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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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PASSIVE SYSTEM RELIABILITY - A Challenge to Reliability Engineering and Licensing of Advanced Nuclear Power Plants

Proceedings of an International Workshop hosted by the Commissariat à l'Energie Atomique (CEA)

Held in Cadarache, France, 4th- 6th March 2002

English text only

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 28 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

ABSTRACT

The workshop provided a forum for the exchange of information on the technical issues associated with assessing the reliability of passive systems in the context of nuclear safety, regulatory practices and probabilistic safety analysis. Special emphasis was placed on the reliability of the systems based on thermal hydraulics, for which the methods are still in developing phase.

Issues and discussions topics included lessons learned from designing passive systems, developing methodologies, performed studies, field experience with passive systems and need for future development.

These proceedings provide a compilation of the papers presented and a summary of the discussions.

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FOREWORD

In March 1999, a proposal was made by Finland to perform a survey of concerning the reliability of passive system (RoPS). The main objective was to map methods developed for passive systems based on thermal hydraulics. The goal was to support developing Finnish national licensing practice and rule generating.

In September 1999, WGRisk decided to extend the survey to a potential new task force. The new task was approved by CSNI at its December 1999 meeting.

One of the conclusions of the survey was that an expert workshop in the area might be useful in order to assess the state of art, to highlight the problems and to find ways forward.

At the same time, the European Union has accepted to launch a project RMPS (reliability methods for passive systems) belonging to the area NFS-II in its Fifth Framework Programme. The idea of this project is to develop methodology especially for systems dependent on functioning of thermal hydraulics.

The workshop was scheduled and held in March 2002 and these proceedings are the final result of the presentations and discussions.

The Chairman of the Technical Committee will be Dr. Nicolas Devictor, Commissariat à l'Energie Atomique (CEA) and co-chairmen Dr. Pekka Pyy, VTT, Finland and M. Michel Marques, CEA, France. The scientific secretary of the workshop will be Luciano Burgazzi, ENEA, Italy. The workshop committee and WGRisk members wish to acknowledge the work performed by them in preparing and guiding this work.

Workshop Committee members included:

Pekka Pyy, VTT, Finland

Luciano Burgazzi, ENEA, Italy

Joseph Murphy, NRC, USA

Klaus Koeberlein, GRS, Germany

Barry Kaufer, NEA, France

Joaquin Martin-Bermejo, EC, Belgium

Michel Marques, CEA, France

Reino Virolainen, VTT, Finland

Joon-Eon Yang, KAERI, Korea

Giacomo Cojazzi, JRC, Italy

Jean-Luc Pelletier, Technicatome, France

Mamoru Fukuda, NUPEC, Japan

EXECUTIVE SUMMARY

The Workshop was held under the auspices of the Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk (WGRisk) and was hosted by the Commissariat à l'Energie Atomique (CEA) and will take place in Cadarache, France, 4th through 6th March 2002.

Observations

Designs for Passive Systems have been and are currently being developed for new nuclear power plants. Based on the results of the survey performed by WGRisk and the workshop papers and discussions it is apparent that progress is being made in the development of methodologies to deal with the reliability of the Passive Systems being designed.

As predicted by the results of the survey participation at the workshop was small. This however was not considered an indication of lack of interest in the subject rather that the subject field is very limited due to current research efforts.

In the three (3) main areas reviewed during the workshop: development and use, methodologies and licensing, the following conclusions were derived:

1. Development and Use of Passive Systems – Any new reactor being designed will most likely contain passive systems. Testing of passive systems has been performed by several groups but further testing and development is still required. Discussions are already in progress to make clear how passive systems should be introduced in a design; e.g., back-up of an active system, BOPHR strategy, etc.

2. Methodologies – While work is being performed on methodologies and progress is being made, a lack of data exists mainly since very little or no operational experience is available. This is especially true in the area of thermal hydraulics and the result is a large amount of uncertainties.

3. Licensing – Very little progress has been made relating to licensing of passive systems. This mainly due to the fact that it is premature at this time until a firm order is made and a request for licensing is received for one of the new reactors under design. It is apparent from the discussions at this workshop and others recently that future reactors will require a more advanced interpretation of defence-in-depth principles integrated with risk-informed thinking (i.e.; risk insights).

In relation to international activities ongoing and planned on passive systems, two (2) research programmes were discussed as follows:

- Reliability Methods for Passive Systems (RMPS) – This is project underway by the European Commission with the objective to propose a specific methodology to assess the reliability of thermal hydraulic passive systems.

- Proposed IAEA Co-ordinated Research Programme (CRP) on Natural Circulation Phenomena, modelling and reliability of passive systems which utilise natural circulation.

Numerous issues were brought forward and advanced in the workshop discussions. Two particularly interesting issues were:

- What is a passive system? While the IAEA definition of a passive system and classifications were mentioned numerous times and participants generally agreed with them, there was detailed discussion on what exactly constitutes a passive system. One point noted that many passive systems are initiated by an active component. While this fits into the IAEA definition, it was noted that perhaps there needs to be clearer differentiation between types of passive systems.

- Human Action and Passive Systems – While not discussed in detail it was noted that in a truly complete Passive System operating human action role should be very low (since there is no need to intervene). Since very high uncertainties are related to maintenance and design related human actions of a full scale system in an NPP does the reduction of it in a passive system create a better situation? In other words does the reduction of the number of operating human actions offset the other uncertainties that are considered in passive systems?

Conclusions

The workshop agreed that: in general that:

1. Passive Systems and Passive PSA are becoming more and more important as technology evolves. The key element as to furthering development and use of passive systems is the decision to proceed with licensing and construction of a new nuclear power plant design.

2. As a corollary to item 1, continued and increased research in to Passive Systems is an essential element for both nuclear regulators and operators to ensure that they are prepared to meet future challenges of newly designed and licensed reactors.

3. While the current IAEA definition and classification is well accepted, it should be reviewed and refined to allow a better common understanding of what is a passive system and what is an active system.

4. There is a clear need to obtain more data, especially related to thermal hydraulics. This necessitated additional development, testing and research.

5. The work being performed under the European Commission Project - Reliability Methods for Passive Systems (RMPS) should be followed closely by WGRisk.

6. Likewise, should the proposal by IAEA for a Co-ordinated Research Programme be started, WGRisk should also follow its development and provide support and co-ordination as necessary.

7. The regulatory bodies in most NEA Member countries do not currently have enough information to license a plant (especially those in which a risk-informed approach is expected) containing passive systems, although those in which new plants may be built are closely following developments.

8. WGRisk should prepare a Technical Opinion Paper describing the current state-of-the-art and providing a continuing basis for Member countries to continue research on passive systems reliability. Additionally, WGRisk should continue to discuss and review the topic annually to assess advancements.

More specifically workshop participants noted:

- The performance of preventive maintenance and in-service testing were noted as areas in which further research and development was still required. How to ensure operability of passive systems is a concern.

- Understanding all of the safety functions of both active and passive systems in the plant and defining the correct success criteria are key requisites in determining the reliability.

- Due to lack of data current knowledge of passive reliability contains large uncertainties mainly in the area of thermal hydraulics.

- There is a need to clearly identify the role of the operator in systems that are fully passive, contain an initiating active component, or have a combination of active and passive components. This is an area where human factors experts could provide help.

- The concept that passive systems are less expensive than active systems was discussed and it was determined this may not be necessarily true.

WORKSHOP

As noted the Workshop was held under the auspices of the Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk (WGRisk) and was hosted by the Commissariat à l'Energie Atomique (CEA) and will take place in Cadarache, France, 4th through 6th March 2002.

Background

Innovative nuclear power plants largely implement passive systems, aimed at both substantial simplification and improved safety, as regards in particular human error and active component malfunctions. Thus, passive system reliability assessment is needed in the framework of innovative reactor probabilistic safety studies. This is especially relevant for passive systems based on thermal hydraulics or chemical reactions. The uncertainties related with the performance of these systems are difficult to quantify due to the fact that not all important factors may have been identified and there is no common agreement about calculating principles.

The new generation of nuclear power plants has also led to the regulatory concern about the streamlines followed in their licensing. Working Group Risk (formerly Principle Working Group 5) of the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) has established a Task Force "Reliability of Passive Systems" to investigate the topic. As a result of this work, a questionnaire has been distributed in the OECD countries. One of the conclusions of that questionnaire was that an expert workshop in the area might be useful in order to assess the state of art, to highlight the problems and to find ways forward.

At the same time, the European Union has accepted to launch a project RMPS (reliability methods for passive systems) belonging to the area NFS-II in its Fifth Framework Programme. The idea of this project is to develop methodology especially for systems dependent on functioning of thermal hydraulics.

The NEA believes that an essential factor in ensuring the safety of nuclear installations is the continuing exchange and analysis of technical information, points of views and data. Especially this is the case in emerging areas as the reliability of passive systems.

Objective

The main purpose was to provide a forum for the exchange of information on the technical issues associated with assessing the reliability of passive systems in the context of nuclear safety, regulatory practices and probabilistic safety analysis. The workshop will discuss the state of the art of the topic, and will treat all types of passive systems, i.e. structural components, systems based on working fluids etc. A special emphasis will be put on the reliability of the systems based on thermal hydraulics, for which the methods are still in developing phase.

Issues and discussions topics will include the lessons learned from designing passive systems, developing methodologies, performed studies, field experience with passive systems and need for future development. Participants will have the opportunity to meet their counterparts from other countries and organisations to discuss technical issues and problems associated with modelling and quantifying passive system reliability.

WORKSHOP PROGRAMME

Welcoming Remarks:

- M. Jean-Claude FRAPPIER Commissariat à l'Energie Atomique (CEA)
- M. Nicolas DEVICTOR, Commissariat à l'Energie Atomique (CEA)
- Barry KAUFER, Nuclear Energy Agency (NEA), Introduction

Introductory Session:

- Findings from WGRisk Survey on "Reliability analysis approaches and licensing requirements for passive systems in NEA member countries", Barry KAUFER, NEA
- Highlights from the Reliability Methods for Passive Systems" (RMPS) Study, Michel MARQUES, CEA, France

Keynote Address:

- What has to be taken into account when Building an NPP with Passive Systems, Risto HIMANEN, TVO, Finland

Session 1 – Passive Systems

- Session 1Passive Heat Removal System with the "Base Operation Passive Heat Removal" Strategy Application with a Primary hat Exchanger, Guy-Marie GAUTIER, BAZIN, CHATAING, GULLY and LAVIALLE, CEA. France
- *Inherent Failure Modes of Passive Safety Systems*, Juhani HYVÄRINEN and Paivi MAARANEN, STUK, Finland
- An ATHLET Case Study of Thermal Hydraulic System Reliability: Active versus Passive, Christoph MÜLLER and H. GLAESER, GRS, Germany

Session 2 – Evaluations and Assessments

- Passive Systems Analysis for Decay Heat Removal, Luciano BURGAZZI, ENEA, Italy
- *Assessment of Fluid Flow Characteristics for Fluidic Device in APR 1400,* Manwoong KIM, S.K., LEE, J.H. LEE, J.I. LEE, S.Y. YOO and M.S. GIE), KINS , Korea and Chungnam University, Korea

- *The REPAS Approach to the Evaluation of Passive Systems Reliability*, F. BIANCHI, L. BURGAZZI, F.D. AURIA and M.E. RICOTTI, ENEA, University of Pisa, and Polytechico Milano, Italy

Session 3 - Plant Designs

- The Passive Safety Systems of the SWR 1000, Werner BRETTSCHUH, Doris PASLER, Framatome ANP, Germany

- Experimental Verification of the New Passive Safety Systems of the SWR 1000, Werner BRETTSCHUH, Johann MESETH and Doris PASLER, Framatome ANP, Germany

- Passivity in Cooled Gas Fast Breeder Reactor, CEA

Session 4 – Reliability of Passive Systems

- *Reliability Methods for Passive Systems (RMPS) Study: Strategy and Results*, F.D. AURIA, G.CARUSO,M.E. RICOTTI, E. ZIO (Cirten Consortium, University of Pisa, University of Roma and Polytechico Milano, Italy

- *Probabilistic Safety Assessment for the Advanced SWR 1000* - Evaluation of the Safety Concept with Active and Passive Systems, Hartmut SCHMALTZ, Framatome ANP, Germany

Closing Session – Open discussions

- Further Research on Passive Systems and Designs

- Implementation of Passive Designs in Plants

- Licensing Issues related to Passive Designs

SUMMARY OF WORKSHOP SESSIONS

The following section provides listing of elements and key points that were raised and discussed during the workshop sessions.

Categories of Passivity

Within the workshop discussions many participants referred to the IAEA categories which has been extensively used throughout the world. IAEA (1991) defined passive systems or components as not requiring external input and especially energy to operate. The same reference classifies passive systems into four classes (in parenthesis the principle of passivity) based mainly on experience in thermal hydraulic solutions:

a) physical barriers and static structures (structural through material selection, condition, design and geometrical arrangement)

b) moving working fluids (by fluid/gas movement, by phase changes, by chemical reactions or/and by neutron flux effects)

c) moving mechanical parts e.g. spring loaded check-valves opening based on pressure difference

d) external signals and stored energy (passive execution / active actuation)

Elements

Several different presentations were made on the different aspects of the problem of the reliability of passive systems. These aspects included:

- classification - in practice to say that a system is passive or a little bit active is not easily determined, especially in the case of category D);
- methodologies used for the assessment of the reliability of passive systems;
- examples of implementation of passive systems in projects of innovative reactors; and
- point of view of utility.

Question : Is the IAEA classification useful in practice?

The problem of passive systems is essentially a problem for innovative reactors. We must take into account the following parameters (point of view of utility):

- investment cost ("always cheaper")
- project delays
- availability and performance ("always higher") : including maintenance, test and inspection
- licensing
- safety ("always higher").

Place of passive systems?

Different positions have been shown in the discussions:

- simultaneously use of active system and passive system (i.e., the 2 systems are started at the same instant) (for example : BOPHR strategy);

  Implication: can we expect lower performance of the passive system because there's a redundancy with an active system ⇒ design of a passive system more compact and lower cost (?)

- use as a backup of an active system
- use as an initiator of an active system
- use to fulfil a safety function (alone, without an active system)

Requirements for diversity of passive systems with the same safety function? / Redundancy with different passive systems?

What is the meaning of a comparison between an active system and a passive system? (same function? same operational conditions?…)

In fact, for any engineer the plant designs for passive systems and active systems are different.

It is necessary to set a meaningful and understandable methodology (for the engineer) to be able compare the effects of active systems and passive systems.

Additional question: Is there a difference in the study of passive systems for use as a safety function or for an operating safely?

Reliability of passive systems

There does not seem to be a critical problem in the methodology to assess the reliability for passive system in category A (mechanical component): tools from the structural reliability.

Operating experience may exist for categories C and D.

A methodological problem exists for thermal-hydraulic (TH) passive system (Category B). While the the mathematical and statistical tools are available, the problem with the methodology, is how to set and solve the problems although there is no clear consensus on this.

It is clear that computed failure probability is conditioned by the chosen transient.

An aim of RMPS project is to propose a methodology for category B passive systems.

Points which require additional clarification include:

- definition of failure mode of a TH passive system (including inherent failure mode) ;
- how to verify that all situations are covered: there is a wide range of TH situations under different conditions (problem of the case of extreme conditions)
- difficulty to model the behaviour of a passive system under any condition that it can be see during its lifetime
- in the most cases, it is a problem to model the behaviour of a passive system: and it is necessary to model the behaviour of all equipment and systems on the loop and in the environment
- what data should be used for the quantification step? lack in databases, use of expert judgement… How to obtain an agreement on the used probability density function?

Passive system and PSA

In general it is considered that Passive Systems are more reliable and available systems.

There are potential problems in representing Passive Systems in a PSA model?

- cut-off methods are used in some computer codes (like Risk-Spectrum) ; then are passive system really taken into account ?

  For example in level 1 PSA in France, structural components like pipes are partly introduced because we considered they are very reliable.

- How to validate the reliability data used ? particularly CCF data.

  One possibly answer could be to modify the level 1 PSA by introducing a simulation model ("continuous" model and not only binary model). But we have always the problem to simulate the behaviour.

Another difficulty has to do with uncertainty and sensitivity analysis: It is necessary to take into account a large number of situations and a high number of simulations are needed.

Passive systems are studied mostly in the case of innovative reactors. It needs therefore to be understood that there are also the inherent difficulties of the development and the use of PSA in the first design steps. For example, safety goals are assigned by safety authorities. In the design steps, because of the uncertainty on the true value of reliability, data goals are always checked. The results have then no meaning; only the sensitivity analysis is useful and important in determining which components or systems are really influent and to try to "orientate" the Research and Development.

Workshop Participants

FINLAND

HIMANEN, Risto Teollisuuden Voima Oy (TVO) FIN-27160 Olkiluoto	Tel: +358 2 8381 3240 Fax: +358 2 8381 3209 Eml: risto.himanen@tvo.fi
MAARANEN, Paivi Inspector Radiation and Nuclear Safety Authority (STUK) PL 14, 00881 Helsinki	Tel: +358 9 7598 8329 Fax: +358 9 7598 8382 Eml: paivi.maaranen@stuk.fi

FRANCE

BLANCHET, Yves D.R.P./SDEC C.E.N. CADARACHE F-13108 St Paul-lez-Durance Cedex	Tel: +33 4 42 25 24 17 Fax: +33 4 42 25 77 67 Eml: yves.blabchet@cea.fr
DEVICTOR, Nicolas CEA CE Cadarache DER/SCC BP 1 F-13108 St Paul-lez-Durance Cedex	Tel: +33 4 42 25 30 05 Fax: +33 4 42 25 77 67 Eml: nicolas.devictor@cea.fr
GAUTIER, G.M. CEA Cadarache F-13108 St Paul-lez-Durance Cedex	Tel: +33 4 42 25 40 98 Fax: +33 4 42 25 36 35 Eml: gmgautier@cea.fr
IOOSS, Bertrand CEA CE Cadarache DER/STR/LCFR BP 1 13108 F-13108 St Paul-lez-Durance Cedex	Tel: +33 4 42 25 72 73 Fax: +33 4 42 25 24 08 Eml: bertrand.ioss@cea.fr
LA LUMIA, Virgile Technicatome	Tel: +33 4 42 60 28 96 Fax: Eml: la-lumia@tecatom.fr

MARQUES, Michel
CEA
CE Cadarache DER/SCC
BP 1
F-13108 St Paul-lez-Durance Cedex
Tel: +33 42 25 71 31
Fax: +33 42 25 24 08
Eml: michel.marques@cea.fr

MATHIEU, Bernard
CEA
CE Cadarache DER/SERI
BP 1 13108
F-13108 St Paul-lez-Durance Cedex
Tel: +33 4 42 25 74 57
Fax: +33 4 42 25 71 87
Eml: bernard.mathieu@cea.fr

NOEL, Brigitte
CEA
CE Grenoble DTP/SETEX/LETS
Tel: +33 4 38 78 42 32
Fax: +33 4 38 78 50 45
Eml: noelb@alpes.cea.fr

PELLETIER, Jean-Luc
Technicatome
Tel:
Fax:
Eml:

PIGNATEL, Jean-Francois
CEA
CE Cadarache DER/SCC
BP 1
F-13108 St Paul-lez-Durance Cedex
Tel: +33 4 42 25 37 26
Fax: +33 4 42 25 36 35
Eml: jean-francois.pignatel@cea.fr

VIDARD, Michel
Chef de Projet, Pôle Industrie
Serv. Etudes & Projets Thermiques
& Nucléaires (SEPTEN)
Electricité de France
12-14 avenue Dutriévoz
Tel: +33 (4) 7282 7565 (secretary:
Fax: +33 (4) 7282 7701
Eml: michel.vidard@edf.fr

GERMANY

BRETTSCHUH, Werner
Head, SWR-1000 Product Management
NGPP
Framatome-ANP GmbH
Postfach 10 10 63
D-63010 Offenbach
Tel: +49 (69) 807 ext. 93527
Fax: +49 (69) 807 94327
Eml: werner.brettschuh@framatome-anp.de

FABIAN, Hermann
Framatome ANP GmbH
FANP NDS
Freyeslebenstr. 1
D-91058 Erlangen
Tel: +49 9131-18-95614
Fax: +49 9131-18-94787
Eml: Hermann.Fabian@framatome-anp.de

GLAESER, Horst
(Chairman TG-THA)
Gesellschaft fuer Anlagen
und Reaktorsicherheit (GRS)mbH
Forschungsgelaende
D-85739 GARCHING

Tel: +49 89 32004 408
Fax: +49 89 32004 599
Eml: gls@grs.de

KOEBERLEIN, Klaus
Gesellschaft für Anlagen-und
Reaktorsicherheit (GRS) mbh
Postfach 1228
D-85739 Garching

Tel: +49 89 32004 445
Fax: +49 89 32004 306
Eml: koe@grs.de

MULLER, Christoph
Gesellschaft für Anlagen- und
Reaktorsicherheit (GRS) mbH
Postfach 13 28
D-85739 Garching

Tel: +49 89 3 20 04 426
Fax: +49 89 3 20 04 599
Eml: mur@grs.de

SCHMALTZ, Hartmut
Framatone ANP GmbH
FANP NDS4
Freyeslebenstr. 1
D-91058 Erlangen

Tel: +49 09131 18 94756
Fax: +49 09131 18 97507
Eml: hartmut.schmaltz@framatone-anp.de

ITALY

BURGAZZI, Luciano
ERG-SIEC-ENEA
Via Martiri di Montesole, 4
40129 Bologna

Tel: +39 051 609 8556
Fax: +39 051 609 8279
Eml: burgazzi@bologna.enea.it

D'AURIA, Francesco
Universita degli Studi di Pisa
Costr. Meccaniche e Nucleari
Via Diotisalvi, 2
I-56126 PISA

Tel: +39 (050) 836653
Fax: +39 (050) 836665
Eml: dauria@ing.unipi.it

KOREA (REPUBLIC OF)

LEE, Sang-Kyu
Integrated Safety Research Department
Korea Institute of Nuclear Safety (KINS)
19 Goosong-Dong, Yusong-Gu
Taejon, Korea

Tel: +82 42 868 0521
Fax: +82 42 861 9945
Eml: sklee@kins.re.kr

UNITED STATES OF AMERICA

THORNSBURY, Eric
Reliability & Risk Engineer
m/s T-10E50
Washington, DC 20555

Tel: +1 301 415 6216
Fax: +1 301 415 5062
Eml: eat2@nrc.gov

International Organisation

KAUFER, Barry
OECD/NEA
"Le Seine St. Germain"
12, Boulevard des Iles
92130 Issy-les-Moulineaux

Tel: +33 1 45 24 10 55
Fax: +33 1 45 24 11 10
Eml: barry.kaufer@oecd.org

PAPERS AND PRESENTATIONS

PASSIVE SYSTEM RELIABILITY
A challenge to reliability engineering and licensing of advanced nuclear power plants

The OECD Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment (WGRisk)

Cadarache, France, 4-6 March 2002

http://www.nea.fr

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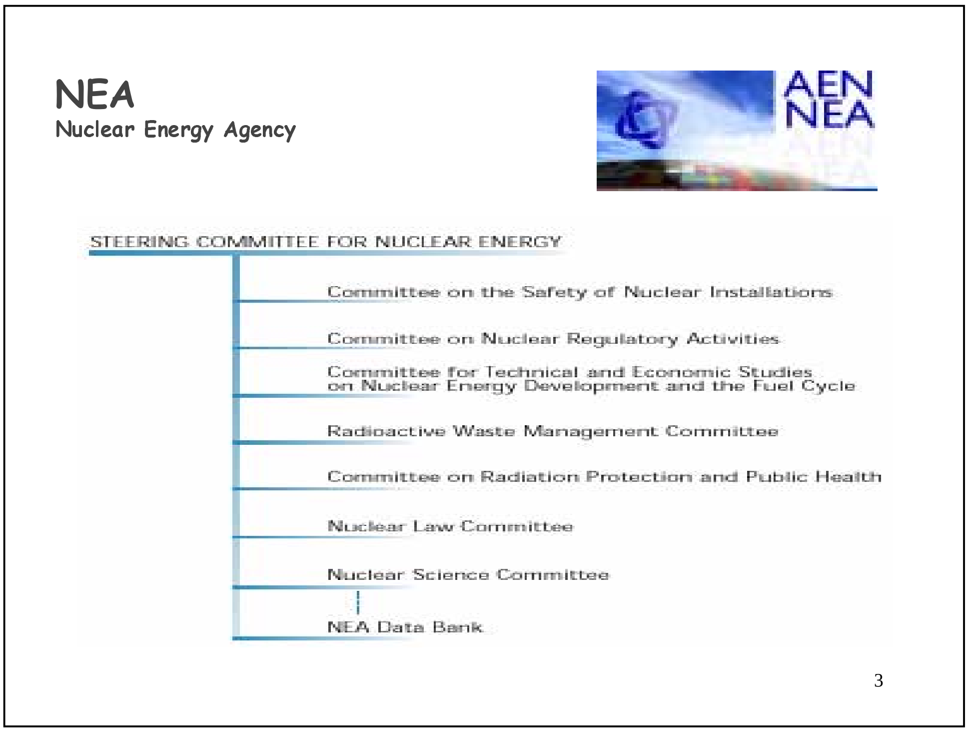
NEA
Nuclear Energy Agency
AEN
NEA
STEERING COMMITTEE FOR NUCLEAR ENERGY
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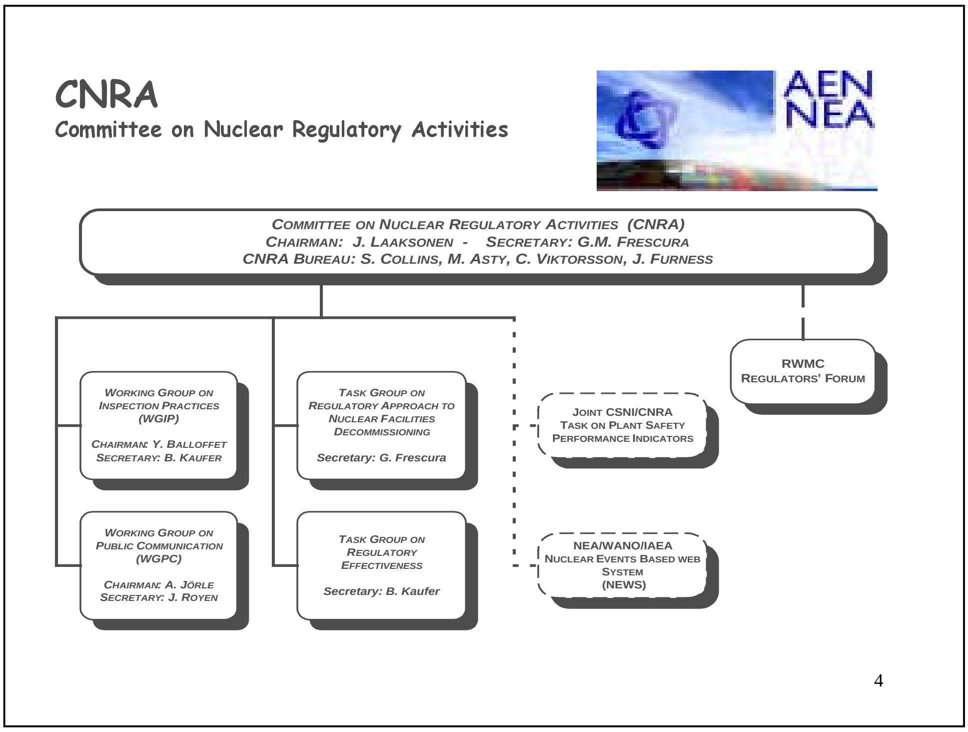
Committee on Nuclear Regulatory Activities
Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle
Radioactive Waste Management Committee
Committee on Radiation Protection and Public Health
Nuclear Law Committee
Nuclear Science Committee
NEA Data Bank
3


