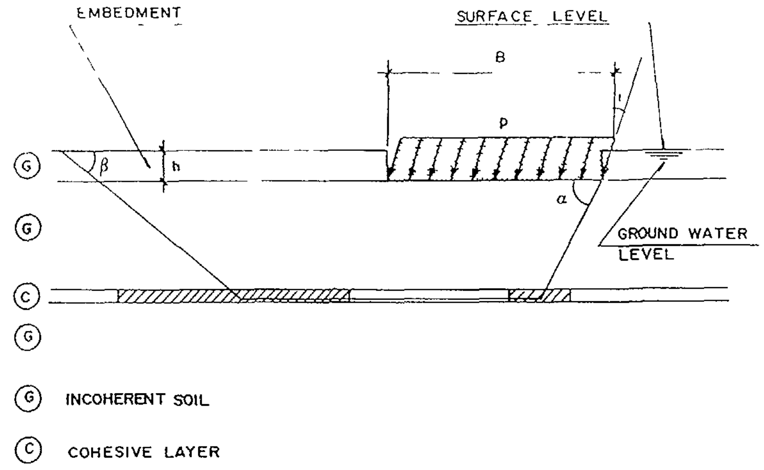


FIG. 10. Soil conditions.

Incoherent soils with interbedded cohesive layers conditions show the influence of cohesive layer depth. Problems about bearing capacity decrease with increasing depth of the layer. In this case further analyses, similar to homogeneous cohesive conditions, should be performed, taking in account possibility of soil improuvement from both a technical and economical point of view (Fig. 10).

5. STANDARDS AND SAFETY FACTORS

In order to evaluate the results shown above comparing them with standards from codes of practice in the following, same consideration are illustrated about the bearing capacity calculation methods and corresponding safety factors

IAEA (Safety Series No 50-SG-S8 and No 50-SG-S2).

The safety factor depends on the calculation method. When traditional type analyses at limit equilibrium (Terzaghi Brinch-Hansen) are performed, it refers to standards of each AIEA member, specifying that some of them require a value of safety factor equal to 3 in static conditions and 2 in dynamic conditions Further analyses are required if inadequate safety factor values are obtained by limit equilibrium method and as no reference value are given, considerations about the safety factor should be developed according to the method of analysis adopted

Italian Code DM LL PP 11 3 88

No particular calculation methods are referred to, it can be assumed that reference is made to traditional methods at limit equilibrium (Terzaghi Brinch- Hansen) The safety factor must be 3, but different values can be adopted if adequately supported, it should be the case of different kind of analyses

Eurocode 7 Geotechnics, Design

The safety factors is generally evaluated in terms of partial coefficients

A multiplicative coefficient higher or equal to one has to be applied to the acting loads and a coefficient lower than one to tne soil resistance parameters

The bearing capacity design value must be greater or equal to the acting vertical load multiplied as above

The design bearing capacity, according to Eurocode 7, can be assessed with the classical approach (Terzaghi Brinch-Hansen) assuming as partial safety coefficient 1/1 2 for the tangent of the angle of friction and 1/1 8 for cohesion

6 CONCLUSIONS

The aspects related to the bearing capacity evaluations in seismic conditions for sites defined as "soft" have been discussed for the buildings of ALWR systems (systems SBWR and PIUS)

Analyses were carried out parametrically, the aim was to reach some preliminary assessments and identify the significant elements that influence the analyses and that need more development in the future

Examination of geotechnical field data collected at real Italian sites, allows to identify some "soft" sites showing the importance of bearing capacity evaluation in dynamic conditions

The development of future studies should concerne

- seismic input definition

- bearing capacity calculating methods

With reference to the last point, the available standards provide safety factor values that can be referred to the use of limit equilibrium methods that should not always be satisfactory In such cases more complex methods should be considered taking into account the effective soil stress-strain behaviour, such finite elements, boundary elements etc , to calculate soil deformations for complex load histories in static and dynamic terms

With these methods the limit load can be higher than that obtained with the traditional techniques and the safety factor can be defined corresponding at an acceptable deformation level

BIBLIOGRAPHY

CHRISTIANO P P , e Al (1974) "Compliance of layer elastic system" Institution of Civil Engineeres London

ELSABEE F , MORRAY J P (1977) "Dynamic behavior of embedded foundation" MIT pub n R77 33

ENEL (1986) "Criteri per la progettazione strutturale", documento G P R 13

HALL MOHRAZ, NEWMARK "Statistical analyses of earthquake response spectra

ISMES, (1991) "Attivita' interazione suolo struttura, Profili geotecnici di progetto", Prog ASP 5961, RAT-CGA 011

JAMIOLKOWSKI, LANCELLOTTA (1977) "Moderni metodi nella valutazione della capacita' portante delle fondazioni superficiali", atti istituto Scienza delle Costruzioni, Politecnico Torino Nov 1977

JANBU N (1957) "Earth pressures and bearing capacity calculations by generalized procedure of slices" 4th Int Conference on S M F E London

JANBU N , (1972) "Earth pressures computations in theory and practice" Eur Conf S M F E , London

JANBU N , e Al (1976) "Effective stress stability analysis for gravity structures", The Norvegian Institute of Technology

JANBU N , (1979) "Design analysis for gravity platform" 2nd Int Conf of B O S S , London

NEWMARK N M , (1980) "ssrt guidelines for sep soil structure interaction review" Dec , U S NRC Letter for all SEP licensees

SERVA L , (1990) "Normative antisismiche per la localizzazione e la progettazione di centrali elettronucleari", estratto da Sicurezza e Protezione, anno 8 n 22 1990 ENEA DISP

VESIC A , (1975) "Bearing capacity of shallow foundation" 3rd Chap , Foundation Engineering Handbook Edited by H F Winterkorn and Hsai-Yang Fang Van Nostrand Reinhold

WROTH C P , HOULSBY G T , (1985) "Soil mechanics Property characterization and analysis procedures", 11th Int Conf on Soil Mechanics and Foundation Engineering, S Francisco 1985

SAFETY SYSTEM AND ANALYSIS

(Session 3)

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INVESTIGATION OF PASSIVE SYSTEMS FOR THE NUCLEAR POWER INTERNATIONAL PWR

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Abstract

In the framework of the cooperation of Siemens and Framatome, NPI is developing and designing an advanced PWR. During the early conceptual phase a systematic investigation of passive systems for possible implementation in the NPI-PWR was made to assess their merits for a large-sized PWR.

The paper gives firstly an overview on:

- objectives of a possible implementation of passive features,
- areas for evaluation of passive systems and evaluation criteria used.

Secondly, a number of examples of candidates for implementation of passive systems will be presented: the relevant plant conditions and system functions will be outlined, and the main results of the conceptual evaluation will be given.

Finally, the most promising innovative passive system under consideration, a secondary-side safety condenser which could replace the emergency feedwater system will be described and the safety assessment done up to now will be presented

1 INTRODUCTION

In the frame of the cooperation of Siemens and Framatome, NPI is developing and designing an advanced innovative PWR which shall meet the requirements on operation and safety to be raised for the next generation of nuclear power plants in France and Germany From the very beginning it was clear that this development had to be based on the broad design and construction experience existing in Siemens and Framatome and on the positive operational experience of the existing plants in France and Germany.

This evolutionary approach does not fit totally to the U S evolutionary advanced reactor which is mainly characterized by a large power size and active safety systems, by contrast with passive advanced reactors which employ smaller reference sizes and mainly passive means for essential safety functions What is called evolutionary advanced in the United States is to a large degree close to existing designs in France and Germany But it is not only the purpose of the NPI cooperation to harmonize present requirements and design features in France and Germany, but also to introduce essential innovations, mainly for the purpose of better balancing contributors to the core melt risk and of improving the containment function for severe accidents

The reference size for the NPI-PWR was chosen mainly on the basis of economical and grid considerations In any case, a nuclear power plant for the future has to compete with fossil power generation This seems for today's technologies and on the basis of present prices only possible above a power size of about 1000 MW For the integrated grids in western Europe, there is no need to deviate from a power size of around 1400 MW which is already presently successfully operated So the NPI product aims preferably at this power size

The question of performing safety functions by active or passive means is in NPI's view not a principal and autonomous safety concern and also not a general question for the whole innovative plant, but it has to be answered for each of the different technical features of the plant separately, taking into account all advantages and disadvantages of the possible solutions mainly in terms of safety and cost

One of the possibilities for innovation is introducing more passivity compared to existing designs Therefore at the beginning of the product development, independent of the relatively large power size, a systematic review of passive features for a possible implementation in the NPI product was done

2 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

and the associated definition of a passive system is

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

Looking around to the developments of passive reactors worldwide, it seems that they are not strictly based on this definition, but more on a definition like it is 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 also solutions according to the EPRI definition were included in the review

3 PURPOSE OF POSSIBLE IMPLEMENTATION OF PASSIVE SYSTEMS

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 The question for NPI was can the use of passive features be extended without loosing the safety advantages of presently operated PWR plants which mainly consist in the well-known multi-barrier concept and the defense-in-depth approach It seemed advantageous to look for passive features mainly in the defense-in-depth level 3 (accident mitigation) for the purpose of

- introducing higher degree of diversity
- allow for enhanced simplification
- decreasing needs for safety-grade electrical power
- improving safety (probabilistically or radiologically)
- increasing grace periods for operator intervention

This paper deals only with possible passive features for defense-in-depth level 3 But it shall be stressed, that also in level 4 (severe accident mitigation) a potential for implementation of passive systems exists

4 INVESTIGATION OF PASSIVE FEATURES

In Fig 1 a list of principal passive system approaches is given which were investigated for possible implementation in the NPI-PWR These features were subject to a systematic assessment with respect to the following criteria

- simplicity of design
- impact on plant operation
- safety
- cost

The first two assessment criteria concern in more general terms simplicity

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 e g necessary 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, maintenance should not be affected by the passive system Spurious actuation of the passive system 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 And 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 cause a clear safety or economical advantage

Questions related to safety concern e g

- diversification of safety systems and thereby improvement of system and plant reliability
- decrease of accident releases

It should be avoided, that new accident scenarios are introduced by the passive system And the system should fit to the well-proven defense-in depth concept and allow for a gradual response in 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 the passive system

Questions related to economics concern all costs related to the passive system development, verification and qualification including analytical and experimental efforts, as well as plant investment costs

It must be stated that most of the principal passive system approaches listed in Fig 1 had to be dropped already after a rough qualitative investigation

Primary-side RHR

1 (high pressure) heat exchanger connected to primary side passive cooling chain

2 heat up of a pool connected to primary side via density locks

Secondary side RHR

1 condenser connected to secondary side air/water cooling on tertiary side

2 secondary depressurization and passive feed

Primary-side make-up

1 high head safety injection gravity driven from make up tank

2 medium head safety injection by steam injectors

3 medium head safety injection by accumulators

4 gravity driven low head safety injection from tank/sump by primary system depressurization

Containment heat removal

1 metal containment outside cooling (water/air)

2 sump cooler with passive cooling chain

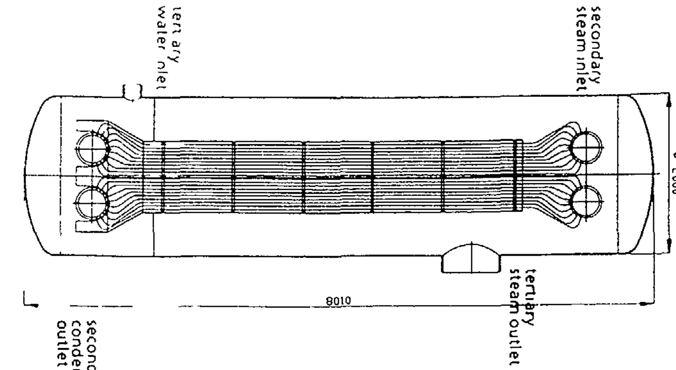
Fig 1 List of Principal Passive System Approaches Investigated

5 SAFETY CONDENSER (SACO)

Only for the secondary side safety condenser, a conceptual development of the component (Fig 2) and the system (Fig 3) was done and a detailed safety and cost assessment is under way

The SACO shall replace the emergency feedwater and atmospheric steam relief systems and provide a confinement of the secondary side steam discharge in case of accidents (e g SGTR)

header design



design data per device (header or tube sheet design)		
number of devices	4	
normal power at t_H = 230 °C	ca 50	MW
power at t_H = 152 °C	ca 15	MW
max sec steam temperature	290	°C
tertiary water temperature	20	°C
tertiary evaporation pressure	< 1 5	bar
tube bundle		
tube outside diameter	40	mm
tube inside diameter	34	mm
pitch (quadratic)	60	mm
number of tubes	351	
pass number tube side	1	
mean tube bundle length	5400	mm
tube bundle diameter	1350	mm
heat exchanger surface	238	m^2

tube sheet design

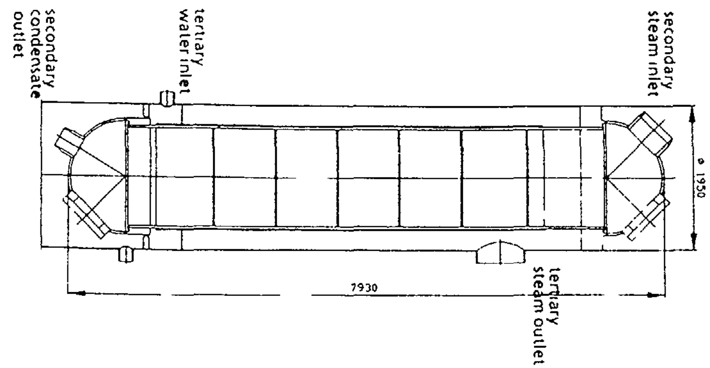


Fig. 2 Safety Condenser Designs and Main Design Data

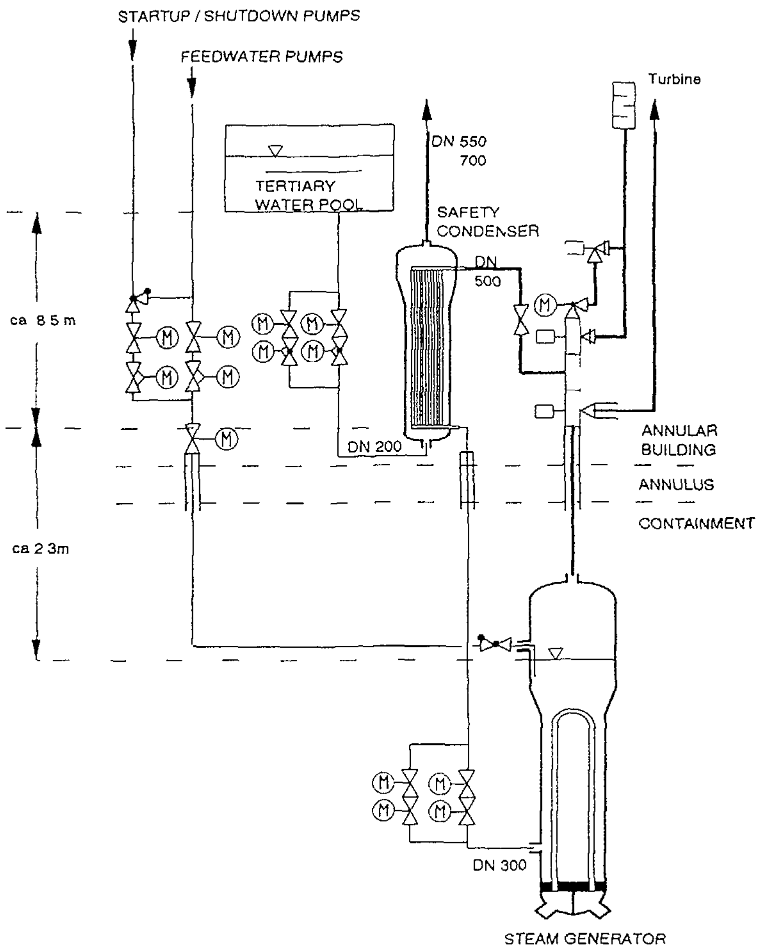


Fig. 3 Safety Condenser, Basic Circuit Diagram

The main functional requirements in the <u>short term</u> are the following

Loss of offsite power, loss of feedwater

By initiation of SACO operation it is the target to remove the heat and thereby prevent the response of secondary-side safety valves If this will not be possible, the response shall be limited to the small and isolatable 15% safety valve in the very short term (≈ first 100s)

Small break LOCA

The SACO shall contribute to heat removal and thereby keep the secondary side at a temperature level enabling primary-side make up by medium-head safety injection (MHSI) at a relatively low pressure head (≈ 80 bar)

Steam Generator tube rupture (SGTR) with coincident loss of condenser

After detection of a SGTR by an activity signal, the SACO shall remove heat from all SGs (including the defective one) to prevent the response of the secondary-side safety valves or at least restrict the response of the 15% safety valve to the very short term After identification and isolation of the affected SG, the associated SACO will have to be cut off Activity release is prevented thanks to a MHSI head well below the secondary safety valve setpoints These measures shall considerably reduce releases in case of SGTR accidents

The main functional requirement in the <u>medium term (< 1 day)</u> is to cool down the reactor to RHR conditions

First safety assessment indicates that these functional requirements could be met

First probabilistic studies show that the reliability of the "passive" SACO depends strongly on the reliability of the active components involved, mainly of the valves on SACO secondary and tertiary side and on sufficient tightness of the main steam and feedwater isolation valves More detailed studies will have to show which measures will be necessary

e g
- diversification of valves
- diversification of valve actuation and/or power supply
- introduction of a passive secondary side make-up

to achieve an overall safety level for secondary side cooling which is compatible with the probabilistic target of the NPI PWR

The overall assessment including costs and comparison with an active secondary side heat removal system is expected in the next year

REFERENCES

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

(2) EPRI ALWR Requirements document

INNOVATIVE SYSTEMS AND COMPONENTS AIMED AT PROVIDING PWRs WITH COMPLETELY PASSIVE EMERGENCY SHUTDOWN AND DECAY HEAT REMOVAL

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Abstract

A new interest for nuclear power generation is expected to start within a near future in several countries. It is time to concentrate research efforts on innovative safety systems able to fit with experienced nuclear designs.
Among the key issues to be considered there are prevention of primary coolant pressure boundary failure, decay heat removal through simple, passive systems, the emergency core shutdown through simple, passive devices.
In the paper, the innovative systems developed for the MARS nuclear plant are described, aiming at i) emergency core cooling for an infinite time, only relying on static mechanical components, one mechanical passive component (a special check valve) and the action of physical laws to perform the protective action, ii) safe core shutdown, through devices sensible to core temperature and able to utilize the core temperature increase and gravity to perform the protective function through passive subcomponents.

INTRODUCTION

The main concern on the safety of nuclear power plants is originated by the impossibility of excluding a-priori an unbalance between power generated and power removed by the core, thus leading to lack of fuel integrity and release of radioactive matter.
Substantial improvements in the capability of the core to maintain its integrity may be achieved through an in-depth re-design of the nuclear plant safety system, utilizing sub-systems and components characterized by a higher standard of reliability in performing the two basic functions of nuclear safety core shutdown and decay heat removal

APPROACH TO SAFETY

Traditional shutdown systems and emergency core cooling systems in PWR's rely on an extended plant operation experience This could lead to the search of proposing only limited modifications to their design in order to achieve higher reliability of intervention and, as a consequence, a lower probability of severe fuel damage. Nevertheless, the approach itself to safety, adopted in their design, prevents achieving substantial improvements through limited modifications, because their intervention is dependent on a complex chain of sensors, instrumentation, active components, which is inherent with the systems' conceptual design.

Hundreds of mechanical and electric components and subcomponents are called to intervene in case of the safety function demand, and the reliability of the global system may be only partially improved through minor simplifications or the introduction of some mechanical passive components instead of active ones
This situation is, on the other hand, confirmed by the vanishing improvement in the fuel damage probability figure, due to the modifications proposed in the last years to traditional shutdown and emergency core cooling systems for PWR's, based on conceptual designs basically unmodified.
That is why if a substantial improvement in the fuel damage probability is desired and is considered as an imperative design goal, only substantial modifications to the design of shutdown and emergency core cooling systems must be proposed. In the following, the main features of completely innovative shutdown and emergency core cooling systems proposed to be utilized in conjunction with traditional PWR nuclear systems are described.

EVENTS LEADING TO FUEL DAMAGE - NEW DESIGN REQUIREMENTS

If the initiating events potentially leading to full damage in PWR's are considered, together with the events which may play a negative role in an accidental sequence, it is possible to identify the issues which should become reference design criteria for innovative shutdown and emergency decay heat removal systems.
The first issue is the protection of the primary coolant boundary so as to avoid (or to reduce substantially the probability of) failures and damages leading to loss of pressure boundary integrity and to loss of coolant. This may be achieved only through the inclusion of the whole primary coolant in a secondary pressurized containment, operating at the same pressure as the primary coolant in order to avoid primary stresses in the pressure boundary, and designed so as to preclude the leak of primary coolant into the secondary pressurized containment in the unrealistic event of pressure boundary failure, and conceived so as to make a potential failure of the secondary containment uneffective with respect to the failure of the pressure boundary.
Once the probability of occurrence of primary coolant loss is abated and the persistence of primary coolant flow is assured in the reactor core, the great part of probability for severe accidents in PWR's, as calculated through PSA's, is eliminated.
The next issue is to provide the plant with a decay heat removal with a drastically improved reliability, in order to allow the primary coolant already assured in the core, play the required role of heat removal.
Events susceptible to prevent the correct operation of emergency core cooling systems include mechanical loads overcoming design values, faults in the construction/assembling process, area events not correctly considered in the design stage, unexpected failures of components and subcomponents, faults in the operation and maintenance procedures. A first-quality design process, relying on experience and suitably sustained by Quality Assurance programs, allows for design errors to be basically eliminated. Nevertheless, even a perfect design cannot eliminate unexpected failures of components and subcomponents, very probable especially in active and energized components, and faults in the operation and maintenance, due to human errors. The only solution, in this case, is the elimination of as many components as possible, which may fail or may be affected by operation and maintenance failures, obviously guaranteeing the requested performance of the system.
If considerable reliability improvements are to be expected, the said philosophy must be brought to the extreme consequences the new design of emergency core

cooling systems shall be based on a few components only (not thousands, not hundreds), no active components shall be allowed, no external energization shall be allowed, a total protection against sabotage shall be required. The new system will have to be extremely simple, with only static and - if not avoidable - passive components.
Natural laws will be called up both in the identification of the protective action requirement and to perform the protection function itself (motion of the coolant and heat removal).
The new design criteria regarding the emergency core cooling system allow to obtain important achievements in fuel damage occurrence probability reduction.
There is a further aspect, though less important, which must deserve a special consideration: the possibility of reducing the probability of core shutdown failures, and thus of ATWS.
Thermal transients in the fuel due to transient power unbalances which may occur during ATWS are to be fight against, in the view of a new generation of nuclear reactors, incorporating advanced features of passive safety. The third issue, therefore, is to provide the plant with an innovative shutdown system to be added to the traditional control rods shutdown system, and designed to satisfy the requirement of core shutdown independently from the operation of the traditional reactor protection system. The innovative shutdown system shall be sensible to thermal transients within the core and shall perform the protective action through a few passive components only, directly driven by natural laws.

TABLE 1

MARS (MOD. 4)
REACTOR CHARACTERISTIC DATA

RATED POWER	600	MWt
FUEL THERMAL POWER	582	MWt
INLET CORE COOLANT TEMPERATURE	214	°C
OUTLET CORE COOLANT TEMPERATURE	244	°C
RATED PRESSURE	72	bar
FUEL ROD EXTERNAL DIAMETER	0.98	cm
FUEL ACTIVE LENGTH	260	cm
FUEL RODS ARRAY	15x15	
FUEL RODS PER FUEL BUNDLE	204	
RODS PITCH	1.3	cm
FUEL BUNDLES	96	
CORE EQUIVALENT DIAMETER	216	cm
CORE HEAT TRANSFER SURFACE	1564	m^2
CORE LINEAR POWER DENSITY	114.3	W/cm^2
CORE AVERAGE THERMAL FLUX	37.2	W/cm^2
CORE AVERAGE VOLUMETRIC POWER DENSITY	63	kW/l

THE INNOVATIVE EMERGENCY CORE COOLING SYSTEM

The innovative emergency core cooling system described in the following has been designed for the MARS plant (Table 1 includes the main data of the plant). It allows removal of decay heat from the reactor core by switching primary coolant flow from the user loop (including circulation pumps and steam generator) to

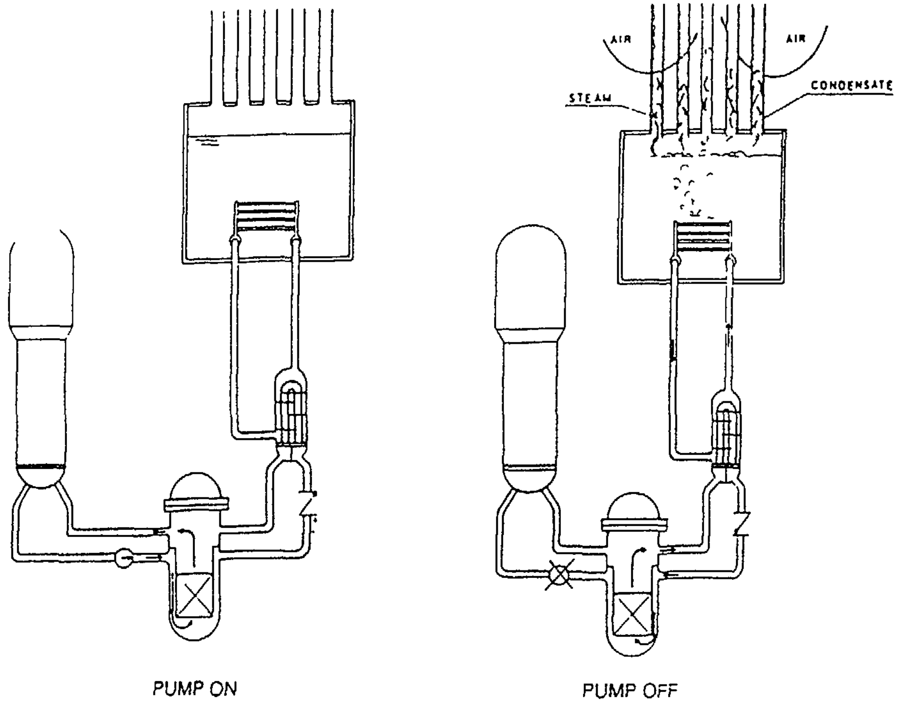
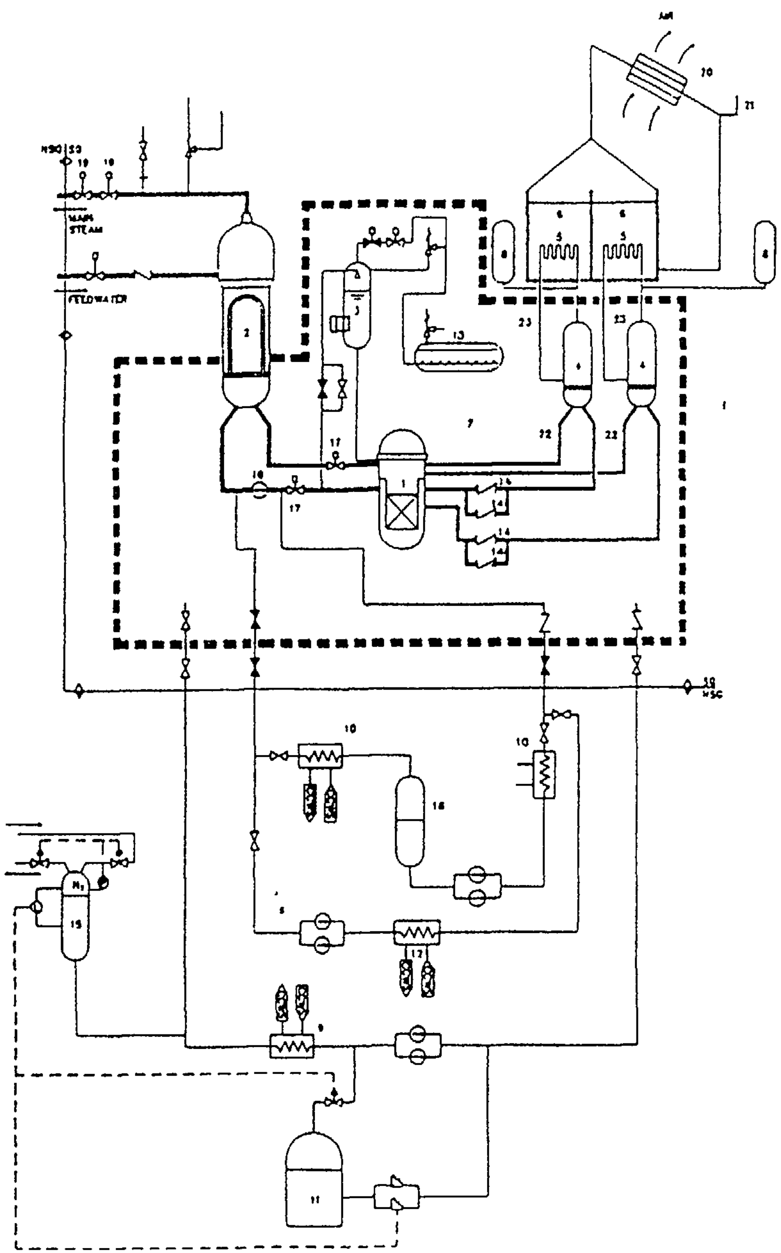


FIG. 1 - OPERATION PRINCIPLE OF MARS ECCS

another loop, where gravity, through thermal gradients, assures the circulation of the primary coolant and of secondary fluids (fig. 1). A cascade of loops transfers the heat from the primary system to the external air, allowing for an indefinite cooling capacity. The switching effect of the primary coolant from the user loop to the emergency core cooling system relies on a special check valve (item N° 14 in fig. 2), which is an innovative component, driven by gravity and designed so as to guarantee a very high reliability on demand. The special check valve will open in any case following a reduction of flow delivered by the user loop circulation pump (item N° 16 in fig. 2).
The emergency core cooling system designed for the MARS reactor plant includes two trains; each train may remove the 100% of the decay power. In an accidental event (e.g., station black-out or steam generator feed-water loss) its activation is automatic and the operation of the system is completely passive.



1) REACTOR
2) STEAM GENERATOR
3) PRESSURIZER
4) HEAT-EXCHANGER (REACTOR COOLANT/INTERMEDIATE COOLANT)
5) HEAT-EXCHANGER (INTERMEDIATE COOLANT/FINAL HEAT SINK COOLANT)
6) WATER RESERVOIR
7) PRIMARY CONTAINMENT SYSTEM
8) INTERMEDIATE LOOP PRESSURIZER
9) HEAT-EXCHANGER (PRIMARY CONTAINMENT WATER COOLING SYSTEM)
10) CHEMICAL AND VOLUMETRIC CONTROL SYSTEM HEAT-EXCHANGERS
11) WATER STORAGE TANK
12) RESIDUAL HEAT REMOVAL SYSTEM HEAT-EXCHANGER
13) PRESSURIZER RELIEF TANK
14) SAFETY CORE COOLING SYSTEM CHECK VALVE
15) PRIMARY CONTAINMENT PRESSURE CONTROL SYSTEM PRESSURIZER
16) MAIN COOLANT PUMP
17) PRIMARY LOOP ON/OFF VALVE
18) CVCS TANK
19) STEAM LINE ON/OFF VALVE
20) ULTIMATE HEAT SINK CONDENSER
21) COMMUNICATION PATH WITH THE ATMOSPHERE
22) SAFETY CORE COOLING SYSTEM PRIMARY LOOP
23) SAFETY CORE COOLING SYSTEM INTERMEDIATE LOOP

FIG. 2 - MARS PRIMARY COOLANT SYSTEM FLOW DIAGRAM

Each train includes:

- a Primary Safety Cooling loop (PSC)
- an Intermediate Safety Cooling loop (ISC)
- a reservoir of cold water (pool) with piping which brings the steam produced to a special atmoshperic-pressure-condenser, cooled by environment air through natural circulation (Pool and Condenser Loop, PCL).

The PSC loop is directly connected to the reactor vessel through 16' piping. In case of system intervention, after a first transient phase, the forcing head for the circulation of the coolant is assured by a difference in level of about 7 m between the vessel outlet nozzle and the primary heat exchanger (item "4" in fig. 2). The heat exchanger that transfers heat from PSC to ISC is a shell-and-tube heat exchanger with vertical straight tubes.
The tubes (about 2000) are 3/4" in diameter, 3 m in length, stainless steel type. In this heat exchanger, the PSC fluid, at a high pressure (72 bar), flows tube-side. The difference of level for the natural circulation in the ISC loop is of about 10 m. The pressure in the ISC loop is 72 bar and has been selected in order to guarantee sub-cooling water conditions of the fluid during any accidental situation or transient. This value of the pressure is maintained by means of a pressurizer connected to the hot leg of the loop. The heat exchanger (item "5" in fig. 2) which transfers the heat from the ISC circuit to the water of the emergency reservoir is realized by means of a tube bundle submerged in a pit filled with water, which is the main water reservoir pool (see fig. 2).

In the MARS plant, during normal operation of the plant, fluid circulation in each PSC loop is prevented by two special check valves on the cold leg.
The emergency water reservoir is built in reinforced concrete, internally lined by stainless steel, and is designed to guarantee the core cooling for a theoretically infinite period.
The reservoir water flows through the pit owing to local recirculation, absorbing heat from the ISC loop; its temperature rises and then it vaporizes.
Steam produced is mixed with air initially present in the dome over the pool. The pressure in the dome tendentially rises and this causes a flow of the mixture (air + steam) towards a connection path with the atmosphere (item "21" in fig. 2). Between the pool dome and the connection path is an inclined tubes heat exchanger (item "20' in fig. 2), where steam is partially condensated. With the progress of vaporization in the pool, the air content in the dome decreases, and the blanketing effect of air on condensation is reduced, thus causing an increase in the condensation rate in the heat exchanger, which is cooled by external air in natural circulation. After a short transient from the beginning of vaporization, the heat transfer capacity of the condenser is such that practically all the steam produced in the pool condenses, no further steam losses occur through the communication path, and the condensate produced goes back to the pool.
The performance of the system allows, without presence of pumps, motors, special diesel generators, or any other active component, a core cooling in safe conditions.
The operation of the emergency core cooling system relies on mechanical components only (no electronic or electric device is needed), no active components are foreseen, and only one component is not static, the innovative, special check valve. Even if PSA of MARS reactor shows that the probability of failure of the emergency core cooling function is several orders of magnitude lower than in traditional PWRs' design even if traditional check valves are used in the innovative MARS emergency core cooling system, nevertheless a special care was devoted to the design of an innovative check valve, because it is the only not static component involved in the operation of the system. In the MARS plant, the valves are four in number, in parallel on two trains, due to reliability reasons (see PSA), but functionally one valve only could perform the safety function.
During normal operation of the plant, the head delivered by the primary pump causes the special check valves to remain in a closed position, when, due to the primary pump coast-down (whatever its cause), the difference of pressure between inlet and outlet of the core decreases, the force which maintained the check valves closed is not sufficient anymore, and the weight of the plug makes it fall, thus opening the flow path through the valve.
The special horizontal check valve, shown in fig. 3, at the moment selected for the installation in the MARS plant, performs the closure of the plug using the force exerted by an active component, as described below. Then it remains in closed position owing to the difference of pressure created by the main coolant pump. When this difference of pressure decreases during a main coolant flow decrease, the considerable weight of the plug opens the valve completely.
A small-size connection exists between the upper plenum of the valve and the ECCS piping (towards the heat exchanger).
One of the systems analyzed to perform the closure of the valve is based on a difference of pressure generated between the suction of the valve and a plenum below the valve, whose pressure is increased during the action of closing the plug, relying on a small-size centrifugal pump. After the valve closure, when the primary pump is delivering the rated flow and before bringing the core to

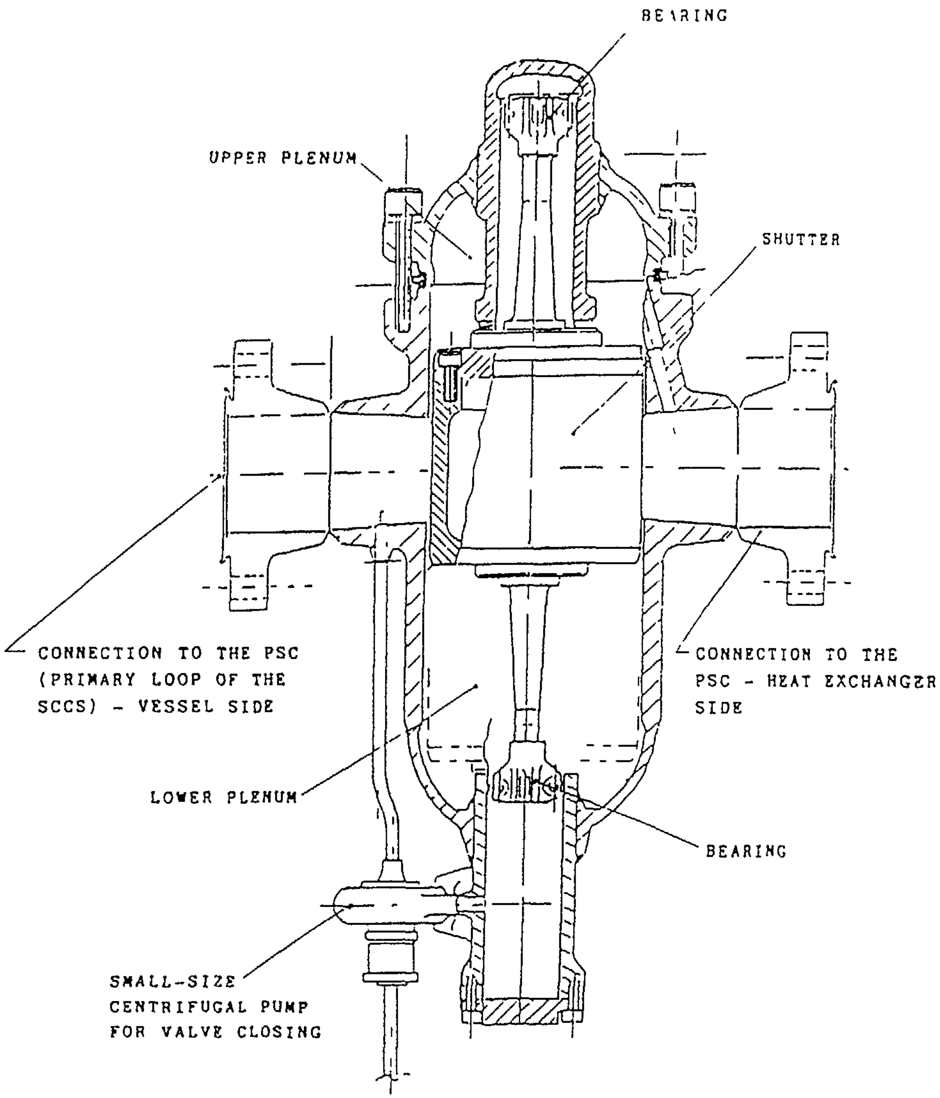


FIG. 3 - HORIZONTAL-TYPE SPECIAL CHECK VALVE

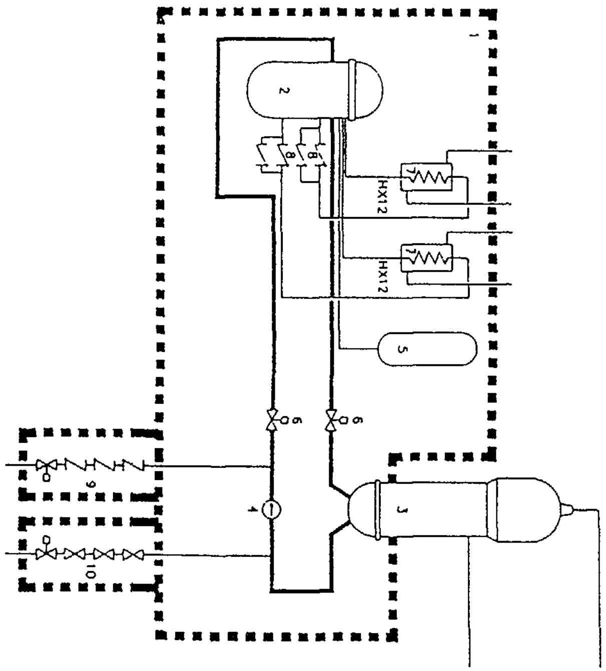


FIG. 4 - PRESSURIZED CONTAINMENT FOR PRIMARY SYSTEM PROTECTION

(1) PRESSURIZED CONTAINMENT (CCP)
(2) REACTOR VESSEL
(3) STEAM GENERATOR
(4) PRIMARY PUMP
(5) PRESSURIZER
(6) PRIMARY SYSTEM ISOLATION VALVES
(7) SAFETY COOLING SYSTEM HEAT-EXCHANGER
(8) CHECK VALVES
(9) CVCS INLET CONNECTION
(10) CVCS OUTLET CONNECTION

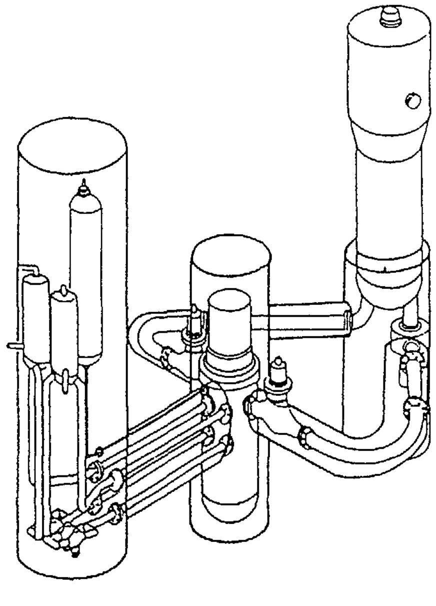


FIG. 5 - PRIMARY COOLANT SYSTEM ENCLOSED IN THE PRESSURIZED CONTAINMENT (CPP)

criticality conditions, the pressure in the lower pressurized plenum is brought to the same value of the primary coolant, through the disconnection of the motor moving the small centrifugal pump. This disconnection must be assured during plant operation. The connecting piping containing the operating centrifugal pump assures the same pressure between the vessel-side PSC piping and the lower plenum of the valve during the normal operation of the reactor.
The characteristics of this design are: the very wide range of selection of the weight of the plug which may allow to freely select a value of the main coolant flow set point for the valve opening.
Preliminary tests on prototypes showed an excellent performance and a very high reliability.
The described, innovative emergency core cooling system is based on the presence, in the reactor vessel, of the primary coolant even in accidental conditions. In the MARS plant, this has been achieved through the inclusion of all the primary coolant pressure boundary in a pressurized containment (CPP).
The MARS CPP is filled with water at the same pressure as the primary coolant, at 70 °C.
In figs. 4 and 5, the simplified sketch and a schematic view of the CPP are reported. This solution is aimed at a global easy disassembling of all components. All components of CPP are in carbon steel, may be built and tested in shop and easily transferred and assembled in site. The special bends are placed on rails in order to simplify the operations of connection and disconnection.

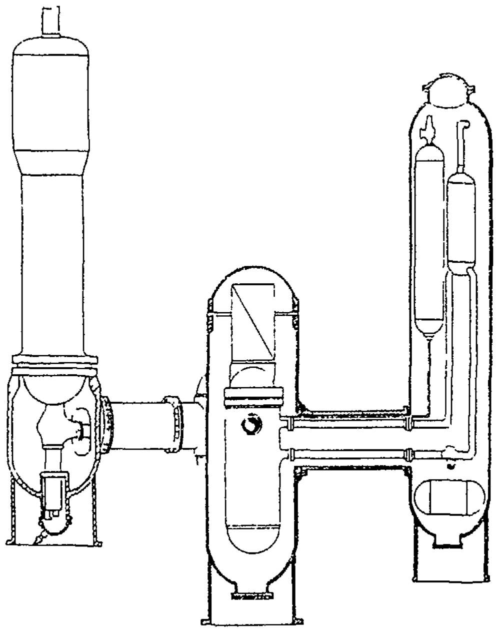


FIG. 6 - PRIMARY SYSTEM AND PRESSURIZED CONTAINMENT

Special mention should be made to the special coupling with the shell of the steam generator, which allows the realization of, also through this component, the double barrier enclosing the primary coolant (the shell of the steam generator is designed to withstand the total pressure of the primary coolant (72 bar)).

A slow flow of water is maintained in the CPP, in order to realize a uniformity in its temperature and in order to make easy, though special check points, the detection of potential small losses of primary flow into the CPP, through the flanged couplings.

The detection is made possible by small underpressure of the CPP boundary with respect to the primary coolant pressure The small difference in pressure is also the reason why the water in CPP is not borated.

Should the concept of the innovative emergency core cooling system be considered for application to PWR's without the enclosure in a pressurized containment, modifications may be proposed to guarantee the coolability of the reactor core, even after accidents involving a break of the primary coolant pressure boundary.

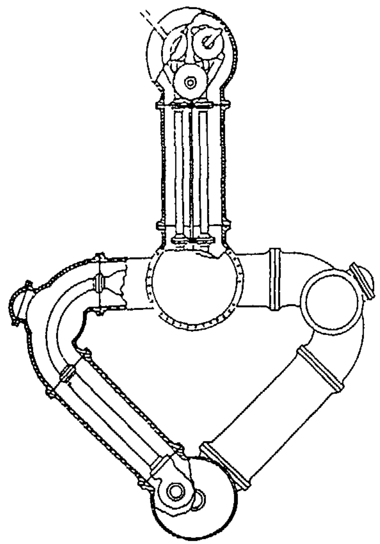


FIG. 7 - PRIMARY SYSTEM AND PRESSURIZED CONTAINMENT

THE INNOVATIVE EMERGENCY SHUTDOWN SYSTEM

In the development of the design of the MARS PWR plant, an innovative design criterion was adopted, regarding the provision of an additional special shutdown system, completely passive and relying on physical laws only, aiming at avoiding ATWS accidents. Therefore, in the MARS plant, two coexistent and mutually independent scram systems have been selected. The first one is a traditional rod cluster type PWR control system, driven by magnetic jack activation and by gravity. Independent from this is an additional, special scram system (ATSS), which was studied and designed to provide the automatic and safe shutdown of the reactor as soon as the fluid temperature in the core rises over a selected maximum operation temperature.

This special scram system eliminates the occurrence of ATWS accidents The reactivity control of the additional scram system is obtained through the insertion, into the core, of control rod clusters with the same geometrical and physical caracteristics as the traditional scram system. The difference between the two systems is in the type of actuator selected.

In order to assure the required inherent safety of actuation, each control rod cluster of the innovative scram system is controlled by a special actuator, based on a simple physical principle: the thermal expansion of a rod, due to the variation of temperature of the core coolant, which leads to the disconnection of hooks holding the control rod cluster.

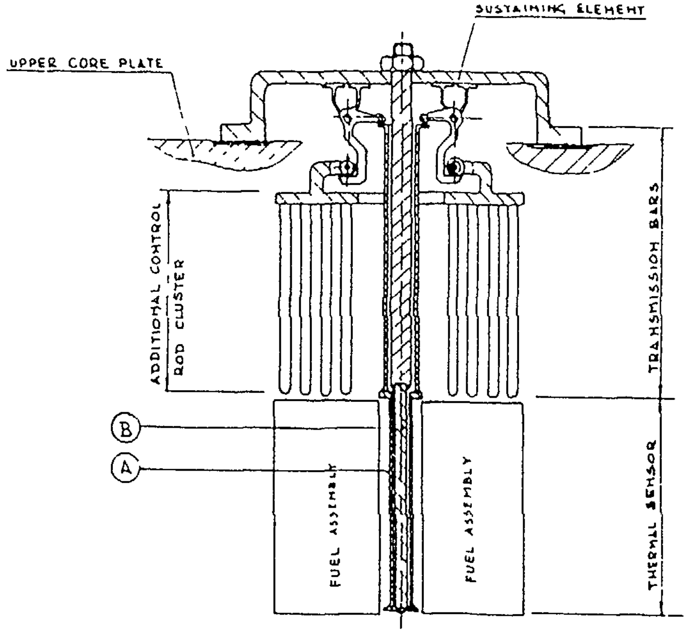


FIG. 8 - WORKING SCHEME OF THE ATSS SENSOR

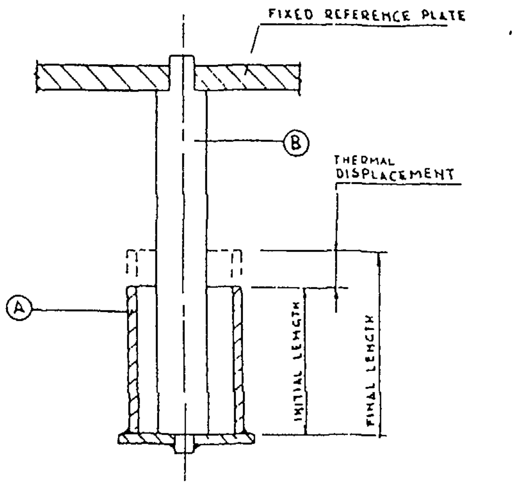


FIG. 9 - WORKING SCHEME OF THE ATSS SENSOR

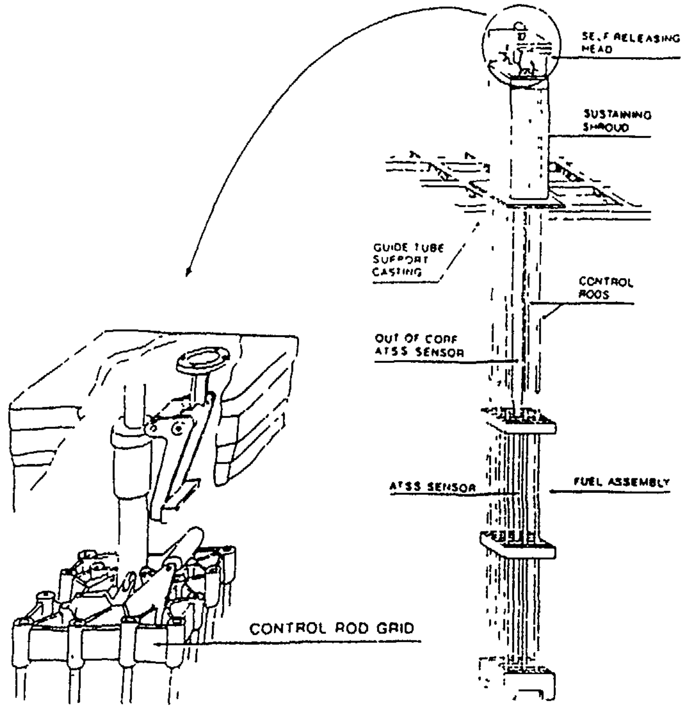


FIG. 10 - INNOVATIVE PASSIVE SHUTDOWN SYSTEM

The geometry and the design of the innovative actuator have been selected so as to assure a considerable mechanical force to operate the mechanical hooks.
In order to describe functionally the proposed new device, two bars, A and B, should be considered, made up by materials with a different thermal expansion coefficient, connected in the lower extremities, as in figs. 8 and 9.
If the upper extremity of B is fixed (on the structure of reactor vessel internals) and that of A is maintained free, owing to a temperature variation the A edge will exhibit a displacement "δ" referred to the B edge.
The differential expansion for a uniform heating is a thermometric parameter correlated to the absolute local temperature and, consequently, it is possible to select the position of the bar edges in order to obtain an expected action, at a selected core temperature.
The geometry of the ATSS sensor for the MARS reactor has been chosen with the aim of being introduced in the fuel element, having the dimensions of a fuel rod; the two-metal sensor is realized using a square section bar, concentric to a 9.6 mm O.D. tube.

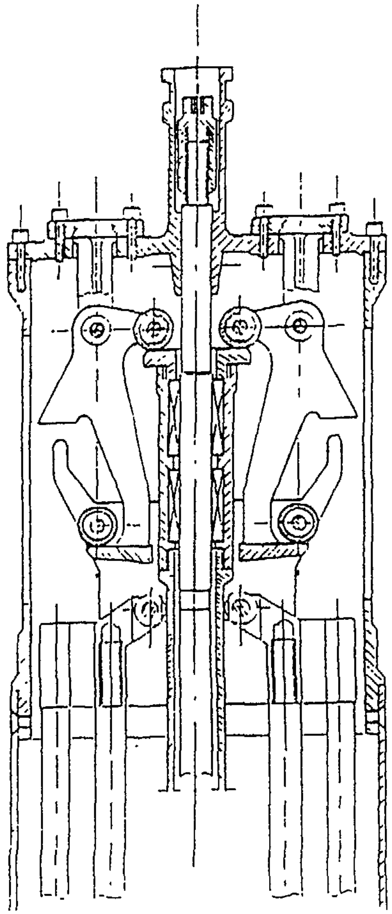


FIG. 11 - ATSS SELF-RELEASING HEAD

The sensors are introduced into the center of fuel assemblies which are expected to host the control rod clusters; with respect to "normal" control rod clusters, the group of control rods in the ATSS cluster is reduced by one unit, i.e. 24 control rods instead of 25 (each cluster).
Coupled with the ATSS sensor located in the fuel assembly, are two transmission elements, aimed at transporting the differential expansion up to the upper extremity, where the control rod disconnection device in placed.
As mentioned, the actuator has been designed so as to use the mechanical force developed in the sensor itself to open directly the hooks sustaining the ATSS control rod cluster.

CONCLUSIONS

Improvements may be achieved in safety of fission nuclear plants, also with reference to traditional designs of nuclear systems. Key issues which should deserve a special consideration concern : i) the prevention of primary coolant pressure boundary failures; ii) the emergency core cooling based on innovative, simple and more reliable systems and, iii) the adoption of dedicated additional core shutdown systems, based on the operation of passive components only, actuated by physical laws. Some of the advanced nuclear designs of PWR's are not compatible with the prevention of primary boundary failures, while the applicability to other PWR's of the innovative emergency core cooling system and of the innovative shutdown system developed for the MARS plant design is likely to be considered.

STEAM INJECTORS AS PASSIVE COMPONENTS FOR HIGH PRESSURE WATER SUPPLY

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Abstract

Steam injectors are static devices in which steam and subcooled water are mixed together and the outflowing subcooled water can reach a pressure significantly higher than both inlet pressures. A specific application, addressed in this paper, concerns the use of this kind of device in Advanced Light Water Reactors (ALWR's) for high pressure makeup water supply; this solution would take advantage of the available steam energy without introduction of any rotating machinery.
The design requirements of this system were defined according to the following considerations:
- water discharge pressure must be at least ten percent more than steam pressure;
- the device will have to operate in a quite wide range of steam pressures (e.g. from 9 MPa down to 1 MPa or less).

At present, two different calculation approaches were developed and used to evaluate the feasibility of a steam injector for this application:
- an overall control volume analysis;
- a more detailed, one-dimensional analysis.

The results, summarized in the paper, confirm the capability of a suitable steam injector to satisfy the mentioned requirements. An instrumented prototype, at present in the design phase, will be used for an experimental activity in order to confirm the theoretical results.

1. INTRODUCTION

The Steam Injector, hereinafter shortly named SI, is a device where a subcooled liquid and its vapour are mixed producing a flow of still subcooled liquid; a special feature is that the outflowing fluid may reach pressures higher than both inlet fluids. The operating principle is in some way similar to conventional ejectors, where a high pressure (driving) and a low pressure (driven) fluid are mixed and the outlet pressure is intermediate between inlet pressures. The distinguishing SI feature is steam condensation that makes available a large heat amount; it can be partly converted into mechanical work useful for pumping the liquid. In this respect, SI can be regarded as equivalent to different devices, like turbine-driven pumps, where steam thermal energy is used to pressurize a liquid; in comparison to this kind of equipment, the main difference is that in SI all thermodynamic processes rely on direct contact transport phenomena (mass, momentum and heat transfer) between fluids, not requiring any moving mechanism. According to the described features, SI can be suitable to different applications, provided that:
- a steam supply is available;
- high pressure, hot water flow is required.

Indeed, SI applications can be found as boiler feed pump for ships and locomotives and for high pressure hot water supply in the food industry; studies have also been found about the possible use in LMHD (Liquid-metal Magneto-HydroDynamic) energy conversion. In recent years, a new nuclear reactor generation has been the subject of intensive research and development activities; the main objective is to design nuclear power plants in which equipment and systems are simplified, and that can be considered safer because their emergency systems operate according to passive features (the different levels of system passivity in Advanced Light Water Reactors (ALWR's) are thoroughly discussed in (1)). In connection with this new reactor generation, the attractiveness of SI's is quite evident, because an high pressure water supply can be essential to very important emergency functions (like emergency core cooling, feedwater supply for decay heat removal and so on) and usually a steam supply is easily available in power plants. Moreover, SI can be regarded to a great extent as a passive system, as it does not require any external energy supply or moving mechanical part.

ENEL, the Italian Electricity Generating Board, which is engaged in a wide range of activities concerning ALWR's, decided to evaluate the applicability of SI in this field; so CISE, which is a research company property of ENEL, was charged to perform a feasibility study in cooperation with ENEL itself, including the following steps:
- review of existing experience on SI's;
- identification of design requirements for a specific application to ALWR's problems;
- set-up of a theoretical model to be used for a preliminary SI sizing.

At present these activities have been completed; the obtained results, described in this paper, appear quite encouraging so that an experimental activity on a scaled SI model is now in preparation.

2. DESIGN REQUIREMENTS

The application of Steam Injectors as basis for high pressure emergency water supply could be attractive in different ALWR's plant configurations, both of the boiling water and the

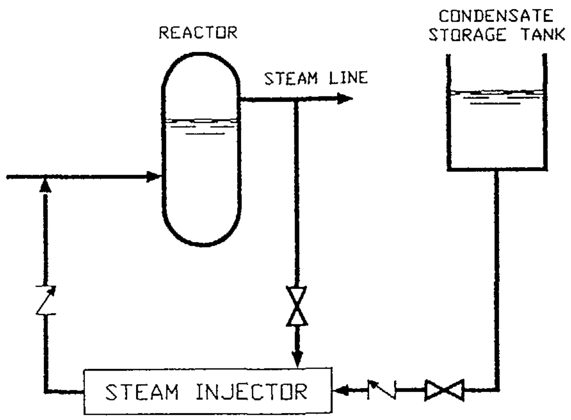

Fig. 1 Schematic arrangement of the steam injector system for SBWR application

pressurized water type. For the purpose of the present study, SI use for high pressure injection system of SBWR's was assumed as a reference. In this case, the driving fluid would be steam present in the reactor vessel, while the cold water supply could be provided by any large water reservoir, operating at atmospheric pressure, available in the power plant (for example the condensate storage tank). The schematic arrangement of the steam injector system is shown in fig. 1. The main design goal is to reduce the ADS (Automatic Depressurization System) actuation frequency, especially in connection with small-break LOCA's; at the same time, plant safety behaviour would be improved.

According to small break LOCA features, a relatively small water flowrate at high pressure is required. This need could be satisfied also by different systems, based for example on electrical or steam turbine driven pumps, but a solution making use of the SI would provide an higher passivity level. In order to establish a precise reference condition for the present feasibility study, the maximum break size and the break location for SI system design have to be specified. So the general requirement assumed is the following:

1) The SI system must be able to supply the reactor system with a water flowrate corresponding to a 2 inches sch. 80 pipe break in the bottom region of the reactor vessel.

Obviously, the range of reactor system conditions to be considered in SI design must also be specified. In this respect, the choice has been to cover a reasonably wide spectrum; so the following additional requirement has been established:

2) The SI system must satisfy the requirement 1) for reactor system pressure ranging form 1 to 8.7 MPa.

For SI system preliminary design, the requirements 1) and 2) must be expressed in a more practical form, establishing the water flowrate and pressure required for any reactor system pressure. The flowrate values, estimated according to the requirement 1), are shown in tab. 1, while for the water pressure the assumed criterion is 10% more than steam pressure, considering that it is sufficient to overcome the pressure drops along the steam and water piping. As far as the cold water reservoir is concerned, the following conditions have been assumed:

- water temperature 25°C;
- water pressure 0.17 MPa.

Finally, the steam at SI inlet is assumed to be dry satured.

TABLE 1 Required water flowrate Q versus reactor system pressure. Notice that the indicated values refer to the water flow taken from the storage tank and do not include the steam flow condensed in the SI

P_{vo} (MPa)	Q (kg/s)
8.7	62
8	54.4
6	50
4	40
2	23.3
1	15

3. STEAM INJECTOR DESCRIPTION

The SI system, in addition to the SI itself, will include all the necessary piping, valves, instrumentation and controls; as this equipment can be considered more or less conventional, all the attention will be devoted to the SI.

A steam injector sketch is shown in fig. 2. It can be subdivided into the following regions:

a) steam nozzle, producing a nearly isoentropic expansion and partially converting steam enthalpy into kinetic energy; it can be a sonic nozzle or, if a stronger expansion is required, a supersonic nozzle with the typical converging-diverging shape;

b) water nozzle, producing a moderate acceleration and distributing the liquid all around the steam nozzle outlet;

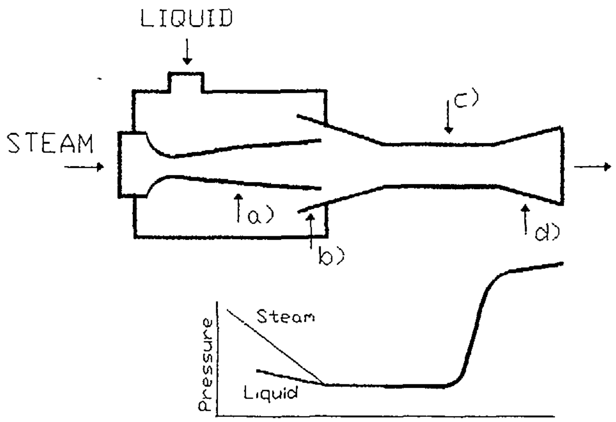

Fig. 2 Steam injector schematic arrangement: a) steam nozzle, b) water nozzle c)mixing section d)diffuser

c) mixing section, where steam and water come into contact. Steam transfer to water heat (because of temperature difference), mass (because of the related condensation) and momentum (because of velocity difference). The final result is the complete steam condensation, with an outflowing subcooled liquid at relatively high pressure. The shape of the mixing section is usually converging, for reasons explained later on;

d) diffuser, where the liquid kinetic energy at mixing section outlet is partially recovered producing a further pressure rise.

The pressure variations along the steam injector are qualitatively shown in fig. 2. It must be noticed that the arrangement of steam and water nozzles could be inverted, creating a circular water nozzle and an annular steam nozzle. However, the arrangement shown seems more convenient because it minimize the perimeter of the steam nozzle and avoids the immediate contact between steam and mixing section walls; in this way viscous dissipations are reduced.

4. CALCULATION MODELS

4.1. Modelling needs

In order to perform a preliminary sizing of a SI able to satisfy the above defined requirements, and to estimate its performaces under different operating conditions, some calculation models of the involved thermodynamic and transport phenomena are required. In this respect the four different SI regions (steam and water nozzles, mixing section, diffuser) can be separately considered. For the most challenging situation, concerning the mixing section where complex interactions between steam and water take place, the development of special calculation procedures was felt necessary; for the other regions, the following simplified approaches were used:

- expansion in the steam nozzle is considered an isoentropic equilibrium process, that can be evaluated by application of the usual mass and energy conservation laws and by steam table use; because of the saturation conditions at the nozzle inlet, the expansion occurs in the wet region, so the hypothesis of an homogeneous steam-water flow has been assumed;
- expansion in the water nozzle is considered an isoentropic process too, neglecting head losses;
- the diffuser has not been modelled at all; its behaviour, expressed by the diffuser outlet pressure, is bounded by the static pressure (corresponding to zero diffuser efficiency) and the total pressure (corresponding to efficiency equal to one) at the diffuser inlet.

For the mixing section, two different calculation approaches were used:

- an overall control volume analysis (OCVA);
- a more detailed, one-dimensional analysis.

The two approaches, that took advantage of previous works (2), (3), will be shortly described in the following sections.

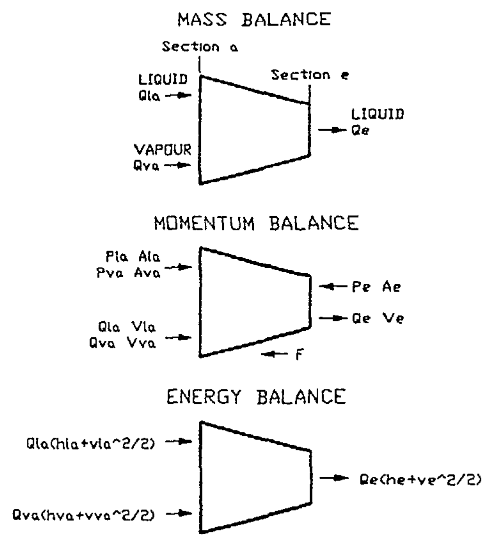

Fig. 3 Overall control volume analysis: conservation equations

4.2. Overall control volume analysis

The considered control volume is the mixing section, included by its inlet section a and its outlet section e (fig. 3).

The following hypotheses have been assumed:

a) steady-state conditions;
b) no wall friction;
c) adiabatic walls;
d) one-dimensional flow (no radial gradient of temperature, pressure or velocity in each phase).

Mass, energy and momentum conservation equations have been written between section a and e, as shown in fig. 3. Moreover, the continuity equation, written for section e, is:

$$Q_e = \rho_e V_e A_e$$

The duct geometry (A_{la}, A_{va}, A_e) and the fluid inlet condition being known, the four equations can be solved to get the outlet conditions (P_e, V_e, H_e, Q_e) provided that an estimate is available for the axial component of the wall forces. A simplifying assumption of constant pressure along the converging stretch of the mixing section has been made, bringing to the following expression:

$$F = P_{la}(A_a - A_e)$$

This assumption has been subsequently verified by the one dimensional approach.

After the calculation of outlet conditions, the liquid subcooling at mixing section outlet must be verified:

$$he \leq h_{ls}\ (P_e)$$

4.3. One-dimensional analysis

The OCVA, very useful for quick performance evaluations, suffers from some limitations:

- the pressure profile must be arbitrarely estimated;
- the effect of geometric parameters like mixing section lenght and flow area variation along the axis cannot be taken into account.

In order to overcome these shortcomings, a one-dimensional model have been established. In addition to hypotheses a) c), and d) reported in section 4.2. (wall friction can be easily introduced in this case) the additional assumption of annular flow pattern with a smooth steam-water interface has been made. Moreover, the momentum transfer from steam to water is related to the condensation mass transfer, neglecting viscous stresses. The conservation equation have been written for three elementary control volumes (steam, water, interface) as shown in fig. 4. They are solved starting from the inlet section, using an iterative technique. The calculation usually stops when a sharp pressure rise (condensation shock) is found.

The calculation procedure has been implemented on a personal computer; a typical run (a few hundred axial nodes) takes about 15 minutes calculation time.

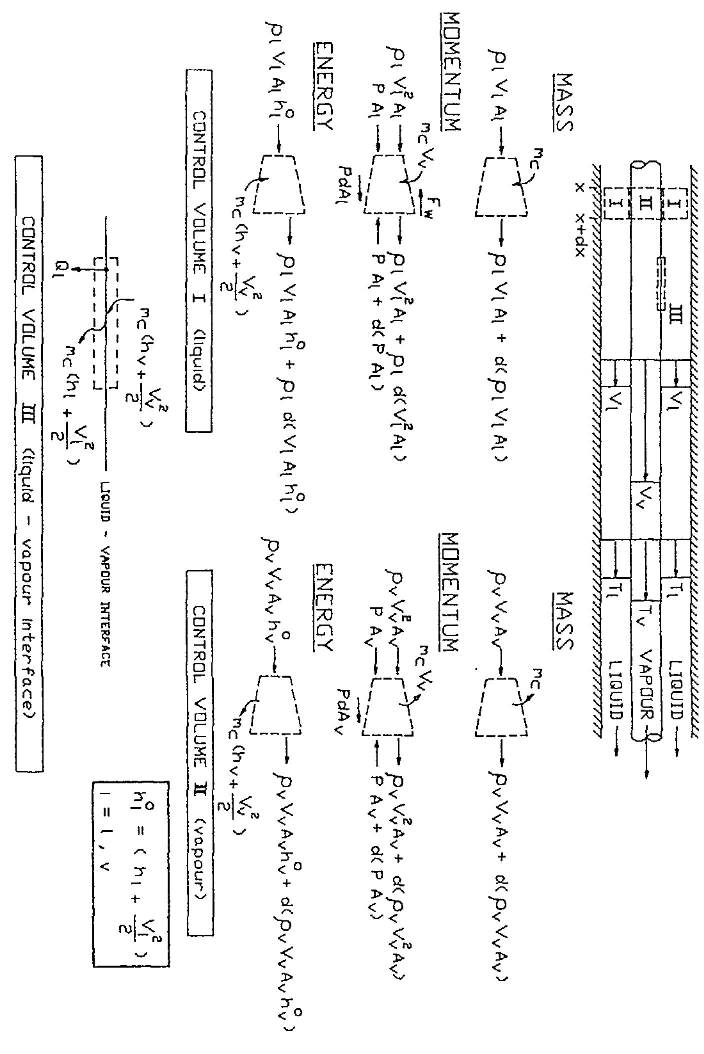

Fig. 4 One-dimensional analysis: conservation equations

5. CALCULATION RESULTS

A large number of calculations, using both the OCVA and the one dimensional model, have been performed in order to clarify the effect of different design parameters (like steam nozzle expansion ratio, mixing section area contraction, mixing section convergence angle, steam/water flowrate ratio and so on). Some highlights of the most significant results will be given here.