CNRA
Committee on Nuclear Regulatory Activities
AEN
NEA
COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES (CNRA)
CHAIRMAN: J. LAAKSONEN - SECRETARY: G.M. FRESCURA
CNRA BUREAU: S. COLLINS, M. ASTY, C. VIKTORSSON, J. FURNESS
RWMC
REGULATORS' FORUM
WORKING GROUP ON INSPECTION PRACTICES (WGIP)
CHAIRMAN: Y. BALLOFFET
SECRETARY: B. KAUFER
TASK GROUP ON REGULATORY APPROACH TO NUCLEAR FACILITIES DECOMMISSIONING
Secretary: G. Frescura
JOINT CSNI/CNRA
TASK ON PLANT SAFETY PERFORMANCE INDICATORS
WORKING GROUP ON PUBLIC COMMUNICATION (WGPC)
CHAIRMAN: A. JÖRLE
SECRETARY: J. ROYEN
TASK GROUP ON REGULATORY EFFECTIVENESS
Secretary: B. Kaufer
NEA/WANO/IAEA
NUCLEAR EVENTS BASED WEB SYSTEM
(NEWS)
4

Accomplishments 2001

- Eleven Reports Issued including Regulatory Effectiveness and Assuring Nuclear Safety Competence
- Collective Statement on Research in a Regulatory Context
- Workshops on Research in a Regulatory Context and Licensing and Operating Experience with Computer Based I&C Systems held
- Working Group on Public Communication established
- Nuclear Events Web-Based System (NEWS) established

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CNRA 2002 POW Highlights

- CNRA/WANO International Forum on Nuclear Regulator Licensee Interface Issues (June)
  - Market Competition, Asset Management and Measuring Safety Performance
- Developing PIs for Measuring Regulatory Effectiveness
- WGIP
  - Inspection Workshop, Veracruz, Mexico, April/May
  - New task on site selection, fabrication & construction
- Reports being prepared on
  - Licensee Self-Assessment
  - Regulatory Approach to Nuclear Facilities Decommissioning
- Joint CNRA/CSNI Group for Regulator - Industry / Collaboration (GRIC) established

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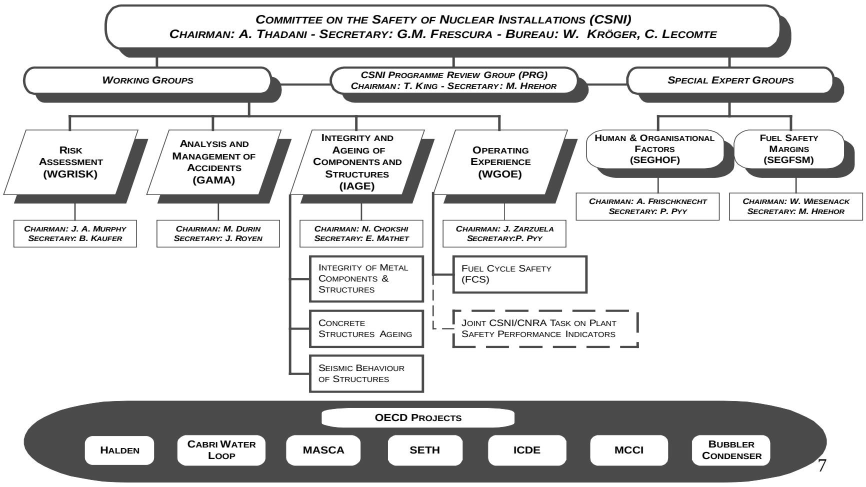
CSNI

Committee on the Safety of Nuclear Installations



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Accomplishments 2001

OECD

- Nineteen reports issued
- Collective Statement on Research in a Regulatory Context
- Seven workshops held
- SETH Programme established - Continued support to 5 existing R&D Projects and 4 new proposals in various stages of implementation
- 35 previously approved tasks currently being carried out

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CSNI
2002 POW Highlights

- Workshops and Specialist Meetings
  - Workshop on Advanced Reactor Safety Issues and Research Needs (February)
  - Specialist Meeting on External Hazards in Nuclear Installations (April)
  - Joint CNRA/CSNI Group for Regulator - Industry Collaboration (GRIC) established.
- PRG
  - Guidelines issued for CSNI reports and Integrated Plans
  - Following up on conclusions from Research Workshop

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CSNI
2002 POW Highlights

- WGOE (Operating Experience)
  - Workshop on How to Prevent Recurring Events, Switzerland, March
  - WGOE Fuel Cycle Safety to begin SOAR on burn-up credit
- SEGHOF (Human & Organisational Factors)
  - Preparing SOAR on Management of Change
- SEGFSM (Fuel Safety Margins)
  - Preparing report on fuel safety criteria
  - Topical Meeting on RIA Fuel Safety Criteria, May, Cadarache
- IAGE (Structural Integrity)
  - Workshop on Concrete Repair, Berlin, April
  - Workshop on Seismic Relations between Data and Engineering Analysis, Istanbul, Turkey, October
- GAMA (Accident Management)
  - ISPs 42, 44 and 45 scheduled to be completed.

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OECD

Working Group on Risk Assessment WGRisk

Dr. Jeanne-Marie Lanore, Chairman
Mr. Magiel F. Versteeg, Vice-Chairperson
Dr. Pieter De Gelder, Vice-Chairperson
Dr. Charles Shepherd, Vice-Chairperson
Barry Kaufer, NEA Secretariat

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Contents

OECD

- **Annual Meeting**
- **Programme of Work**
  - **Tasks**
  - **Technical Opinion Papers**
  - **Workshops**
  - **Co-operation with Others**
  - **Integrated Plan**
- **Summary and General Issues**

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WGRisk Tasks

- Human Reliability - Proposal being made on Exchanging Information
- Risk Monitor - SOAR in progress (co-ordinated with IAEA)
- Software Reliability - SOAR in progress
- Low Power/Shutdown PSA - Issuance of survey delayed due to meeting postponement.
- Passive Systems Reliability - SOAR in Progress awaiting results of March 2002 Workshop.

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WGRisk Tasks

- Use and Development of PSA in Member Countries - delayed due to missing responses.
- Integrated Plan - delayed due to meeting postponement. Plan to present to CSNI in June 2002.
- PSA in Installations different than Nuclear Reactors - awaiting new proposal.
- Level 2 PSA - to be discussed at next WGRisk meeting.
- Fire Data Exchange - to be discussed under OECD Projects.

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WGRisk TOPs

(Technical Opinion Papers)

OECD

- Fire Risk Assessment -
  - PRG comments received and being incorporated.
  - Review and comment by WGRisk (1/2002), PRG (3/2002). For CSNI Approval (6/2002) if accepted by WGRisk and PRG.
- Seismic Risk Assessment, Human Reliability and Living PSA delayed.
  - In order to achieve consistency the intent was to revise other TOPS to the style and format of the final Fire Risk Assessment, but due to delay work will proceed in parallel.
  - Review by WGRisk (1/2002), PRG (4/2002). For CSNI Approval (6/2002) if accepted by WGRisk and PRG.
- General Comment
  - PRG and CSNI comments during the drafting phase are encouraged to assist group in expediting the process.

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WGRisk Workshops

OECD

- Building the new HRA: Strengthening the Link between Experience and HRA, 28th to 30th January 2002, Munich, Germany, hosted by GRS, co-sponsored by WANO.
- Passive System Reliability: A Challenge to Reliability Engineering and Licensing of Advanced NPPs, Cadarache, France, 4th-6th March 2002, hosted by the Commisariat à l'Energie Atomique (CEA), co-sponsored by WANO.
- Workshop on the Development and Use of Risk Monitors in NPPs, 19th to 21st November 2002, hosted by CSN, co-sponsored by WANO and IAEA (?)
- Workshop on Low Power and Shutdown PSA, Winter 2003

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WGRisk Co-operation (1)

- OECD/NEA
  - CNRA has requested WGRisk and WGOE to make presentation on Low Power/Shutdown at June 2002 meeting.
  - WGOE
    - Human Performance, PSA Event Analysis / Precursor Analysis, Low Power/Shutdown Issues, ICDE and COMPSIS.
  - SEGHOF
    - Human Performance
  - Others
    - OPDE (IAGE), Level 2 PSA (WGAMA).etc.
- OECD Systemic Risk Project
  - Public Perception of Risk

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WGRisk Co-operation (2)

- EC
  - Well co-ordinated activities including work performed by JRC.
- IAEA
  - Co-ordination in areas such as the Risk Monitors task and workshop, PSA for non-Reactor nuclear facilities, etc.
- COOPRA
  - No existing concerns at present.
- Issues/Concerns
  - All 3 other organisations performing work in the area of risk-informed decision-making (RIDM). WGRisk proposal for SOAR will take into account these activities.
  - WGRisk organised a joint meeting of all international organisations to discuss issues (I.e., RIDM) for annual meeting which has been delayed.

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WGRisk Integrated Plan

OECD

- Draft Outline
  - Integrated plan is formulated on the WGRisk Mandate, CSNI Strategic Plan and CSNI High Level Safety Issues
    - Executive Summary
    - Introduction
    - Background: What is the purpose of WGRisk, What does WGRisk Do? and How does WGRisk Accomplish its Objectives?
    - Mandate
    - WGRisk Programme of Work
    - Assessment of PSA Topics and Issues
    - Summary and Conclusions - Recommendations

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Reliability Methods for Passive Safety functions

RMPS

RMPS

Coordinator

- CEA France

Contractors

- CIRTEN Italy
- ENEA Italy
- GRS Germany
- JRC Italy
- TECHNICATOME France

RMPS Project 2

RMPS Objectives

- The objective of the project is to propose a specific methodology to assess the reliability of thermal-hydraulic passive systems.

- The methodology is tested on examples of industrial T-H passive systems.

RMPS Project 3

Secondary Objectives

- Identification and quantification of the sources of uncertainties and determination of the important variables.
- Propagation of the uncertainties through a T-H model and reliability assessment of the T-H passive system.
- Methodology for linking the passive system T-H unreliability with other sources of unreliability (failure of active systems, human errors...) in an accident sequence.

RMPS Project 4

Identification of the Sources of Uncertainties

The uncertainties are in the:

- Approximations in modelling the process physics.
- Approximations in modelling the system geometry.
- The input variables.

This identification must be based on the opinion of experts of the physical process and of the thermal hydraulic codes.

RMPS Project 5

Quantification of the Sources of Uncertainties

Selecting :

- The range of uncertainty.
- The probability density function.

Based on :

- Particularity of the data base (amount of data).
- Physical considerations, correlation between parameters.
- End use of the uncertainty analysis.

RMPS Project 6

Determination of the Important Variables

Goal: identify the main contributors to passive system performance :

- guide further code development.
- prioritize experimental investigation.

Qualitative or quantitative :

- Standardized Regression or Partial Correlation coef.
- Rank correlation coefficients.
- Non-linear sensitivity analysis (Sobol indice, Fast indice).

RMPS Project 7

Propagation of the Uncertainties Through a T-H Model

The methods for reducing the number of T-H calculations :

- Variance reduction techniques in the Monte-Carlo simulations : Latin Hypercube, Importance Sampling, Directional Simulation…
- Response surface techniques: polynomial surfaces, non-linear response surface obtained by neural networks....

The approximate methods such as First and Second Order Reliability Methods (FORM/SORM) used in structural mechanics .

RMPS Project 8

Introducing the passive system T-H unreliability in an accident sequence

Link the passive system T-H unreliability with other sources of unreliability in an accident sequence.

Assess the relative contribution of the passive system unreliability in the failure probability associated to the sequence.

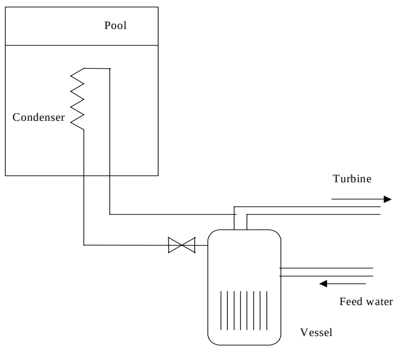
Compare active versus passive systems.

RMPS Project 9

Tests on passive systems

- Isolation Condenser System (ICS).
- Residual Passive heat Removal system on the primary circuit (BOPHR/RP2)

- Benchmark on RELAPS, CATHARE and ATHLET calculations



RMPS Project 10

Reliability of Passive Systems (RoPS)

WGRisk Task Group

Results of Survey

1

Background

- In March 1999, VTT Automation distributed a questionnaire concerning reliability of passive system (RoPS). Especially, the idea was to map methods developed for passive systems based on thermal hydraulics. The goal was to support developing Finnish national licensing practice and rule generating.
- In September 1999, the WGRisk decided to extend the survey to a pre-study for a potential new task force. At the same time, more time was given to the member countries to reply to the questionnaire. This was due to the fact that very few replies had been obtained in due time.
- In December 1999, the CSNI approved a WGRisk proposal to prepare forming a task force on the topic and a future workshop.

2

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Survey

OECD

1. Have you developed or are you currently developing approaches for passive system reliability / availability / safety assessment (from reliability engineering point of view)?

2. The same question as 1) for licensing practices?

3. If so, can you send us articles or reports related to your work? Maybe there is a chance for co-operation.

4. If you do not know directly about the topic, do you know somebody who is carrying out active work on the area? Can you send us references or contact information?

3

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Responses

OECD

- Twenty (20) replies were obtained from 12 countries and from the IAEA.
- In four replies, methods for passive system reliability assessment were developed.
- Apart from this, physical barrier reliability was discussed in six replies, and some replies discussed physical and structural uncertainty more generally.

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Conclusions

OECD

- The replies demonstrate that it was premature to develop a state of art report about reliability of passive systems, especially if the systems based on thermal hydraulics are considered. This was the case even if information obtained outside the OECD questionnaire was included.

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Conclusions (cont'd.)

OECD

- Three key areas were identified for continuation of the RoPS task:
  - to collect different aspects and points of view about passive system reliability
  - define major sources of uncertainty / ways to model them and
  - to generate recommendations and good practices to be taken into account in (future) licensing and rule generation.

6

Conclusions (cont'd.)

- to collect different aspects and points of view about passive system reliability
  - It may be a good idea to postpone launching the group for a while and think about its aims. This could, possibly, take place in the form of an international workshop under auspices of CSNI

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Conclusions (cont'd.)

- define major sources of uncertainty / ways to model them and
  - Following further development, work relevant to uncertainty of physical processes has to be taken into account. In fact, a passive system could be just one application area. Thus, there is even some potential for a combined task force.

8

Conclusions (cont'd.)

- to generate recommendations and good practices to be taken into account in (future) licensing and rule generation.
  - This is obviously a long term goal following completion of the first 2 elements. Although recent developments relating to new reactors brings higher priority to this requirement.

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Summary

- This workshop, to look at the current status of research and development relating to reliability of passive systems is the first step towards achieving progress in:
  - Development and use of passive systems (i.e.; looking at future R&D efforts)
  - To develop and apply methodologies for assessing the reliability of passive systems which utilize natural circulation and to introduce this reliability into example accident sequence analyses.
  - To make progress in risk informed decision making methods and bring them available for end users; develop licensing practices in problematic areas; develop methods for importance & uncertainty analysis; assure the competence transfer to the new generation and to contribute to international perspectives to risk analysis.

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Final Observation

- Consideration by participants of this workshop on IAEA proposal for:
  - Co-ordinated Research Programme (CRP) on Natural Circulation Phenomena, Modelling, and Reliability of Passive Systems That Utilize Natural Circulation

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WGRisk Survey on Reliability of Passive Systems (RoPS)

Introduction

In March 1999, VTT Automation distributed a questionnaire concerning reliability of passive system (RoPS). Especially, the idea was to map methods developed for passive systems based on thermal hydraulics. The goal was to support developing Finnish national licensing practice and rule generating.

In September 1999, the OECD/NEA/CSNI/PWG5 decided to extend the survey to a pre-study for a potential new task force. At the same time, more time was given to the member countries to reply to the questionnaire. This was due to the fact that very few replies had been obtained in due time.

In December 1999, the CSNI asked the PWG5 to prepare forming a task force on the topic. Preconditions for such a working group will be discussed under the conclusions about the replies.

The questionnaire included four questions as follows:

9. Have you developed or are you currently developing approaches for passive system reliability / availability / safety assessment (from reliability engineering point of view)?

10. The same question as 1) for licensing practices?

11. If so, can you send us articles or reports related to your work? Maybe there is a chance for co-operation.

12. If you do not know directly about the topic, do you know somebody who is carrying out active work on the area? Can you send us references or contact information?

Responses

20 replies were obtained from together 12 countries and from the IAEA. In four replies, methods for passive system reliability assessment were developed. Apart from this, physical barrier reliability was discussed in six replies, and some replies discussed physical and structural uncertainty more generally.

A table including shortened versions of the replies is presented. References given in the replies are included in the reference list with some other sources of information.

Conclusions

The replies demonstrate that there is no potential of directly writing a state of art report about reliability of passive systems, especially if the systems based on thermal hydraulics are considered. This is the case even if information obtained outside the OECD questionnaire is included (see reference list).

If a task force about RoPS is launched, the aim should be rather to:

1. to collect different aspects and points of view about passive system reliability

2. define major sources of uncertainty / ways to model them, and

3. to generate recommendations and good practices to be taken into account in (future) licensing and rule generation.

The point a) should give an answer to the question about what should be studied. It may be a good idea to postpone launching the group for a while and think about its aims. This could, possibly, take place in the form of an international workshop under auspices of OECD.

If and when the work approaches to phase b), also other work relevant to uncertainty of physical processes has to be taken into account. In fact, a passive system could be just one application area. Thus, there is even some potential for a combined task force.

Phase c) is obviously a long run goal. For that purpose, at least, a three year working period is required. Based on the replies, This estimate about a sufficient time period may even be optimistic.

Responses

In the following table, the replies from the various countries are summarised. The author's own comments are shown in parentheses.

I	II	III	IV	V
Country, respondent, date, points of view	Developing passive system reliability / availability / safety assessment methods?	Developing passive system licensing practices?	Articles / references / projects	Other related work known and add-on information
Belgium, P. De Gelder, AVN, 17.4.1999 View: The questionnaire should be sent to the vendors, who have both examples and licensing needs	No	No		ESREL '96-PSAM III, N. Siu on "Identifying New Failure Modes for an Advanced Reactor Design" (Proceedings p. 1660)
Belgium (Tractebel), A.M. D'eer, 18.5.1999	No	No	No	(ENEL) Mr.Bassenelli has been performing reliability analysis on passive components in the framework of the EPP design and its corresponding PSA.
Finland (VTT), P.Pyy (added afterwards)	There are two lines of work a) study about international practices and b) developing methods related to analysing phenomenological uncertainties	Research is going on related to the point b.	See Pulkkinen et al. in ESREL 2000 in the reference list.	Finland has been active in compiling this material.
France, J.M. Lanore, IPSN, 21.2.2000	Nothing very relevant			The future Franco-German plant EPR is an evolutionary design, with mainly active systems.
France, B. Magondeux, EdF, 29.2.2000	The process of OMF-structures (Reliability Centred maintenance) is a present approach for passive system reliability : in that process, PSA is used to list the passive parts (of pipes) that impact significantly the Core Damage Frequency. Then, we have to identify their possible degradation modes, before deciding the best maintenance rule for them (the reply is applicable to structures and not to physical phenomena)	The reported approach is not a licensing practice (EdF was responding)	Several papers have been presented and published on the topic of OMF-Structures (see reference list)	Contact: The manager of OMF process is Alain DUBREUIL-CHAMBARDEL, from Nuclear Production Division of EdF.
Germany, K. Koeberlein, GRS, 8.7.2000	Included in the analysis of thermal hydraulic uncertainties – also work on uncertainties related to crack behaviour in transient conditions		Hofer & Kloos in the reference list	Materials experts wish to initiate a project on passive system reliability evaluation.

Germany, H-P. Balfantz, TUV, 27.6.1999	RI-ISI methods for pipes and similar structures are developed (systems based on natural laws were not mentioned in the reply)	See column II, the principles are used in decision making	ESREL '99 "reliability based inspection of safety systems + 14 references (mainly USNRC reports) were given	
Hungary, E. Hollo, VEIKI, 17.4.1999 and 21.2.2000	Reliability of physical barriers is being studied. Two comprehensive projects are to address the subject in some details: - the level-1+ seismic PSA study started January 2000, and - the level-2 internal event PSA study under preparation. Both studies have to evaluate the reliability of passive safety engineering systems as well as of structural barriers within the containment to some extent.	No	No	
IAEA, A. Gomez-Cobo, 29.2.2000	No work is carried out in the area of "Passive system reliability" at the IAEA. however some work on reliability of passive components is carried out to develop estimates of pipe failure frequency		Working Material IAEA-J4-2000-CT-00280 which mainly discusses how to deal with statistical analysis of service data to develop estimates of pipe failure frequency as function of key reliability attributes (e.g., pipe material, size) and influence factors (e.g., operating conditions).	
Italy, L. Burgazzi, ENEA, 21.2.2000	ENEA, Italian Commission for New Technology, Energy and Environment, is currently carrying out an activity about passive system reliability studies, in co-operation with Italian Universities, focusing particularly on natural circulation systems (type B, according to IAEA categorisation). The activity seeks to evaluate the passive system behaviour from the safety and reliability point of view in the frame of the PSA studies, therefore in developing fault trees and event trees and finally in the assessment of the end sequence frequencies.	The final goal of the activity should be directed towards licensing practices. Up to now the activity is conducted at a research level.	Articles are presented in the reference list.	

Japan, K. Muramatsu, JAERI, 26.6.1999 View: it is worth studying the possibility of using passive systems because they may allow simplification of the plant. The simplification may contribute, not only to cost reduction, but also to enhancement of reliability against seismic events, which might be important for Japan.	Several years ago, JAERI made a design study of SPWR (System Integrated PWR). It had a passive core flooding system using gravitation. They assumed a generic failure rate value for usual check valves because of similarity.	The kind of reliability estimation (under column II) was enough for a PSA to assist the designers at conceptual design stage. If the PSA were for regulatory safety review (although Japan has no formal requirement for PSA for licensing), it is believed that some kind of reliability test data should have been requested.	The JNC (Japan Nuclear Fuel Cycle Development Institute) has recently made a presentation at a domestic conference of the Japan Atomic Energy Society on their research program on PSA of a Fast Breeder Reactor where the safety enhancement effectiveness of two passive reactivity control systems is being assessed. Their work is ongoing.	In Japan, the research program on passive system is currently not very active. One reason may be that current interest of the utilities are construction of ABWRs and APWRs, which have very low core damage frequencies due to higher redundancy of active safety systems than older types of BWRs or PWRs.
Japan, R. Nakai, JNC, 24.2.2000	They are developing a method to estimate failure probability in function of a passive shutdown device: e.g., Curie point electromagnet (CPEM) type of self-actuated shutdown system (SASS), and gas expansion module leading to an increased neutron leak (GEM). a) Reliability of SASS: a computer program computes a relationship between various input parameters, related uncertainties and the effect on output (cdf). b) Reliability of GEM: Estimating reactivity uncertainty due to nuclear calculation uncertainty assuming the normal distribution and as computing occurrence probability of accident scenarios (i.e., power reduction and stable cooling, delay of accident progression, and no delay effect) depending on input negative reactivity inserted by the GEM.	No activity.	Two articles from Mihara et al. were attached to the reference list	Future plan for SASS: plant response analysis under accident condition, evaluation of uncertainty of input parameters, and examination of computation method for frequency of inadvertent shutdown rod drop from SASS. Future plan for GEM: More accurate analysis about delay of accident progression and estimation of operator reliability under accident condition (i.e., human factor).
Japan, M. Fukuda, NUPEC, 22.3.2000' Comment: it was difficult to find the right people to reply	Not directly - for SPWR JAPC used the PSA methodology, but the passive component failure data came from US AP-600 PSA.	Japanese utilities investigate single failure criteria, safe shutdown criteria, etc. for advanced reactors with passive systems	An article was attached about PSA for a plant with passive systems, see Reference list	The studies of ABWR-II and APWR+/NP-21 have been conducted jointly by Japan's BWR utilities and vendors and by Japan's PWR utilities and vendors, respectively. The study of JSBWR and SPWR has been conducted jointly by JAPC, Japan's utilities and vendors. NUPEC and JAERI have also studied about passive safety systems in specific areas.

Korea, J-E. Yang, KAERI, 24.2.2000	Yes, in KAERI, the passive systems PSA is included as part of KALIMER (KAERI Liquid Metal Reactor) PSA. However, the research on passive system safety is not so active (they only survey the state-of-art technology on this topic)	No	Currently not available.	
Korea C-J., Lee, KSC, 2.3.2000	The utility has developed a methodology for safety assessment of a passive component, such as PAR (passive autocatalytic recombiner) which can automatically control the containment environment against hydrogen explosion. The result will be used in Level 2 PSA of KNGR (Korean Next Generation Reactor). Except PAR, there are no other passive systems.	Because the performance of PAR in an adverse environment during the mitigation of severe accidents is still arguable, they are concerned about the reliability of PAR, and are developing the regulatory guideline for ensuring the function.	Currently not available. The work result will be presented in ongoing phase 3 (final) PSA report, later on.	No
Netherlands, M. Versteeg, 25.2.2000	Apart from the piping failure data, there are no developments in the Netherlands in the field of passive system reliability.			ECN (nowadays NRG) is doing some development work for a small modular HTR. Jan Schuller from NRG should be contacted.
Mexico, A. Huerta Bahena, CNSNS 7.4.2000	No	No	No	No
J. Calvo, Spain, CSN, 1.3.2000 View: some more in-depth work could be considered for the future.	a) The activities about passive systems reliability are only related to the structural components: containment fragility study, uncertainty analyses of containment failure calculations, pipe rupture frequencies (statistics + fracture mechanics) and RI-ISI. b) Spain has had no activity about other passive systems reliability analysis. Natural circulation is being considered and credited in some plants for some scenarios.	Spain has no experience about licensing practices in this area.	Information on the topics (mentioned under column II) are included in Spanish in the utility-owned documentation of each of our PSAs. If there is interest, utility permission should be asked.	There has not been other work on passive systems reliability in Spain.