A first analysis, performed by the OCVA, has concerned the SI performances at varying mixing section contraction ratios. The results are sinthetized in fig. 5, where both the static and the total pressure at mixing section outlet are plotted. As can be seen, the outlet pressure satisfies the design goal (P_{min} equal to steam pressure plus 10%) for contraction ratios less than 0.1. This result can be understood by considering the momentum equation (fig.3): other things being constant, the smaller A_e is, the greater P_e is obtained. At very small A_e/A_a ratios the static pressure P_e shows a maximum, because the kinetic energy term becomes larger and larger, while the total pressure P_{oe} seems to increase indefinitely (remember that in OCVA wall friction is neglected).

A possible critical point of the assumed design requirements is the very wide variation of the operating steam pressure. Being the pressure at mixing section inlet approximately constant, that means that the steam nozzle expansion ratio should have large variations (let say 87/1 to 10/1 if pressure at mixing section inlet is assumed to be about atmospheric). On the other hand, a supersonic nozzle is characterized by a well defined expansion ratio: if the back pressure is too low in respect to

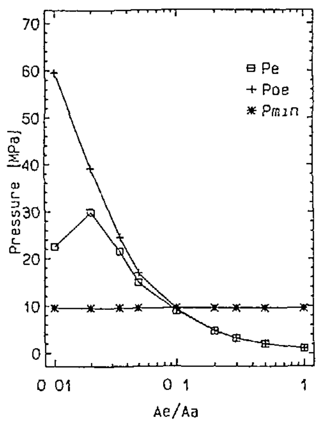

Fig. 5 OCVA approach: effect of mixing section contraction ratio

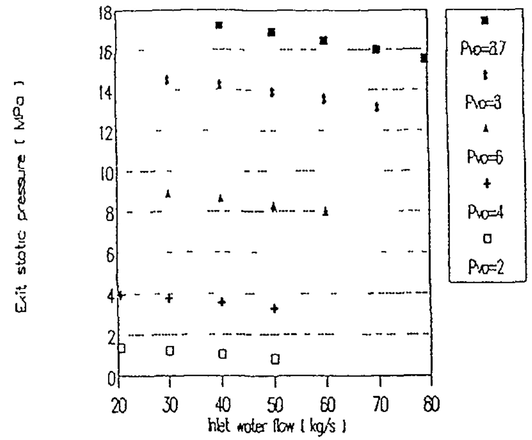

Fig. 6 OCVA approach: SI performance at different steam pressures, for steam nozzle expansion ratio 87/1

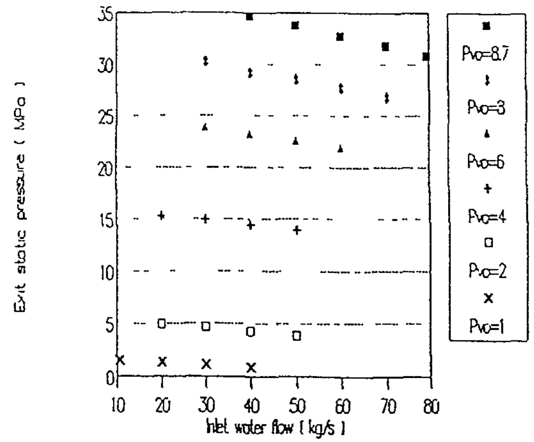

Fig. 7 OCVA approach: SI performance at different steam pressures, for steam nozzle expansion ratio 40/1

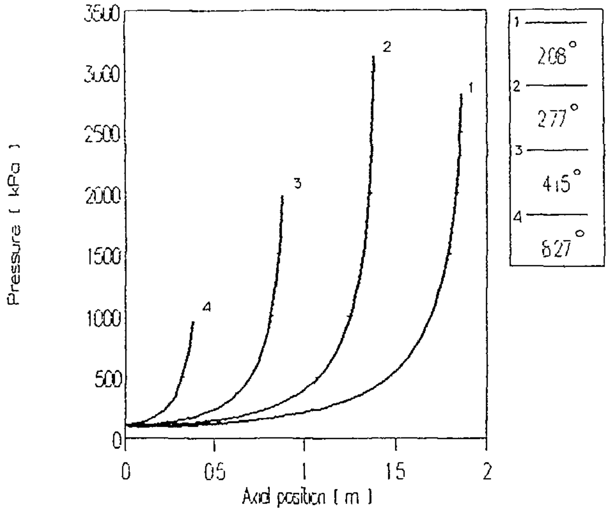

Fig. 8 One dimensional analysis: effect of the mixing section convergence angle

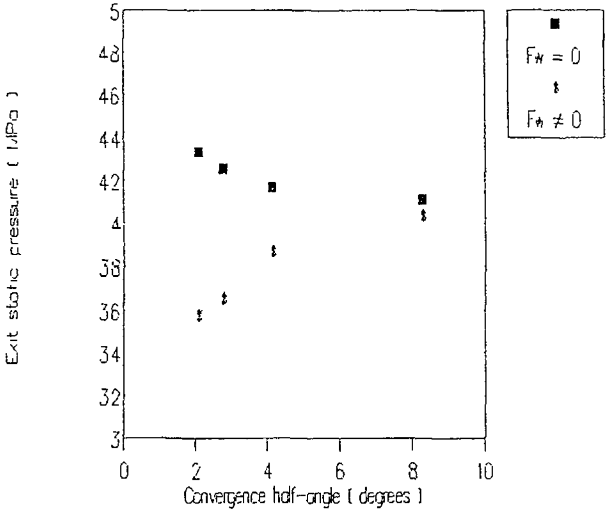

Fig. 9 Effect of the mixing section convergence angle, analyzed by combined use of one-dimensional analysis and OCVA

this ratio, a free expansion occurs downstream the nozzle, while in the opposite case a compression occurs in the diverging nozzle section, involving complex shock wave phenomena. Therefore the question arise what is the best steam nozzle expansion ratio in order to get acceptable performances over all the required steam pressure range. A preliminary answer to this question has been obtained by use of the OCVA again: the results are shown in figs. 6 and 7. Fig. 6 represents SI performance for steam nozzle expansion ratio 87.1; exit pressure is not acceptable for steam pressures less than 4 MPa. In fig. 7 the behaviour for expansion ratio 40/1 is acceptable over all the required range.

One of the applications of the one-dimensional analysis has been the effect of the mixing section convergence angle. Fig. 8 shows the pressure variation along the mixing section for different angles. One can observe:

- the pressure profile is quite flat at the beginning, showing that the assumption made in the OCVA is reasonable;
- the smaller the angle is, the faster is the pressure recovery (as can be understood considering the steam phase compression in a converging channel);
- the calculation stops casually at different pressure values, so a final answer for the effect of angle on SI performances cannot be drawn from this calculation.

In order to obtain a significant comparison between different angles, the one-dimensional calculation results have been truncated at the same contraction ratio and the OCVA have been applied from that point on. The results are shown in fig. 9. The performances calculated without considering wall friction seem to indicate an improvement at decreasing angles: that can be ascribed to the greater length of the converging section, that allows a larger condensation and therefore a better kinetic energy recovery during steam recompression. In the same figure, results obtained taking wall friction into account are also shown: in this more realistic case the trend is the opposite, because the friction effect becomes more and more important as the convergence angle decreases and the duct lenght increases. The optimum angle can be estimated at about 8°.

6. PRELIMINARY SIZING OF A SI

According to the results obtained with the theoretical analyses, a preliminary sizing indicated in Chapter 2 has been possible. The main design parameters are summarized in tab.2. To refine this preliminary sizing and to get more confidence in the SI performances, the following points have to be clarified:

- effect of boundary layer and non-equilibrium phenomena on steam nozzle behaviour;
- performance trend at different steam pressures;
- mixing section length actually required to obtain the complete steam condensation.

For this purpose, an experimental activity is foreseen on a scaled-down model (flowrate scale about 1:9) operating at prototypical temperature and pressure. At present, the SI model design is under way.

TABLE 2 Main design parameters of a SI for SBWR's

Steam nozzle expansion ratio	40/1	
Steam flow at maximum pressure (8.7 MPa)	47	kg/s
Steam nozzle throat diameter	69	mm
Steam nozzle outlet diameter	174	mm
Steam nozzle length	0.8	mm
Mixing section contraction ratio	1/20	
Mixing section convergence half-angle	8°	
Mixing section constant section length	2.3	m
Diffuser half - angle	6°	
Total length	4.5	m

NOMENCLATURE

A cross section area
F axial component of wall pressure force
F_w wall friction force
h enthalpy
P pressure
P_o total pressure
m_c condensing mass flowrate
Q mass flowrate
V velocity

GREEKS
ρ density

SUBSCRIPTS
a mixing section inlet
e mixing section outlet
l liquid
ls saturated liquid
min minimum
V vapour

REFERENCES

1) Safety Related Terms of Advanced Nuclear Plants. International Atomic Energy Agency, Final Report, February 1991.
2) G.A. Brown, E.K. Levy. Liquid-Vapour interactions in a constant area condensing ejector. Journal of Basic Engineering, March 1972.
3) L.S. Tong. Private communication, April 16, 1984

SBWR — ISOLATION CONDENSER AND PASSIVE CONTAINMENT COOLING: AN APPROACH TO PASSIVE SAFETY

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Abstract

Two key systems of the SBWR, the advanced passive BWR developed by General Electric with the cooperation of an international team, are being designed according to the simplification and reliability requirements that should characterize a new generation of water cooled reactors.

The Isolation Condenser System and Passive Containment Cooling System provide, for 72 hours without operator actions, core and containment heat removal, relying on natural circulation to transfer residual decay heat outside containment through heat exchangers of original design immersed in water pools.

Both systems are to be defined "passive" (respectively, category D and B of IAEA proposed classification) since they rely completely on passive components to perform their function.

The concept of the steam condensers, the main components of these systems, is a modular one, which is expected to have a high reliability, due to its unique design, supports configuration and minimum number of welds; additionally in-service visual and volumetric examination can be easily performed.

1. INTRODUCTION

General Electric is leading an international cooperation effort on the design of the Simplified Boiling Water Reactor (SBWR), an advanced passive power plant of small size (600 Mwe). In the frame of the Italian program for developing

new generation reactors, sponsored by ENEL and ENEA, ANSALDO is partecipating to this effort and has primary responsibility for the design of two systems for passive core and containment heat removal.
These systems have been conceived and are being designed according to the simplification and reliability requirements of safety related systems which are expected to characterize a new generation of improved nuclear power plants. Simplification is a proper means to improve the operability and maintainability as well as the reliability of a system, a key point in achieving an enhanced level of safety.
As a result of this approach both systems rely only on natural forces and passive components to perform their functions; pumps have been eliminated and no operator action or support system is needed during their operation for a time period of 72 hours.
The two systems and their main components, the steam condensers, are described hereafter.

2. ISOLATION CONDENSER SYSTEM (ICS)

2.1 System Functions

The function of the Isolation Condenser System is to remove decay heat when the reactor becomes isolated during power operations as a consequence of a transient; the system shall control reactor coolant pressure and temperature within a range so that Safety/Relief Valves will not lift and automatic reactor depressurization will not occur.
These functions, when the Isolation Condenser System has come into operation, are to be performed in a completely passive way with no need of both operator actions or control and external AC power sources or forces.
The Isolation Condenser System is not defined as an "Engineered Safety Feature" since other ESF's provide reactor protection and incident mitigation should the ICS be unavailable; the system, however, is designed as a safety related system to prevent unnecessary reactor depressurization and operation of Engineered Safety Features.

2.2 System Description and Operation

The Isolation Condenser System consists of three totally independent loops, connected directly to the steam region and the downcomer region of the reactor pressure vessel; they provide core heat removal relying on natural circulation to transfer residual heat outside containment through heat exchangers of original design (IC condensers) immersed in water pools vented to the atmosphere.
Each loop is designed for 30 MW heat removal capacity; three loops are provided to obtain the required system capacity (60 MW) and redundancy.

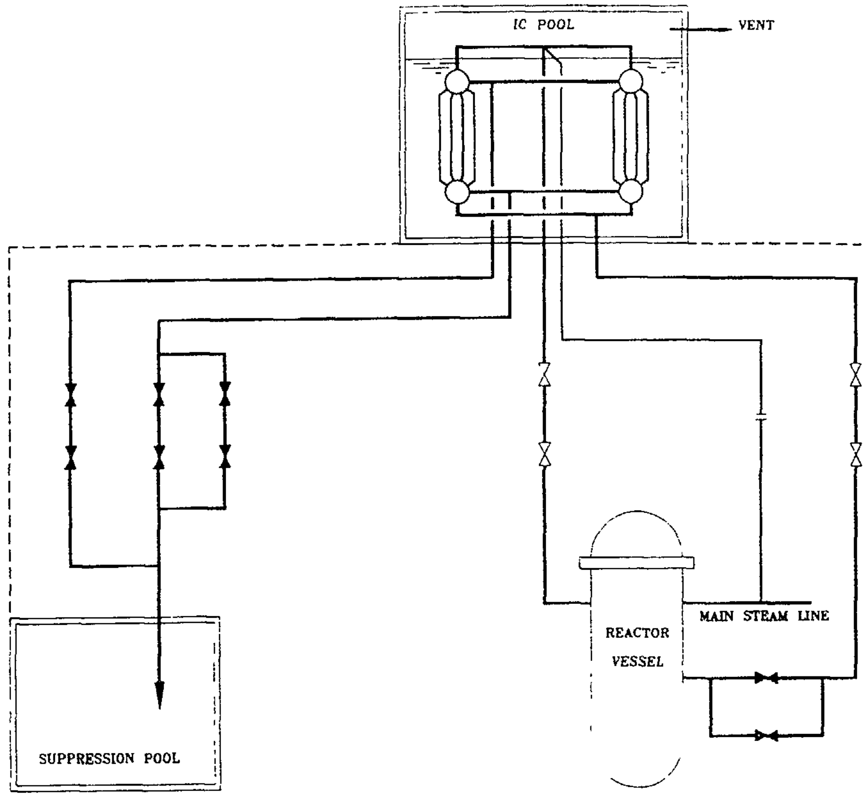

Figure 1 . ISOLATION CONDENSER SYSTEM

Figure 1 is a simplified diagram of a typical loop.
A single steam supply line connects the reactor vessel to each IC unit placed in a large pool of water outside containment at a higher elevation.
On the run of the steam supply line, inside the containment boundary, are located two normally-open isolation valves in series; the line vertically penetrates the containment roof slab and, up to the connection to the IC unit, is enclosed in a guard pipe in order to avoid any large steam-LOCA outside containment.
On the condensate return piping, also located inside containment, are provided two normally-open isolation valves in series and, just upstream the reactor vessel entry point,

a loop seal and a pair of condensate-return valves in parallel . These two valves are closed during normal power operation and since the steam supply line valves are normally open, condensate will fill the IC unit up to the steam distributor (Figure 2) above the upper headers.
On a high reactor pressure signal or a main steam line isolation signal one condensate-return valve in each loop opens starting the ICS into operation; the condensate inside the IC units is drained into the reactor vessel downcomer and the steam-water interface in the IC tube bundle moves downward, below the lower headers to a point in the main condensate return line.
When non-condensable gases build up in the condenser, vent valves open and the gases are routed to the suppression pool.
According to the classification of passive systems proposed by IAEA Consultant's Group, the Isolation Condenser System, in its current configuration, should be included in Category D since an external signal triggers the passive process of decay heat removal opening the condensate-return valve.
However the possibility is being investigated of activating the system opening the condensate-return valve by means of mechanical actuators triggered directly by an increase of reactor coolant pressure, thus increasing the level of passivity of the system and avoiding possible failures on demand due to an external "intelligence" failure.

2.3 Isolation Condenser Unit Description

The IC is operated simply by opening a normally closed condensate-return valve This means that the component is normally at RPV pressure and that it will experience a significant thermal transient every time it is called upon to function.
These very high and rapid thermal transients constitute one of the component main characteristics; in fact, if the equipment is too rigid, the material plasticization is possible but local plastic behaviour must be avoided due to the possible presence of the physical conditions generating the Intergranular Stress Corrosion Cracking (IGSCC).
Another conflicting requirement consists of the need to resist high pressure while at the same time assuring a good thermal performance through the tubes. Therefore, the thermal capacity of the various parts assumes a great importance.
Other important issues such as flow differences inside the different tubes, possible vibrations, ISI accessibility requirements and long term immersion have been considered and resolved in compliance with the arrangement requirements.

The conceptual configuration proposed by ANSALDO is designed so that each part of the unit is free to move, in order to meet its thermal expansion needs without experiencing significant restrictions imposed by other interconnecting parts
Figures 2 and 3 show the configuration of the component
Each module is obtained by connecting an upper header with a lower one by means of a vertical tube bundle· a very well-known and common configuration which eliminates the need of a thick plate and reduces the material volume
Each header is closed at both ends with double-sealed flanged covers allowing the complete opening for the tubes/header welds inspection
The connection with the vertical steam supply line is obtained through a X-shaped forging provided with four outlet lines (two for each module), each featuring a built-in Venturi orifice
A guard pipe is provided around the main supply steam line in the pool outside the containment
Each lower header is drained by a flexible line, connected through a tee to the main condensate line, which penetrates the primary containment and discharges into the RPV above the core.
The whole weight of each module is sustained by the upper support which provides also the required seismic restraint
The lower restraint allows the heat exchanger tubes vertical downward expansion and the header axial expansion, but prevents any seismic movement in either horizontal direction.
Each heat exchanger tube may also expand with some delay or in a different way compared with the bundle global behaviour. In fact, each tube is provided with two small additional bends which will take care of the additional stresses
The main advantages of this solution are the following:

- A modular configuration,
- A very flexible design in order to accomodate possible surface changes or changes in the available pool depth and width according to the arrangement configuration needs,
- Low additional stresses due to the various parts thermal expansion,
- Low pressure drops for natural circulation,
- A good in-place accessibility and the possibility of complete removal,
- Outlets provided with orifices in case of pipe ruptures;
- A modular full scale testability.

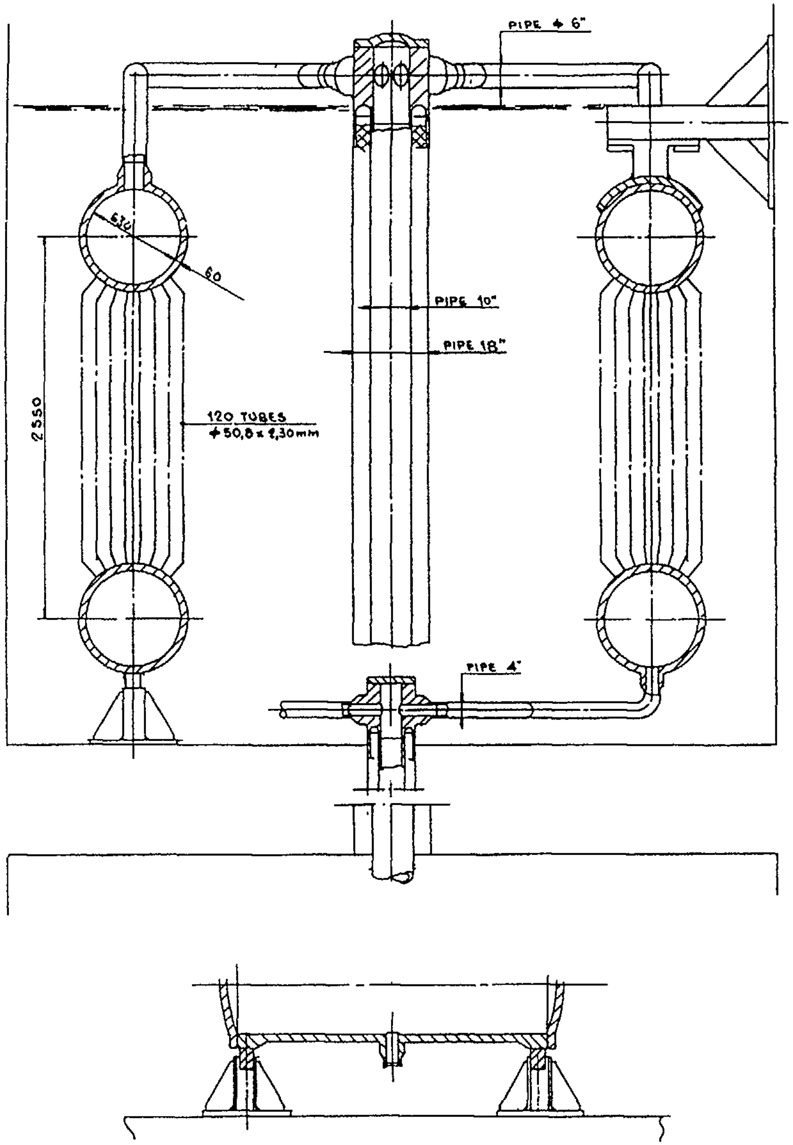

Figure 2. Isolation Condenser. Vertical Section

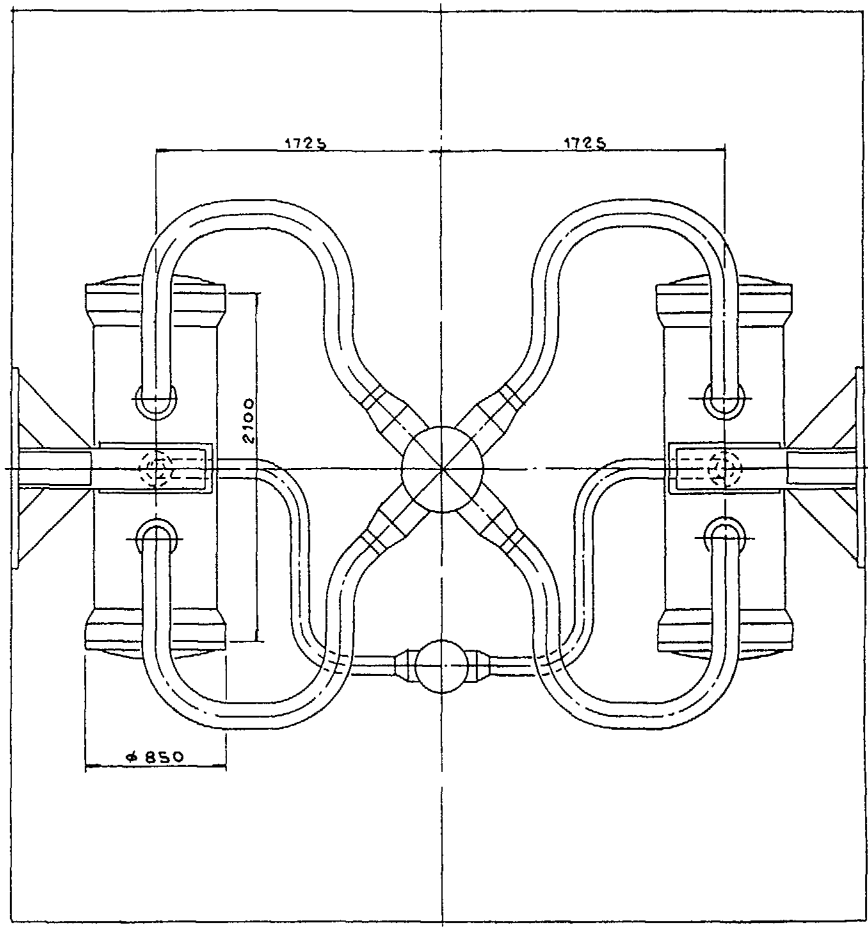

Figure 3. Isolation Condenser. Plan view

3. PASSIVE CONTAINMENT COOLING SYSTEM (PCCS)

3.1 System Functions

The Passive Containment Cooling System function is to remove decay heat from the Containment after a Loss of Coolant Accident, maintaining the containment pressure within the design limits.

The containment heat removal function shall be provided for a minimum of 72 hours, in a passive way, without any external intervention by support systems or operators.
The system is an Engineered Safety Feature and as such is designed as a safety related system.

3.2 System Description and Operation

The Passive Containment Cooling System consists of two totally independent loops open to the primary containment, connecting the drywell with a steam condenser (the Passive Containment Cooling Condenser) located outside containment in a large pool of water vented to the atmosphere.
Figure 4 is a simplified diagram of a typical loop.

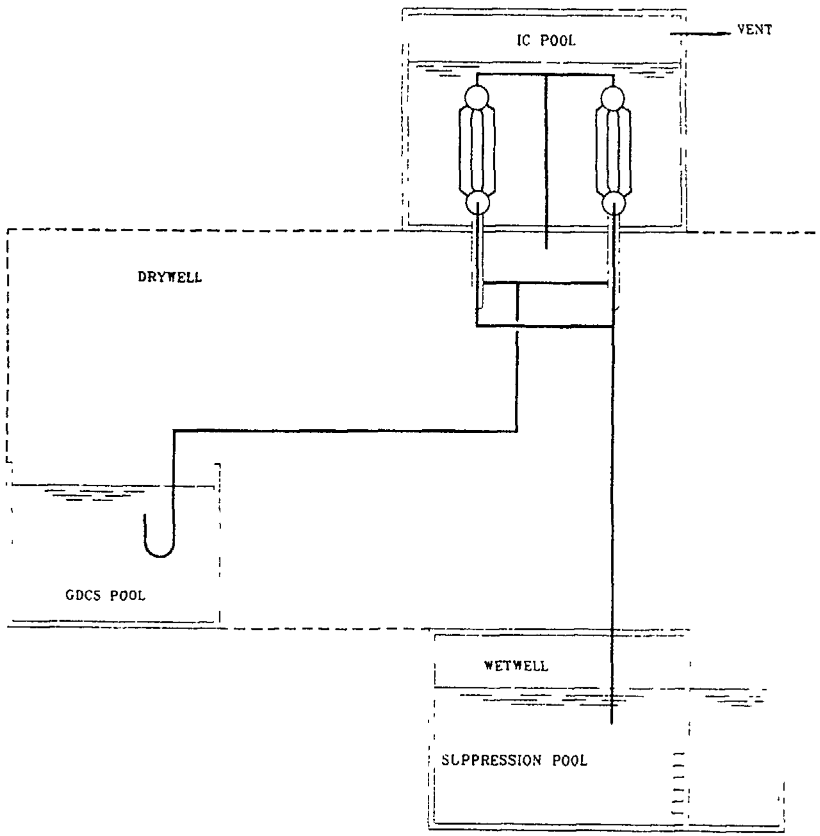

Figure 4 . PASSIVE CONTAINMENT COOLING SYSTEM

The condenser receives a steam-gas mixture directly from the drywell through a central supply line which penetrates the containment roof slab vertically and has no isolation valves.
Steam is condensed inside the vertical tubes of the PCC condenser and drained to the Gravity Driven Cooling System pool located inside the containment drywell.
On the drain line and submerged in the GDCS pool, just upstream of the discharge point, is a loop seal : it prevents back flow of steam and gas mixture.
Non-condensible gases are routed to the suppression pool in the wetwell through a vent line, which, like the drain line, has no valves.
The Passive Containment Cooling System, therefore, is always in a "ready standby" condition. Its safety function is initiated by the pressure increase in the drywell due to the loss of coolant accident itself : the difference in pressure between drywell and wetwell initially provides the driving head for the steam-gas mixture flow through the condenser. Then condensation promotes the steam flow causing a local pressure reduction and drainage of condensate to the GDCS pool relies on gravity.
The PCCS is therefore a completely passive safety related system as far as the initiation of the process is concerned and the functions execution as well : it should be classified in category B of passive systems according to the IAEA proposed classification.

3.3 Passive Containment Cooling Unit Description

Considering the opportunity of maintaining the same configuration already adopted for the IC, which meets the requirements of modularity and the arrangement needs, the PCC unit has been developed based on the IC concept, taking into account its less severe design conditions.
In fact:

- Its operating pressure and temperature are much lower;
- Only one significant operating cycle is considered for the entire working-life;
- Standby stress levels are insignificant so IGSCC is not a problem;
- The equipment is normally empty and the tubes are in contact with the containment atmosphere.

On the other hand, the amount of expected non-condensable gases mixed with the condensing steam is much larger and requires greater diameter vent pipes.
Figures 5 and 6 show the resulting design, which is a free-standing condenser placed on the bottom of the pool and expanding upwards without any need of sophisticated seismic

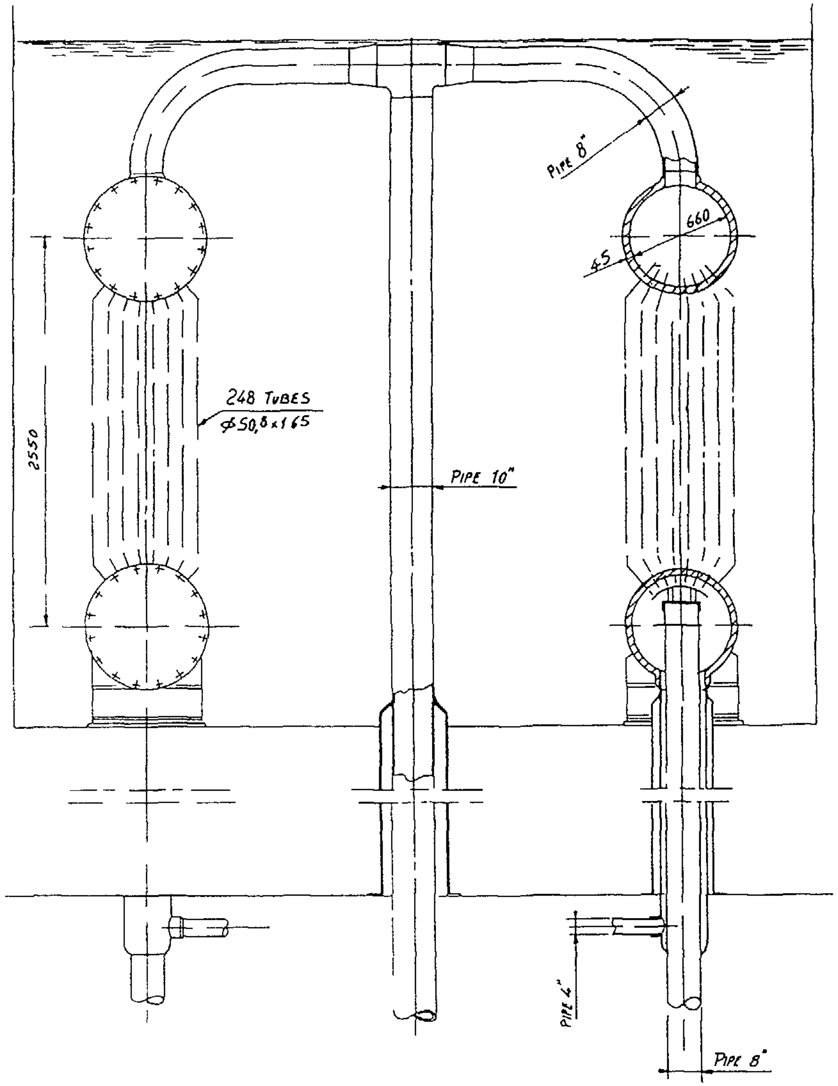

Figure 5. PCC Condenser. Vertical Section

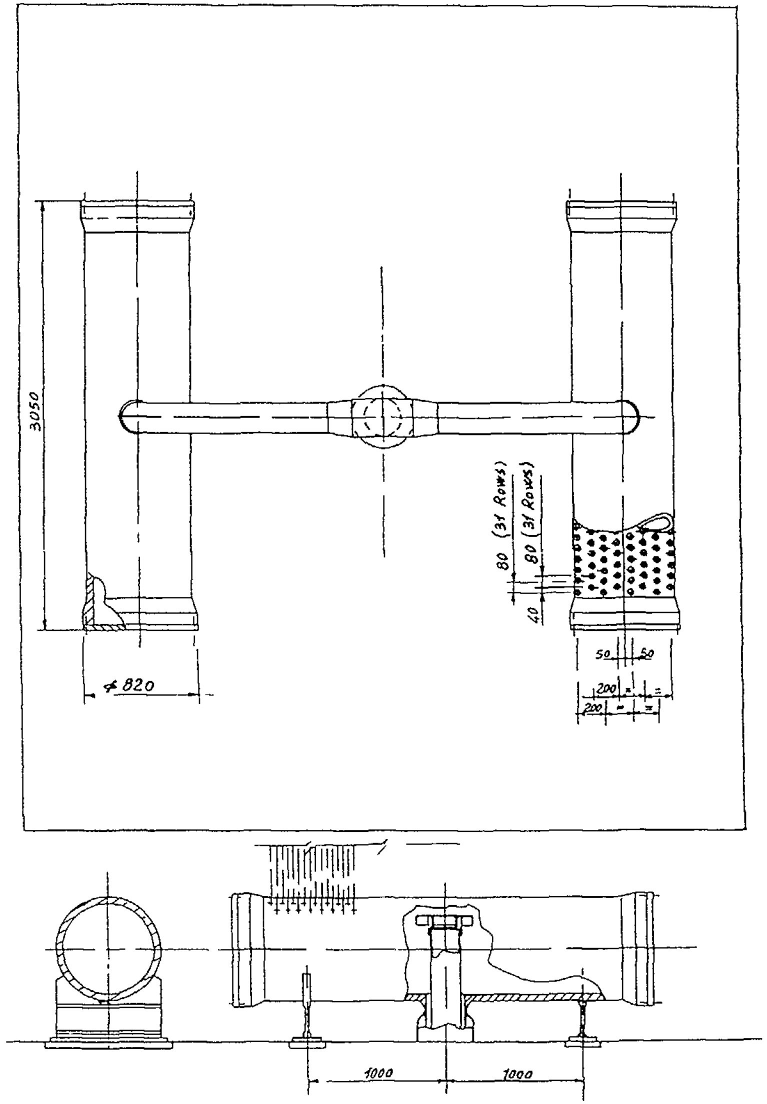

Figure 6. PCC Condenser. Plan view

restraints; the penetrations are simplified, the header covers may be flat and all the material to be used can be stainless steel.
The condensate return line and the non-condensable gases vent line have been arranged in a single concentric tube, thus reducing the number of penetrations and maintaining the possibility of disassembling the unit from the top.
Considering the very delicate function performed by both the IC and PCC components with respect to the plant safety and taking into account their location outside the containment, the number of welds for large pipes has been reduced to a very low minimum in order to improve the units reliability.
Actions such as inspection and possible plugging of the heat exchanger tubes are expected to be easily accomplished and, if desired, they could also be automated by means of a proper robotic device.

4. TESTING

The ICS and PCCS are two key features of the SBWR and their functioning is strongly dependent on the proper design of the two condensers.
As a consequence, since both the condensing units are first-of-a-kind devices, their expected performance has to be confirmed through a full scale test campaign, which represents the third step of the integrated testing program set up by General Electric, based on the MIT and UCB test campaign and on the small scale PCC Toshiba tests.
The IC and PCC units testing will be conducted at SIET (Società Informazioni Esperienze Termoidrauliche) facilities located in Piacenza, Italy, under the lead responsibility of ENEA.

4.1 Isolation Condenser Testing

A prototypical module of the IC will be tested to demonstrate the adequacy of the design from both a thermal-hydraulic and structural standpoint.
Specifically, the IC heat removal capability over the expected range of SBWR conditions will be measured as well as the tube side heat transfer and flow rates to confirm that the condenser operation is stable and not affected by large fluctuations.
Additionally, it will be demonstrated that a specified fraction of the IC thermal cycles expected during its lifetime (60 years) will not result in excessive deformation, crack initiation or excessive crack growth rate in the module.

4.2 PCC condenser testing

Similarly to the IC testing, the thermal-hydraulic and structural adequacy of the PCCS will be verified through the prototypical testing of the PCC unit.

The heat removal capability over the expected range of SBWR conditions will be measured and also the scheme for venting the non-condensible gases from the PCC will be tested.
The stress levels at critical locations on the PCC will be measured to confirm that they do not exceed design values.
Finally, the behavior of the unit under cyclic loads typical of the PCC system operation will be investigated.

Currently, the design of the test facility is completed and the equipment for both loops (ICS and PCCS) is being procured

DESIGN OF BACK-UP PROTECTION SYSTEMS FOR NEW GENERATION NUCLEAR REACTORS

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Abstract

The addition of a Back-up Protection System (BPS) for the prevention and mitigation functions, meets the aim to reduce at the minimum possible level the contribution of the common mode failure (CMF) to the frequency of high release.

Protection Systems (PSs), are based on microprocessor technology and, consequently, on the related software. In these cases experience has proven that hardware and, chiefly, software may be prone to CMF. But the PS is required to be highly reliable and for this reason a Back-up Protection System, to be used in the new generation nuclear reactors, has been studied and is under design.

Owing to the necessity of not having CMF with the PS, the BPS design has been based on the following general design criteria:

- The BPD design is required to be based on operating principles which are diverse with respect to the ones adopted in the PS.

For this reason BPS design doesn't use microprocessor technology;

- components used in the BPS are required to be simple, reliable and passive.

For this reason the BPS design foresees extended use of LADDIC MODULES.

Finally the use of 2-out-of-4 (2004) logic characterizes BPS design.

The paper presents the design of a BPS aimed to guarantee the Containment Isolation (Mitigating Function). BPSs design for the preventing function will follow.

In the end of the design activities a representative divisional panel will be constructed and, successively, subjected to a qualification program in accordance with the IEEE procedures for class 1E equipments.

1. INTRODUCTION

The Protection System (PS) as well as the Control System (CS) for Passive Plants is based on digital technology and is characterized by enhanced features in terms of reliability compared to protection systems based on conventional technology.

The Passive Plant PS design adopts the same component both safety and non safety systems.

The common cause failure of these components, although unlikely to occur, represents a significant contribution to the Core Damage Frequency (CDF) and to the high radioactivity release frequency, as PSA evaluations indicate. This is due to the fact that the PS failure affects both preventing and mitigating functions.

In particular, three major protection functions need very high reliability:

- Reactor Shut-down and Reactor Vessel Depressurization to reduce the CDF and to make the probability of sequences leading to core melt at high reactor vessel pressure negligible;
- Containment Isolation to guarantee the containment tightness during any accident sequence independently of the possible failure of the preventing systems called for operation during the accident evolution.

The addition of a Back-up Protection System (BPS) for the aforesaid prevention and mitigation functions, fulfils the requirement for reducing to the minimum possible level the contribution of the PS failure to the frequency of high release.

One of the most important concepts to be included in the passive plant design is the requirement for the diversity of alternate systems which are relevant for safety. For this reason, since the PS is based on the microprocessor technology, a Back-up Protection System (BPS) should be provided, based on a diverse technology, using simple, reliable, low consumption and possibly passive (without moving parts) components.

The solution presented in this paper is based on the use of LADDIC modules, already employed in other nuclear plants and in particular foreseen for the BPS of the Italian standardized PWR plant (PUN).

2. BPS SAFETY CHARACTERISTICS

To design a system having the features reported in the previous section, it is necessary that the BPS is based on the following safety criteria:

1. In conjunction with the primary Protection System, the Safety Systems and the Containment Protection and Isolation Systems, the BPS is to be designed to minimize the probability of accident sequences leading to core melt with the containment formerly by-passed or at high reactor vessel pressure, and to guarantee the containment tightness during any acccident

sequence, independently of the possible failure of preventing systems called in operation during the accident evolution.

2. The BPS is to be designed to perform its intended safety functions when exposed to natural phenomena including earthquake, tornado, wind, flooding and man-made hazards such as aicraft crashes. The BPS should be designed to operate in the presence of severe environmental conditions caused by such events.

3. The BPS is to be designed and located to minimize the probability and effects of postulated hazards such as fire and explosions. The BPS is to be capable to perform its safety function in the presence of such hazards.

4. The BPS is to be designed and located to minimize the effects of inadvertent actuation of plant systems such as fire suppression.

5. The BPS is to be protected against the effects of missiles and conditions that may be caused by pipe breaks or other equipment failures, including non-Class 1e equipments which may be located in the same area as BPS.

6. The BPS is to be designed to perform its intended safety functions when exposed to the environmental conditions associated with normal operation, maintenance, testing and postulated accidents.

7. The BPS is to be designed for high functional reliability and inservice testability, in agreement with the overall reliability targets.

8. The BPS is to be designed to permit periodical testing of its functions as far as feasible with no restriction on the reactor operation.

9. The BPS is to be designed to assure that (1) no single random failure results in loss of protective function and (2) removal from service of any component or channel results neither in loss of required minimum protection nor in an increased probability of spurious actuation.

10. The BPS is to be designed to fail into a safe state if condition such as system disconnection, loss of energy supply, or adverse environmental condition are experienced.

11. The BPS is to be separated and diverse from the primary Protection System so that any failure of the latter does not impair the correct operation of the BPS. Possible exclusion from these criteria could be represented by sensors and actuators when complete diversification and/or separation is not practical or not required.

12. BPS set-points shall not reduce the operating margins of the primary Protection System both during normal or transient operations.

13. Once initiated, any protective function of the BPS shall continue to completion.

14. The BPS design shall minimize the spurious actuation occurrence.

15. When actuating equipments are shared between PS and BPS, the latter shall provide actuation signals to these equipments overriding possible faulted stop or non-actuation signals generated by the PS. In case of a spurious actuation of BPS, the resetting should be made only according to the general criteria for passive plants on this matter.

3. BPS DESIGN CRITERIA

The following criteria of redundancy, independence and diversity are followed in designing the BPS.

Redundancy: The BPS is to be designed to perform its intended safety functions from any single random failure originated within the system. This requirement is met by incorporating redundancy into the BPS design, providing a fault-tolerant design as follows:

- Four redundant guard-lines must be considered for gathering plant data, for transmitting them to the system and to generate the system level actuating signal according to a 2 out of 4 logic.
- Single failure within the BPS must not cause a spurious reactor trip or a spurious actuation of a safety system.
- In case of random failure of a single channel the BPS must automatically reconfigure to a 2 out of 3 logic in order not to affect plant availability in case of a further failure or a spurious signal.

Independence The BPS design must have an adequate degree of independence among the equipments costituting the system itself. To this purpose, the electrical isolation among redundant components and sub-systems is to be guaranteed. Redundant components and/or subsystems are to be arranged in separate cabinets, although physical segregation between the cabinet is not required. The BPS is to be completely independent of the primary Protection System to the following extent:

- No signal processed and/or calculated by the PS is necessary for the BPS to perform its intended functions.
- No failure of the BPS impairs the capability of the PS to perform its functions, and vice-versa.
- As for the components, if any, actuated by both the primary and back-up systems, the actuation devices guarantee the independence between the systems.
- The PS and BPS derive the respective power supplies from separate sources.

Diversity: In order to prevent to the maximum extent the probability of common mode of failure between PS and BPS, as reported in section 1, the following diversity concepts are applied when designing the BPS:

- The BPS must use a technology diverse from the one used for the PS design.
- Possible exceptions to this requirement are the sensors that are required to be neither diverse nor separate from the sensors of the PS.
- The BPS must shut the reactor down by means of mechanisms which are diverse from those used by the PS.

4. LADDIC MODULES

The LADDIC module (See Fig. 1) is a passive component, constituted by a ferrite magnetic core (ladder shaped) with several windings. Four of these, named control (or hold) windings, are continually fed by output signals coming from the TUs (hold currents). Other two windings are the reset and output devices.
The last one is the set winding.

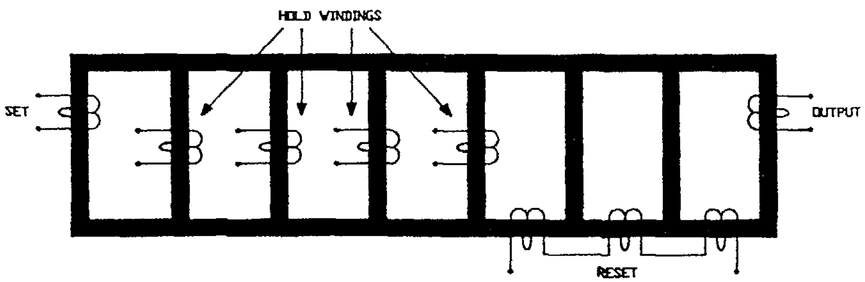

FIG.1 SCHEMATIC CORE AND COIL ARRANGEMENT OF A TYPICAL LADDIC MODULE

Each module is associated to a divisional signal input parameter, and it is in series with other modules. The output winding of a module is connected with the reset winding of the successive module.

The reset signal of the first module is supplied by a clock circuit, called Pulse Generator, which generates a stream of pulses which pass through the magnetic core.

The output from the last module is sent to a power inverter that changes its status (open-closed) if an interruption of the pulse stream is detected.

For the BPSs presently under design, an actuation signal is generated for each trip parameter when at least two out of four divisions of that parameter exceed the related set-point (2oo4 logic).

The control windings on each LADDIC control the magnetic flux present in the ferrite core. An output current pulse is produced by the LADDIC only if a set pulse is present and a reset pulse from the previous LADDIC or from the pulse generator is present and three out of four of the hold signals are present.

In this manner, the control windings act as switches determining whether or not a pulse from the reset is transmitted to the output.

An "OR" configuration among different trip parameters is achieved by putting several LADDICs in series (Fig. 2, 4).

5. BPS UNDER DESIGN

The BPS at present under design concerns the function of: ATWS, Reactor Depressurization and Containment Isolation. They are substantially costituted by three parts. The first is located in the process islands and is constituted by the sensors and associated signal transmission lines. The signals from the sensors are transmitted to the BPS divisional cabinets. Four signal divisions are assumed, as for the PS.

The second part of the BPS is constituted by the four cabinets which contain the guard lines TUs/LADDIC/Inverter sets. Each divisional parameter is sent to the input of the four trip unit sets as shown in Fig. 2. For each of the N parameters used for the safety function, four trip signals are generated inside each trip unit. Four buses (one for each division), constituted by the signals for all the process parameters, exit from each Trip Unit set. These buses are the inputs for the divisional LADDIC sets. If the parameter trip setpoint is reached or exceeded in at least two out of four related TUs, all the four associated LADDICs are tripped and no pulse stream is trasmitted to the four divisional inverters connected to each LADDIC set. This event causes a 4 out of 4 trip signal that is sent to the actuation logic and device. (Fig. 2).

The Actuation Logic modules perform the 2 out of 4 logic function on the signals coming from the four division inverters. These modules are necessary to avoid that the trip of a single division or a single failure of one component of a chain (normally the inverter, since the failure of a LADDIC is very

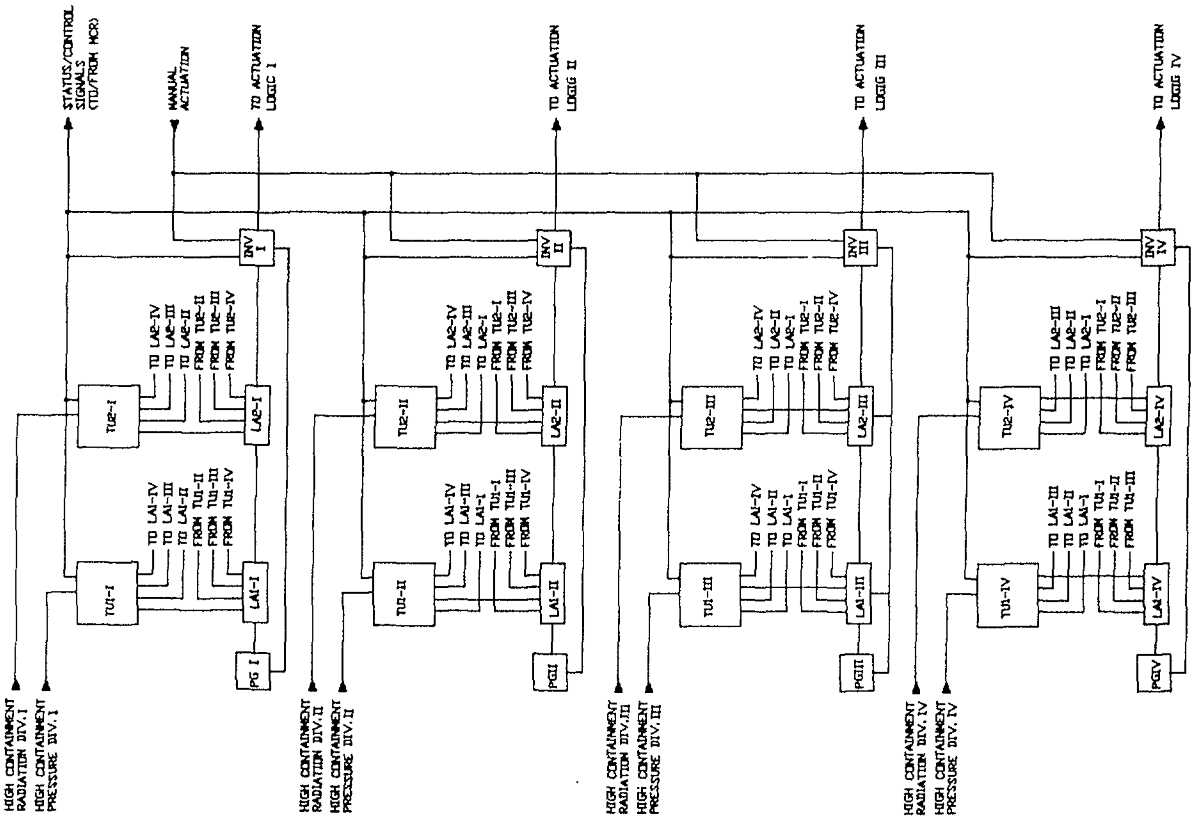

FIG.3 BPS-CONTAINMENT ISOLATION (for PWR)

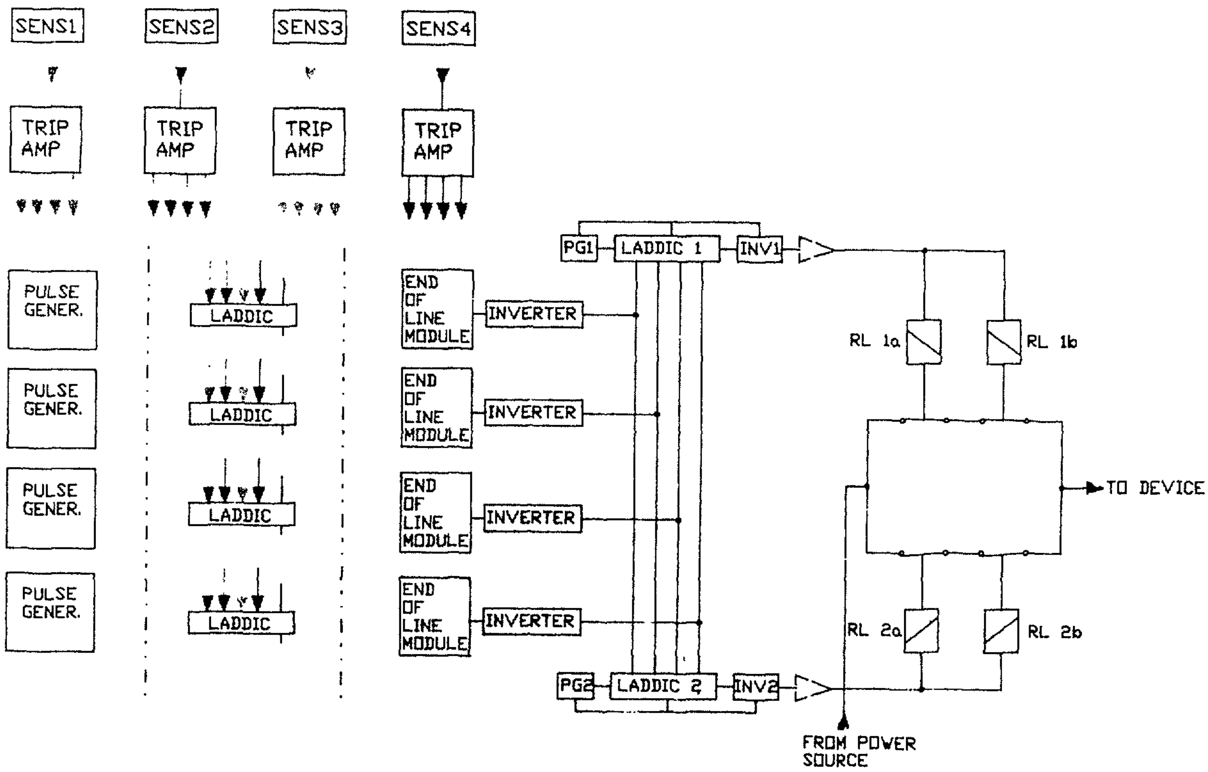

FIG.2 TYPICAL ONE-PROCESS PARAMETER LINE

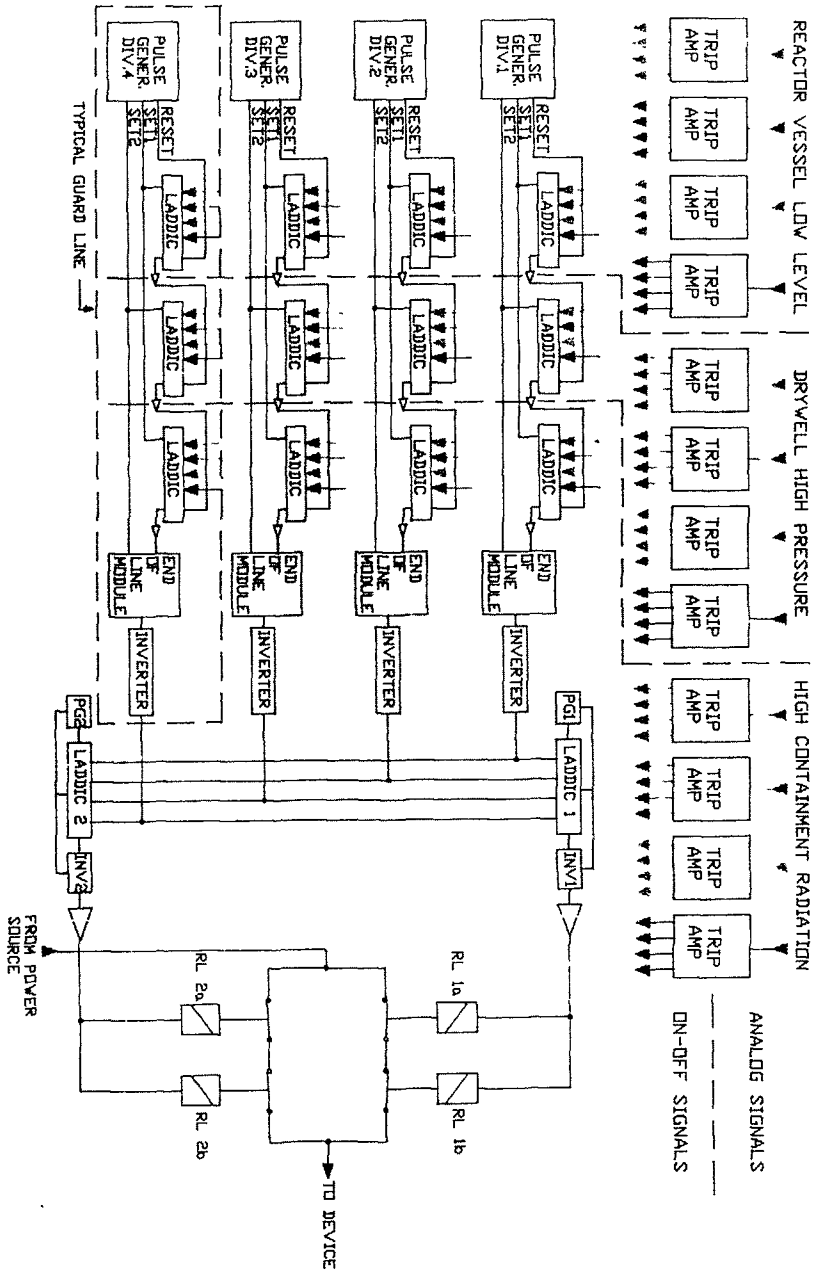

FIG.4 BPS-CONTAINMENT ISOLATION (BWR)

unlikely due to its passive features) causes the spurious actuation of the related function.

The Actuation Logic can be implemented using again LADDIC modules as the principal component.

The logic is obtained using two LADDIC modules which perform a 2 out of 2 function. If a trip is required all the four divisional inverters generate signals that actuate both the modules. The inverters of these modules cause the closing or opening of the four relays on the actuator power/control line.

Conceptual design schemes both for the BPSs relevant to the preventing functions and for the BPSs relevant to the mitigating functions have been developed for reactors type BWR and PWR. Similar schemes for other reactors will be developed, if necessary, in the future.

Process signals used to actuate the BPSs relevant to the above mentioned functions are reported in table 1.

Figure 3 presents one of the mentioned conceptual schemes.

TABLE 1. PROCESS SIGNALS USED FOR BPS ACTUATION

Function	Process Signal	
	Boiling Water Reactor	Pressurized Water Reactor
ATWS	- Reactor water level - High reactor pressure	- Low feed water flow - High thermal power
Reactor vessel depressuri-zation	- Reactor vessel low level L1	- SI signal in AND logic with - CMT low level
Containment isolation	- Reactor vessel low level - Drywell high pres-sure - High containment radiation	- High containment pressure - High containment radiation

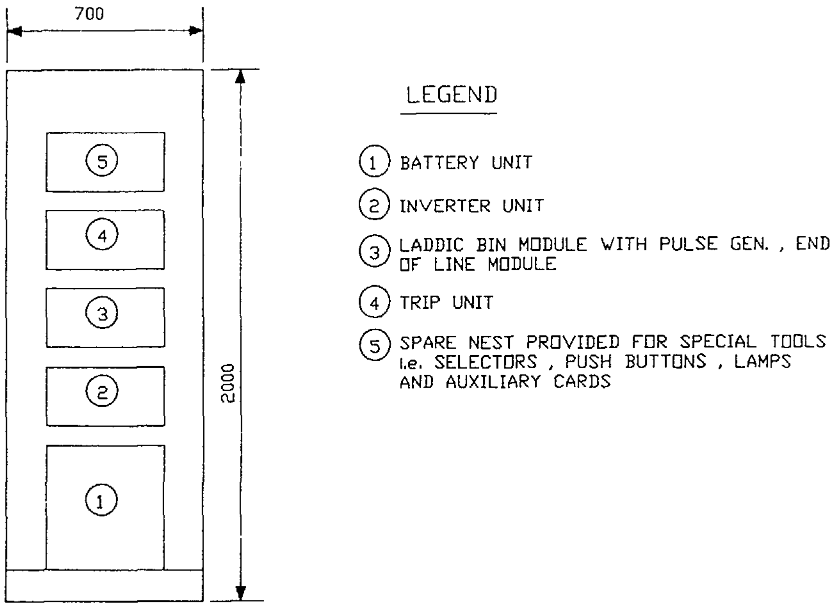

FIG.5 GENERAL LAY-OUT OF A DIVISIONAL PANEL

6. BPS FOR CONTAINMENT ISOLATION FUNCTION

The BPS for the containment isolation function is a completely additional system (LADDIC logic and actuation equipments). The process signals, used for BPS actuation, are reported in table 1.

Each signal is sufficient for actuating the containment isolation. The "OR" logic is obviously performed by the LADDIC modules themselves.

The design of this BPS is in an advanced phase over the other BPSs. Fig. 4 is a functional scheme up to the Actuation Logic of this BPS and fig. 5 shows a lay-out of various equipments in a typical divisional panel.

A suitable battery to cope with an accident with loss of any a.c. power supply has been added to the panel. Its capacity is foreseen, at present, for a period of 72 post-accident hours, according to the general design criterium of passive plants in this field.

7. FUTURE ACTIVITIES

After performing the conceptual design of the BPSs ENEL asked CISE for the detailed design of the BPSs. After this, a divisional panel of the BPS related to the "Containment Isolation function" will be constructed. An extensive qualification program according to the procedures and criteria coded in IEEE 323 and IEEE 344 will be successively performed on the panel. This qualification program will deal with both environmental and vibrational aspects using analytical methods and real tests.

8. CONCLUSIONS

The addition of a BPS to the main protection system is considered to be a major step forwards as far as the safety of new generation plants is concerned. With such an addition it is in fact possible to neglect the common mode failure at the main protection system microprocessor level (either hardware or software) which might cause at the same time both a core damage and the failure of the containment isolation function, thus giving rise to high release into the environment.

The use of LADDIC technology using simple, fail safe, low consumption and passive components in line with the peculiar features of the passive plants and completely diversified from the main protection system, has been seen by ENEL as the best way to solve the problem.

On the other hand the possibility of an automatic reconfiguration of the system from a 2oo4 logic to a 2oo3 one in case of a spurious trip of one line, guarantees that the plant reliability is maintained at the same high level assured by the main protection system itself.

Eventually no human action is necessary to allow the proper operation to be carried out by the back-up system, neither as regards its safety function, nor to maintain the required level of plant availability.

The activity under way consists in the detailed design of the complete system, the construction of a typical panel related to the containment isolation function and the execution of a thorough qualification program.

ADVANCED REACTOR DESIGN PHILOSOPHY AND APPLICATION — WAYS AND MEANS TO PREVENT CORE MELT

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Abstract

Looking at the energy supply situation in the world it is obvious that the global energy demand will rise drastically, in particular in the less developed countries. It is also clear that energy use will shift towards electricity, ie., the demand for electricity generation capacity will increase significantly.

There is also an increasing awareness that, for environmental reasons, a large scale increase in utilization and burning of fossil fuels must be avoided. Hence, nuclear power should be a strong option for meeting the increase in energy demand.

Unfortunately, public confidence in nuclear safety has eluded the present generation of reactor designs to such an extent that the phasing out of operating plants rather than expansion is considered in some countries. Steadily increasing safety requirements have resulted in an ever-increasing complexity of the current designs which makes them more or less "incomprehensible" to laymen.

A prerequisite for a real revival of the nuclear option may be a drastic design change yielding a quantum jump in perceived safety and reduced plant complexity. From the public point of view the important question is not whether the risk of core melt is once per a million years or once per ten million years; but whether they will have to evacuate their homes in the event of "accidents" in nearby power plants. Hence, there is a distinguishable trend in design goals for new plants - do your utmost to prevent core melt.

The approach to nuclear safety is based on the "defense-in-depth" concept which can be characterized as a design logic with four major steps - prevention, protection, mitigation and emergency management (or off-site actions). A reduction in the requirements of the latter category implies a shift in emphasis to the other three; in some of the advanced designs today strengthening of the mitigation is utilized, in others improved protection, and in a few the emphasis is put on the prevention.

One of the latter advanced designs is PIUS and this plant design will be utilized to illustrate the philosophy and its application in the plant design with the aim of "eliminating" the risk of core melt.

Introduction

It seems to be a general consensus, among both technicians and politicians, that the global energy demand will rise drastically in the years to come, in particular in the countries that are less developed today, - a necessity for improving their standard of living. The growth in energy demand is furthermore expected to involve a strongly increased demand of electric energy; historically, an increasing standard of living has implied a tendency to use more electricity.

It is also widely agreed that the number of possible options to meet such demands on increased energy supply is quite limited, taking into account the potential environmental impacts and hazards of the different large-scale energy sources. What kind of energy source(s) can then be used?

Today, coal and other fossile fuels are utilized widely for producing heat and electricity, but firing of fossile fuels results in emissions of carbon, sulphur and nitrogen oxides, particulates and other gases with detrimental effects on the environment. Oil and gas are "cleaner", but still emit carbon dioxides. And there is an obvious question: Should oil and gas be used for generating electricity or saved for other purposes?

Nuclear power contributes considerably to the world's energy supply today; last year roughly 20% of the electricity generation came from nuclear. And it will continue to be needed. It's environmentally clean, but it has had its problems; instabilities in safety requirements, public perception, etc. An accident receives much more publicity than years of safe, clean electricity production, and technical ignorance is easily exploited by opponents. If nuclear power is to remain an option for the future, it must be commonly perceived to be really safe and economical and without threat of degrading the environment.

It is considered as one of the major options for electricity generation of the near future in many countries, and looking forward to a revival of the nuclear power plant market the major nuclear vendors are developing a number of advanced reactor designs; water-cooled reactors, gas-cooled reactors, and liquid metal reactors. Basically, these new (or advanced) designs fall into two kinds or categories; "evolutionary" types and more "innovative" types.

The "evolutionary" types, mostly Light Water Reactor (LWR) plant versions, can be characterized as modest modifications (or improvements) of present-day plant designs with a strong emphasis on maintaining "proven designs" and avoiding need for additional experimental verification. In other words, these types represent an evolution of existing plant designs.

The more "innovative" LWR reactor types are also based on well established LWR technology, but incorporate departures from, or supplements to, current LWR designs in various aspects. For some of these types, the major development goal is stated to be simplification and enhanced use of passive safety features based on current LWR plant designs; this is the simplified "passive" advanced light water reactor program pursued in the U.S. by EPRI (the Electric Power Research Institute). Others have more ambitious development goals -

eg , striving for PRIME safety which necessitate more radical modifications of plant structures or system arrangements [**PRIME** is an acronym introduced by Dr Charles Forsberg, ORNL for **P**assive, **R**esilient, **I**nherent, **M**alevolence resistant, and **E**xtended time safety]

Are advanced designs needed?

From a nuclear vendor, and perhaps also from a power utility, point of view construction of new plants exactly as previous ones (repeats") would imply advantages as it represents reduced engineering and construction work, less errors and significant cost savings On the other hand, a plant design cannot live forever, it has to be "up graded to implement new technology developments and new generations of equipment, to incorporate improvements based on experience gained from construction, commissioning and operation of the plants, and to adapt to new regulatory requirements There is also an ambition, by both vendors and utilities, to improve their product, - as far as possible with respect to economy, - making the nuclear plants safer, more reliable and easier to operate Without such an ambition the risk of getting a reduced product quality is imminent

From the point of view of competitivity, striving for excellence is a necessity for the vendors , but it is also important for the utilities from the public perception point of view - good averages are necessary but not sufficient, since the public will always base their judgement on the poorest performers This was recently underlined by Lord Marshall, chairman of WANO (World Association of Nuclear Operators), in a statement at its general meeting in Atlanta in April 1991 it is no good having 99% of nuclear plants safe if the other 1% have low standards '

This implies that there has been, and will always be, a "continuous development of new, advanced designs, the situation today, with a rather large number of advanced designs under development, is remarkable only due to the long period of no nuclear market a revival of a nuclear market can be envisaged within this decade

The safety level of modern nuclear power plants in operation is high and further improvements of the safety performance is not generally a dominating concern in the requirements on new, advanced designs, requests for plant simplifications and reliability improvements are given a higher priority The safety of present day modern plant designs is based on rather complex interactions between a large number of redundant and diverse safety systems, as a result of numerous system add-ons over the years to meet escalating safety concerns From the technical point of view, this represents a challenge to plant designers and operators with respect to ensuring reliable and safe operation, and from the public perception point of view, it makes it very difficult to inform- a layman can not fully understand the interactions Besides, the plant complexity involves much equipment, building volumes, engineering and analysis work, discussions with authorities etc , it costs a lot of time and money when con structing the plant

The complexity of safety systems may also be a detrimental factor for plant oper ation reliability and that will obviously affect the economy of the power plant Safety grade systems and components require extensive periodic testing, inspections and maintenance, and a large number of such systems and components represent a heavy work load on the utility personnel Much of this work has to be carried out in connection with the refuelling outage and therefore tends to extend the plant shutdown, the plant economy is affected

The economy of nuclear power, as well as its safety, has been under much debate, both in the public and within the industry, - and this explains why vendors and utilities strongly emphasize the need for simplification and improved reliability

From the utility point of view, the socalled evolutionary types of advanced reactors would normally be preferable, since they do not differ much from their predecessors, even though some simplifications are introduced For the more innovative types the utility will have to take a risk, since these plants deviate more from the present-day designs

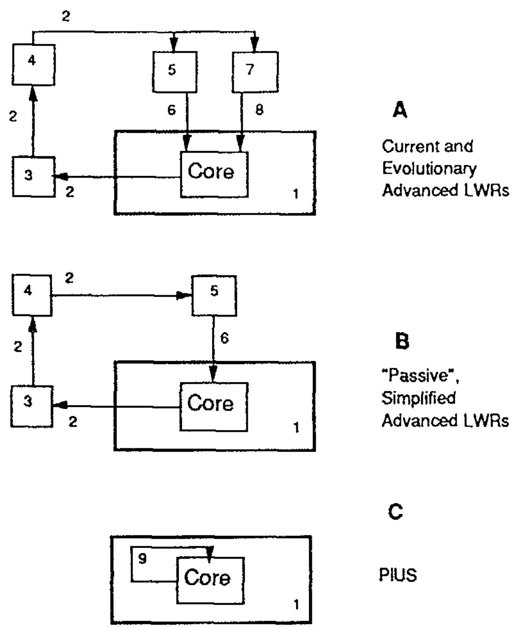
And, for plants to be built in industrialized countries, the benefits of these changes and differences may not be of decisive importance, the infrastructure and the safety culture of the society are adequate for safely constructing and operating also the evolutionary types under development The anticipated economy of the plant, its provenness and experience background, reliability and efficiency, flexibility and maintainability will in many of these countries be the most important evaluation parameters

In less developed countries, the situation is different, there is often a lack of infrastructure - scarcity of qualified and trained workers, shortage of material and equipment, no tradition and experience of operating industries, - and lack of a well-educated middle class that could provide a natural basis for dedicated operators Hence, construction and operation of nuclear power plants that are closely similar to the present-day types in less developed countries might well prove to be in conflict with Lord Marshall's statement on 'the other 1% above It seems that implementation of nuclear power in the less developed countries should be based on significantly simplified and more forgiving nuclear power plant designs

Advanced reactor design features

As noted above, the design goals for the advanced reactors are to a large extent focusing on a rather small number of typical common design considerations, which sometimes are differentiated by supplementary national concerns

The two most fundamental goals are related to plant simplification and to increased design margins - both with the prime objective of making plant operation simpler, from the point of view of the operator Simplification improves operator comprehension of plant functions and implies that he more easily can supervise the actual operating conditions in the plant, i e , he can devote more of his attention to keeping the plant in operation and generating kWhs In-



1. Reactor coolant system
2. Instrument leads, cables
3. Diagnostic equipment
4. Signal processing, logics
5. Flow rerouting/reactor shutdown; valves and control rod drives
6. Piping and control rods
7. Forced core coolant supply; diesel generators and pumps
8. Piping
9. "Self-protective" thermo-hydraulic feedback

Figure 2 - Functions and equipment needed for core protection in LWR plants

In present-day reactor designs as well as in the evolutionary types and in most of the more innovative types reactor shutdown is initiated/triggered by a control and monitoring system. A substantial improvement of this function can only be gained if and when the process itself can "sense and evaluate" the operating conditions and adjust itself to a safe state.

The next functional step relates to the "tool" that is utilized to shut down the reactor or at least to bring it to a safe state. As noted above most reactor designs utilize control rods which are inserted into the core by means of stored energy (in gas accumulators) or dropped by gravity, with borating systems as backup. The control rods represent a rather reliable means of shutting down the reactor, but they also generates a number of problems and safety concerns - reactivity insertions and power distortions by erroneous rod manoeuvering or equipment failures. From this point of view, elimination of control rods, both for power control and reactor shutdown, should be an advantage, since - as some critics say - control rods are needed to handle the problems that arise from having control rods. Arranging the boron supply to the core in such a way that the process itself can control it,, represents a practical problem that is not easily solved in normal reactor design configurations.

Finally, what about the core cooling? Systems that operate in a passive mode, i.e., without external input of energy or control signals, following an initial actuation, are utilized in many of todays advanced reactor designs to a varying degree. A characteristic feature of such systems is that only few and simple components (with well-defined function) are allowed to be involved in the "passive" function in order to ensure simplicity and a rather reliable function. Still, further simplification and improvement can be gained if the function can be accomplished in an entirely passive way, without any components such as check valves, and without need for actuation.

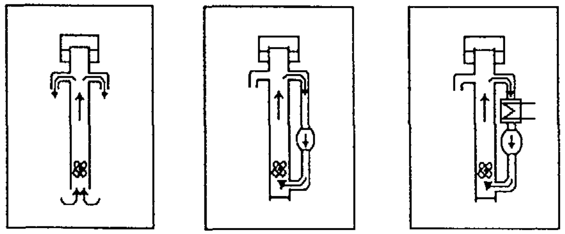
A reactor design that can fulfil the requirements of these three paragraphs, with a self-protecting reactor process, with a shutdown system without moving parts, and with a core cooling system that does not rely on activation and movement of components nor on external energy input, will meet the basic prerequisites for a "no core melt" plant. These are necessary but not sufficient conditions; careful engineering and design of the complete plant are quite obviously absolutely necessary supplements.

Locating the reactor core in a large pool of water will beyond doubt provide core cooling without need for external actuation or energy supply etc. if the power level of the reactor can be restricted and prevented from exceeding the cooling capability of the pool water. The latter condition is easily met by dissolving boron in the water; an adequate boron concentration in the water will stop and prevent fission reaction in the reactor core. Having a core located at the bottom of a large pool of borated water does not make anyone happy, however. The crucial question is whether this can be utilized as a basis for a power reactor design in which normal operation is accomplished and ensured by means of control systems and pumping forces etc. but where the system will revert to its "cooled, shutdown condition" by itself when operating conditions exceed acceptable limits. If a practical solution to this problem can be found, a "no core melt" plant can be built.

The use of quotation marks here indicates that this may not be valid in absolute terms but within certain limits with respect to initiating events etc. (cf. comment above on need for a threshold of occurrence frequency). This observation is also in correspondence with the discussion of safety concepts in the Description of "Safety related Terms for Advanced Nuclear Plants" that has been prepared by the IAEA. Care is recommended to add qualifications to stated terms - e.g., valid under this and that condition. In line with that and the "Good neighbour" approach above the practical solution and its implementation in a reactor design should ensure that no core melt will occur in the event of credible occurrences and accidents. These include all events in the traditional "Design Basis Event" packages used for licensing analysis of present-day reactors, as well as less probable events ("beyond" these "Design Basis Event" packages), if possible extending to individual events and event sequences with estimated occurrence frequencies down to about 10^{-6}.

Designing for core melt prevention

ABB Atom in Sweden has for more than a decade been working on reactor designs in which core melts are prevented - the PIUS type reactors. These are based on the "core submerged in a borated pool"-principle outlined above as one departure point for designing a "no core melt" plant, and a unique physical arrangement which makes it possible to operate the plant at power in normal operation with an "isolated" reactor primary loop even though it is in reality in open connection with the surrounding borated reactor pool water. [Cf. figure 3] Reactor coolant pumps draw reactor coolant from the top of the riser, through steam generators or heat exchangers, and bring it back to the reactor core inlet, and by speed control of the pumps the flow rate can be adjusted to match the natural circulation flow rate through the core and up the riser, as determined by the thermal and gravitational conditions of the primary loop compared with those of the reactor pool. This adjustment of pump speed (pump flow rate)



A B C

A Heating of water in a pipe results in a water flow up through the pipe The pipe is connected to a pool of water at top and bottom, heated water flows out at top, cold water enters at bottom.

B By means of a pump and speed (flow rate) adjustment all rising water is returned to pipe bottom, no water outflow to the pool and no inflow from it A separated pump loop has been accomplished with permanent openings to a surrounding pool of water

C By means of a heat exchanger (or steam generator), inside or outside the pool, reactor heat can be utilized When pump work is lost, or reactor power increases, hydraulic balance will be disturbed, providing automatic "inherent" self-protective feedback

Figure 3 - Operating principles of PIUS

establishes a "no flow" region and pressure equilibrium between the primary loop coolant and the reactor pool water in the opening below the reactor core, and due to the natural circulation flow up through the riser also in an other opening located in the vicinity of the riser top.

By this arrangement the "isolated" primary loop can be operated at a low boron content and elevated temperatures, as needed for the power operation, whereas the reactor pool is kept at low temperature and high boron content. As long as the power production in the core is balanced by the heat extraction in the steam generators (or heat exchangers) the reactor operation will be quite stable without water exchange in the openings, but in the event of disturbances the pressure balances will also be disturbed and water may flow into or out of the primary loop. For small, frequent disturbances which do not call for a reactor scram, the pump speed control will adjust the pump flow rate in such a way that there will not be any net in- or outflow; temperature monitors in the openings (designated "density locks") will detect displacement of the interface between warm and cold water and "order" pump flow in- or decrease. An increase is needed when the plant load decreases, as the primary loop temperature then increases; and plant load increases similarly result in pump flow decrease. It should be noted that the locks are provided with buffer volumes in order to ensure stable power plant operation during normal situations.

In the event of more severe transients the pump speed control can no longer counteract the changes and borated pool water will enter the primary loop, either through the lower density lock or through the upper one. The ingress of pool water tends to stop the nuclear reaction (to shut the reactor down) but in some transients the reactor will not shut down completely but continue operation at a safe power level.

The natural equilibrium state of this arrangement is "a shutdown reactor with the core cooled in natural circulation by borated pool water"; the operational point can only be maintained by adding pump work and adjusting this pump work to match the operating conditions.

This brief description has outlined the general arrangement of PIUS reactors with the intent of showing that the general principles for accomplishing a "no core melt" plant can be adapted and realized in a practical reactor design, but to provide background for qualifying evidence a more detailed review of its different design features is needed.

The PIUS reactor

The PIUS reactor concept is based on well established LWR technology. Basically, it is a passive and simplified, reconfigured PWR incorporating also some BWR features. The current standard version has a nominal power output of 600 MWe. Its basic arrangement is outlined in Figure 4, and the main plant parameters are provided in Table 1. The modified arrangement reflects the goals of achieving increased simplicity and safety, in particular with respect to protection of the reactor core in possible accident scenarios.

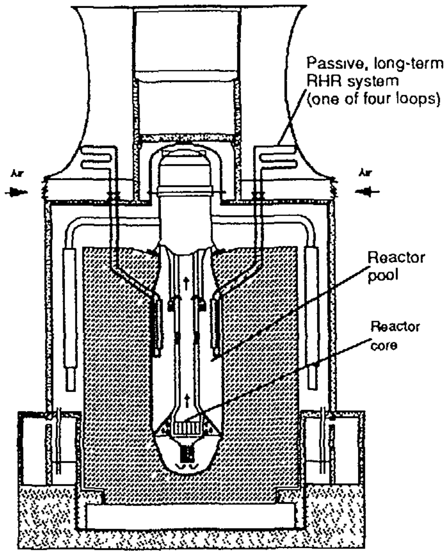


Figure 4 - Safety-grade structures of PIUS

The core is a low-rated, open PWR type core ensuring large fuel margins. And in correspondence with the core cooling concerns above, it is located near the bottom of the reactor pool, containing a highly borated water mass. The reactor does not use control rods, neither for shutdown nor for power shaping. Reactivity control is accomplished by means of reactor coolant boron concentration and temperature. Burnable absorber in the fuel rods is utilized to suppress excess reactivity and to control power distribution. This way, the coolant boron concentration can be kept relatively constant over the fuel operating cycle. Hence, a strongly negative moderator temperature coefficient of reactivity can be achieved under all operating conditions, and this is utilized for a convenient power control of the reactor.

The reactor pool is enclosed in a large prestressed concrete reactor vessel, and the leaktightness of the cavity is ensured by avoiding penetrations and providing a double barrier system, a stainless steel plated liner on the inner surface and a carbon steel membrane embedded into the concrete. The pool water is cooled by two systems; one with forced circulation through out-of-vessel heat exchangers and one with entirely passive function, utilizing natural water circulation and dry, natural-draft cooling towers. The passive system ensures the pool cooling in accident, and station blackout, situations and prevents boiling of the reactor pool water inventory. In the hypothetical case that all pool cooling systems fail, the water inventory ensures the core cooling for a protracted period of time (several days).

Table I: PIUS 600 Main Data

Core thermal power	2,000 MW
Net electric power	640 MWe
Circulating water temperature	15°C
No of fuel assemblies	213
Core height (active)	2 50 m
Core equivalent diameter	3.76 m
Average fuel linear heat rate	11.9 kW/m
Average core power density	72.3 kW/l
Core inlet temperature	260°C
Core outlet temperature (mixed mean)	290°C
Operating pressure (pressurizer)	9 MPa
Core coolant mass flow	13,000 kg/s
Average burnup	45,500 MWd/t
Equilibrium core ingoing enrichment (12 months)	3.5%
Concrete vessel cavity diameter	12 m
Concrete vessel cavity volume	3,300 m^3
Concrete vessel total height	43 m
Concrete vessel thickness	7-10 m
No. of steam generators and coolant pumps	4

From the reactor core, the heated coolant passes up through a riser pipe and leaves the reactor vessel through nozzles on the sides of an upper plenum. The coolant continues in four hot leg coolant pipes to steam generators, mounted on two sides of the concrete vessel. The main coolant pumps which are located below the steam generators, are glandless, wet motor design pumps similar to those that have been utilized in the ABB Atom BWR plants. The cold leg piping enters the reactor vessel through nozzles at the same level as the hot leg nozzles, and the return flow is directed downwards to the reactor core inlet via a downcomer.A siphon breaker arrangement is provided to prevent siphoning off the reactor pool water inventory in the hypothetical event of a cold leg pipe rupture. During normal operation, the siphon breaker does not affect the water circulation. At the bottom of the downcomer, the return flow enters the reactor core inlet plenum.

A pipe that opens to the enclosing reactor pool, is located below the core inlet plenum. A tube bundle arrangement inside this pipe minimizes water turbulence and mixing and ensures stable layering of hot reactor loop water on top of the colder reactor pool water. This pipe, with the bundle arrangement and the stratified water, is called the lower "density lock" and is one of the special components required to implement the basic design philosophy. Temperature measurements are used to determine the position of the hot/cold interface, for con trolling the speed, and hence the flow rate, of the main coolant pumps Another density lock arrangement is provided at a high location in the pool, connected to the upper riser plenum This reactor configuration, with the two continu ously open density locks connected to the high boron content pool water, pro vides the basis for the PIUS principle

A natural circulation loop from the pool, through the lower density lock, to the core, up the riser, through the upper density lock and back to the pool, is always present, but during normal plant operation, it is kept inactive by means of the main coolant pumps In the event of a severe transient or an accident, the natural circulation flow loop is activated, providing both reactor shutdown and continued core cooling

In a similar way as in other LWR plants, the nuclear steam supply system is enclosed in a containment structure, experience has shown that this mitigative feature must be maintained, for acceptability reasons The containment is in turn partly enclosed in a reactor building Pools for storing spent fuel and for reactor internals during refuellings are arranged in the reactor building, on top of the reactor containment The containment and the fuel pool portion of the reactor service room are designed with sufficient strength to provide protection against a crashing aircraft

PIUS Safety Performance

The main emphasis in the development work has been to prevent core degra dation accidents under any credible conditions, **without** recourse to the function of safety equipment that need actuation signal or power, or to operator actions or interventions, in other words, in an entirely passive way But, in line with the concerns stated in the general discussion above, the economy and oper- ability of the plant must not be sacrifized to achieve this, however

Accident analyses performed so far confirm that the safety goals are fulfilled No accident sequence leading to core degradation has been identified

The plant response to a postulated large LOCA, a double ended cold leg pipe rupture at a low location, can illustrate the favourable safety performance Initially, primary loop water flows out through both the hot and cold leg pipe nozzles, borated water enters the core from the pool and shuts the reactor down (Cf figure 5 curves)

Outflowing primary loop water flashes to steam in the containment, and its gas atmosphere is compressed The containment is of the pressure-suppression type and most of the compressed gas will be displaced to the gas compression chamber in the wetwell, via the blowdown pipes and the condensation pool The steam flowing down into the pool condenses, and the peak pressurization of the containment will be limited by the heat capacity of the pool

The hot leg pipe outflow stops when the water level in the vessel has dropped below the nozzles Then the containment pressure equals the reactor pressure,

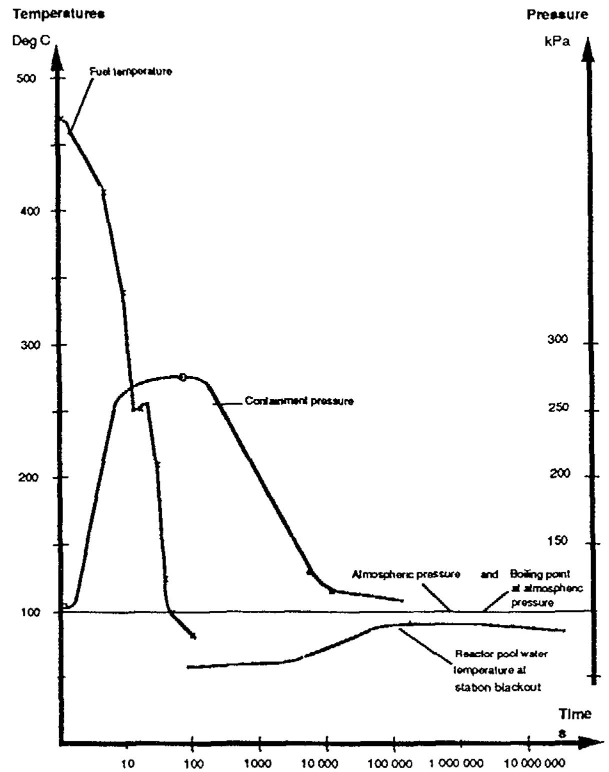


Figure 5 - PIUS response to large LOCA

and the cold leg outflow stops due to the siphon breaker arrangement The core is cooled by reactor pool water in natural circulation, and the decay heat is ab sorbed in the pool The pressure in the containment attains a peak and then decreases due to steam condensation on containment walls and structures In about 2 hours it is down to slightly above atmospheric pressure again

The reactor pool is cooled by the passive system arranged in four groups, each with a cooling tower on top of the reactor building Postulating failure of one group, the reactor pool water temperature will still be kept below boiling tempe rature at atmospheric pressure

The accident does not result in fuel damages, and the release of radioactive material to the containment is determined by the amount of such products that was present in the primary loop water prior to the accident [The increased iodine concentration following from spiking will remain in the reactor pool water since there will be no subsequent boiling.] Hence, release to the containment will be very small and combined with moderate and short pressurization of the containment, the release to the environment will be minimal; the whole body dose at the plant fence has been calculated to about 1 mrem.

Conclusion

These deliberations show that the "good neighbour" requirement of "no-release plants" can be met by present-day LWR plants, and by evolutions from these designs, by adding some equipment and systems. They also show that the prime threat itself, the core degradation, can be eliminated as a practical concern, and reveal a feasible way of doing so, the PIUS design concept.

In PIUS, the protection against core degradation accidents is ensured by the laws of physics alone. Intervention of active systems is needed to keep the reactor in operation, not for safety, preventing it from reverting to a state of shutdown and natural circulation core cooling. [Cf. figure 6] It is based on established LWR technology, and the existing regulatory framework should be sufficient as a basis for licensing. From a licensing point of view, the concrete vessel and the absence of control rods represent important departures from current technology, but they are also, together with the totally passive safety systems, the key elements for the favourable safety performance. The absence of control rods is actually an advantage since mechanical devices and interacting detector and insertion systems are eliminated. The risk of serious reactivity insertion due to control rod malfunctions is also eliminated.

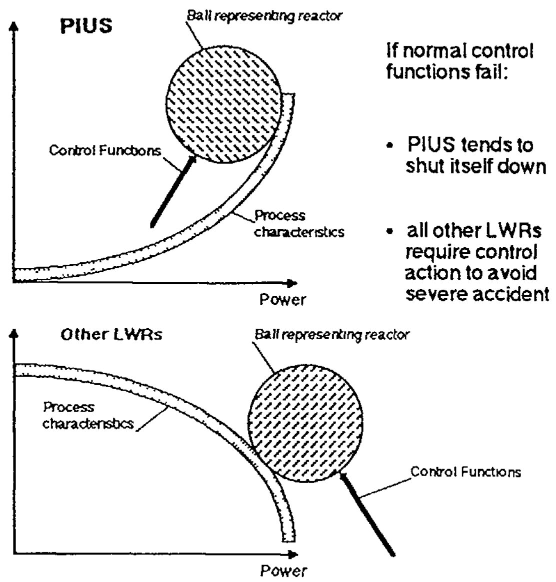


Figure 6 - Operating principles of PIUS vs other LWR plants

Compared with existing LWR designs, a number of safety-grade systems have been eliminated, allowing major simplification. Thus, PIUS should be simpler to operate and maintain than present-day LWR plants.

The self-protective thermal-hydraulics have been successfully demonstrated in normal and under severe transient conditions. Experiments that have been carried out, have provided adequate evidence of the feasibility and practical operability, but further tests are planned for verification of detailed design and performance of the new features.

References

[1] C. Sundqvist, K. Hannerz, L. Nilsson, T. Pedersen; Concept Status and Marketing Strategy of ABB PIUS and SECURE Reactors; Special Session on Development of Thermal Reactors, Jahrestagung Kerntechnik '91, Bonn, Germany

[2] T. Pedersen; Reactors take a large step towards "inherent safety"; Power Generation Technology 1990/91

[3] C. Sundqvist, L. Nilsson, T. Pedersen; Aspects on Containment - Philosophy and Design, IAEA TCM on Progress in Development and Design Aspects of Advanced Water-cooled Reactors, Rome, Italy

[4] C.Pind; Some Natural Convection Phenomena in the PIUS Reactor Design; Problems, Calculations and Tests.; Special Session on Natural Circulation Cooling Problems, Jahrestagung Kerntechnik '91, Bonn, Germany

[5] EPRI; Advanced Light Water Reactor Utility Requirements Document, Volume I, March 1990

SUMMARY OF THEORETICAL ANALYSES AND EXPERIMENTAL VERIFICATION OF THE PIUS DENSITY LOCK DEVELOPMENT PROGRAM

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Abstract

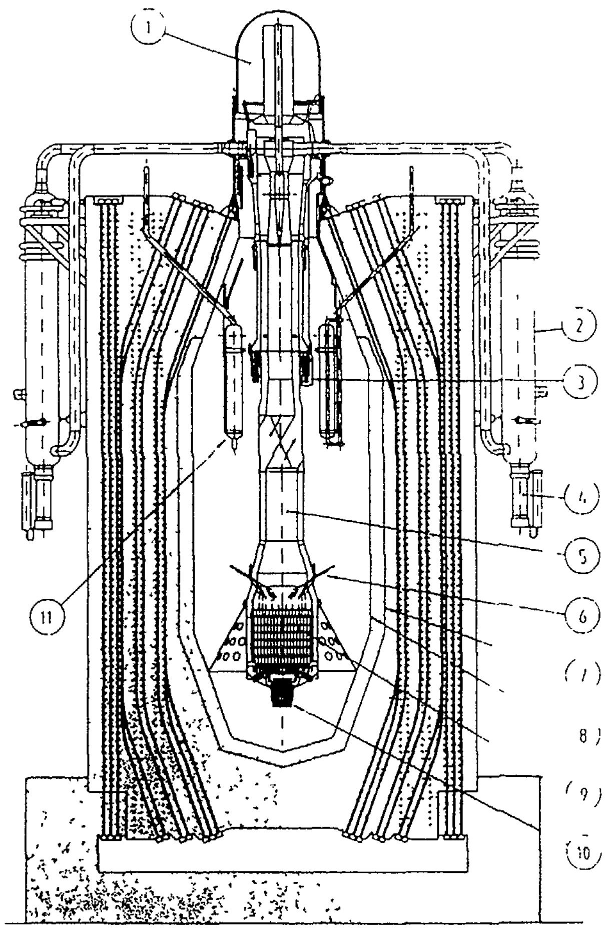
PIUS belongs to the next generation of LWR plants and has been under development by ABB Atom during the last decade. The core and the primary loop coolant guiding structures are submerged in a large pool of highly borated cold water. The cold borated pool water is separated from the hot primary water with low boron content by Density Locks where hot water is stably layered over cold water building a density interface that most effectively blocks transport of boron from the pool water to the primary water

In case of pump trip or station black out, the density locks "open" fully and with a low pressure loss admits pool water into the primary system where it shuts down fission reaction in the core and by natural circulation takes the decay heat via coupled natural circulation systems to the ambient air. No mechanically moving parts are needed to open this cooling circuit.

The Density Locks represent new and unique PIUS features that have been tested extensively in ABB Laboratories. Theoretical studies of the transport phenomena in the Density Locks have been verified experimentally and give as a result that the transport of boron through the locks during normal operation has very little influence on operation and economy The tests have covered small scale experiments as well as full scale and high temperature and pressure tests. A final test in full scale but limited to atmospheric pressure and 100 deg C is planned for the near future and will give information of the stability of and the boron transportation through the density lock for the operation conditions where stability is most endangered.

The PIUS reactor has been under development by ABB Atom during the last decade and is known to the nuclear public through many articles (Ref 1, 2, 3).

The NSSS of the PIUS reactor are submerged in a large pool of highly borated water as can be seen from Fig 1 showing the main components of the nuclear steam supply system. The pool and the primary flow guiding structures are pressurized to 95 bar and located in a concrete vessel with an upper steel extension. During normal operation the primary circulation flow of 13000 kg/s is driven through the core at the bottom of the pool by 4 main coolant pumps.



1 Pressurizer steam volume
2. Steam generator (4)
3 Upper density lock
4 Main coolant pump (4)
5. Riser
6 Core instrumentation
7 Embedded steel membrane
8 Pool liner
9 Core
10 Lower density lock
11 Submerged pool cooler, cooled in natural circulation by ambient air.

Fig 1.
PIUS - Main features of NSSS

The coolant pumps are suspended below the 4 external steam generators, where the 2000 MWth power is transformed to secondary steam of 40 bar, giving about 600 MW electric power from the station.

The primary flow guiding structures in the pool keep the primary circuit with water temperatures at 260-295 °C separated from the pool water of 50 °C. The reactor power is controlled by the boron content in the primary system water. This primary water has a relatively low boron content of 100-200 ppm of boron in contrast to the 2000 ppm boron content of the pool.

The primary water is in always open connection with the poolwater through the upper and the lower density locks. In the density locks hot primary water with low boron content is layered above cold water with high boron content. The interface between hot and cold water is very stable and prevent penetration of poolwater into the primary system. The level of the interface between cold poolwater and hot primary water is in the lower density lock controlled by the pumpspeed. If one or more of the pumps fail, the control of the density lock level is disturbed and the cold highly borated poolwater enters the density lock and the primary system, thereby shutting of the reactor and cooling the core. A natural circulation path is then formed from the pool through the lower density lock to the core and up through the riser to the upper density lock and out to the pool again. This circulation is driven by the core decay heat warming the incoming poolwater to higher temperature and lower density, giving the driving pressure difference across the core.

Main new elements in the PIUS design are the density locks and the big pool. The density locks have been studied at ABB Atom since 1983 theoretically as well as with experiments. The upper density lock in PIUS is influenced by natural circulation in the pool and calculations for pool flows had to be made to correctly investigate the behavior of the density lock.

Density locks

The purpose of the density locks is to separate the primary system cooling water from the cold highly borated pool water. The separation of the two water volumes must function without a physical membrane or valve that could jeopardize the natural circulation cooling of the core during shutdown. Although the density lock is new as a reactor component, separation by density differences has been used for many years in other non-nuclear applications, e.g. in heat storage tanks, and is abundantly used by nature.

A key issue of the design of the density locks is that the driving pressure for the natural circulation cooling is limited by the hydrostatic pressure difference between the primary side and the pool side. It is therefore important to design for a low flow resistance in the density locks.

The level difference between the upper and the lower density locks must be so that under normal operation the pressure drop over the core equals this level difference times the density difference between poolwater and primary water. This imposes a limitation specially on the height of the core in PIUS.

When the natural circulation cooling system is transporting the decay heat from the core to the pool, the flow goes through the lower density lock, the core and the upper density lock. These three components give rise to most of the flow resistance in this path. The flow resistance of the density locks must therefore be low.

Another key demand on the density locks is that the mass transport of boron through the locks at normal operation must be very low. At normal operation of the plant there is no net flow through the locks but by diffusion and mixing there can be an influx of boron into the primary system and this influx has to be removed by an online purification system, which in its other end has an evaporator to separate boron from the primary water. The evaporator system is rather costly, so the influx of boron into the primary system must be kept low during normal operation of the plant.

The important mechanisms for mass transfer through the density lock are convection and turbulent and molecular diffusion. The classical literature about mixing across stable density interfaces describes the mixing as intermittent breaking of waves in the density interface. This led to the thought of making the horizontal distance for travelling waves very small, and the idea of a honeycomb structure was born. The honeycomb structure is located with vertical cell axis, so that the flow resistance to vertical flow is minimized. The reduction in transport observed in small scale experiments with honeycomb structures was almost three orders of magnitude against those without.

The first set of experiments was made at atmospheric pressure and the density difference was produced by salinity. The turbulence was produced by an oscillating grid on one side of the density interface. The presence of a honeycomb structure was found to reduce the mixing strongly provided the interface was sufficiently stable.

The most important stability parameter characterizing the interface is the so called overall Richardson number defined by

$$Ri = \frac{\Delta\rho \cdot g \cdot h}{\rho_0 u^2} \tag{1}$$

$\Delta\rho$ is the density difference between the two sides of the interface, g is the acceleration due to gravity (9.81 m/s^2), h is the honeycomb height, ρ_0 is the mean density and u is the turbulence rms-value just outside the honeycomb. The Richardson number can be interpreted as the ratio between the stabilizing force $\Delta\rho g h$ and the turbulent kinetic energy $\rho_0 u^2$ of the fluid with density ρ_0 and velocity u. A large value of Ri corresponds to a stable interface, and a smaller value of Ri gives a more instable interface. Ri must be well above 1 in all cases of interest where a honeycomb is used to reduce mixing.

Initial experiments were followed by a systematic investigation of the dependence on the most important parameters. The extended set of experiments were performed at the ABB laboratories in Västerås, and at The Royal

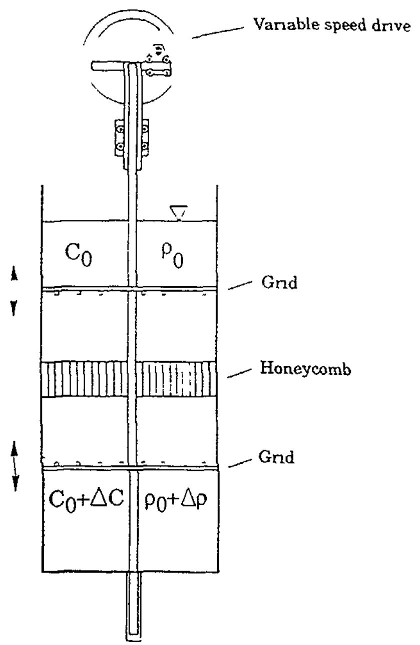


Fig 2
Sketch of the apparatus used in the experiments

Institute of Technology, Stockholm. They were performed at room temperature in a container with a honeycomb across a density interface. The density difference was established first by salinity and later also by temperature. The turbulence was generated on both sides of the density interface by oscillating grids as shown schematically in fig 2. Stroke and frequency of the grids were adjustable.

To correlate the experimental results one uses the effective entrainment velocity Ue defined by the equation

$$\frac{dC1}{dt} = U_e \cdot \frac{C2 - C1}{H} \tag{2}$$

Where C1 and C2 respectively are the concentration in the two layers, H is the depth of each of the two layers (which is assumed to be equal). $\frac{dC1}{dt}$ is the rate of

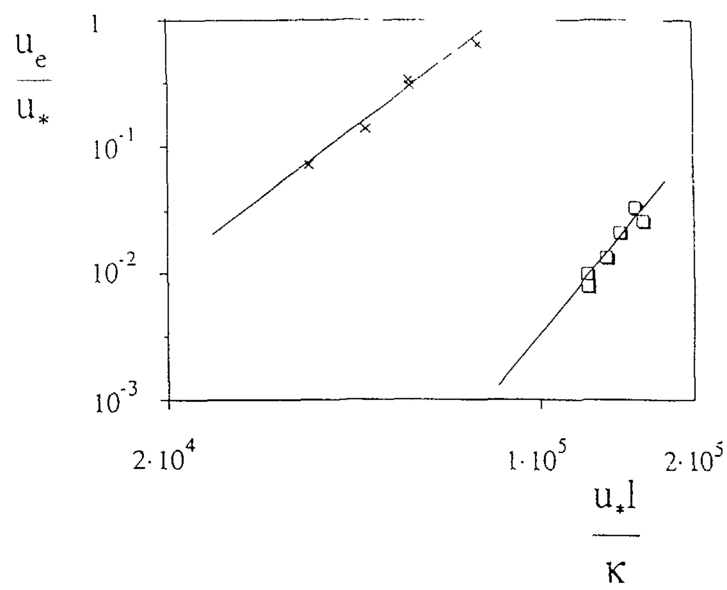


Fig 3
Comparison of mixing with and without honeycomb The slope of the upper line is 3 and that of the lower line is 4 The density ratio $\frac{\Delta\rho}{\rho}$ = 0.045, the honeycomb - length = 0.1 m, and the length - to - width ratio = 10 The symbols in the figure are: X for the unprotected interface and □ for the interface within a honeycomb

change of concentration in the top layer. The entrainment velocity is related to an effective diffusivity κ_e by

$$\kappa_e = U_e \quad \Delta h \tag{3}$$

where Δh is the length (hight) of the gradient.

The reduction of the transport by introduction of a honeycomb across the density interface was found to be substantial. A direct comparison by Gebart (ref. 6) showed that the reduction could be as large as 3 orders of magnitude (see Fig. 3).

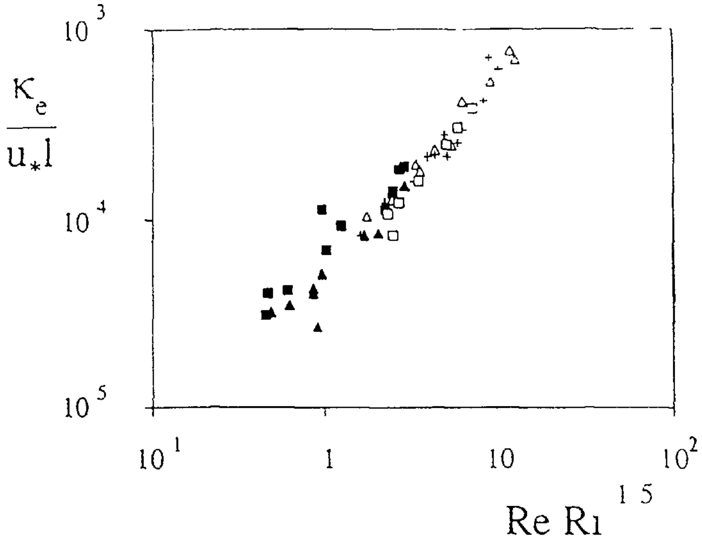


Fig 4

Effective diffusivity for 3 different density ratios and 3 different viscosities The data collapse when plotted against ReRi 1 5 The honeycomb length = 0 1 m and the length to width ratio = 10

Symbol	$\frac{\Delta\rho}{\rho}$	ν (m /s)
△	0 01	1 10^{6}
▲	0 01	3 10^{6}
+	0 02	1 10^{6}
■	0 04	? 10^{6}
□	0 045	1 10^{6}

Collapse of the measured data for various density ratios and viscosities is obtained if the nondimensional effective diffusivity is plotted versus Re Ri 1 5 (Fig 4) A good fit to the data from the salt experiments is obtained with

$$\frac{\kappa_e}{u\ l} = A\ \ \mathrm{Re}\ \ \mathrm{Ri}\ 1\ 5 \qquad (4)$$

where the Reynolds number is based on turbulent intensity, u, and the turbulent length scale l (dimension of grid) The constant A is 2 5 10 ^{5} in the salt experiments

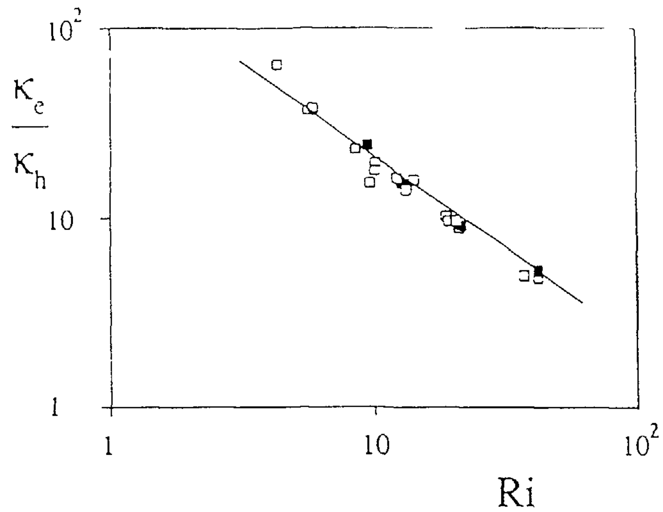


Fig 5

Effective diffusivity for heat transport normalized with the heat diffusivity in water versus Richardson number The slope of the line is 1 The honeycomb length = 0 1 m and the length to width ratio = 10 The results for $\frac{\Delta\rho}{\rho}$ = 0 01 is represented by □ and the results for $\frac{\Delta\rho}{\rho}$ = 0 02 is represented by ■

A number of experiments were performed in water but with temperature as the density affecting agent When these data were normalized with the mole cular diffusivity for heat κ as in Fig 5, all the data collapsed around a single line The effective diffusivity for these heat experiments can be expressed as

$$\frac{\kappa_e}{\kappa} = B\ \ \mathrm{Ri}\ 1 \qquad (5)$$

In some of the heat experiments a low concentration of salt was added (as in the PIUS density locks) The resulting effective heat flux was identical to the flux measured without salt The most interesting result from the combined heat and salt experiments is that the slope of the salt transport curve is diffe rent from that for the heat curve

The result of all the experiments is that the transport through a density profile in a honeycomb is very strongly dependent of the Richardson number

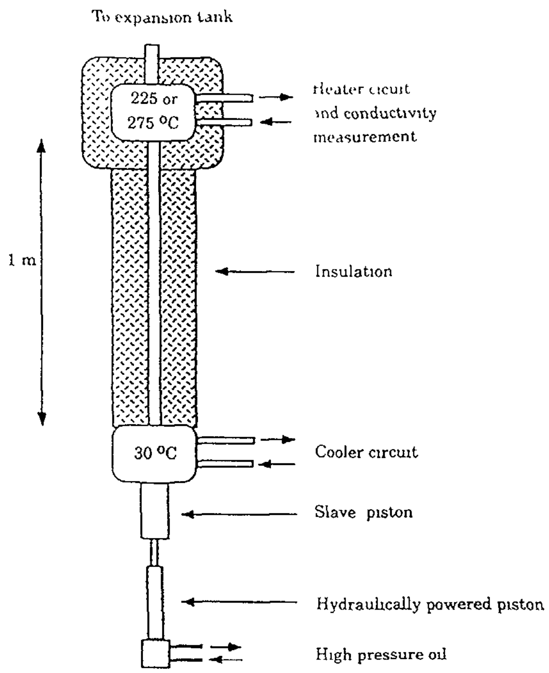


Fig 6.

One-pipe HT apparatus

After the low temperature tests of the density lock a full temperature full scale test was made on a density lock pipe. The pipe dimensions were as intended for the PIUS density lock. The high temperature test equipment is shown in Fig. 6. The apparatus was placed in a high pressure vessel, that allowed the high water temperatures and pressures. The hydraulically powered piston force disturbances with varying amplitudes and frequencies onto the lower pipe end. The temperature gradient lies in the pipe with 1 m height, and the transport of dissolved matter is measured. The measured transport gave a good fit to the low temperature experiments.

It has been found earlier that the transport through the density lock will increase if pressure disturbances with a certain frequency was imposed on the tube end. This frequency give rise to oscillations of the water in the tube as a whole, thereby increasing transport through the tube. These oscillations (sinusoidal movements of the water up and down in the tube) had been measured in an experiment earlier on a full size density lock, but will in this paper be mentioned in the part concerning natural convection in the pool. The frequencies and amplitudes from the full size density lock experiment were fed to the hydraulic piston in one series of experiments. In other series of experiments the hydraulic piston imposed sinusoidal movements to the pipe end at constant amplitude and frequency, and in a third serie the piston moved with constant amplitudes but with the frequency sweeping over a narrow band.

Applied to the PIUS design at an estimated disturbance level of 3 cm/s, the effective diffusivity was estimated to be $1.7 \cdot 10^{-7}$ m^2/s. This means for the PIUS design that the increase in primary boron content would be less than 1 ppm in 24 hours with the purification system disconnected. This value is so low that it has no influence on the economy of the PIUS reactor.

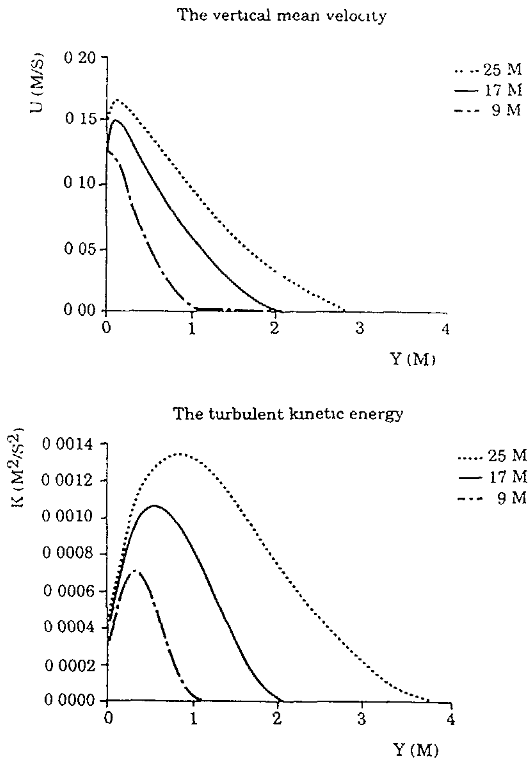


Fig 7.

Velocity and turbulent energy profile

Natural Convection in the Pool

The large pool of highly borated water in the PIUS vessel surrounds the primary internal parts consisting of mainly the core, the riser and the downcomer. The primary system has a mean temperature of 275 °C and the pool keeps a temperature of 50 °C at normal operation. Thermal energy is transmitted through the primary parts thermal insulation and this gives rise to natural circulation in the pool with mainly upgoing velocity near the internal parts and downgoing velocities nearer the walls of the concrete vessel. The computer code Phoenics has by Lena Hedgran (ref. 7) at the ABB Research Center been used to simulate the thermal boundary layers along the riser in the reactor pool in order to estimate the turbulence intensity. The simulation of natural convection by means of Phoenics was checked by a comparison with experiments on thermal boundary layers from the literature.

The result of these calculations are seen on Fig. 7. The boundary layer calculations for PIUS resulted in a maximum mean velocity of 0.17 m/s and a maximum turbulent kinetic energy of $1.35 \cdot 10^{-3}$ m^2/s^2. The heat flux through the thermal insulation had a constant value of 10.0 kW/m^2. The max temperature increase above the pool temperature was found to be only 2.5 °C. The calculations gave with hand that an upper limit for the magnitude of the fluctuating vertical velocity would be 0.03 m/s - 0.052 m/s. The question was now how much these velocities can disturb the upper density lock whose underside is placed in the vertical upflow of this natural circulation.

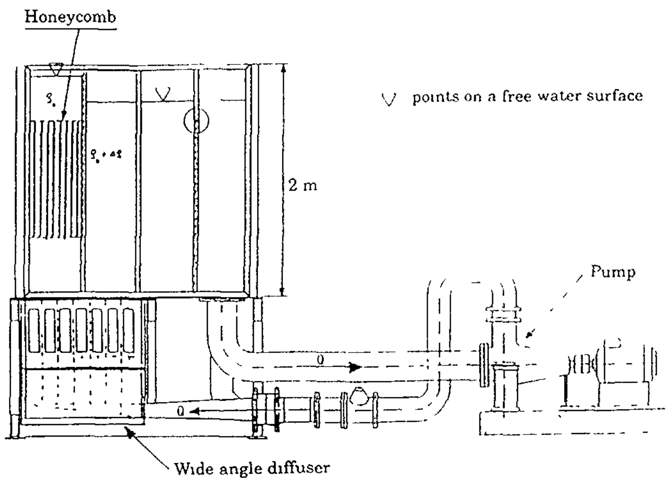


Fig 8.

Density Locks - Arrangement of atmospheric test

A full scale experiment was conducted to clear up this question. The apparatus used is shown in Fig. 8. The pump circulates water in the system and the waterflow is directed upwards against a simulated density lock. Before hitting the honeycomb grid from below the water was led through a wide angle diffusor and through adjustable slots. These slots were adjusted so that the water at hitting the honeycomb had the velocity profile as calculated in the above related Phoenics calculations. Above the honeycomb the water had a lower density than in the circulating water. The density difference could be controlled with temperature or with salinity. The exchange through the Honeycomb (Density lock) was then measured, giving an effective diffusion coefficient as earlier defined in this paper. During the experiments it was observed that a sloshing movement appears in the vertical honeycomb channels. The vertical channels resonate at a given frequency (the Brunt Väsilä frequency) and in the experiments the vertical velocity inside individual tubes was measured. The disturbance level and the velocity profile under the density lock were varied. This was done to give information on how these alterations resulted in typical oscillations inside the density lock. Later this information was used to simulate the sloshing movements in the high temperature experiment already mentioned. The vertical velocity inside the honeycomb tubes was measured with an ultrasonic anemometer capable of resolving very low (and reversing) velocities. The velocity signal was integrated in time to give a vertical displacement which was found to be almost purely sinusoidal in time but with varying amplitude. The characteristic "sloshing" frequency in the experiment agrees very well with theoretical calculations.

Final full scale test on the density lock

The final step in developing the density lock design for PIUS will be a full scale test at about 100 °C to test the instrumentation and measure the transport velocity at the lowest Richardson number that is encountered during operation of the PIUS reactor.

REFERENCES

1. Nuclear Technology Vol. 91 (July 1990) pp 81-88.
   Kåre Hannerz et. al.
   "The PIUS Pressurized Water Reactor: Aspects of Plant Operation and Availability"

2. ENC '90 Proceedings Vol. II pp 701-718.
   Kåre Hannerz, Lars Nilsson and Tor Pedersen
   "Nuclear Power Facing Human Fallibility - The PIUS Design"

3. Nuclear Science and Engineering: 90, pp 400-410 (1985)
   Dusan Babala, Ulf Bredolt and John Kemppainen
   "A Study of the Dynamics of the SECURE Reactor: Comparison of Experiments and Computations"

5. Sixt Proceeding of Nuclear Thermal Hydraulics 1990 Wintermeeting November 11-16, 1990 Washington DC.
Ulf Bredolt
"PIUS - Verification of Primary System Boron Transport Phenomena by Comparison of Experimental Results with Simulations using the RIGEL-Code"

6. The Royal Institute of Technology, Department of Mechanics. TRITA-MEK-88-02. ISSn 0348-467-X
B. Rikard Gebart
"Transport Mechanisms at a stable Density Interface - Influences of Vertical Rigid Walls"

7. Hedgran, L. 1985
"Numerical simulation of the thermal boundary layers along the riser in the reactor pool of Secure"
ABB Corp. Res. internal rep: TR KZBH 85-075

LONG TERM DECAY HEAT REMOVAL WITH PASSIVE FEATURES IN ALWRs

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Abstract

In Advanced Light Water Reactors the efforts towards the use of passive mechanisms for the most relevant safety systems lead the designers to introduce innovative solutions to the problem of post-accident containment cooling.
For example, in the PIUS type reactors a natural circulation water loop cools the reactor pool and transfers heat to an external dry cooling tower. SBWR type reactors make use of an "Isolation Condenser" submerged into a large water pool; following a postulated accident the pool water boils off, releasing steam to the atmosphere and assuring passive containment cooling for at least three days.
A further improvement could be the Isolation Condenser Pool Cooling System (ICPCS), at present under study in Italy. Its expected benefits are:

- the elimination of constraints on the "grace period" duration;
- the possibility to avoid an extended release of a visible and potentially radioactive steam plume.

According to the design philosophy of Advanced Light Water Reactors, a natural circulation system using the atmosphere as final heat sink is envisaged.
Among the topics to be solved in the feasibility study of the ICPCS, a particular problem arises from the presence of a large amount of non condensible gas in the pool atmosphere, that can greatly impair the natural circulation and heat transfer mechanisms involved.
After examination of several possible solutions, very promising features have been pointed out in the use of reflux condensing heat exchangers directly connected to the pool gas space; non condensible gases can be vented during the earlier phase of operation by means of valves operating in passive way.

1. INTRODUCTION

A new generation of nuclear power plants is at present being studied and developed in many different countries. The main effort is directed to increase the acceptance of nuclear power by evolutions of existing reactor concepts, with the introduction of simpler plant configurations. This is especially true as far as safety related systems are concerned: whenever is practical, use is made of "passive" mechanisms, which means that system operation does not rely on human action, external energy supply and, if possible, on moving machinery or fluid motion (1).

A specific problem of nuclear plants, which may have a significant importance from the safety viewpoint, is decay heat removal after reactor shutdown or postulated accidents. In next generation reactors this need is often satisfied by use of passive mechanisms, like natural circulation of fluids. For example, in the PIUS reactor a cooling loop, containing subcooled water, connects an heat exchanger submerged in the reactor pool to dry cooling towers located on the top of reactor building. In this solution, heat generated inside the reactor pool is transferred to the cooling loop, producing natural circulation of the loop water; heat is then released to the cooling towers, where natural circulation of atmospheric air takes place. The final result is therefore the decay heat release to the atmosphere without use of any external energy supply or fluid makeup.

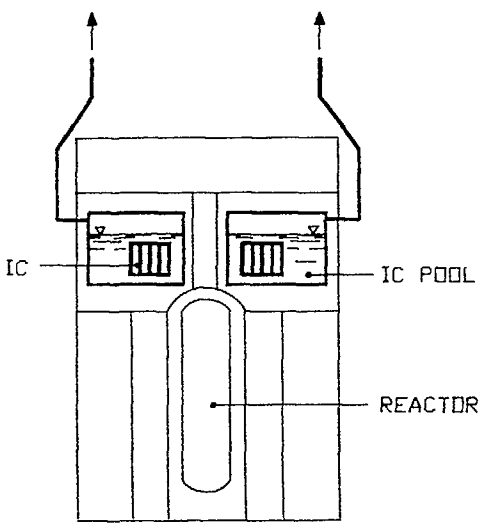

Fig. 1 Isolation Condenser schematic arrangement

In SBWR reactors a different solution to the same need has been envisaged. Several heat exchangers, named "Isolation Condensers" (IC's) and connected to the reactor system or to the containment drywell, are submerged in a large water pool, located inside the reactor building at a suitable elevation above the reactor (fig.1). In case of reactor isolation or following a postulated accident producing a release of decay heat to the containment, steam is condensed in the IC's and the condensate is returned by gravity to the reactor system or to the containment pool. Water present in the IC pool heats up to the saturation temperature and starts boiling; produced steam is directly released to the atmosphere. This process goes on, releasing decay heat to the atmosphere without any human intervention or external energy supply, till IC pool water has been completely boiled off. The presently established pool volume is sufficient to ensure decay heat removal for a period of at least 72 hours, named "grace period" according to the definition reported in (1).

In the frame of the studies performed in Italy by ENEL/DSR/VDN on next generation reactors, the possibility to increase as much as possible the grace period duration for the above described conditions was felt very attractive. Therefore CISE was charged to perform a feasibility study of a system able to satisfy this goal, that was named "Isolation Condenser Pool Cooling System" (ICPCS).

Main objectives of the study are the following:

- to identify the possible system configuration;
- to perform a preliminary sizing, in order to be able to estimate physical size, to predict performances and to evaluate costs;
- to identify critical areas for system design and to define research needs, if necessary.

At present the study has attained the first and the second of the mentioned objectives, while the last is in progress. This paper summarizes the main results till now achieved.

2. BENEFITS OF THE PROPOSED SYSTEM

The benefits expected from the introduction of the ICPCS system mainly concern the extension of the grace period duration and the environmental impact. Indeed, the capability to cool the IC pool water avoiding an extensive boiling and releasing heat directly to the atmosphere could allow decay heat removal for a theoretically infinite time period, independently of the pool water volume (it must be noticed, however, that a large water amount is nevertheless useful in order to exploit its thermal capacity during earlier transient phases, when decay heat power is largest). Moreover, the elimination of a prolonged (about two days) steam release to the environment could achieve two benefits:

- no effect of a plant malfunction would be visible outside the reactor building;
- the risk of radioactive release to the atmosphere, possible in case of IC tube bundle leakage, would be avoided.

3. MAIN HYPOTHESES ASSUMED

For the purpose of the ICPCS feasibility study, the following assumptions have been made:

a) reference is made to a SBWR type reactor, with a full thermal power of about 2000 MW;

b) ICPCS shall be sized for low pressure IC operation (that refers to IC's connected to containment drywell);

c) IC pool initial temperature is 50 °C, initial pressure is 0.1 MPa;

d) IC pool gas space is initially occupied by an air/steam mixture, with the steam partial pressure in equilibrium with the pool water temperature;

e) ICPCS design power corresponds to the decay power at the time when the pool water reaches 100 °C, considering a uniform temperature distribution inside the pool; this hypothesys provides a value of about 17 MW (see Fig. 2);

f) the total elevation available for natural circulation, measured from the top of IC to the top of reactor building, is 15 meters;

g) atmospheric air temperature has a conservative value of 40 °C.

It must be noticed that all these hypotheses have been used for the reference sizing of ICPCS; the effect of changing hypotheses e) and f) has been also studied, as will be explained later.

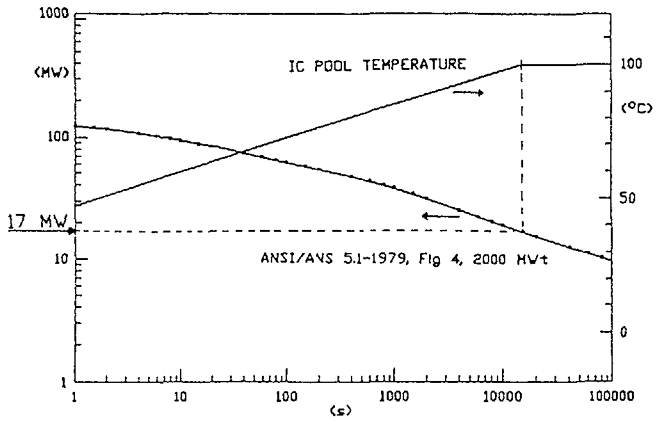

Fig. 2 ICPCS design power evaluation according to decay heat curve

4. DESIGN CRITERIA

The feasibility study has been performed taking into account, in addition to the requirements resulting from the above listed assumptions, the following criteria:

- ICPCS system must operate according to passive mechanisms, fulfilling as far as possible the passivity definitions reported in (1);
- the stresses on reactor building, due to differential pressure between IC pool and the atmosphere, must be kept as low as possible;
- the physical size of the equipment needed for ICPCS operation must be minimized, in order to get a reasonably small effect on reactor building general architecture and to reduce the impact on plant costs and construction time.

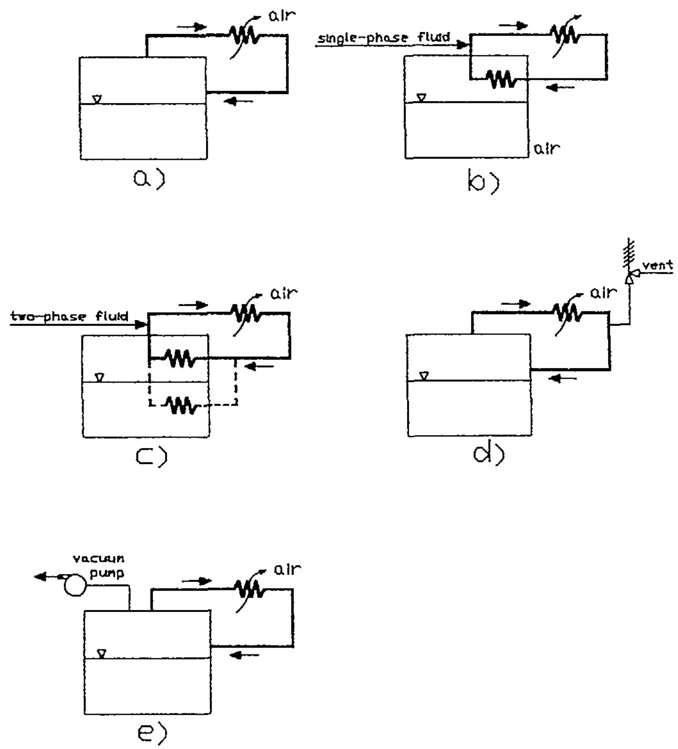

Fig. 3 Possible ICPCS configurations

5. IDENTIFICATION OF POSSIBLE CONFIGURATIONS

The most obvious way to satisfy ICPCS system requirements seems to be the condensation of steam produced *inside the IC* pool, releasing latent heat to an heat exchanger cooled by air in natural circulation and returning the produced condensate to the pool. Unfortunately, this process is hindered by the presence of a large air volume above the IC pool (hypothesis d) in Ch.3) that can greatly impair both steam/water natural circulation and condensation heat transfer. Therefore, a number of different configurations, schematically shown in fig.3, have been taken into consideration.

a) This solution, shown in Fig.3a, is the simplest one: a steam pipe directly connects the IC pool gas space to an heat exchanger cooled by air in natural circulation; condensate is returned to the pool. The main drawback is the presence of a large air amount in the pool cover gas (air mass concentration may be about 70% assuming a pressure of about 0.2 MPa during ICPCS operation); the air would accumulate in the heat exchanger stopping natural circulation and making the system uneffective.

b) In this case, presented in Fig. 3b, heat is transferred from the IC pool to the external, air-cooled heat exchanger via an intermediate loop full of liquid, establishing a single-phase natural circulation. That eliminates any non-condensible gas concern for fluid circulation. It can be noticed that this solution is very similar to the PIUS one. Disadvantages regard: the necessity of a second, internal heat exchanger which increases system complexity, size and cost;the introduction of additional temperature differences, due to the intemediate loop, that increase the external heat exchanger size if the pool pressure is assumed to be unchanged.

c) This solution (Fig. 3c) is quite similar to the previous one, but the intermediate loop operates with a two-phase fluid, boiling in the internal heat exchanger and condensing in the external one. In comparison with case b), further advantages are connected to a more efficient natural circulation, thanks to the bigger density differences, and to the practically isothermal intermediate loop that improves thermodynamic behaviour.

d) System configuration (Fig. 3d) is similar to case a), but the non-condensible gases are vented during early system operation by a valve actuated by system overpressure. When air vent has been performed, a sufficient heat transfer in natural circulation occurs, system pressure decreases and the valve can close, stopping the steam-air mixture release. This solution combines the simplicity of configuration a) with a thermodinamic efficiency even better than case c); the main shortcoming is that the system is not completely closed to the environment, reducing some benefits mentioned in Ch. 2. It must be considered, however, that steam-air release would be limited to a very short time (a few minutes).

e) This last solution (Fig. 3e) makes use of a vacuum pump in order to reduce air partial pressure during steady-state operation, improving system behaviour in comparison with case a). However the non condensible gases cannot be completely vented in this way, while the necessity of an active pump and the vacuum conditions in the pool during normal operation make this configuration less attractive than the previous ones.

According to the mentioned advantages and shortcomings, subsequent evaluations have been restricted to solutions c) and d); for case c) two different configurations, named c1) and c2) have been taken into consideration, corresponding to different locations of the internal heat exchanger: in the IC pool gas space or directly submerged in the pool water.

6. PRELIMINARY SIZING OF THE SELECTED CONFIGURATIONS

6.1 Solution c1) (intermediate two-phase loop, internal heat exchanger in gas)

6.1.1 Basic design

In this configuration, ICPCS is composed by the following equipment:

- an internal heat exchanger, located above the IC pool water level, composed by nearly horizontal plain tubes in which the intermediate loop fluid (supposed to be water) boils; steam produced by the IC condenses on the outer side of the tubes;
- an external heat exchanger, made of nearly horizontal finned tubes; the heat exchanger is located around the upper part of the reactor building, as shown in Fig. 4, and is cooled by air in natural circulation;
- all the necessary piping connecting the internal and the external heat exchangers.

The mentioned equipment has been sized according to well established heat transfer and pressure drop correlations; the results can be summarized as follows:

- the internal heat exchanger surface is about 2500 m^2, while the external one has about 38000 m^2 that require to locate the heat transfer panels all around the reactor building for a width of about 6 meters;
- the maximum pressure reached in IC pool is about 0.19 MPa for the hypotheses assumed (17 MW power, 40 °C air temperature).

6.1.2 Sensitivity studies

In order to check if performances of the above described basic design can be significantly improved with an acceptable increase of ICPCS system size and cost, two possible modifications have been taken into consideration:

- an increase of natural draft elevation from 15 m, used for basic design, to 30 m;
- an increase by a factor two of external heat exchanger surface.

The former change would produce a strong impact on reactor building size and cost without an appreciable performance improvement: the IC pool maximum pressure could be reduced to about 0.18 MPa. The latter modification could be done without big

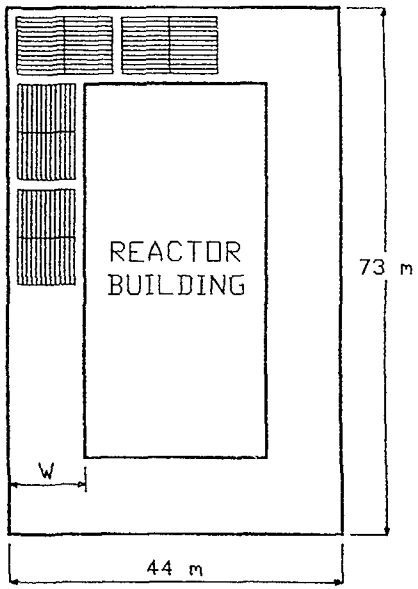

Fig. 4 Air-cooled heat exchanger schematic plan

effects on reactor building (notice that a twofold heat transfer surface is compatible with the same building enlargment if a 60° arrangement of finned tubes is used instead of the horizontal one); the IC pool maximum pressure would decrease in this case to about 0.17 MPa. So the heat transfer surface increase is possible and can be convenient if the heat exchanger additional costs are compensated by the reactor building stress reduction.

6.1.3 Effect of IC pool partitions

In the preliminary sizing performed reference has been made to the uniform temperature assumption in the IC pool water (Ch. 3, hypothesis e)) that brings to a ICPCS design power of 17 MW. This assumption is reasonable if IC pool is internally open, while it becomes unrealistic and non conservative in case of internal partitions. To evaluate the effect of this different situation, a division of IC pool into 12 smaller modules has been considered; in this case the water heat capacity of just one module has been taken into account, while a completely mixed gas space has been assumed. In this way, the ICPCS design power can be evaluated to be about 43 MW, because of the much faster water temperature increase. Assuming the same ICPCS sizing as in the basic design, the IC pool maximum pressure would be about 0.3 MPa, while an increase of all heat transfer surfaces by a factor 2.5 would be required in order to get the same performances as in the previous case.

6.2 Solution c2) (intermediate two-phase loop, internal heat exchanger in water)

6.2.1 Basic design

In this configuration, ICPCS is composed by the same equipment as in case c1), but the internal heat exchanger is located below the IC pool water level. A more efficient heat transfer mechanism occurs at the outer surface of this heat exchanger, i.e. free convection with single-phase liquid instead of steam condensation with large non-condensible gas concentration. Considering exactly the same heat transfer surfaces as in basic design of solution c1), a better thermodynamic performance is therefore obtained: maximum pressure inside the IC pool is about 0.15 MPa.

It must be noticed that in this case the uniform temperature hypothesis in the pool water should be critically evaluated; the water free convection certainly requires some temperature gradient between different pool regions. Therefore, the maximum water pool temperature could be appreciably higher than temperature of water flowing along the cooling surface, with the consequence of a maximum pressure in the IC pool higher than predicted. Some numerical analysis of pool velocity and temperature fields is required in order to get confident of the above indicated performance improvement.

6.2.2 Sensitivity studies and effect of IC pool partitions

The same checks of the influence of ICPCS sizing, as described in 6.1.2, have been performed also in this case, with very similar results.

The assumption of a IC pool partition has in this case two different effects:

- the design power increases to about 43 MW, as explained in 6.1.3;
- the internal heat exchanger must be inserted in the same pool module where the IC is located, with strict space limitations.

The latter constraint imposes to reduce the heat exchanger surface from 2500 to about 1400 m², with a further performance decrease in comparison with the basic design. Assuming the same size of the external heat exchanger, the maximum IC pool pressure has been estimated to be about 0.3 MPa.

6.3 Solution d) (non condensible vent)

6.3.1 Basic design

In this configuration, ICPCS is composed by the following equipment:

- an external heat exchanger, made of 60° inclined, finned tubes; the heat exchanger is located around the upper part of the reactor building, as in previous solutions, and is cooled

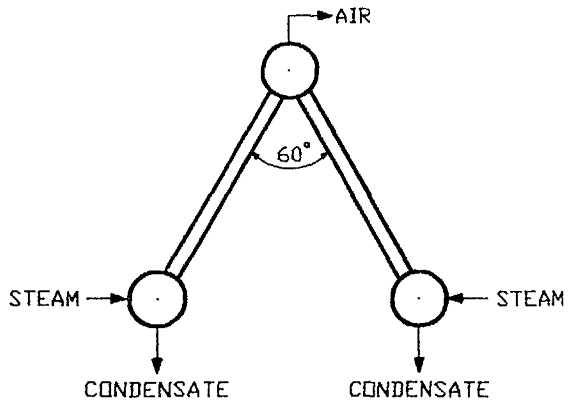

Fig. 5 Air-cooled, reflux condensation heat exchanger

by air in natural circulation. Inside the tubes the steam produced by pool water evaporation is condensed in upflow and the condensate produced flows down to the inlet header, creating a countercurrent flow situation (*reflux condensation*). A schematic arrangement of the heat exchanger is shown in Fig. 5. The general arrangement around the reactor building is similar to the previous ones (Fig.4);

- an overpressure operated valve for non condensible vent, connected to the heat exchanger upper header; it may be of the spring loaded type or preferrably could be a "fluidic" device (water seal) that would even better satisfy the passivity criteria;
- a vacuum breaker valve, required to avoid excessively low pressures in the IC pool that may result from steam condensation after non condensible vent; this valve could be easily integrated with the previous one, especially if a water seal solution is choosen;
- all the required piping connecting the IC pool to the reflux condenser.

The described system appears to be quite simple and reliable; in particular, its self-regulating features are very attractive. Indeed, a pressure increase produces a valve opening with non condensible release; that improves condensation heat transfer, lowering pressure and eventually leading to valve closure. This process goes on until the air has been sufficiently evacuated to allow a completely "close" operation mode. Symmetrically, the system regulates itself in case of subatmospheric pressure. Moreover, the gas vent location allows a preferential air release, minimizing both the steam outflow to the environment and the pool water inventory loss.

The mentioned equipment has been sized according to well established heat transfer and pressure drop correlations; particular care has been devoted to verify possible flooding problems in the reflux condenser tubes in all the possible operating conditions (notice that a 60° inclination is nearly the optimum one from the flooding viewpoint). A 0.15 MPa opening pressure has been assumed for the vent valve; the reflux condenser surface has been kept about the same of previous cases (39000 m^2), with different tube length and diameter to make them suitable to a reflux condenser application.

Main results of the basic design are:

- maximum IC pool pressure obviously is 0.15 MPa;
- heat exchanger panels occupy about 3.5 m around the reactor building;
- air/steam release would have a total duration of about 3 or 4 minutes and the water pool inventory loss would be less than 1000 kg.

6.3.2 Sensitivity studies and effect of IC pool partitions

The basic design has been performed keeping the same heat transfer surface as in previous solutions; however the thermodynamic performance of this case is considerably better thanks to the elimination of the intermediate loop and to non-condensible gas vent, so the system is oversized in respect to the assumed design power (17 MW). Therefore no sensitivity studies at increasing heat transfer surface or natural draft heigth have been performed. On the contrary, a surface reduction with further decrease of heat exchanger geometric size seems to be possible; no increase of maximum pressure, which is limited by the vent valve, would result, but the valve opening duration could increase if the heat transfer surface became too small. In order to quantify this possible effect and to define an optimum heat exchanger size, a more detailed, transient analysis is needed and this is beyond the purpose of the feasibility study.

The effect of introducing partitions in the IC pool in the present case would be an earlier vent valve opening due to the faster temperature and pressure increase; it has been evaluated that after air vent the basic design is already sufficient to exchange the required power (obviously greater than 17 MW) without any further steam release to the environment.