Sweden, R. Nyman, SKI, 23.4.1999	During the last years, the so called SLAP project has worked with issues like collection of data on passive piping systems (welds, pipe, bends, components etc in primary and secondary systems (SLAP database)		References: The SLAP reports, the BLAP report, conference papers. SKI Nucleus articles. VTT is one of the organisations that are included in the mailing list of reports from SKI. SKI is introducing an approach on the so called "LBB - leak before brake criterion" and on how it can applied in Sweden. The concept is under external review these days.	Other domestic sources: - Passive components where treated and analysed in the so called RAMA investigation, (Filtra in Barsebäck, scrubbers in other plants) - SAQ work optimisation of RB ISI (now an active research project), the former NKS/RAK-2 activities. - Passive components are in some sense treated and analysed in the PSA Level-2 studies (via the sensitivity studies) - Passive components taken into account in the reviewing of PSA level-1, fire-, flooding analysis.
USA, N. Siu, NRC, 20.3.2000	a) NRC has reviewed PRAs performed for passive reactors (e.g., the AP600). Some minor methods work was done to support the review process, e.g., to search for new failure mechanisms associated with the new designs. b) Pressurised Thermal Shock (PTS) work involves an integrated treatment of PRA, thermal hydraulic, and probabilistic fracture mechanics concerns. The problem does not involve a purely passive system, but some of the issues – like thermal hydraulic uncertainties, errors of commission, improved integration of PRA and thermal hydraulics - of concern for PTS are probably relevant to passive systems.	NRC is currently involved in developing the technical basis for a rule change regarding pressurised thermal shock (PTS). It is intended that the PTS rule change process will be risk-informed. Work is underway to define what are the appropriate acceptance criteria, how should they be set, how should uncertainties be addressed etc. (only indirectly related to passive system licensing).	The NUREG/CR report on the ageing study (see column V) and a white paper addressing uncertainties in PTS are available from USNRC on request. See reference list for passive reactor failure modes, paper from Siu et al in PSAM-III conference.	NRC has also sponsored a technical feasibility study looking at incorporating ageing effects into PRA. This study focused on a specific ageing mechanism relevant to the reliability of piping systems. For information on NRC's regulatory activities regarding the AP600 interested persons should contact: Dr. Adel El-Bassioni, Office of Nuclear Reactor Regulation

Literature

OECD SOURCES:

1. Ph. Bryla, F. Ardorino, K. Aubert, P. Aufort, H. Boyelle, B. Vérité, 1998, OMF-Structures : A Maintenance Optimisation Procedure for Structural Components in Nuclear Power Plants. In ESREL '98 Conference, Trondheim, Norway.
2. Principles and Applications (EdF - also a related paper from PSAM4)
3. Hofer, E., Kloos, M. 1999. Lessons learnt from an uncertainty and sensitivity analysis of a transient simulation for a PSA. PSA '99, Washington DC, August 1999. pp. 363-368.
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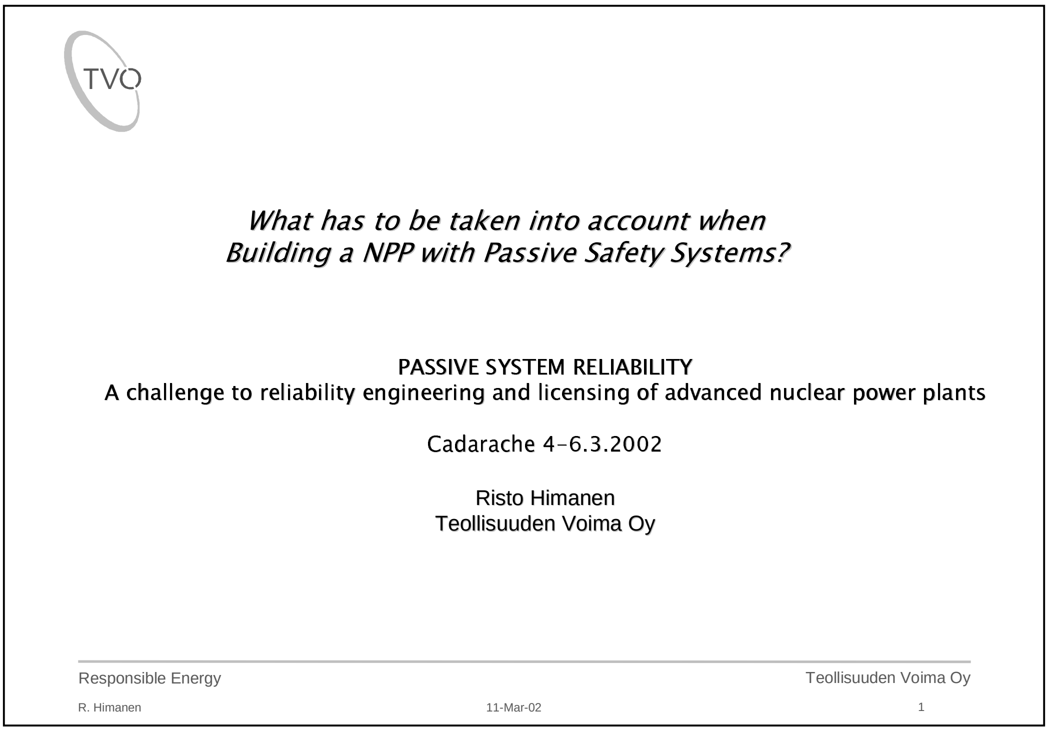
TVO
What has to be taken into account when
Building a NPP with Passive Safety Systems?
PASSIVE SYSTEM RELIABILITY
A challenge to reliability engineering and licensing of advanced nuclear power plants
Cadarache 4-6.3.2002
Risto Himanen
Teollisuuden Voima Oy
Responsible Energy
Teollisuuden Voima Oy
R. Himanen
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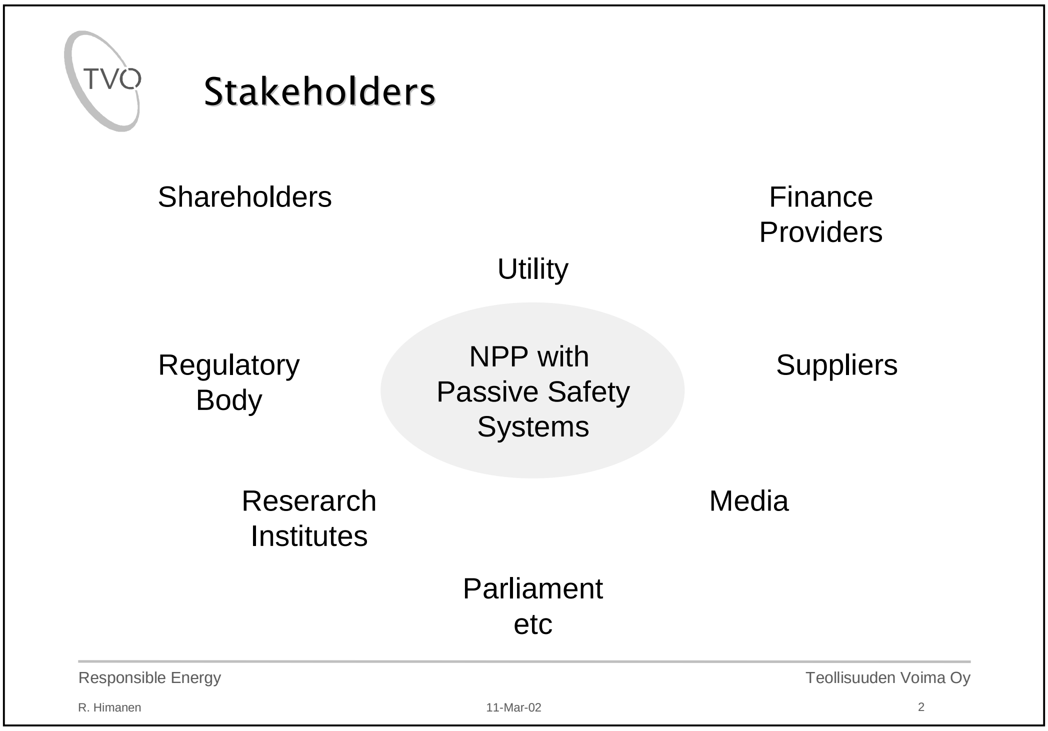
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Stakeholders
Shareholders
Finance Providers
Utility
Regulatory Body
NPP with Passive Safety Systems
Suppliers
Reserarch Institutes
Media
Parliament etc
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Key matters – related to new NPP investment from the utility's point of view

Investment

Construction

Life Cycle Cost

Availability Performance

Licensing

Nuclear Safety (PSA)

Categories of passivity

IAEA-TECDOC-626, 1991

	Cat A	Cat B	Cat C	Cat D
Signals Power Forces	**No**	**No**	**No**	**Manual initiation excluded Stored energy sources**
Moving mechanical parts	**No**	**No**	**Yes**	**Controls, Instrumentation, Single action valves relying on stored energy**
Moving working fluid	**No**	**Yes**	**N/A**	**N/A**

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Passive features – prevention of core damage Front Line Safety Functions – Level 1 PSA (1)

Primary circuit overpressure protection (Cat B,C)
- Explosive (squib) valves
- Passive signal system

SCRAM (Cat B,C)
- Gravity
- Pressurized gas
- Saturated steam
- Passive signal system

HP core cooling (Cat B,C)
- HP accumulators
- Heat exchanger – natural circulation
- Includes pressure control

LP water injection (Cat B,C)
- Internal flooding of reactor pressure vessel

Core Cooling (Cat A)
- Radiation/Conduction

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Passive features – prevention of core damage Front Line Safety Functions – Level 1 PSA (2)

Residual heat removal from core – isolation condenser (Cat B)
- inside containment
- outside containment

External cooling of reactor pressure vessel (Cat B,C,D)

Residual heat removal from containment (Cat B,C,D)
- Cooling circuit open to the containment
- Cooling circuit open to an external pool
- External cooling of the containment

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Passive features – mitigation of severe accidents Systems, Structures, Phenomena – Level 2 PSA

Hydrogen treatment (Cat A)
- Passive autocatalytic recombiners
- Transport of hydrogen by means of steam pressure

Containment overpressure protection (Cat C)
- Rupture disks
- Filters, scrubbers

Direct Containment Heating
Steam Explosion (Cat A,B,C,D)
- Keeping Corium inside pressure vessel
- LDW tolerates corium
- Flooding of lower drywell
- External cooling of the bottom head of the pressure vessel

Hardened building structures (Cat A)

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Passive features

Explosion (squib) valves (Cat D)

Own medium operated valves

Passive instrumentation (Cat B)

Directed choke valves (Cat B)

Operator errors (Cat A?)
- 24…72 h without operator actions and without electric power

Deep reactor pressure vessel
- Large water volume above core (Cat A?)

Long grace time (Cat A?)

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Power sources for passive safety functions

Gravitation (water pools high up)

Steam pressure

Density difference of fluids based on temprature difference

Density difference of non-condensible gas and steam

Residual heat from the core

Expansion of fluid and gas as a function of temperature (passive instrumentation)

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IAEA TECDOC 626, 1991:

It should be emphasized that passivity is not synoymous with reliability or availability, even less with assured adequacy of the safety feature

Active designs employing variable controls permit much more precise accomplishment of safety functions; this may be particulary desireable under accident management conditions.

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Problems in PSA

The equipment is simple and the components reliable
- Traditional PSA produces very small CDF and LERF
- PSA results are not plausible

Irreversible operation of components

Depleting energy sources

Phenomena related uncertainties

- Edge zones of the phenomena may be common for separate systems or safety functions
- The parameters of phenomena (temperature, concentration, pressure) may have opposite effect on success of separate safety functions

Factory tests in conservative circumstances may be unrepresentative for success criteria

In-service testing is difficult

What about?

Use of level 2 PSA methods in the level 1 PSA
- Models are based on parameters of the phenomena
- Success criteria are no more binary, but distributed
- Stress – strength –approach in modeling uncertainty

Quantitative model based on event tree simulation instead fault trees

TVO

Utility's fears

Plants with passive systems only may be cheaper, but are they licensable?

The Regulatory Body or public opinions may require expensive active systems during construction project, or later!

Availability - Length of refueling outage!

Prototype problems!

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Utility's requirements of the PSA methodology for Passive Systems

Should be generally accepted when licensing starts

Should not delay the construction project

Coordination with licensing procedure

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Development of acceptable methodology
PSA for a NPP with Passive Safety Systems

Questions
- Research initiation
- Method development
- Financing of research

Integration or cooperation
- Availability analysis
- LCC analysis
- Other licensing analyses

Stakeholders
- Utilities
- Suppliers
- Regulatory Bodies
- Research Institutes

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What is this facility?

Biggest, strongest and fastest in the world

Passive safety features of best possible design

Less attention paid on active safety systems and accident mitigation, because they seemed to be unnecessary

Extreme strength of phenomena were outside design basis

No attention paid in CCF

Strong belief that it was unsinkable, shown by calculations

Answer: Titanic

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Summary

The interests of separate stakeholders may be different

It may be dangerous to fall in blind love with passive systems

One should not isolate within PSA, phenomena or severe accidents

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PWR Passive heat removal system with the "Base Operation Passive Heat Removal" strategy Application with Primary Heat Exchangers

Guy-Marie GAUTIER
Commissariat à l'Energie Atomique - Direction de l'Energie Nucléaire
DEN/DER - Cadarache
13108 Saint-Paul-Lez-Durance FRANCE
Phone : 33 4 42 25 48 24, Fax : 33 4 42 25 36 35
E-mail : guy-marie.gautier@cea.fr

P. BAZIN, Th. CHATAING, Ph.GULLY, G. LAVIALLE
CEA- DEN/DTP - Grenoble - FRANCE

The interest of the passive heat removal systems in a PWR, is their ability to extract the decay heat without any continuous external energy input and their independence from operator actions. Yet these systems can have some drawbacks : installation of large components, difficulty in monitoring the operating point such as the cooling rate. Furthermore, the passive systems have to be studied with their accidental procedure management and their compatibility with the other safety systems.

In this study we propose a new management principle, termed "Base Operation Passive Heat Removal" strategy (BOPHR), for an accidental transient. The passive heat removal systems are designed to be used simultaneously alongside active systems for all emergency shutdowns. Using the thermal inertia of the steam generators, these systems must be able to remove the decay heat in a satisfactory manner should the active systems fail.

In this document, we present an evaluation of this strategy associated to a passive system connected to the primary circuit of a French three loops PWR. The accidental transients known to be the most penalising are studied with the CATHARE code. The safety analysis confirms the advantage of the system-strategy couple by showing that it meets the safety requirements for future reactors.

Keywords Passive system, Decay heat removal, PWR, Safety assessment, Procedure management

1 INTRODUCTION

For several years now, the CEA/DEN has studied the passive systems with a view to diversifying the heat removal systems of any LWR. The aim of these studies is to improve the defence in depth, especially from the point of view of prevention through the simplification of procedures and systems, by designing systems with forgiving characteristics, all the while respecting the future reactor safety requirements and the economic constraints.

In most of present pressurised water reactors, decay heat removal is usually carried out by means of active systems: auxiliary feedwater (AFS) supply of the steam generators (SGs) by motor-driven pumps or turbo-driven pumps, and by heat extraction performed by a steam release into the atmosphere through the blowdown valves.

In order to diversify the means of decay heat removal, the use of passive systems can be envisioned : the Passive Residual Heat Removal System : PRHR from the AP600 [1], or the Secondary Condensing System : SCS from the SIR project [2], or the Refroidissement du Réacteur au Primaire : RRP (Primary Circuit Cooling System) which resulted from work carried out at the CEA [3].

The recent evaluation studies made on these systems adapted to a 900 MWe reactor of the same type as those being used by EdF (the national electricity utility in France) [4], have led to a new strategy named "Base Operation Passive Heat Removal" Strategy (BOPHR) for accident management.

2 PRINCIPLE OF THE BOPHR STRATEGY

In a standard PWR, if no intervention whatsoever is carried out at the time of a power loss (i.e. no auxiliary feedwater supply to the SGs, no safety injection) the transient occurs in a manner characterised by the two following stages:

- Removal of the residual power is carried out thanks to the thermal inertia of the SGs : evaporation of the secondary water contained in the SGs with a release of steam into the atmosphere through the safety valves. The primary cooling circuit is in natural convection. This phase lasts as long as the SGs remain operational, in other words, as long as all the secondary water has not completely evaporated. The duration of this phase ranges from 30 minutes to an hour.

- When the preceding phase has ended, the decay heat is removed by evaporation of the primary water through the safety relief of the pressuriser. This phase is therefore accompanied by a degradation in the primary water inventory, and ends with an adiabatic overheating of the core leading on a short term basis to its meltdown under high pressure (assuming that no Automatic Depressurisation Safety -ADS- system is provided which is the case for many current PWRs).

To this standard reactor, a Passive Decay Heat Removal System (PDHRS) is added. This system is designed to have a low power extraction capacity. Its actuation is automatic on the emergency shutdown signal. Furthermore, it is capable of operating simultaneously with a power extraction by the SGs. In this case, at the time of a blackout without any intervention of the active systems, a natural convection occurs in the primary circuit on one hand between the core and the SGs, and on the other hand between the core and the PDHRS. The residual power is removed by both the SGs through their thermal inertia and the PDHRS.

At the beginning of the transient, the residual power is therefore adequately removed, and the primary circuit remains at a constant temperature owing to the pressure regulation on the secondary side thanks to the SG safety relief valves. This isothermic phase continues either up to the exhaustion of the thermal inertia of the SGs or until the residual power becomes inferior to that of the PDHRS.

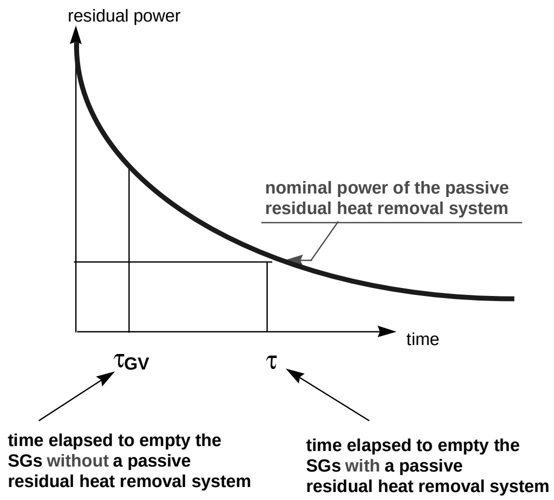


Figure 1:
Diagram illustrating the basic principle of a residual power removal according to a "Base Operation Passive Heat Removal" strategy for a blackout transient

If we seek a PDHRS with the lowest possible power in order to disturb as little as possible the power removal performed by the active systems while they are in operation, and if we refuse any loss of a primary water inventory when the active systems fail, the PDHRS must entirely guarantee residual heat removal when the SGs no longer have any more available thermal inertia (fig. 1). This means that the nominal power of the PDHRS must be equal to the residual power when the power removal carried out by the safety relief valves of the secondary circuit becomes unavailable.

In these conditions, the PDHRS functions in the same manner as a standard reactor in base load operating, which explains just why it is referred to as "Base Operation Passive Heat Removal" strategy.

3 DESIGN POWER OF THE PDHRS

The goal of the PDHRS functioning according to the BOPHR strategy is to extract a small amount of this power as soon as emergency shutdown occurs. At the beginning of a blackout transient, the power is removed by thermal inertia of SGs for as long as this inertia remains available.

As an initial approximation, the residual power of a PWR can be expressed in the following form:

$W_{res}(t) = W_o \cdot k \cdot t^{-a}$ (1) with W_{res} : the residual power of the reactor,
W_o : the nominal power of the reactor,
k and a : two constants.

The thermal inertia of SGs is :

$I_{th} = M \cdot L$ with M : water mass contained in the SGs
L : latent vaporisation heat

The design criterion of PDHRS is : " The nominal power W_{nd} of this system is equal to the residual power when the SGs no longer have any more available thermal inertia". This criterion is expressed by the following relation:

$W_{nd} = W_{res}(\tau)$ with τ : the time after emergency shutdown at the end of which the SGs no longer have any available thermal inertia.

In these conditions, the energy balance is written :

$$\int_o^{\tau} W_{res}(t)\,dt = I_{th} + \int_o^{\tau} W_{res}(\tau)\,dt \quad (2)$$

The first term expresses the decay heat produced during the first τ seconds after emergency shutdown. The other two terms express the heat extracted on one hand from the thermal inertia of the SGs, on the other from the PDHRS.

From this equation, τ is deduced :

$$\tau = \left(\frac{M \cdot L}{W_o \cdot k} \cdot \frac{1-a}{a} \right)^{\frac{1}{1-a}} \quad (3)$$

The nominal power of the PDHRS is then :

$$W_{nd} = W_o \cdot k \cdot \tau^{-a} \quad (4)$$

Observations:

If there is only the thermal inertia of SGs to extract the residual power, the time τ_{GV} during which the SGs are available is obtained by solving the equation (2) in which we have suppressed the term representing the power extracted from the PDHRS, let :

$$\tau_{GV} = \left(\frac{M . L}{W_o . k} \; . \; (1 - a) \right)^{\frac{1}{1-a}} \qquad (5)$$

The increased factor grace period is obtained by carrying out the ratio of the equation (3) and (5), i.e. let :

$$\tau / \tau_{GV} = \left(\frac{1}{a} \right)^{\frac{1}{1-a}}$$

For a given reactor, the build up factor of the grace period only depends on the time exponent in the formula giving the residual power, it does not depend on the initial amount of water present in the SGs.

Application to a French three loops 900 MWe PWR of EdF type:

For this type of reactor, the residual power is approximated by the following coefficients:

Wo = 2830 MW k = 0.146a = 0.28

The entire water liquid mass contained in the all SGs is M = 120 000 kg (40 t per SG)

We obtain :

F The time at the end of which the SGs no longer have anymore thermal inertia :
τ = 17200s i.e. approximately 4.8 hours (with PDHRS).

F The nominal power of the PDHRS is
W_{nd} = 27 MW. To take account of the single failure criterion, we assume that this power is removed by two of three files. That means, each file removes ≈14 MW

F The grace period is increased by a factor : $\square / \square_{GV} = 5.8$

This numerical application using simple analytical formulas demonstrates the interest of the BOPHR strategy. In fact without any intervention, the dryout time of the SGs is about 50 mn. With the passive system, this lapse of time reaches 4 to 5 h or thereabouts a factor 6 that considerably extends the grace period allowing the operator to recover the active systems (electricity power supply, auxiliary feedwater supply for the SGs).

An evaluation of this strategy has been made using a passive system including heat exchangers integrated in the vessel [6]. Here we evaluate this strategy using a system composed of immersed heat exchangers placed in derivation in the primary circuit. This circuit, designed to operate according to the BOPHR strategy, is called RP2 (**R**esidual **P**assive heat **R**emoval system on the **P**rimary circuit). This evaluation is made by studying the accidental transients usually considered as the most penalising. The calculations have been made with the CATHARE V1.3U for a 900 MWe reactor [5].

4 DESCRIPTION OF THE RP2 SYSTEM

The system is implemented on a reactor primary loop like it is shown on the figure 2. One exchanger loop is implemented on each reactor loop. The water coming from the hot leg of the reactor primary circuit is cooled within the immersed passive exchanger and then returns to the cold leg ; a natural convection flow is supposed to be established. The nozzle on the hot leg is up oriented as it is recommended in [7]. The nozzle on the cold leg is located upstream from the primary pump in order to be sure that its rotation on inertia cannot perturb the natural convection flow coming from the exchanger.

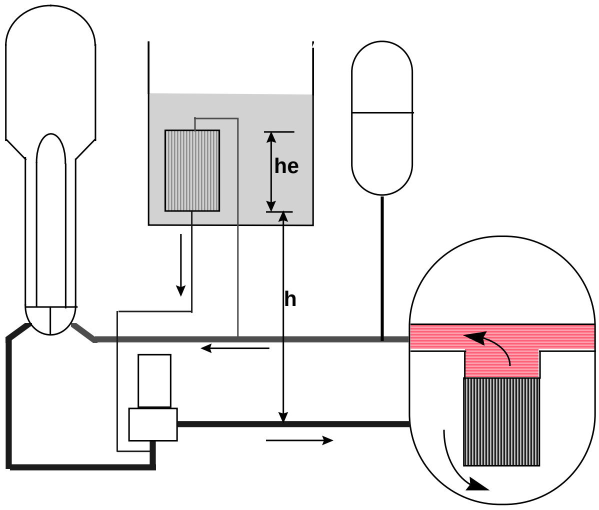


Figure 2: Passive system implementation

The exchanger is a one-through type. In nominal condition, it is designed to operate in single phase liquid flow conditions. On the primary side, water flows in tubes. On the secondary side, a natural convection regime sets up within the pool. The flow inside the tubes and the flow outside the tubes, in the vicinity of the exchanger, have opposite directions.

Main Geometric Characteristics of the heat exchanger:

External tube diameter 42.2. mm
Tube thickness 2.77 mm
Tube length 2 m
Number of tubes per heat exch. 170
Elevation of heat exchanger
above the primary leg 5.7 m

We assume that the autonomy of the heat sink, i.e. the pool, is infinite.

Remark :

The RP2 system is quite similar to the PHRS from AP600, but their missions are different. AP600 only rely upon passive systems for Design Basis Accidents. For complex sequences and severe accidents, active systems are credited as participating to the defence in depth and mitigation accident. In the present strategy, the residual power is removed jointly by active and passive system. Therefore, the passive system (RP2) is sized to ensure a redundancy should the active systems fail.

5 ASSESSMENT OF THE BOPHR/RP2

The assessment of the BOPHR strategy associated to the RP2 system is aimed at knowing the capabilities in management of accidental transients, i.e. the capability to remove heat in passive mode, and the possibility of avoiding core melt down under pressure.

The assessment is made by comparing the calculation results between the reactor without RP2 (reference case) and the reactor with RP2. For all the calculations, no active system is simulated, which means that there is no safety injection in high or low pressures, no Auxiliary Feedwater Supply system (AFS), and that the primary pumps stop on their own inertia after the emergency shutdown. Only the accumulators are available.

The retained transients are : the Blackout, LOCAs (Loss Of Coolant Accident) transients with breaks of 7.5 and 2.5 cm on a cold leg, and a steam generator tube rupture. During the calculations with the RP2, we considered a failed train to take into account the single failure criterion. The calculations were stopped when a stable situation was reached or when the temperature of the cladding had reached excessive values.

5-1 Calculation of a blackout transient (fig.3)

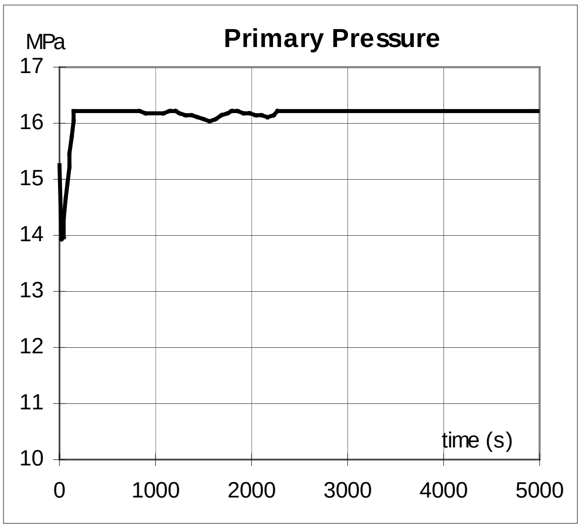
Without RP2 :

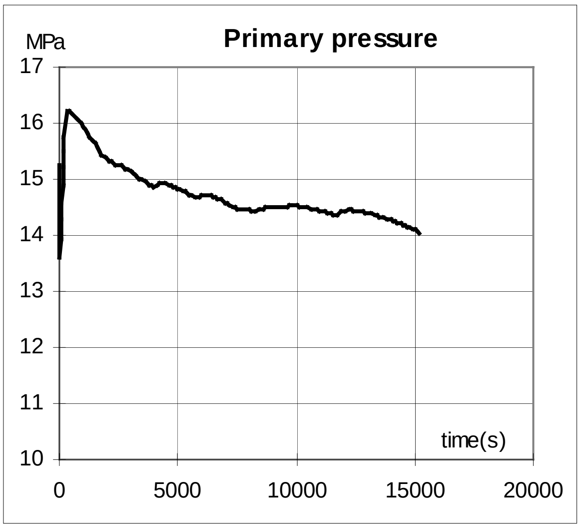
In the absence of RP2, the SGs are empty after one hour. At this time the primary water mass inventory starts to decrease through the opening of the pressure relief valves of the pressuriser. The transient continues towards a core melt down under high pressure.

With RP2 :

The SGs are empty after 5 hours. At this time, the RP2 removes all the decay heat. During the simulated transient, the primary temperatures remain under saturation level, and the primary pressure slowly decreases after a slight peak following the emergency shutdown. There is no loss in the inventory of primary water. This operating state is stable and can be considered as a safe state as long as the pool plays its role of cold source.

In the RP2 circuit, the power is removed in liquid single phase with a power per train of 15 MW at the beginning, then 13 MW at the end of the transient. The difference in temperature in entry and exit of the heat exchanger is close to 50°C.