7. CONFIGURATION COMPARISON

Some significant findings of the preliminary sizing performed are summarized in Tab.1. The comparison is made for approximately the same external heat exchanger surface. The following observations arise:

- solution d) is better from the viewpoint of maximum pressure in the IC pool (notice that in this case maximum pressure is dictated by the vent valve opening value and can be easily decreased even more);
- solution d) is notably simpler and therefore cheaper and easier to be integrated in the reactor building;

Tab. 1 Summary of the preliminary ICPCS sizing

CONFIGURATION	c1)	c2)	d)
Maximum IC pool pressure (MPa)	0.192	0.152	0.15
External heat exchanger width w (see Fig.4) (m)	6	6	3.5
External heat exchanger surface (m^2)	38000	38000	39000
Internal heat exchanger surface (m^2)	2500	2500	N.A.
Maximum pressure for pool partitioning (MPa)	0.3	0.3	0.15
Completely closed system	YES	YES	NO
Intermediate loop	YES	YES	NO

- the effect of introducing partitions in the IC pool is practically negligible in case d), while it can significantly penalize the remaining solutions;
- solution d) main shortcoming is that the system is not completely closed to the environment; however the short duration and the limited amount of the air/steam release can be considered substantially acceptable in respect to ICPCS requirements;
- among the c1) and c2) cases, both having the advantage of a completely closed system, c2) supplies a lower maximum pressure, but the actual free convection behaviour in the pool must be carefully analyzed and the effect of pool partitions is more penalizing.

A final choice between the different candidate solutions has not been made at the moment; however, solution d) can be considered as the most attractive.

8. CONCLUSIONS AND FUTURE WORK

The study till now performed has shown that the ICPCS system is technically feasible with reasonable size and cost and is able to supply acceptable performances. The most suitable solution seems to be a reflux condenser cooled by air in natural circulation.

Next activity steps will concern:
- identification of critical areas for system design;
- planning and execution of the required research activities (numerical calculations, experimental tests);
- detailed system design.

Some possible subjects of the research activities are finned tube heat transfer with natural circulation air at very low velocity and, in case the adopted solution is d), reflux condensation phenomena with non-condensible gases, vent/vacuum breaker valve (or water seal) behaviour and reliability.

REFERENCE

(1) Safety Related Terms of Advanced Nuclear Plants.International Atomic Energy Agency, Final Report, February 1991.

SURVEY OF ACTIVITIES FOR ASSESSING THERMALHYDRAULIC ASPECTS OF NEW REACTORS

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Abstract

The present paper deals with an overview of some activities carried out in Italy in the field of safety related thermalhydraulics studies after the decision of the Government to shut down the nuclear power plants.

The described activities have been performed in the frame of cooperations supported by ENEA and involve the Pisa University.

Results obtained from the operation of PIPER-ONE (gravity driven reflood) and from Accident Management studies performed with reference to a loss of feedwater and a small break LOCA experiment, are briefly described.

All the above studies are directly applicable to the technology of the new generation reactors.

1. INTRODUCTION

In Italy the production of electric energy from nuclear power plants has been suspended after the Chernobyl accident; nevertheless the research in the field from various organizations including university is active and concerns applied technology aspects as well as studies of basic phenomenologies having as objective the safety of the new generation reactors.

In the area of thermalhydraulics, activities on the codes assessment, like the International Standard Problems (ISP) or the evaluation of the experimental data base obtained in the facilities available in various countries, have been maintained in this period; these have a role in the knowledge of the scenarios foreseen in accident situations in the nuclear reactors.

The failure of the emergency systems, the operator actions, the exploitation of the whole spectrum of available systems in the plant, are among the issues of current interest.

The main purpose of the present work is to give an outline of few relevant activities carried out in the last years in this frame. Emphasis is given to researches performed by ENEA in cooperation with Pisa University, dealing both with experimental studies and codes assessment. Some of the researches are directly related to the study of aspects relevant to the new generation reactors, e.g. the experimental investigations of the behaviour of gravity driven core cooling system with the PIPER-ONE facility /1/. Additional activities in this context relate to the evaluation of transient scenarios in AP-600 and SBWR plants and are documented in another paper in this conference /2/.

Finally, a synthetic view is given of recent work performed at Pisa University in cooperation with ENEA related to the analysis of Accident Management situations in the present LWR: some aspects can be extrapolated to the new generation reactors. In particular Accident Management have been investigated in SPES and BETHSY facilities /3,4/.

2. AVAILABLE FACILITIES AND CODES ASSESSMENT ACTIVITY

Several activities have been in progress in Italy in '70 to evaluate the safety of nuclear power plants. In particular two facilities were designed and built at the beginning of '80, SPES and PIPER-ONE simulating PWR and BWR respectively.

Activities in code assessment concerned the utilization of RELAP5 code series (ENEA, ANSALDO, CISE, Universities) and of CATHARE code available at Pisa University since 1986.

In this chapter an outline of experimental facilities and codes is given.

2.1 Spes facility

The SPES facility /5/, is designed to simulate the whole primary circuit, the relevant parts of the secondary circuit (steam generator secondary sides, feedwater lines downstream the isolation valves, main steam lines upstream the turbine valves), and the most significant auxiliary and emergency systems (charging and letdown system, safety injection systems, Emergency FeedWater, steam dump, etc.) of the PWR plant.

The basic features of the facility are:
- volume scaling ratio equal to 1:427
- three active loops to simulate a three loop reactor;
- design pressure 20 MPa, design temperature 638 K, allowing tests with primary pressure above the reactor design value;
- core simulator consisting of 97 electrically heated rods, with uniform flux (local hot spots are simulated by means of three rods, with a peaking factor of 1.19);
- maximum channel power corresponding to about 140% of the reactor nominal power (roughly 6.5 MW) allowing the simulation of reactor power excursions;
- the height of all the components is the same as in a real plant, except for the pressurizer which is shorter, in order to preserve the volume scaling ratio and to maintain, at the same time, an acceptable flow area.

A sketch of the facility is reported in Fig. 1.

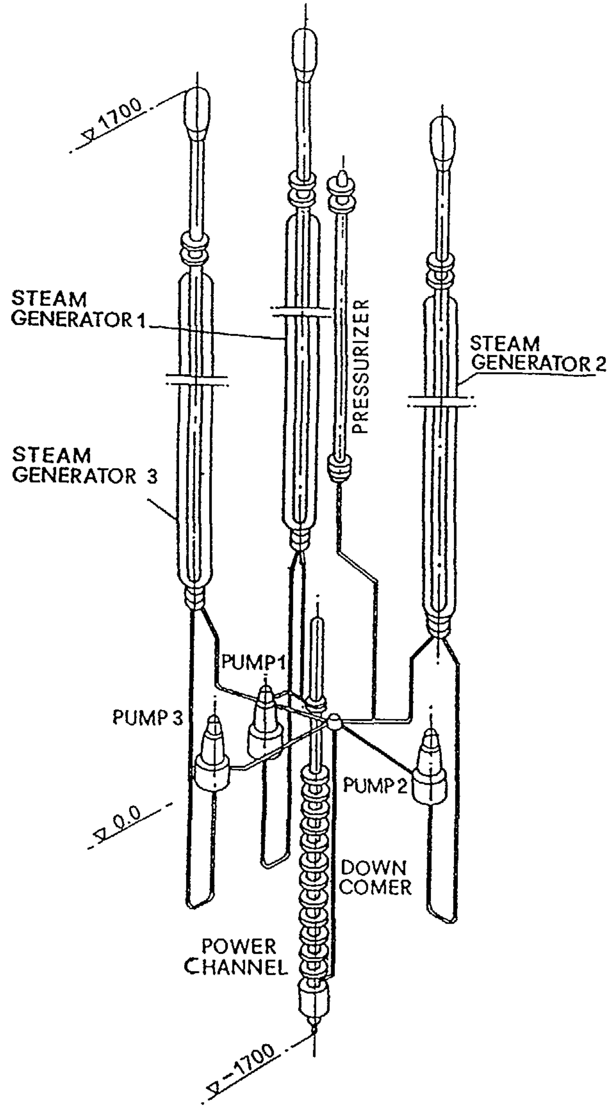

Fig. 1 - Sketch of SPES integral test facility (PWR simulator).

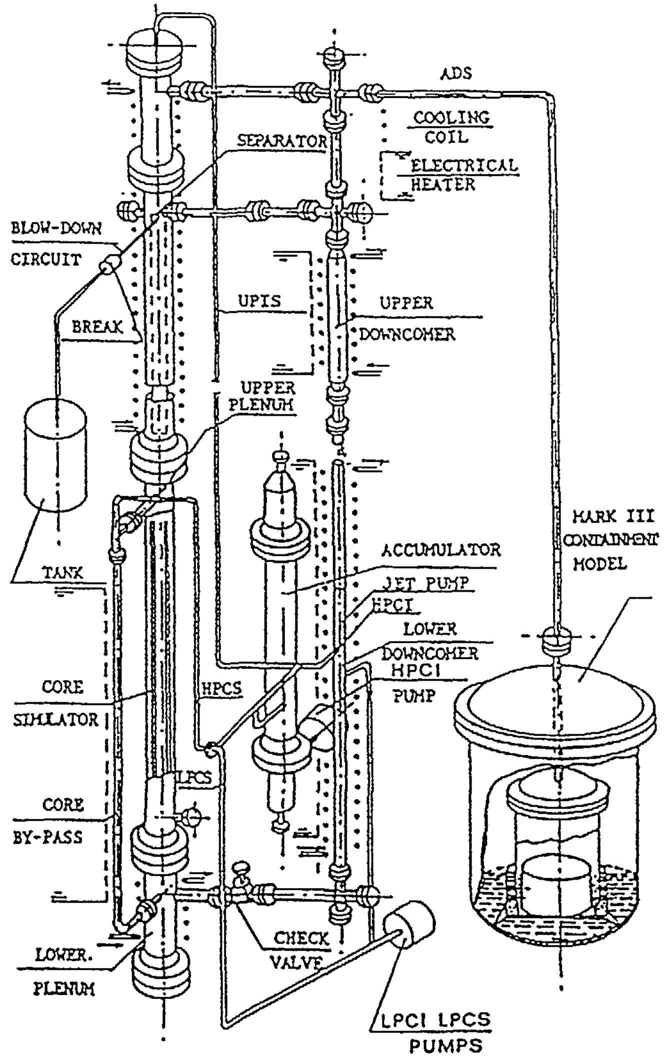

Fig. 2 - Sketch of PIPER-ONE integral test facility (BWR simulator).

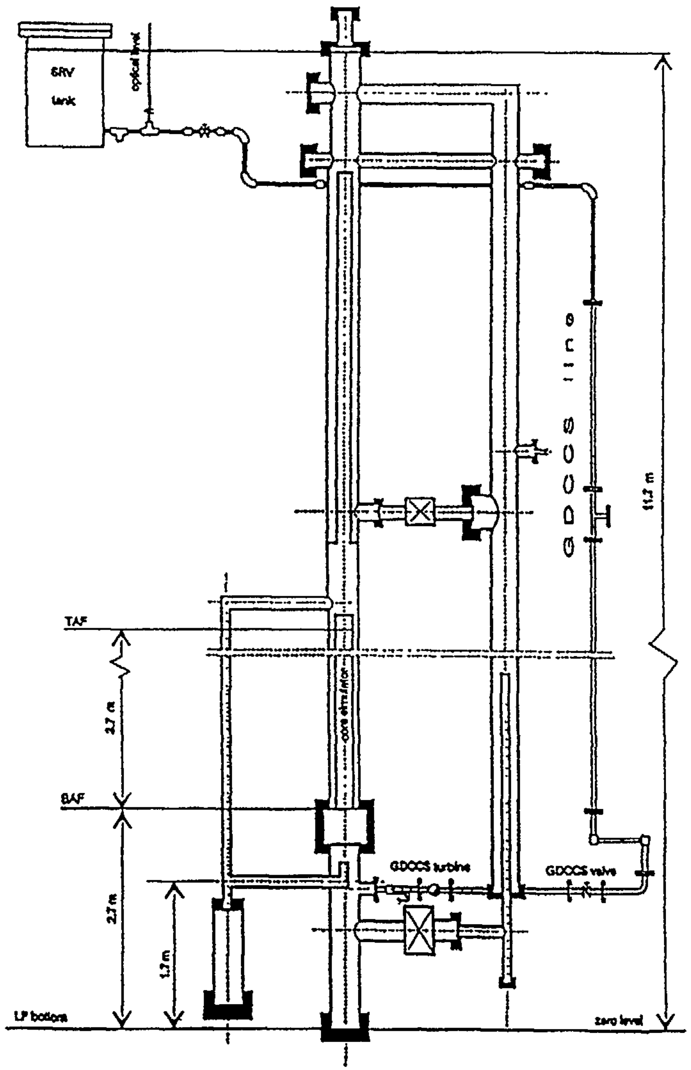

Fig. 3 - Sketch of the GDCCS system installed in PIPER-ONE facility.

2.2 PIPER-ONE facility

The simplified sketch of the PIPER-ONE apparatus is shown in Fig. 2 /6/. The main loop simulates the BWR vessel and can be subdivided into nine zones: lower plenum, core, core bypass (external to the core), control rod guide tubes, upper plenum, separators and dryers, steam dome, upper downcomer and jet pump. The ECCS simulators (ADS, LPCI/LPCS, HPCI/HPCS), the Steam Relief Valve (SRV) and steam lines, as well as the blow-down line, complete the apparatus.

The facility comprises also structure heating and cooling systems; the former is used to compensate the heat losses to the environment; the latter has the aim of mitigating the impact of heat exchange between walls and coolant that cannot be correctly scaled down in such a type of apparatus.

The heated bundle consists of 16 (4x4) indirectly heated electrical rods, whose height, pitch and diameter are the same as the fuel elements in the reference plant. The maximum available power is 250 kW, sufficient to simulate the power decay of the nuclear reactor.

The volume scaling factor is about 1/2200, while the piezometric heads acting on the lower core support plate are the same in the model and in the reference plant.

The facility hardware was slightly modified for studying the behaviour of a Gravity Driven Core Cooling System (GDCCS). The SRV discharge tank, 80.1 dm^3 capacity, able to withstand an internal pressure of 0.2 MPa greater than the atmospheric one, was connected to the lower plenum of the primary circuit (Fig. 3). The SRV tank is located at the 4th floor of the main frame, roughly 10 m above the lower plenum inlet nozzle. In order to avoid distortions in the transient due to the low tank capacity, an automatic system was added, which mantains constant (± 2 cm) the liquid level in this tank.

2.3 Computer codes features

Several versions of Relap5 and Cathare codes have been in use; nevertheless, the discussion is restricted hereafter to Relap5/mod2 /7/ and Cathare 1 v. 1.3 /8/.

Relap5/mod2

Relap5/mod2 is a transient analysis code for complex thermal-hydraulic systems. The fluid and energy flow paths are approximated by one-dimensional stream tube and conduction models. The code contains system components peculiar to pressurized water reactors. In particular point neutronics, pumps, turbines, valves, separators, and control systems are simulated. The code also contains a jet pump component that has been used for modeling boiling water reactors.

The code is based on a non-homogeneous non-equilibrium set of 6 partial derivative balance equations for the steam and liquid phase. A non-condensable component in the steam phase and a non-volatile component (boron) in the liquid phase can be treated by the code.

A fast, partially implicit numerical scheme is used to solve the equations inside control volumes connected by junctions.

Tab. I - Overview of code assessment activities carried out at DCMN of Pisa University by CATHARE and RELAP5 codes (Status Aug. 1991).

	PROBLEM TITLE	REFERENCE REACTOR	BASIC MODELLING STUDIES	CODE ASSESSMENT AGAINST FUNDAMENTAL PROBLEMS	CODE ASSESSMENT AGAINST SEPARATE EFFECT EXPERIMENTS	CODE ASSESSMENT AGAINST INTEGRAL EXPERIMENTS	CODE ASSESSMENT AGAINST PLANT DATA	CODE APPLICATION TO SAFETY AND DESIGN
	Interfacial drag	-	XO					
	Heat transfer at steam liquid	-	XO					
	Interface	-	XO					
	Wall to fluid friction	-	XO					
	Flow regimes	-	XO					
	T - branch	-	O					
	Reflood heat transfer	-	O					
	Countercurrent flow limitation	-	O					
	Critical nozzle flow	-		O				
	Condensation at interface	-		O				
	Pressurizer noding	PWR		O				
	Loop seal noding	PWR		O				
	Cold leg - downcomer connection	PWR		O				
	Noding			X				
	Parallel channels noding	BWR		X				
	U-tube noding	PWR		X				
	CCFL at UTP without bundle	both			XO			
	CCFL at UTP with 4x4 bundle	both			XO			
	CCFL at UTP with 16x16 bundle	both			XO			
	Blowdown - Piper vessel	both			XO +			
	Pressure drop in critical flow	both			XO			
(1)	Reflood Achille facility	PWR			XO +			
	Containment - Lace facility	-			X			
	Nat. circ. in glass facility	BWR			X			
	Gravity driven reflood in PIPER-ONE (PO-SD-7)	both			X			
	Instability in PIPER-ONE (PO-SD-5)	BWR			X			
(2)	SBLOCA - LOBI A2-81	PWR				XO *		
	SBLOCA - LOBI A1-83	PWR				XO		
	SBLOCA - LOBI A1-84	PWR				O *		
	SGTR - LOBI BL-21	PWR				X		
	Nat. circ. LOBI A2-77A	PWR				X		
	Nat. circ. LOBI A1-92	PWR				X		
	ATWS - LOBI A2-90	PWR				X		
	LOFW - LOBI BT-00	PWR				X		
	SLB - LOBI BT-01	PWR				X		
	ATWS - LOBI BT-03	PWR				X		
(3)	LOFW - SPES SP-FW-02	PWR				X		
(4)	SBLOCA - ROSA IV - SB-CL-18	PWR				X		
	Nat. circ. ROSA IV- ST-NC-02	PWR				X		
	Nat. circ. ROSA IV - ST-NC-09	PWR				X		
	IBLOCA - LOFT - LB-FP-2	PWR				X		
(5)	SBLOCA - PMK	WWER				X		
(6)	SGTR - PMK	WWER				X		
*(7)	SBLOCA - PIPER-ONE PO-SB-7	BWR				X		
	SBLOCA - PIPER-ONE PO-SB-7c	BWR				X		
*	Nat. circ.- PIPER-ONE PO-SD-4/I	BWR				X		
*	Nat. circ. PIPER-ONE PO-SD-4/II	BWR				X		
(9)	SBLOCA - BETHSY 9.1.b.	PWR				X		
	SBLOCA - SPES SP-SB-03	PWR				X		
	LBLOCA - JP failure PIPER-ONE PO-LB-1	BWR				X		
	Nat. circ. SPES SP-NC-04	PWR				X		
	SBLOCA - TMI	PWR					X	
(10)	SGTR - DOEL	PWR					X	
	LOFW - LEIBSTADT	BWR					X	
	SBLOCA - Montalto	BWR						X
	ATWS/MSIV - Montalto	BWR						X
	ATWS/SORV - Montalto	BWR						X
	ATWS/MSIV - Caorso	BWR						X
	ATWS/SORV - Caorso	BWR						X
(8)	SBLOCA - TRINO	PWR						X
	TWT - SBWR	BWR						X
	SLB - SBWR	BWR						X
*	LBLOCA - Montalto	BWR						X
	LBLOCA - AP600	PWR						X
	IBLOCA - AP600	PWR						X +
	LBLOCA - JP failure	BWR						X
	Nat. circ. Doel	PWR						X
	AM Studies related to SPES	PWR						X

X = RELAP5/MOD2
O = CATHARE 1 V1.3
+ = RELAP5/MOD3
* = CATHARE 2

* counterpart test
1) CSNI ISP 25 2) CSNI ISP 18
3) CSNI ISP 22 4) CSNI ISP 26
5) IAEA SPE 2 6) IAEA SPE 3
7) CSNI ISP 21 8) only modelization qualification at DCMN
9) CSNI ISP 27 10) CSNI ISP 20

Cathare

Essentially Cathare code has the same main peculiarities as Relap5. The difference, from a general point of view, outcomes from the consideration that the code is mostly devoted to the analysis of transients in Pressurized Water Reactors with U-tubes steam generators.

In particular the structure of the code is modular, each module describing a particular plant feature: pipes, large capacities, connections among pipings, as tee junctions and branches, boundary conditions like dead ends, assigned flow rates, etc. The basic hydrodynamic module, adopted for pipes and other 1D parts of the plant, is based on six balance equations expressing the conservation of mass, momentum and energy for the two phases. Space discretization of these equations is made using a finite difference scheme, based on the definition of "scalar nodes" where mass and energy equations are solved and "vector nodes" where momentum equations are solved, making extensive use of the "donor cell principle" and of the "staggered mesh" technique.

2.4 Assessment methodology

The ultimate goal of codes use, in any case, consists in the application to nuclear plant systems in order to evaluate the safety margins, to optimize the plant design and the operating procedures, and, possibly, for a better training of the operators.

System codes such as those previously presented, contain a number of approximations when applied to the prediction of the behaviour of a real continuous system. These approximations are necessary due to the finite storage capabilities of computers, the need to obtain calculated results in a reasonable time, and, in many cases because of limited knowledge about the physical behaviour of components and processes that are modelled. Very little objective information is available regarding the source of errors introduced by the mentioned approximations; almost all of this outcomes from the comparison between measured and calculated transients.

The actual framework of code assessment results from the above considerations. The code assessment is a quite complex process and can be distinguished into the following main steps /9/:

a) in depth analysis of code manuals considering the basic models and the description of input data deck;
b) user qualification through study and applications of nodalizations set up by experienced users;
c) application of the code to fundamental physical and numerical problems;
d) application of the code to separate effect test facility data including extensive comparison with experimental data;
e) application of the code to transients measured in integral test facilities possibly of the same type of the plant to be analyzed;
f) code use for plant design and safety.

An overview of the activity carried out at University od Pisa results from Tab. I, with reference to both Cathare and Relap codes; the various steps of the code validation process, can be recognized in the first column of the table.

3. GRAVITY DRIVEN REFLOOD TESTS

The tests PO-SD-6 A, B and C performed in PIPER-ONE were designed by taking into account the peculiarities of the facility with the objective of simulating phenomena that are typical of gravity driven reflood.

The GDCCS plays a crucial role for the safety of new LWR concepts, particularly of the General Eletric SBWR (Simplified BWR). For this reason, the first step in the research was to identify the limitations of the available facility in simulating SBWR related scenarios. The active height of the core (2.2 m and 3.7 m in SBWR and in PIPER-ONE, respectively) and the distance between the bottom of the Active Fuel (BAF) and the initial liquid level in the GDCCS pool (22 m in the reactor and 9 m in the facility) were identified as the major potential sources of distortions. The heat exchange across the fuel rod box constitutes an additional difference between the model and the prototype. Thus, the direct extrapolation of the phenomena measured in the PIPER-ONE facility to the plant situation is not possible. Therefore, the objectives of the experiments were limited to:

- the experimental study of the gravity driven reflood phenomenology;
- the achievement of a data base suitable for validating system codes, also utilized in the licensing process of the SBWR; low pressure conditions are challenging from the code assessment point of view.

The core power was fixed considering the needs to be representative of the decay power and to compensate for heat losses. The GDCCS injection was located in the lower plenum to avoid the influence of the jet pumps simulator in the PIPER-ONE facility.

Code analyses were performed by Relap5/Mod2, utilizing the available nodalization, properly updated to simulate the GDCCS. The calculated data were used to determine a suitable diameter for the GDCCS discharge line consistent with the core power.

Details about initial and boundary conditions are given in Tables II and III.

Tab. II - Initial conditions of PO-SD-6 experiments.

QUANTITY	UNIT	TEST PO-SD-6A	TEST PO-SD-6B	TEST PO-SD-6C
LP absolute pressure	MPa	0.23	0.17	0.18
Core level [a]	m	-0.4	-0.7	-0.2
DC level [b]	m	2.2	1.9	2.4
Liquid temperature in LP	K	370.	360.	360
Liquid temperature in LDC	K	410.	400.	390.
Max cladding temp. (level G)[c]	K	460.	460.	420.
GDCCS pool temperature	K	285.	285.	285.
GDCCS pool level[a]	m	9.	9.	9.

a- Starting from BAF (2.7 m above LP bottom)
b- Starting from downcomer bottom (1.1 m above LP bottom)
c- The levels are shown in Fig. 2.

Tab. III - Boundary conditions and sequence of events during PO-SD-6 experiments.

QUANTITY	UNIT	TEST PO-SD-6A	TEST PO-SD-6B	TEST PO-SD-6C
Core power	KW	50.	50.	50.
Opening of the GDCCS valve	s	60.	40.	65.
Initial pressure in GDCCS pool	MPa	0.1	0.27	0.18
GDCCS reverse flow	s	70.-130.	NO	NO
Test end signal (PCT or H)[d]	K or m	900 K	6.5 m[e]	6.5 m[e]
Test end	s	300.	140.	220.

d- The test ends when PCT > 900 K (no core quench) or when the liquid level in the core (H) reaches TAF.
e- Starting from LP bottom.

Heating power, rod surface temperatures, pressure and levels in both the downcomer and the core regions are represented in Figs. 4 to 9.

The data emphasize the relevance of the pressure boundary conditions on the scenario.

Test PO-SD-6A

Water coming from the GDCCS rewetted the core bottom but the steam, produced mostly at the quench location, caused a local pressure increase which prevented quench front progression. The pressure drops at the connections between the primary circuit and the containment play a crucial role in this respect. Reverse flow occurred in the GDCCS line in the period 70-130 s (Fig. 4). Following a period of electrical power cycling, power was finally switched-off roughly at ~300 s (Fig. 5).

Test PO-SD-6B

The addition of an equivalent gravity head, through the control of GDCCS pool pressure (0.1 MPa above the primary loop ressure), allowed a prompt reflood of the whole core. This was completed in about 70 s after GDCCS activation (Fig. 7). It should be noted that the steam produced by the core quench caused an increase of about 0.1 MPa in the lower plenum pressure.

Test PO-SD-6C

A configuration that, under some extent, is more close to the plant (i.e. the primary system discharges the steam into the pool) allowed the rewet of the rods in a few hundreds of seconds (Fig. 9).

The experimental research was carried out with the aim of characterizing phenomena occurring during gravity driven reflood situations, which are expected to be typical of the new nuclear reactor concepts. Several limitations of the facility (mostly the active height of the core rods) prevent a direct extrapolaion of the data to the reactor situation. As a relevant finding, the data obtained in three different tests confirmed the huge impact of boundary conditions related to the GDCCS pool and primary circuit pressure on the involved phenomenology. Changes in these conditions, that are negligible if pumps are available, cause the occurrence or not of the core reflood. The resulting data base appears useful for the assessment at low pressure of the models implemented in the codes. So far, the Relap5/Mod2 code has only been used in the design of the experiments (pre-test calculations).

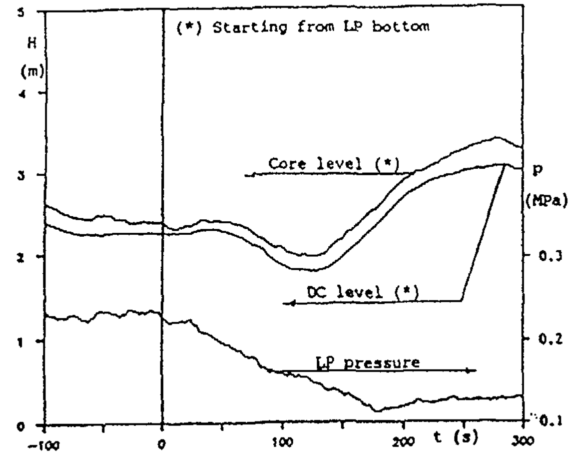

Fig. 4 - PIPER-ONE GDCCS experiments: Trends of LP and core region in test PO-SD-6A.

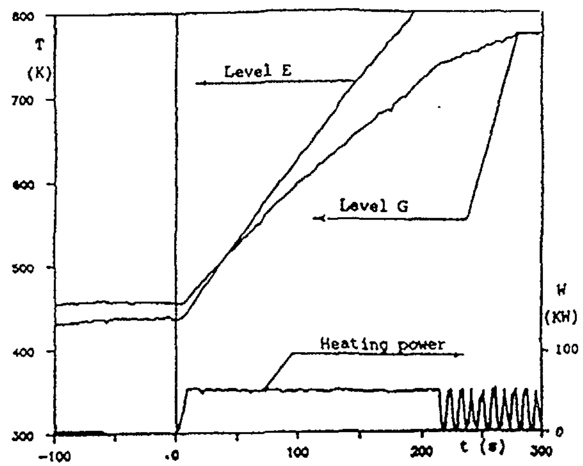

Fig. 5 - PIPER-ONE GDCCS experiments: Trends of heating power and rod surface temperature in test PO-SD-6A.

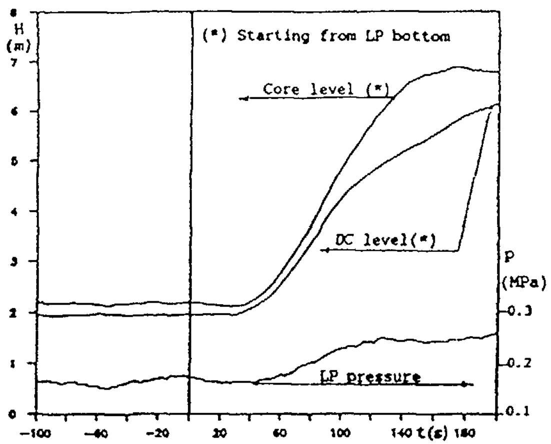

Fig. 6 - PIPER-ONE GDCCS experiments: Trends of LP pressure and liquid levels in DC and core region in test PO-SD-6B.

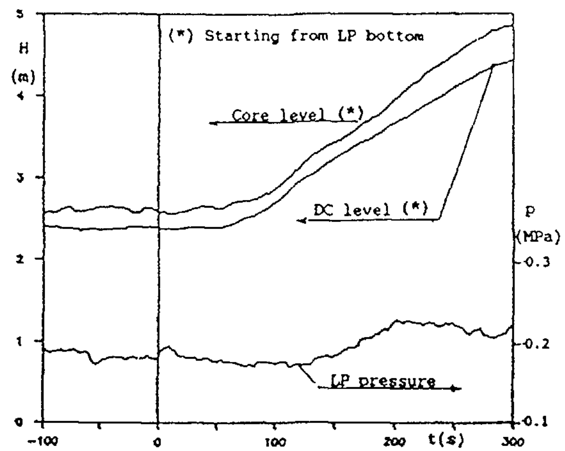

Fig. 8 - PIPER-ONE GDCCS experiments: Trends of LP pressure and liquid levels in DC and core region in test PO-SD-6C.

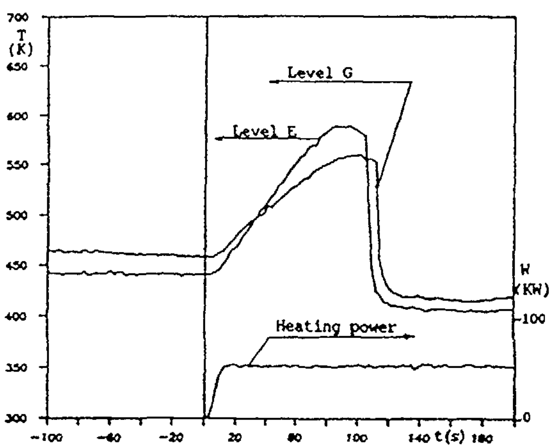

Fig. 7 - PIPER-ONE GDCCS experiments: Trends of heating power and rod surface temperature in test PO-SD-6B.

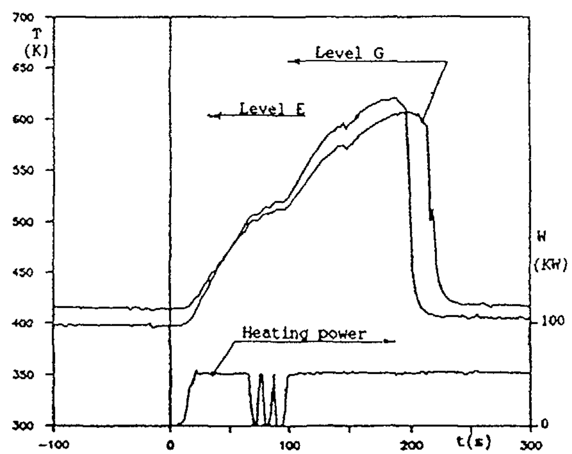

Fig. 9 - PIPER-ONE GDCCS experiments: Trends of heating power and rod surface temperature in test PO-SD-6C.

Tab. IV - SPES AM study: overview of the performed calculations and of the reference experiment.

TEST IDENTIFICATION	EFW ACTUATION	PRESSURIZER PORV	PUMPS	ACCUMULATORS & LPIS	DNB TIME (s) I/II	PCT (K) I/II	DNB DURATION (s) I/II	END OF CALCULATION (s)	PRIMARY SIDE PRESSURE/ RESIDUAL MASS (MPa)/(% of nom.value)
Experiment/CALC-00 (SP-FW-02)(PRE-TEST)	LOOP 1	CYCLING	TRIP OFF AT LOW SUBCOOLING	NOT ACTIVE	6600/N.O. EXP 4565/N.O. CALC	750 EXP 705 CALC	145/N.O. EXP 95/N.O. CALC	8000	9.2 /N.A. EXP 8.1 / 43% CALC
CALC - 01	LOOP 2	CYCLING	TRIP OFF AT LOW SUBCOOLING	NOT ACTIVE	4565/N.O.	715	95	8000	8.1 / 43%
CALC - 02	LOOP 2 & 3	CYCLING	TRIP OFF AT LOW SUBCOOLING	NOT ACTIVE	4565/N.O.	690	95	8000	6.0 / 44%
CALC - 03	LOOP 1,2 & 3	CYCLING	TRIP OFF AT LOW SUBCOOLING	NOT ACTIVE	4565/N.O.	685	60	8000	3.0 / 42%
CALC - 04	LOOP 2	CYCLING + STUCK OPEN	TRIP OFF AT LOW SUBCOOLING	NOT ACTIVE	4565/5100	715/>1473	95/*	5100	6.0 / 33%
CALC - 05	LOOP 2	CYCLING + STUCK OPEN	TRIP OFF AT LOW SUBCOOLING + RESTART	NOT ACTIVE	4565/6950	715/>1473	95/*	5600	3.0 / 11%
CALC - 06	LOOP 2	CYCLING + STUCK OPEN	TRIP OFF AT LOW SUBCOOLING + RESTART	ACTIVE	4565/5700	715/745	95/500	8200	1.9 / 48%

* The calculation end is due to too high rod temperature.

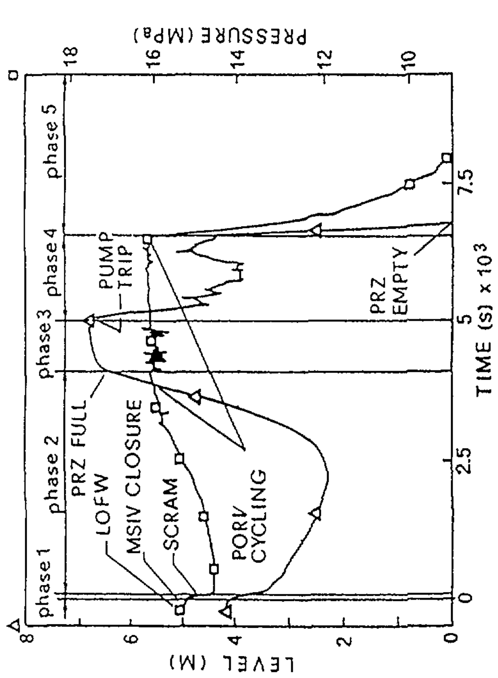

Fig. 10 - SPES experiment SP-FW-02: pressurizer pressure and level.

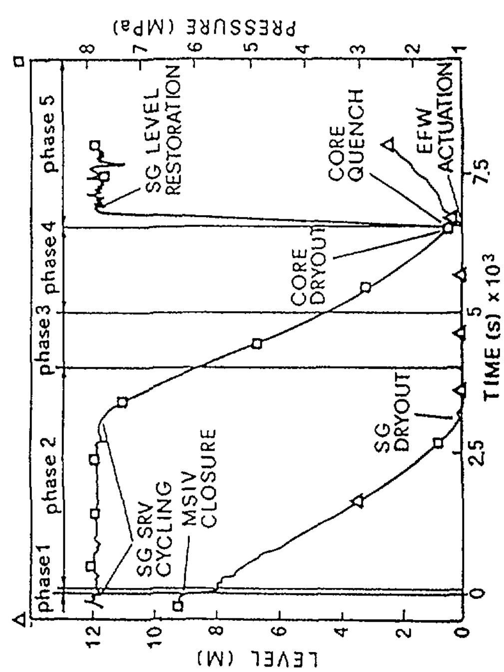

Fig. 11 - SPES experiment SP-FW-02: steam generators pressure and level.

4. ACCIDENT MANAGEMENT

Accident management (AM) quantifies all the actions which aim to anticipate or mitigate accidents in industry. In nuclear power plants the accident management, within the prevention limits, tries to maintain core coolability and containment integrity for Design Basis Accident events using safety related and operation systems inside the design limits. In accident with core melt, accident management attempts to mitigate the consequence of core distruption and the release of radioactive material to the environment, by making the best use of all existing systems, even outside the design limits.

The activities have been performed with reference to the analysis of transients performed in SPES and BETHSY facilities.

The described results demostrate the capabilities of AM procedures in mitigating accident severity.

4.1 Accident Management in case of Loss of FeedWater

The transient utilized at the basis activity, performed in the Spes facility, is a Loss of FeedWater in all the three steam generators with delayed activation of Emergency FeedWater. The transient evolves through 5 phases reported in Figs. 10 and 11.

The code demonstrated to be able to predict the relevant phenomena occurring during the transient as can be seen from the comparison between the calculation and the experimental data /10/.

The most important phenomena occuring during the experiment were:
- core dryout;
- pressurizer overfill and emptying;
- core quench;
- heat transfer across the steam generator at variable mass inventory in primary side.

Preliminary sensitivity calculations were performed in relation to the position and the number of Emergency FeedWater systems available for plant recovery. The summary of all the performed calculations is given in Tab. IV.

The results demonstrated that in all cases (calcs. 00 to 03) core quenching and consequent recovery of the plant occurred. Calc-01 led to the worst situation (Fig. 12) as far as core temperature was concerned. Considering the calculated scenario with Emergency FeedWater in loop 2 as the reference test, additional calculations were performed:
- PORV in "stuck open" position (TMI type accident) when the pressurizer pressure fell below the nominal closure set point (calc-04);
- as above, assuming pumps restart when rod surface temperature was greater than 750 K (calc-05);
- as above, assuming available accumulators and LPIS (calc-06).

Significant results of these calculations are shown in Figs. 13 to 17.

As a main conclusion, with respect to the reference LOFW experiment, it has been demonstrated that further failures such as PORV in "stuck open" position, involves a much more severe accident scenario, making impossible, the plant recovery /3/ (Figs. 13 to 15). The combination of operator intervention aiming at restarting the pumps, and the availability of safety systems at low pressure, permits a complete core cooling (Figs. 16 and 17).

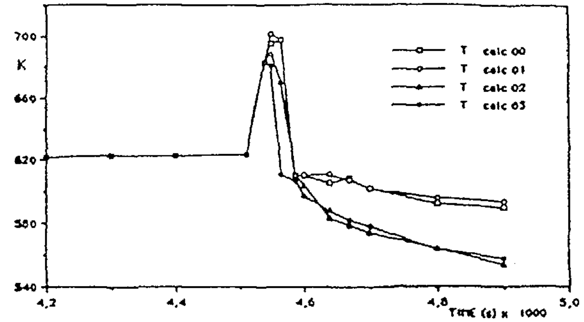

Fig. 12 - SPES AM study: rod temperature trends at the core top in sensitivity calculations with different EFW injection locations.

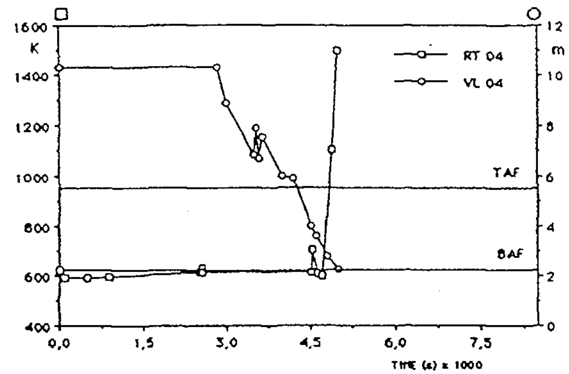

Fig. 13 - SPES AM study: trends of vessel level and rod temperature at the core top in the case of PORV in "stuck open" position.

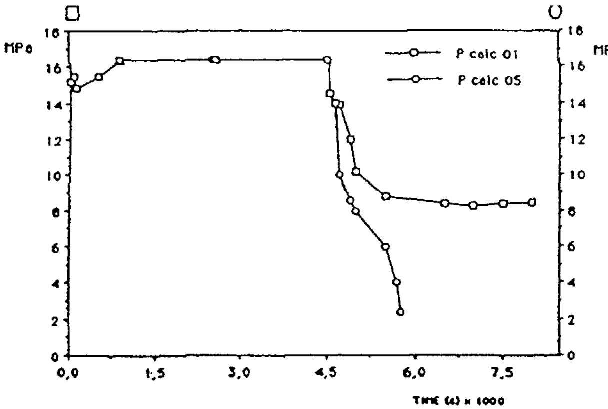

Fig. 14 - SPES AM study: trends of the primary pressure in the reference calculation and in the case with PORV in "Stuck open" position and and pumps restart.

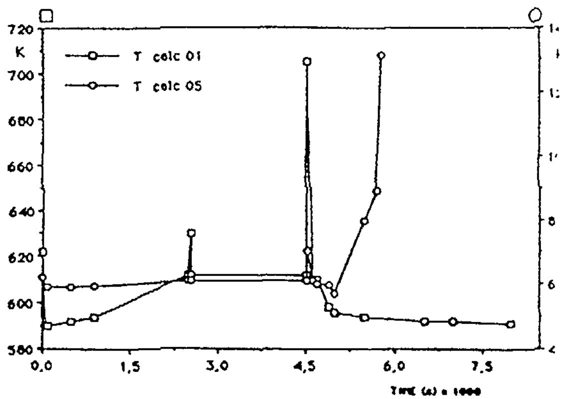

Fig. 15 - SPES AM study: trends of the rod temperature at the core top in the reference calculation and in the case with PORV in "Stuck open" position and pumps restart.

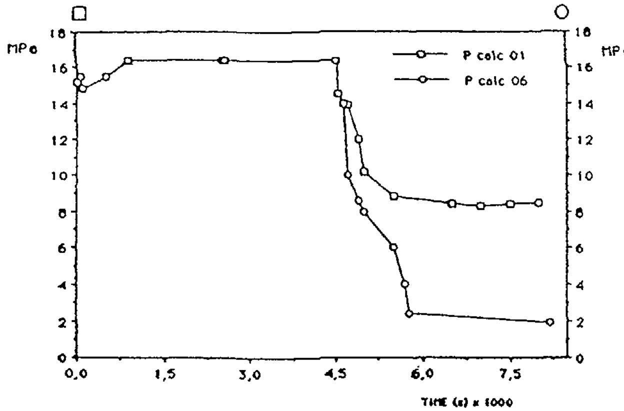

Fig. 16 - SPES AM study: trends of the primary pressure in the case reference calculation and in the case with PORV in "Stuck open" position, pumps restart availability of the low pressure emergency systems.

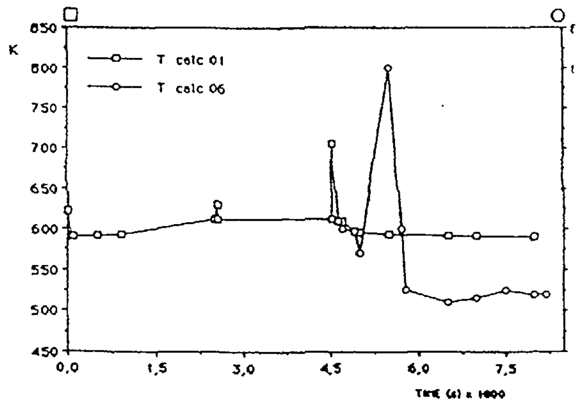

Fig. 17 - SPES AM study: trends of the primary pressure in the reference calculation and in the case with PORV in "Stuck open" position pumps restart and availability of the low pressure emergency systems.

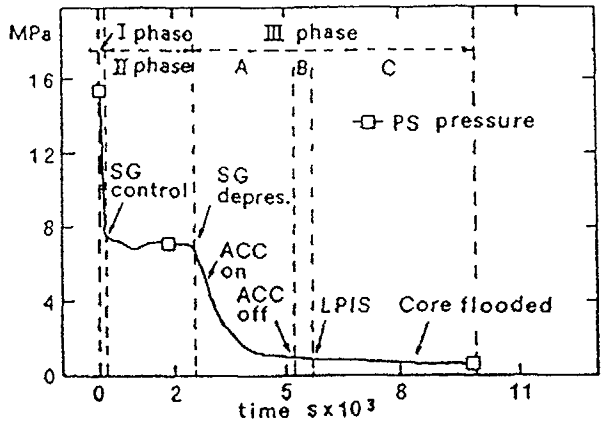

Fig. 18 - Bethsy experiment 9.1.b: primary side pressure.

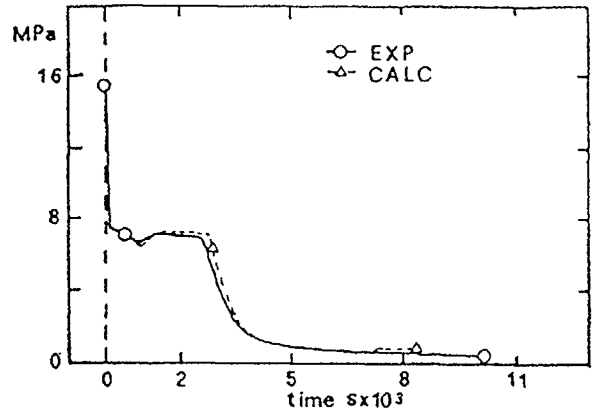

Fig. 19 - Bethsy experiment 9.1.b: comparison between measured and calculated trends of primary side pressure.

It should be noted that the same accident in the plant would evolve in less severe way than that resulting from this calculations.

4.2 Accident Management in case of small LOCA

The test, performed in the Bethsy facility is a SBLOCA with the break (roughly 0.4%) located in the cold leg of the loop with the pressurizer; HPIS is not available. Three different phases can be recognized during the transient (Fig. 18) /4/:

1. subcooled blowdown;
2. mass depletion in primary side;
3. ultimate procedure.

1. <u>subcooled blowdown</u>: following the break opening the primary pressure falls down and scram occurs when the pressure reaches 13.1 MPa. Safety injection signal (SI) occurs at 11.9 MPa. Following SI signal, turbine bypass occurs and main feed water is off.

2. <u>mass depletion</u>: the second phase is characterized by mass depletion and almost constant pressure and temperature in primary loop (saturation values). Oscillations in break flowrate in the first period of phase 2 testify of voiding of the cold leg of the broken loop. Secondary side conditions (mostly levels) remain constant in this period. At the end of this phase two dryout situations in core region occur; the first one causes the trip for the ultimate procedure;

3. <u>ultimate procedure</u>: due to the accumulators and LPIS actuations, three different parts can be distinguished during the last phase of the transient (A, B and C, respectively).

In the A period, starting with the opening of dump valves in the secondary side and ending with accumulators isolation, two rewet phenomena occur (the first rewet could be the consequence of liquid coming from the

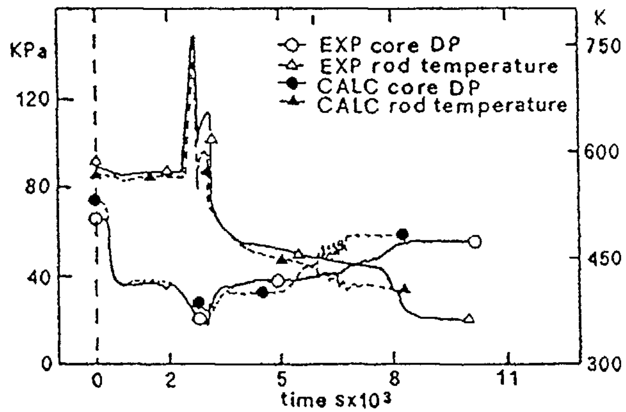

Fig. 20 - Betshy experiment 9.1.b: comparison between measured and calculated trends of core DP and rod temperature.

upper zones of the loop - mostly, SG U-tubes - and the second caused by the liquid injection from accumulators). Fast depressurization of secondary side also occurs during the A period

Period B occurs from the accumulators isolation up to the LPIS actuation. A continuous mass depletion of primary side without ECC injection characterize this phase.

Very early in period C, LPIS flowrate becomes larger than break flowrate leading to the recovery of the plant. The achievement of conditions for RHR operation mode, that fix the end of the simulated transient, largely depends upon LPIS flowrate.

In this case the ultimate procedure was sufficient for recovery of the plant

Relap5/Mod2 was utilized in pre-test analysis to predict the transient evolution /4/. The results were in very good agreement with the experimental data as shown in Figs. 19 and 20.

The ultimate procedure was planned with the U-tube of the primary circuit empty and the secondary circuit full. It appears interesting to verify the possibility to have still recovery of the plant in this transient when considering the trip for the ultimate procedure at low secondary mass inventory.

5 CONCLUSIONS

An overview has been presented of relevant activities concerning thermalhydraulic studies in LWR safety, performed in the frame of cooperations between ENEA and Pisa University.

Two main facilities, SPES and PIPER-ONE, have been maintained into operation in the last years at research centers of Pisa and Piacenza. The assessment of system codes like Relap5 (Mod2 and Mod3) and CATHARE (one and two) also continued in the same period.

Three aspects have been discussed in some detail in the paper:

a) advanced codes assessment (Relap5 and Cathare) has been performed at Pisa University considering a wide spectrum of situations;
b) the gravity driven reflood experiments performed in PIPER-ONE facility demonstrated the large dependence of reflood phenomenology upon boundary conditions like the connection modalities between primary system and containment (pressure drops between primary system and containment);
c) the accident management studies carried out with reference to two situation typical of PWR, can be directly applied to scenarios in new reactors

In definitive, the activities demonstrated that some presidium and competence in this field has been mantained in Italy.

ABBREVIATIONS

ACC	: Accumulator
ACCU	: Accumulator
ADS	: Automatic Depressurization System
AM	: Accident Management
AP-600	: Advanced PWR
ATWS	: Anticipated Transient Without Scram
BAF	: Bottom of Active Fuel
BWR	: Boiling Water Reactor
Calc/CALC	. Calculation
CCFL	Counter Current Flow Limitation
CMT	: Core Make-up Tank
CSNI	. Committee on the Safety of Nuclear Installations
C1	: Calculation 1
C2	: Calculation 2
DC	Down Comer
DCMN	: Dipartmento di Costruzioni Meccaniche e Nucleari
DNB	: Departure from Nucleate Boiling
DP	. Pressure Drop
ECCS	: Emergency Core Cooling System
EFW	: Emergency FeedWater
ENEA	: Italian Committe for Nuclear and Alternative Energies
EXP	: Experimental
GDCCS	: Gravity Driven Core Cooling System
H	. Core Height
HPCI	: High Pressure Coolant Injection
HPCS	: High Pressure Core Spray
HPIS	: High Pressure Injection System
IAEA	: International Atomic Energy Agency
IRWST	: In Containment Rad-Waste Storage Tank
ISP	: International Standard Problem
IBLOCA	: Intermediate Break LOCA
JP	· Jet Pump
LDC	: Lower Downcomer
LOCA	: Loss of Coolant Accident
LOFW	: Loss of FeedWater
LP	: Lower Plenum
LPCI	· Low Pressure Coolant Injection
LPCS	· Low Pressure Core Spray
LPIS	: Low Pressure Injection System
LWR	: Light Water Reactor
MSIV	: Main Steam Isolation Valves
PCT	: Peak Cladding Temperature
PORV	: Pilot Operated Relief Valve
PRHR	: Pressurizer Residual Heat Removed
PRZ	. Pressurizer
PS	: Primary System
PWR	: Pressurized Water Reactor
SBLOCA	· Smale Break LOCA
SBWR	Simplified BWR
SG	: Steam Generator
SGTR	: Steam Generator Tube Rupture
SI	: Safety Injection
SL	: Steam Line
SLB	. Steam Line Break
SORV	: Stuck Open Relief Valve
SRV	: Steam Relief Valve
TAF	: Top of Active Fuel
TM	: Three Mile Island
UPIS	: Upper Plenum Injection System
UTP	: Upper Tie Plate

Symbols

$\dot{m}_{AV}$ = average mass flow rate
P = pressure
T, RT = core rod temperature
VL = vessel level

REFERENCES

/1/ Bovalini R., D'Auria F., Mazzini M.
"Experiments of Code Coolability by a Gravity Driven System performed in PIPER-ONE Apparatus"
ANS Winter Meeting, San Francisco (Ca), Nov. 10-15 191.

/2/ Andreuccetti P., Barbucci P., Donatini F., D'Auria F., Galassi G.M., Oriolo F.
"Capabilities of Relap5 in simulating SBWR and AP-600 thermalhydraulic behaviour"
IAEA TCM on progress in Development and Design Aspects of Advanced Water Cooled Reactors - Rome (I), Sept. 9-12, 1991.

/3/ Billa C., D'Auria F., Galassi G.M.
"Accident Management in Case of Loss of FeedWater in PWRs"
1991 ANS Annual Meeting - Orlando (Fl), June 2-6 1991
Submitted at J. Nuclear Technology.

/4/ D'Auria F., Galassi G M., Billa C., Debrecin N.
"OECD/CSNI Isp. 27. pre-test prediction of the Betsy Small Break LOCA experiment 9.1.4 performed by Relap5/Mod2 code"
DCMN report, University of Pisa NT 168 (91) March 1991

/5/ Cattadori G., Rigamonti M.
"SPES: Description and Specification for OECD/CSNI International Standard Porblem N 22 (SP-FW-02 TEST), Vol. I: Spes System Description"
ENEA Term.-Risil, MFAG ITPMB 87073, November 1987.

/6/ Benigni E , Billa C , Bovalini R , D'Auria F , Di Marco P., Giannecchini S., Mazzini M , Piccinini L , Vigni P
"Description of PIPER-ONE Facility-Final Report" (In Italian) DCMN, Pisa University, RL 251(86) - Pisa (I), Dec 1986.

/7/ Ransom V H , Wagner R.J , Trapp J.A , Feinhauer L R., Johnson G.W., Kiser D M., Riemke R.A
"Relap5/Mod2 Code Manual Volume 1 Code Stucture, System Models and Solution Methods"
NUREG/CR-4312, Aug. 1985

/8/ Boulet M.
"User Manual of CATHARE 1 V.1.3 Code" - Dossier d'Exploitation D 4-s.
Ceng Technical Note SETH-LEML EM 87-86, Grenoble (F), 1987.

/9/ D'Auria F , Galassi G.M
"Code Assessment methodology and results" TAEA Technical Workshop/Committee on Computer Ailed a Septy Analyses - Moscow (USSR) May 14 17, 1990

/10/ Bonuccelli M., D'Auria F , Galassi G.M., Lombardi P
"art-test Analysis of Ses Test SP-FW-02 W.M. Relap5/Mod2 code"
DCMN Report, University of Pisa NT 175(91), July 1991

RESEARCH ACTIVITIES IN THE FIELD OF SEVERE ACCIDENT ANALYSIS

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Abstract

The paper presents issues on Source Term and Containment studies which have been approached by ENEA Dipartimento Reattori Innovativi in the field of severe accident analysis in advanced water-cooled reactor
In a deterministic approach to safety, which includes the evaluation of consequences of severe accidents, progresses are needed in reducing uncertainties in the evaluation methodologies, making reference to three critical areas description of phenomena occurring in severe accident sequences, thermo-mechanical behaviour of containment structure, source term release, transport and behaviour in the containment atmosphere
The analysis of the consequences of severe accidents, able to challenge the containment integrity, requires to model phenomenologies not completely known and not easely reproducible in experimental conditions
This has brought to a limited number of research program with a large international partecipation The ENEA partnership in these programs and the use of available codes and results in the safety evaluation of innovative reactors, are shortly described
Some application of code system CASTEM 2000-TRIO to the structural verification of the containment are described, with specific attention to material behaviour, singularity points in the containment structure, dynamic load effects
In the Source Term analysis accuracy of methodologies for the evaluation of transport, diffusion and deposition of aerosol particles should be increased, taking into account, in an implicit way, the variations of thermodynamic conditions in the containment atmosphere induced by the condensation on suspended aerosols and the presence of hygroscopic particles Some considerations related to this problem are briefly reported

Introduction

This document presents an overview of the R&D activities in the field of severe accidents that the Dipartimento Reattori Innovativi of ENEA is in charge of developing in the next three years, as contribution to the italian program on nuclear development oriented to the study of innovative reactors with enhanced inherent and passive safety characteristics

Facing the indications given by the referenda and the following government decisions, the approach to safety, necessary for a possible restart of the nuclear program in Italy, requires the definition of new safety objectives and criteria, more restrictive in terms of admissible risk with respect to those adopted in the past In the same way, for their achievement, technical solutions must be researched being simple, transparent for the public, preferably with inherent and passive characteristics

The major safety goal in this new approach is to limit the environmental impact and the off site radiological consequences in such a way that no specific evacuation plan shall be needed, nor any significant long term land contamination shall occur
Consequences of the accidents shall be ensured acceptable by a deterministic analysis for all conceivable events, even extremely rare, as severe accident with extended core degradation and large radioactive release

In this approach, which continues to be based on the "defense-in-depth" concept, the containment system has a key role His performance and integrity, and the adequacy of all auxiliary systems, must be demonstrated in all accidental circumstances, even though their probability of occurrence can be considered very low, in order to assure the achievement of the safety goal

On the development of the containment system and related safety evaluations, a relevant contribution is provided by different national operators (the Utility ENEL, the Industries and ENEA), either in the field of research and development, either in more specifically industrial areas, taking advantage of the existing qualified experience

The Dipartimento Reattori Innovativi of ENEA, who has a specific role for research, experimental support and industrial promotion, has identified different areas of research, shortly described in the following, considering these main objectives:

- to develop and validate new methodologies for a deterministic approach to safety, reducing uncertainties and conservativism in the evaluation procedures;

- to provide background experience for an improvement of the containment and other auxiliary systems, including new design solutions and making extensive use of inherent and passive concepts for the mitigation of the consequences of severe accidents.

A relevant part of these activities will be performed in the framework of national and international collaborations. In more details:

- a Research Contract has been signed with ANSALDO and FIAT (now consociated in the GENESI Consortium) and contacts have been established with italian Universities (e.g. Pisa and Roma) to take advantage of synergies existing at national level;

- a collaboration agreements have been defined by ENEA and ANSALDO with industrial companies proposing innovative reactor designs with enhanced inherent and passive safety characteristics (G.E. for the SBWR and PRISM; Westinghouse for the AP 600; ABB for the PIUS), in order to receive all information needed for the evaluation process requested by the italian government;

- a Research and Development Agreement on Future Reactors has been signed with the Commissariat à l'Energie Atomique (CEA France) and a collaboration on themes of specific interest for safety has been started; contacts have been established with BMFT and KfK in Germany for collaboration on containment studies related to severe accidents

The increasing consideration given to the containment system for the future reactor brought the Commission of European Communities to propose a "Reinforced Concerted Action" on this subject. Further discussion will certainly create new opportunities for improving collaboration at international level.

The research activities selected by the Dipartimento Reattori Innovativi, can be grouped in three main areas:

- the study of the phenomenologies occurring in severe accident conditions and the evaluation of the consequences in terms of mechanical and thermal energy, able to challenge the containment integrity;

- the development and validation of a methodology for the structural analysis and integrity evaluation of the containment system, with particular reference to dynamic load conditions;

- the study of the radiological source term and the development and validation of a methodology for the evaluation of the production, transport, diffusion and deposition of fission products in the plant and in the containment system.

Phenomenological studies for the evaluation of the consequences of severe accidents

The main objective of these studies is the possibility of developing a deterministic approach to the safety evaluation of the future reactors, which is considered a possible way to increase the public acceptability.

The activities of the ENEA Dipartimento Reattori Innovativi are addressed to the analysis of the consequences of severe accidents, which are able to challenge the containment integrity; this study requires to model phenomenologies not completely known and not easily reproducible in the experimental conditions.
The considerable engagement required for the research on severe accidents (experiments, code validation) has brought to limit the number of the research programs, enhancing a large international partecipation; examples are the SARP (NRC), ACE (EPRI), PHEBUS (CEA), FARO (CEC).
ENEA is in the partnership of these programs and has access to results and developed codes. Integral codes as STCP and MELCOR and other mechanicistic codes developed in the SARP NRC program, are extensively used for severe accident evaluation in innovative reactors.
The french code ICARE for the core degradation studies has been recently implemented at ENEA and will be developed and validated with PHEBUS P.F. experiments in the framework of the ENEA-CEA Collaboration Agreement.

Among the phenomenologies expected during a severe accident sequence, events as the high pressure melt ejection and the direct containment heating, the steam explosion, the hydrogen detonation have the potential of energy generation inside the containment able to challenge its integrity

The analytical effort will be devoted to reduce the uncertainties already existing in the computational methodologies through an extended validation program on the existing experimental data base and taking benefit of international code assessment programs A tight collaboration with University of Pisa is envisaged in this area

In the same time, the possibility of mitigating the consequences of severe accidents by using passive design solution will be evaluated On this possibility, a large international consensus exists Conceptual solutions are developed for reactors of the present generation and can be extended to innovative designs

This activity, conducted with the support of the industries, will refers to reactor cavity design, cages and screens, passive heat removal through the containment wall, hydrogen burners, pressure discharge systems and filters

A development of a post-accidental instrumentation system will be a part of this study, mainly referring to the needs of mitigation management procedures

Thermal and mechanical analysis of the containment structure

The requirement imposed to the containment system, to be able to withstand severe accident and the need of a deterministic verification in spite of any probabilistic consideration requires to improve structural analysis methodologies through the optimization of the computation techniques and a better description of the material behaviour under dynamic load conditions

The Code System CASTEM 2000-TRIO, whose development and validation for research and industrial application purposes is object of a specific agreement between ENEA and CEA, will be used for the structural analysis of the containment A validation program has been defined, which will be supported by reduced scale experimental tests, with the aim of reducing uncertainties in the modeling of composite material in the containment structure

A development of the PLEXUS code is also foreseen, to improve its capability to treat impulsive phenomena, as hydrogen detonation, steam explosion, internal and external impacts

In the same time, the Dipartimento reattori Innovativi of ENEA is supporting GENESI in design activities aimed to optimize those subsystems relevant for the containment tightness and integrity during an accident, as penetrations passive heat removal system, confinement of molten material in the reactor cavity

The possibility of conducting an experimental program on small scale or prototypic subsystems and component has been envisaged ENEA has the potentialities to perform this experiments in his Research Centres and in associations with GENESI and the University of Pisa as soon as the specifications of the verification program will be defined

The activities already mentioned identify some characteristics of the containment system which can be investigated without precise reference to the reactor type As a parallel action, the collaboration with Westinghouse and General Electric will give the opportunity of participating in the design verification on the reference containment design and the possibility of proposing improved solutions to be adopted in Italy and in other countries having similar requirements

Source Term evaluation studies

In the Source Term analysis, the activity of the Dipartimento Reattori Innovativi is aimed to increase the accuracy of the methodologies for the evaluation of transport, diffusion and deposition of radioactive products in the containment atmosphere Improvement of existing codes will be obtained through a validation program in the framework of existing international cooperations, where ENEA intends to maintain an active partecipation

The PHEBUS F P Program is the most relevant experimental program for the coming years in the area of Source Term Evaluation His major motivation is to give integral test results for code validation in a in-pile facility specially designed to obtain the best possible representativity of the source and the environmental conditions encountered by the fission products in their route during an accident ENEA considers these objectives important for the assessment of a deterministic approach, therefore, considering the opportunities of the agreement with CEA and the partnership with CEC, it is intended to propose a partecipation of ENEA personnel in the PHEBUS P F Program technical structure

Furthermore specific actions for the development and validation of mechanicistic codes for source term studies have been defined in the framework of the ENEA-CEA collaboration Independent calculations on reference test cases will be performed with FUMO, a containment thermodynamic code developed by University of Pisa, and IDRA, an aerosol behavior code developed by ENEA, in order to assess uncertainty areas and verify the needs of further developments

In the area of source term mitigation studies, the experimental program on the SPARTA facility, conducted by ENEA at Casaccia Centre and aimed to determine the decontamination factor exerted by water pools for aerosols generated during severe accidents, is scheduled to start in the middle of next year, with the support of the CEC Ispra The test matrix will be discussed with different organizations participating to the "Pool Scrubbing Group" A first scoping test was conducted in December '89, with satisfactory results

Consideration is also given to the need of reducing weakness and uncertainties already existing in the analytical modeling of the source term behaviour in the containment atmosphere A key aspect is to determine the distribution of the condensed mass of steam onto the suspended aerosol particles as well as onto the containment structure surfaces Condensation onto walls characterize a diffusiophoretic deposition (Stefan flow) while condensation onto particles influence their deposition by gravitational settling, thus acting both of them as a natural reduction process of the source term available for a potential release to the environment
Uncertainties in the evaluation of condensed mass onto particles are related to same specific non equilibrium phenomena (the Kelvin and solute effects) and to the presence of hygroscopic particles which may induce condensation even at subsaturated conditions This effect must be rightly considered also in connection with aerosol dynamics, in fact hygroscopic matters are produced during LWR severe accidents and may enter the containment atmosphere, producing a not negligible effect

Because the saturation conditions are strongly affected by the loss of steam and the release of latent heat during condensation, the condensation phenomena will influence so much the thermodynamics in the containment as the aerosol dynamics Sensitivity calculations have shown the importance of a fully coupled computation of the thermodynamic transient and the source term esimate

On these bases a new code is under development through a collaboration between the Dipartimento Reattori Innovativi of ENEA and the Dipartimento di Costruzioni Meccaniche e Nucleari of the University of Pisa The code structure will be organized in such a way to have all the necessary models, describing phenomena occurring at the same time, physically and functionally connected Due attention will be given to the possibility of maintaining a suitable computation time
Individual models will be also qualified, checking validity limits of correlations derived in specific experimental conditions and extrapolation difficulties due to particular geometries and physical conditions of the containmnent system during severe accidents

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PHYSICS AND THERMOHYDRAULICS

(Session 4)

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USE OF SLIGHTLY ENRICHED URANIUM IN A PHWR IN ARGENTINA

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Abstract

The nuclear power plant Atucha 1 (Atucha 1 NPP) uses natural uranium as fuel. Due to the manufacturing process, the fuel has a strong impact on the electricity cost.

Different possibilities are analyzed in order to improve the exit burn-up.

Assuming that the present fuel design and fuel operation limits remain unchanged as much as possible, the alternative of an enrichment of 0.85 w % in U235 seems to be the most adequate, since increase in the burnup brings economic benefits and optimization of uranium resources of about 30 % with respect to natural uranium.

I) INTRODUCTION

Argentina has two operating nuclear power plants. The first one, Atucha 1, located at about 100km from Buenos Aires, is of the pressure vessel type. The fuel is natural uranium and the coolant and moderator are D20.

It has a thermal power of 1179 Mwth. and started commercial operation in 1974.

It is refuelled on power and the spent fuel is kept in decay pools with capacity for the whole life of the plant.

The fuel element consists of a 36-pin cluster of UO2 pellets and an additional structural rod. The uranium mass is 153.5 kg and it has an active length of 5300 mm.

The present exit burnup is of about 6 Mwd/kgU which requires a refuelling rate of 1.3 fuel elements / full power days (f.e/fpd).

Due to the manufacturing process, the fuel has a strong impact on the electricity cost; thus, any increase in the exit burnup produces an important economic benefit.

In this work, we analize the possibility of using fuel elements with different degrees of enrichment in U235, in order to increase the fuel burnup.

This analysis was made assuming that the fuel element design and the present operating limits are maintained as well as the reactivity reserve necessary to perform power cycles.

At this stage it is supposed that no significant changes exist in the effectiveness of safety and control systems, so that the safety analysis for the natural uranium core is valid.

II) PRESENT FUEL MANAGEMENT FOR NATURAL URANIUM. OPERATING LIMITS

The reactor core is divided into three fuel burnup zones, (figure 1).

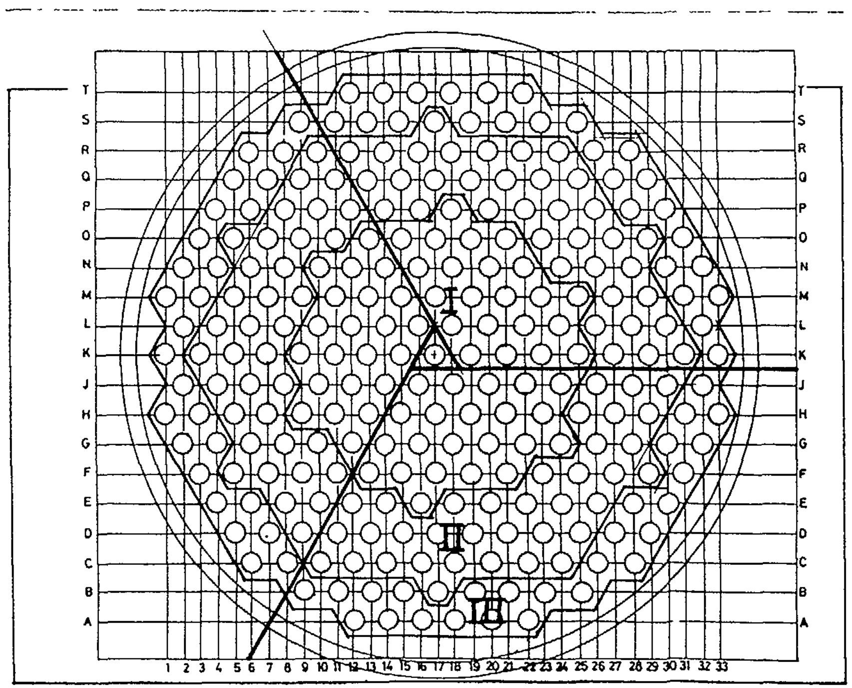

FIG. 1: ATUCHA 1 CORE; FUEL BURNUP ZONES

The fresh f.e. is entered in the intermediate zone, and once it has reached a burnup of 2.8 Mwd/kgU it is moved to the central zone. At 5.1 Mwd/kgU it is again moved to the external zone and from here to the decay pool with an exit burnup of 6 Mwd/kgU. The average residence time is of 200 full power days (fpd).

The core is further divided into eight thermohydraulic concentric zones, having decreasing coolant flows from the center to the periphery, thus imposing the maximum channel power in each zone.

The flows are restricted by nozzles at the coolant channel inlets. A change in the flow distribution would require structural modifications and a long shut down of the reactor probably not compensated by the economic benefits of the change of type of fuel.

In order to avoid fuel failure due to p.c. interaction the maximum linear power is limited. Moreover, limitations are also imposed to linear power steps produced by fuel shuffling.

As no modification of the f.e. design is considered, all these limits should be taken into account.

III) MODEL USED

A simplified model of the core was used.

The lattice calculations were performed with WIMSD4 (1) and the reactor calculations with PUMA (2) in its time-average option.

The core is divided into uniform burnup zones. The exit burnup is fixed keeping the same value for the reserve reactivity as in the natural uranium case.

For each zone the f.e. are represented by cross sections having averaged values between the input and exit burnups. An iteration process is performed between flux calculation and burnup, reaching convergence in a few steps.

The control rods are explicitly represented and their insertion is kept constant with an average value corresponding to full power.

Results were obtained for natural uranium, and enrichments of 0.85 %, 1% and 1.2 % in U235.

IV) LATTICE CALCULATIONS

Figure 2 shows k-effective as a function of burnup for the different enrichments

Figure 3 shows the relative power of the outer ring of fuel rods as a function of burnup For higher enrichments this variation is more important than for natural uranium

This parameter is important because it is related to the f e power limits

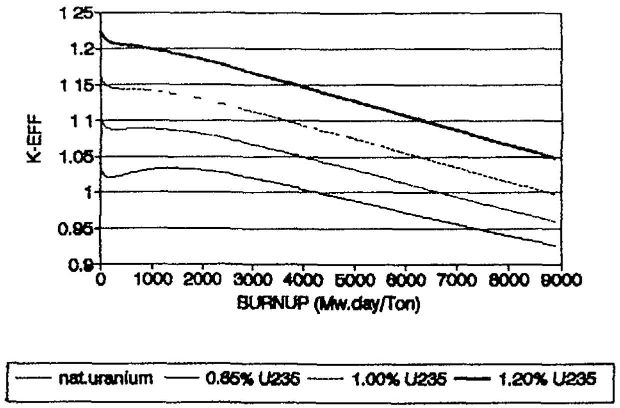

FIG 2 K-EFF vs BURNUP FOR DIFFERENT ENRICHMENTS

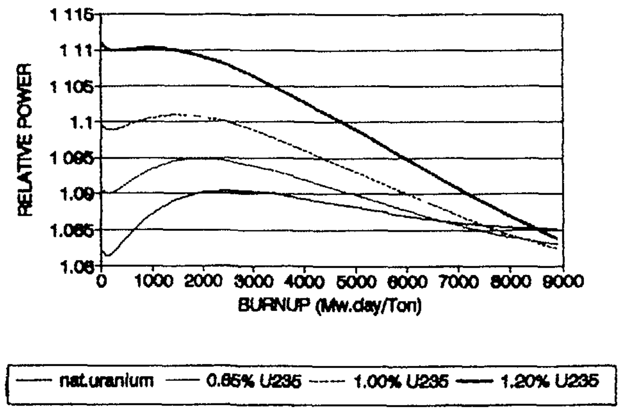

FIG 3 RELATIVE POWER IN THE OUTER RING FOR DIFFERENT ENRICHMENTS

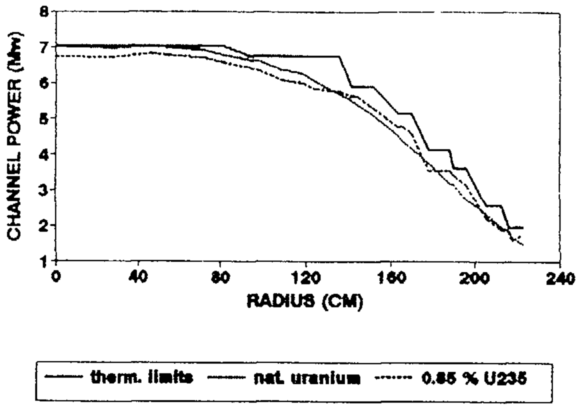

FIG. 4 : MAXIMUM CHANNEL POWER vs. RADIUS

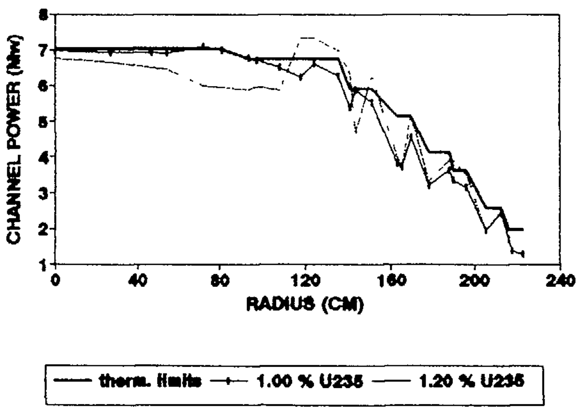

FIG. 5 : MAXIMUM CHANNEL POWER vs. RADIUS

V) COMPARISON AMONG DIFFERENT CYCLES

The following table summarizes the global results for the four cycles.

CYCLE	EXIT BURNUP	GLOBAL FORM FACTOR	REACTIVITY	ϵ (*) (%)
NATURAL URAN	6.2	1.91	-1.13	0
0.85%	11.4	1.74	-1.14	-3
1.0%	16.0	1.76	-1.12	1
1.2%	21.0	1.86	-1.15	9

(*) ϵ is the maximum difference between the channel power limit and the actual channel power.

Figures 4 and 5 show the maximum channel power as a function of the radius of the core, for natural uranium and the different enrichments, as compared to the allowed limits.

It may be noticed that the channel power limits are exceeded for the 1.2% cycle; this fact rules out this alternative although the exit burnup is triplicated.

The 1% cycle seems to be the most adequate proposal, although more detailed calculations are required to ensure that the limits are preserved. Changes in the f.e. design will be probably required since for high burnup the risks related to pellet cladding interaction may be increased.

The 0.85% cycle is clearly advantageous because it presents greater margins with respect to the channel power limits and duplicates the exit burnup.

The form factor is smaller for higher enrichments, that is to say, the power distribution is flatter.

VI) PROGRAM FOR THE CORE HOMOGENIZATION TO 0.85% IN U235

A test period and a slow, gradual introduction of 0.85% enriched f.e. is programmed in order to homogenize the initial natural uranium core, transforming it into a 0.85% enriched equilibrium core.

In the first step, 12 f.e. will be introduced as a demonstration, following a refuelling path equivalent to a natural uranium f.e. Their exit burnup will be 8.3 Mwd/kgU.

The following steps will be 24, 50 and the 100 enriched f e.(0.85%) to be introduced in a period of two years approximately, with a gradual increase of the exit burnup

As the refuelling schene in Atucha 1 allows the re-entrance of spent fuel elements from the pool, it is possible to consider a strategy of spiking of half-burnt natural uranium f.e. in a slightly enriched homogeneous core in order to optimize costs This possibility has not been studied yet

VII) ECONOMIC BENEFITS OF 0.85% ENRICHED F.E.

As the exit burnup is duplicated, the number of f.e. needed assuming an annual load factor of 85 %, is decreased from 396 to 205. This implies an annual saving of 30 % in fuel cost, and 31 % in uranium resources.

This evaluation was performed with international values for the important cost parameters.

It should be mentioned that the average exit burnup of a 0.85 % enriched f.e. must be 7.9 Mwd/kgU in order to obtain the same generation cost as for the natural uranium f.e.

The economic benefit is thus evident from the initial test stage.

As an additional argument, we should also mention that there is a reduction impact in the number of fresh f.e that are necessary in order to be able to operate the plant during 6 months with a load factor of 80 %. 184 elements are required, while only 95 f.e. are needed in the presently 0.85% case.

We should also take into account that within 30 years of operation all the vacant positions in the decay pool will be occupied, whereas if the enrichment program is initiated, vacant positions will still be found for 8 to 10 years more. thus, this will be not be a limiting factor for the eventual life extension of the plant.

The economic benefits due to the less frequent use of the transport system, as well as the less degraded inventory of D2O produced by the utilization of 0 85 % enriched f.e. could not be calculated

VIII) CONCLUSIONS

Under the assumptions of maintaining the same thermohydraulic design of the plant and producing the least number of changes in the f.e. manufacturing, the alternative of 0.85 % enriched fuel seems to be the most adequate for Atucha 1 NPP

This alternative duplicates the fuel exit burnup and implies an annual saving of approximately 30% in fuel cost and uranium resources

It also results in vacant positions in the decay pool which facilitates the eventual life extension of the plant

A program based in a gradual introduction is proposed in order to be able to perform a detailed following of the f e. until a completely homogenized enriched core is obtained.

A check of the results obtained with the simplified model will be required with a more detailed one, simulating the fuel management for a long period and studying the variations of the control and shut down effectiveness as a measure of the plant reliability.

REFERENCES

{1} 'A summary of WIMS-D4 input options" M. H. Halsall- AEEW M1327 (1980)

{2} "PUMA: Descripción y manual de uso'. C. Grant. IT 1043/90 (1990)

PERSPECTIVES ON INNOVATIVE FUEL AND ASSOCIATED CORE PHYSICS

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Abstract

Increasing the safety level of future water reactor may be reached by different ways

One way consists in keeping the present configuration of the core and increasing the efficiency or reliability of safety devices in order to ensure a better global safety In some cases the margin optimization is obtained by lowering the linear power of the fuel

A second way might be followed it tries to get a more forgiving reactor physics by optimization of different reactivity coefficients, avoiding some reactivity accident initiators, using a colder fuel In doing so, one increases the level of prevention rather than the protection itself

In order to illustrate this approach this paper emphasizes 2 aspects

* PWR without soluble boron (excepted for safety injection) this goal will be reached by using burnable poisons (heterogeneous and/or homogeneous) optimizing moderating ratios, cycle length, primary temperature versus load, others absorber rods In doing so, an important reactivity induced accident initiator would be eliminated

* Another way is related to 'cold fuels' The objectives are to reduce stored energy, pellet clad interaction fission gases release and increased margin on DNB

This paper describes the different possible "cold fuels' and the very first associated core physics studies in term of spectral effects on different temperature coefficients

1 INTRODUCTION

Feed back experience from the 56 PWRs which are now (Dec 1990) under operation in France, has demonstrated that actual Nuclear Water Reactors (W R) can reach a high standard of safety and operational reliability

However, looking to the future reactor generations, it seems always possible to further reduce the residual risk and to improve reliability

At the CEA development programmes for future reactors are directed simultaneously according to two approaches

1) The evolutionnary approach on the basis of the actual core design, the efficiency or reliability of safety related equipments (and procedures), materials etc , is increased so as to reach an overall higher level of safety
2) The "revolutionnary" or innovative approach consists in inquiring into reactor physics, for conditions which tend to ensure more forgiving systems through the optimization of the various reactivity coefficients, the elimination of initiators of reactivity accidents, the utilization of a colder fuel etc

This second approach improves the level of prevention rather than increasing the protection against the risk

The subject of this paper is to illustrate specifically the second approach

For this purpose two aspects, now under investigation at the DRN* will be developed These aspects are related to

1) Boron free PWRs (coolant without soluble boron)
2) Implementation of "cold fuels"

2 BOROW FREE PWRs

2 1 Interest in suppressing soluble boron

Safety Soluble boron appears as the major initiator of reactivity accidents in case of untimely dilution Indeed, probabilistic studies showed an occurrence frequency which was not neglectible both during operation and on cold shutdown

Limitation of cycle length Boron content in the coolant is limited to < 1200 ppm to fulfill the requirement of a negative temperature coefficient For instance with a pure uranium core in a 1300 MWe with 1/4 core reloading, the average burn up limit is 55 GWd/t , with Mox fuels, due to hardenning of the spectrum and subsequent loss of absorption properties of boron, cycle length would be further shorten (by a factor close to two)

Limitation of load follow Because of limited dilution rate load follow with boron adjustments is not possible during the last part of the cycle

Waste Potential chemical polution with boron and a source of tritium could be removed Also, the efficiency of the CVCS purification on ion exchangers (i e for Cs) will be improved without the interference of boron

Simplification of the systems and of operation Boron injection systems, CVCS, waste processing systems (boron recycling) will be consistantly simplified For operators, it is also the end of the constraining specification on coordinated boron to lithium ratio

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* DRN Direction des Réacteurs Nucleaires (Nuclear Reactor Directorate)

Corrosion   Suppresion of boric acid is expected to

- improve corrosion rates of austenitic materials, wich are at the origin of dose rates on out of core surfaces,
- presumably have a positive influence on IASCC in the S G. tubes (as one knows, 90 % of these craks are originated from the primary side),
- presumably, improve corrosion resistance of zircaloy cladding, this parameter being one of the barrier to extend burn-up (it has been shown that Li^+ recoil from $^{10}B(n,)^7Li$ reaction was penalizing to zircaloy corrosion rates)

However one of the positive aspect of the boron is the usually homogeneous absorbance minimizing the axial offset.

2.2. Quantitative aspects of boron in reactivity control

The proeminent part of soluble boron and the orders of magnitude of the associated negative reactivity are, for a 1300 MWe PWR :

- fuel burn-up : 8500 pcm,
- Sm and Xe poisoning : 1500 pcm,

(load follow 100 % -> 30 % NP with control rod cluster ~ 3000 pcm)

- fuel Doppler : 500 pcm,
- moderator density · 6500 pcm,
  (cold shutdown)

safety subcriticality margin 5000 pcm,

TOTAL · 22000 pcm

2.3. Alternative options

a) Fuel burn-up the reactivity to ensure cycle length implies the extensive use of burnable poisons (heterogeneous and/or homogeneous with respect to the fissile materials.

Among the criteria for the candidate burnable poisons a difficulty (especially on large commercial reactors) is to accomodate the poison and fuel burn-up rates and to minimize their mismatch, over the entire fuel life (keff close to 1) (mismatch is ajusted by grey rods).

Theoretical considerations show the benefits on combining various poisons (since no single one will burn-out at the same rate as the fuel itself)

Fortunately, many parameters can be considered for this adjustment i.e. :

- nature, isotopic content and density of the poison
- *poison in an inert support (i.e. Al_2O_3),*
- self shielding, by adjusting the spatial location in the core and using discretization of the material (grains with adjusted sizes)

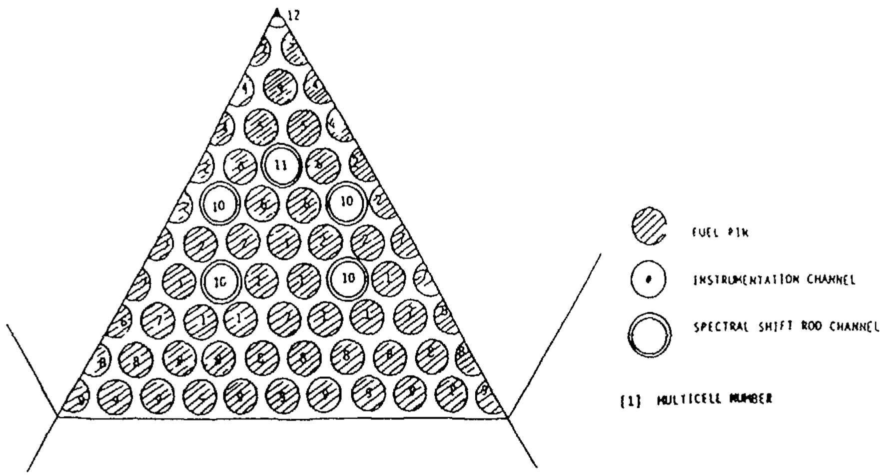

FIGURE 1 : 1/6 OF THE SSCR ASSEMBLY

Minimization of the penalty on fuel lifetime (residual poison) will require refined neutronic studies and a new philosophy of in pile fuel management

The safety interest, is to lead to a global reactivity coefficient close to one as a constant for the cycle length and so to approach the inherent safe assembly, concerning the excess of reactivity criterion, respecting the definition given by [1].

A preliminary study, at the fuel pin level, has been performed, concerning a SSCWR (Spectral Shift Convertible Water Reactor) MOX fuel loaded (Pu/U + Pu = 9,14 %) core, moderating ratio close to 1,4 [2], where several rods (depleted UO_2, Gd_2O_3, ZrB_2...) were fitted into the 30 guide thimbles of the 331 locations of the assembly (figure 1). The results of the study indicated that, at first, the water displacement caused by the poisoning rods induces a hardenning of the spectrum and its associated non neglectible variation of the moderator temperature and density coefficients caused by both the lowering of the volumetric ratio of water and the hardenning of the spectrum Then the major result is the spectral interaction between Gd_{155}, Gd_{157} and UPu which grows stronger the intrinsic direct negative reactivity of Gd_2O_3 (direct effect of $Gd_2O_3 \sim$ - 2600 pcm, induced spectral effect on Pu ~ -4000 pcm).

An another degree of freedom is the abundance of the Gadolinium in the support (figure 2) and the isotopic weight abundance.

The effects are rather different for the burn-up kinetic and the Pu induced spectral efficiency (table 1). *Figure 2 shows that between 0 and 500 (mg/cm^3), natural Gd_2O_3 is a burnable poison with a* kinetics able to roughly change the affinity of the physics (of the curves), between the reference case and the perturbed one. Between 500 and 1000 (mg/cm^3). Gd_2O_3 is an absorber and it does not change the physics because its burn-up kinetic is not well fitted to the fuel one.

TABLE 1 : REACTIVITY EFFECTS FOR A 0 TO 10000 MWd/t BURN-UP

	Gd_2O_3 150 mg/cm^3				Gd_2O_3 1000 mg/cm^3			
ISOT	Direct	Spectral	H.O	TOTAL	Direct	Spectral	H.O.	TOTAL
Gd155	512	-1450	848	-90	714	-259	60	515
Gd157	2275	-6552	6125	1848	897	-597	143	442
Pu241	996	125	12	1135	1156	-142	-15	998
Pu239	-2683	933	-85	-1835	-1840	-146	9	-1978
F.P	-3150	0	0	-3150	-3168	0	0	-3168
Oth	49	-12	-1	-72	+6	-78	0	+33
Total	-2001	-7063	+6900	-2164	-2132	1222	196	-3158

	Water hole/Gd_2O_3 150 mg/cm^3 0 MW				Water hole/Gd_2O_3 150 mg/cm^3 10000 MWd/t			
ISOT	Direct	Spectral	H.O.	TOTAL	Direct	Spectral	H.O.	TOTAL
Pu241	0	-1059	0	-1059	399	-1304	-45	-950
Pu242	0	115	0	115	33	114	-2	145
Pu239	0	-4441	0	-4441	166	-4451	-25	-4310
Pu240	0	1395	0	1395	161	1276	-15	1421
Cm144	0	0	0	0	570	0	0	570
Gd155	-790	0	0	-790	-663	0	0	-663
Gd157	-1796	0	0	-1796	-518	0	0	-518
Oth	-212	+72	-21	-161	-682	+3934	-2911	-178
Total	-2798	-3918	-21	-6737	1054	-4311	-2998	-4483

	Water hole/Gd_2O_3 150 mg/cm^3 225000 MWd/t			
ISOT	Direct	Spectral	H.O.	TOTAL
Pu241	0	-1059	0	-1059
Pu242	0	115	0	115
Pu239	0	-4441	0	-4441
Pu240	0	1395	0	1395
Cm144	0	0	0	0
Gd155	-790	0	0	-790
Gd157	-1796	0	0	-1796
Oth	-212	+72	-21	-161
Total	-2798	-3918	-21	-6737

UNIT = p.cm = 10^{-5}

REACTIVITY EFFECTS
FOR - reference case : water holes
- perturbed one : Gd_2O_3 150 mg/cm^3
FOR - 0, 10 and 22.5 GWd/t

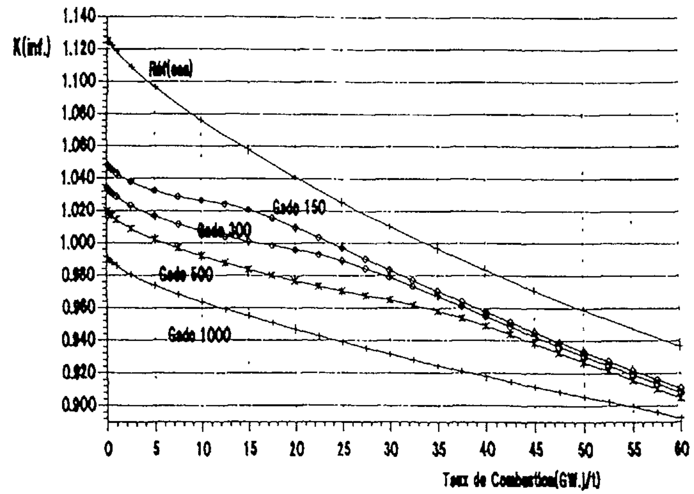

FIGURE 2 :

K∞ VARIATION VERSUS BURN-UP FOR DIFFERENT GADOLINIUM CONCENTRATIONS

Other degrees of freedom are to be studied, particularly the self shielding. For example the efficiency of the heterogeneous (in Al_2O_3) Gd_2O_3 is better (+ 5 %) than the mixed (homogeneous) in fuel one. Concerning the feedback coefficients, heterogeneous Gd_2O_3 leads to roughly lower the moderator density coefficient (absolute value, - 21 %), decreasing the need for negative reactivity to control the cold shutdown. The transposition of this result to a 1300 MWe PWR gives a gain of about 1500 pcm for the cold shutdown . Without optimization, neither of the assembly, neither of the poison and neither of the moderating ratio, the use of burnable poison (natural Gd_2O_3) saves about 10000 pcm, compared to the 22000 requested to control the reactor with soluble boron it is 45,5 %. So is doing by mean of a core conception where mechanical or mobile facilities are not requested, and where no human decisions or actions are necessary.

b) Other means for reactivity control

Other investigations are being pursued on alternate options to replace soluble boron for reactivity control.

- Variations of the primary flow rate and/or variations of the inlet temperature. These techniques which are commonly used in the PWRs operation to strech out the cycle length and to a larger extend in BWRs lead to compensate Sm and Xe poisoning during the load follow. An estimation shows that a 5 % variation of the rotating speed of the primary pump induces a variation of 800 to 1000 pcm for the reactivity. However the sensitivity of this approach is strongly corellated to the moderator density coefficient and to the fuel Doppler, and if the second one is stronger with Gd_2O_3, the first one is lowered (by 21 %).

As an attempt to differrentiate control systems, calculations have been carried out with gazeous poisons rods replacing of grey rods in a 1300 MWe PWR [3] Gazeous rods poisons consist of either BH_3 (natural or enriched boron) or mixed He_3/He_4 in the guide thimbles A part from any considerations on mechanical faisability (and reliability), the equivalent control rods worth is obtained with a pressure of 70 bar for BH_3 (nat) and 35 bar for 25 % of He_3 in mixed He_3/He_4 or with 15 bar for in 10B enriched (90 %) and 8 bar for 100 % He_3

The overall reactivity of such a system has to be kept low enough to make sure that depressurization of gazeous rods does not induce reactivity accident

The major interests, from the neutronical and safety point of view, are .

a rough decrease of the temperature coefficient with the gazeous rods,
the absorbance is axially homogeneous and does not induce Xenon tilts and so restricts the axial power tilts.
- it is a fine tuning control system

Concerning the gazeous rods, many other investigations have to be carried out ; spatial location of the rods, optimization of the radial power shape, mechanical faisability and safety analizis of such a system. However, it should be pointed out that some facilities are operating with equivalent systems : GEM (Gas Expansion Modules) in FFTR, aeroballs in some BWR's and Helium3 gazeous rods in the safety test reactor CABRI therefore, it can be reasonably assumed that this existing technology can be further developed.

3. COLD FUELS

The increasing trends of the Nuclear Community to save fissile materials creates a strong demand to further optimize fuel performance.

Parameters to be possibly increased to satisfy this goals are :

- linear power density,
- peaking capability,
- average burn up at discharge, by increasing the number of cycles and / or cycle length.

However, it is required that the safety criteria and the margins be maintained, at least at the same level of reliability as they are at present time for all the conditions of operation.

The technological limitations of the above mentioned parameters are

1) Cladding resistance to waterside corrosion . oxyde growth and consecutive hydriding of the zircaloy causes a loss in ductility of the metal together with thinning of the clad The actual requirement is expressed as a temperature limit at the metal oxyde interface
2) Fission gases release :
   - increase of the fission gases release with the increase of the fuel centerline temperature,
   - increase of the clad internal pressure,
   - increase of the clad-pellet interaction (iodine).
3) Clad-pellet-interaction during operation :
   - operation : closing of the gap (20 GWd/t),
   - load follow : local linear power density on hot rods may exceed failure threshold.
4) No local melting of the pellet and no DNB (for cat. 1 and 2)

The more penalizing of these two parameters permit to define a limit to the maximum local heat flux so called "critical flux".

Whereas the dry out conditions are dependent only on the thermohydraulics, melting temperature of the fuel is mainly dependent on the burn-up, many manufacturing factors, and, evidently, is an intrinsec property of a particular compound Even if there is an important margin between the fuel melting point and operating conditions, the existence of the critical flux help to avoid failure threshold on load follow From the point of new fuels like metals, with relatively low melting point, these criteria need to be reconsidered in details. Some of these requirements can be achieved on using oxyde fuel under its actual configuration, for instance by decreasing linear power density (as this is a general trend for Advanced - Water - Reactor) Benefits from this limitation are expected on cladding corrosion (decrease in metal to oxide interface temperature), fission gas release, on CPI and DNB margins, and on decreasing the stored energy in case of accident

A study [4] shows that the decrease of the Linear Power Density (LPD), by mean of the decrease of the central temperature in the fuel pellet would lead to :

a) A decrease of the cladding corrosion because of lower oxyde-metal interface temperature. The range of this decrease can be calculated with CEA corrosion code for stress relieved zircalloy 4.
   - - 10 % LPD -> - 30 % corrosion,
   - - 20 % LPD -> - 50 % corrosion
b) A decrease of the fission gas release which leads to increase margins versus internal pressure criterion.
c) Increase of the DNB margin.
d) Increase of the CPI margins.
e) Gain concerning mechanical and chemical aspects.

As a result, higher burn-up can be reached without having to consider new cladding materials.

From the safety point of view, the decrease of the stored energy, by decreasing the pellet temperature is a favorable step in case of accident.

It should be noted that the specific power could be kept unchanged by decreasing the fuel pin diameter and increasing their number.

Other attempts currently under investigation consist in improving UO_2 conductivity with metal additives, or to use a large grain fuel, limiting the diffusion of fission gases.

However these improvements proceed from the first evolutionnary approach and are not precisely the subject of this talk Our "innovative" approach focuses on developing new type of fuels with such characteristics as a high thermal condictivity, a significant high density in fissile atoms, compatibility with the coolant and cladding material and so on.

Four options of new fuels are presented below

- metallic fuels (UAl, UZr), already extensively used in research reactors studied for the LMFBR PRISM [5] From the safety point of view, the disavantage for WR is to drastically increase the inventory of oxizidable metal in the core (during a severe accident the hydrogen from the oxidation of the fuel would add to this originating from the oxidation of zircalloy cladding, possibly leading to explosive reaction) On the other hand metallic fuels present a strong swelling rate on irradiation property which imply to limit the weight density to 60/70 % of the theoretical density,
- carbide (UC) already used in LMFBR, a japanese study shows that it could a very promising fuel for FBR [6], but in WR, the compatibility with hot water (in case of cladding failures) is questionable In addition the fuel reprocessing is not praticable with current technology,

- complex compounds (complex silicides, vitreous ceramics...) these compounds are being studied presently and too few results are avalaible to date to make an analysis,
- nitride fuels (UN). The reprocessing seems to be easy and the limited swelling rate is a point of interest. Used by the USSR [7] for LMFBR. Interest as WR fuel has not been demonstrated, so a tentative answer is discussed in the following lines.

Nitrides :

The theoretical density is = 14,32 g/cm^3 (UO_2 = 10,12 g/cm^3).

The averaged pellet temperature in nominal operation is 462°C (650°C for UO_2), for a average coolant temperature at 306°C.

The thermal conductivity is :

- 12 Wm^{-1} K^{-1} at 230°C,
- 21 Wm^{-1} K^{-1} at 1000°C.

The fusion point of non irradiated materials is > 3100 °C (2700°C for UO_2).

Comparizons were made using the APOLLO code [8] for the infinite medium calculations and using the DAPHNIS code [9] to analyze the results in terms of perturbation theory. The geometry of the fuel pin is the French standard PWR one. The selected density for UN was 90 % of the Theoretical density (Td).

At the beginning of cycle, results show an important decrease of the k of UN with respect to UO_2 (- 7100 pcm), the thermal flux depression is attributable to ^{14}N, due to the following reactions :

- ^{14}N (n,p)^{14}C : 77,6 %
- ^{14}N (n,t)^{12}C : 0,4 %
- ^{14}N (n,)^{11}B : 19,1 %
- others (n,γ; n,d ; n,2α) 2,8 %

It can be noted that the cut off of the thermal component by N14 leads to harden the spectrum. This effect being amplified by the strong concentration of ^{238}U. This spectral shift induces a lowering of the soluble boron efficiency : - 7,4 pcm/ppm for UN and - 12 pcm/ppm for UO_2.

The variation of the reactivity with the weight density shows (figure 3) that by decreasing the density the negative reactivity due to the decrease of ^{235}U is easily compensated by the decrease of ^{238}U and N14 concentrations inducing a positive reactivity. However, we have shown that this contribution was due for its larger part to the depletion of absorbing ^{238}U, more than *to the increase of fissile* ^{235}U.

In view of these facts, it can be concluded, that natural UN induces prohibitive absorption of thermal neutron and prohibitive gas production. In addition to these disadvantages UN oxidises with hot coolant which leads to more or less rapid ruin of the pellet. In case of accident, contact with hot vapor will also yield to production of consistent amount of hydrogen. Ternary nitride compounds such as UZrN, UAlN and USiN present a better thermodynamical stability and a better resistance to corrosion. Alternatively, the coating of UN by a fire proof oxide like Al_2O_3 is of interest.

The compound recently under development is the ternary nitride UZrN including a 80 % enrichment in N15 (the very low absorption rate of N15 is less than 100 pcm at beginning of life), the reactivity equivalent UO_2 ternary nitride is :

U_{80} % Zr_{20} % (U_5 = 3,7 %) $N15_{80}$ % d = 70 % Td

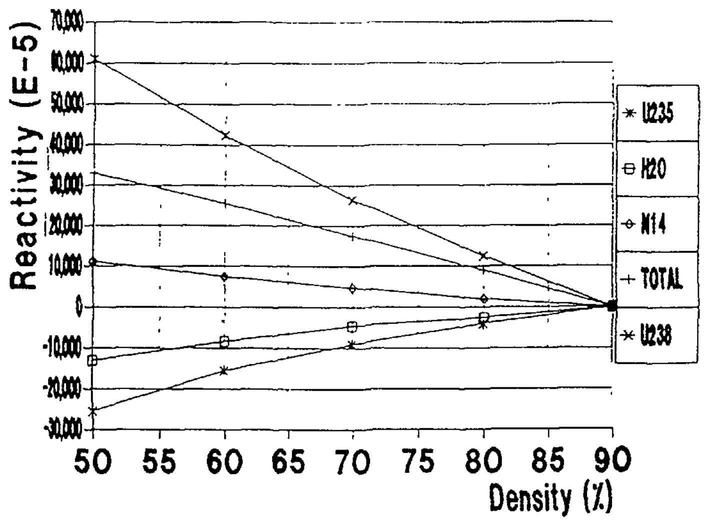

FIGURE 3

INDIVIDUAL EFFECT OF THE DENSITY VARIATION VERSUS 90 % Td UN

Voided core : a 50 and 80 % voided core are calculated, the effect is strongly negative with UZrN more than with UO_2 :

50 % UO_2 = - 8600 pcm UZrN = 9050 pcm
80 % UO_2 = - 27200 pcm UZrN = 28300 pcm

Feedback coefficients on the following tab one can see that if all the temperatures coefficients are more important for UZrN the lower DT for cold shutdown ensures a less requirement for negative reactivity : 1250 pcm for UZrN, 1600 pcm for UO_2. The very strong moderator density coefficient in the UZrN spectrum requests 5500 pcm of negative reactivity versus 5150 for UO_2.

	UZrN		UO_2	
	Doppler ΔT = 442°C	Moderator ΔT = 286°C	Doppler ΔT = 630°C	Moderator ΔT = 286°C
Temp.Coeff. pcm /°C	- 2,84	- 19,23	- 2,48	- 18,01
Average Temp. coeff. pcm /°C	- 15,27		- 10,71	

The global effect for cold shutdown is 6750 pcm for UZrN and 6750 pcm for UO_2. The use of UZrN requires the same control rods for the cold shutdown as for UO_2 but UZrN will be more forgiving on power or reactivity transients

Studies on burn-up

Figure 4 shows the burn-up behaviour of different nitride fuels. The different slopes of the curves can be interpreted as the result of a very different production consumption rates for ^{235}U and ^{239}Pu. Table 2 shows that, in fact the in-out masses balances are rather different

The most important physics features is the effective moderating ratio variations with the type of fuels. Starting from a standard PWR assembly (moderating ratio MR = 2), one obtains on effective MR of 1,3 for UN, 1,8 for UZrN d = 90 % Td, 2,3 for UZrN d = 70 % Td So varying the type and the density of the fuel it is possible to increase the consumption of ^{235}U and ^{239}Pu (V_m/V_f = 2,3 over moderated core) or to increase the production (and in quality too) of ^{239}Pu (V_m/V_f = 1,3 under moderated core). Doing so the impact of these fuels is very strong on the strategy and cycle, particularly for the minimization of the Actinide production. Table 2 shows that UN and UZrN cycle lengths are shorter, versus reactivity, than for UO_2, but corrosion and gases release are roughly lowered it becomes thus possible to increase cycle lenght by mean of ^{235}U enrichment. Concerning the safety, including a metal in a ternary compound is not a favorable step because the oxidation and hydrogen release in case of accident, it will be preferable to sudy compound including unoxizidable material (i.e. Al_2O_3).

From the economical point of view, several estimations [7] [10] give a 10 to 20 F/g price for N15 and show that the N15 enrichment cost presents very low variations between 30 and 90 % of 15N. A recent synthesis [11] gives, for LMFBR a 16 F/g price as a threshold of profitability.

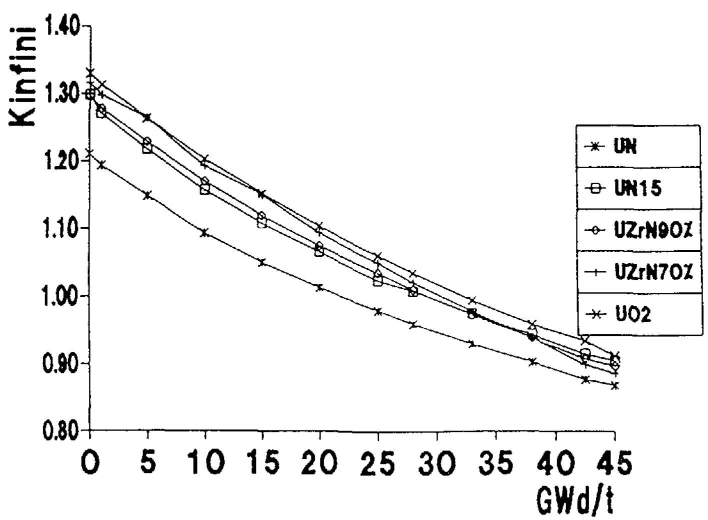

FIGURE 4
VARIATION OF THE KINFINI WITH THE BU

TABLE 2
CRITERION EQUIVALENT k INFINI FOR UO_2 B U = 23,75 GWd/t

		UO_2	UZ_rN 70%	UZ_rN 90%	UN_{14}	UN_{15} 80%
Tc MOYEN	MWj/t	23750	24600	22600	15200	21800
LOT	MWj/t	10300	9840	9040	6080	8720
DUREE	JEPP	256	240,2	220,3	151,6	217,5
Energie	TWh	5,53	4,39	5,18	4,36	6,26
Fin de cycle	MWj/t	41200	39360	36160	24320	34880
Pu 9	%	51,56	51,6	56,36	67 83	59,06
Pu 0	%	24,1	24,85	82,54	17 44	20,62
Pu 1	%	14,37	14,04	14	11,45	13,62
Pu 8	%	2,3	2	1,42	0 8	2,08
Pu 2	%	7,2	7,14	5,29	2,24	4,25
Am	%	0,42	0,4	0,39	0 25	0,37
N Pu TOT		2,3031-1/1	1 7806-1/0,773	2,4839-1/1,08	2,8381-1/1,23	3,3863-1/1,47
Pu Fis	%	65,93	65,64	70,35	79,28	
$U5^f$		1,7-1	1,5-1	2,13-1	5,3-1	3 514-1
$U8^f$		21,075	17,926	23,06	29 05	28,747
T rejet	%	0,8	0,83	0,92	1,79	1,206
ΔU5		6 7572-1/1	5,6705-1/0,84	7,0892-1/1,049	6,224-1/0 92	8,01-1//1 185
ΔU8		0,658/1	0,501/0,76	0,632/0,96	0,565/0 86	8,18-1/1,243
ΔU5/NU5	%	79,9	79,08	76,89	54	69 5
ΔU8/NU8	%	3,02	2,72	2,67	1 91	2,76
Δ (U5+U8)		1 33372/1	1,068/0,8	1,34092/1,005	1 874/0 89	1 619/1 214
Δ(U5+U8)/(U5+U8)ⁱ	%	6,13	5,58	5,45	6 09	5,26
P		900 MW	763 MW	981 MW	1200 MW	-
Masse		72 5 t	60,4 t	77,6 t	97 t	-
P/lin		175 5 W/cm	148 8 W/cm	191,3 W/cm	234 W/cm	-

Estimations give a possible price for 15N close to 2 F/g for a 80 t/y production Considering a 10 F/g averaged cost of 15N it is only 3 % of the fuel cycle cost (including reprocessing) and 0 7 % if the price is 2 F/g

The resistance to oxizidation of different ternary compounds is a part of on going experimental studies conducted at CEA/Cadarache The results (avalaible at the end of 1991) will allow to interest the studies in qualification core calculations for the candidate compounds Associated with economic and strategic calculations these studies will allow to define the place that cold and dense fuel would take in the future reactors

4 CONCLUSION

The review of the physics exposed here leads to define the major axes of a three points programm of studies requesting several competencies

- Convertible boron free PWR
- Lowered linear power density convertible assembly with locations for a control using burnable poisons
- Cold and dense fuel allowing to increase the performancies in terms of cycle length, economy , while respecting (or increasing) the level of the safety

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REFERENCES

[1] IAEA consultants group safety related terms of advanced nuclear plants Final report Vienna 3 6 December 1990

[2] J PORTA and al
Search of a few group structure for plutonium fueled advanced reactors
4th ICENES, 30th June, 4th July 1986 MADRID

[3] P THOMET
Emploi d'absorbants gazeux dans les reacteurs a eau pressurisée
Consequences sur la neutronique du coeur Internal CEA/DER/SIS note 91 1022 June 1991

[4] M NOE
Internal DER/SIS Working group and private communication

[5] VAN THUYLE and al
Summary of advanced LMR evaluations PRISM and SAFR NUREG/CR 5364 October 1989
SALERNO and al
ALMR fuel cycle economics San Diego August 1989

[6] H TEZUKA and al
Conceptual desing study of LMFBR core with carbide fuel
Int top ANS meeting Paris April 1987

[7] I S SLESAREV and al
The high safety and economy NPP with liquid lead cooled reactors
Seminar Moscow 22 24th October 1990

[8] A KAVENOKY
Neptune a modular system for the calculation of LWR
Conf 750413 Charleston 1975
R SANCHEZ J MONDOT
Specifications detaillees APOLLO 2 internal CEA note SERMA T 1632

[9] J PORTA and al
Sensitivity studies on high conversion power reactor via perturbation methods and information conditionning
Nucl Sci Eng 95(4) 266 1987

[10] M NOE J ROUAULT
Internal CEA working group DER/SIS DEC/CPCE and private communication

[11] M NOE
Justification d'une étude technico economique de l'enrichissement isotopique en N15 pour les combustible RNR futurs
Internal CEA DER/SIS note 91/1027 April 1991

LOW POWER REACTIVITY INITIATED ACCIDENTS IN NEW GENERATION LIGHT WATER REACTORS

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Abstract

Plant low-power operation had been generally considered to be a safer condition than high power operation. Results from Probabilistic Risk Assessment studies for French PWR's pointed out the importance of risk when the reactor is shut-down or in refuelling.

The low-power concern is in part directed at loss of decay heat removal accidents. This paper is focused however on reactivity accidents, which are worst at zero/low power than at rated conditions due to the larger amount of reactivity that can be potentially added, to the lower moderator feedback available to quench the excursion and because control systems operative at power are often inoperative during shutdown.

A review of low-power reactivity accident scenarios for SBWR and AP 600 reactors is provided.

In particular, 3-dimensional space-time analyses are presented for control rod drop/control rod ejection accidents in the SBWR; the possibility to mitigate such events by a suitable startup procedure is discussed.

Reactivity excursions due to the insertion or dropping of a fuel assembly into the core during plant refueling are discussed for SBWR and AP 600.

Accident scenarios for AP 600, involving the possibility of rapid boron dilution, are discussed; the possibility to improve the plant performance by the use of burnable absorbers is highlighted.

1 Introduction

Plant low power operation (i e < 5%) had been generally considered to be a safer condition than high power operation

Most of the events occurring at high power conditions, like for example turbine trip, loss of feedwater, MSIV closure, are less severe at low power or are not possible at all

The same is true for LOCA's, due to the reduced level of stored energy in the fuel and of radionuclides inventory

Furthermore LOCA's are by far less probable at zero power due to the drastic reduction of the primary system pressure

In spite of these considerations, PRA studies for French PWR's put emphasis on the importance of risk when the reactor is shutdown or when it is in refueling These states account for over half (55 %) of total calculated core melt risk in the 1300 MWe plants and almost one-third in the 900 MWe plants These high values may result from the fact that there are generally no automatic systems to counter accident situations in shut down conditions and that human intervention is therefore necessary

Though the concern is in part directed at loss of residual heat removal system events, this paper will focus on reactivity accidents initiated at zero / low power conditions which represent the most limiting power level for many reactivity accidents in LWR's due to

i) the larger amount of reactivity that can be potentially added (with obvious exceptions like void reactivity in BWR overpressurization transients),

ii) the lower moderator feedback available to quench the excursion,

iii) possible unavailability or uneffectivness of reactor protection system

In particular this latter point was stressed in the US Nuclear Regulatory Commission report subsequent to the Chernobyl accident [1], which recommended to focus on sequences that involve relatively large reactivity additions due to one or more of the following

- low or zero power or testing as an initial condition
- inadequate response of shutdown systems
- a positive moderator temperature coefficient
- human error
- deliberate bypassing or disabling of any safety feature

As an outgrowth of such recommendations, Ref [2] investigated, from a deterministic and/or probabilistic point of view, LWR reactivity event sequences for their potential to result in fuel fragmentation and prompt dispersal

In this paper we intend to review reactivity initiated accidents (RIA) during zero and low power operation for some of the new generation light water reactors, that is the Simplified Boiling Water Reactor (SBWR) and the PWR type AP600, highlighting design and procedural features that improve inherent plant reponse to RIA's

Analyses for the SBWR, discussed in Section 2, were performed within the framework of a more general agreement between General Electric and ENEL, they are at a final stage and are based on detailed three dimensional space-time calculations

Preliminary evaluations for the AP600, discussed in Sections 3, were aimed at the individuation of accident scenarios worthwhile to be analyzed in detail by dynamic calculations Section 4 summarizes the conclusions drawn in this report

Finally, some details about calculational models are presented in Appendix 1

2 SBWR Reactivity events

Low power BWR reactivity event sequences can be grouped as follows [9]

i) Events due to the rapid removal of control absorbers

1) Control rod drop accident
2) Control rod ejection accident
3) Continuous control rod withdrawal
4) Boron dilution during ATWS,

ii) Events due to the insertion of fuel bundle (Refueling accidents)

5) Fuel bundle loading accident

6) Fuel bundle drop accident

iii) Events with rapid insertion of cold water

7) ATWS with boron injection failure

Sequences 1), 2), 5) were explicitly analyzed for the SBWR plant, by 3 dimensional space time calculations (see Appendix 1 for a short description of analytical methods), in support to the SBWR PRA study and the results are discussed in the next sections A qualitative discussion is provided also for the sequences 3) and 6)

Sequence 4) consists in the flushing of the boron injected by the Standby Liquid Control System during the latter phase of an ATWS, by unborated ECCS water when the pressure is abruptly lowered below the low-pressure injection set-point by the Automatic Depressurization System (ADS) inadvertant actuation The possibility that boron flushing might lead to a severe power excursion had been of concern for standard BWR's [2]

The accident was recently analyzed for a BWR-4, the Browns-Ferry plant, by RELAP5/MOD2 [3] The injection of cold unborated water causes indeed a severe power excursion (with power peaks of around 3 4 times the rated power), but inherent feedbacks limit the peak fuel enthalpy to less than 220 cal/g Although fuel prompt dispersal is not expected, fuel damage may occurr as a result of excessive clad temperatures This may be caused directly by the high power levels leading to boiling transition or indirectly because the energy generated in the core cannot be dissipated and excessive containment suppression pool temperature can lead to failures of systems cooling the core

Previous sequence is very similar to the case where there is a boron injection failure after an ATWS (Sequence 7 above) with ADS actuation and low pressure injection In that case, the power is not shut-down prior to depressurization, but once that cold water enters the system, a power excursion is produced, due to the decrease in saturation temperature and subsequent void collapse

Sequences involving plant depressurization and rapid inlet of cold, unborated water from the Gravity Driven Cooling System (GDCS) is less important in the SBWR, due to the automatic inhibition of ADS Furthermore, GDCS injection flow is very sensitive to the vessel backpressure, so that flow is reduced or stopped by the pressure surge caused by the power excursion Anyway, the event is much slower than in a standard BWR

2.1 Control rod drop accident analysis

The control rod drop accident (CRDA) is the postulated dropping of a control rod at its maximum velocity In order to be able to fall by gravity, control rod blade must separate from the control rod drive, with the blade sticking in the inserted position while the drive is withdrawn

Control rod drive design in the SBWR is completely different from standard GE design, to allow control rod fine motion As a consequence of such modifications, features has been introduced, like a modified drive-blade coupling and a separation detection device, which make the possibility of a control rod drop even more remote On the other hand the control rod blade free fall velocity is much higher than in standard BWR's (about 3 versus 0 9 m/s average speed), due to the elimination of the control rod velocity limiter

In the same way than in standard BWR's, control rod worth is minimized by the hard-wired Rod Worth Minimizer system (RWM), which constrains the sequence of control rod withdrawal

Even with multiple errors (that is with multiple failures of RWM), control rod worth is significantly less in SBWR than in a conventional BWR maximum worths with single and multiple errors are respectively 1 9% and 3 0% (SBWR with 9ft long fuel at the equilibrium cycle, 20 °C) to be compared with typical values of about 2 5% and 4 0% for a conventional BWR core [4],[5] This important inherent feature is reflected by the increased shut down margin allowed by the heavy use of burnable absorbers and by the reduced power density of the SBWR plant

Control rod drop accident has always been considered the bounding reactivity accident for BWR's generic sensitivity studies [4], [6] concluded that zero power conditions are the limiting ones for this accident, since, when the reactor is at power, control rod worth is significantly reduced and voids exist in the core to quench readily the reactivity excursion

TABLE 2 1

Summary of SBWR CRDA results

RUN	DESCRIPTION	ADDED RIA (%)	MODERATOR INLET SUBCOOLING (°C)	PEAK FUEL ENTHALPY (cal/g)	PEAK POWER (MW)	PEAK POWER TIME (s)	VOID ONSET TIME (s)
1	Single Error	1 85	98	179	60724	0 503	0 862
2	Multiple Error	3 03	98	321	85117	0 393	0.588
3	Multiple Error with scram	3 03	98	299	63182	0 423	0 646
4	Multiple Error	3 09	38	303	94310	0 389	0 440
5	Multiple Error	3 09	1	21	15949	0 379	0 378
6	Single Drop	2 25	38	175	34660	0 311	0 421
7	Single Drop	2 34	18	187	38756	0 307	0 331
8	Multiple Drop	5 65	38	>375	276424	0 253	>0 260
9	Multiple Drop	5 72	28	306	281844	0 252	0 252
10	Multiple Drop	5 74	18	259	281868	0 251	0 250

Beginning of equilibrium cycle

Runs 1-4 SBWR 9ft Single control rod drop

Runs 5 9 SBWR-8ft Multiple RWM failures

Core power 10^{-8} rated power

Core flow 15% rated flow

Moderator Saturation Temperature 118 °C (Runs 1-4 6-8) / 81 °C (Run 5)

CRDA was analyzed at cold zero power conditions (moderator temperature less than 100 °C), for two different SBWR core configurations, that is

- the 2000 MWth core with 9 ft fuel active length (SBWR-9ft, the present core configuration), during the equilibrium fuel cycle,
- the 1800 MWth core with 8 ft fuel active length (SBWR-8ft), at first cycle beginning

Analyses for the SBWR-9ft considered a single control rod drop, with single or multiple errors in the control rod withdrawal sequence Analyses for the SBWR 8ft considered instead single or multiple control rod drops, with multiple errors in the withdrawal sequence

The results obtained are summarized in Table 2 1, while figures 1-3 plot some key data for the most severe single CRDA analyzed (case 2 in Table 2 1)

Most of the accidents were analyzed assuming no scram scram of course would be required anyway in the long term to keep the reactor shut down at zero power, but in the short term it is scarcely effective, reducing fuel peak pellet enthalpy by less than 10% (compare cases 2 and 3) In fact, though scram is triggered very early (0 006 s into the transient, plus 0 2 s delay) by a low period signal of the startup range neutron monitor system (which is a new feature respect to standard BWR's), too few rods are available to add anti reactivity to be effective in reducing fuel enthalpy

The dependence of CRDA behaviour on accident parameters is now discussed

The initial power burst is limited by Doppler feedback (with the exception of the low subcooling cases 5 and 10), note in Figure 1 that, at the power peak, the total reactivity is 1 \$, as it should be in a super-prompt critical excursion, fuel temperature rise decreases the neutronic importance above the control rod tip This effect, in turn, reduces control rod reactivity

SBWR-9ft CONTROL ROD DROP ACCIDENT
MULTIPLE RWM FAILURE 20 C (CASE 2)

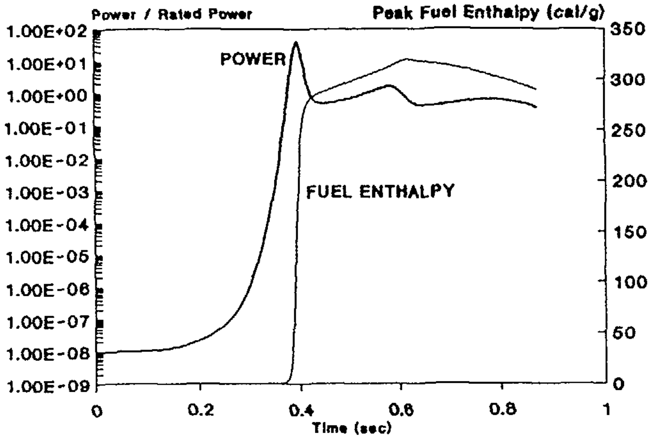

FIGURE 1

SBWR-9ft CONTROL ROD DROP ACCIDENT
MULTIPLE RWM FAILURE 20 C (CASE 2)

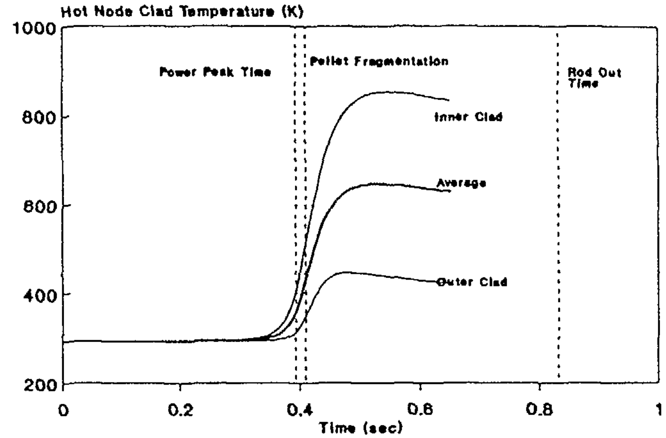

FIGURE 3

SBWR-9ft CONTROL ROD DROP ACCIDENT
MULTIPLE RWM FAILURE 20 C (CASE 2)

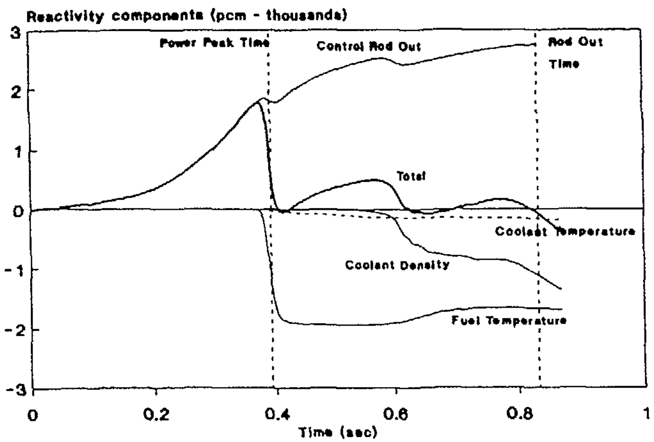

FIGURE 2

SBWR CONTROL ROD DROP ACCIDENT AT CZP
HOT NODE FUEL ENTHALPY VS REACTIVITY

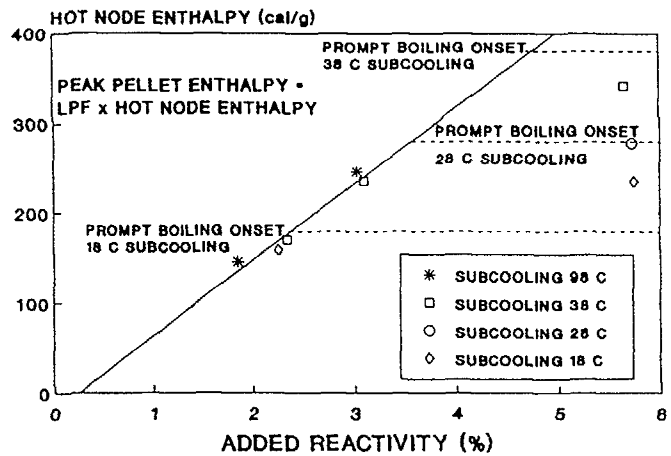

FIGURE 4

Energy is transferred to the coolant by heat conduction and by neutron slowing down and gamma rays absorption Delayed gamma absorption and all energy contributions deposited within fuel cladding were not taken into account in our analyses a conservative value of 3% was therefore used as moderator to fuel energy deposition ratio

The initial feedback, caused by moderator temperature rise doesn t contribute significantly to the reactivity balance

Though the "effective" heat conduction time constant during severe CRDA's is much lower than at quasi steady state conditions, being of the order of tenth of seconds instead of seconds, the initial energy transfer is essentially due to prompt moderator heating

However as soon as void formation occurs, the reactor is quickly shutdown in an inherent way the larger is moderator subcooling, the longer is void onset time

For a fixed core design, key parameters affecting the accident behaviour are the added reactivity and moderator subcooling The interplay between these two parameters is summarized in Figure 4, which plots the hot node average enthalpy (peak pellet enthalpy divided by the local peaking factor) versus the added reactivity (excluding case 3, with scram , and 5 at different saturation temperature) If the subcooling is high enough, fuel enthalpy doesn t depend significantly on moderator temperature (compare cases 2 and 4) in such case in fact voids come into play just to limit the "tail of the accident, but fuel enthalpy is primarily determined by the added reactivity and Doppler feedback

If the moderator temperature is rised enough, prompt moderator heating by alone (i e with no contribution from heat conduction) will be able to overcome moderator initial subcooling and get saturated boiling It can be easily shown, by a plain energy balance, that, neglecting subcooled voids and assuming stagnant moderator (which is justified in view of the short accident time scale), node average fuel enthalpy at saturated boiling onset is given by

$$dH_f = R_{w/f} C_p (T_s \quad T_{in}) / ß$$

where

$R_{w/f}$ moderator to fuel mass ratio,

T_s moderator saturation temperature,

T_{in} moderator inlet temperature

C_p water specific heat,

ß moderator to-fuel energy deposition ratio

dH_f node average fuel enthalpy rise at saturated boiling onset

In such regime, fuel enthalpy will depend essentially on subcooling alone, because larger reactivity additions are easily compensated by an increased core voidage At the low pressures typical of cold zero power conditions, such increased voidage is obtained at the expense of a small fuel enthalpy increase, so that a plateau in the reactivity enthalpy curve is obtained Data in Figure 4 (corresponding to cases 7, 8, 9 in Table 2 1) show that node enthalpies from 3 D dynamic analyses are in fair agreement in the plateau region with the "prompt boiling onset enthalpy calculated from previous equation

Data in Table 2 1 show that, even in the event of a CRDA with a single RWM failure, core damage would not be very extensive Fuel peak pellet enthalpy is just above the irradiated fuel rod cladding failure threshold (about 140 cal/g, [7]) and below that one for unirradiated fuel (205 to 225 cal/g [7]) assuming about 60 cm of failures per rod, about 110 irradiated fuel rods are expected to fail, while fresh fuel should not fail at all

Should a CRDA with multiple RWM failure and high inlet subcooling (> 25 30 °C) occur, the hottest pellets would be fragmented (peak pellet enthalpy > 280 cal/g, [7]), about 80 rods (both irradiated and fresh) would be broken up and fractured (fuel enthalpy > 220 cal/g, [7]) Further 350 irradiated rods will have the cladding cracked and failed (fuel enthalpy > 140 cal/g)

However, if moderator subcooling is low enough, even an extreme and unrealistic addition of more than 5 5% reactivity (simultaneous drop of 5 control rods, with multiple RWM failures) would not produce fuel pellet fragmentation

The original SBWR startup plan was to use nuclear heating from cold zero power (with typical moderator temperatures around 50-60 °C) up to rated conditions

The current SBWR startup plan requires instead to withdraw control rods only after completion of the following steps [8]

Establish a vacuum in the reactor vessel to about 0 1 atm,

Begin heating of reactor coolant by terminating shutdown cooling operation and, if necessary, by external heatup to about 80 °C

This procedure allows to get a core inlet subcooling close to zero case 5 in Table 2 1 shows (CRDA with multiple RWM failures) the subsequent dramatic reduction of fuel enthalpy to inconsequential values (peak pellet enthalpy of about 20 cal/g)

The revised startup procedure therefore eliminates, in an inherent way, the possibility of fuel fragmentation, even for low probability multiple failure CRDA scenarios

The only remaining possibility to have a serious CRDA is then during the shutdown margin demonstration test, performed to verify the existence of an adequate subcriticality margin at the end of each refueling, with the reactor vessel still open In that case the containment is open and a direct path to the environment exists for the fission products in such conditions not only fuel fragmentation, but even the lack of fuel cladding integrity is of concern Fission product release from core would be more severe than for the fuel handling accident (drop of a fuel assembly, with subsequent fuel rod mechanical damage), the refueling accident usually considered for off-site dose evaluation, due also to the much larger fission product inventory released by each fuel rod in the CRDA case However, the probability of a CRDA during the shutdown margin demonstration test is even lower than during startup, due to the low frequency occurrance of such testing condition, and can be considered negligible

2.2 Control rod ejection accident analysis

Control rod ejection accident (CREA) is usually not analyzed in standard BWR's because the control rod housing support system, provided for safety, has been judged sufficient to reduce the probability of the event to levels below those requiring standard analysis

The redesign of control rod drives in the SBWR affects this conclusion control rod ejection would be possible for different reasons, including major breaks of the control rod drive housing or failures of the drive insert line from the hydraulic control unit (HCU) Since each HCU is connected to two control rod drives, in the latter case a double control rod ejection could be possible however the control rods assigned to the same HCU are sufficiently far apart to be neutronically decoupled

Control rod ejection speed will depend on the system pressure at the rated core pressure the average speed is about 1 2 m/s, significantly less than the control rod free fall velocity

The peculiarity of CREA is that it is physically possible only if the reactor is pressurized the bounding conditions are those with the lowest possible core void inventory, that is at zero power In the SBWR, plant pressurization is accomplished by nuclear heatup of the coolant, so that the moderator is anyway close to saturated conditions moderator subcooling in the upper part of the core is in fact less than about 1 °C

TABLE 2 2

Summary of SBWR CREA results

RUN	DESCRIPTION	ADDED REA (%)	MODERATOR INLET SUBCOOLING (°C)	PEAK FUEL ENTHALPY (cal/g)	PEAK POWER (MW)	PEAK POWER TIME (s)	VOID ONSET TIME (s)
1	Single Ejection	1 29	1	32	3684	1 127	0 85
2	Double Ejection	1 95	1	37	8335	0 941	0 93
3	Double Ejection	2 05	1	31	5560	1 121	1 10
4	Double Ejection	2 21	4	26	1457	1 140	0 93

Run 1 Beginning of equilibrium cycle

Runs 2 3 4 End of equilibrium cycle

Pressure 1019 psi (Runs 1 2) / 750 psi (Runs 3 4)

Core Power 10^{3} (Runs 1 2 3) / $9x10^{-3}$ (Run 4) rated power

Core Flow 15% rated flow

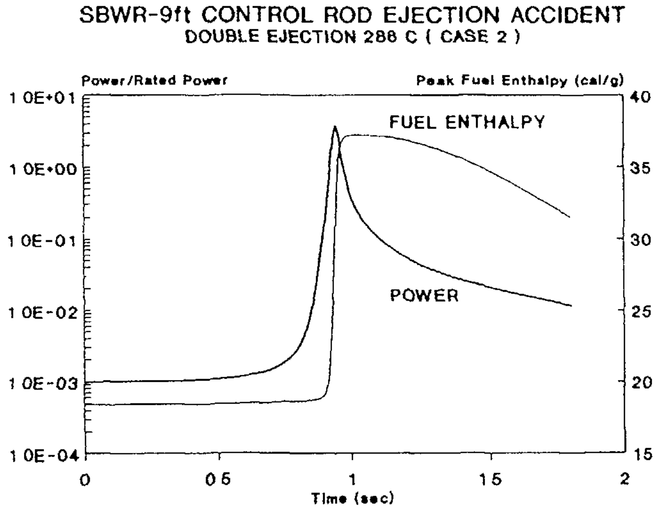

FIGURE 5

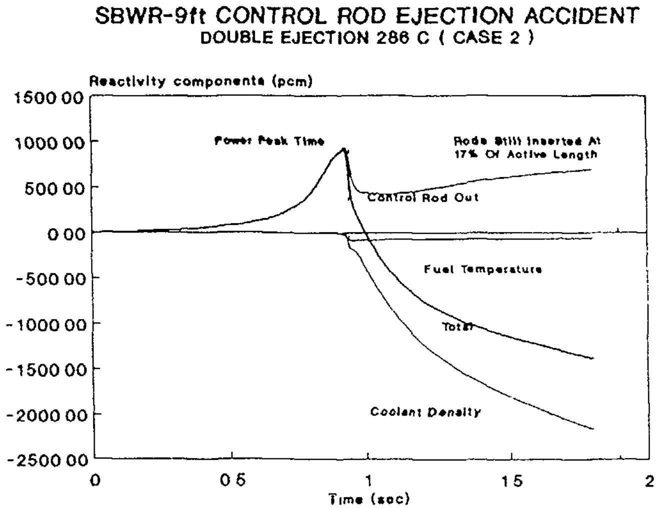

FIGURE 6

CREA was analyzed for the SBWR 9ft core configuration, during equilibrium fuel cycle at different combinations of system pressure / inlet subcooling, considering single and double ejections, assuming to have no scram

The results are summarized in Table 2 2, while Figures 5, 6 present some key data for the most severe case analyzed (Case 2 in Table 2 2)

CREA is a very mild event in fact subcooling is so low that voids come into play even before the power peak, the added reactivity is compensated essentially by void feedback (see Figure 6), while Doppler feedback is of secondary importance Fuel enthalpy is quite independent on the added reactivity and stays very low, with peak values of less than 40 cal/g

2.3 Refueling reactivity accident analysis

The rapid insertion or dropping of a fuel assembly into uncontrolled core locations could result in a reactivity excursion To preclude this event from occurring, refueling interlocks exist which require all control rods to be inserted prior to loading fuel in the core The administrative bypassing of such interlocks, to allow multiple control blade removal for maintenance, is permitted as long as several criteria, including removal of fuel assemblies surrounding the control rods to be withdrawn, are satisfied

The sequence of events leading to the accident [9],[10] requires that at least two control blades have been removed from the core and made inoperable The surrounding fuel bundles would also have been unloaded As part of the refueling (in violation of administrative procedures), fuel is then loaded back into the locations adjacent to the removed blades When the last fuel bundle is placed into the core, the reactivity insertion would be sufficient to cause a severe power excursion, scram would be impossible because control rods are still inserted or are inoperable

The analysis done by General Electric [10] for a standard BWR took into account two face adjacent withdrawn control blades, with a resulting fuel bundle worth of 6 25%, with the bundle inserted at the maximum fuel grapple speed of 20 32 cm/s Calculations were performed with an external coupling between a 2 dimensional (R Z geometry) adiabatic code and a thermal-hydraulic code to predict moderator feedback The calculated fuel peak pellet enthalpy (about 400 cal/g) was well above the fuel fragmentation threshold (280 cal/g) In view also of the relatively high frequency (somewhat greater than 10^{-6}/ry), this event was considered in Ref [2] the most severe reactivity accident for LWR's

Specific calculations were performed for the SBWR 9ft core configuration, at beginning of the equilibrium cycle, with the same hypotheses described above removed control rods and fuel grapple lowering speed

However, due to the large SBWR shutdown margin, the core is still subcritical withdrawing two face-adjacent control rods (60 °C) Therefore it was postulated that the refueling floor operator misplaces fuel assemblies, clustering the most reactive ones in and possibly around the two uncontrolled cells

Respectively with 4 and 10 misloadings, core is supercritical by 3 2% and 4 0% a number of errors larger than 10 is not possible, because otherwise the core would not be subcritical **before** the last fuel assembly is lowered

The dynamic analysis of the case with 4 misloadings was performed (60 °C) and the main results are shown in the figures 7, 8

The initial power peak, at 0 956 s, is 1896 MW, less than the rated power The added reactivity at this point is compensated essentially by Doppler feedback, continued bundle insertion adds more reactivity, which is compensated by the fuel temperature rise, while power stays essentially constant The reactivity addition rate is so low that heat conduction significantly contributes to the energy transport to the coolant Saturated boiling onset is at 1 714 s, void feedback quickly reduces reactor power to about 1 3% rated and fuel starts to cool-down peak pellet enthalpy is 95 cal/g After the complete insertion of the fuel bundle, the reactor stabilizes around 1% power level