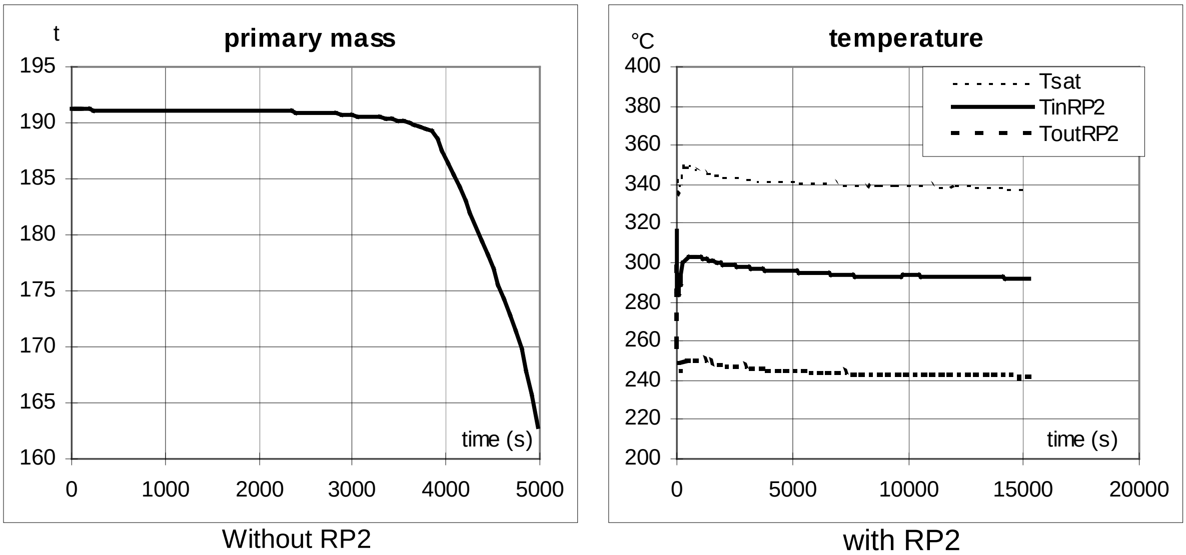


Figure 3 : Blackout transient

5-2 Calculation of the breaks

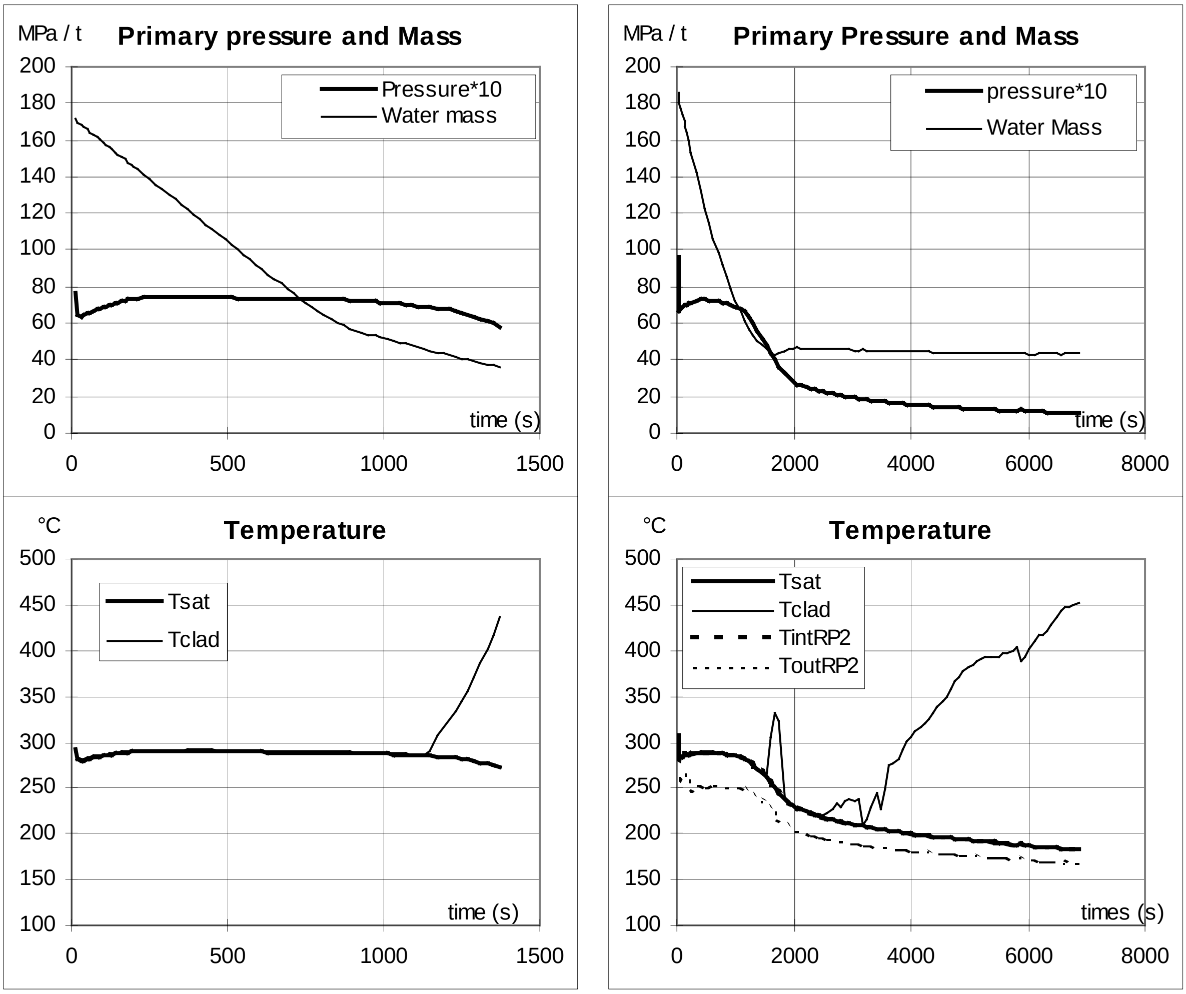
The calculations of the breaks are made by considering an adiabatic expansion of the nitrogen in the accumulators. The pressure of the injection threshold through the accumulators is of 4.2 MPa.

7.5 cm LOCA Without RP2 (fig. 4):

The primary pressure drops rapidly towards a pressure plateau (which is linked to the set pressure of the secondary circuit pressure relief valves (7.2 MPa). During this phase of 900s, the heat is removed through the SGs and through the break. After this phase, the break flow rate is essentially in steam, the heat removed through the break increases, which stops the thermal exchanges with the SG. The loss in primary water is not compensated for by the accumulators as the primary pressure is too high. At 1200s, the cladding temperature of the core starts to increase excessively which can lead to core meltdown under high pressure

7.5 cm LOCA With RP2 (fig. 4):

The primary pressure rapidly reaches 7.3 MPa. The heat removed by the SGs is very small. The core power is essentially removed by the two RP2s and the break. Depending on the two phase flow regimes in the RP2, the power of the two heat exchangers reaches a power of 44 MW. Towards 1000 s the break goes into steam, which is evidence of an increase of the heat removed by the break and a rapid depressurisation of the primary circuit. The RP2 continues to operate with a power of 25 MW (fig. 7). The drop in primary pressure allows the accumulators to compensate for the losses in water. The core dewatering starts around 3200 s at a pressure of 1.8 MPa. The calculation is stopped at 6850 s with a partially dewatered core for a maximum clad temperature of 435°C and a pressure of 1.0 MPa. There will be core meltdown at low pressure.



Without RP2 With RP2

Figure 4 : 7.5 cm LOCA

2.5 cm LOCA Without RP2 (fig.5):

The primary pressure drops in 800 s to reach 7.5 MPa. The two phase natural convection settles in the primary circuit. The removed heat through the SGs becomes insufficient at 2000 s. The SGs operate in heat pipe regime with a low secondary inventory. The break goes into steam regime at 3500 s. The calculations are stopped at 5700 s, the core is partly dewatered, its temperature is of 1100°C and the primary pressure is of 15.9 MPa, which can lead to core meltdown under high pressure.

2.5 cm LOCA With RP2 (fig.5):

During the first 2000 s, the behaviour of the reactor is very much like that of the case without RP2. From 2000 s on, the SGs no longer remove heat. Between 4000 and 6000 s, the power of the RP2 is greater than the residual heat (fig. 7) with consequently, a drop in the primary pressure. At 6000 s the primary inventory stabilises at 70 t thanks to the accumulator injection. The depressurisation continues slowly. The calculations are stopped at 21 000 s with a cladding temperature of 210°C and a pressure of 1.9 MPa. In the long term, there will be core meltdown at low pressure.

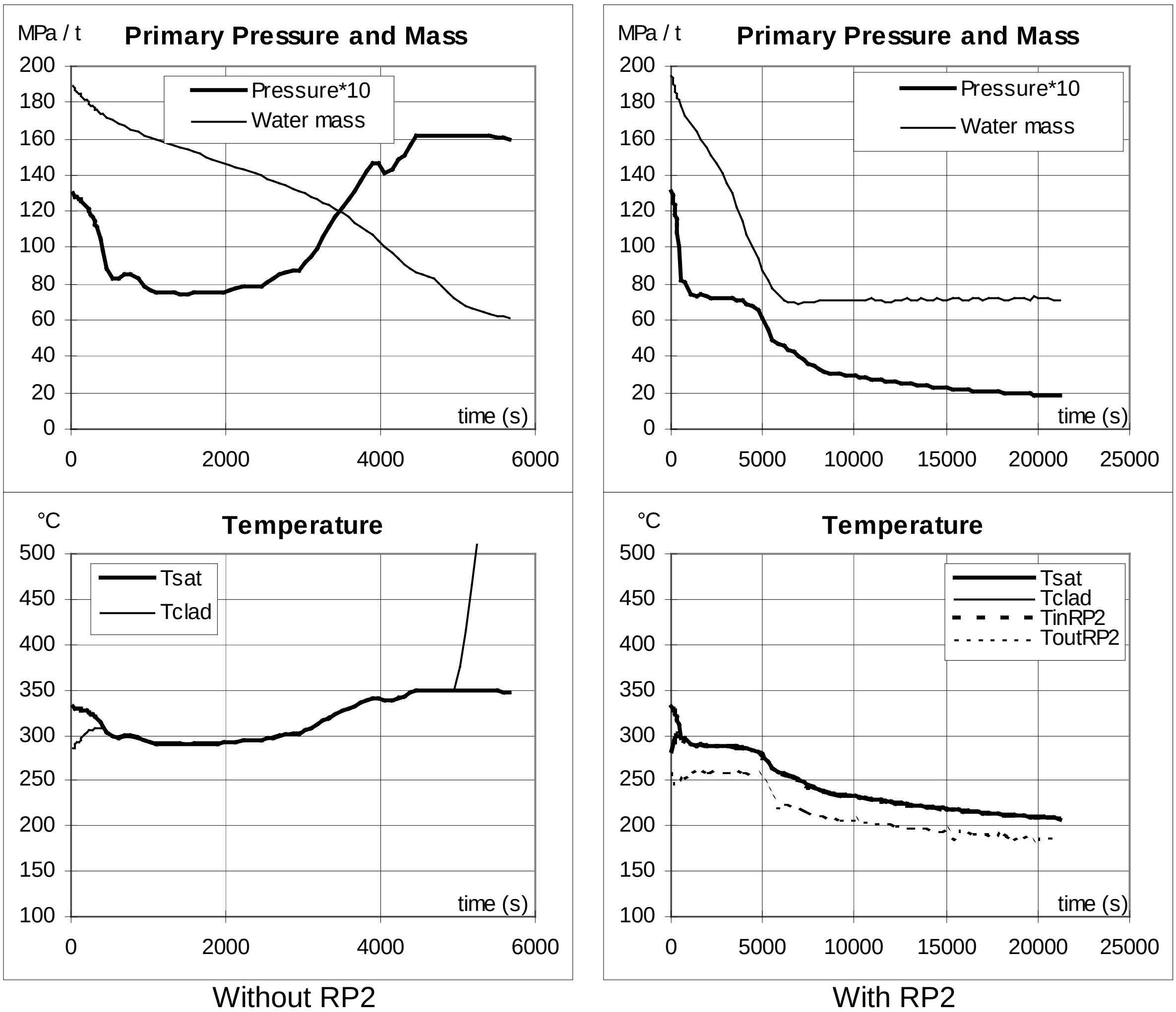


Figure 5 : 2.5 cm LOCA

5-3 Calculation of one tube SGTR (fig.6)

Without RP2:

The primary pressure drops to 7.5 MPa in 500s, the heat removed by each SG is roughly of 20 MW in the first 1500 s. From 2000 s onwards, only the ruptured SG removes practically all the heat. The calculation is stopped at 9000 s. The core is partly dewatered, the pressure is of 10.5 MPa, the releases into the atmosphere through the ruptured SG are significant (roughly 160 t). There is a risk of core meltdown under high pressure.

With RP2:

The primary pressure reaches 7.3 MPa in 1300 s. The change to two phase regime of the RP2s from 300 s onwards leads to an increase in their power. The SGs remove a significant heat up to 1000 s. Up to 1100 s, the ruptured SG operates in feed and bleed through the break, which leads to a steam release of 32 t. From 8000 s on, there is a slight flow back from the ruptured SG towards the primary circuit. The calculation is stopped at 20 000 s, the primary pressure is of 7.2 MPa. There is no leak at the break and the core is properly cooled. Core meltdown has been avoided.

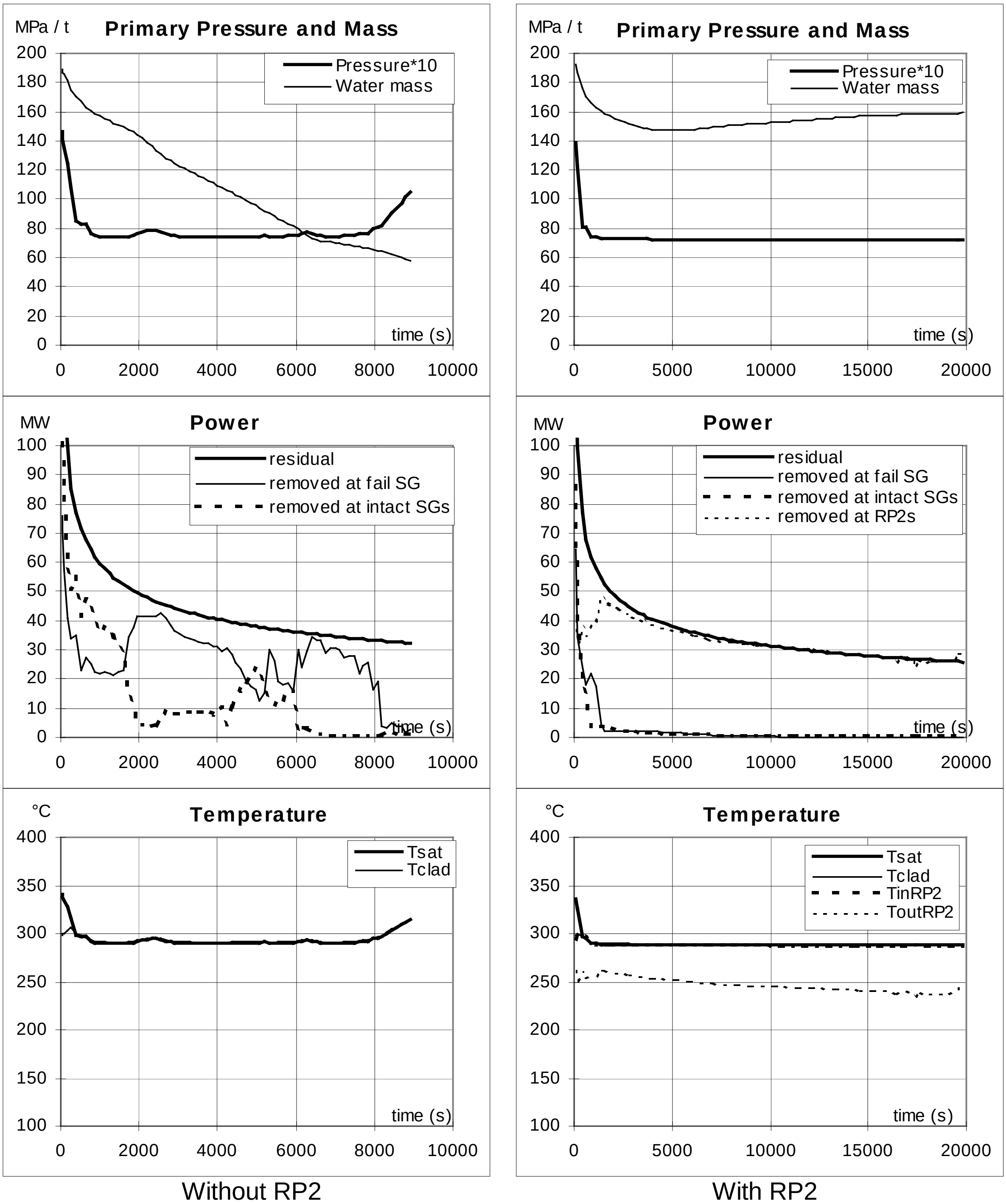


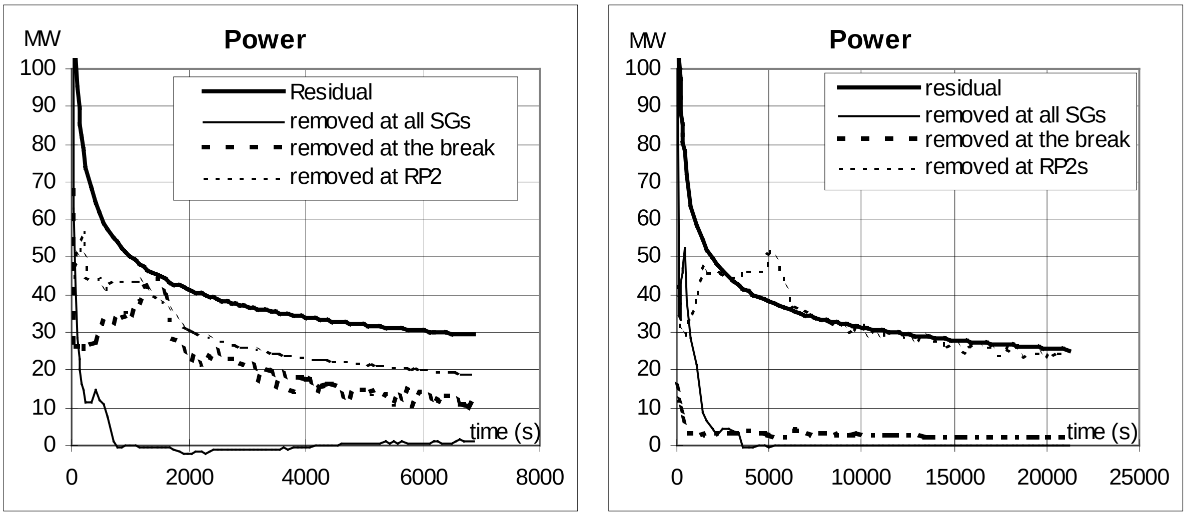
Figure 6 : One tube Steam Generator Tube Rupture

5-4 Conclusion on the study of transients

The results of the calculations show the advantage of the RP2 in managing a blackout : passive management, without opening the pressure relief valves of the pressurise and an autonomy depending solely on the pool capabilities.

In the case of the small and intermediate sized primary breaks, this system considerably increases the grace period. The first core dewatering occurs at a pressure of less than 2.0 MPa. The unacceptable temperatures for the core occur for a pressure lower than 1.0 MPa (we recall that no active safety injection has been simulated). It must also be noted that the fact that the heat removed through the RP2 increases when it goes into two phase regime, in other words with a deterioration of the primary mass inventory, which is a very favourable characteristic of this system (fig.7). This increase in the heat results from an increase of the heat transfer coefficient in the primary side during the change in flow regime (from liquid to steam).
In the case of a SGTR, the RP2 prevents core meltdown and considerably reduces releases into the atmosphere.

The performances of the RP2 show the advantage in using this system in future projects of reactor in order to simplify the safety systems. The most important potential simplifications are : the non-implementation of an Automatic Depressurisation Safety (ADS) system through a blowdown valve on the pressurise and the possible suppression of the high (or medium) pressure safety injection. In the last case, it will be necessary to define the exact mission of the RP2, and to make a Probability Risk Assessment to evaluate the pertinence of this assumption.



7.5 cm LOCA with RP2 2.5 cm LOCA with RP2

Figure 7 : Removed power for small LOCA

6 IMPLEMENTATION OF THE RP2 IN A REACTOR

In a normal condition, residual power is removed by both the active and passive systems. Management of the transient is performed in priority, with the help of the active systems or by the intervention of an operator or a automatic device. The passive system has not been designed to remove all residual power immediately following an emergency shutdown. The passive system is designed to act as backup to the active systems and it takes over without any intervention. Indeed, the passive system is automatically activated at each emergency shutdown. Quite obviously, in cases where the active systems fail, the operating condition should correspond to a change in accident category from N to N+1 with admissible limits that are less rigid (fig.8).

7 QUALITATIVE SAFETY ASSESSMENT

The brief safety analysis is conducted by analysing this system versus the recommendations of the safety authorities for the reactors of the future. [8 and 9]. This analysis (tab.1) is made by examining the characteristics of the RP2 and comparing them to the defence in depth and the safety functions. The assessment carried out here, demonstrates the potential interests of the RP2 with the BOPHR strategy, for forgiving reactor concept.

8 OTHER POTENTIAL ADVANTAGES OF THE BOPHR STRATEGY

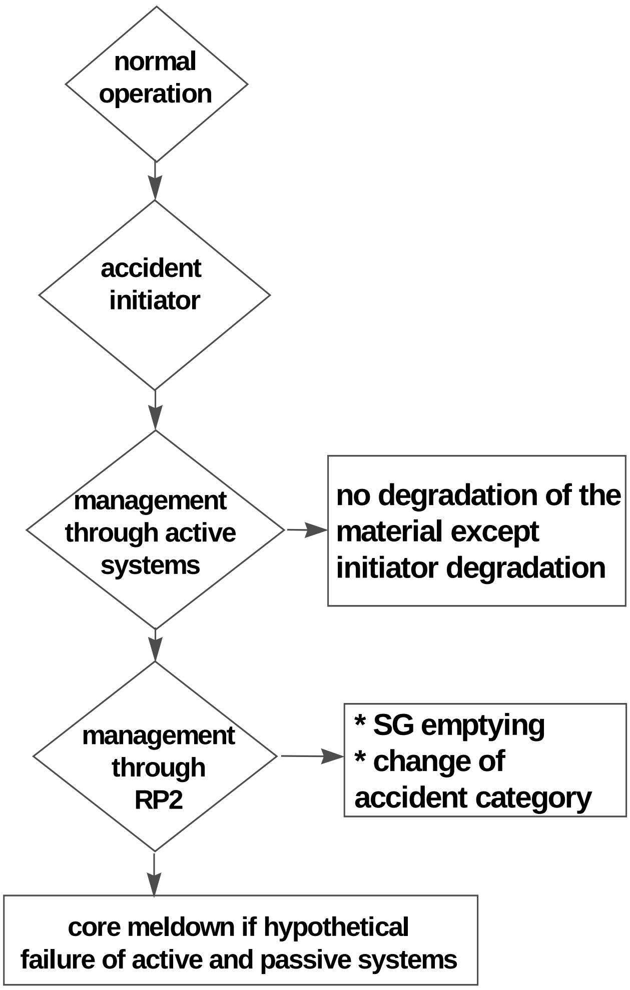


figure 8
Defence lines in BOPHR Strategy

The excessive initial investment involved in the installation of a passive system is offset against :

- a passive system of moderate dimensions in comparison to the same system but designed according to larger dimensions rendering it capable of removing all residual power at the very outset of the transient,
- a simplification of safety systems such as : no implementation of an ADS on the pressurise, the suppression of the HPSI, the downgrading of certain components like the emergency diesel generator sets,
- a reduction in the size of the active heat removal systems owing to the assistance provided by the passive system.

Defence in depth

grace period	considerable increase (roughly factor 5)	+
Autonomy	passive system of which the autonomy depends on the pool volume and on its cooling capabilities in the long term	+
simplification of systems	• elimination of HPSI • elimination of ADS • complication of primary circuit (increase of primary circuit)	+ + -
procedures	• simplified procedure : the start-up of the RP2 is not submitted to a specific analysis, as the failure of another system • Progressivity in the implementation of the defence lines	+ +
inadvertent actuation	low perturbation of the normal operation in case of sudden start-up of RP2	+
operator error	less operator errors because of : • simplified procedures • elimination of HPSI • no ADS	 + + +
diversification of systems	• simultaneous operation compatibility with other active systems (AFS) • redundancy and diversification of the residual power removal function	+ +
range of operability	• passive system • all pressures and all temperatures • removed power increased with deterioration of primary water mass inventory	+ + +
Mitigation of accidents • blackout • small breaks • RTGV	 • no core meltdown • * considerable increase of grace period * core meltdown at low pressure in case of a hypothetical failure of the LPSI • * no core meltdown * limitation of releases into the atmosphere	 + + + + +

Safety functions

reactivity control	without effect	=
heat removal :	• temperature control of the core coolant, • pressure control of the core coolant, • flow control of the core coolant, • inventory control of the core coolant, • depressurisation capability, • pressure boundary integrity.	+ + + + + =
containment of radioactivity	release limitation in case of SGTR	+

\+ item favourable to RP2/BOPHR
\- item unfavourable to RP2/BOPHR
= item without effect

Tableau 1 : Safety assessment

9 CONCLUSIONS

The prospective studies carried out at the CEA/DEN on passive systems for residual power removal have led to an accident management strategy proposal based on low power extraction performed by a passive system: the BOPHR strategy. Extraction of residual power is performed simultaneously jointly by the standard active systems and the passive system automatically actuated by the emergency shutdown signal. The management of the transient, such as control of the cooling rate, is normally ensured by the active systems. In the case of a total failure of all active systems, the passive system automatically takes over. The residual power is removed partly by the thermal inertia of the SGs thanks to the evaporation of the secondary water through the safety relief valves and by the passive system.

This strategy is associated to a heat removal system (RP2) connected to the primary circuit of a French three loops PWR. An evaluation has been made on accidental transients by simulating a totally passive behaviour. The results show that this strategy associated to the RP2 system allows the blackout and the SGTR to be correctly managed. In the case of small breaks, the primary pressure is controlled by the design of the RP2 which allows the heat removed to be increased through the deterioration of the primary water mass inventory.

The major potential interests of this type of management are : - a use of low power passive systems keeping any disturbance of normal operation to a very minimum in case of an unexpected actuation, - a backup and diversification of the active systems, - a simplification of the safety systems (no HPSI, no ADS) - a simplification of accident management operating procedures - an exclusion of a high pressure core melt - and above all, an NSSS design pointing in the direction of a forgiving reactor concept.

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INHERENT FAILURE MODES OF PASSIVE SAFETY SYSTEMS

Passive System Reliability Workshop, March 4 - 6, 2002 Cadarache, France

Päivi Maaranen, Juhani Hyvärinen
Radiation and Nuclear Safety Authority (STUK)
PO BOX 14
FIN-00881 HELSINKI
FINLAND

ABSTRACT

The paper discusses inherent failure modes of passive safety systems and their influence on passive system reliability. With "inherent failures" means a failure of the system to perform its intended function due to physical reasons and/or phenomena. Such phenomena may occur inside the system, or be due to an external cause. Focus will be on passive cooling or heat removal systems that rely on natural circulation or stored pressure energy. The paper discusses the physical mechanisms due to which system malfunctions can arise, gives examples of phenomenological analyses for several model systems that are typical to proposed advanced passive reactors, discusses various external influences that may cause system malfunctions (drawing insights from operational experience of current plants), and concludes with a discussion of the relevance of such phenomena regarding the reliability of passive safety systems. This work shows that the reliability of passive systems can be strongly influenced by external factors, although comprehensive quantification of external disturbance frequencies is not yet available.

introduction

In recent years, many different passive safety system concepts have been proposed for advanced nuclear power plants. Advantages of passive safety systems are independence of external power sources and of human factors. Disadvantages of passive safety systems are smaller capacity (small driving forces) and the limited amount of practical experience of their use.