The SBWR fuel loading accident is a relatively mild event no fuel cladding failure is expected, sensitivity studies are in progress to assess the effect of larger reactivity additions (up to the theoric maximum of about 4%) and reduced moderator temperatures, below 60 °C

In the most unlikely event of a bundle drop in one of the uncontrolled cells, concurrent with the misplacement of high-reactivity fuel assemblies herein, consequences would be of course much more severe since the reactivity addition is similar to that of CRDA with multiple RWM failures, the occurrance of fuel fragmentation and extensive fuel damage can be anticipated, but the expected frequency is negligible

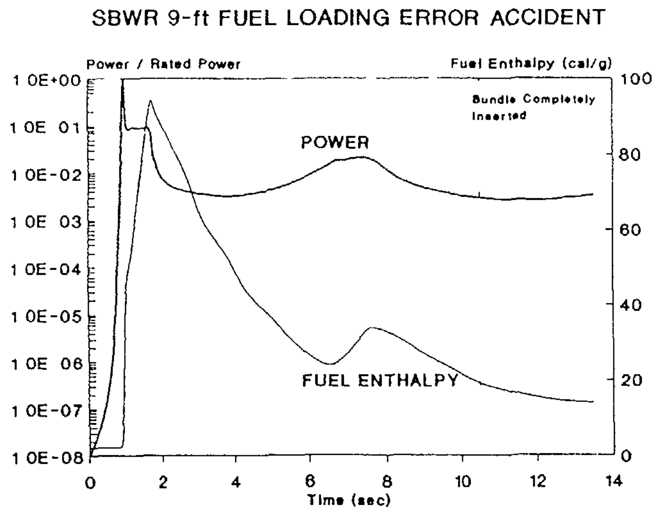

FIGURE 7

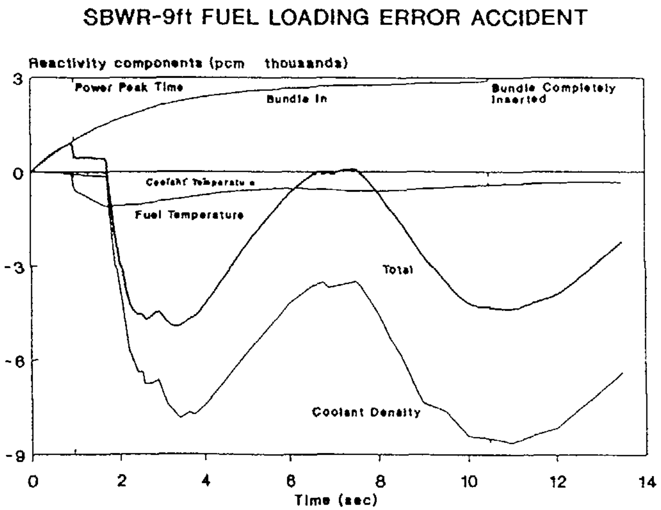

FIGURE 8

2 4 Uncontrolled control rod gang withdrawal

In the SBWR control rods can be withdrawn in a ganged way gangs in the first half of all rods to be withdrawn (control rod groups 1 4) include more than eight control rods by gang Various active features exist to mitigate the uncontrolled withdrawal event (rod block and scram due to short period high flux scram) however due to the low control rod withdrawal speed (3 cm/s) time would exist to compensate the reactivity addition by void feedback keeping fuel enthalpy very low It was evaluated (by point kinetics) that withdrawal of 50 % control rods which would bring the reactor supercritical by 3 2% (20 °C) would result into a core average enthalpy of about 30 cal/g

3 AP600 Reactivity events

An extensive analysis of RIA s will be undertaken for the AP600 plant in support to the PRA study in the same way as for the SBWR However for the time being only a preliminary survey of accident scenarios has been completed [11] mainly to individuate sequences worthwhile to be analyzed in detail by 3D space time calculations The limited and preliminary scope of this survey must be emphasized
PWR low power reactivity events can be grouped as follows [2]

i) Events due to the rapid removal of control absorbers
   1) Control rod ejection accident
   2) Boron dilution accidents during plant shutdown refueling or startup
   3) Continuous control rod withdrawal

ii) Events due to the insertion of fuel bundle (Refueling accidents)
   4) Fuel bundle loading accident
   5) Fuel bundle drop accident

iii) Events due to thermal hydraulic perturbations
   6) Steam line break
   7) Thermal hydraulic transients with positive moderator temperature coefficient (MTC)

Sequences 1) 2) 4) 5) are discussed in the next sections Sequences 3) 6) 7) represent extensions of events normally analyzed as part of plant s safety analysis The amount and rate of reactivity that can be potentially added is so low that no rapid fuel damage is expected as the result of such extensions [2] [11]

3 1 Control rod ejection accident

The rod ejection accident has always been considered to be the worst reactivity event in PWR s Extensions to the design basis accident that must be considered are sequences with multiple rod ejections or with excessive control rod worth due to violations of operational procedures
On the basis of recent sensitivity studies for standard PWR s [12] it can be evaluated that the simultaneous (within about 0 1 s) ejection of respectively two and three control rods would be needed to exceed the licensing limit (200 cal/g enthalpy in irradiated fuel) or to fragment the fuel (peak pellet enthalpy > 280 cal/g) respectively this conclusion will be however verified by specific dynamic analyses

3 2 Boron dilution accidents

A large number of event sequences leading to boron dilution can be devised
A first category include dilution from the chemical and volume control system (CVCS) by hardware failure and/or operator error or from diluted demineralized water from the purification system entering the core, during shutdown or refueling when all control rods are inserted and cannot be used to counter the criticality At the end of 1984 there had been 25 reported instances of boron dilution by CVCS during maintenance or refueling none of which resulted in criticality [2] The rate of reactivity addition is low so that criticality would result into coolant boiling and subsequent shutdown by void feedback The concern with this accident is therefore the possibility of core uncovery in particular during midloop operation in the French PWR PRA study its contribution to the core damage frequency was estimated $3x10^{-6}$/ry [13]

Though, taken per se, slow boron dilution could result at worst into core melt and not into rapid fuel damage it could be the initiator of a refueling accident, as discussed in section 3 3

Various event sequences leading to rapid boron dilution were identified in Ref [2] for standard PWR's The most significant were

3 2 1) Addition of diluted water, from accumulators or from the refueling water storage tank (RWST), during shutdown due to slow leakage or blowdown through single valve or inadvertant safety injection,

3 2 2) Large loss of coolant accident with diluted water from accumulators or RWST,

3 2 3) Startup of a reactor coolant pump after improper boron dilution

As for sequences 3 2 1 and 3 2 2 above, the frequency of a potentially significant event was > 10^{-7}/ry, taking into account the sensitivity to human error probability, only for the cases with inadvertant safety injection and for a LOCA with diluted accumulators [2] However, no thermal hydraulic calculation with boron transport was completed to verify if an unacceptable power excursion would occur, or if sufficient mixing with borated water would exist to prevent rapid fuel damage

Category 3 2 3) includes those sequences in which there is the restart of reactor coolant pumps after a period when boron dilution has been carried out

Swedish investigators [14] considered various initiating events in this category and in particular a dilution sequence that could occurr during recovery from a steam generator tube rupture the analyses (performed by a 3 dimensional model of boron transport in the vessel and by point kinetics to describe core behaviour) showed that the maximum supercriticality was about 7%, added in about 10 s, and resulted into a delta fuel enthalpy of 149 cal/g, that is without appreciable fuel damage

The incorrect startup of an isolated loop was analyzed by Finnish investigators [15] for a VVER plant (the six-loop Soviet-type PWR), by a flux-synthesis model, which accounts for the extreme asymmetry of neutron flux shape during the accident, in an approximate way Reduction of boron concentration by 660 ppm in one coolant loop, resulted into a peak pellet enthalpy of 234 cal/g, so that limited fuel damage occurs

Different event sequences in category 3 2 3 was considered in the French PWR PRA study [13], the most important one can be schematized as follows

- i) Normal dilution operation in progress,
- ii) Loss of offsite power supply,
- iii) Dilution operation goes on (backed up by auxiliary power supply),
- iv) Recovery of offsite power supply,
- v) Starting of primary pump #1 according to procedure

As result, a water plug, without boron and cooler than primary coolant, is pushed within the core The probability of this sequence was reduced to 1 $2x10^{-6}$/ry implementing an automatism that, on reactor pump trip, swithched make up pump suction on RWST and reset boron dilution [13]

Using neutron point kinetic approximation and assuming no water mixing, French investigators evaluated that, if core reactivity at accident conditions (HZP, all control rods inserted, primary temperature reduced by 30 °C) is larger than 5%, the peak fuel enthalpy would exceed the licensing acceptability threshold for RIA's (> 200 cal/g in irradiated fuel), which coincides with extensive cladding damage onset and is well below fuel pellet fragmentation threshold Such condition is satisfied in the first part of the fuel cycle of French PWR's [13]

However, it must be stressed *that analyses based on point kinetics are inconclusive*, due to the extremely skew flux profile and to the sensitivity of core reactivity to boron distribution

The limits of reactivity additions following postulated boron dilutions were estimated for each of the rapid boron dilution sequences 3 2 1, 3 2 2 3 2 3

On the basis of the comparison of reactivity balance, reactivity insertion rate, reactivity feedback and moderator subcooling, it was concluded [11] that the reactor coolant loop startup event would represent the bounding case

After having evaluated the specific AP600 plant design features that could impact or mitigate this event, detailed analyses will be performed to assess the resulting core damage

3.3 Refueling accidents

The possibility to initiate a reactivity accident loading a fuel assembly during refueling was not considered in Ref [2] for PWR's, as it was instead for BWR's A possible sequence of events is

i) Boron has been diluted for some reason, like for example CVCS malfunctioning, leakage from a diluted accumulator or RWST dilution

ii) The refueling floor operator misplaces fuel assemblies, clustering the most reactive ones and/or loads fuel assemblies without control rod however the reactor is still subcritical due to the presence of water holes in the core

iii) When the last fuel assembly is lowered into the core, a reactivity excursion takes place

Dynamic analyses will be performed to assess the accident severity in fact, if the fuel assembly is lowered by the *fuel grapple (fuel loading accident), the reactivity addition rate could be so low, to allow for enough heat transfer* and subsequent coolant voidage, avoiding in this way extensive fuel damage, even for large reactivity additions On the basis of the comparison with the SBWR case, it is expected that no less than 4 5% reactivity is needed to damage fuel

In the case of a bundle drop, concurrent with boron dilution and bundle misplacements 3% reactivity could be enough to fragment the fuel but the frequency for these sequence of events is expected to be insignificant

3 4 Benefits of burnable absorbers use

Current PWR core design employs 3 batches out-in loading scheme, with no burnable absorbers (BA) use in the equilibrium fuel cycle Extensive use of BA would offer instead several advantages

i) Increased fuel management flexibility, with the possibility to implement low leakage fuel loading schemes [16]

ii) More negative moderator temperature reactivity coefficient (MTC) at beginning of cycle, which will improve plant performance during ATWS heatup events (for example loss of normal feedwater, LONFW), usually the more severe ones in PWR's Sensitivity of peak system pressure, the limiting parameter, to MTC changes is very significant for standard PWR's a decrease of about 20 [17] 60 [18] psi per 1 pcm/°C MTC decrease (more negative) was evaluated

On the other hand, overcooling accidents, like steam-line break accident, will not be affected from this point of view, because bounding conditions are at end-of cycle, when MTC is approximately unchanged

iii) Mitigation, or even elimination by design, of boron dilution accident concern, due to the decreased core excess reactivity

iv) Mitigation, or even elimination by design, of refueling RIA concern, due to the reduced worth of high reactivity fuel bundles

Table 3 1

PARAMETER	STANDARD	BA CORE
FUEL DISCHARGE BURNUP (MWD/Kg)	42.55	42 28
MAX ASSEMBLY AVG POWER PEAKING FACTOR	1 27	1 39
MAX PIN POWER PEAKING FACTOR	1 42	1.57
CRITICAL BORON BOC HFP ARO EQUILIBRIUM XENON ppm	1361	921
CRITICAL BORON BOC CZP ARI NO XENON ppm	1260	643
EXCESS REACTIVITY BOC WZP ARO NO BORON NO XENON (%)	10 2	2 2
MTC BOC HFP ARO (pcm/°C)	23 1	35 6
MTC BOC HZP ARO (pcm/°C)	1.55	5 13
SHUTDOWN MARGIN BOC (%)	2 1	5 2
STRONGEST CONTROL ROD WORTH (*) BOC HZP (%)	2 3	1.5
MAXIMUM BUNDLE WORTH BOC CZP (%)	8 2	5.5

ARI All control rods inserted
ARO All control rods withdrawn
CZP cold (20 °C) zero power
WZP zero power moderator temperature reduced by 30 °C respect to HZP
BOC Beginning of cycle equilibrium samarium
(*) All but the strongest control rod inserted

STANDARD CORE
HOT ZERO POWER • NO BORON • NO XENON
CORE REACTIVITY

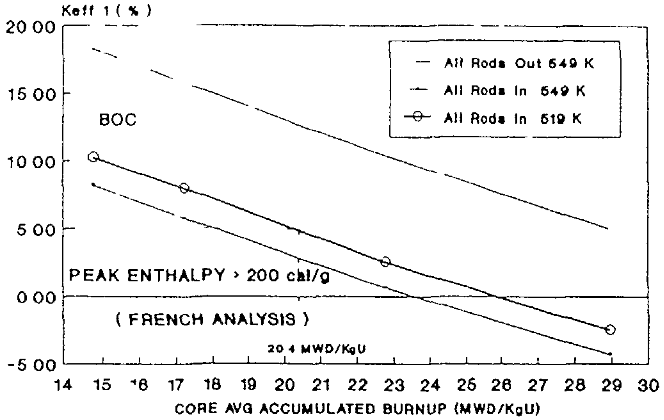

Figure 9

STANDARD CORE
COLD ZERO POWER • FUEL LOADING ACCIDENT
LAST LOADED FUEL BUNDLE WORTH

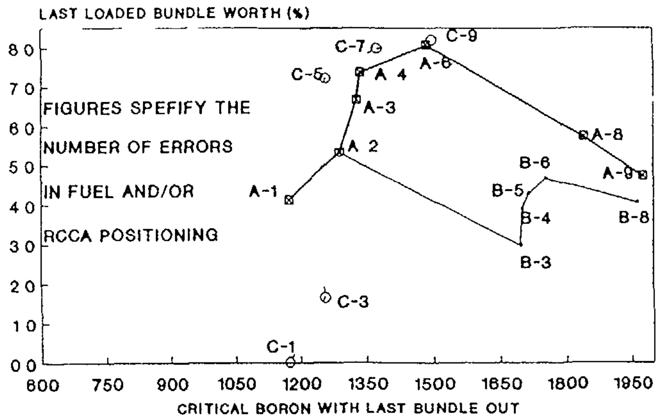

Figure 11

CORE WITH BURNABLE ABSORBERS
HOT ZERO POWER • NO BORON • NO XENON
CORE REACTIVITY

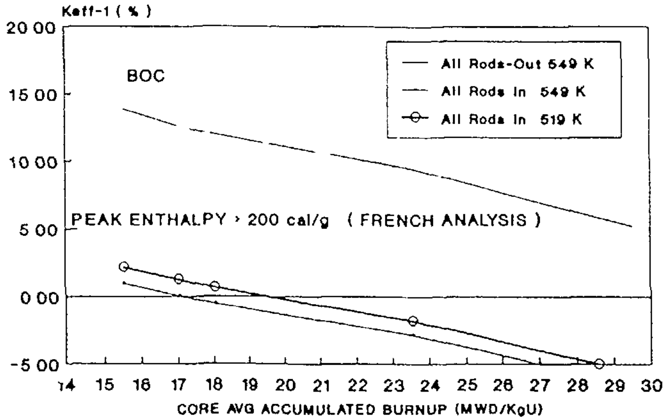

Figure 10

CORE WITH BURNABLE ABSORBERS
COLD ZERO POWER • FUEL LOADING ACCIDENT
LAST LOADED FUEL BUNDLE WORTH

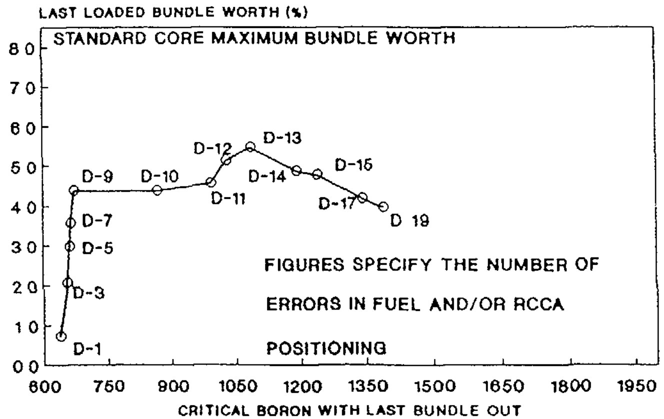

Figure 12

A conceptual study to help quantify the design configuration and performance parameters for a PWR with BA in the equilibrium cycle is currently in progress at ENEL Integral BA in the form of gadolinia doped fuel rods, were selected for this study, due to advantages over boron bearing BA in terms of increased fuel management flexibility, better control of intra assembly power peaking and elimination of additional storage and disposal
Only the equilibrium fuel cycle is modeled, by two dimensional calculations
Results are briefly discussed for an example case comparing a standard PWR core (18 months, 3 batches, in out loading scheme, no BA use) to an alternative core design employing a 3 batches in out in fuel loading scheme with 12 rods, doped with 7% Gd oxide in each fresh fuel assembly Table 3 1 compares some key results

Fuel discharge burnup is about the same in both cases indicating that the reduced radial leakage approximately compensates for Gd parasitic residual absorption (about 1% in this case)

The higher peaking factor of the BA core, by about 10%, will imply some loss of thermal margins however, preliminary estimates suggest that approximately 15% margin still exists to DNB Power distribution can be improved optimizing BA loading strategy, by the use of fresh fuel sub-batches with proper Gd loading, and improving the intra assembly peaking factor, by enrichment shaping or by alternative neutron poisons, like erbium [19]
Critical boron is reduced at BOC by about 440 ppm, which allows to get a significatively more negative MTC, both at HFP and at HZP Based on generic sensitivity studies quoted above [16],[17], a reduction of 240 720 psi in peak pressure after a LONFW ATWS at HFP could be expected, but, due to the great sensitivity to plant parameters, this should be verified by plant specific ATWS studies
Shutdown margin is significantly improved, by about 3%, which would reduce the concern for core recriticality after a steam-line break accident
The concern for rapid boron dilution accidents is essentially eliminated by design
Calculations, performed for the standard core with different boron / moderator temperature non-uniform distributions and with assumptions consistent to the French event sequence depicted in Section 3 2 indicated that

i) At beginning of cycle (BOC), the core can be brought to prompt-criticality by a relatively small, completely diluted, water plug (about 0 8 m^3)
ii) Neutron flux is of course extremely asymmetrical, with radial peaking factors in the range 7 19
iii) Plugs of 3 5-6 m^3 (25-45 % core water volume) can add 10% reactivity (BOC), more than 98% of all the available boron reactivity reactivity addition depends very significantly on boron spatial distribution At the rated loop flow, such plugs could be pushed into the core in less than one second
iv) If the plug is large enough (approximately more than 2 5 m^3), reactivity addition and flux distribution become fairly independent on the inital boron concentration, due to the low neutronic importance of the still borated core region

If the French criterion described in Section 3 2 is applied to the standard core, it indicates that, in the first part of the cycle, enough reactivity (WZP, ARI, no boron, no xenon conditions) is available to damage fuel (see Figure 9)
However, if we use the same criterion for the BA core, we can see that core reactivity is significantly below 5% along the whole cycle (Figure 10), so that no fuel damage is expected by the injection into the core of a cold, diluted water plug On the other hand, reactivity events involving addition of diluted water at CZP, ARI conditions (sequences 3 2 1) would require much higher dilution of accumulators or RWST down to about 640 ppm, which further decreases their probability
In order to assess the impact of BA on refueling accidents fuel bundle worths were evaluated for the two cores at beginning of the equilibrium cycle as a function of the number/ location of misloadeed bundles, number of water holes into the core and boron concentration For any given fuel bundle whose worth has to be calculated, the minimum possible boron concentration corresponds to the value that makes the reactor just critical **before** the bundle itself is loaded, such concentration, of course, maximizes core reactivity **after** the bundle has been loaded
Results from different misloaded bundle arrangements are summarized in Figure 11 (standard core) and 12 (BA core), where the last-loaded bundle reactivity is plotted versus the boron concentration defined above
The higher the number of misplaced bundles, the lower the boron dilution necessary to allow the accident in the standard core case, boron dilution is actually not needed at all if the number of misloadings is high enough With the nominal 2000 ppm boron concentration, the standard core could be made prompt critical lowering the last bundle if four fuel assemblies (including the last one) were misplaced The maximum reactivity obtained was slightly less than 5%, with 9 misloaded assemblies (including the last one)

The worth of a single misplaced bundle in the standard core is around 3 4%, depending on distribution of water holes, with critical boron concentrations of 1100-1200 ppm
The standard core fuel bundle worth is maximum at intermediate boron dilution / # of misloadings maximum worths of about 8% were obtained in some cases, with 6 9 misplacements and 1400 1500 ppm boron
Refueling reactivity accidents also would be significantly mitigated by BA use not only the maximum bundle worth is reduced to 5 5% (Figure 12) due to the reduced k infinite of the most reactive assemblies, but a larger boron dilution and/or number of misloadings would be required to initiate the accident

4 Conclusions

Low power reactivity accidents have been discussed for SBWR and AP600 reactors

In the SBWR case, control rod drop, control rod ejection and fuel loading accidents have been analyzed in detail the only identified possibility to have severe fuel damage are a control rod drop accident with RWM failures, during cold zero power nuclear testing, or fuel assembly drop, concurrent with multiple violations of refueling administrative procedures However, the frequency of such events is expected to be negligible Key inherent features, for the well behaved plant performance with respect to these accidents, are the reduced core excess reactivity, which significantly reduces control rod and fuel bundle worths respect to standard BWR's, and a revised startup procedure, which allows to achieve criticality with the moderator close to saturated conditions

No definite conclusion can be drawn at the present stage about the potentially more severe reactivity events identified for AP600, that is rapid boron dilution and fuel loading accidents, which still need to be investigated from both deterministic and probabilistic point of views Should such analyses identify any concern, it could be significantly reduced, or even eliminated by design, by the extensive use of burnable absorbers in the fuel

Appendix I

Some details about calculational models used are presented in this Appendix

Nuclear data libraries at rated and off rated conditions were generated using the CASMO 3 code [A 1]

SBWR steady state state core follow up, to obtain nodal parameters like fuel exposure and void history, was performed with the 3 dimensional, 1 5 groups, coarse mesh simulator CETRA [A 2]

PWR calculations were carried out with the 2 dimensional code MBS [A 3], which solves the 2-group neutron diffusion equation by finite difference tecnique assuming homogenized cross sections within each fuel assembly in this study each assembly was subdivided into 4x4 meshes

Dynamic analyses were performed with the QUANDRY EN code [A 4], which solves the steady state and time dependent two group neutron diffusion equation by the well known Analytic Nodal Method [A 5], including discontinuity factors The thermal hydraulic model is based on the COBRA 3C/MIT model and has the capability to treat coolant transverse cross flow (PWR s only) and boron transport

The present model simulates the reactor core only time dependent forcing functions are used for thermal hydraulic boundary conditions Constant values for inlet core flow and system pressure were used in all the analyses reported herein
QUANDRY EN has been benchmarked against Peach Bottom 2 turbine trip (BWR overpressurization transients) and SPERT IIIE (control rod ejection transients) tests

REFERENCES

[1] U S Nuclear Regulatory Commission "Implications of the accident at the Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States" NUREG 1251 V2 (April 1989)

[2] D.J Diamond C.J Hsu R Fitzpatrick "Reactivity accidents a reassessment of the design basis events NUREG/CR 5368 BNL NUREG 52198 (January 1990)

[3] D Mirkovic D.J Diamond "Effect of flushing boron during a boiling water reactor anticipated transient without scram Nuclear Technology 95 163 (1991)

[4] C J Paone R.C Stirn J.A Woolley "Rod drop accident analysis for large boiling water reactors NEDO 10527 (March 1972)

[5] L Fiorino "Evaluation of control rod worths in standard BWR s beyond licensing design basis accident scenarios ENEL/DSR VDN internal memo (February 1990)

[6] H Cheng D J Diamond "Analyzing the rod drop accident in a boiling water reactor" Nuclear Technology 56 40 (1982)

[7] P E MacDonald et al "Assessment of Light Water Reactor fuel damage during a Reactivity Initiated Accident" Nuclear Safety 21 5 (October 1980)

[8] C K Tang "SBWR startup plan" private communication

[9] R C Stirn "BWR Reactivity initiated accidents" presented to "BWR Safety Technology Workshop" Lugano Switzerland (April 7 10 1987)

[10] D C Cambra R.C Stirn "Fuel bundle loading accident Supplemental information to refueling interlock bypass procedure" General Electric memo CFSD-81299 (December 1981)

[11] E Fiorino "Preliminary survey of AP600 reactivity initiated accident scenarios" ENEL internal report (to be published)

[12] M Gonnet C Neaume T Bisiaux "Rod Ejection Analysis in PWR s with a 3D kinetic new code" in Proceedings of the Conference PHYSOR 90 Marseille France (April 1990)

[13] J Brisbois "Probabilistic Safety Assessment for French PWR 900 MWe Dilution Accidents" Presentation to U S Nuclear Regulatory Commission (March 5 1990)

[14] S.Jacobson "Some local dilution transients in a Pressurized Water Reactor" LIU TEK LIC 1989 11 Linkoping University Institute of Technology

[15] M.Antila R Kirky Rajamaki M Rajamaki H Raty P Siltanen T Vanttola "Application of the synthesis model in an asymmetric reactivity disturbance of the VVER-440 type Loviisa reactors" in Proceedings of the American Nuclear Society Topical Meeting on Safety of Thermal reactors Portland USA (July 1991)

[16] W Boehm H D Kiehlmann A Neufert M Peehs "Gd_2O_3 up to 9% w/o an established burnable poison for advanced fuel management in PWR s" in Proceedings of the Topical Meeting on Advances in Fuel Management Pinehurst USA (March 1986)

[17] W E Burchill "Anticipated transients without scram" in "Guidebook to Light Water Reactor Safety Analysis" Hemisphere Publishing Corporation (1985)

[18] B S Pei G P Yu G C Lin Y P Ma "Assessment of the safety function for the anticipated transient without trip mitigation system actuation circuitry at Maanshan nuclear power station" Nuclear Technology 90 49 (1990)

[19] H G Joo et al "A feasibility study of utilizing erbia burnable absorber in Korean PWR s in Proceedings of Topical meeting on Advances in Mathematics Computations and Reactor Physics Pittsburgh USA (April 1991)

[A 1] M Edenius A Ahlin B H Forssen "CASMO 3 A fuel assembly burnup program STUDSVIK/NFA 86/7

[A 2] G Alloggio E Brega R Guandalini M Mengoli "Outlines of the BWR-oriented ENEL Integrated Code System (EICS) CISE 5793 (May 1991)

[A 3] O Norinder E Jonsson K Olsson "MBS a 2 D core analysis diffusion theory code MBS user s manual version 1 7 STUDSVIK/NR-85/80 (1985)

[A 4] G Alloggio F Brega L Salina QUANDRY EN a computer code for the neutronic and thermal hydraulic space and time dependent analysis of Light Water Reactor cores by advanced nodal tecniques (Rev 3) Report in progress

[A 5] K S Smith "An Analytic Nodal Method for solving the 2 group Multidimensional Static and Transient neutron diffusion equations Nuc Eng Thesis Dept of Nuc Eng MIT Cambridge MA (February 1979)

ADVANCED BOILING WATER REACTORS

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Abstract

Power distribution, fuel cycle, void reactivity and the first insight of the thermal-hydraulics problem of an Advanced Boiling Water Reactor were analysed. The calculations have showed that it is possible to obtain a long fuel cycle, with a 3% Plutonium enrichment. The moderator void coefficient, even in a complete voiding of the reactor, is negative. The thermal-hydraulic calculations have demonstrated that it is necessary to reduce the coolant flow rate in order to maintain the same pressure drop as in a reference reactor.

INTRODUCTION

Studies have demonstred that it is possible to increase the conversion ratio of the Pressurized Water Reactor near to 1 [1,2,3]. These reactors,called Advanced Pressurized Water Reactor(APWR), have been searched for almost two decades and the R & D are still going on. The main characteristics of these reactors is the hexagonal core structure with a low moderator to fuel volume ratio. The main physical effect of this change is a shift of the neutronic spectrum from the thermal region to the high energy region, increasing thus the resonance capture in the fuel fertile materials and consequently the conversion ratio.

Boiling Water Reactors present in the average harder neutronic spectrum than that of the Pressurized Water Reactors. If the geometrical change done in the PWR core were considered in the BWR core, the effect should be much more enhanced and the conversion ratio also should go up. Based on such concept we have studied the feasibility of an "Advanced Boiling Water Reactor" (ABWR).

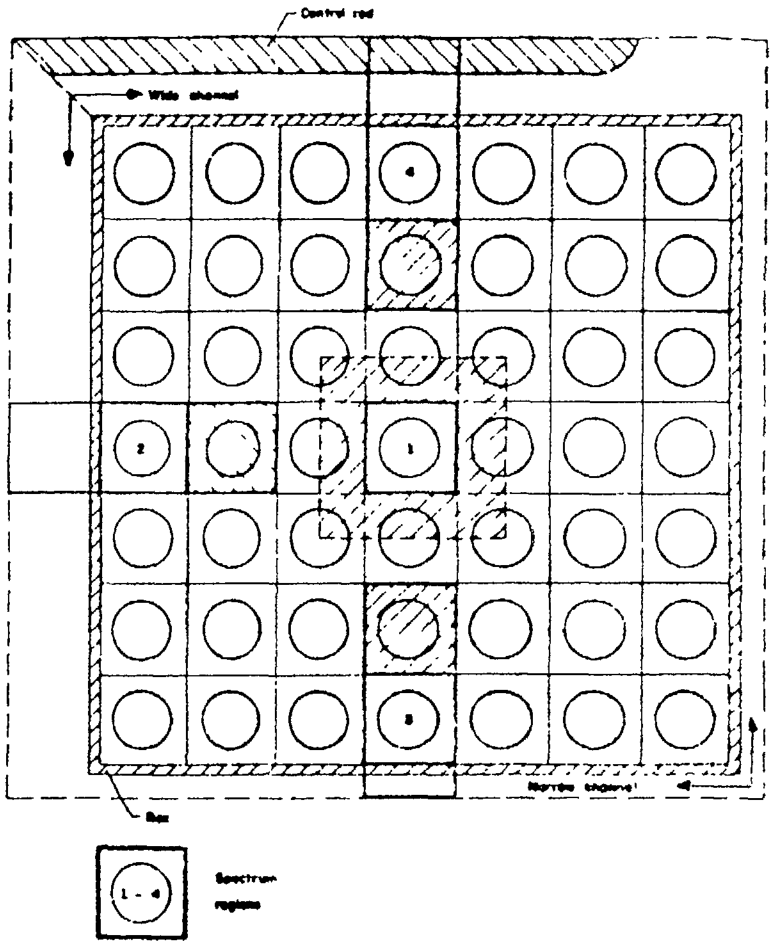


FIG. 1. BWR fuel bundle cell.

Parametric studies[4,5] has shown that the physics of the ABWR is feasible. From this study we selected two basic conceptions, named as Alternative 1 and 2. This paper deals with the calculation of the power distribution, fuel cycle, reactivity control and the first insight on the thermal-hydraulic problems of the ABWR-Alternatives.

METHODS and CODES

The calculational methods employed in this study are briefly described herein. The calculation was done with the RSYST-program[6]. The BWR fuel bundle cell is shown in Figure 1. The cell was divided in four typical regions. For each region the neutron spectrum generated was condensed to build up a basic 60-groups library. this 60-groups structure covers the fast and the thermal energy ranges. This library with 30 - fast energy groups (14 MeV - 1.86 eV) was generated with the GAM code and the 30 - thermal energy groups (1.86 - 0 eV) was generated with the THERM code. The group structure was strongly oriented toward the Plutonium and U - 238 resonances. The group constants were generated for a 4 % mixed UO_2-PuO_2 fuel for a volumetric moderator to fuel ratio of : 2,0 ; 1,0 ; 0,6 and 0. The collapsed spectrum was generated for three temperatures : 20 ºC 280 ºC and 663 ºC.

THE REFERENCE CONCEPTION

From the parametric study[4] we selected two basic conceptions for the ABWR reactor. Figure 2 shows the fundamental properties of PuO_2-UO_2 BWR-lattice as a function of the moderator fraction in the core. For the three different enrichments of fissile Plutonium (Pu-239 and Pu-241) the infinite multiplication factor and the conversion ratio are plotted against the effective moderator to fuel volume fraction. By reducing the moderator volume, the multiplication factor decreases.

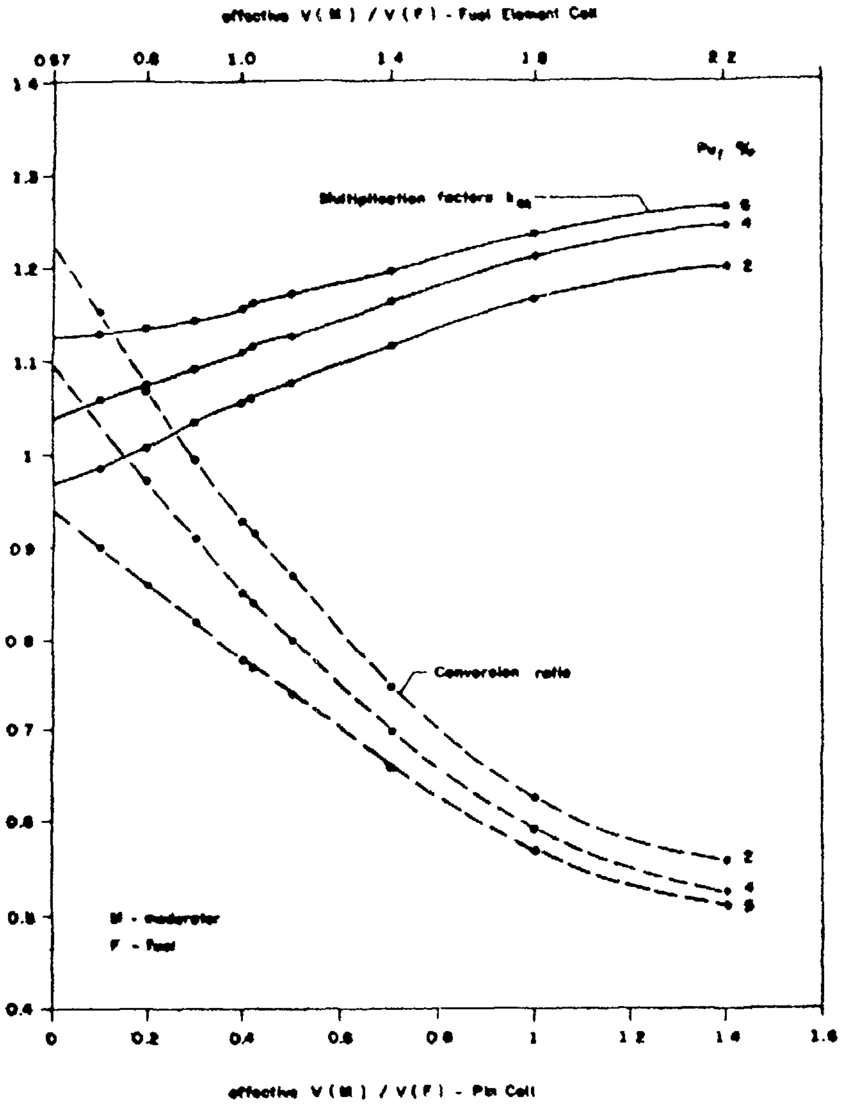
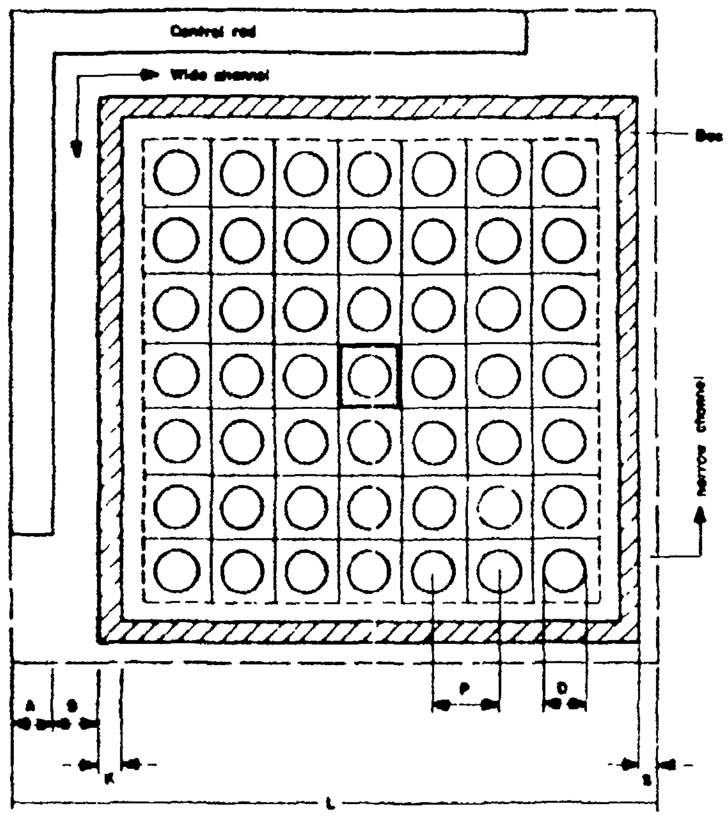


FIG. 2. K∞ and conversion ratio in a UO_2–PuO_2 fuel bundle lattice



Dimens. (mm.)	A	B	D	K	L	P	S
alternative 1	3.95	8.62	14.[illegible]	2.1	128.5	15.3	4.7
alternative 2	3.96	8.62	14.[illegible]	2.1	131.3	15.7	4.7

FIG. 3 Geometrical parameters of ABWR alternatives.

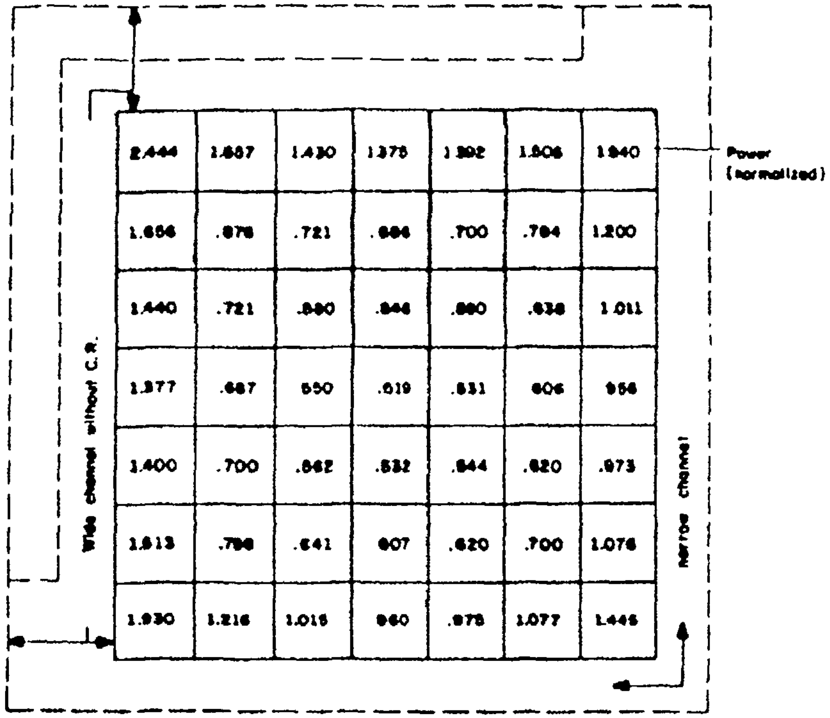
Table 1. Basic Data Constitutive for ABWR Alternatives

Alternative	1	2
. Fuel rods/Element	7 x 7	7 x 7
. $V(H_2O)/V(fuel)$ - Cell	0,6	0,7
. $V(H_2O)/V(fuel)$ - Element	1,3	1,7
. Fuel rod diameter (mm)	14,3	14,3
. Pin pitch (mm)	15,3	15,7
. Core active lenght (mm)	2300,0	2300,0
. Fuel (mixed oxide)	UO_2 - PuO_2	UO_2-PuO_2
. Plutonium enrichment (%)	3,0	4,0
. Average Burnup (MWd/kg)	50,0	-
. Conversion ratio	0,88	0,80

The conversion ratio goes up because of the increased resonance absorption. Figure 3 shows the geometrical configuration of the ABWR basic fuel bundle. Table 1 shows the main parameters of the two concepts extracted from the parametric study. Forthcoming we present the power distribution, fuel cycle, reactivity control and moderator reactivity coefficient for the first ABWR Alternative.

POWER DISTRIBUTION

Firstly the power distribution was calculate in an homogeneous core with 4 % Plutonium enrichment. Figure 4 shows the homogeneous power distributi-n The outer fuel rods in the Fuel Assembly have a high power factor than the inner rods. This non-uniform power dis tribution is caused by the soft neutronic spectrum due to the wide moderator channel in the periphery of the Fuel Assembly.

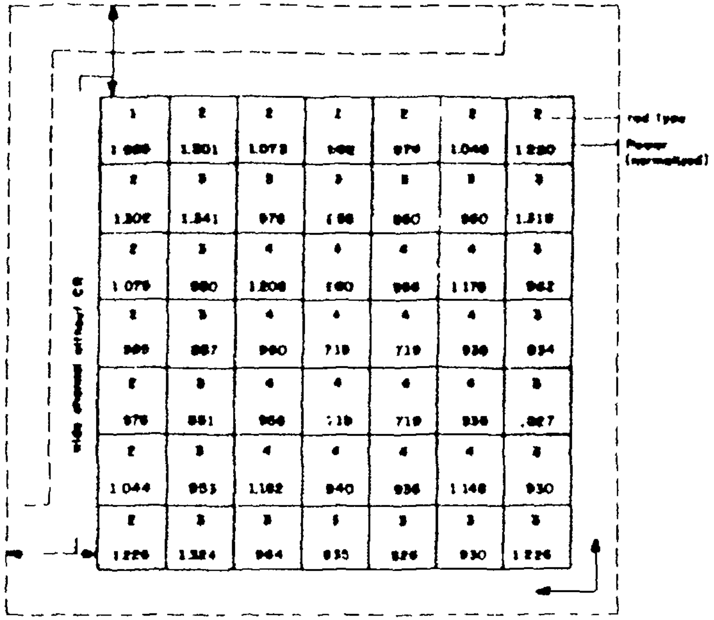


Keff = 1.07647

CR = 0.87

FIG. 4. Power distribution (PuO_2 : 4%)

To reduce the power factors the calculation was done for a heterogeneous core with a four different Pu - enrichments. Figure 5 shows the power distribution the effective multiplication factor and the conversion ra tio. We can see that the inner rods even with a high Pu-enrichment have a lower power factor than the outer rods. The cause for that is the inner region neutronic spectrum being more energetic than the outer region neutronic spectrum. The power distribution was accepted once the power factor is lower than the permited value.



rod type	# of rods	Pu$_f$ %
1	1	6
2	12	1.0
3	20	2.0
4	16	4.0

$\bar{P}u_f$ = 3.0 %

K_{eff} = 1.08612

CR = 0.88

FIG 5 Power distribution (distributed enrichment)

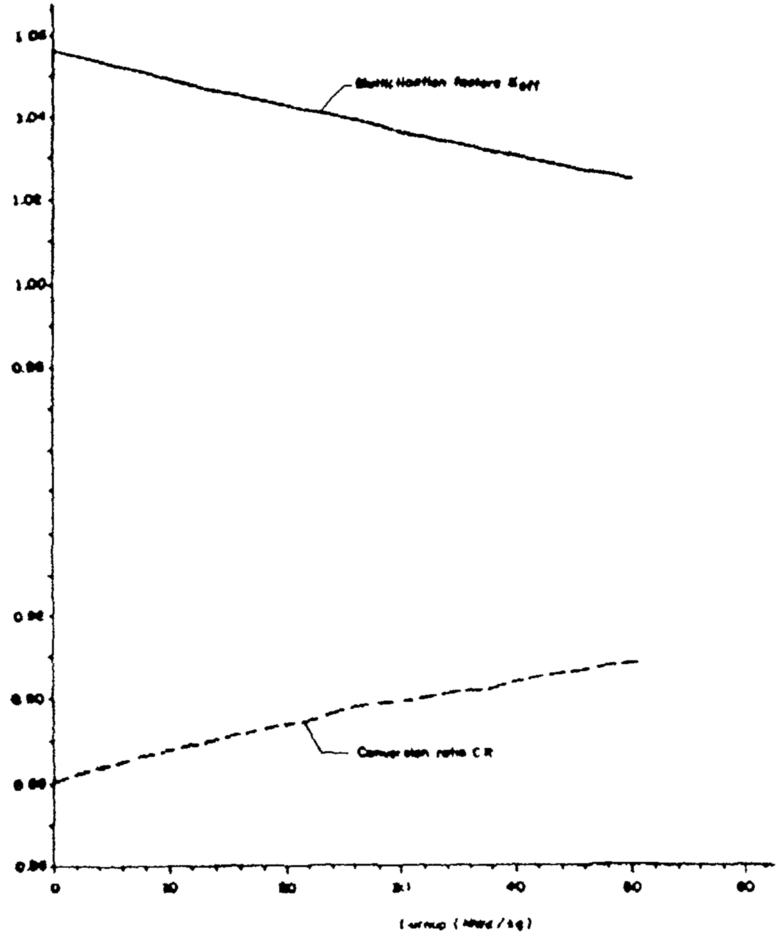


FIG 6 K_{eff} and conversion ratio vs burnup

FUEL BURNUP

The burnup characteristics for an heterogeneous core with a 3% average enrichment is shown in Figure 6. From this Figure we can see that with a 6% reactivity it is possible to have a long fuel cycle higher than 60 MWd/kg. Calculations show that it is possible to reduce the average Pu-enrichment in order to increase the conversion ratio.

VOID REACTIVITY

One of the most important criteria to evaluate the safety of power reactor is the behavior of the moderator void coefficient in case of loss of coolant accident. Figure 7 shows the void characteristics of an ABWR for three different enrichments of the fissile

Plutonium. In the calculation of the loss of coolant it is demonstrated by a reduction of moderator to fuel volume ratio. We can see that the reactivity has a clear negative tendency even for a high Pu-enrichment. These results point out that the physics of the ABWR is fea sible.

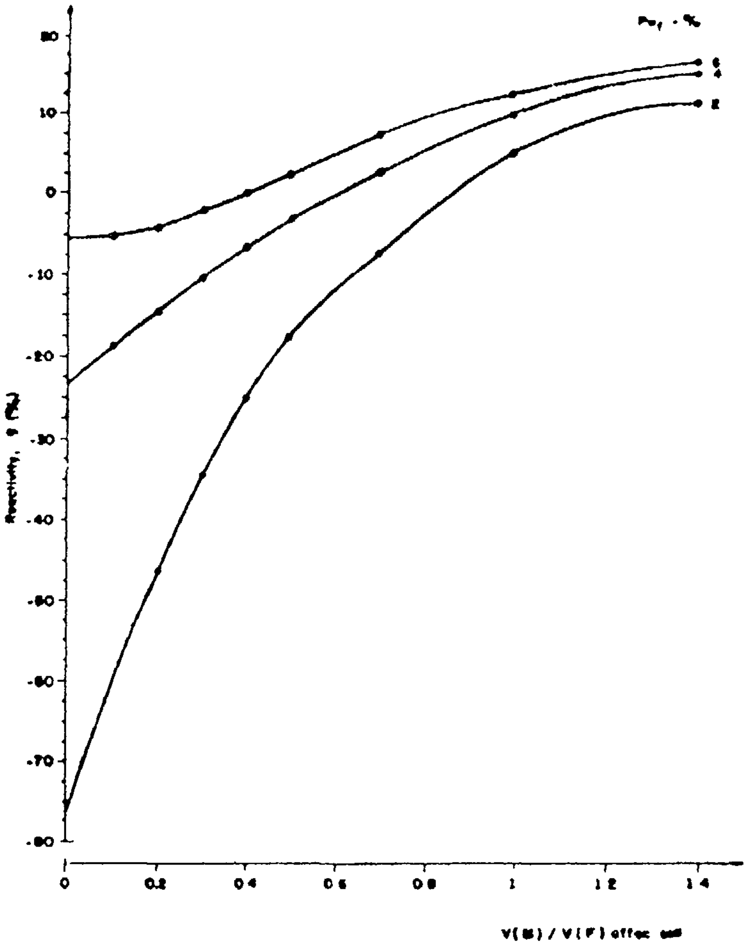


FIG 7 ABWR reactivity characteristics (central pin cell)

Table 2. Control rod characteristics

. Type	cruciform
. Material	s.s.
. Lenght	104,84 mm
. Wing size	7,90 mm
. Wing thickness	1,38 mm
. Absorber active lenght	2300,00 mm
. Absorber rod diameter	4,70 mm
. B_4C - density	1,76 g/cm^3
. Absorber rods/wing	15

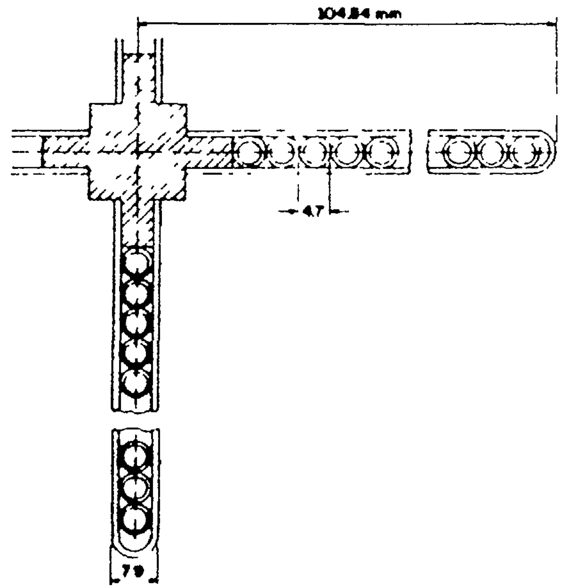


FIG. 8. Control rod

REACTIVITY CONTROL

The reactivity for the ABWR can be controlled by the BWR-conventional control rods. The control rods were fitted to the new geometry of the fuel element cells. Figure 8 shows the control rod structure and Table 2 lists the main characteristics.
The calculation has been done for the cold condition (worse) and hot condition(operational). The cold condition is interesting since the absorber worth has a minimum value. Figure 9 shows the effective multiplication factor as a function of the rod insertion for cold and hot conditions. The efficacy of the control rod is strong since for a 70% insertion the subcriticality is 4% for the hot condition. In comparison with the cold conditions, this subcriticality will be reached only with a 100% of insertion. With a fully inserted control rod the reactivity is 13,6% for the hot condition. This reactivity is enough for a safety shutdown margin, calculations uncertainties and manufacturing tolerances.
A first evaluation for the ABWR reactivity is presented in Table 3.
Some data was taken from a 770MWe BWR reference reactor. This assumption is conservative since the reactivity for Advanced Light Water Reactor is smaller than that for normal Light Water Reactors. Table 3 shows that the

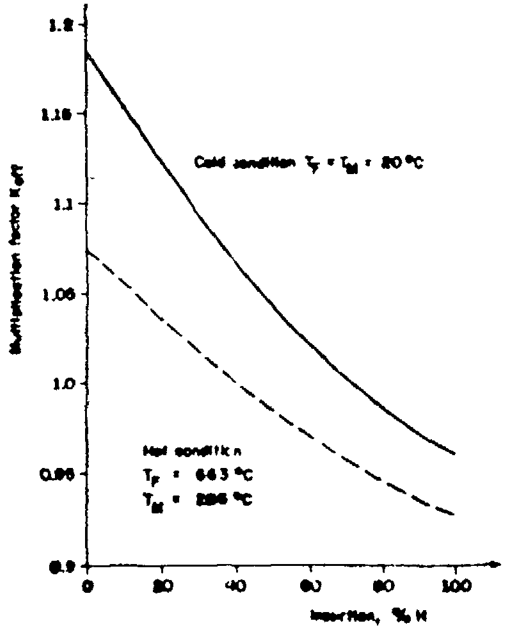


FIG. 9. K_{eff} vs control rod insertion.

Table 3. Reactivity balance

		BWR	ABWR
k_{eff}	, cold, clean	1.2140	1.1879
Δk	, cold - hot	+0.0100	+0.0100
Δk	, Doppler	-0.0070	-0.0070
Δk	, Moderator	-0.0230	-0.0230
Δk	, Xe, Sm	-0.0330	-0.0330
Δk	, Burnup	-0.1560	-0.0400
Δk	, Trim	-0.0050	-0.0050
Δk	, Total	-0.2140	-0.0680
Δk	, Control rod, cold	-0.1740	-0.2260
Δk	, Gd , cold	-0.1110	-
Δk	, Total(full controlled)	-0.2850	-0.2260
K_{eff}	, cold, clean (full Controlled)	0.9290	0.9620

reactivity equivalent to the rapidly satured fission products is one order of magnitude smaller than in the traditional BWR. The reactivity reserve for a long fuel cycle, almost 60 MWd/kg, is a factor 4 lower than that in the normal BWR reactor. Another interesting point in Table 3 is the lower reactivity reserve for the ABWR (6,8%) in comparison with the normal BWR(21,4%). It is advised that the control rod calculation presented here should not be taken as a final one but only as a first insight into the problem.

THERMAL-HYDRAULICS

The subchannel analysis of the ABWR core were done with a well-known code COBRA-4I [7]. The presented evaluation of critical heat flux is valid only as a first approximation for ABWR thermal-hydraulics conditions.

The calculation was performed for the following pairs of mass flow rate and heat flux:

1. $q'' = 48.8\ W/cm^2$, $\dot{m} = 15.75\ kg/s$ (reference BWR)
2. $q'' = 48.8\ W/cm^2$, $\dot{m} = 8.0\ kg/s$ (ABWR)
3. $q'' = 58.6\ W/cm^2$, $\dot{m} = 8.0\ kg/s$ (20% overpower)

The point 1 are the actual parameters of a conventional BWR data. Point 2 differs from 1 in the mass flow rate since this is the minimum mass flow rate to reduce the core pressure drop. Values under this limit will generate unacceptable core void fractions with a possible low safety margin for a DNBR. Point 3 simulates 20% overpower. Figure 10 shows schemactic of the core channels, gridspacers positions and the axial power distribution.

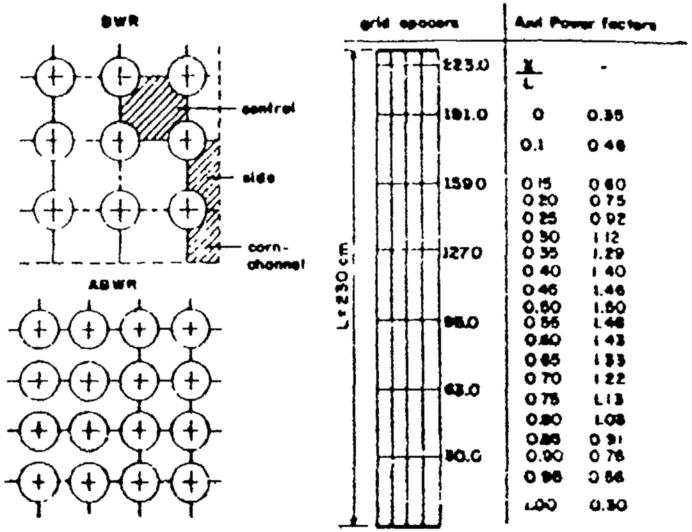


FIG. 10. Grid spacer positions and axial power factors.

RESULTS

a. PRESSURE DROP

The tight fuel element lattice to the ABWR lead to a expanded core pressure drops because of the 50% reduced coolant flow cross section. Figure 11 shows the ABWR-core pressure drop. For comparison it was included in the Figure the reference BWR pressure drop (dotted line). The vertical dotted line indicates the grid spacer contribution to the core pressure drops.
To keep the ABWR-core pressure drop in the order of the reference BWR pressure drop, the coolant flow rate must be reduced to a very low value. As exposed the coolant flow rate limit is 8.0 kg/s. At this point Alternative 2 has almost the same core pressure drop. For Alternative 1 the core pressure drop is 60% higher than the reference pressure drop. Fortunately at the BWR only

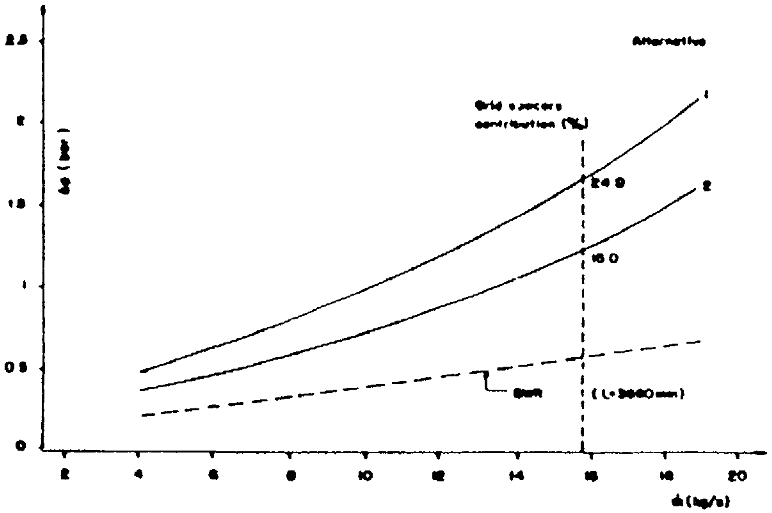


FIG. 11. ABWR core pressure drop.

a small portion of the total pressure drop of the primary circuit ocurs in the reactor core. Therefore if necessary the mass flow rate can be increased, e.g.,to 10.0 kg/s, and the core pressure drop will be less than 1.0 bar.

b. TEMPERATURE DISTRIBUTION

The axial temperature distribution for the hot rod of Alternative 1 is show in Figure 12. For comparison the axial temperature distribution of the BWR-reference reactor(dotted line) was included. At the nominal condition for the ABWR(point 2) the fuel central temperature and the clad surface temperature is lower than the temperature at the operational point in the reference BWR reactor. Even with a 20% overpower and 50% reduction in core coolant (8.0kg/s) the ABWR central fuel temperature is 45ºC higher than the fuel central temperature in the reference BWR.

The calculations show that the maximum fuel central and clad surface temperatures in the ABWR conception is far down the safety limits.

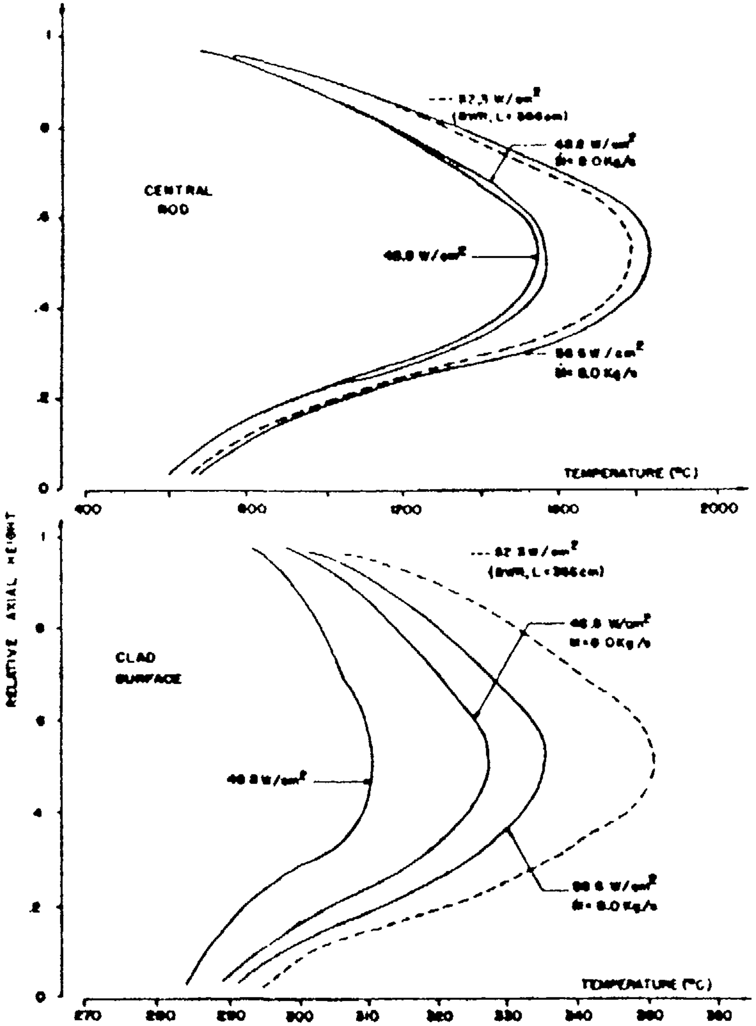


FIG. 12. Temperature distribution (alternative 1).

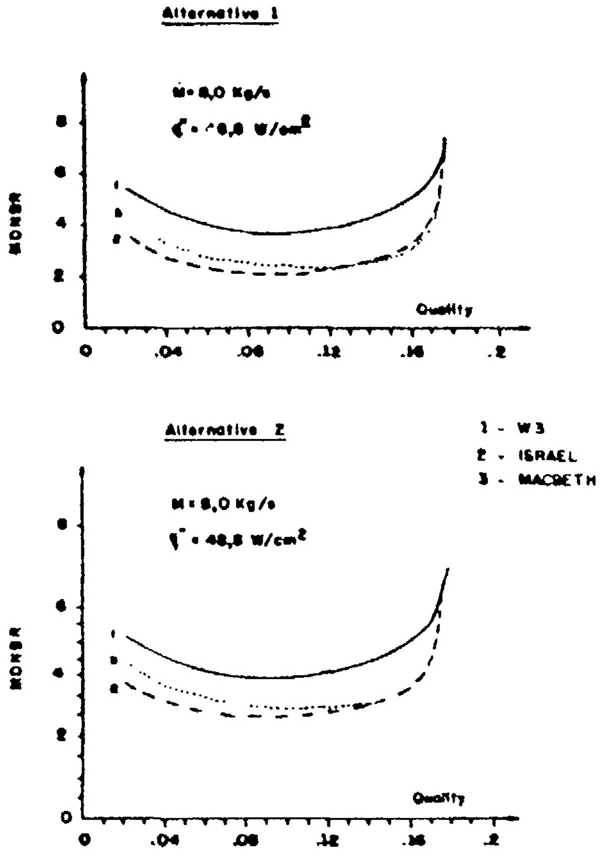


FIG. 13. MDNBR vs quality.

c. CRITICAL HEAT FLUX

The critical heat flux was calculated with the well-known W-3, MacBeth and Israel correlations[8]. In the calculation the minimum DNBR was searched for each fuel rod, coolant channel and axial height. The thermal-hydraulics limits imposed for safety operational conditions were:

- maximum heat flux 127 W/cm²
- maximum linear heat rate 570 W/cm²
- minimum DNBR (MDNBR) $>$ 2

Figure 13 shows the MDNBR versus quality for Alternative 1. The minimum DNBR is larger than the safety limits ($>$ 2). Nevertheless this should not be taken as an absolute value, but only as an indicative. The critical heat flux evaluation must be done with appropriate correlations and further experimental rigs.

CONCLUSIONS

The results presented were based on many conservative assumptions, however we found out some conclusions about the physics and the thermal-hydraulics to set up the basic conception for an Advanced Boiling Water Reactor:

- From the parametric studies we selected two basic conceptions (Alternatives) with conversion ratio of the order of 0,8.
- It is possible to have a long fuel burnup (50 or 60 MWd/kg) with a low reactivity (3 or 4%).
- The conventional BWR-control rod seems to be enough to control the reactivity of the ABWR.
- The moderator reactivity coefficient even with complete core voiding is negative.
- To keep the core pressure drop in the same order of that in the reference BWR, the coolant flow rate should be reduced to about 8,0 kg/s (50% reduction).
- The fuel central and clad surface temperatures of the hot rod is far from the safety limits.
- The minimum DNBR, even with a low mass flow rate is higher than the safety limit.

Finally, the present study must not be understood as an affirmation to the feasibility of the ABWR, however it can be the base to estabilish a detailed program to consolidate the conception of an Advanced Boiling Water Reactor.

REFERENCES

[1] M.C. Edlund, Physics of the Uranium-Plutonium Fuel Cycle in Pressurized Water Reactors. Trans. Am. Nucl. Society, 24, 508 (1976)

[2] Penndorf,K., Schult,F., Buenemann,D., "Abschaetzung der wesentlichen neutronenphysikalischen Eigenschaften hochkonvertierender DWR in U/Pu Zyklus" Proc. Jahrestagung Kerntechnik 80,Berlin,FRG,1980.

[3] Dalle-Donne,M., "Fluid-and Thermodynamics Calculations for Advanced Pressurized Light Water Reactor with a Tight Fuel Rod Lattice", Proc. Jahrestagung Kerntechnik, Duesseldorf,FRG,1981.

[4] Rodrigues,V.G., Stegemann,D., "Consideracões Neutronicas para o Desenvolvimento de Reatores BWR Avançados" VII-ENFIR : Brazilian Meeting on Reactor Physics and Thermal Hydraulics, Vol 1, Recife-Brazil,1989.

[5] Rodrigues,V.G., Stegemann,D., "Análise Termoidraulica de Reatores BWR Avançados" VII-ENFIR : Brazilian Meeting on Reactor Physics and Thermal Hydraulics, Vol 2, Recife-Brazil,1989.

[6] Ruehle,R., "RSYST, ein integriertes Modul system mit Datenbasis zur automatisierten Berechnung von Kernreaktoren" IKE-Bericht N.4-12,1973.

[7] COBRA-IV-I : An Interim Versions of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores. BNWL-1962,UC-32 Battele-Pacific Northwest Laboratories,1976.

[8] Two-Phase Flow and Heat Transfer in Rod Bundles. ASME-Winter Annual Meeting(1969).

SOME THERMAL HYDRAULIC STUDIES RELATED TO INDIAN AHWRs

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Abstract

Design of an Advanced Heavy Water Reactor (AHWR) is in progress in India to enable effective thorium utilization. The proposed system envisages a pressure tube type of heavy water moderated reactor with vertical channels using boiling light water as coolant. The reactor core consists of Th-U233 fuel driven by plutonium-uranium mixed oxide driver assemblies. As part of the core design exercise, preliminary thermal hydraulic analysis of the core has been carried out. Core flow distribution, among other parameters, is largely dependent upon hydraulic resistances at channel inlets. This, in turn, influences the steam quality at channel exit. Analysis has been carried out for different values of inlet resistances. One of the passive safety features proposed to be incorporated in the AHWR is to maintain the flow of coolant through the core by natural circulation. Two phase thermosyphon analysis has been carried out to study the effect of variation of loop height and core power on core flow. Details of the analyses carried out along with the results are discussed in the paper.

1.0 Introduction

The first phase of the Indian Nuclear Power Programme envisaged the installation of Pressurised Heavy Water Reactors (PHWRs) with natural uranium fuel. This phase is well under way with five units operating, three units nearing completion and several others in various construction stages. Under the second phase a Fast Breeder Test Reactor has been commissioned and the design of a Prototype Fast Breeder Reactor (PFBR) is in an advanced stage. An important objective of this phase of the programme and the third phase which envisages the installation of

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U233 fuelled reactors is the effective utilisation of the vast reserves of thorium in India The thorium reserves in India are estimated to be around 363,000 Te [1] The Advanced Heavy Water Reactor (AHWR) being designed in India is aimed at extracting sizeable fraction of energy from thorium with minimum consumption of plutonium The reactor core of the proposed AHWR thus consists of Th-U233 fuel driven by Plutonium-Uranium driver assemblies The proposed system envisages a pressure tube type of heavy water moderated reactor with vertical channels using boiling light water as coolant Some of the desirable design features are i) effective utilisation of thorium with minimum fissile material consumption, ii) use of light water as coolant instead of heavy water to reduce capital as well as running costs, iii) achieving a negative void coefficient of reactivity and iv) incorporation of passive safety features.

One of the passive safety features proposed to be incorporated in the AHWR is the natural circulation of coolant through the primary loop. The thermal hydraulic studies being carried out on the proposed AHWR, as part of the preliminary investigations being conducted to establish its feasibility, include analysis of the core comprising of two types of fuel assemblies and also two phase thermosyphon analysis of the primary loop. Details of the system, analyses carried out and the results obtained are discussed in the following sections.