The power sources of passive safety systems are based on natural circulation and on stored pressure energy. In order to function properly, both of these power sources require a certain internal and external thermodynamic and physical state. Inherent failure modes of a passive safety system discussed in this paper are the disturbances of these states. This topic has been discussed more extensively in [1].

Background

The passive safety systems are typically simple heat exchangers that need no power source to function but may need external control to initiate the operation. Also mechanical or moving components are rare except for check valves. Therefore the passive safety system failures due to malfunction of their components are rare compared to complex active systems.

There are two main types of inherent failures:

- a failure due to unexpected change in the internal physical state and
- a failure due to environment of the system, e.g. foreign material.

Inherent failures can occur also in active systems but appear relatively rare compared to active component failures. Moreover the large driving forces of active systems can eliminate some concerns due to internal state changes. As to external factors operating experience from current plants shows many cases in which the foreign material have got in to a system. For example, pieces of plastic are found in a suppression pool and also the things that are needed in the maintenance and in the outage of the power plant. A few examples of these cases are discussed later in this paper.

Physical possibilities of inherent failures can be reduced by extensive enough development testing. Consequences of inherent failures can be mitigated to some extent by design measures. Passive safety devices tend to require less maintenance which less expose passive systems to human errors. The failures of passive safety system due to foreign material can be reduced by proper plant housekeeping throughout the plant life time. In the light of past experience, prudent housekeeping is not a trivial task (see chapter 3.1).

For the time being there is not much of operational experience of most passive safety system concepts. This again highlights the importance of thorough development, verification and review testing. It is supposed that there are some features in safety systems, in their environment and also in using them, which will be prevailed when gaining the operational experience. That means, when gaining the operational experience the systems can be developed further more reliable. In general, the passive safety systems are rarely needed. For this reason, it is essential to design such passive safety systems of which proper operating conditions can be verified during in-service inspection and testing.

Foreign material found in safety systems

In this chapter a few examples of foreign material in safety systems are presented. There are cases reported in which the foreign material is found in a suppression pool [2], [3]. In two cases the emergency core cooling system (ECCS) suction strainers were covered by debris. In the first case the strainers were almost entirely covered by thin mat of material consisting of fibres and sludge. In the second case only a small amount of debris was found on the strainers. In both cases the floor of the suppression pool was covered by sediment. The debris on the suction strainers was analysed and it consisted of mostly iron oxides and fibres, which were of a polymeric nature. Also other material was found in the suppression pool. In the first case those things were pieces of wood, nails and hose. In the second case mostly the operational debris was found on the pool floor. It included hardhat, a pair of anti-contamination coveralls, 15,2 m of Tygon tubing, 3 nuts, 4,6 m length of duct tape, four lengths of hose, 8 to 46 m, a short length of wood, 5 cm by 10 cm and a flashlight. In the first case, totally about 635 kg of debris was removed from the suppression pool.

In the first case the debris was found when the safety relief valve (SRV) had stuck open and the suppression pool cooling was initiated. An additional pressure drop across the residual heat removal (RHR) pump strainer was detected and the reason for this was revealed to be the debris found on the strainers. In the second case the debris was detected while the plant was in a refuelling outage and the inspection of the ECCS suction strainers and suppression pool were made.

It was concluded that reported experiences of inadequate suppression pool cleanliness could have lead to unacceptable build-up of foreign material, debris and corrosion products on the strainers. This could have prevented the ECCS from providing long-term cooling following a LOCA.

There are also cases reported [3] where the remains of plastic bag were found in the RHR cooling/test return valve within the anti-cavitation trim. In the same unit a drain plug was found on the volute of the RHR pump. After a closer inspection a 10 cm diameter wire brush wheel and a piece of metal were found wrapped around a vane of the pump.

Third type of foreign material case was a part of plastic bag found in a standby liquid control (SLC) system storage tank [4]. Pieces of plastic which were 25 cm square pieces, were found while drawing a routine monthly sample from the tank. When inspecting the tank closer, several additional pieces of plastic were found floating on the liquid surface. Also further inspection revealed that there were pieces of plastic also at the bottom of the tank. Because of the low flow velocities in the tank and the buoyancy of the plastic, the SLC storage tank was thought to perform its function in a steady state condition even with plastic in the tank. Before the monthly chemistry sampling of the SLC tank contents, it is agitated by air during 10 minutes. During this agitation and immediately following it, the location of plastic material in SLC tank could not be predicted. Intended safety function may not have been able to provide during those periods.

These examples of foreign material in a system show that foreign material in the safety systems has caused risks to active safety systems. Because of lower driving forces in passive safety systems, foreign material in passive safety systems may cause a higher risk.

Examples of Passive Safety Systems

Core makeup tank and accumulator

The paper presents two examples of passive safety system concepts, typical of ones proposed for future passive plants. The first is a system of a core makeup tank (CMT) and an accumulator (Figure 1). Both of them are large tanks of borated water discharging to an emergency injection line. The pressure in the CMT is the same as it is in the primary circuit and the pressure is maintained by a pressure balance line (PBL). The accumulator, pressurised be nitrogen, has lower pressure than the normal primary pressure. CMT is a naturally circulating coolant injection system and accumulator operates with stored gas pressure. Tanks are normally isolated with check valves and may also feature (motor) isolation valves that the plant protection system operates.

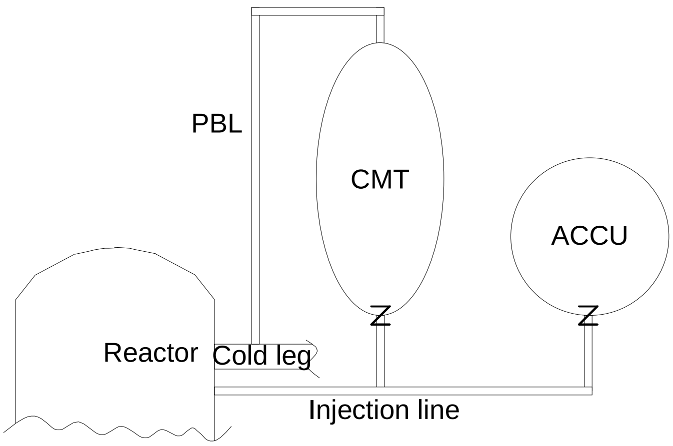

Figure 1 CMT and accumulator.

Containment cooling condenser

The other example of a passive safety system is a containment cooling condenser (CCC). It is located in the roof of the containment and in the bottom of the storage pool (Figure 2). CCC is U-shaped system of tubes. The vertical tubes are open to the storage pool and the horizontal heat exchanger tubes are below the containment building roof. CCC is meant to condense the steam accidentally released in the containment atmosphere. By condensing the steam, the pressure in the containment building decreases. CCC is a naturally circulating heat transfer system with no valves.

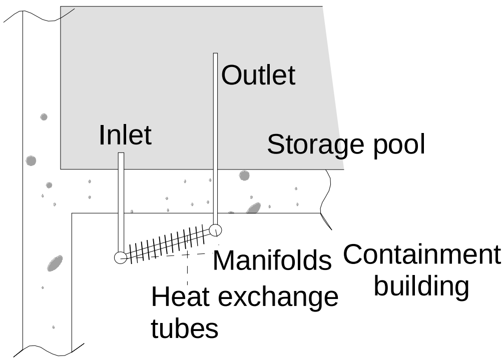

Figure 2 Containment cooling condenser (CCC).

Failure Modes

As mentioned earlier, there are two possible reasons for the inherent failures: the disturbance of the physical state inside the system or foreign material entering the system from outside. The disturbance of the physical state can eliminate the natural driving force or otherwise disrupt system operation. Foreign material in a passive safety system may restrict or blockage the flow, hence reducing the efficiency of heat transfer.

Blockage in a flow channel

The passive safety systems that are based on natural circulation are sensitive for a blockage in the flow channel. The blockage could be formed, for example, of corrosion products or of foreign material accumulated near the system inlet. If the blockage occurred in the CCC, it depends of the geometry if CCC may still perform its action. The blockage in the inlet of the CCC may cause counter-current-flow in the outlet tube. A blockage of outlet would completely prevent the operation of CCC.

Example: Porous material in the inlet manifold of CCC

The effects of the porous material in the inlet manifold of CCC on the heat transfer capacity were examined. The porous material could be for example insulation material which has accumulated in the inlet manifold of the CCC (figure 3), due to careless maintenance or housekeeping. [2] [3] [4].

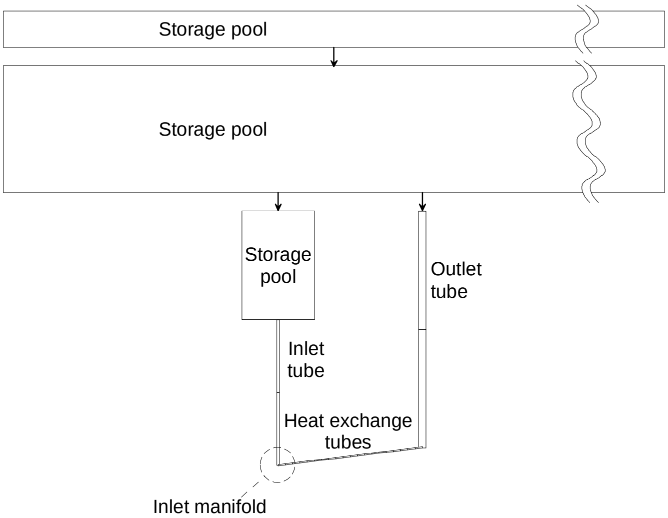

Figure 3 CCC and the porous material in the inlet manifold. The nodalisation of the Relap-model.

The CCC was modelled by the thermohydraulic code Relap5/MOD3.2.2β. The simulation was made by varying the coefficient *B* that depends on the thickness of the porous material. The coefficient *B* determines pressure drop caused by the porous material. The pressure drop is defined as follows:

$$\Delta p = k \cdot \frac{1}{2}\rho v^2 . \qquad (1)$$

The flow channels in the porous material are so small that the flow there can be assumed laminar. For the laminar flow the coefficient *k* is

$$k = \frac{B}{Re} \qquad (2)$$

The flow in the heat transfer tubes was unstable because of the boiling in the tubes. The system was simulated during 20 minutes. The average of heat transfer capacity and the outlet mass flow were calculated for a wide range of coefficient *B* (figure 4). In the figure 4 the heat transfer capacity and the outlet mass flow are shown as proportional to nominal values (in both measures the nominal value is equal to one).

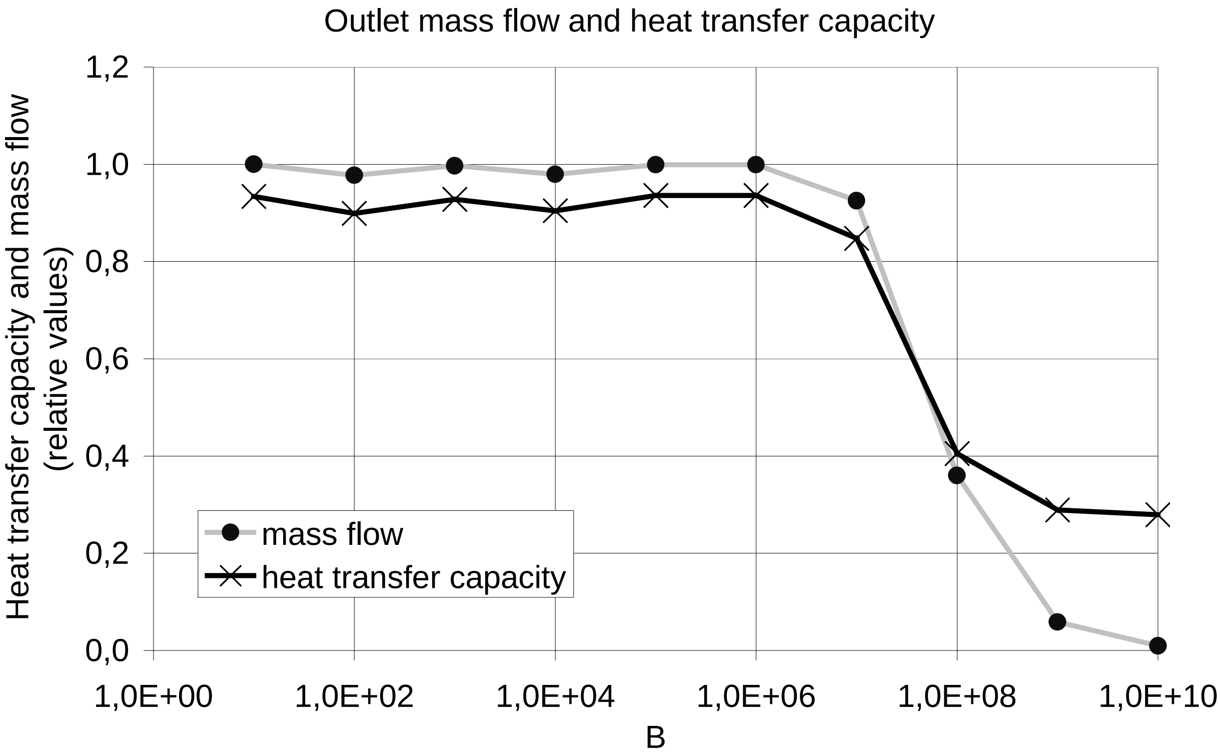

Figure 4 The average heat transfer capacity and the outlet mass flow in function of *B*.

The figure 4 shows that the heat transfer capacity remains close to its nominal value if the coefficient *B* is smaller than 10^7. Above that the capacity drops quickly to 30 % of nominal value. This equals about 10 litres (< 1 kg) of typical thermal insulation fibres. The fibres form a highly porous plug and high porosity causes small resistance. This amount of insulation fibres equals about three quarters height of the inlet tube in CCC. Figure 4 also shows that CCC can maintain its heat transfer capability even if the flow channel was not fully open.

When the inlet of CCC is blocked the minimum heat transfer capacity is limited by counter current flow limitation (CCFL). The steam flowing out of the tube limits the amount of water flowing in from the pool. The outlet of CCC is not likely to clog from outside because it expels the fluid. Instead, the inlet sucks fluid from the pool bottom and inlet is more likely to gather the debris inside the CCC. If the debris in the inlet tube is transported towards the outlet and blocks it, the heat transfer capacity descends to zero.

External disturbance

A passive safety system that uses the natural circulation as a power source requires a certain thermodynamic state to perform its action. The natural circulation is disturbed if the required temperature difference, the heat source or heat sink, is lost. The heat source of CCC may be lost if the tubes are surrounded by an insulator. For example nitrogen of the containment atmosphere or hydrogen released from the primary circuit may form an insulator. Such failure modes can be eliminated by proper design of the whole containment system, not only the CCC.

Internal thermodynamic imbalance

Initially the CMT is full of cold water. It communicates with primary system, which during a LOCA is mostly in saturation. Hence there is thermodynamic imbalance between water and the steam entering the CMT from PBL. Normally, when CMT discharges coolant, the layer of warm water forms in the upper part of the CMT. This layer prevents the steam and the liquid from mixing. The steam entering the tank from PBL forces the liquid to flow forward to the injection line. The thermodynamic imbalance is lost in this system if the steam is condensed in the cold liquid. The mixing of the fluids can happen, for example, if a strong steam jet enters the tank and disrupts the isolating warm water layer.

Initial state

The failure of passive safety system may arise if the initial state of the system was not the intended. Assume an accumulator that begins to discharge to the injection line. If the CMT is already empty and if the check valve between the CMT and injection line fails, a part of the flow from the accumulator flows into CMT and the flow to the reactor is not as big as it was intended to be.

Example: CMT and accumulator

The situation described above was simulated by Relap5/MOD3.2.2β. The system of the figure 1 was modelled. The volume of CMT was about 70 m^3 and the accumulator about 60 m^3. The initial primary pressure was 15,5 MPa and the accumulator pressure 4,25 MPa. These parameters and measures were thought to be representative of such system.

The simulation started with the normal primary pressure (figure 5). The following changes of primary pressure are thought to occur in LOCA with incomplete functioning of automatic depressurisation system (ADS). The primary pressure descends during first 10 minutes. The steam flowing from the PBL towards the CMT forces the CMT to discharge to the plenum (figure 6). The primary pressure remains on the level above the pressure of the accumulator during 50 minutes while the CMT empties. The primary pressure continues to descend and remains on 0,2 MPa. After the primary pressure has descended below the pressure of the accumulator it begins to discharge. A part of the water flow from the accumulator flows to the CMT and the other part to the reactor (Figure 6).

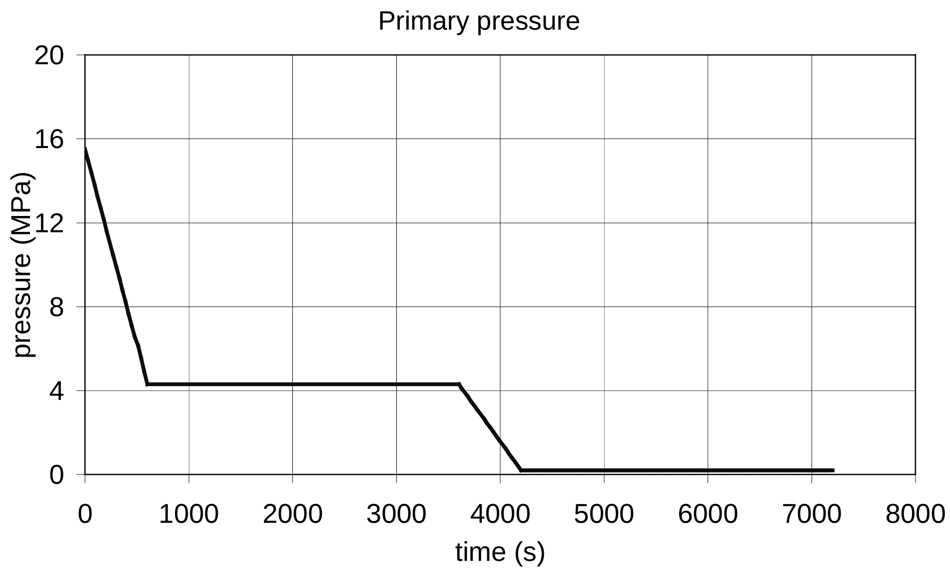

Figure 5 Primary pressure in function of time.

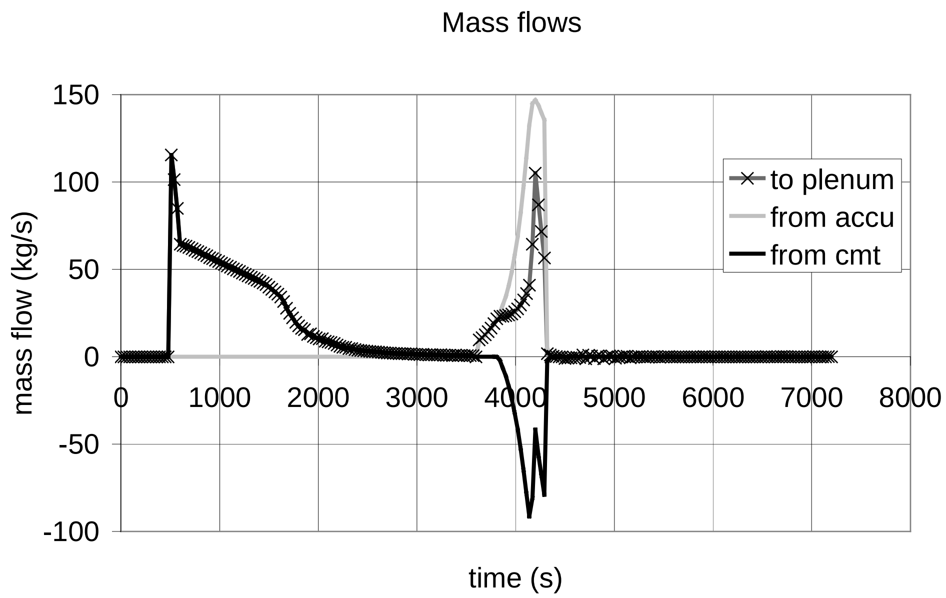

Figure 6 Mass flows from the CMT and from the accumulator to the plenum in function of time.

This example shows that unexpected failure of another system (ADS in this case) may disturb the functioning of passive safety system. More detailed analysis of whole primary circuit response, including all associated safety systems, is needed to make more concrete conclusions regarding actual plants.

Conclusion

Passive safety systems use natural circulation and stored pressure energy as power sources. These power sources have small capacities, which cause their sensitiveness to internal and external physical and thermodynamic conditions. The major advantage of passive safety system is their independence of external power sources and human factors. In this paper inherent failures of passive safety systems were discussed. There are two main types of these failures. Those failures are a failure due to unexpected internal change of physical state and a failure due to environment of the system, e.g. foreign material.

When dealing with passive safety systems the environment of these systems plays bigger role compared to active systems. Thorough development, verification and review testing of passive systems should be considered including the environment and surrounding systems. This appears also in simulated examples of inherent failures presented in this paper. Those examples were about consequences of foreign material inside the system and functioning of an emergency injection with disturbance in the internal thermodynamic state. The foreign material cases presented in chapter 3.1 show that the cleanliness and maintenance of passive safety systems and their environment is essential and should be cared of during the whole life time of the power plant.

Note also that a "passive system failure" may be more difficult to define than in case of active systems. Inherent failures may result in degraded performance of a passive system instead of a complete failure. Adequacy of degraded performance needs to be evaluated carefully for actual applications.

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INHERENT FAILURE MODES OF PASSIVE SAFETY SYSTEMS

Workshop
PASSIVE SYSTEM RELIABILITY
4th - 6th March 2002
Cadarache, France

Päivi Maaranen, Juhani Hyvärinen

STUK

2

Contents

- Background
- Examples of passive safety systems
  - CMT, Accumulator, CCC
- Failure modes
  - Blockage, external disturbance, loss of internal imbalance, initial state
- Conclusions

Passive safety systems

* PSS concepts for ANPP
* Features:
  - Simple
  - Power sources
    - natural circulation
    - stored pressure energy

- Advantages
  + independence
    · external power sources
    · human factors
- Disadvantages
  - small driving forces
  - little operating experience

Inherent failures

- Sensitive to physical and thermodynamic state
- No (or few) mechanical or moving components
  ➜ Component failures are rare

Types
- unexpected change in the internal physical state
- environment of the system
  e.g. foreign material

Can be reduced
- development testing (experience)
- housekeeping

Can be mitigated
- design measures

In-service inspection & testing

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Foreign material: operational experiences

[Refs: NRC GL 98-04, NRC IN 2002-05]

Suppression pool
- strainers: fibres & sludge
- sediment
- pieces of plastic
- operational debris
- 635 kg

RHR
- check valve: remains of plastic
- pump: drain plug

SLC tank
- closed tank
- pieces of plastic

Passive systems:
- lower driving force
- ➜ higher risk

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Work Shop on PASSIVE SYSTEM RELIABILITY, March 3th - 6th 2002, Cadarache France, Päivi Maaranen Juhani Hyvärinen

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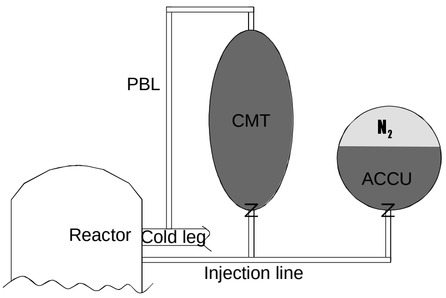
Examples of passive safety systems

Core Makeup Tank and Accumulator



- Large tanks, borated water
- CMT naturally circulating system
- Accumulator pressurized by N_2
- Isolated by check valves

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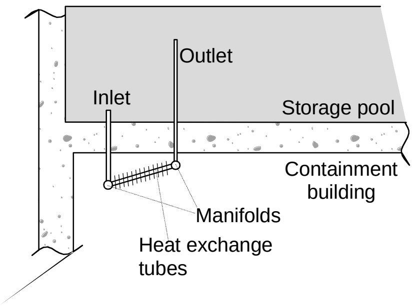
Examples of passive safety systems

CCC Containment Cooling Condenser



- Heat exchanger
- Condensates the steam in containment, *p*↗
- Naturally circulating system
- No valves or other moving parts

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Failure modes

- Blockage in a flow channel (example)
  - foreign material
- External disturbance
  - insulator
- Loss of internal thermodynamic imbalance
  - condensation
- Initial state (example)
  - valve fails

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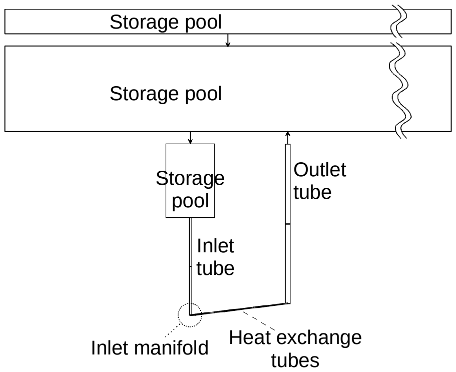
Failure modes

Blockage in a flow channel

Porous material in the inlet manifold of CCC

Calculated by RELAP5



- Effects on heat transfer capacity and mass flow
- Pressure drop: $\Delta p = k \cdot \frac{1}{2} \rho v^2$
- For the laminar flow: $k = \frac{B}{Re}$
- Coefficient *B* was varied
- Simulation: 20 min
- Unstable flow ➜ averaged values

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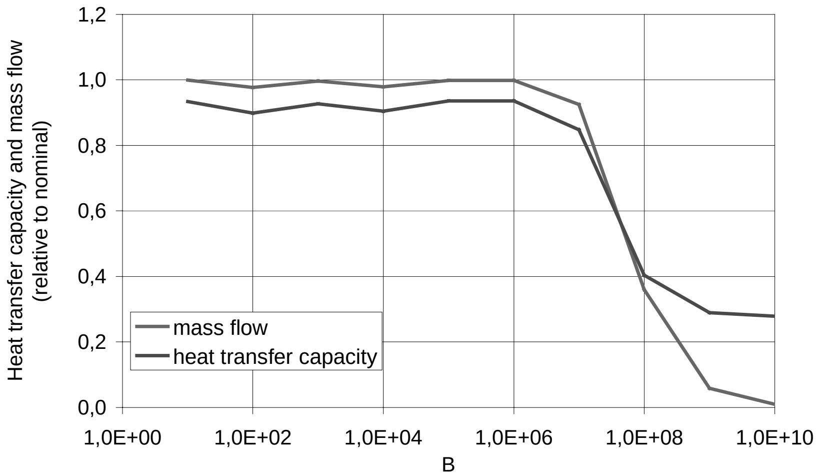
Failure modes

Blockage in a flow channel: results

Porous material in the inlet manifold of CCC

Outlet mass flow and heat transfer capacity



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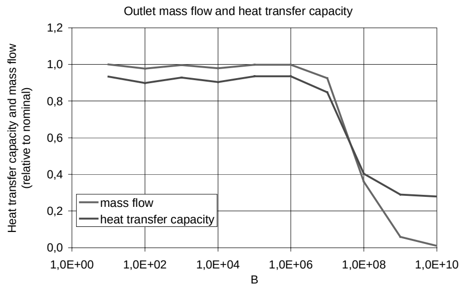
Failure modes

Blockage in a flow channel: results (2)

Porous material in the inlet manifold of CCC



- Heat transfer capacity ≈ nominal value if $B < 10^7$
- $B = 10^7$ corresponds 3/4 of inlet tube height of insulation fibres
- $Q_{min} \approx 0{,}3{*}Q_{nominal}$
  - limited by CCFL at outlet
- Inlet more likely to clog

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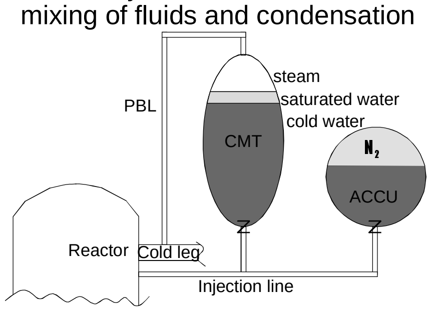
Failure modes

External disturbance

- Natural circulation needs ΔT
- Loss of ΔT: insulation
  - N_2 in containment
  - (H_2 in core melt)
- Important: proper design of whole system

Loss of internal thermodynamic imbalance

- Steam entering the CMT from PBL
- Thermodynamic imbalance lost if mixing of fluids and condensation



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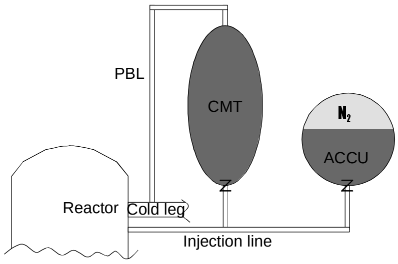
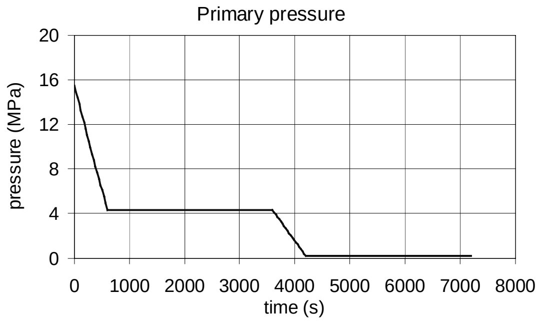
Failure modes

Initial state

Example:
- CMT empty
- accumulator begins to discharge
- CMT check valve fails
- CMT 70 m^3
- ACCU 60 m^3





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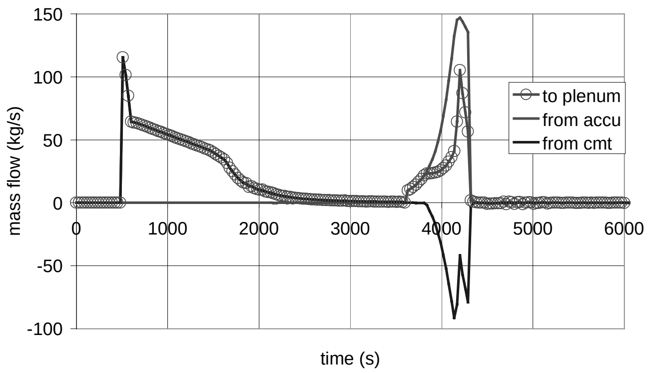
Failure modes

Initial state: results



- CMT empties
- ACCU begins to discharge
- Part of the ACCU-flow to CMT

➜ Unexpected failure of another system may disturb the passive safety system momentarily

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Conclusions

- Passive safety systems:
  - natural circulation, stored pressure energy
  - small capacities
  - sensitive to physical and thermodynamic conditions
- Failure modes:
  - unexpected change of physical state
  - environment (foreign material)

- Development, verification and review testing
  - system with the environment
- Housekeeping
- Difficulties in definition:
  - degraded performance
  - complete failure
- Adequacy of degraded performance

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An ATHLET case study of thermal hydraulic system reliability: active versus passive

H.Glaeser, C.Müller, GRS

Abstract

In order to study the specific reliability aspects and the differences between active and passive thermal hydraulic systems ATHLET simulations have been carried out on a cooling loop. This loop is similar to the PANDA isolation condensation system and it is run both by natural convection and by the use of a pump with a small pump head just enough to produce the equivalent of the natural convection. In the computer simulations remarkable differences in the system performance are found which are evaluated and analyzed. The simulation shows that in the passive system key variables behavior is much smoother than in the active system. Performance figures are derived to quantify the differences in a mathematical way. This gives deeper insight in the different behavior of active and passive systems and reliability aspects connected with this specific behavior.

In the study it is also investigated how the performance changes due to design modifications. The results indicate that passive systems have to be designed more carefully that active systems and that the design of passive systems is much more sensitive to design modifications than active systems.

It is also investigated how the reliability of both systems is affected by beyond design operational conditions and if it is true that active system can cope better with these conditions. The simulations results support this current view but it is not possible to quantify the observations in mathematical fashion.

An ATHLET Case Study of Thermal Hydraulic System Reliability: Active versus Passive

H.Glaeser, W.C.Müller
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
OECD Workshop „Passive System Reliability"
Cadarache, France
4.-6.March 2002

14-Mar-02

Overview

Topics:

- Passive System Reliability
- Hypothesis on Passive System Reliability
- Thermal hydraulic reference problem: PANDA natural convection
- Measuring the Failure Probability
- Design conditions: active versus passive
- Improved design
- Beyond design basis performance: active versus passive
- Conclusions

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Probabilistic safety analysis

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(„Probabilistic passive" not the same as „Thermal hydraulic passive")

Usual Procedure

- System Reliability
- Fault tree analysis
- Thermal hydraulic systems:
  Unavailability negligible

Presumed disadvantages of passive T-H-systems

- Behavior details not well-defined
- Behavior details hard to predict

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Different criteria for reliabilty?

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Lessons learned from car pollutant emissions

	gaseous hazardous	particulate hazardous
gasoline	yes	no
diesel	no	yes

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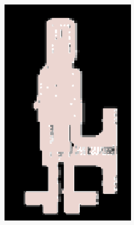
Differences in behavior: active versus passive

Sample Relief Valve:

- Active: Power Operated Relief Valve
  - Exact control of set points
  - Exact control of mass flow
  - Manual action in case of stuck valve
  - No System Pressure Required for Operational Testing
  - Depending on external power
- Passive: Spring Loaded Relief Valve
  - Set points vary depending on situation
  - Mass flow may cause valve chatter
  - No options in case of stuck valve
  - Independent of external power



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Objective of Investigations

Hypothesis:

- Passive system show strong deviations from nominal design conditions
- Passive systems have a higher probability of „rare events", which may lead to failure
- Passive systems have a higher failure probability in beyond design basis conditions
- Passive system performance is more difficult to predict by simulation (not investigated here)

Investigation by numerical experiments with the thermal hydraulic ATHLET code
Test case: Natural convection similar to the PANDA IC loop

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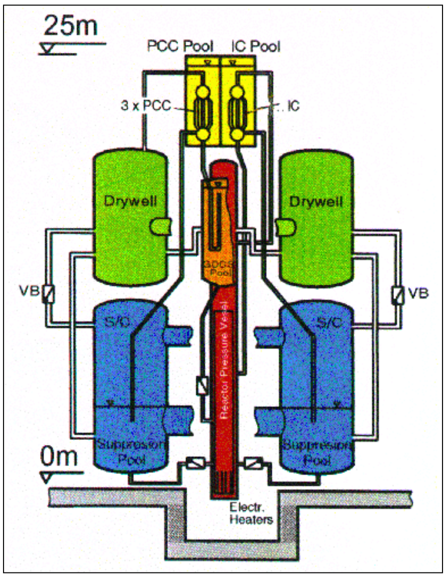
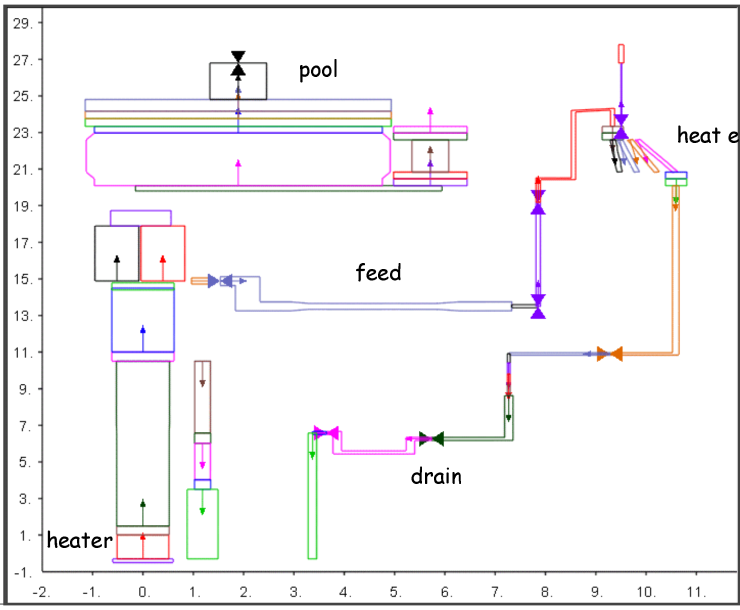
PANDA test facility



Diameters: RPV ~ 1m, IC feed ~ DN 150-DN100, IC drain DN 40-DN 50, IC pool ~ 6m

ATHLET calculation of passive reference case



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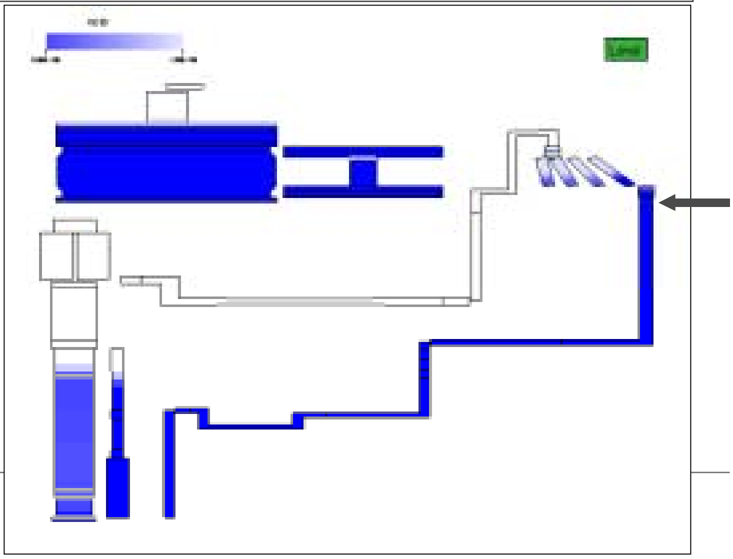
ATHLET simulation modell

ATHLET calculation of passive reference case

Starting conditions: Draining of the heat exchanger after valve opening
Transient: System cools down to a stationary value in 15000 s
Stationary system while pool evaporates until 75 000s
Afterwards : Breakdown



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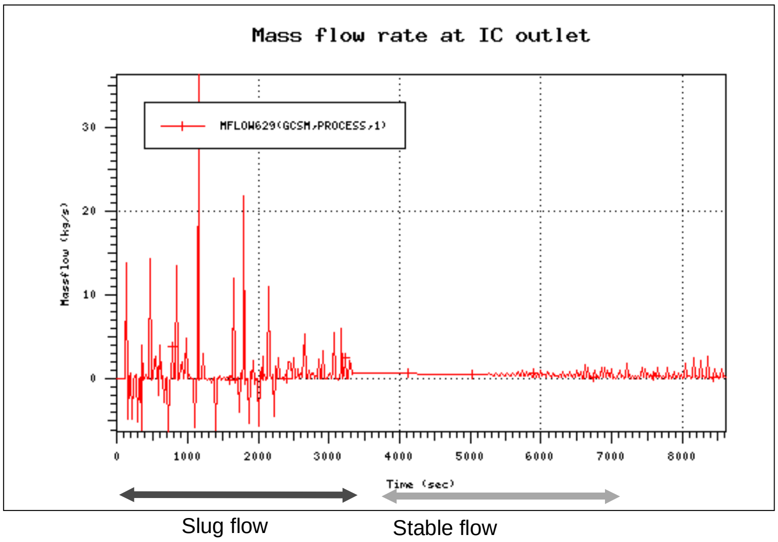
passive reference case

Mass flow rate at IC outlet



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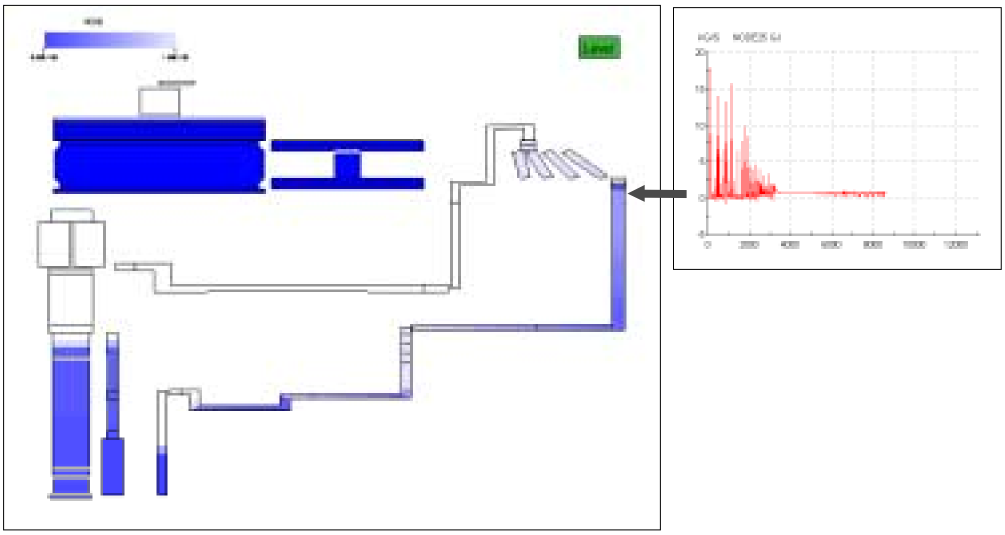
passive reference case

Unstable U-tube oscillations



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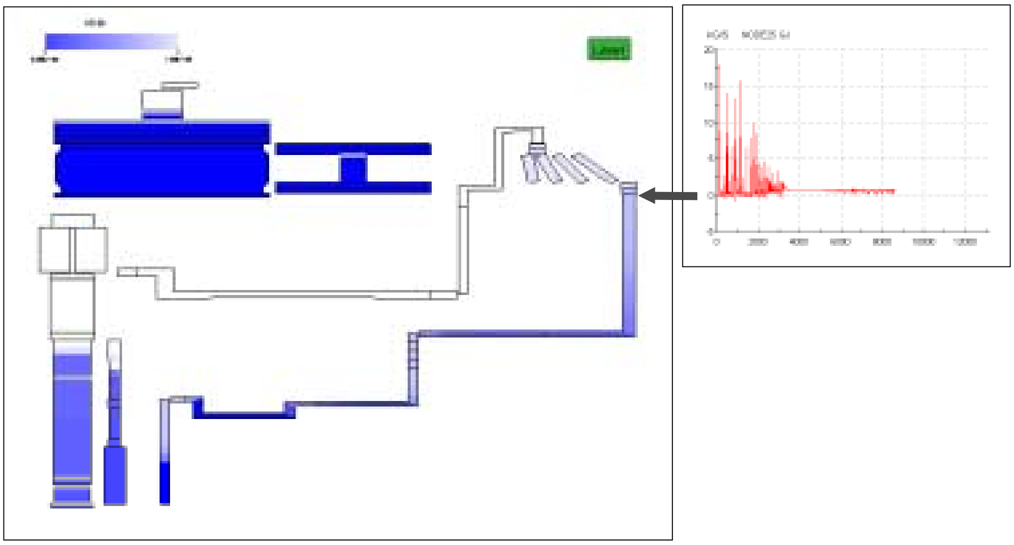
passive reference case

Natural convection stable mass flow



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Increased probability of rare events

Hypothesis:
Strong spurious oscillations and increased probability of rare events indicate imminence of system failure, e.g.
by dynamic pipe loadings or circulation breakdown

- Deviation from diagonal in normal probability plot
- Recurrence time (not investigated here)
- Deviation from prediction (not investigated here)

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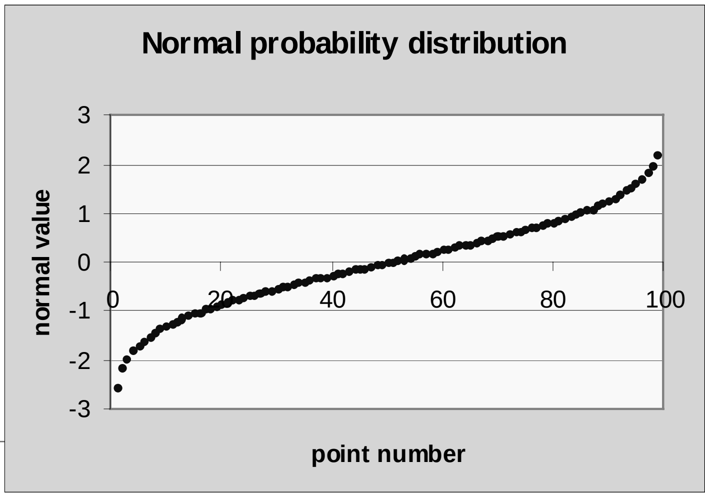
Normal probability plot

All points ordered in ascending order
Normal probability distribution
Each point is assigned its normal value
The same is done for the observed distribution.



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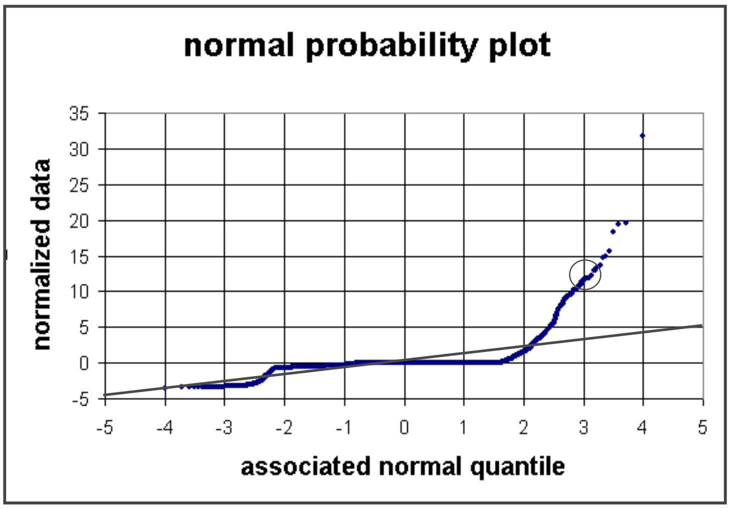
reference case: Test for rare events

Normal probability plot of mass flow



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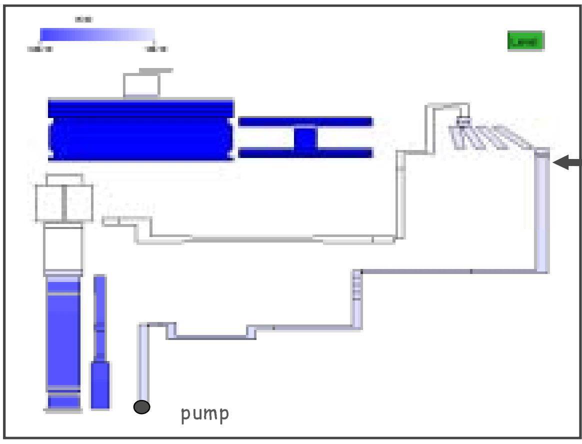
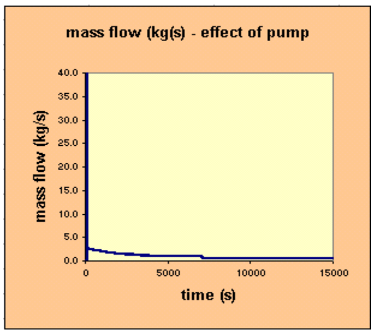
active case with pump

Same transient, but no visible oscillations at all !!
Normal distribution





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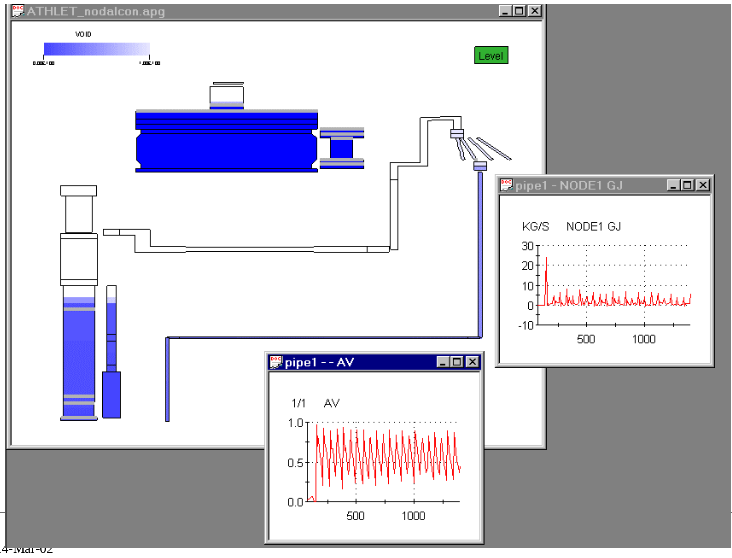
passive case improved design

New drain pipe design: no U's and long straight runs
-> regular oscillations and reduction of rare events



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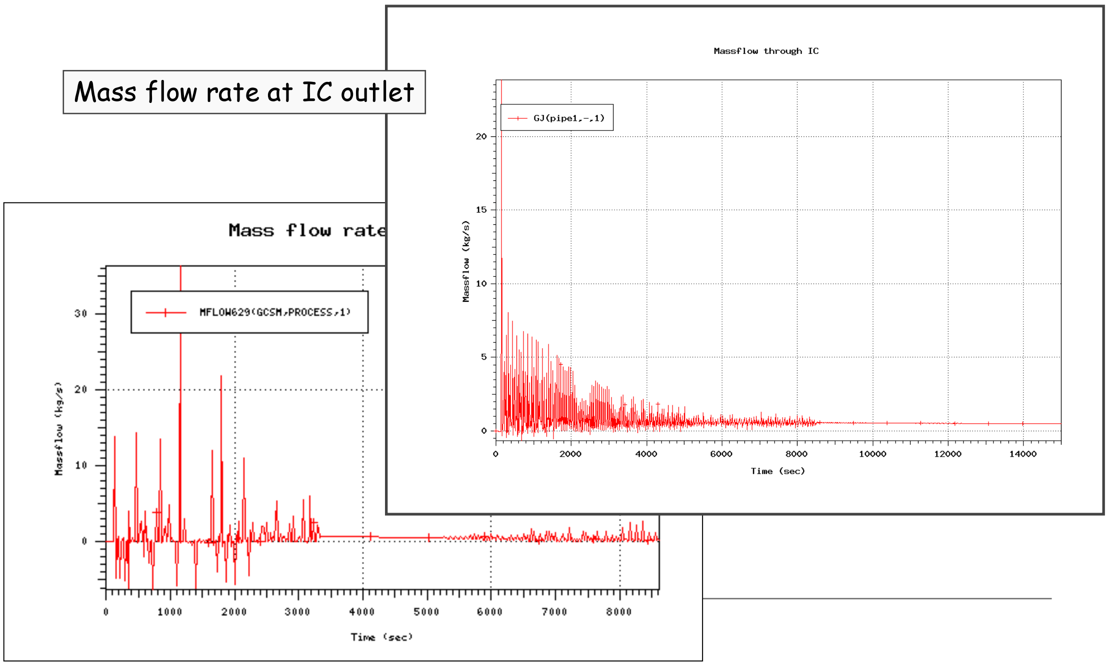
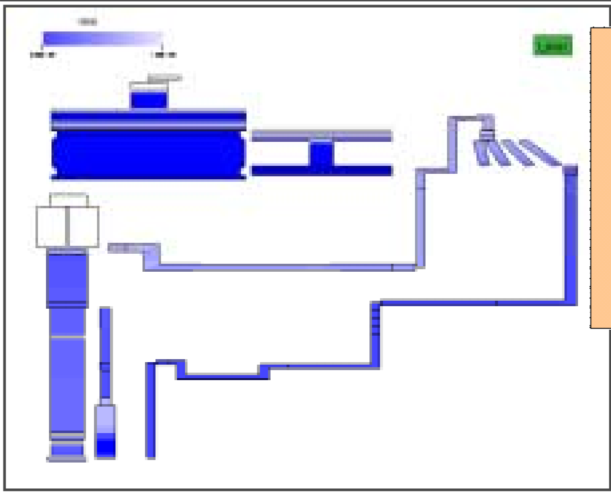
passive case, improved design

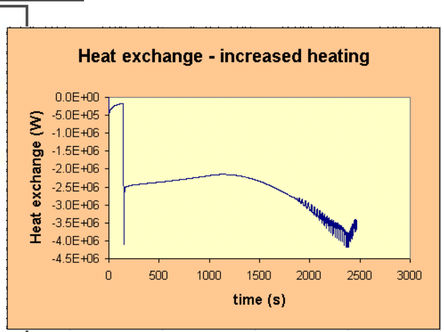
Mass flow rate at IC outlet



Beyond design basis conditions: test 1

Overpressurization due to heater power increase
Failure at 220 bar
Failure time the same for active and passive system





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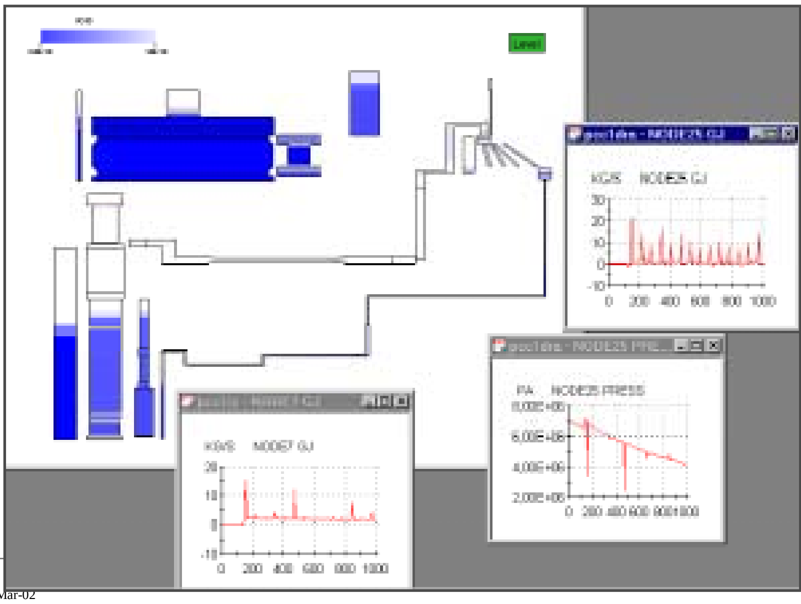
Beyond design basis conditions: test 2

Effect of non-condensables, insufficient venting



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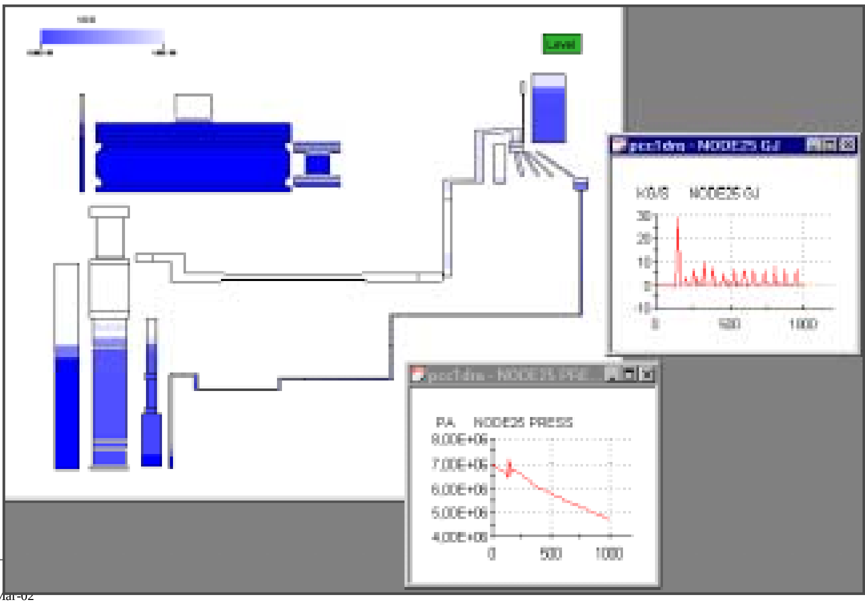
Beyond design basis conditions: test 2