2.0 Primary Heat Transport (PHT) System

A schematic of the proposed system is depicted in fig.1 The aim is to maintain the flow of the coolant through the PHT system by natural circulation From the inlet header , the light water coolant enters the core, comprising of vertical fuel channels, through feeder pipes and lower channel extensions Fuel channels are housed in the calandria filled with heavy water moderator Steam-water mixture from the fuel channels is led to the outlet header via the upper channel extensions and feeder pipes. From the outlet headers, steam water mixture flows through the risers to the steam drums Separators inside the steam drums separate steam from water and the steam goes to the turbine Feed water enters the system through the steam drums and the subcooled water from steam drums flows to Secondary Steam Generator (SSG). Heat is rejected to the secondary side of the secondary steam generators before the coolant enters the inlet header. An isolation condenser (not shown in the figure) is provided to remove decay heat when the reactor is shut down.

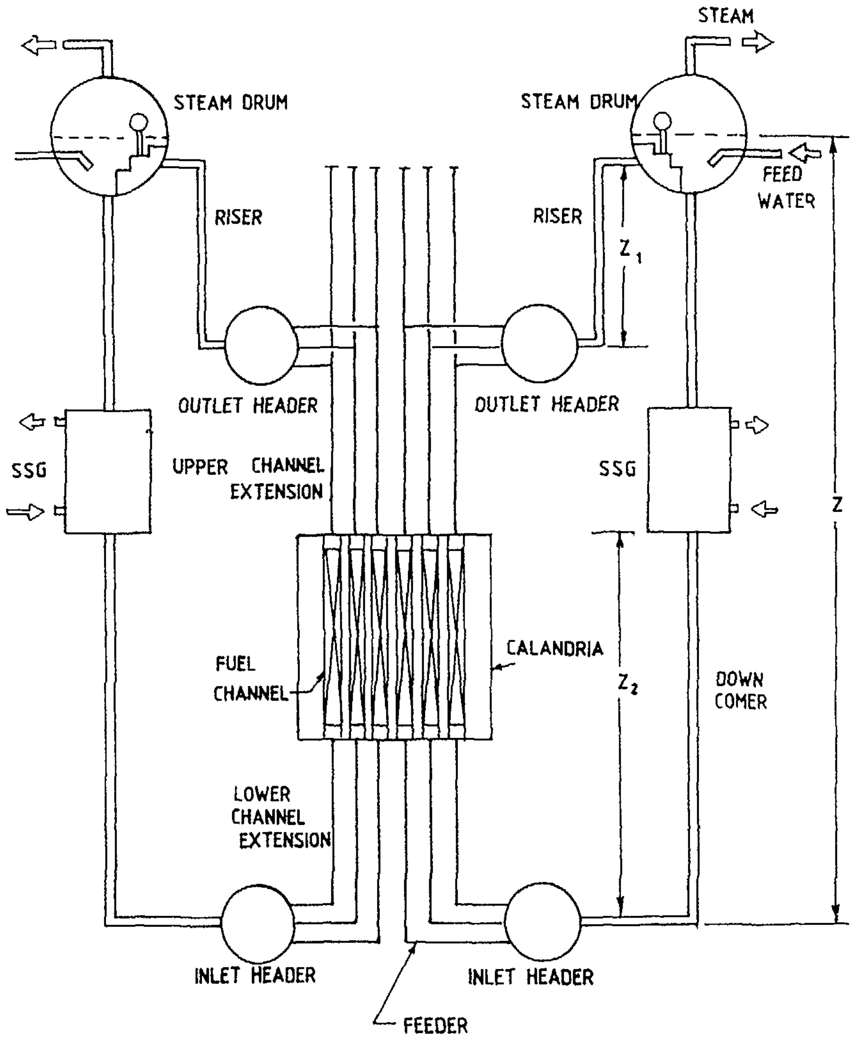

Fig 1 SCHEMATIC OF AHWR PRIMARY HEAT TRANSPORT SYSTEM

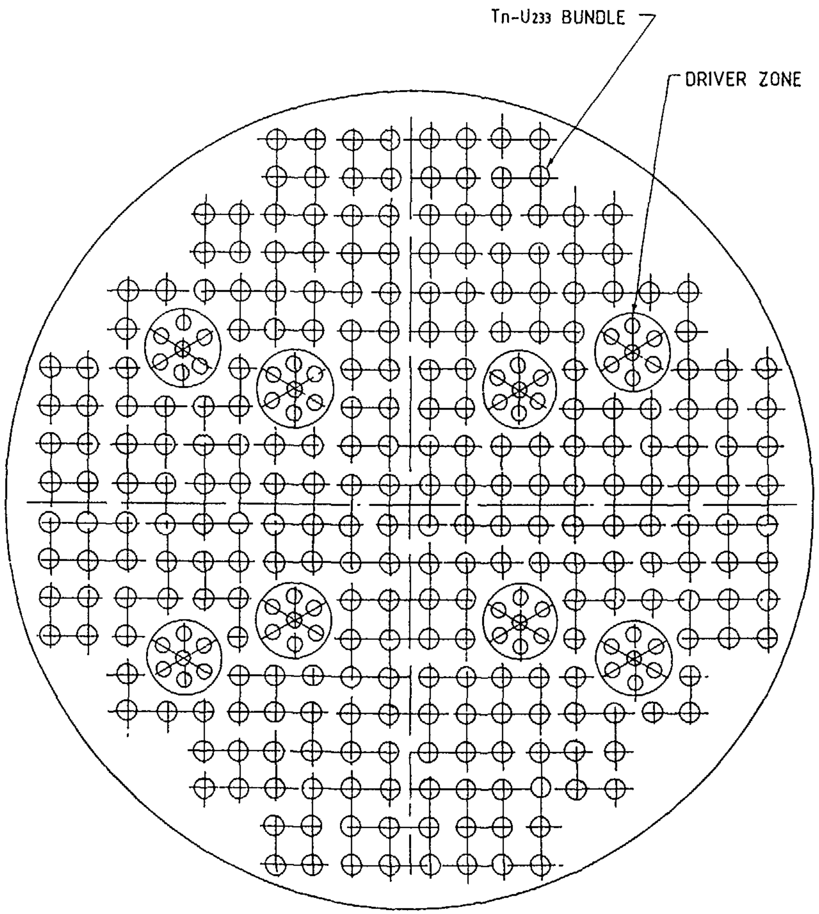

Fig.2 CORE CONFIGURATION

3.0 Reactor Core

The major part of the core consists of thorium oxide fuel clusters enriched with U233. However, this lattice is designed to be some what sub-critical [2]. The additional reactivity is provided by driver zones consisting of mixed plutonium-uranium oxide fuel. The boiling light water functions both as moderator and coolant in this zone. The voiding of this zone and consequent removal of moderator results in an overall negative void coefficient of reactivity for the core even though the Th-U233 zone has a small positive void coefficient of reactivity. As part of the feasibility investigations, a number of core configurations are being studied. The core configuration considered for the present study is depicted in fig.2 [3]. The core consists of 328 fuel channels out of which 272 channels have Th-U233 fuel and the rest have PuO2- UO2 fuel. Locations of the eight driver zones are shown in fig.2. Seven fuel channels having PuO2-UO2 fuel are located in each of these eight zones. Each of the fuel channels in both the zones houses a 37 rod fuel bundle. Each of these bundles has 36 fuel rods and a central structural rod. Ten spacers and two tie plates have been considered for the present analysis.

4.0 Core Thermal Hydraulic Analysis

The hydraulic circuit of the core consists of a number of parallel channels connected to the headers. The header pressure and resistance of individual channels as well as the thermal hydraulic conditions in them determine the flow through these channels. In the AHWR, having boiling light water as coolant, the core power distribution also has a major effect on the channel flow distribution. This is because of the effect of power on steam quality and hence on the two phase pressure drop. An important aspect of the thermal-hydraulic design of the core is coolant channel orificing. In a core, with varying power and boiling coolant in the channels, orificing is resorted to, to even out the quality of vapour-liquid mixture emanating from various fuel channels. Orificing also reduces the dependency of the flow on power, though at the cost of additional pressure drop. Considering the various factors mentioned above, the steady state thermal hydraulic analysis of the proposed AHWR core was carried out with the objective of determination of following main parameters with and without orificing.

- Channel flow distribution
- Channel pressure drop
- Non-boiling length
- Channel exit steam quality
- Channel exit void fraction.

TABLE - 1: Data Used for Core Thermal Hydraulic Analysis

Channel Type	Type of Fuel	No. of Channels in each Type	Radial* Power Factor
1	Th	27	1.4016
2	Th	27	1.5454
3	Th	27	1.5336
4	Th	27	1.2978
5	Th	27	1.1996
6	Th	27	0.9540
7	Th	29	0 7698
8	Th	27	0.6330
9	Th	27	0.5242
10	Th	27	0.5184
11	Pu	4	0.6450
12	Pu	24	0 7903
13	Pu	4	0.7184
14	Pu	24	0.9103

Total core power : 750 MWth

Core Av. pressure· 70 bar

Total core flow : 1622 kg/sec

Inlet enthalpy : 1055 kJ/kg

Th-U233 bundles

Clad O.D. . 14.6 mm

Pressure tube I.D: 120 mm

Heated length 5 m

PuO2-UO2 Bundles

Clad O.D : 7.35 mm

Pressure tube I.D : 70.08 mm

Heated length 5 m

$$\text{* Radial Power Factor} = \frac{\text{Actual Power in Channel}}{\text{Average Channel Power}}$$

4.1 Computer Code THABNA-M

The analysis has been carried out using the computer code THABNA-M (Thermal Hydraulic Analysis of Boiling Nuclear Assemblies - Modified) which is the modified version of the code THABNA [4]. For the analysis, the fuel assemblies in the core are divided into a number of channel types having same axial and radial power factors and other thermal hydraulic characteristics. The modified code can accommodate 71 channel types and 24 axial nodes for each channel

Either the total core flow or the core pressure drop can be specified as input for the code. When one of the two is specified as input the other is calculated. The coolant flow distribution between the various fuel assemblies is estimated on the basis of equal pressure drop in all the assemblies. For each channel type, the code carries out a node by node calculation from bottom to top. The following pressure drop components are taken into account in calculating the total pressure drop.

- Friction pressure drop (in non-boiling and boiling region)
- Local pressure drop (in orifice, spacers, tie plates etc.)
- Acceleration pressure drop
- Elevation pressure drop.

The coolant flow through the various channel types is so adjusted that the core pressure drop or the total core flow agrees with the specified input value, within the convergence limits specified. The steam quality, void fraction, pressure drop etc. at each node are calculated by the code.

4.2 The Analysis

The steady state analysis has been carried out by dividing the core comprising of vertical fuel channels into 14 groups or channel types. Channels in each type have identical geometry and thermohydraulic characteristics. Heated length of each channel is divided into 24 axial nodes. The analysis has been carried out for

a) identical inlet local loss coefficients for all channels and
b) Different combinations of inlet local loss coefficients for different channels.

TABLE - 2: Results of Core Thermal Hydraulic Analysis

Channel Type	Non - Boiling Length (m)		Channel Exit Quality (%)		Channel Exit Void Fraction (%)		Channel Pressure Drop (bar)	
	case 1	case 2	case 1	case 2	case 1	case 2	case 1	case 2
1	1.24	2.03	38.6	16.5	84.6	69.0	0.61	1.42
2	1.08	1.99	46.6	16.9	87.7	69.5		
3	1.09	2.01	46.0	16.5	87.5	69.1		
4	1.40	2.00	32.5	17.3	81.7	70.0		
5	1.59	2.05	26.7	16.7	78.2	69.3		
6	2.37	2.11	12.7	16.3	64.0	68.9		
7	3.32	2.12	4.6	16.8	44.3	69.4		
8	4.20	2.18	0.5	16.3	15.4	68.9		
9	5.00	2.18	0.0+	16.8	0.0	69.4		
10	5.00	2.21	0.0+	16.5	0.0	69.0		
11	0.96	1.69	37.7	16.4	84.2	69.0		
12	0.71	1.64	53.9	16.6	90.0	69.1		
13	0.81	1.67	46.3	16.4	87.6	69,0		
14	0.60	1.56	65.1	17.6	92.9	70.3		

+ Subcooled Water

Analysis carried out for the case of identical values of local loss coefficient for all channels (case 1) and one case of different inlet local loss coefficient values for different channels (case 2) are presented and discussed in this paper. The data used in respect of channel types, number of channels, radial power factor etc. are given in Table 1.

4.3 Results and Discussions

Channel exit quality and void fraction, non-boiling length and channel pressure drop for the two cases defined in section 4.2 are depicted in table 2. With identical value of inlet local loss coefficient for all channels, the coolant channel exit quality varies from 0.0 to 46.6% for thorium bundles and 37.7% 65.1% for Pu bundles. Values of exit quality in Pu bundles are higher than those in Th bundles because of higher hydraulic resistance of Pu bundles which have smaller flow area. Variation of exit quality in channels having same type of fuel is mainly due to variation of channel power. For case 2, by adjusting the inlet local loss coefficient, the variation of exit quality has been reduced to a great extent. The variation is from 16.3% to 17.3% for Th bundles and 16.4% to 17.6% for Pu bundles. However, exit quality has been evened out at the cost of additional core pressure drop which has increased from 0.61 bar (case 1) to 1.42 bar. The increase in pressure drop is because of the increase in the value of inlet local loss coefficient. The pressure drop can be reduced if variation of exit quality to a larger extent is found to be acceptable.

5.0 Thermosyphon Analysis of the PHT System

Two phase thermosyphon analysis of the PHT system has been carried out using the computer code RELAP4/MOD6 [5] for different cases.

5.1 Cases Analysed

i) Effect of variation of loop height Z (see fig.1) on natural convection flow. The height Z was varied by varying Z_1 and Z_2. The values of Z (in metre) considered are 10, 15, 20, 25, 30, 38, 46 and 60. The core power was maintained constant at 750 MWth for all these cases.

ii) Effect of variation of core power on natural convection flow. Analysis was carried out over a range of values for core power from 50% to 100% with steps of 5% (100% corresponds to 750 Mwth). Loop height Z was maintained constant at 38 m for these studies.

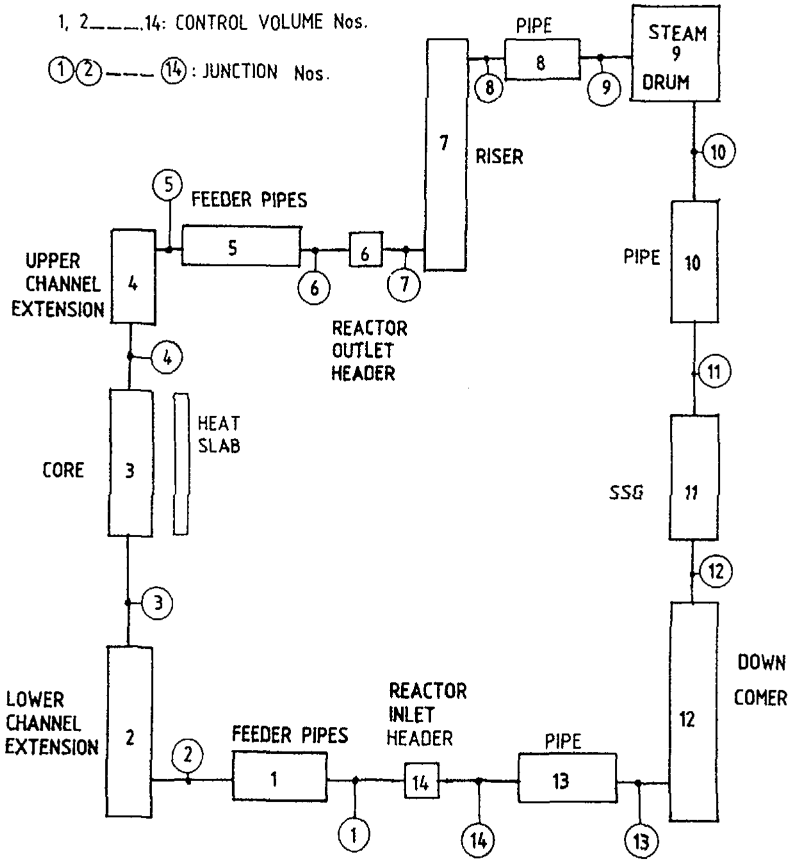

FIG. 3 NODALIZATION SCHEME FOR THE AHWR THERMO SYPHON ANALYSIS

5.2 Analysis

The detailed nodalisation scheme adopted for the analysis is depicted in figure 3. The PHT system was simulated as a set of control volumes and junctions. Fourteen control volumes and fourteen junctions were considered for the analysis as shown in the figure. The fuel was modelled as heat generating cylindrical slab. Core inlet hydraulic resistance corresponding to case 2 of Section 4.2 was considered for the analysis. The steam drum pressure was maintained constant at 69 bar.

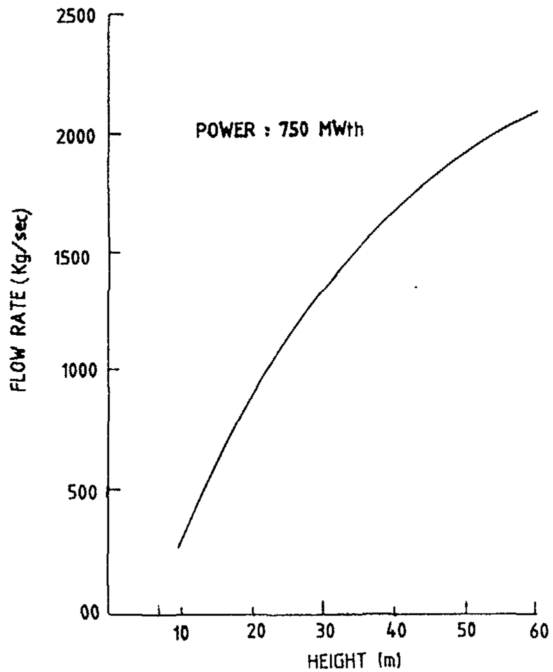

FIG. 4 VARIATION OF CORE FLOW WITH LOOP HEIGHT (Z)

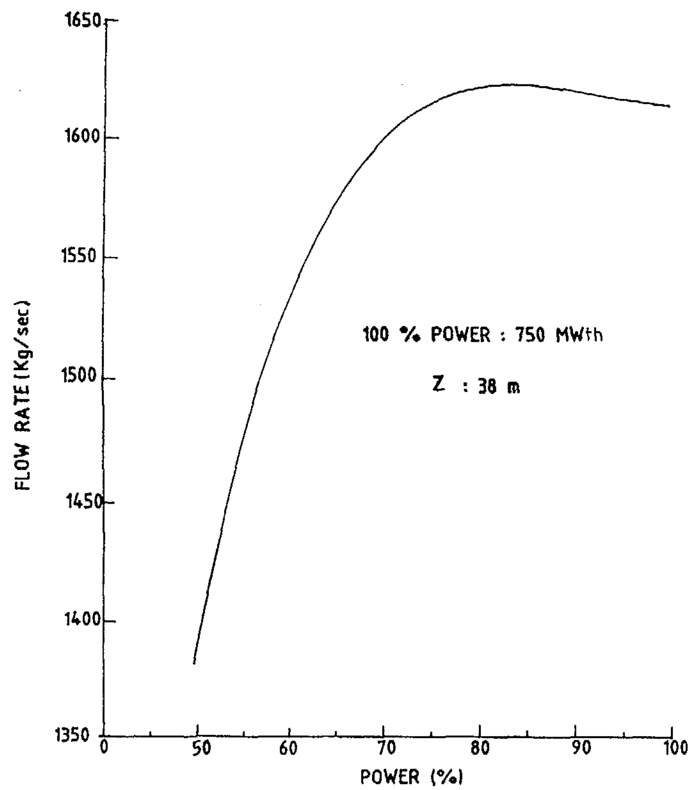

FIG. 5 VARIATION OF CORE FLOW WITH CORE POWER

5.3 Results and Discussions

Variation of PHT system flow rate with loop height, Z, is depicted in fig.4. Over the range of parameters studied, flow rate increases considerably with increase in Z. However, the slope of the curve reduces with increase in height. This is mainly because of larger increase of flow resistance compared to the increase in driving head. It may be noted from the figure that the flow rate used for the core thermal hydraulic analysis (Section 4.2) is obtained when the loop height is 38 metre. The variation of flow rate with power for this height is depicted in fig.5. The flow rate initially increases considerably with power upto a power of about 85% and then reduces slightly. This reduction of flow rate can be attributed to the increase in flow resistance which dominates over the increase in the driving head caused by the increasing steam quality. Similar trends are noted in the results presented in reference 6.

6.0 Conclusions

i) In the absence of orificing, exit quality in some of the channels is very high while at the exit of some of the channels water is subcooled. This is due to the variation of channel hydraulic resistance and power.

ii) By orificing almost uniform channel exit quality can be achieved but the channel pressure drop increases by a factor of about 2. However, pressure drop can be reduced if non-uniformity of exit quality to a reasonable extent is allowed.

iii) Results of the thermosyphon analysis indicate that the intended core flow rate can be obtained when the loop height is about 38 metre.

iv) Flow rate increases with increase in power, attains the maximum value at about 85% and then reduces.

Acknowledgement

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References

1. A.C. Saraswat, "Uranium and other Resources for Indian Nuclear Power Programme", Conference on Nuclear Science and Technology, Bombay, January, 1989.

2 K Balakrishnan and A Kakodkar Preliminary Physics Design of Advanced Heavy Water Reactor (AHWR)' IAEA Technical Committee *Meeting on* Technical and Economic Aspects of High Converters Nurenberg, FRG, March 1990

3 Communications from K Balakrishnan, Reactor Engineering Division, BARC, India

4 V. Venkat Raj, A K Anand, D Saha "THABNA- A Computer Program for the Thermal Hydraulic Analysis of Boiling Nuclear Assemblies" Second National Heat and Mass Transfer Conference, Kanpur, India, December, 1973

5 S.R Fischer et al , "RELAP4/MOD6, A Computer programme for transient thermal hydraulic analysis of nuclear reactors and related systems, User's manual", CDAP-TR- 003, Jan., 1978

6. R.B. Duffey and J.P. Sursock, "Natural Circulation Phenomena Relevant to small breaks and transients". Nuclear Engineering and Design, V 102, PP 115-128, 1987.

CAPABILITIES OF THE RELAP5 IN SIMULATING SBWR AND AP600 THERMAL-HYDRAULIC BEHAVIOUR

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Abstract

This paper deals with the application of the RELAP5/Mod2 code for the analysis of thermal hydraulic transients in the SBWR and AP600 nuclear reactors.

The RELAP5 limits and capabilities as far as the development of the nodalization and the simulation of each plant are discussed. In particular, limitations have been found in the applications of code models to low pressure scenarios where gravity is the dominant driving force for core cooling

1 INTRODUCTION

In the framework of ENEL activities for the safety evaluation of passively or intrinsecally safe nuclear reactors, SBWR and AP600 plants are under study.

SBWR and AP600, proposed by General Electric and Westinghouse respectively, are the result of an evolution of BWR and PWR technologies widely used for the electricity production over the last thirty years. In both cases the electric power ranges around 600 MWe and substantial simplifications have been introduced in all the main systems to improve the reliability and reduce the cost of power generation Passive safety features, mostly based on gravity, characterize the abnormal operation systems, large water pools are available at higher elevations than the core and allow cooling in any low pressure accident scenario

For the analysis of thermal-hydraulic response of the two plants ENEL and Pisa University have developed two RELAP5/Mod2 models and have undertaken a program in order to assess and validate them before the use for safety evaluations.

RELAP5/Mod2 /1/ is a well known transient analysis code for complex thermal hydraulic systems. It is based on a non-homogeneous non-equilibrium set of six one-dimensional balance equations for the steam and the liquid phases.

The RELAP5/Mod2 has been intensively used at DCMN of University of Pisa in the analysis of transients in experimental facilities /2/, /3/ as well as in real plants /4/, /5/ Limits and capabilities of the code have been identified with reference to the various phenomena occurring in nuclear reactors in accident conditions.

When applying the code to the SBWR and AP600 it has to be consider that the typical accident scenarios in these plants can be subdivided into two parts a) time before the equalization between primary and containment pressures; b) long term cooling period, with gravity heads at near atmospheric pressure which are the driving forces for different natural circulation loops in the plant. Furthermore, the interaction between the primary circuit and the containment is a fundamental aspect in the safety of these reactors.

Now, while the RELAP5 code has been assessed worldwide /6/ in relation to phenomena occurring in part a) of the scenario, very limited experience in the application to phenomena typical of the part b) is reported

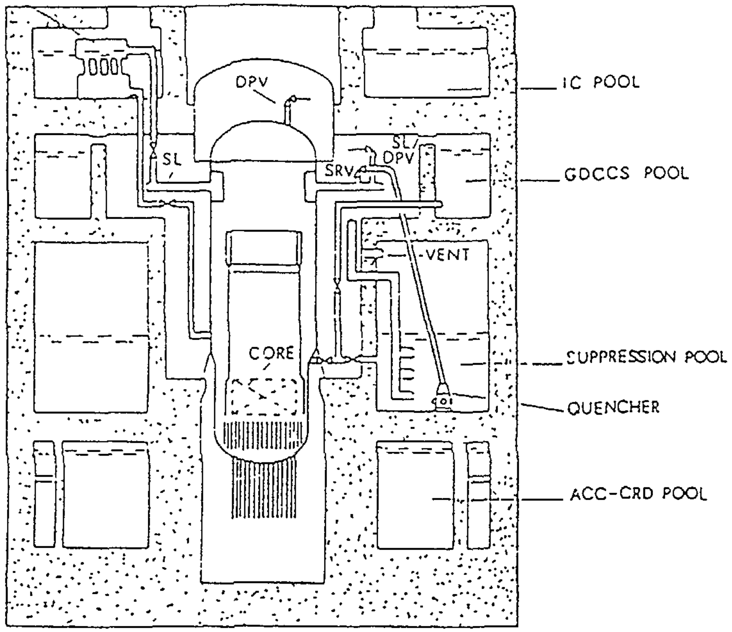


Fig. 1 - Sketch of SBWR including safety systems

The purpose of this paper is to discuss the main results of RELAP5/Mod2 code application to the analysis of four scenarios (two for each plant) in SBWR and AP600 reactors. It has to be noted that both the plant models have been developed starting from plant data updated to design status in the early 1990 and that some details of particular components of the plants were not available. Moreover the developed models are already under validation. Therefore the results discussed in this paper have to be considered as preliminary.

2. SBWR ANALYSES

The SBWR plant /7/,/8/,/9/ (Fig. 1) is designed to use natural circulation to cool the core both in normal and accident conditions. In accident conditions the reactor is cooled by the following systems:

- Isolation Condenser (IC), High Pressure Injection System (HPIS) and Control Rod Driving (CRD) for high pressure transients;
- Gravity Driven Cooling System (GDCS) at low pressure conditions.

The IC system allows the energy exchange between the vessel and an external pool, by the condensation of the steam coming from the vessel without fluid discharge from primary system. The GDCS injects into the vessel the water coming from a pool at an elevation higher than the top of the core, in order to provide the adequate head for system operation at low pressure.

Tab. I - Boundary and initial conditions for SBWR Feedwater Trip and Steam Line Break accidents

			FWT	SLB
1	Steam dome pressure	(MPa)	7.04	=
2	Core Power	(MW)	1800.	=
3	Feed Water Temperature	(K)	488.	=
4	Core Flow Rate	(Kg/s)	7556	=
5	Steam Line Flowrate	(Kg/s)	970.	=
6	Lower Plenum Temperature	(K)	551.	=
7	Downcomer Level	(m)	15.03	=
8	HPIS and CRD fluid temperature	(K)	300.	=
9	HPIS flow rate	(Kg/s)	19.6	=
10	Liquid temperature of the pools	(K)	300	
11	Break occurrence	(s)	===	0.
12	Feed water closure	(s)	0.	0.
13	Scram	DC level	=13.98 m	=13 98 m
14	Actuation of CRD	DC level	==	=13.98 m
15	Actuation of HPIS	DC level	=13.98 m	=13.98 m
16	MSIV closure	DC level	=12.98 m	=12.98 m
17	Actuation of IC	DC level	=12.98 m	=12.98 m
18	Actuation of SRV	DC level	==	=10.53 m (+)
19	Actuation of SL-DPV	DC level	==	=10.53 m (++)
20	Actuation of RPV-DPV	DC level	==	=10.53 m (+++)
21	Actuation of GDCS	DC level	==	=10.53 m

(+)90 s delay
(++)135 s delay
(+++)150 s delay

The primary system depressurization is allowed by groups of valves on the steam lines that discharge fluid in the containment suppression pool.

The containment cooling is provided by the Passive Containment Cooling System (PCCS), which performs the heat transfer to environment, condensing the steam of the containment atmosphere with a three days operational capability.

For SBWR two accidents have been selected among those which contribute to the overall risk of the plant. They are a complete Feedwater Trip (FWT) and a double ended Steam Line Break (SLB).

The main boundary and initial conditions, are given in Tab. I. The considered conditions do not necessarily reflect the nominal reference values of the plant.

2.1 SBWR nodalization

The SBWR model consists of 200 fluid volumes, 220 junctions and 183 heat conducting structures (Fig. 2). The core is subdivided into four parallel channels to account for the radial power distribution; an additional channel simulates the bypass region between the fuel assemblies. The isolation condenser, the two steam lines, and the GDCS can be easily identified in the nodalization.

Each large tank (GDCS and Isolation Condenser) is connected with a node having assigned conditions because it is impossible to model the presence of air in the control volume using the available code version.

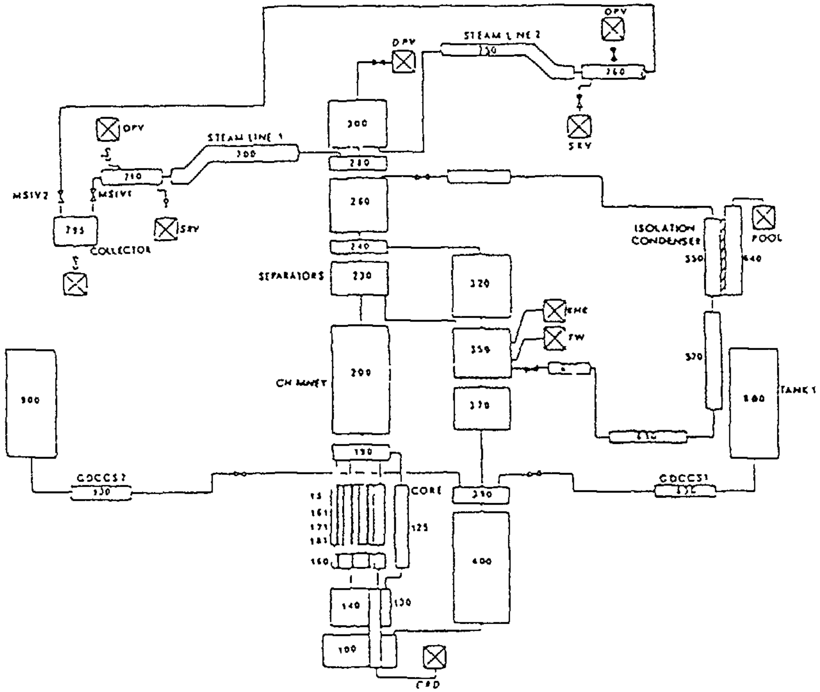


Fig. 2 - Simplified RELAP5/Mod2 nodalization scheme for SBWR

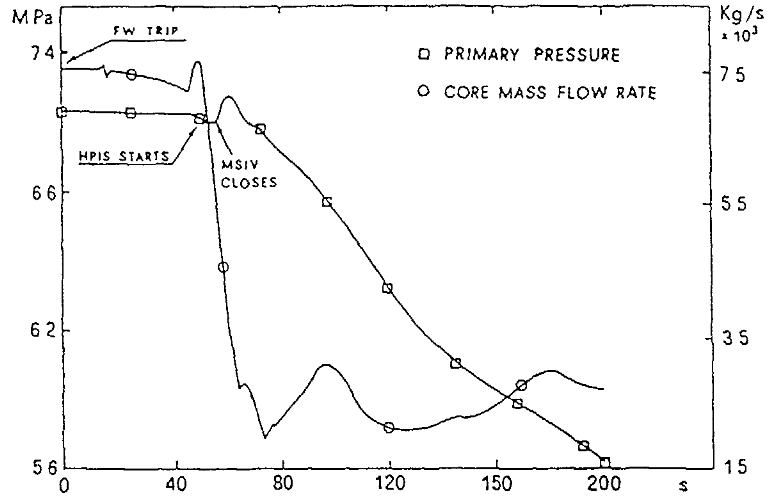


Fig 3 Primary system pressure and core mass flow rate following a feedwater trip in the SBWR plant

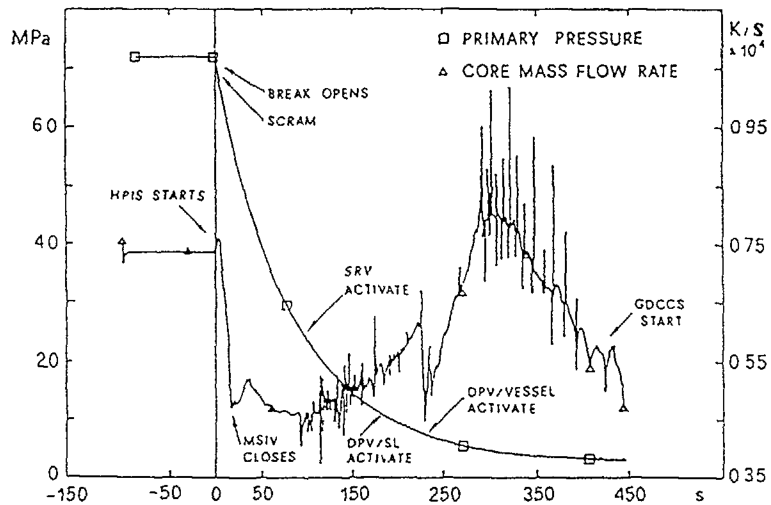


Fig 4 - Primary system pressure and core mass flow rate following a main steam line break in the SBWR plant

2.2 SBWR transient results

Preliminary results about SBWR thermal hydraulic response to the feedwater trip and to the main steam line break transients can be seen in Fig 3 and 4, respectively

In the transient originated by feedwater trip, the vessel isolation occurs after about 100 s. The scram is actuated earlier due to low level in the downcomer The HPIS and the isolation condenser activations are sufficient to keep the core in a stable cooled situation at relatively high pressure Natural circulation also occurs between core and bypass leading to negative values of the bypass flowrate

The transient calculation for the SLBA has been stopped shortly after the GDCS has been activated The primary depressurization is enhanced by the opening of various groups of discharge valves (see Tab I) The low pressure at the top of the vessel leads to an increase in the core flowrate during the transient up to values typical of nominal conditions. Core power at this time corresponds to decay power values and bypass flowrate remains positive for almost all the transient. Wide oscillations occurs in the core flowrate In the latest period of the transient, mainly following GDCS actuation, the isolation condenser flowrate becomes negative An apparent effect of the GDCS intervention is a step increase of subcooling in the lower plenum that improves the core heat transfer potential

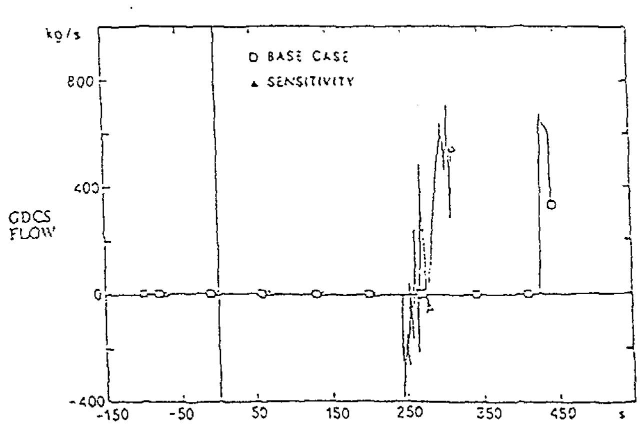
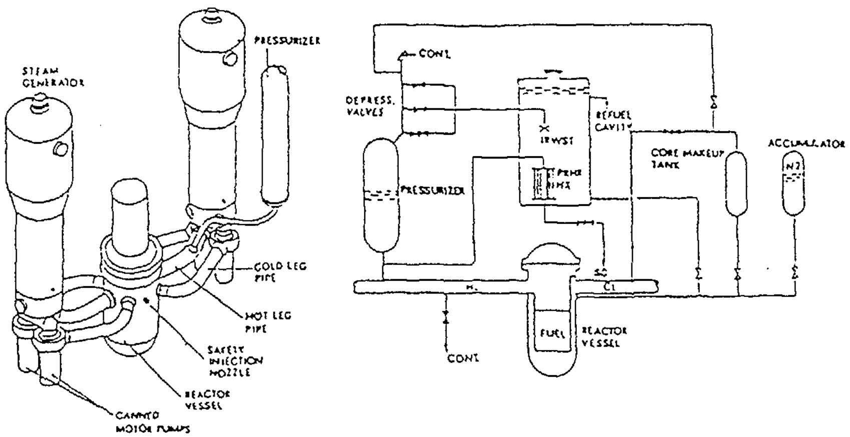


Fig. 5 - Flowrate delivered by GDCS in the SLB transient in the SBWR, for different operation modes of check valves



a) b)

Fig. 6 - Sketch of AP600: a) primary loops; b) safety systems

With reference to this accident, the effect of two different GDCS flowrates has been investigated and the obtained results are given in Fig. 5.

The reference case, where GDCS valve is assumed to work properly, is compared with the result of a calculation where it is assumed that the valve opens (and after that time it remains always open) when the pressure difference between the primary system and a point upstream to the valve is still positive.

3. AP600 ANALYSES

The AP600 plant /10/, /11/ is cooled by a simplified primary system with two loops, employing inverted canned motor pumps feeding cold liquid to two separate cold legs (see Fig. 6). AP600 safety features include:

a) the Passive Residual Heat Removal (PRHR), whose function is similar to the I.C. one in the SBWR plant;
b) the Inside containment Refueling Water Storage Tank (IRWST), which provides the reactor cooling at low pressure condition through a line connected to vessel downcomer;
c) the Depressurization System, consisting of four group of valves;
d) two Core Make up Tanks, essentially consisting in large reservoirs of fluid;
e) two accumulators pressurized at 4.9 MPa by nitrogen.

Transients that are widely investigated for conventional PWR have been selected in the case of AP600. Both of these are Loss of Coolant Accidents (LOCAs): an intermediate LOCA (ILOCA), originated by a break in the safety injection pipe (piping connecting the accumulator and CMT to the reactor vessel downcomer), and a double ended LOCA in one of the cold legs of the loop with pressurizer.

The main boundary and initial conditions are reported in Tab. II.

3.1 AP600 nodalization

The model consists of 245 fluid volumes, 264 junctions and 253 heat conducting structures (Fig. 7). There are three parallel channels connected by cross junctions in the core model. The two cold legs including canned motor pumps, were modelled in each primary loop. The engineered safety features were modelled with different levels of detail.

3.2 AP600 transient results

Preliminary results about AP600 thermal-hydraulic response to the intermediate break LOCA and the large break LOCA can be seen in Fig. 8 and 9, respectively.

In both the considered scenarios the primary pressure becomes lower than the secondary pressure early in the transient: consequently the secondary sides of the steam generator that remains isolated from the turbine and condenser (Tab. II), do not play an important role unless as heat source late in the transient.

In the case of the ILOCA the break is in one of the two main delivery lines of the emergency system and causes the discharge of the liquid contained in one accumulator and in one CMT, directly to the containment without contribution to core cooling. The behaviour of the various systems can be summarized as follows:

- reverse flow (from the cold leg to the pressurizer surge line) occurs after the activation of the PRHR owing to the pressure balance in the loop; afterwards a stable circulation is established in the right direction;
- the Depressurization System causes a positive steam flow from the pressurizer towards the IRWST and the CMT;
- in the early period of the transient, the accumulators discharge prevents the liquid fall from the CMT connected to the intact line owing to a counterpressure in the delivery lines.

No dryout situation results during the ILOCA transient.

On the other hand, an extended dryout mostly in the central part of the core characterizes the LBLOCA scenario which appears similar to the one predicted for the "conventional" PWR. Early core rewet in the lower

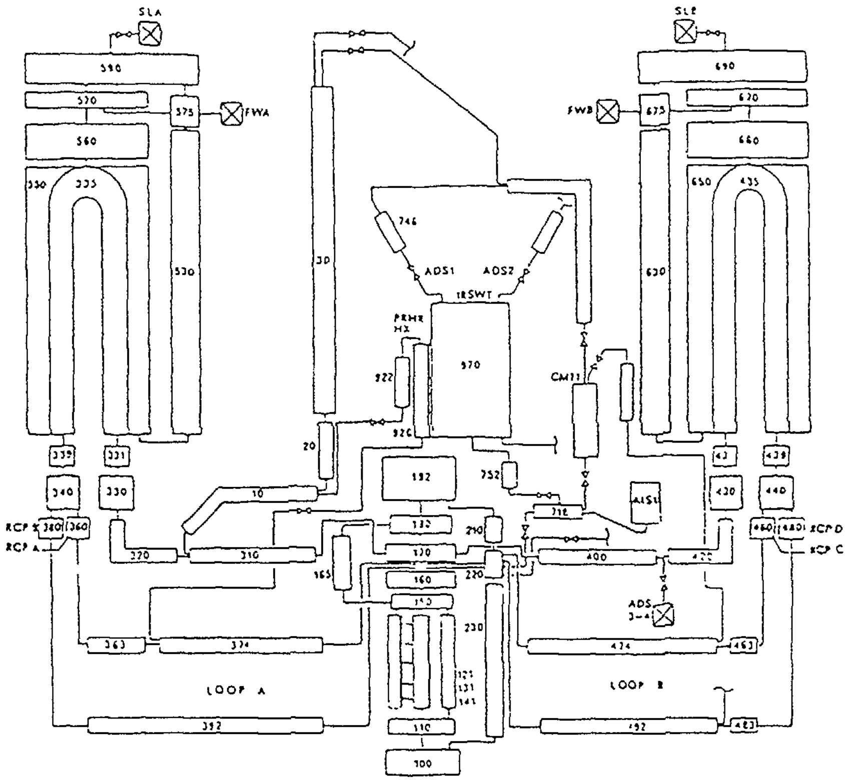


Fig. 7 - Simplified RELAP5/Mod2 nodalization scheme for AP600

part of the fuel bundles is also observed in the calculation. The accumulators and the liquid discharge from the CMT are effective in bringing the plant to a safety condition.

It has to be noted that code results for calculations in the low pressure range are strongly influenced by variations in plant lay-out, initial conditions or plant nodalization.

In the following some examples of this are given.

<u>Position of sparger in the IRWST</u>

The reference ILOCA calculation was performed assuming that the sparger of the depressurization system in the IRWST pool is located near the top of the volume occupied by the liquid. In this case only partial condensation of the steam discharged from the primary system is observed; the remaining part of the steam contributes to the pressurization of the pool volume, allowing the drainage of the liquid from the pool towards the core.

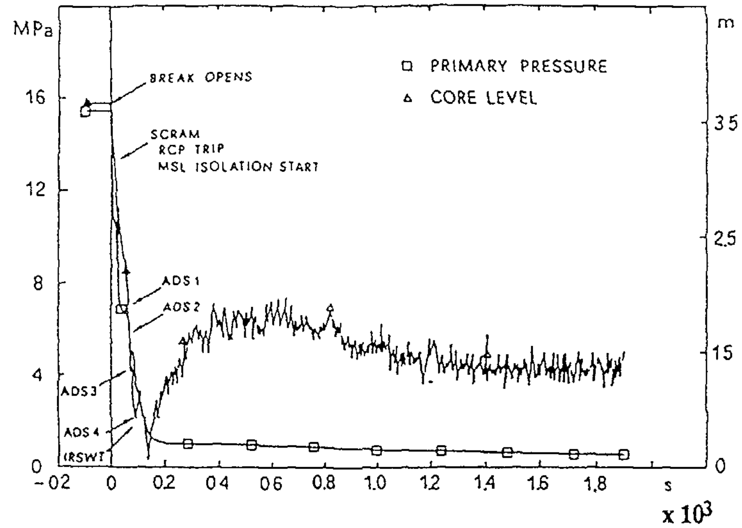


Fig. 8 - Primary system pressure and core level trend following an Intermediate Break LOCA in the AP600

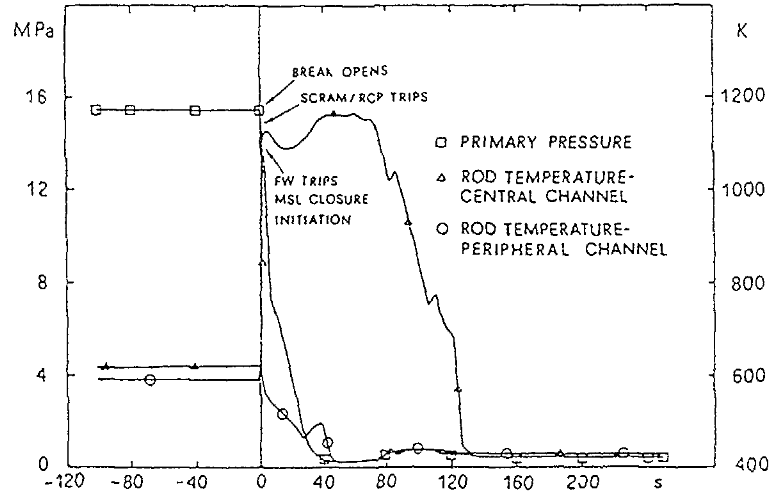


Fig. 9 - Primary system pressure and fuel rod temperature following a Large Break LOCA in the AP600

Tab. II - Boundary and initial conditions for AP600 Intermediate and Large Break LOCA

	QUANTITY	VALUE OR SETPOINT	
1	Pressurizer pressure	(MPa)	15.53
2	Core Power	(MW_{th})	1812.
3	Hot leg Fluid temperature	(K)	596.4
4	Core How rate	(Kg/s)	8921
5	Steam generator secondary pressure	(MPa)	6.33
6	Feedwater temperature	(K)	508.15
7	Feedwater how rate	(Kg/s)	515.84
8	Steam generator recirculation ratio	(-)	4.
9	Break occurrence	time =	0 s
10	Reactor trip	PRZ pressure =	12.7 MPa
11	CMT actuation	PRZ pressure =	12.7 MPa
12	Feedwater trip	PRZ pressure =	12.7 MPa
13	Steam line trip	SG level =	11.1 m
14	RCP trip	PRZ pressure =	12.7 MPa
15	RCS depressurization actuation: 1st stage	CMT level =	4.4 m
	2nd stage	CMT level =	3.97 m
	3rd stage	CMT level =	2.33 m
	4th stage	CMT level =	10% initial level
16	PRHR actuation	CMT level =	4 4 m
17	Start at delivery at IRWST liquid into the RCS	Prcs < Pirwst	

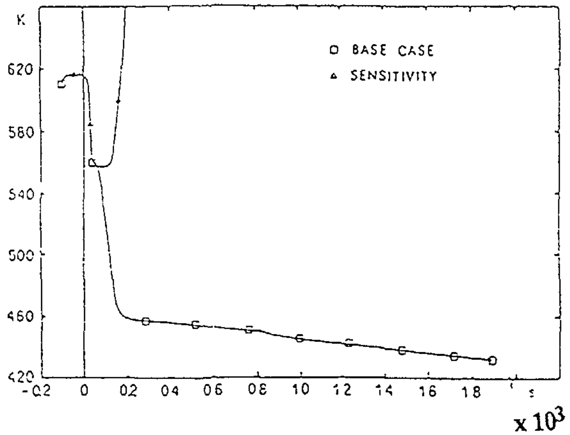


Fig. 11 - ILOCA analysis for AP600: effect of different liquid temperature in the CMT

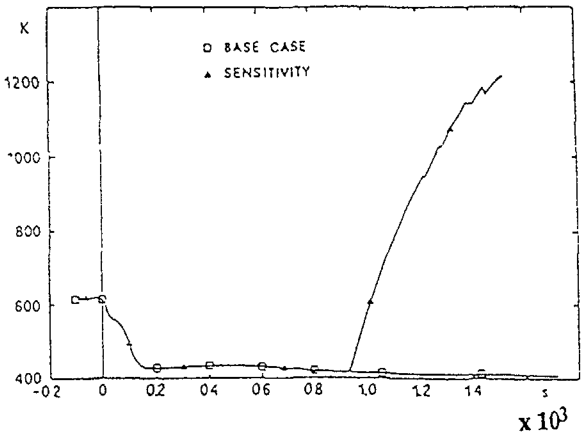


Fig. 10 - ILOCA analysis for AP600: effect of different positions of the depressurization system sparger in the IRWST

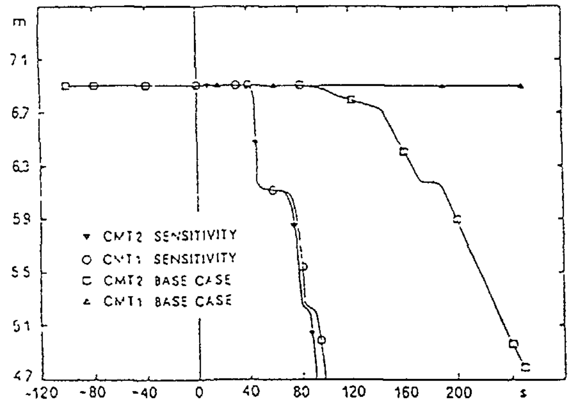


Fig. 12 - LBLOCA analysis for AP600: effect of nodalization of the CMT discharge lines

A second calculation was carried out assuming that the sparger is located in the bottom of the pool. In this case, a nearly complete steam condensation prevents the pool liquid from discharging and leads to extended core dryout (Fig. 10).

Effect of liquid temperature in the pool

Again with reference to the ILOCA in the AP600 plant, while the reference calculation was done assuming an initial temperature of the liquid in the Core Make up Tanks equal to 373 K (to prevent the stop of the calculation owing to numerical problems), a second calculation was run considering a lower (323 K) liquid temperature in the Core Make up Tanks. In this case the large vaporization in the connection zone between the CMTs lines and the vessel creates a steam upward flow in the lines that prevents liquid drainage and causes dryout early in the transient (Fig. 11).

Effect of nodalization of the CMT discharge lines

The line connecting the CMT to the vessel presents a complex layout in the vertical plane. Different approaches in modelling the line result in quite different prediction of CMT discharge. The CMT levels in the reference case for LBLOCA are compared with those obtained in the sensitivity study in Fig. 12(*). In the reference case the fairly coarse nodalization of the discharge lines modifies the pressure drop situation in the plant causing a large positive flow from the cold leg to one of the CMT, that does not experience any level decrease despite the liquid delivery to the vessel. The nearly constant level situation in one tank prevents the activation of the depressurization system.

The more detailed nodalization used in the sensitivity calculation allows the complete discharge of both CMT's.

4. EVALUATION OF CODE LIMITS

As already mentioned, in the evaluation of code results two periods in each of the four transients should be distinguished:

a) primary system pressure greater than about 0.5 MPa;
b) subsequent period, including the interaction between the primary loop and the containment system and the long term behaviour of the passive safety systems.

Phase a) is within a range of physical situations for which an accurate code qualification process based on international standard problems, already exists.

Phase b) results must be considered outside the qualification boundary limits of the code, considering the international assessment activity.

As all the phenomena relevant to the safety in the analyzed transients(**) occur in the period a), the obtained results have to be considered reliable. Nevertheless some limits of the code models with main reference to the capabilities required to simulate low pressure scenarios governed by gravity, should be highlighted.

Specific limits of the code are as follows:

- at low pressure ($P \leq 0.2$ MPa), very wide oscillations occur in the physical quantities. This requires the use of very small time steps and cause the interruption of the calculation;
- the simulation of the fluid behaviour downstream of a critical section (supercritical flow) is not allowed in any geometric configuration;
- the transition between critical flow model and Bernoulli model leads to instabilities in the calculation, mainly at low pressure;
- the temperature stratification in pool with large vertical dimension is not well calculated by the code. This may results in wrong prediction of gravity heads and in large errors in mass flowrates;
- the natural circulation in the system constituted by primary loops and containment, largely depends on local loss coefficients in complex three dimensional geometries that must be supplied as input by the user;
- the zero-dimensional neutronic kinetic model is not suitable for simulating the 3D behaviour of large cores, and in particular of the SBWR one where the control rods and the peculiar moderator distribution are strong sources of non uniformities;
- the condensation at the "ECCS port" is a further source of instability that stops the calculation.

5. CONCLUSIONS

The RELAP5/Mod2 code has been used to develop models for the SBWR and AP600 plants and to carry out preliminary analyses of two typical accidents for each plant.

As it is well know, passive LWR's are characterized by the interaction between primary circuit and containment system, the occurence of the most important scenarios at low pressure and the existence of several potential natural circulation loops.

The obtained results showed the safe response of the plants to the analyzed accidents. Some plant features that could require additional investigations by designers, and code limits in the simulation of the complete thermal-hydraulic transient have also been identified.

As far as the qualification of the code is concerned, its validation in the physical conditions that are specific of the new reactors is strictly necessary. The behaviour of components like the isolation condenser in SBWR and the cooling pools injecting liquid in the core by gravity, are an example of items requiring an in depth study.

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REFERENCES

/1/ V.H. Ranson, R.J. Wagner, J.A. Trapp, L.R. Feinhauer, G.W. Johnson, D.M. Riser, R.A. Riemke: "RELAP5/MOD2 Code Manual Vol. 1, Code Structure, system models and solution methods" NUREL/CR - 4312, August 1985

/2/ F. D'Auria, G. M. Galassi: "Code Assessment Methodology and Results" IAEA Tech. Committee Workshop on Computer Aided Safety Analysis - Moscow (USSR), May 14-17, 1990

/3/ F. D'Auria, G. M. Galassi: "Assessment of RELAP5/MOD2 code on the basis of experiment performed in LOBI Facility"J. Nuclear Techonology, Vol 90 (3), p. 340-355, 1990

(*) The steps observable in the curves for the reference case occur when the level crosses the volume boundary and constitute a clear nodalization effect.

(**)The only exception is the LBLOCA event in the AP600. In this case the past experience in the analysis of large LOCA in PWR's can help in the evaluation of the results.

/4/ P Camiciola, F D Auria, P Marsili, P Petrini
"OECD CSNI ISP 20 post test calculation and sensitivity analyses by RELAP5/MOD2 performed with reference to the SGTR accident Doel 2 NPP"
University of Pisa Report, DCMN RL 290 (87), Pisa 1987

/5/ F D Auria, P Maugeri
"Analysis by RELAP5/MOD2 of a Loss of Feedwater Transient in the Leibnstand BWR-6 plant"
Int Conf on Thermal Reactor Safety, Avignon (F) October 3 7, 1988

/6/ CSNI Group of Experts.
"Thermalhydraulics of Emergency Core Cooling in Light Water Reactors
CSNI Report 161, Paris October 1989

/7/ B Wolfe, D R. Wilkins.
"Future directions in Boiling Water Reactor Design"
J Nuclear Engineering and Design, Vol 115 pag 281 288, 1989

/8/ J D Duncan, C D Sawfer
"Capitalizing on BWR Simplicity at Lower Power Rating"
ANS/ENS International Conference, November, 1984

/9/ J D Duncan
"SBWR, a Simplified Boiling Water Reactor"
Nuclear Engineering and Design, Vol 109, n 1 and 2, 1988

/10/ R.A. Bruce, R.P Vijuk:
"Conceptual Design for an advanced Passive 600 MWa PWR"
ASME IEEE Power Generation Conference, Miami Beach, USA, October 4 8, 1987

/11/ C.M Vertes.
"Passive Safeguards Design Optimation Studies for the Westinghouse AP600"
1989 ANS Winter Meeting, San Francisco, USA, November 20-30, 1989

ASSESSMENT OF THE PIUS PRIMARY SYSTEM USING ANALYTICAL AND EXPERIMENTAL MODELS

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Abstract

This paper deals mainly with the application of the RELAP5 code to the thermal hydraulic analysis of PIUS system

In the first part, the RELAP5 model and the analysis of plant steady state under forced and natural circulation conditions are described The analysis outlines the influence of different modelling of heat conducting structures and pressure losses on the prediction of plant response

The second part describes the analytical and experimental models under development in order to analyse particular separate effects typical of the innovative nature of the PIUS plant

1 INTRODUCTION

The assessment of PIUS primary system requires the development of analytical and experimental models in order to simulate the thermal hydraulic behaviour in normal and accident conditions

In the frame of a cooperation between ENEL-CRTN and the "Dipartimento di Energetica" of Politecnico di Torino, a RELAP5 model of PIUS reactor has been developed and implemented on ENEL CRAY XMP/14 computer system

The selection of RELAP5 code derives from the following considerations

RELAP5 is an advanced best estimate thermal hydraulic code, developed at INEL under the USNRC support for safety analysis of nuclear reactors, and submitted to a wide qualification program based on experimental facilities

RELAP5 has been adopted by ENEL also for the thermal hydraulic analysis of SBWR and AP600 reactors, in order to provide an homogeneous tool for the comparison of the three different systems.

Some sensitivity studies have been carried out to evaluate the influence of the constitutive models implemented in the code (heat transfer, wall friction and form losses, etc)

On the other hand the innovative features of PIUS system require an accurate assessment of code capability in simulating particular phenomena like natural circulation, syphon breaker behaviour, boron diffusion, etc In order to start this assessment process an experimental test facility for lower density lock

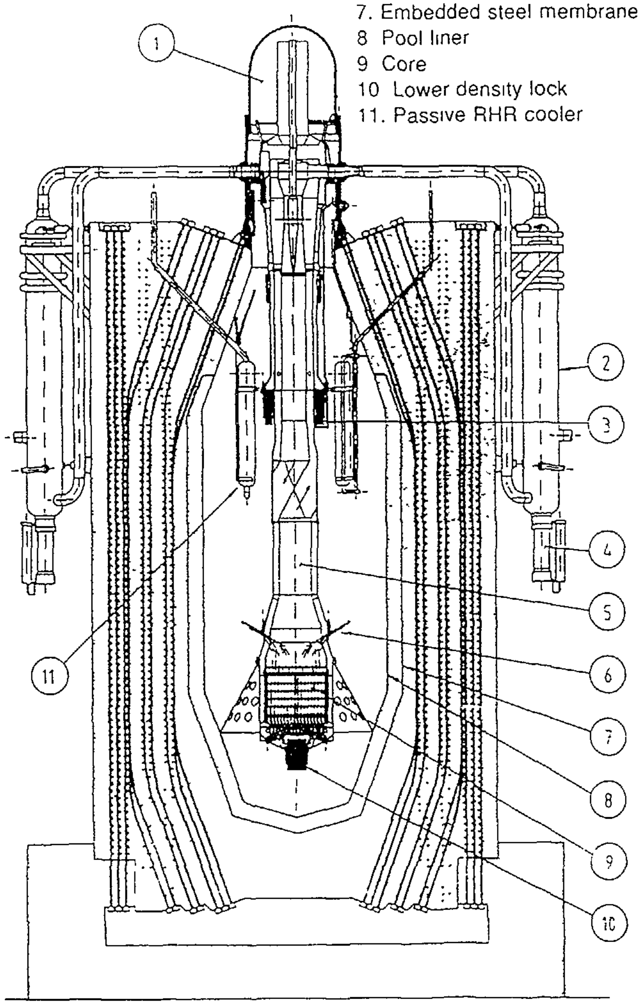

Fig. 1 - Sketch of PIUS Reactor

simulation has been designed and 2-D fluid dynamic analyses have been carried out using FLUENT code. FLUENT code, which solves the Navier Stokes equations in complex geometries, provides an usefull tool in the analysis of separate effects as support to plant design.

2. *PIUS SYSTEM DESCRIPTION*

PIUS (Process Inherent Ultimate Safety) /1/ is a modified PWR where the main difference compared with the current reactors is the safety concept which leads to a different layout of primary system and design of out-of-core components. The name implies that safety against serious accident is inherent in the design of heat removal process and not a result of outside safety features such as emergency core cooling systems. In addition this inherent safety is always present as ultimate protection against serious accidents, even though such accidents are usually averted by normal plant control system.

To preserve the core integrity and to avoid radioactive releases to the environment the PIUS core is submerged in a large pool of borated water enclosed in a prestressed concrete vessel (Fig.1). The reactor core is a PWR type core with 213 fuel assemblies with standard PWR fuel rod diameter and reduced height. The 2000 MWt core is located at the bottom of the reactor pool and does not use control rods, neither for reactor shutdown, nor for power shaping. Reactivity control is accomplished by means of reactor coolant boron concentration and temperature control. Reactivity compensation for burnup is provided by burnable absorber (gadolinium) in selected fuel rods, while the moderator temperature reactivity is strongly negative throughout the operating cycle.

The coolant leaving the core at 290 °C, passes up through the riser pipe and leaves the reactor vessel through nozzles connected to the side of the upper plenum. The coolant enters the 4 hot legs and goes to the 4 once-through steam generators. The main coolant pumps are located in the steam generators outlet plenum and are structurally integrated with the steam generators. The cold leg piping enters the reactor vessel through nozzles in the upper plenum at the same level as the hot leg nozzles and return flow at 260 °C is directed downward to the reactor core inlet via a downcomer. On its way down the flow is accelerated and there are open connections between downcomer and riser providing a siphon-breaker arrangement.

The siphon-breaker prevents siphoning off the reactor pool water inventory in the event of a cold leg rupture. During normal operation the siphon-breaker does not affect the water circulation. At the bottom of the downcomer the return flow enters the reactor core inlet plenum. A pipe that is open to the enclosing reactor pool, is located below the core inlet plenum. A tube bundle arrangement inside this pipe minimizes water mixing and ensures stable layering of hot reactor loop water on top of colder reactor pool water. This pipe is called "lower density lock" and the position of the interface between hot and cold water, defined by temperature measurements, is used to control the speed of the main coolant pumps and hence the flow rate in the primary system in order to maintain the interface level at a constant position during normal operation.

There is another density lock at a higher location in the pool, connected to the upper riser plenum; here the temperature measurements for the interface level are used to control the primary loop water volume. This reactor system configuration with the two continuously open density locks connected to the high boron content water pool, provides the basis for the PIUS principle. There is always an open natural circulation path from the pool through the lower density lock to the core, through the core itself, up the riser, from the upper riser plenum to the upper density lock and back to the pool.

During normal plant operation this natural circulation circuit is kept inactive by controlling the speed of the main coolant pumps, maintaining the hot-cold interface level in the density locks. The coolant flow rate in the core is always determined by the thermal conditions at the reactor core outlet relative to the reactor pool. The resulting pressure drop across the core and up to the riser must correspond to the static pressure difference between the interface levels in the upper and lower density locks. The main coolant pumps are operated to establish a pressure balance in the density locks during normal steady state and load following operations. A sudden collapse in this pressure balance, as would occur during a severe transient or an accident, would result in natural circulation of borated pool water through the core providing both reactor shutdown and continued core cooling.

The hot parts of primary system are insulated from the cold reactor pool water by means of a wet heat insulation of metallic type. This insulation consists of a number of thin, parallel stainless steel sheets with stagnant water in between.

3. INTEGRAL SYSTEM MODELIZATION WITH RELAP5 CODE

3.1 RELAP5 MODEL DESCRIPTION

RELAP5/Mod2 /2/ is a well known transient analysis code for complex thermal-hydraulic systems. It is based on a non-homogeneous set of six partial derivative balance equations for the liquid and vapor phases. A set of constitutive equations accounts for the properties of two-phase mixtures, as well as for heat transfer regimes. A zero-dimensional model for neutron kinetics simulates fission and decay power behaviour.

The RELAP5 model of the PIUS plant was based on the plant data included in Preliminary Safety Information Document (PSID) /3/. Two versions of the model were developed. In the first one only the core and the steam generator tubes were simulated as heat slabs (elsewhere adiabatic) while in the second, with the same number of volumes, all the structures, inside and outside the pool, were taken into account.

Two primary loops were modeled: one representing the single loop of PIUS plant and the other representing the remaining three loops. The pressure losses between inlet and outlet are the same for both the loops. Each primary loop contains a steam generator and a main coolant circulation pump. A constant heat transfer coefficient was assumed in input for the steam generator secondary side to take into account heat removal from the primary loop and to verify the thermal balance.

The reference model consists of 201 hydraulic control volumes, 205 junctions and 181 heat slabs and it is shown in Figg. 2 and 3.

3.2 STEADY STATE ANALYSIS

Two phases were necessary in order to reach steady state conditions. Using the first model of PIUS reactor (without heat structures), initial guessed pressure and temperature distributions in the primary system were assumed and the system conditions were let to evolve up to 200 s while keeping pressure in the pressuriser and pool level constant and the path between reactor and pool through the density locks inactive. In fact, in this time period, trip valves 199 and 599 were closed, trip valve 886 (pressure control) was opened and time dependent junction 866 was operating.