Effect of non-condensables, insufficient venting, active pump



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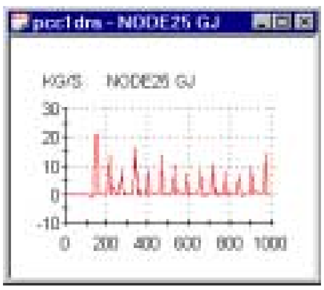
Beyond design basis conditions: test 2

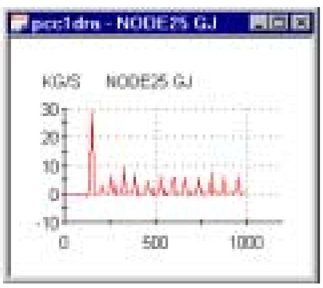
Effect of non-condensables, insufficient venting:

Main effects:
- reduced oscillations
- improved IC heat transfer
- lower final pressure level (15 instead of 20 bars)

passive



active



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Active versus Passive

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Conclusions:

- Hypothesis partly supported by numerical experiments
- When combining all aspects, passive systems may not be superior
- Passive systems as a backup system for active systems
- Results suggest further investigations by simultaneous experiments and numerical simulations

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News on ATHLET reference case

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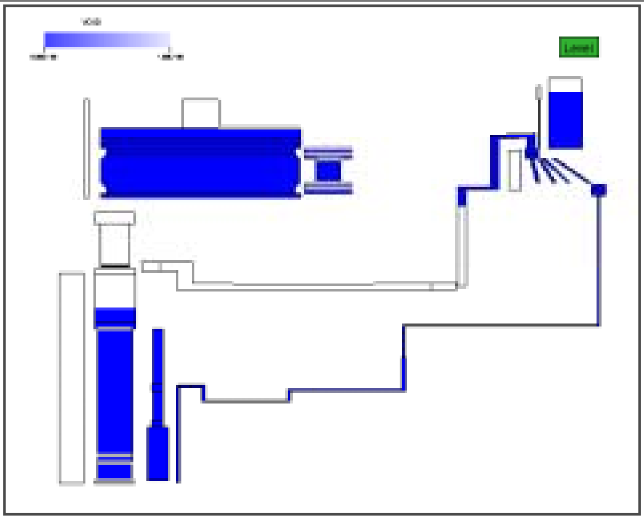
Additional elementsd for automatic setting of initial levels



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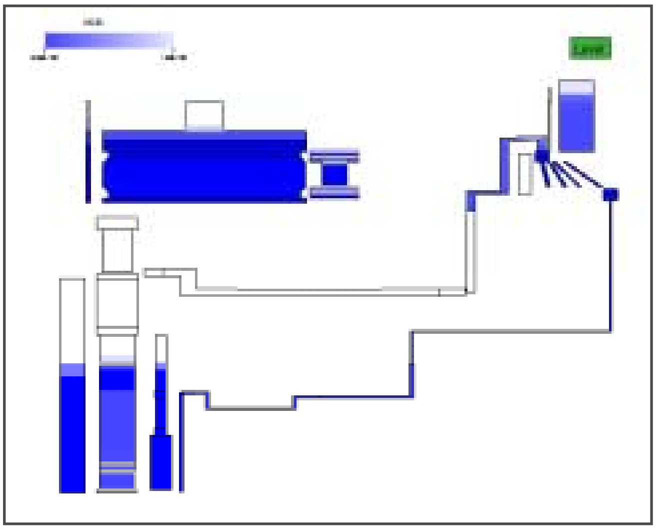
News on ATHLET reference case

Levels after 50 s



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News on ATHLET reference case

Automatic run of 75 "discrete" cases

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athlgo.22.e82449  athlgo.47.e82474  athlgo.72.e82499  panda.58.pd
athlgo.22.o82449  athlgo.47.o82474  athlgo.72.o82499  panda.59.pd
athlgo.23.e82450  athlgo.48.e82475  athlgo.73.e82500  panda.6.pd
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14-Mar-02

PASSIVE SYSTEMS ANALYSIS FOR DECAY HEAT REMOVAL

L. BURGAZZI
ENEA FIS-NUC BOLOGNA
Via Martiri di Monte Sole, 4 40129 Bologna Italy
Tel. +39 051 6098556 Fax +39 051 6098279
Email: burgazzi@bologna.enea.it

Abstract

A systems analysis is deemed the initial step to be accomplished as support for the development of a methodology aimed at the reliability assessment of passive safety systems, providing important input to further more detailed quantitative studies (e.g. uncertainty and sensitivity analysis), in order to qualify and endorse the methodology itself (cf. RMPS project).

The present study is concerning Passive Systems designed for Decay Heat Removal relying upon natural circulation which foresee, for the most part, a condenser immersed in a cooling pool (i.e. the Isolation Condenser System) and consists of two parts: the qualitative safety and systems analysis and the reliability analysis.

The former one adopts a hazard identification used qualitative method, as FMEA (Failure Mode and Effect Analysis), to assess the potential failure modes and their consequences associated with the passive system operation. Main purpose of this analysis, in the present context, is the identification and evaluation, on a qualitative basis, of the sources of uncertainties related to passive system performance, in terms of parameters judged critical for the natural circulation performance/stability and which are characteristic of the recognised failure modes: in addition a qualitative viewpoint importance and sensitivity analysis is presented.

The latter part focuses on the reliability study about the Isolation Condenser System, performed applying the standard reliability tools (i.e. fault trees) on a simplified model, where the natural circulation, upon which the system relies, failure probability is estimated in terms of failures of systems/components designed to assure the best conditions and mechanisms for natural circulation stability and passive operation: the result provides a preliminary appraisal of the whole system unavailability, included therefore also the natural circulation together with the relative weight.

1. INTRODUCTION

Advanced Light Water Reactors use passive systems for both accident prevention and mitigation: although their unavailability due to hardware failure and human error should be significantly smaller with respect to the active ones, the deviations of the natural forces or physical principles, upon which they rely, from the expected conditions can impair the performance of the system itself, adding therefore uncertainty in the system behaviour.

For instance, with reference to passive systems relying upon natural circulation (i.e. thermal hydraulic passive systems), the coolant flows predicted to be delivered to these systems can be subject to significant uncertainties, which in turn can lead to a significant uncertainty in the predicted thermal hydraulic performance of the plant under accident conditions. This calls for the evaluation of the thermal-hydraulic unreliability to be accounted for in the probabilistic safety analysis studies.

It’s worth noting that consideration of passive function reliability assessment, that is the probability of failure of the natural circulation upon which the system operation relies makes a subtle difference, as compared to the reliability assessment of passive systems.

A detailed systems analysis is required in order to support the approach to passive systems reliability assessment, providing important input to further more detailed quantitative studies (e.g. uncertainty and sensitivity analysis).

The matter is treated with regard to the previous thermalhydraulic analysis performed on a system designed for decay heat removal, basing on natural circulation and provided with the Isolation Condenser, chosen as reference system in the preliminary phase of the RMPS project (ref.1): this is in order to accomplish a more relevant study and to further endorse the aforementioned analysis.

The present study consists of two parts: the qualitative safety and systems analysis and the reliability analysis.

The former one adopts a hazard identification used qualitative method, as FMEA (Failure Mode and Effect Analysis), to assess the potential failure modes and their consequences associated with the passive system operation. Main purpose of this analysis, in the present context, is the identification and evaluation, in a qualitative way, of the sources of uncertainties related to passive system performance, in terms of critical parameters for the natural circulation performance/stability, identified by the failure modes: in addition an importance and sensitivity analysis from the qualitative viewpoint is introduced.

The latter part presents a reliability study about the Isolation Condenser System, performed applying the standard reliability tools (i.e. fault trees) on a simplified model, where the natural circulation, upon which the system relies, failure probability is estimated in terms of failures of systems/components designed to assure the best conditions and mechanisms for natural circulation stability and passive operation: the result provides a preliminary appraisal of the whole system unavailability, included therefore also the natural circulation together with the relative weight.

2. SYSTEM DESCRIPTION

In accordance with IAEA passive system categorisation (ref.2), the Isolation Condenser is classified as a type B passive system, that is relying on natural circulation The Isolation Condenser system is designed to remove excess sensible and core decay heat from the BWR reactor by natural circulation, when the normal heat removal system is unavailable, after any of the following events:

- Sudden reactor isolation from power operating conditions ;
- Reactor hot standby mode ;
- Safe shutdown conditions.

Its main purpose is to limit the overpressure in the reactor system at a value below the set-point of the Safety Relief Valves (SRV), preventing unnecessary reactor depressurisation. The IC system (Figure 1) basically consists of a number of totally independent loops, taking into consideration a redundancy degree, each loop contains a heat exchanger that condenses steam on the inner tube side and transfers heat to the water in a large pool, localised in the reactor building and above the reactor containment, which is vented to the atmosphere. The IC is connected by piping to the reactor pressure vessel, and is placed at an elevation above the source of the steam in the RPV. The steam connection between the vessel and the IC system condenser is normally open and the condensate line is normally closed. This allows the IC and drain piping to fill with condensate which is maintained at a subcooled temperature by the pool water during normal power operation of the plant. When operation of the IC system is required, the valves are opened, the steam flows directly from the reactor into the condenser, and when steam is condensed the condensate drains into the reactor vessel by gravity via a condensate return line. The flowrate is determined by natural circulation. The primary side of the condenser is also provided with vent lines to remove non-condensable gases which may reduce heat transfer rates during extended periods of operation.

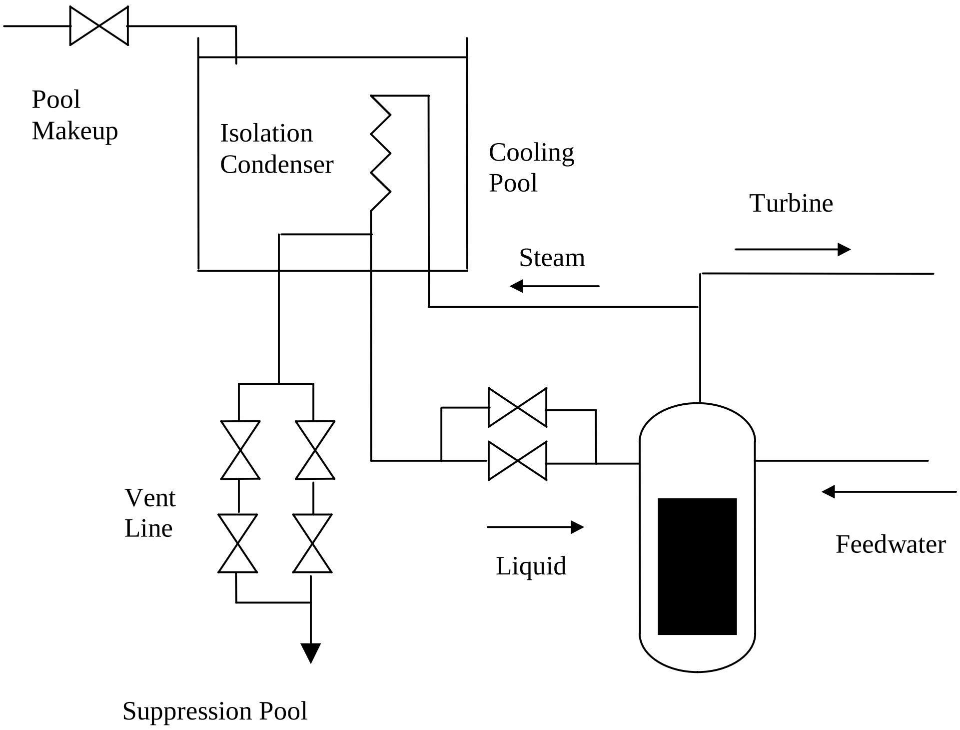


Fig. 1. Isolation Condenser of a BWR

3. OVERVIEW ON UNCERTAINTIES RELATED TO PASSIVE SYSTEMS

Correlations, data and codes needed for the deterministic description of the mission already exist to calculate nominal plant parameter evolution.

The overall uncertainty relating the thermal hydraulic analysis consists of two kinds of uncertainties:

– Uncertainties related to thermal hydraulic code

– Uncertainties related to natural circulation performance

With reference to the former class, uncertainties may have different origins ranging from the approximation of the models characterising any physical phenomena, to the approximation of the numerical solutions, to the lack of precision of the values adopted for boundary and initial conditions, and to the parameters that are the input to the phenomenological models. The amount of uncertainty that affects a calculation strongly depends upon the involved area in the technology and upon the sophistication of the adopted models and modelling techniques. This is even more relevant as far as natural circulation is concerned, due to the small engaged driving forces and the thermal hydraulic phenomena affecting the system performance that may induce large uncertainties.

Consequently various methodologies have been developed in order to evaluate the overall uncertainty in the physical model predictions and some efforts have been made aimed at the internal assessment of uncertainty capability for thermal hydraulic codes (ref.3, 4).

In the present study the uncertainties pertaining to the code are not accounted for, focusing the attention on the uncertainties relative to natural circulation. It’s worth stressing that this class of uncertainties is of epistemic nature, that is related to the state of knowledge about the studied object/phenomenon.

In general other sources of uncertainties derive from:

- model uncertainty, being a model a mathematical representation of a relationship. A model is uncertain if there is a lack of knowledge about the true relationship and/or about the effect of simplifications and omissions (intentional and unintentional).

- completeness uncertainty due to intentional or unintentional modifications, simplifications, omissions of phenomena and processes: the first ones are treated as model uncertainties while the second ones can naturally not be the subject of the uncertainty analysis.

These kinds of uncertainties fall within the code uncertainties.

4. QUALITATIVE ANALYSIS

The approach to the functional reliability, defined as the probability of the mission failure, entails, first of all, the identification of the relative failure modes (e.g. loss of the primary boundary, presence of non-condensables, thermal stratification, cracking, oxidation, etc.) together with the related critical parameters and the uncertainties associated with their evaluation.

Therefore the main steps of the qualitatively conducted uncertainty analysis are the following:

1. Identification of the uncertainties related to the physical process, that is identification of either the design parameters of the passive system or the parameters that characterise the mission (i.e. natural circulation) failure

2. Quantification of the uncertainties, that is assignment of range and distributions associated with the l parameters

3. Sensitivity Analysis, devoted to the identification of most contributors to the final result and to the overall uncertainty

Starting point of the analysis is the identification of the design parameters (RPV pressure, RPV collapsed level, pool level, etc.) coming from the connection between the passive system and the complex system, into which the passive system is inserted and by which the passive system is affected; the assessment of the potential failure modes affecting the system and their consequences on its performance and finally the identification of the parameters associated with the failure modes, defined as “critical parameters” cover the rest of the first step.

A qualitative analysis is deemed necessary in order to identify potential failure modes and their consequences associated with the passive system operation. The aim of this analysis is to identify the parameters judged critical for the natural circulation performance/stability allowing to associate to each of the failure modes a proper parameter direct indicator of the failure cause.

The methodology is combined with a hazard identification used qualitative method as FMEA (Failure Mode and Effect Analysis).

The FMEA approach application has implied the introduction, in addition to mechanical components of the system (piping, drain valve, etc.), of a "virtual" component identified as Natural Circulation and its evaluation in terms of potential "phenomenological" factors (the list of these includes e.g. non-condensable gas build-up, thermal stratification, surface oxidation, cracking, etc.), the consequences of which can impair or stop the natural circulation upon which the system is relying, and the identification of the relative critical parameters (non-condensable fraction, undetected leakage, heat loss, etc.).

The analysis pointed out several factors leading to disturbances in the Isolation Condenser system;
the list of these includes:

- Unexpected mechanical and thermal loads, challenging the primary boundary integrity
- HX plugging
- Mechanical component malfunction, i.e. drain valve
- Non-condensable gas build-up
- Heat exchange process reduction: surface oxidation, thermal stratification, piping layout, etc.

Finally, as previously assumed, this qualitative analysis allows to detect a set of critical parameters affecting the natural convection reliability and to be accounted for in further probabilistic analysis (ref.1):

- Non-condensable fraction
- Undetected leakage
- POV(Partially Opened Valve) in the discharge line
- Heat loss
- Piping inclination
- HX plugged pipes

The second step consists in the quantification, expressed in probabilistic terms, of the present status of parameter uncertainties by subjective probability distributions.

Since data uncertainty is being treated without the possibility to use information from a likelihood function, the simplest form of prior knowledge is based on plausible considerations. Such considerations provide for instance limits beyond which the fixed true yet unknown parameter value justifiably can not lie. Additional information may justify not to use a uniform distribution over the intervals derived from the limits but to choose a subjective probability distribution that exhibits some characteristic behaviour towards the endpoints of the interval (for instance a triangular or truncated normal or lognormal distribution, etc). Consequently, because of the lack of operational experience and data, item 2 is satisfied by assigning either the range or the probability distributions pertaining to both design and critical parameters arbitrarily, basing on expert judgement and engineering assessment, and discrete pdfs have been selected (ref.1): clearly this is deemed as a critical issue that ought to be further inquired in order to add credibility to the whole methodology and to foster and qualify the predicted results.

The task of the sensitivity analysis (item 3) is to identify main contributors to the performance of passive safety systems, thus allowing to reduce the final uncertainty in the predicted results.

The difficulties in the identification and quantification of the sources of uncertainty associated with thermal hydraulic performance of passive systems, arosen in the previous chapter, forces the analysis to be performed in a qualitative way aimed at identifying, for each failure mode, both the level of uncertainty associated with the phenomenon and the sensitivity of the natural circulation failure probability to that phenomenon. For example, even if a phenomenon is highly uncertain (because of deficiencies in the physical modelling) this may not be important for the overall failure probability. Conversely a phenomenon may be well understood (therefore the uncertainty is small) but the failure probability may be sensitive to small variation in this parameter. The grading scheme is as follows:

Table 1 Grade Rank for Uncertainty and Sensitivity

	Grade	Definition
Uncertainty	H	The phenomenon is not represented in the computer modelling or the model is too complex or inappropriate which indicates that the calculation results will have a high degree of uncertainty.
	M	The phenomenon is represented by simple modelling based on experimental observations or results.
	L	The phenomenon is modelled in a detailed way with adequate validation.
Sensitivity	H	The phenomenon is expected to have a significant impact on the system failure
	M	The phenomenon is expected to have a moderate impact on the system failure
	L	The phenomenon is expected to have only a small impact on the system failure

The results of a preliminary qualitative point of view analysis as outcome of expert judgement assessment can be summarised in the table below, regarding the failure modes of the natural circulation.

Table 2 Failure Modes related Uncertainty and Sensitivity

TOPIC	UNCERTAINTY	SENSITIVITY
Envelope failure	L	H
Cracking	L	L
Non-condensable gas	H	H
Thermal stratification	H	H
Surface characteristics modification (e.g. oxidation)	M	L

From the above qualitative table it seems that only for the structural failure modes there is a deep knowledge of the relative phenomenology. In the other cases the level of uncertainties is high or medium and in every case an effort must be devoted for their quantification. The sensitivity grades are the results of only a qualitative analysis: the natural circulation failure is very sensitive either to the loss of primary boundary or to the thermalhydraulic phenomena arising during the system performance.

It’s clear that the worst case is characterised by "high" rankings relative to either sensitivity or uncertainty (see e.g. non-condensable gas and thermal stratification), making the corresponding phenomena evaluation a critical challenge.

5. RELIABILITY ASSESSMENT

This section reports the reliability study with reference to the Isolation Condenser system, providing the system analysis and modelling and reliability assessment taking into account also the loss of natural circulation.

The analysis is not focused on one defined BWR advanced reactor, nevertheless the functions and the general requirements for the system and its arrangement are drawn from the available literature, although the general validity of the present study is not affected by the peculiar IC considered.

The approach to passive system assessment consists of two parts: the first one entails the classical reliability analysis of components, the second one concerns the passive function (i.e. natural circulation), which is evaluated through the reliability analysis of the components designed to assure the best conditions for passive safety function.

Accordingly to this approach, the unavailability of a passive system is the sum of two contributions:

- Failures of actuation devices, that is, failures of the components that must change state in order to start the passive operation. An example is the condensate return line valve of the Isolation Condenser.

- Failures that defeat or degrade the natural mechanisms which are the principles for the operation of the passive systems.

The first contribution is treated in the classical way; that is, as a failure of a component that must change its state or a failure of their supporting systems, such as the electrical power supply. The second contribution requires the identification of the mechanisms and the boundary conditions needed for starting and maintaining the intrinsic phenomena: the failure probability is evaluated “in the classical way” as an unavailability of components, or as the probability of occurrence of a failure mode which would violate the needed boundary conditions or mechanisms.

The major components of the system are (see Figure 1):

- heat exchanger (straight tube bundle),
- one main valve and a bypass valve in parallel with the main valve located on the drain line,
- piping.

The most critical components of the system are the motor operated valves on the condensate line which are required to actuate during transients, for instance upon high reactor pressure or low reactor water level.

In Table 3 the system component reliability data, derived from fission reactor PSA experience, are reported, while in Figure 2 the related fault tree is shown.

Table 3 IC Component Reliability Data

COMPONENT	FAILURE MODE	FAILURE RATE	REFERENCE
Valve	Fails to open	3.0E-3/d	IREP (Interim Reliability Evaluation Program)
Valve	Fails to remain open	1.0E-7/h	IREP
Valve	Fails to open CCF	3.0E-4/d	Eng judgement
Valve	Fails to remain open CCF	1.0E-8/h	Eng judgement
Heat Exchanger	Single pipe rupture	3.0E-10/h	IREP
Heat Exchanger	Multiple pipe rupture	3.0E-11/h	IREP
Heat Exchanger	Single pipe plugging	3.0E-10/h	IREP
Heat Exchanger	Multiple pipe plugging	3.0E-11/h	IREP
Piping	Rupture	2.4E-8/h (1.2E-9/hm*20m)	Eng judgement

h: hour
d: demand

As previously stated the quantification of the thermal-hydraulic unreliability is at present a challenging endeavour, mostly due to the large number of uncertainties, related to the failure modes related to the onset of thermal-hydraulic phenomena that would impair the passive function of the system, and the difficulties in their evaluation.

In order to overcome these obstacles, the present effort aimed at the natural circulation assessment is developed through an approach that implies the evaluation of components designed to assure the best conditions for passive function performance, thus simplifying the problem significantly. Therefore the natural circulation failure probability is assessed by the development of the fault tree reported in Figure 3, which includes three main failure modes, that is Loss of Heat Transfer, Presence of Non-Condensables and Loss of Primary Boundary, while the overall unavailability relative to the Isolation Condenser system will include both contributors.

The reliability values for the natural circulation failure are reported in Table 4.

Table 4 Reliability Data for Natural Circulation

COMPONENT	FAILURE MODE	FAILURE RATE	REFERENCE
Valve	Fails to operate	3.0E-3/d	IREP (Interim Reliability Evaluation Program)
Valve	Fails to open CCF	3.0E-4/d	Eng judgement
Heat Exchanger	Excessive pipe fouling	3.0E-11/h	IREP
Primary Boundary	Rupture	1.2E-9/hm*20m	Eng judgement

h: hour
d: demand

Loss of Heat Transfer addresses the failure of the IC heat transfer to an external source (IC pool water) which is assessed through two possible failures:

- insufficient water in the IC pool (makeup valve),
- degraded heat transfer conditions due to heat exchanger pipe excessive fouling.

The envelope failure (i.e. loss of primary boundary) failure mode is given the failure rate relative to piping rupture (1.2 E-9/hm) and has already been taken into account in the IC fault tree; excessive pipe fouling failure mode is assigned the failure rate relative to multiple pipe plugging for heat exchanger.

Finally the non-condensable build-up is modelled through the failures of the vent valves, whose actuation is required to enable the non-condensables venting towards the suppression pool.

As previously underlined the analysis is not peculiar of a definite plant and is perfumed on a rather simplified model of the system that implies the following assumptions and simplifications :

- the logic for the system control and for the component actuation is not accounted for,
- the electrical power supply system is not considered,
- the manual remote action by the operator from the control room is not taken into account,
- a main or a bypass valve located in parallel are required to initiate the Isolation Condenser operation,
- a Make-up Water System supplies water for replenishing the IC pool, the interface being a motor operated valve,
- two main located in series or two bypass vent valves are required to purge non-condensables from the lower header of the IC to the suppression pool.

The reliability assessment of IC system is performed at component level by means of RISK SPECTRUM code, a PC software package for system risk and reliability analysis based on fault tree technique (ref.5). The results obtained from both systems reliability evaluation are shown in Table 5: the numerical values are reported for illustrative purposes, being the analysis carried out on the basis of quite generic numerical values assigned to an oversimplified model of the system.

Table 5 Failure Probabilities for IC

System	Failure Probability	Contributor
Isolation Condenser	3.1E-4	8%
Natural Circulation	3.3E-3	92%
Isolation Condenser System	3.6E-3	

From the table one infers the natural circulation failure probability, which is evaluated with regard to the failure of specific system components, is the main contributor to the total system unavailability.

Moreover it has to be pointed out that the aforementioned degree of redundancy of the units (for example the IC foreseen for SBWR consists of three redundant units, each unit made of two identical modules which act as heat exchangers), leads to a reduction in failure probability values.

The MCS (Minimal Cut Set) analysis identifies the failure of heat transfer to the external source due to insufficient water in the IC pool (in the study this failure is represented by the basic event consisting of the makeup valve fault) as the most important contribution to the final reliability value of the system.
Other important contributions are given by the common mode failures relative to either the valves on the drain line or the vent valves for uncondensables purging.

Figure 2. Isolation Condenser fault tree

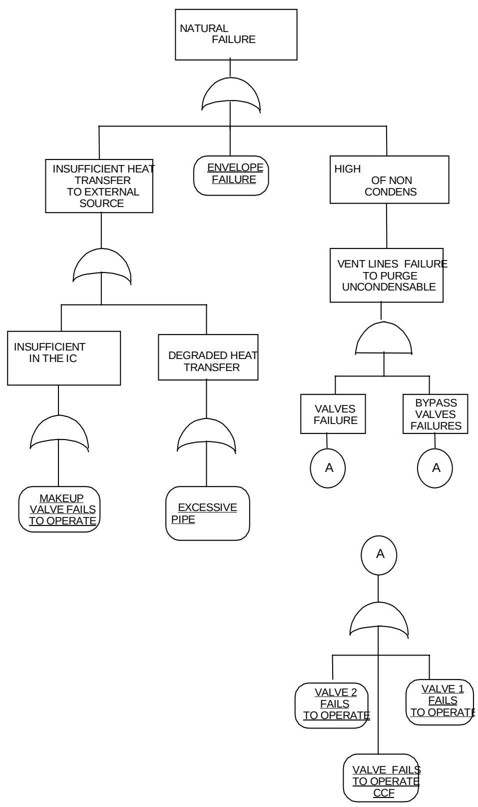
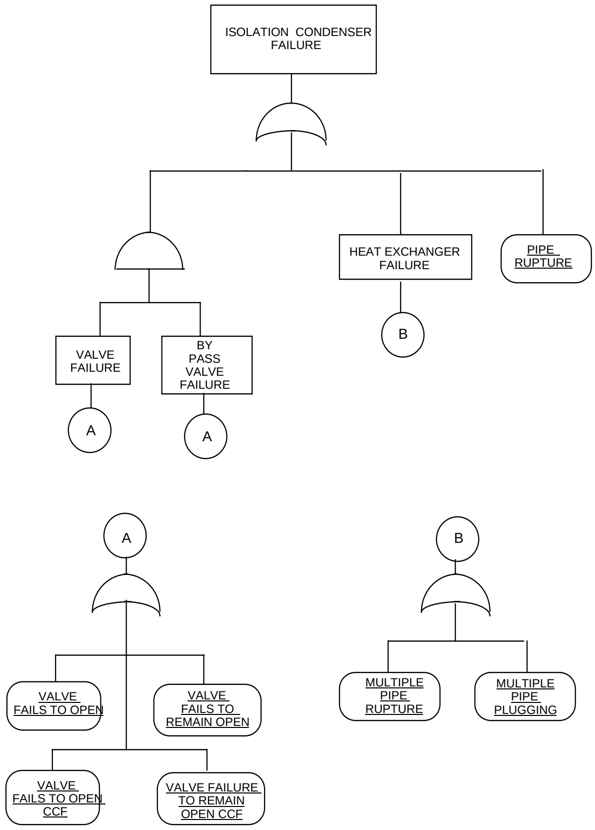


Figure 3. Natural Circulation fault tree



6. CONCLUSIONS

The systems analysis of a thermal-hydraulic passive system has been judged necessary in order to make out the issue in the frame of the methodology aimed at passive system reliability assessment and to add credibility and validate the methodology itself.