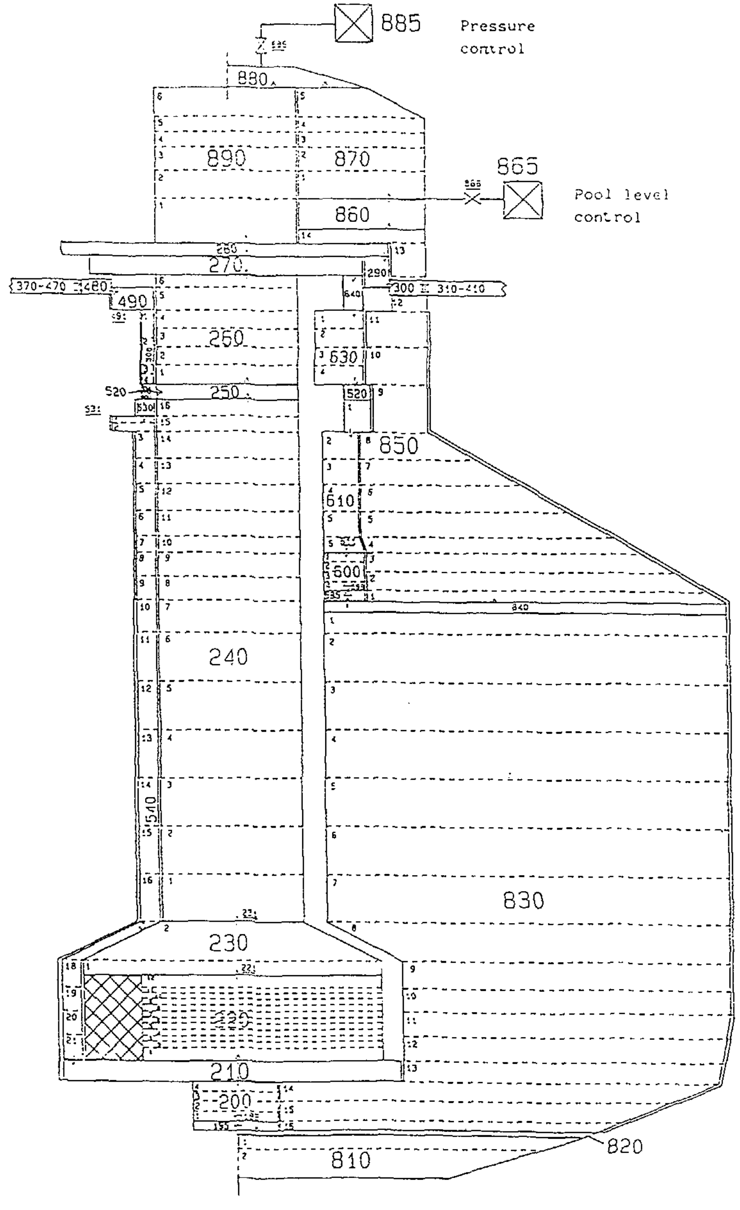

Fig. 2 - PIUS primary system nodalization

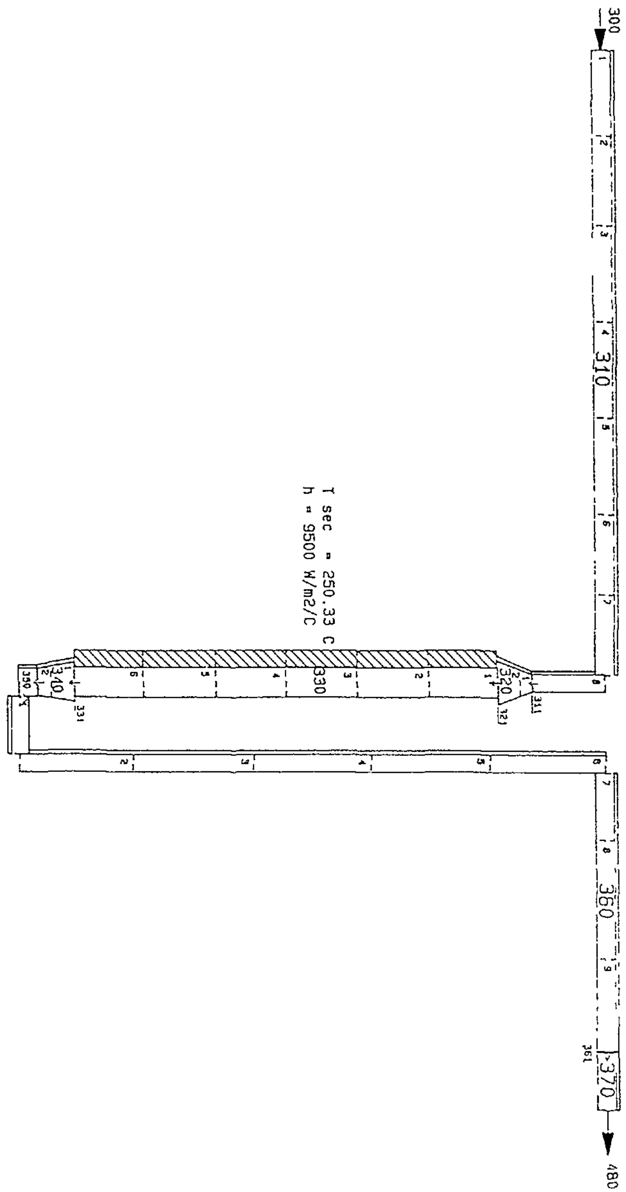

Fig. 3 - PIUS loop nodalization

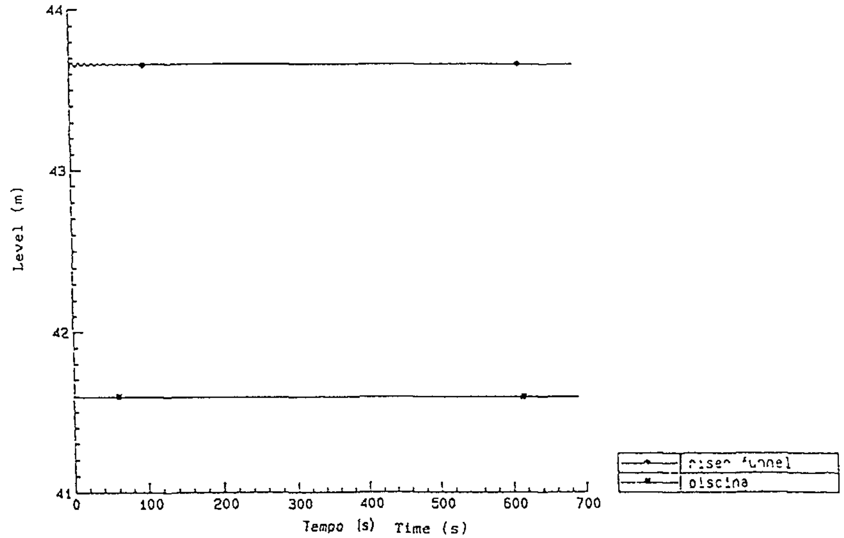

Fig. 4 - Pressurizer liquid level

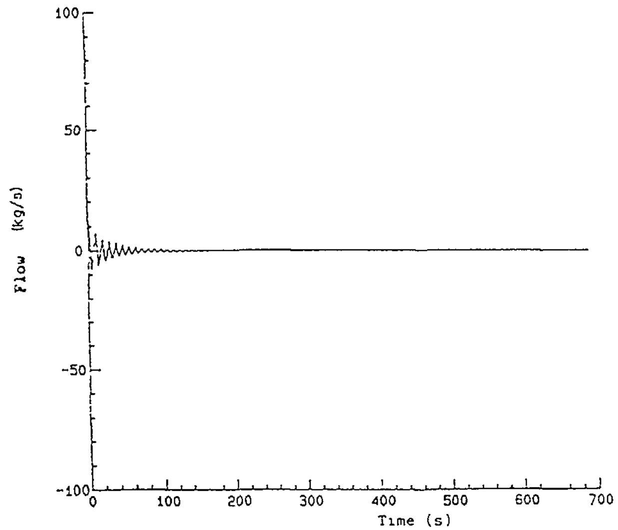

Fig. 5 - Flow through the lower density lock

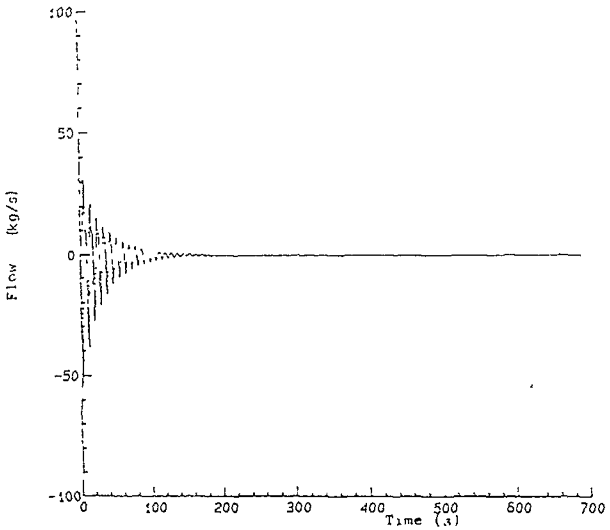

Fig. 6 - Flow through the upper density lock

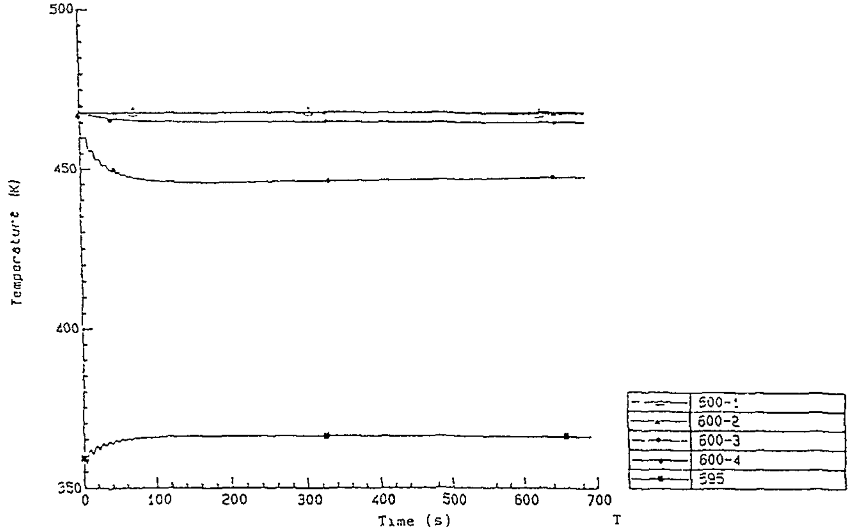

Fig. 8 - Temperature in the upper density lock

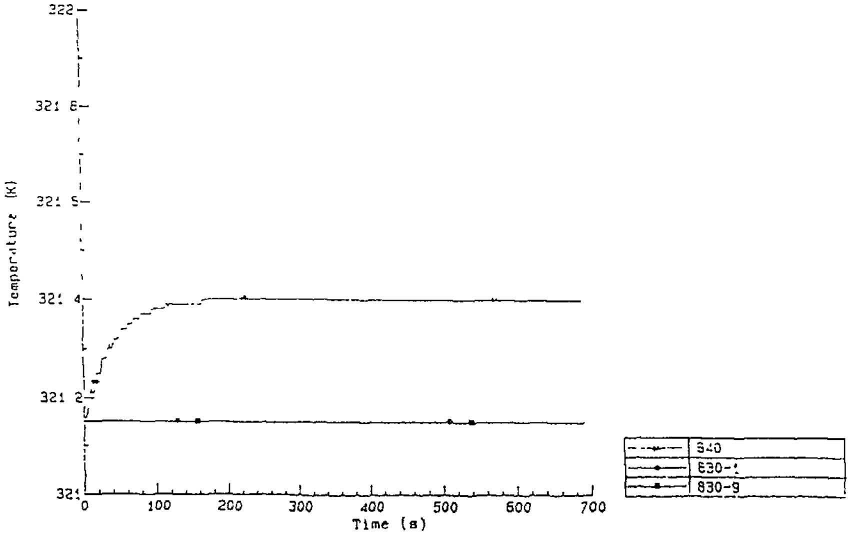

Fig. 7 - Pool temperature

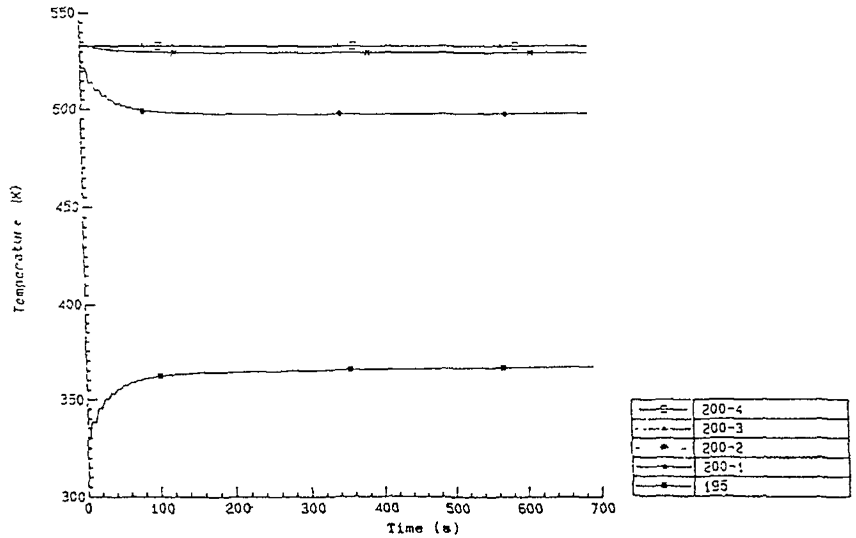

Fig. 9 - Temperature in the lower density lock

Tab I - Comparison between reference and calculated values for steady state condition

VARIABLES	CALCULATED VALUES				REFERENCE VALUES	
	WITH heat slabs		WITHOUT heat slabs			
Pressurizer level						
pool	41 59	m	39 85	m	40 8	m
- riser funnel	43 665	m	43 641	m	43 65	m
Core flow	13001	kg/s	13002	kg/s	13000	kg/s
Total flow	13602	kg/s	13601	kg/s	13600	kg/s
Bypass flow	600 28	kg/s	599 8	kg/s	600	kg/s
Loop 1 flow	3400	kg/s	3399 9	kg/s	3400	kg/s
Loop 2 flow	10202	kg/s	10201	kg/s	10200	kg/s
Inlet core temperature	260 02	°C	260	°C	260	°C
Outlet core temperature	289 91	°C	289 89	°C	290	°C
Inlet SG temperature	288 50	°C	288 52	°C	290	°C
Outlet SG temperature	260 05	°C	260 05	°C	260 3	°C
Inlet core pressure	9 3598	MPa	9 3325	MPa	9 27	MPa
Outlet core pressure	9 2955	MPa	9 2681	MPa	9 20	MPa
Loop 1 SG power	497 096	MW	499 964	MW	500	MW
Loop 2 SG power	1491 41	MW	1500 01	MW	1500	MW

This phase was necessary in order to

evaluate the influence of some uncertain parameters (both geometric and thermo fluid dynamic) on the system behaviour;

set uncertain parameters and conditions so as to obtain, at 200 s, the reference values of the main thermal hydraulic system parameters (pressuriser pressure, inlet and outlet core temperatures, power, liquid level in the riser funnel, flow rates through core, siphon breakers and loops, hot-cold interface level in density locks),

set uncertain parameters and conditions so as to reach steady state conditions and minimise the differential pressure across the density locks trip valves 199 and 599 and therefore reduce the flow rate that goes through the density locks as soon as they are opened

In the second phase, from 200 s to 500 s, the natural circulation path through the pool was activated by opening trip valves 199 and 599, pressure and liquid level controls in the pool were turned off and the system was let free to evolve to stationary conditions.

Once obtained steady state conditions by the model without heat structures, the same procedure was applied to the second model, which simulates all heat structures. The obtained steady state conditions do not differ significantly from those evaluated without heat structures.

The behaviour of the main thermal hydraulic variables after the opening of density locks trip valve (at 0 s in the case of the second model) are shown in Fig 4 through 9 and Tab 1 compares steady state conditions obtained by the two models to the reference values /2/

Moreover, the effects of opening times of density locks trip valves, pressure and pool level control systems and pool liquid level height on the system behaviour were studied The main results can be summarised as follows.

system behaviour is not affected if only the lower density lock trip valve is opened, while when only the upper density lock trip valve is opened relevant oscillations are observed in the flow rate through density locks, liquid level in the pressuriser and pressure throughout the reactor These phenomena can easily be explained by taking into account that the lower liquid inertia and friction forces that characterise the flow path through the upper density lock, riser upper plenum, riser funnel in the pressuriser and reactor pool induce a slower smoothing of disturbances with respect to the circulation path through the lower density lock,

steady state conditions are reached more quickly if the upper density lock is opened before the lower one In fact, when opening the upper density lock, only two parallel circulation paths are present in the system if the lower density lock is still closed, while, if the lower density lock has already been opened, the presence of three parallel circulation paths allows a greater and more complex disturbances propagation through the reactor system and reduces the oscillation smoothing velocity;

pool liquid control system helps to smooth oscillations of flow rate through density locks, of pressuriser liquid levels and pressures in all the system,

higher pool liquid level increases the dumping velocity of disturbances, but also increases the initial amplitude of flow oscillations through the density locks.

4. SEPARATE EFFECTS ANALYSES

4.1 EXPERIMENTAL FACILITY FOR CODE ASSESSMENT

The innovative features of PIUS plant require an accurate assessment of code capability in simulating particular phenomena like natural circulation, density locks, boron diffusion, etc In order to begin this assessment process an experimental test facility for the lower density lock simulation has been designed and some fluid dynamic analyses have been carried out using FLUENT code /4/

The experimental apparatus is shown in Fig 10, where T1 is the lower density lock model, C is the core and T2 is the storage tank used for costant driving force tests. For forced circulation conditions the other loop, with heat exchanger and pump, will be operated

The aims of the experimental studies are

to characterize the lower density lock behaviour in terms of pressure losses versus flow, from zero up to the natural circulation conditions,

to analyse the stability of hot-cold interface in operating conditions,

to analyse the transients during the phase from forced to natural circulation conditions, with regard to boron mixing and diffusion

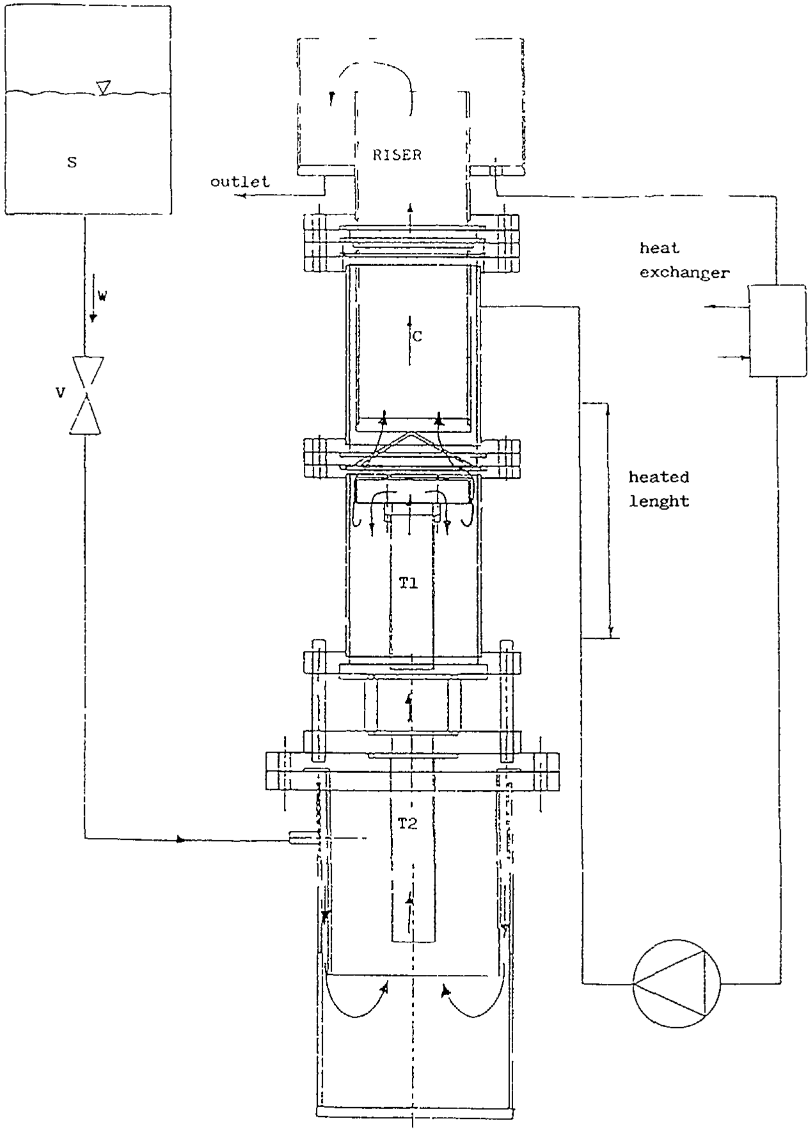

Fig. 10 - Experimental facility simulating the lower density lock of PIUS plant

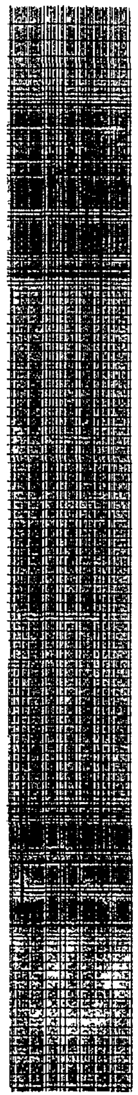
Fig. 11 - FLUENT model of lower density lock

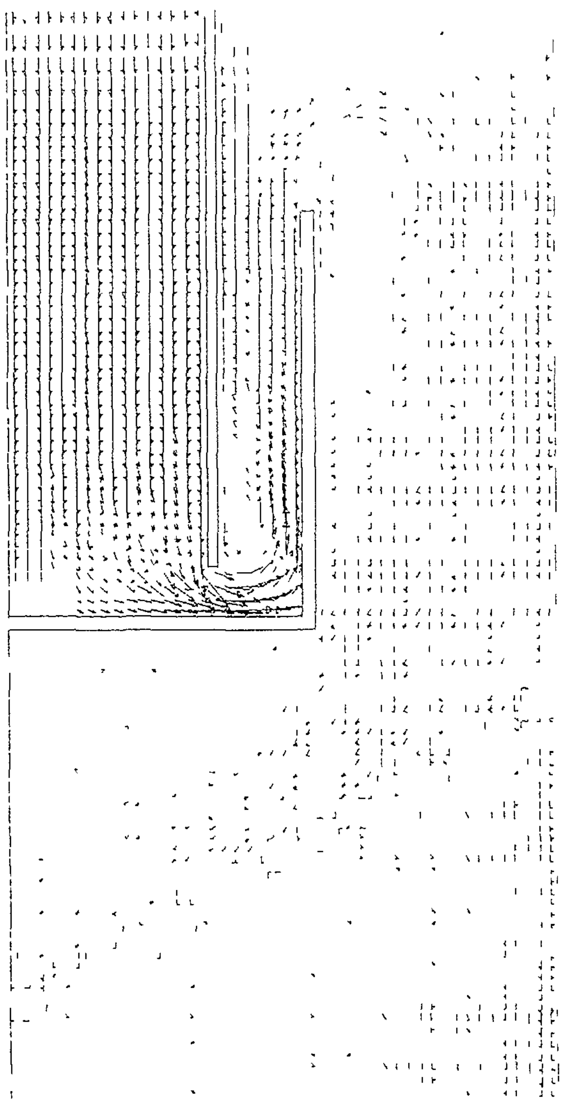

Fig. 12 - Local velocity distribution in the lower density lock calculated by FLUENT code

A bidimensional FLUENT model of this facility has been set up consisting of about 16000 cells Figg 11 and 12 show the grid and the calculated local velocity distribution.

5. CONCLUSIONS

The PIUS reactor represents an innovative approach to nuclear safety, even if it maintains established and proven LWR technology, then further analyses are required, especially in thermal-hydraulic field.

Up to now the application of RELAP5 to the evaluation of PIUS steady state condition demostrates the capability of the code in simulating the plant behaviour under forced and natural circulation operations.

ACKNOWLEDGEMENTS

Authors are grateful to Dr. P Gheri and Dr. G. Mariotti of ENEL-CRTN, who developed the FLUENT model of the experimental facility for lower density lock simulation and provided some preliminary results of their analyses.

REFERENCES

/1/ T. Pedersen, G M Crovato
"The Intrinsic Safety: Possible Break-through for the Nuclear Issue in Italy"
Energy Technologies from Italy - 1990, Publisher "L'Annuario" - Italy

/2/ V. H. Ramson et al.
"RELAP5/Mod2 Code Manual"
Idaho National Engineering Laboratory, March 1987.

/3/ PIUS 2000 - Preliminary Safety Information Document
ABB Atom - 1988

/4/ FLUENT
"User Manual v. 3 02"
Creare Incorporated, NH, 1990.

SIMULATION OF OPERATIONAL AND ACCIDENT TRANSIENTS OF THE PIUS REACTOR BY TRIP MODEL

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Abstract

A simplified but complete simulation model for describing the operational and accident transients of PIUS reactor has been implemented in the computer program TRIP Reactor accident transients are well coped with by the operation of the density locks, but those implying lower temperatures in the primary system, in this last case the plant behaviour is similar in the short run to that of conventional PWRs System stability does not appear to be affected by the density locks, but boron concentration seems an important parameter Reactor startup appears a feasible procedure to be better investigated and optimized

1 INTRODUCTION[1]

The PIUS (Process Inherent Ultimate Safety) is a well known reactor concept aimed at improving inherent and passive safety features of present light water nuclear reactors Starting from PWR (Pressurized Water Reactor) technology it uses two new components the density lock and a cold, pressurized and borated water vessel containing the reactor core and a portion of the primary circuit Its features are described in a number of publications

The analysis of this reactor concept concerns two aspects safety and operability As far as safety is concerned, large loss of coolant accidents (LOCA) do not appear the critical ones, being the density locks straightly activated by the strong pressure unbalances caused by these accidents In this case, the cold borated water, coming from the containment vessel through the lower density lock is capable to cool the reactor and shut down rapidly the neutron power On the contrary some doubts about a satisfactory inherent reactor behaviour may arise in the case of 'slow' accidents where the water injected through the density locks is smoothed down by the limited pressure unbalance across them Slow transients include loss of flow accidents (LOFA), due to pump loss of energy flow and temperature transients in the secondary side of the steam generator disturbance in the boron controlling circuit, slow depressurization of the primary circuit

As far as operability is concerned the hydraulic circuit behaviour may result rather 'soft', so as to enhance any possible disturbance in a way to facilitate borated water injection through the density locks in this case a steady state reactor power may result rather troublesome to obtain In this contest, startup operation seems to be a critical one, because the continuous change of equilibrium conditions going from zero to full power, might involve an oscillatory exchange between the water inside the primary circuit and that of the outside containment so as to impede a regular power augmentation

These considerations led us to the conclusion that a reactor computation model was necessary to better focus the quantitative aspects of these transients This model should be simple but complete, in order to obtain a flexible instrument for parametric calculations, to be performed by Personal Computers The slowness of the transients to be analyzed and the difficulty to reach and maintain two-phase conditions in the primary circuit (this supposed a priori, was thoroughly verified a posteriori) allowed us to adopt the hypotheses suitable for an incompressible fluid, and, in particular, to disregard the sound speed effects However, two-phase flow is taken into account in a simplified manner by applying the same equations adopted for the liquid phase, corrected for the increased specific volume

This model, called TRIP (TRansitori Impianto Pius), is detailed in this paper together with its applications to practical cases the conclusion so far reached about the above mentioned aspects will be briefly illustrated

2 THE THERMOHYDRAULIC MODEL

The thermohydraulic model, implemented in the TRIP program, is based on a subdivision of the circuit components in two categories those described by distributed parameters (constant cross section pipes including the reactor and the steam generators) and those described by lumped parameters (transition cross section between two different components density locks, plena pumps, containment vessel)

For distributed parameter analysis the three conservation equations of mass momentum and energy are written by applying different simplifying hypotheses The pipe component is assumed monodimensional disregarding the lateral exchange Then the three conservation equations, for a constant cross section pipe become as follows

$$\frac{\partial \rho}{\partial t} + \frac{\partial G}{\partial x} = 0 \qquad \text{(mass)} \qquad (1)$$

$$\frac{\partial G}{\partial t} + \frac{\partial}{\partial x}\left(\frac{G^2}{\rho}\right) = -\frac{\partial p}{\partial x} - f\frac{G|G|}{2D\rho} - \rho g \qquad \text{(momentum)} \qquad (2)$$

$$\rho\frac{\partial h}{\partial t} + G\frac{\partial h}{\partial x} = \frac{\phi\,\Pi}{\Omega} + \frac{\partial p}{\partial t} + \frac{G}{\rho}\left(\frac{\partial p}{\partial x} + f\frac{G|G|}{2D\rho}\right) \qquad \text{(energy)} \qquad (3)$$

where the unknowns are five G mass flux, ρ fluid density, p pressure, h specific enthalpy, f friction factor while the assigned variables are

[1] The conceptual part of this activity was carried out with the financial support of MURST (Ministero dell'Università e della Ricerca Scientifica e Tecnologica), computer program implementation and systematic validation were carried out under a research contract sponsored by ENEL/CRTN of Milan

Ω cross section area, D equivalent diameter, Π heated perimeter, ϕ heat flux, g is the gravity acceleration

Two further equations are given by fluid density state equation and the empirical correlation for friction factor

$$\rho = \rho(h,p) \quad , \quad f = f(h,p,G,\text{geometry}) \tag{4}$$

where the dependence on h comes about because of the need to determine the fluid viscosity μ The fluid is assumed incompressible, but thermally expandable, and calculated as it were at saturation condition, then

$$\frac{\partial \rho}{\partial p} = 0 \quad , \quad \rho = \rho(h)|_{p=p_{sat}} \tag{5}$$

In the energy equation, the terms connected to the pressure variation and the friction losses are neglected,

$$\frac{\partial p}{\partial t} = 0 \quad , \quad \frac{\partial p}{\partial x} = 0 \quad , \quad f\frac{G|G|}{2D\rho} = 0 \tag{6}$$

therefore the energy and the momentum equations are decoupled

In the mass conservation equation, the term $\partial\rho/\partial t$ is neglected, by assuming slow transient conditions

$$\frac{\partial \rho}{\partial t} = \frac{\partial \rho}{\partial h}\frac{\partial h}{\partial t} = 0 \quad \Rightarrow \quad \underset{\text{(thermal expandability)}}{\frac{\partial \rho}{\partial h} \neq 0} \quad \text{and} \quad \underset{\text{(slow transients)}}{\frac{\partial h}{\partial t} = 0} \tag{7}$$

This yields

$$\frac{\partial G}{\partial x} = 0 \Rightarrow G(x,t) = G(t) \tag{8}$$

then, the acceleration term in the momentum equation becomes·

$$\frac{\partial}{\partial x}\left(\frac{G^2}{\rho}\right) = G^2\frac{\partial}{\partial x}\left(\frac{1}{\rho}\right) \tag{9}$$

By substituting to G and ϕ the total flowrate Γ and the linear power w, i e

$$\Gamma = G\,\Omega \quad , \quad w = \phi\,\Pi \tag{10}$$

and taking into account the above mentioned simplifying hypotheses, the eqs (1) (2) and (3) become

$$\Gamma = \Gamma(t) \tag{11}$$

$$\frac{\partial p}{\partial x} = - f\frac{\Gamma^2}{2\Omega^2 D\rho} - \rho g - \frac{\Gamma^2}{\Omega^2}\frac{\partial(1/\rho)}{\partial x} - \frac{1}{\Omega}\frac{\partial \Gamma}{\partial t} \tag{12}$$

$$\rho\Omega\frac{\partial h}{\partial t} + \Gamma\frac{\partial h}{\partial x} = w \tag{13}$$

Obviously Γ can vary stepwise, connected with lumped parameter components, and in particular when some fluid is injected in or extracted from the circuit, as through the density locks and the pressurizer

For lumped parameter components, a zero dimension solution of mass, momentum and energy equations is carried out, as follows

$$\frac{dM}{dt} = \Gamma_{in} - \Gamma_{out} \quad \text{(mass)} \tag{14}$$

$$p_{in} - p_{out} = \rho g\Delta z + \frac{d\Gamma}{dt}\frac{L}{\Omega} + K\frac{\Gamma^2}{2\rho\Omega^2} + \frac{\Gamma^2}{2\rho}\left(\frac{1}{\Omega^2_{out}} - \frac{1}{\Omega^2_{in}}\right) \quad \text{(momentum)} \tag{15}$$

$$\frac{dU}{dt} = \Gamma_{in}h_{in} - \Gamma_{out}h_{out} \quad \text{(energy)} \tag{16}$$

where Ω, L, Δz and K are the component cross section, length, elevation, and total form loss coefficient respectively, while M and U are the mass and the internal energy of the fluid contained inside the component When the volume V is constant, another conservation equation is given by

$$\frac{dV}{dt} = \frac{d(M/\rho)}{dt} = 0 \quad \text{(volume)} \tag{17}$$

The time derivative of internal energy is equal to

$$\frac{dU}{dt} = \frac{dH}{dt} - p\frac{dV}{dt} - V\frac{dp}{dt} \tag{18}$$

where H is the available fluid enthalpy The second term on the right side of the equation is zero for all components with constant volume, and for those where it is different from zero (the funnel and the containment), it is a small quantity in any case, for this reason it is always neglected The third term is really small for the accidents which can be analyzed by this program (i e accidents where pressure changes slowly), and it is neglected as well

3 THE SOLVING METHOD

The solving procedure is based on the usual finite difference method, once a space-time grid is defined For the time derivatives of the above mentioned equations the grid is obtained by a fixed time interval Δt Then the solving procedure differentiates according to lumped or distributed parameter components

The lumped parameter components are subdivided in four categories I) transition cross section components and pumps, II) plena, III) density locks, IV) containment The transition component cross section represents the boundary between two adjacent components, their axial length is zero

For first category components, only the momentum equation (15) is solved, by modifying the pressure by an amount equal to the concentrated friction pressure drops and the kinetic terms (the head term in the case of the pump)

For all other components the calculation procedure is as follows (see eqs (5), (14), (16) and (17), taking into account the simplifications indicated at the end of the previous paragraph)

$$M^{n+1} = M^n + (\Gamma^{n+1}_{in} - \Gamma^{n+1}_{out})\,\Delta t \tag{19}$$

$$h^{n+1} = \frac{1}{M^{n+1}}\left[M^n h^n + \left(\Gamma^{n+1}_{in}h^n_{in} + \Gamma^{n+1}_{out}h^n_{out}\right)\Delta t\right] \tag{20}$$

$$\rho^{n+1} = \rho(h^{n+1}) \tag{21}$$

$$\frac{M^{n+1}}{\rho^{n+1}} = V = \frac{M^n}{\rho^n} \tag{22}$$

where the superscripts n and n+1 represent the variable value at time t and t+Δt respectively Γ_{in}^{n+1} is known, as explained below, while Γ_{out}^{n+1} h^{n+1}, ρ^{n+1} and M^{n+1} are obtained by solving the above four equations by a rapidly converging iterative procedure Then the outlet pressure is determined by the momentum equation (15) as

$$p_{out}^{n+1} = p_{in}^{n+1} - \rho^{n+1} g\Delta z - \frac{\Gamma_{out}^{n+1} - \Gamma_{out}^{n}}{\Delta t}\frac{L}{\Omega} - \frac{\Gamma_{out}^{2}}{2\rho^{n+1}}\left(\frac{1}{\Omega_{out}^{2}} - \frac{1}{\Omega_{in}^{2}}\right) \tag{23}$$

For the density lock component the above equations are applied by taking into account the presence of a separation level between cold and hot water (see next paragraph) The inertia term of eq (23) is particularly important in this component, where the flowrate may undergo sharp accelerations, due to continuous changes in direction Some simplifications are adopted for the containment, as explained in the next paragraph

For distributed parameter components the energy equation is solved by adopting the "method of characteristic", recalling that

$$\frac{\partial h}{\partial x} = \frac{dh}{dx} - \frac{\partial h}{\partial t}\frac{dt}{dx} \tag{24}$$

$$\frac{dx}{dt} = v \tag{25}$$

where v is the fluid speed, and substituting eqs (24) and (25) in eq (13), one obtains

$$\frac{dh}{dx} = \frac{w}{\Gamma} \tag{26}$$

which is solved by finite difference method, along the characteristic line, defined by eq (25)

The variables at the time t^{n+1} are calculated by an implicit method The space-time grid is obtained by the same fixed time interval Δt of lumped parameter components, and a variable space intervals Δx ("mobile grid), as defined here below

The eqs (25) and (26) turn into

$$\frac{\Delta x}{\Delta t} = \langle v \rangle \tag{27}$$

$$\frac{\Delta h}{\Delta x} = \frac{\langle w \rangle}{\langle \Gamma \rangle} \tag{28}$$

where <v>, <w>, <Γ> are the values averaged along the characteristic line, by assuming a linear trend The stability condition which is equal to

$$\Delta t \le \min\left\{\frac{\Delta x}{|v|}\right\} \tag{29}$$

is satisfied by eq (27), which defines the value of Δx interval, variable in time and space The subdivision of pipe components according to this mobile grid, is shown in fig 1

The solving equations become

$$\frac{x_p^{n+1} - x_s^{n}}{\Delta t} = \langle v \rangle \quad , \quad \frac{h_p^{n+1} - h_s^{n}}{x_p^{n+1} - x_s^{n}} = \frac{\langle w \rangle}{\langle \Gamma \rangle} \tag{30}$$

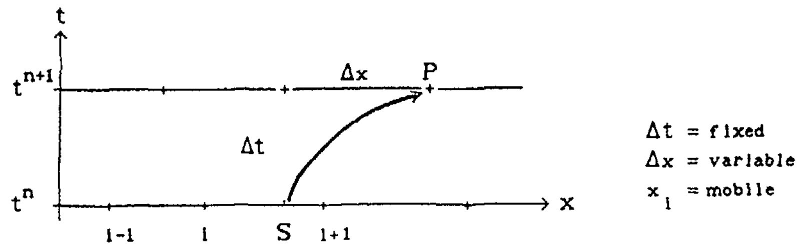

Fig.1 - Subdivision of pipe components according to mobile grid.

subscripts P and S refer to the corresponding points of fig 1 <Γ> is constant versus x, as previously assumed (eq (11)), <w> is obtained by the neutron kinetic equation and/or heat transfer equation, <v> is the arithmetic mean of the values obtained in S and P (see fig 1)

$$\langle v \rangle = \frac{v_p^{n+1} + v_s^{n}}{2} \tag{31}$$

and h_s^n is obtained by a linear interpolation between the corresponding values in x_i^n and x_{i+1}^n

The space interval Δx is determined by <v>, which depends on Δx through the enthalpy h_p^{n+1} and the corresponding ρ_p^{n+1} Thus, a short iterative process is implemented on the relative speed difference, within a fixed convergence error (typically 10^{-4})

The momentum equations (12) and (23) are integrated spatially, by using for distributed parameter components the same space grid assumed for the energy equation, and by adopting in each component the fluid density and the viscosity values as obtained by the energy balance The friction term of eq (12) includes both the distributed pressure drops and the concentrated ones (if present)

The integral of ∂p/∂x extended to the whole circuit must be nihil, being the initial cross section coincident with the last one This is obtained by an iterative procedure on Γ, which remains the last unknown to be determined

In conclusion, at t^{n+1}, Γ is assumed initially equal to the value of the previous time step t^n, then, the equations (30) and (19) through (22) are solved, with the resulting ρ and μ values the eqs (12) and (23) are integrated and the value of integral of ∂p/∂x is determined, if this integral is not zero, within a prefixed convergence error (typically 10^{-5} kPa), the procedure is repeated, by assuming a more correct Γ value, by adopting the null method The procedure is repeated until the convergence is reached

According to the hypotheses done, the solution is affected by an error, which, in particular, reflects on a difference between the space integrals of mass and energy ("photographic) and the corresponding time integrals ("historic"), normalized to the same initial conditions Assuming binding the historic integrals, the photographic ones are continuously corrected, by adding in the circuit, in each time step, the historic-photographic difference, this difference should be distributed along the whole circuit, but for the sake of simplicity it is introduced in the pressurizer

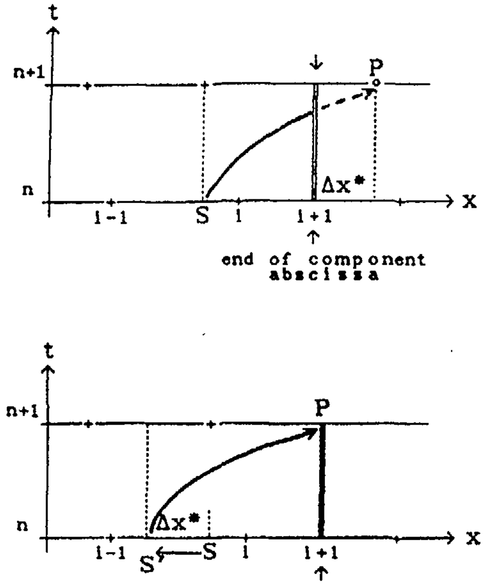

Fig.2 - Procedure to shift upstream point S in order to coincide with component end.

The two-phase mixture, when present, is treated in a simplified way. The steam quality is determined by the excess of fluid enthalpy over the saturation one, which is checked in every cross section of the cell ends or inside the lumped parameter components, by referring to the pressure calculated for the previous time step. The mixture specific volume is calculated by the homogeneous model and assuming slip ratio equal to one. Pressure drops are obtained by using the same equations adopted in single phase, corrected for the mixture specific volume, without referring for the time being to specific correlations valid for this condition.

The mobile spatial grid implies that the last cell of each distributed parameter component does not, in general, coincide with the actual component end. Thus, the abscissa of the starting point S is shifted upstream, as indicated in fig.2, until the convergence is obtained.

The boron moves along the distributed parameter components coherently with flowrate, without any axial mixing; when it enters a concentrated parameter component it is instantaneously mixed in the whole volume; boron distribution along the density lock is described in the next paragraph.

4. THE CIRCUIT COMPONENTS

The PIUS primary circuit is discretized as shown in fig.3. In particular, the four external circuits are lumped together, so that no dissymmetry effects between the four circuits can be described. This hypothesis should be beared in mind, when LOFA, concerning less than four pumps, are analyzed.

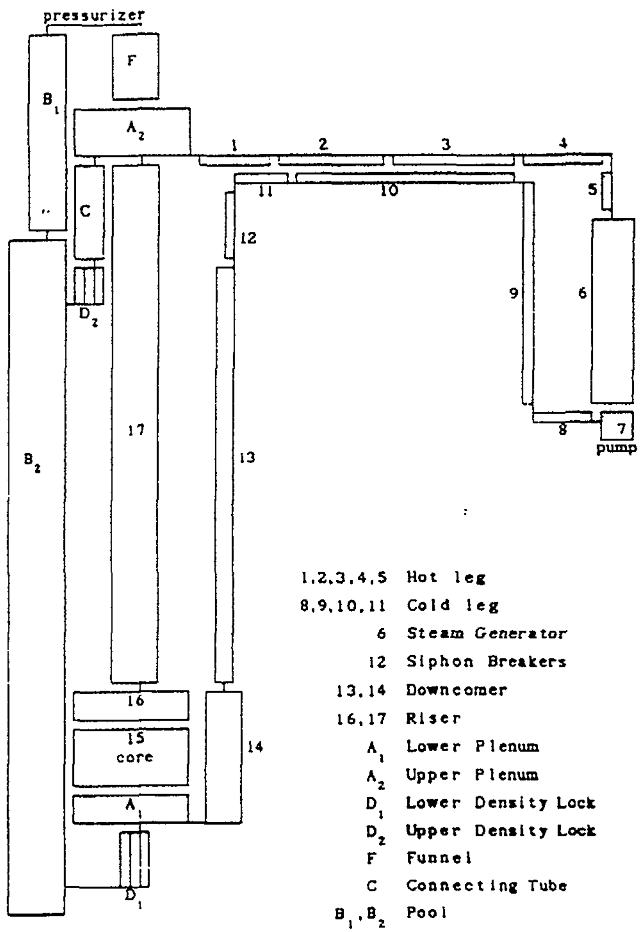

Fig.3 - Discretization of PIUS reactor system in TRIP model.

Some components require a special modelization i e the reactor, the steam generator, the pump, the density lock the siphon breaker the pressurizer and the containment The reactor core model is rather complex and it will be presented in the following paragraph

The power exchanged in the steam generator is calculated by assigning the temperature of the secondary fluid This is uniform along the whole steam generator and it corresponds to the steam saturation temperature, its value might not be constant with time, but, in this case, it is to be assigned as an input The overall total heat resistance is normalized in order to exchange the nominal reactor power with the actual heat exchange surface Its value is then subdivided in two terms a constant one, representing the thermal resistance of the tube thickness, the crud deposition and the secondary side boiling, and a variable one, representing the primary side heat transfer, which is proportional to the flowrate to an exponent of 0 8

The pump characteristic is assumed having the following form

$$\frac{H}{H_0} = A_1\left(\frac{Q^2}{Q_0}\right) + A_2\frac{n}{n_0}\frac{Q}{Q_0} + A_3\left(\frac{n^2}{n_0}\right) \tag{32}$$

where H, Q, n are the head, the volumetric flowrate and the revolutions per minute, respectively, while A_1, A_2, A_3 are constants, characteristic of the pump and subscript (o) indicates the nominal values

The pump is described in three different conditions normal condition, loss of energy condition, cavitation condition

In normal condition, the pump volumetric flowrate is obtained by the value of Γ calculated during the procedure previously explained, while the revolutions per minute are constant during the time step Their value is calculated by setting to zero the lower density lock flowrate, or optionally setting within a prefixed interval the interface level between cold and hot water in the lower density lock, always referring to the previous time step conditions Therefore, at the end of each time step, the rotation speed is adjusted through an approximate procedure The maximum speed cannot exceed 5% of the nominal speed, no pump inertia effect is taken into account in this adjustment In loss of energy condition, the pump decay speed is calculated by solving the usual dynamic equation, where the pump is characterized by an overall rotational inertia, which is assigned, and by an efficiency kept constant during the transient Loss of energy may involve 1, 2, 3 or 4 pumps of the circuit The model takes into account the behaviour of a system, having in parallel some pumps regularly fed by external energy, and some pumps without power supply Cavitation condition is obtained when the assigned NPSH (Net Positive Suction Head) value is reached at the pump entry In this case the head is set to zero

The density lock is a vertical cylinder filled by a bundle of open tubes At a certain elevation, there is a separation interface between hot and cold water Across this interface, the water temperature and boron concentration undergo a stepwise change no turbulence mixing is here considered

Density lock flowrate is determined through eq (23), where the inertia term is slightly increased to take into account the contribution of the movement of the water mass surrounding the density lock The density lock flowrate, when entering the plenum to which it is connected, is instantly mixed with all the water contained in the plenum itself The entering water remains at the conditions (temperature and boron concentration) existing above the density lock separation interface, till this is displayed beyond the highest section of the density lock, beyond this point, cold and borated water (containment water) enters the plenum When the water is flowing in the direction from the primary circuit to the containment, the conditions above the interface are obtained by an instantaneous mixing between the entering water and that already existing in this zone

The siphon breaker is simulated as a localized pressure drop No coolant bypass, between riser and downcomer, is taken into account, being its effect probably limited, but really complicated to describe

The pressurizer is kept at constant pressure in any condition, and this should imply a small effect in the transient to be studied, however its value may change against time, if defined in input The pressurizer part connected to the main circuit is divided in three volumes (see fig 3) the funnel (F), the upper plenum (A_2), the vertical annular connection to the upper density lock (C) Each volume is treated separately, according to the above said procedure adopted for plena During startup operation, A_2 and C enthalpies may be quite different and in this case a rather complex heat transfer process should take place, mainly between the downcomer (12) and volume C For the time being, a simplified model is adopted, which implies only a heat exchange between A_2 and C, calculated by a term proportional to their temperature difference (the temperature is uniform in each volume according to our model)

The funnel and the containment levels are correctly calculated, by considering any water spill over between the funnel and the containment and the density lock flowrates The containment water temperature and boron concentration are assumed constant in space and time Therefore, the containment is assumed with its actual volume for level calculation and with an infinite volume for enthalpy and boron calculation

5 THE REACTOR CORE

The reactor core modelization presents three aspects channel thermohydraulic, fuel rod behaviour, and neutron power kinetic

From a thermohydraulic viewpoint the core may be approximately described by a bundle of channels, assumed separated, each one containing one fuel element The channels, with different power inputs, have also slightly different flowrates, in order to satisfy the boundary condition to obtain the same core pressure drop During transient conditions, a steam-water mixture may be obtained in the hottest channel and this would imply a slightly higher asymmetry in channel flowrate distribution and above all a "void" generation, which produces a negative reactivity feedback A thorough description of this thermohydraulic configuration is really too complicated for our program aims and then a simplification is adopted

The reactor core is described by a single power channel representing the core average channel This is easily modelized as a 'pipe of constant cross section, with a power input calculated by the fuel rod subroutine However, at the end of each time step, keeping constant the already calculated core boundary conditions, the parallel channel behaviour is determined out of line Three channels are examined, each one representing those of the central, intermediate and peripheral zone In this way, a more precise temperature and void distribution is calculated, which determines the reactivity feedback data This out of line calculation follows an iterative procedure to correct the flowrate of each channel, for the same overall flowrate, in order to converge on the same pressure drop Two hypothetic conditions were calculated saturation conditions are obtained at the outlet of the average channel, either halving the nominal flowrate or doubling the nominal power The negative reactivity feedback, due to moderator temperature and void, is calculated for the average channel or as the average of the three channels 512 pcm against 1369 pcm and 779 pcm against 2150 pcm are obtained for the first and second case respectively This example clearly shows the importance, for the reactivity effects, to consider three different channels in parallel Actually, the overall effect on the transient is damped out substantially, because a higher void reactivity feedback reduces more rapidly the reactor power, which, in its turns, reduces the void formation (inherent feedback phenomenon)

The fuel rod temperature distribution is described only in the radial direction The cross section is subdivided in five annular zones, three in the fuel, one in the gap and one in the cladding, and their temperature are the unknowns The general Fourier equation was discretized by referring to these five temperatures The boundary conditions are the power generated in each zone which is proportional to its area (coming from the neutron kinetic subroutine), the coolant temperature and the corresponding heat transfer coefficient In conclusion, five linear algebraic equations are obtained, the solution of which yields the desired temperature distribution and then the thermal power transferred to the coolant, and a suitable mean fuel temperature for Doppler reactivity calculation

The conductivities and the specific heats of zircaloy and uranium oxide are taken as temperature dependent, while their density is assumed constant Gap conductivity is supposed to be burnup and linear power dependent an empirical correlation based on a rather complex fuel performance model was developed

This model is applied to the average fuel cross section of the reactor The coolant boundary values are taken at the average conditions as well This conclusion was reached after some calculations showed that the adoption of several fuel cross sections (six for each of the three above mentioned channels) yielded a reactivity feedback value which was only few per cent different from the single cross section value

The neutron power is calculated out of line, at the end of each time step, the resulting value is the power input for the calculation of the fuel rod temperature distribution in the next time step The reactivity is calculated as the sum of four terms due to fuel temperature (Doppler effect), moderator temperature, moderator voids, boron coolant concentration Each one is calculated by multiplying the corresponding parameter variation with respect to a reference condition, by a given input constant (reactivity coefficient) As for Doppler reactivity, an effective fuel temperature is defined as

$$\theta = \ln\left(\frac{4}{9}\,\theta_0 + \frac{5}{9}\,\theta_2 + 669\right) \tag{33}$$

where θ_0 and θ_2 are the fuel center and peripheral temperature respectively

The neutron kinetic equations (Nordheim-Fuchs) are written for six delayed neutron groups and solved by the simplified Runge-Kutta method, according to the procedure indicated in ref [1] The solution is obtained by a time step shorter then the one used in the thermohydraulic part of the program (typically 10 ms against 100 ms) Therefore, the neutron kinetic integration is carried out with constant input data during the whole thermohydraulic time step The delayed neutron decay time constants and their fractions are input data, burnup dependent When a reactor moderator parameter (temperature, void, boron) is averaged, this can be done weighing the quantity both linearly or on the square of the actual neutron flux (axially and radially)

Present version of simulation program TRIP-2 is implemented in about 6000 statements, subdivided in 34 subroutines The program is currently run on a 386-20 Mhz PC(IBM Compatible) Significant improvements of TRIP model are in progress, namely pressurizer steam generator and axial core kinetics

6 ACCIDENT TRANSIENTS

The PIUS steady state condition has been obtained by the TRIP-2 program as the asymptotic result of a transient having input data equal to the nominal ones, but the flowrate This last one is obtained by this calculation, coherently with the assumed distance between the two density locks and the condition of zero flowrate through them Then the density lock distance can be adjusted to obtain the nominal flowrate

Different incidents and accidents have been simulated and namely loss of flow of 1 to 4 primary pumps, uncontrolled boron dilution, primary depressurization by a fixed rate, loss of flow of feed water pumps, steam line break, bypass of steam cycle preheaters and feed water flow increase These last four have been simulated by assuming prefixed transients for the heat transfer coefficient, and feed water temperature in the steam generator

A typical transient of a loss of flow in two primary pumps is detailed in fig 4, where the resulting main circuit parameters versus time are shown

The analysis so far carried out may be synthesized as follows

- In all transients, but those relevant to reheaters bypass, the density locks work as expected, thus showing a true passive behaviour The power may initially increase, anyway to a limited extent, but the asymptotic value is always lower than the nominal one, including in many cases the zero value
- Bypass of steam cycle preheaters and feed water flow increase implies a progressively reduced feed water temperature In this situation the PIUS behaviour does not seem substantially different from a conventional PWR one, being the lower density lock not activated However, borated water injection through the upper density lock gives, in the long run, some advantages in coping with cold water reactivity effects
- The density locks, once activated during a slow transient, show basically an oscillating behaviour In fact, as soon as the cold water enters the primary circuit and it displays along the reactor core and the riser, the driving head is progressively lowered, then opposing any further water entrance In these conditions, the driving head increases (the water heats up in the primary circuit) and the cold water enters again the primary circuit and this oscillation goes on until the transient is terminated
- An increase of density lock flow resistance appears effective both in improving safety transients (the final reactor power is lowered) and in dampening the above mentioned oscillating behaviour
- Steam-water mixture is difficult to obtain for long periods of time inside the primary circuit, because as soon as it is formed the density locks are quickly activated, and high flowrates of cold and borated water is injected inside the primary circuit, thus collapsing rapidly the existing steam and shutting down the reactor This result justifies our choice, done a priori, to disregard any sound speed effect
- During normal operation the reactor shutdown can be obtained by simple operations on the circuit components, through purely thermohydraulic phenomena
- In general, it seems that the density locks render the system safety behaviour rather insensitive to many design parameter values, thus giving the designer a certain freedom in their definition

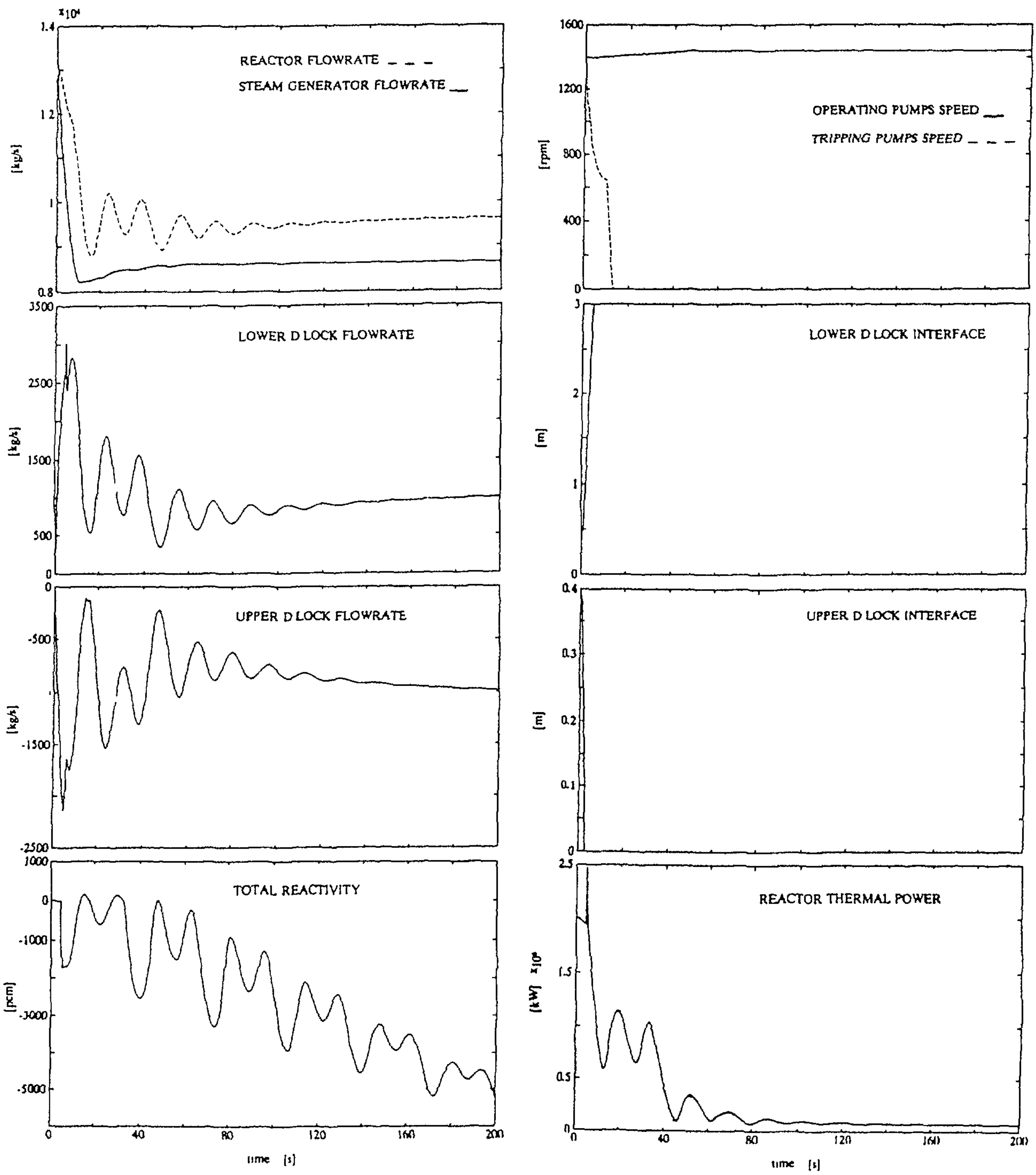

Fig.4 - Typical system parameter trends vs. time in the case of loss of flow of two out of four pumps.

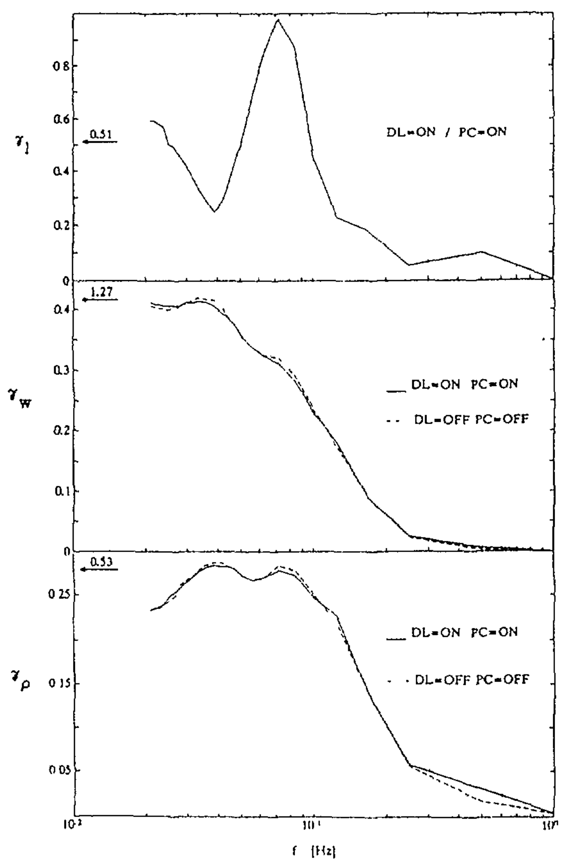

fig -5- Relative amplitude of reactor power, lower density lock level, reactivity divided by relative amplitude of steam generator secondary side temperature disturbance versus frequency (DL=density lock, PC=pump control)

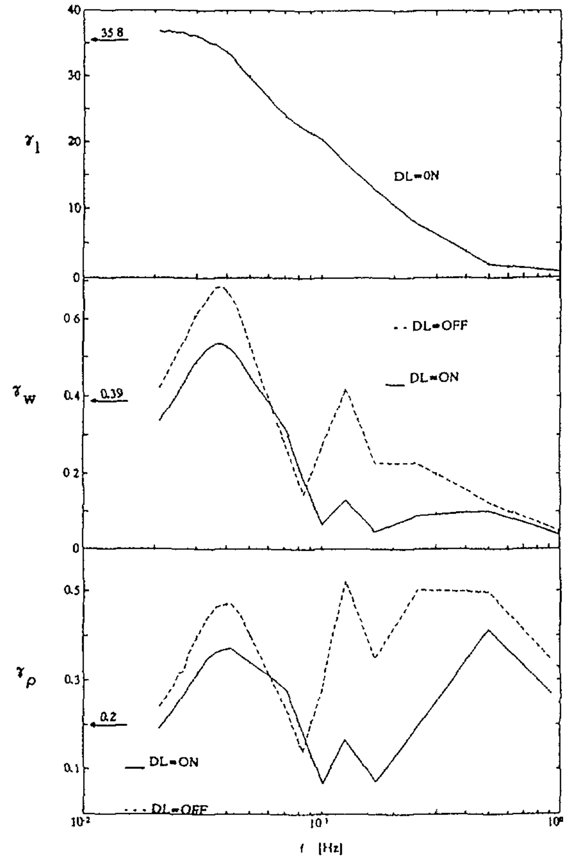

fig -6- Relative amplitude of reactor power, lower density lock level, reactivity divided by relative amplitude of pump speed disturbance versus frequency (DL=density lock).

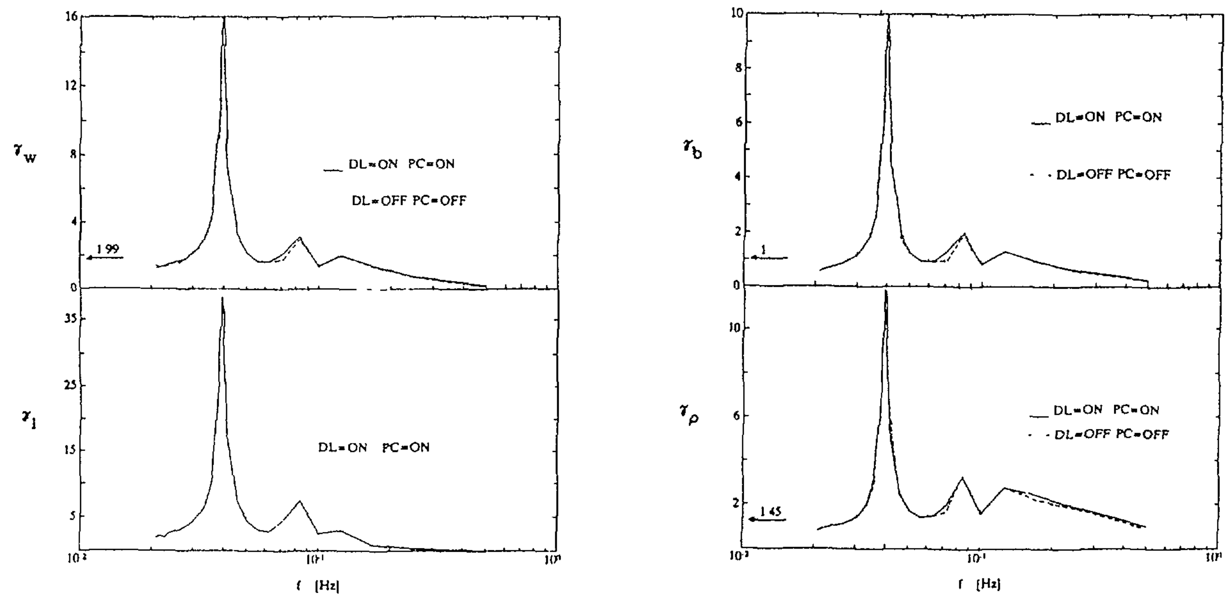

fig -7- Relative amplitude of reactor power, lower density lock level, reactor boron concentration and reactivity divided by relative amplitude of boron concentration disturbance at pump exit versus frequency (DL=density lock, PC=pump control).

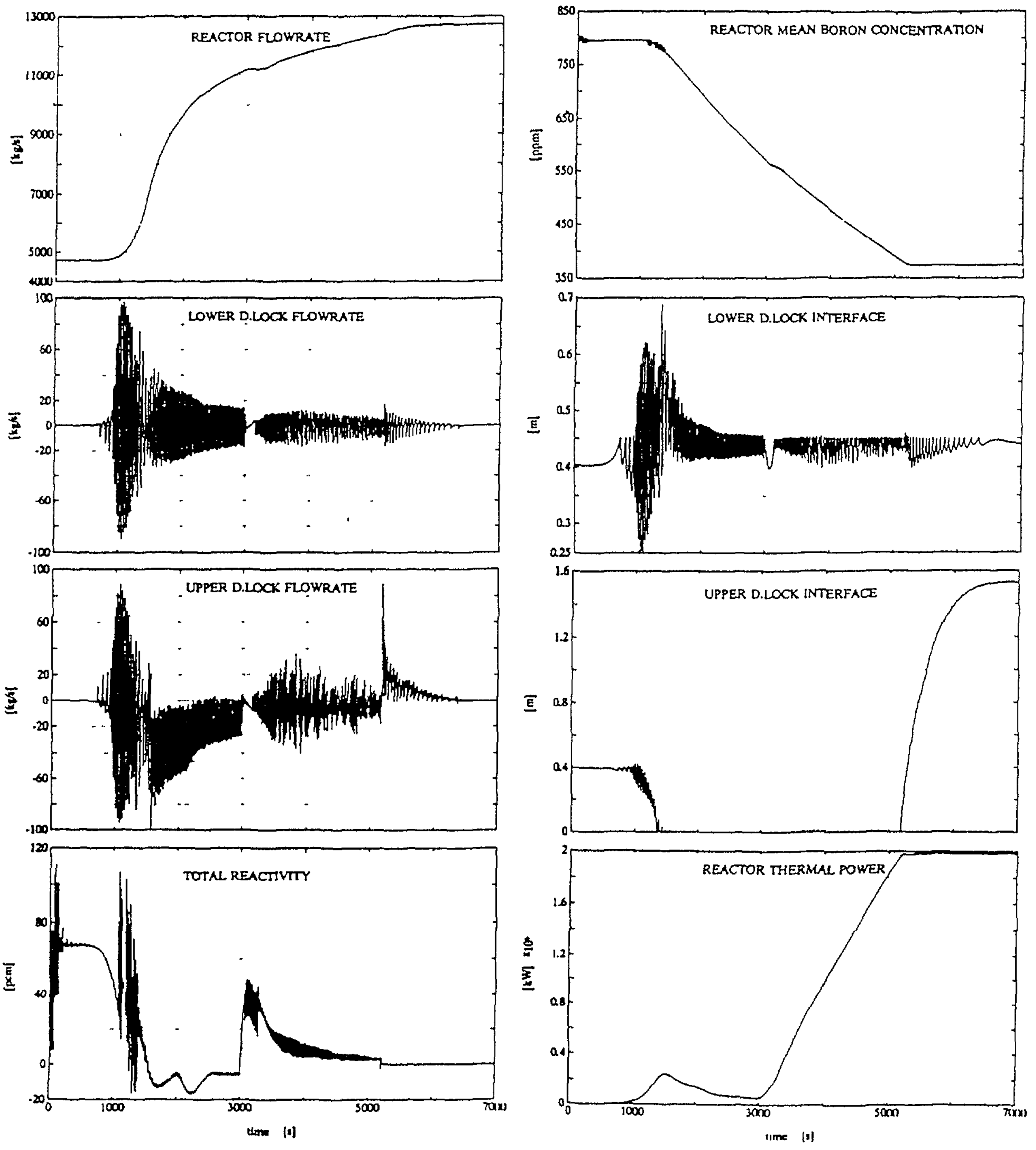

Fig.8 - Typical system parameter trends vs. time during reactor startup (actual startup duration is longer than the showed one).

7 SYSTEM STABILITY

System stability has been studied by analyzing the effect of a given parameter disturbance, varying sinusoidally with an assigned frequency, on the other main parameters, mainly reactivity reactor power, and lower density lock interface elevation The results have been synthesized plotting the relative amplitude of the parameter under consideration divided by the relative amplitude of the parameter purposelv varied (disturbance) versus the frequency of this last one (symbol γ) In order to understand the density lock effect, the analysis has been repeated by closing the density locks, condition which should be rather near that typical of a conventional PWR Obviously in this case the control of the primary pump speed has been disconnected

Up to now the following parameters have been varied temperature of the steam generator secondary side (uniform in our model), pump speed (pump speed control is out in both cases), boron concentration The results are shown in figs 5,6 and 7 on each plot the stationary value of γ (frequency equal to zero) is indicated on the ordinate scale

From these plots the following conclusions may be drawn

- Temperature of the steam generator secondary side the density lock effect is negligible under all circumstances Reactor power and reactivity are always lower than the steady state ones (1/3-1/2), while interface level (in the case of density lock open) shows a limited resonance in correspondence of $f= 0\ 07s^{-1}$, almost two times the steady state value
- Pump speed in this case the density lock effect is not *negligible*, but they stabilize the system, because they are able to absorb part of the flowrate variation by changing their interface level Reactor power and reactivity show a limited resonance, roughly two times the steady state value
- Boron concentration the density lock effect is negligible under all circumstances However, in this case a rather marked resonance (an order of magnitude) is present at $f= 0\ 04s^{-1}$, while progressively lower resonances are evident at multiples of this frequency The coolant transit time is in our case equal to about 25s, equivalent to the above mentioned frequency

In conclusion, the PIUS stability does not appear to be affected by the presence of the density locks Boron concentration seems an important parameter for stability behaviour, the effect of which should be better investigated

8 STARTUP OPERATION

Reactor startup from cold conditions (100°C) to full power takes place by lowering the boron concentration in the primary circuit, leaving the density locks open This has been obtained by injecting a fixed flowrate of clean water at the reactor inlet (60 kg/s) This flowrate is started or interrupted when the reactivity is lower or higher than 30 or 40 pcm respectively (40-50 initial 1000s) The steam generator starts extracting thermal power only when the primary temperature exceeds the nominal secondary temperature (248 5°C) *Because of unavailable data about the pool* and the funnel level control systems, only the pump speed one is simulated by TRIP, having as control variable the lower density lock level (reference value 0 4±0 05m) for this reason the startup asymptotic values are not exactly equal to the nominal values calculated by TRIP The results are synthesized in fig 8 From these plots it may be concluded that the startup procedure appears feasible, but the long transit time of boron content in the circuit (from 75s to 25s according to flowrate) may promote reactivity oscillations Moreover, the thermal power does not show a permanent monotonic behaviour, appearing a single long oscillation around 150 MW Thermal power extraction in the steam generator results stabilizing on the overall procedure (t≥3000 s on the graphs) then it seems advisable to anticipate its operation at lower primary temperatures, as typically occurs in PWRs

NOMENCLATURE

Symbols

D	equivalent diameter
f	friction factor frequency
g	gravity acceleration
G	mass flux
h	specific enthalpy
H	total enthalpy, pump head
K	total form loss coefficient
L	component length
M	fluid mass
n	pump revolutions per minute
p	pressure
Q	volumetric flowrate
t	time
U	internal energy
v	fluid speed
V	component volume
z	elevation
x	axial coordinate
w	linear power
Δ	difference
ϕ	heat flux
γ	relative amplitude of a parameter/ relative amplitude of disturbance
Γ	total mass flowrate
Π	heated perimeter
Ω	cross section area
ρ	fluid density
μ	fluid viscosity

Subscripts

b	boron concentration
i	generic space grid position
in	inlet
l	density lock interface elevation
o	nominal value
out	outlet
p	point P of fig 1
s	point S of fig 1
sat	saturation
w	reactor power
ρ	reactivity

Superscripts

n	at time t
n+1	at time t+Δt

REFERENCE

1 Yung-An Chao, A Attard *A Resolution of the Stiffness Problem of Reactor Kinetics* - Nuclear Science and Engineering, N 90 1985

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