The qualitative analysis, by means of the FMEA approach, has enabled to identify the most relevant uncertainties related to passive system performance and provided as output a set of critical parameters (i.e. physical quantities) that characterise the system failure and thus to be considered in the relative assessment process (see RMPS project supported by the European Community).

Furthermore a preliminary appraisal of the system unavailability has been presented: failures that defeat or degrade the natural mechanism upon which the passive system relies have been treated as an unavailability of components which challenges the boundary conditions or mechanisms needed for assuring the passive operation.

Under the assumptions taken in the study and considering the "exploratory" character of the present effort, the results show the important weight of the natural circulation failure probability on the overall unavailability of the system.

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ASSESSMENT OF FLUID FLOW CHARACTERISTICS FOR FLUIDIC DEVICE IN APR 1400

S.-K. Lee, M. Kim, J.-H. Lee, J.-I. Lee
Korea Institute of Nuclear Safety
19 Goosong-dong, Yusong, Daejon, Korea

S.-Y. Yoo and M.-S. Gie
Chungnam National University
220 Goong-dong, Yusong, Daejon, Korea

ABSTRACT

New evolutionary light water reactors (LWRs) are being developed all over the world today. Some of those LWRs employ so-called passive safety components that use natural phenomena as the driving force such as expansion of pressurised gas, natural circulation, and so forth. These passive safety components and their systems are applied to improve the safety of nuclear power plants, further achieving not only to simplify the safety system, but also to improve its reliability and to provide a sufficient time margin to enable the operators to cope with accidents.

The APR 1400 (Advanced Pressurised Water Reactor), which is a Korean Next Generation Reactor, adopted fluidic device as one of its passive pieces of safety equipment. The fluidic device is installed at the bottom of the inner space of the Safety Injection Tank (SIT) to control the flow rate of emergency cooling water during a large break loss of coolant accident (LBLOCA). During the past two years, a scale model test to obtain the required flow characteristics of the device under the APR 1400 specific conditions has been performed in Korea. The performance verification test was conducted to obtain the optimum characteristics and design data of full fluidic device (FD).

In this study, the thermal-hydraulic characteristics for the fluidic device are analysed using SIMPLE algorithm with an aim to develop the assessment and verification guidelines for the APR 1400. To assess the flow characteristics for fluidic device, a three dimensional numerical model is developed and its results are compared with those of experiments.

I. INTRODUCTION

The fluidic device (FD) is a specific type of vortex valve. In general, the vortex valve consists of a flat, cylindrical vortex chamber with three ports. The main fluid is introduced into the chamber via a radial port and leaves the chamber at the centre through the axial port. The third port, known as the control port, is mounted tangentially to the chamber. In the vortex valve's fully open condition, the main fluid simply flows through the chamber and encounters very little resistance with no fluid flowing into the control port. As control flow through the control port is increased, the main fluid is deflected from its straight path and begins to form a vortex due to the circular shape of the chamber. This vortex flow has a higher resistance so that the flow is reduced. As more control fluid is added, the main flow is progressively reduced until a condition is reached where the pressure drop across the vortex field is high enough to present any fluid from entering the chamber at all. This is the vortex valve's fully closed position. Figure 1 shows the typical characteristics of the aforementioned operation in non-dimensional form [1].

Mitsubishi (Japan) has approached more closely to utilising this concept and combining it as vortex damper in the accumulator of their APWR (Advanced Pressurised Water Reactor) design. In this system, the supply port of the valve is directly connected to a standpipe inside the accumulator. In its standby state, the accumulator is full of water and the standpipe is submerged. When the water level in the accumulator, which is similar to SIT, reaches the level of the standpipe, the water level in the standpipe will start to fall rapidly [2].

The APR 1400 (Advanced Pressurised Water Reactor), which is a Korean Next Generation Reactor, adopted FD as one of its passive pieces of safety equipment. The FD is installed at the bottom of the inner space of the Safety Injection Tanks (SITs) to control the flow rate of coolant during large break loss of coolant accident (LBLOCA). There are four SITs are provided in the APR 1400. The discharging piping of each SIT is routed to a direct vessel injection (DVI) nozzle on the reactor vessel. The SITs automatically discharge the borated water into the reactor coolant system (RCS) when the RCS pressure drops below the SIT pressure as a result of a LOCA. A fixed internal device in the tank regulates the flow rate with changing level. The SITs are normally pressurised to a nominal operating pressure of 610 psig (42.9 kg/cm^2) for normal operation. The SITs contain borated water at a maximum concentration of 2.5 weight percent boric acid (4,400 ppm). The tank gas/water fraction, gas pressure and outlet pipe size are selected to allow three of the four tanks to flood and cover the core before significant core damage and before a significant zirconium-water reaction can occur following a LOCA. The volume of water in the tank is conservatively calculated assuming that all water injected prior to the end of the blow-down is lost. By using the FD, the APR 1400 can integrate both the functions above, and automatically the FD switches from high to low flow rates without the actuation of any active components throughout the transient [3,4].

The application of the FD in PWR safety injection systems is in the conceptual design stage. Under static conditions, a satisfactory performance of FD is feasible. However, the dynamics of switching from a high to a low flow rate under conditions which may be experienced in the SIT are not fully understood and cannot be satisfactorily modelled at present. The fundamental question which needs to be answered is whether the response of the device will be sufficiently fast in achieving a correct switching from the target high to the target low flow rate. In this regard, a development test program is required and is being implemented in Korea.

II. ASSESSMENT OF FLUID FLOW CHARACTERISTICS

1. Experimental Facilities

During the past two years, a scale model test to obtain the required flow characteristics of the device under the APR 1400 specific conditions has been performed using the experience and existing facility of AEA Technology (UK) with appropriate modifications by Korea Atomic Energy Research Institute (KAERI). The performance verification test was performed to obtain the optimum characteristics and design data of FD.

The test pressure vessel's internal diameter is 1.25 m while the full size pressure vessel's internal diameter is 2.74 m. The scale-down on area (and there the ideal flow scale-down) is 4.8. The SIT system design gas pressure is 570 Psig (39.3 bars). The operating pressure during the tests are varies from 38 to 43 bars since an operating parameter is about 42.1 bars.

The initial gas volume at full scale is 806 ft^3 ($22.8m^3$). During the tests. the target gas volume in the test pressure vessel becomes 166.7 ft^3 ($4.75\ m^3$). This corresponds to a height of water above the vessel base of 10.6 m (equivalent to about 11.1 m as the indicated level).

At full scale, the amount of water lost before the standpipe is uncovered is 800 ft^3. At model scale, this is equal to 4.7 m^3. This is equivalent to a level change of 3.8 m in the 1.25 m vessel. The standpipe should therefore be positioned 6.75 m above the vessel base. For most of the test conditions investigated to date, a figure of 7.3 m has been used as a reasonable approximation to the required level. The test loop is shown in Figure 1.

2. Analytical Assessment

To assess the characteristics of fluid flow in FD, the governing equations such as continuous equation, momentum equation and turbulent mode are used as following [5].
The continuous equation is given as Equation (1).

$$\frac{\partial U_i}{\partial x_k} = 0 \qquad (1)$$

The momentum equation is given as Equation (2).

$$U_k \frac{\partial U_i}{\partial x_k} = -\frac{1}{\rho}\frac{\partial P}{\partial x_i} + \frac{\partial}{\partial x_k}\left(\upsilon \frac{\partial U_i}{\partial x_k} - \overline{u_i u_k} \right) \qquad (2)$$

In this study, there are three turbulent models, a standard k-ε model, RNG (Re-normalisation Group) k-ε model, and Reynolds Stress turbulent model, are adopted to compare the sensitivity of the results respectively. The standard k-ε model is given as Equations (3) and (4).

$$\overline{U_k} \frac{\partial k}{\partial x_k} = \frac{\partial}{\partial x_k}\left(\frac{\nu_T}{\sigma_k}\frac{\partial k}{\partial x_k} \right) + \nu_T \left(\frac{\partial \overline{U_i}}{\partial x_k} + \frac{\partial \overline{U_k}}{\partial x_i} \right)\frac{\partial \overline{U_i}}{\partial x_k} - C_D \frac{k^3/2}{L} \qquad (3)$$

$$\overline{U_k} \frac{\partial \varepsilon}{\partial x_k} = \frac{\partial}{\partial x_k}\left(\frac{\nu_T}{\sigma_\varepsilon}\frac{\partial \varepsilon}{\partial x_k} \right) + C_1 \nu_T \frac{\varepsilon}{k}\left(\frac{\partial \overline{U_i}}{\partial x_k} + \frac{\partial \overline{U_k}}{\partial x_i} \right)\frac{\partial \overline{U_i}}{\partial x_k} - C_2 \frac{\varepsilon^3}{k} \qquad (4)$$

Where velocity of k components, $U_{k,}$ the turbulent kinetic energy, k, the kinetic energy dissipation rate, ε, and constant, C_1 and C_2 .

In addition, the two-equation models based on the standard k-ε model such as the RNG k-ε model and the Reynolds Stress Model (RSM) are also applied to assess.
The RNG k-ε model is given as Equations (5) and (6).

$$\overline{U_k} \frac{\partial k}{\partial x_k} = \nu_t S^2 - \varepsilon + \frac{\partial}{\partial x_k} \alpha \nu_t \frac{\partial k}{\partial x_k} \qquad (5)$$

$$\overline{U_k} \frac{\partial \varepsilon}{\partial x_k} = C_{i\varepsilon} \frac{\varepsilon}{k} \nu_T S^2 - C_{2\varepsilon} \frac{\varepsilon^3}{k} - R + \frac{\partial}{\partial x_k} \alpha \nu_T \frac{\partial \varepsilon}{\partial x_k} \qquad (6)$$

The Reynolds Stress Model is given as Equations (7) and (8).

$$\frac{\partial(\rho U_k \overline{u_i u_j})}{\partial x_k} = \rho (P_{ij} - \varepsilon_{ij} + \phi_{ji} + d_{ijk}) \qquad (7)$$

$$\frac{\partial(U_k \varepsilon)}{\partial x_k} = C_\varepsilon \frac{\partial}{\partial x_k}\left(\frac{x}{\varepsilon}\overline{u_i u_j}\frac{\partial \varepsilon}{\partial x_i} + \frac{1}{2} C_{\varepsilon,1}\frac{x}{\varepsilon} P_{ij} - C_{\varepsilon,2}\frac{x^2}{\varepsilon}\right) \qquad (8)$$

Where

$$P_{ij} = -\overline{u_i u_k}\frac{\partial \overline{U_j}}{\partial x_k} - \overline{u_j u_k}\frac{\partial \overline{U_i}}{\partial x_k} \qquad (9)$$

and velocity of j components, $U_{j,}$ and the pressure-strain covariance tensor, ϕ_{ij}, the turbulent diffusion tensor, d_{ij}

In this study, the fluid flow inside the FD is so complicated that the RNG k-e model is mainly applied and the results is compared with both those of the standard k-ε model and the Reynolds Stress model.

The governing equations and turbulent equations could be generalised as following Equation (10);

$$\frac{\partial}{\partial x_k}(\rho U_k \phi) = \frac{\partial}{\partial x_k}(\Gamma_\phi \frac{\partial \phi}{\partial x_k}) + S_\phi \qquad (10)$$

Where, the left-hand side is convection term and the first description in the right-hand side is diffusion term and the second is the source term. As for ϕ =1, the equation means the continuous equation and the momentum equation for $\phi = U_i$, the turbulent equations for ϕ = k and ε. Using the finite volume method, the equation (6) could be generalised into the differential equation as shown Equation (11).

$$\phi_p \sum_i (A_i - S_p) = \sum_i (A_i \phi_i) + S_c \qquad (11)$$

In this study, the numerical calculation is performed using the power-law scheme, three kinds of turbulent models such as the standard k-ε model, the RNG k-ε model, Reynolds Stress Model with SIMPLE algorithm [6].

III. RESULTS AND DISCUSSION

The schematic diagram for fluid flow in the FD is shown in Figure 2. The level water, which is operating fluid, is above standpipe, the water flows into the FD through the supply port (A) and control port (b) connected to the standpipe and flows out through the discharge port. At this time, the fluid flow becomes to be maximised since the water flows into through both the supply port and control port as shown in Figure 3. The pressure drop in the FD is dependent with both the tangential and radial velocities of the incoming fluid flow into the FD. In other word, the pressure drop is depended in the amount of the fluid swirling flow. As the water level of the SIT is below the top of the standpipe, the water flows into FD through the control port only. Therefore, the discharge flow becomes to be minimised because of the high swirling flow. The grid is generated for the prediction of both the velocity and the pressure in FD as shown in Figure 4. As the supply port and control port are symmetric along 90 degrees in the cylindrical structure of FD, the computational grid is modelled as a quarter of full scale and the cyclic boundary condition is applied in the symmetric faces.

1. Sensitivity Study

A. Effects on Grid Generation

To investigate the effect of grid generation in the FD, three cases are analysed for the sensitivity study such as 91,360, 165,529, and 209,266 cells respectively. As the grid is generated finely, the convergence for computation goes stable as shown Table 1. However, there is some deviation due to the re-circulation in the vicinity of the discharge port as the flow rate is increased. Therefore, the discharge port is extended with an aim to avoid the re-circulation flow in the exit and the grid is generated as a number of 165,529. Using the RNG k-ε turbulent model, the discharge flow rate is 241.89 m^3/hr and is well agreed with experimental flow rate of 227.35 m^3/hr within 6.4% deviation.

B. Effects on Turbulent Model

As for the effect of turbulent model, three turbulent models are adopted such as the standard k-ε model, the RNG k-ε model, and the RSM model. The results of the sensitivity study is as shown in Table 2. While the standard k-ε model and the RNG k-ε model enable to well converge during computing, the computation using the RSM model is diverged in the cases of both high discharge flow rate and low discharge flow rate. Using the RNG k-ε model, the predicted flow rate is well agreed with that of experiment rather than the standard k-ε model for the high flow rate. As for the RNG k-ε model and the standard k-ε model, there is a deviation between the numerical calculation and experiment for low flow rate.

2. Fluid Flow

In the experimental results, after initiating the safety injection (SI), the discharge flowrate goes to be maximum between about 82 sec and 100 sec. The discharge flowrate decreases after 111 sec and reaches the minimum flowrate until about 400 sec.

Figure 5 shows the discharge flowrate with comparing the experimental results at 80, 100, 115, and 260 sec, respectively after initiating the SI. At about 80 sec. the discharge flowrate is 241.89 m^3/hr after SI injection and the deviation is 6.4% between prediction and experimental results. At 100 sec. The discharge flowrate is 213.70 m^3/hr in prediction and 240.6955m^3/hr in experiment. The deviation is about 11 %.

The velocity vector in fluidic device is shown in Figures 6 and 7. At 80 sec, the flowrate is 189.78 m^3/hr at control port and 85.72 m^3/hr at supply port respectively. The maximum flowrate is supplied at 100 sec. At that time, the flowrate is 74.96 m^3/hr at control port and 163.96 m^3/hr at supply port. At 115 sec, the flowrate is 157.17 m^3/hr at control port and it increases up to 135.38 m^3/hr at 260 sec.

The fluid flow supplied through supply port and control port mixes at the entrance of fluidic device and smoothly flows into the discharge port when the maximum flowrate reached. However, a few of the secondary flow occurred in the vicinity of outside fluidic device and the fluid flow increases since the swirl flow occurred in centre of axis at discharge port.

3. Pressure

The Figures 8 show the pressure profiles in the FD. As the discharge flow rate is maximised, the pressure profile is various since the fluid flow is complicated in the FD. In particular, the pressure is low in the region of the centreline and discharge at the discharge port due to the formation of the vortex flow, which seems to form the secondary flow. As the discharge flow rate goes to be minimised, the pressure profile in the vicinity of the centreline of the FD is low rather than other region due to the effect of the vortex flow formed inside the FD. This pressure pattern is, in general, less complicated rather than that of the maximum flow rate.

IV. CONCLUSION

The fluid flow characteristics of the FD, which is passive flow control device in the SIT of APR-1400 are assessed using SIMPLE algorithm using FLUENT code [7] and are compared with the experimental results. In this study, the fluid flow rate and pressure in the FD are predicted on the basis of the boundary condition at the entrance of supply port, control port, and discharge port respectively, so as to conclude the followings:

1. Comparing the predicted discharge flow rate at the discharge port with that of experimental results, it is well agreed in such a way to apply the RNG k-ε turbulent

model. However, there is some deviation between the predicted results and experimental ones as the water level goes below the stand pipe in the SIT. This implies that the fluid flow could not be the fully developed vortex flow in the FD due to the formation of the secondary fluid flow. In this regard, there is a need to further study in both the analytical and the experimental results.

2. The pressure is low in the vicinity of centreline along the supply port due to the vortex fluid flow. In the meanwhile, there is very low pressure in the centreline of discharge port due to the secondary fluid flow. The pressure profile along the axial region at the minimum flow rate is, in general, lower than that at the maximum flow rate.

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Table 1 Results of Sensitivity Study on Grid

Case	Cell Number	Discharge Flowrate (m^3/hr)	Continuity	X-vel	Y-vel	Z-vel	k	
case 1	91,360	188.26	3.76e-3	3.81e-5	4.26e-5	4.86e-5	2.42e-5	4.9e-5
case 2	209,266	170.69	8.15e-4	4.19e-6	2.51e-6	1.23e-5	1.01e-5	1.63e-5
case 3	165,529	241.89	3.67e-4	2.52e-5	3.35e-5	2.53e-5	1.49e-5	3.53e-5

Table 2 Results of Sensitivity Study on Turbulent Model

Turbulent Model	Discharge Port Flowrate (maximum flowrate, 80sec)			Discharge Port Flowrate (minimum flowrate, 115sec)		
	Experiment (m^3/hr)	Computation (m^3/hr)	Error (%)	Experiment (m^3/hr)	Computation (m^3/hr)	Error (%)
Standard k-ε	227.35	291.48	28.20	77.74	155.13	99.54
RNG k-ε		241.89	6.40		141.78	82.38
RSM		Diverse	-		Diverse	-

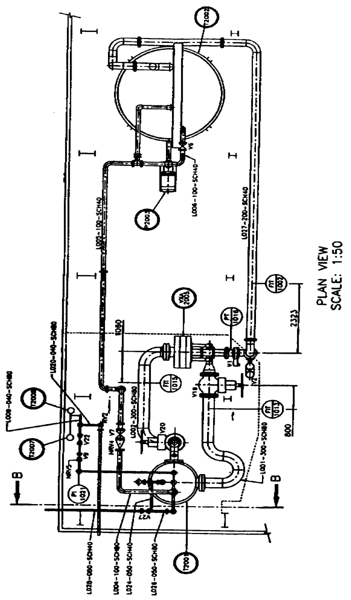


Figure 1 SIT Valve Test Rig Layout

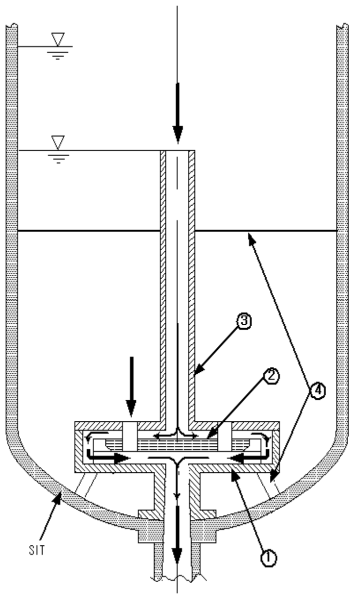
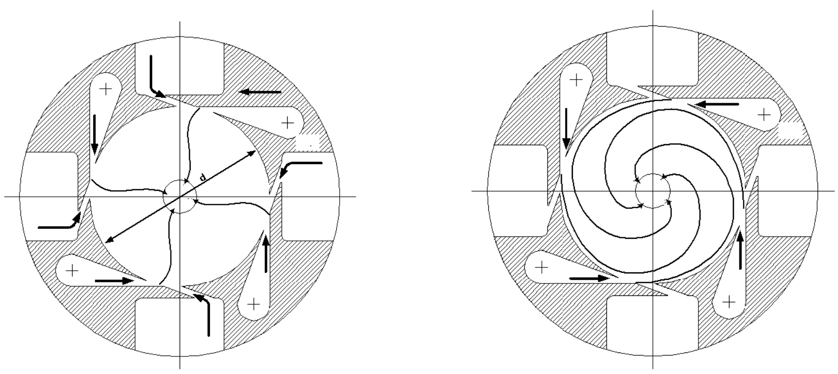
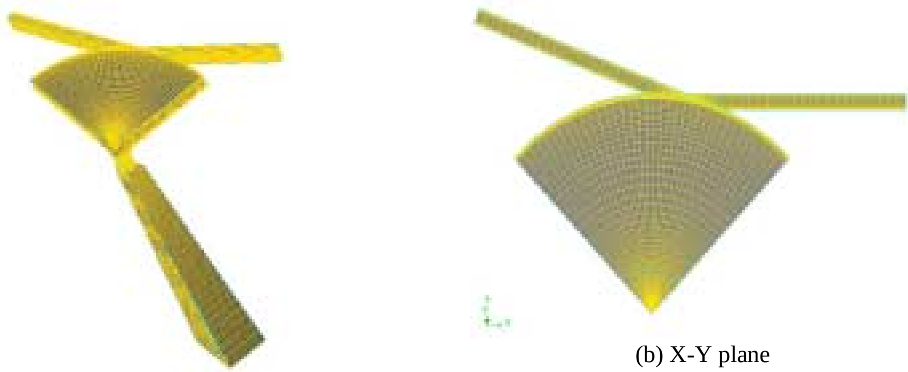


Figure 2 Schematic Diagram of SIT



(a) Maximum Flowrate (b) Minimum Flowrate

Figure 3 Schematic Diagram of Fluid Flow in Fluidic Device



(b) X-Y plane

(a) 1/4 Section

Figure 4 Grid Generation in Fluidic Device

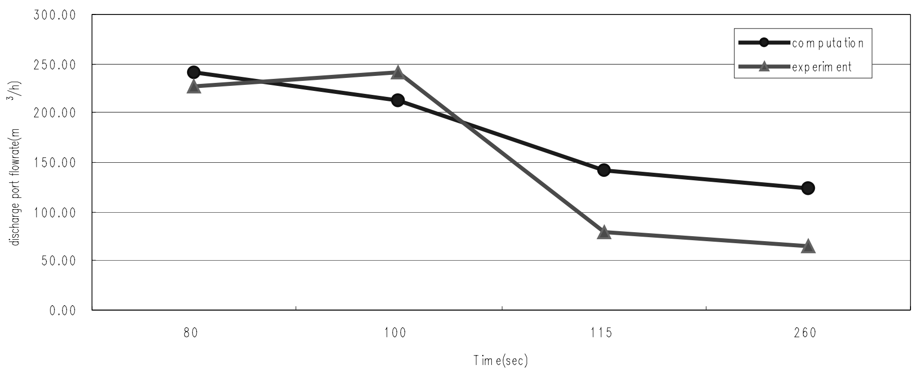
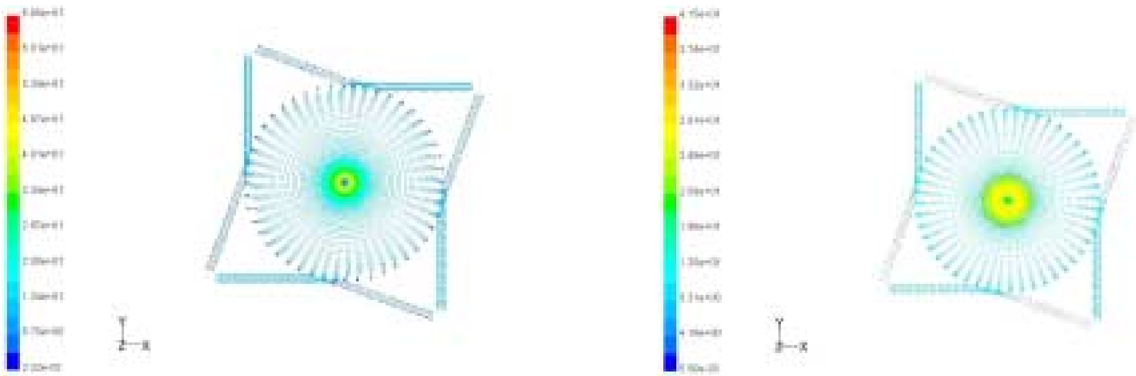


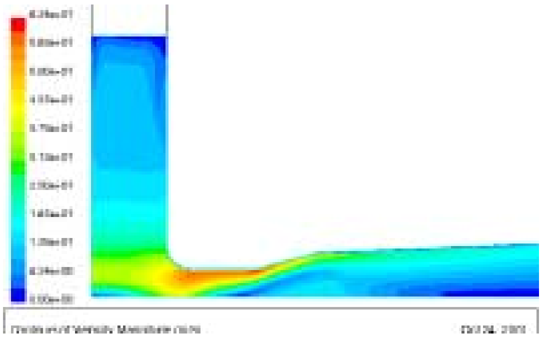
Figure 5 Comparison for Discharge Flowrate

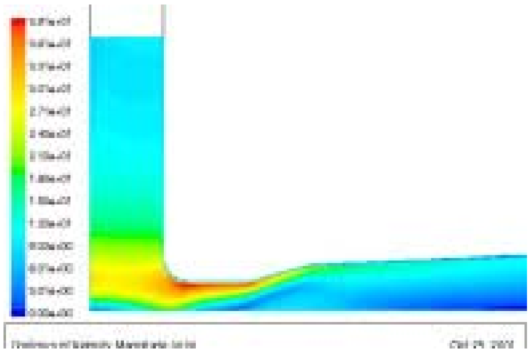


(a) 80 sec

(b) 115 sec

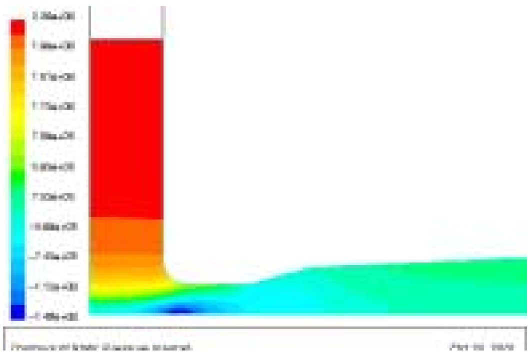
Figure 6 Velocity Profile in Fluidic Device

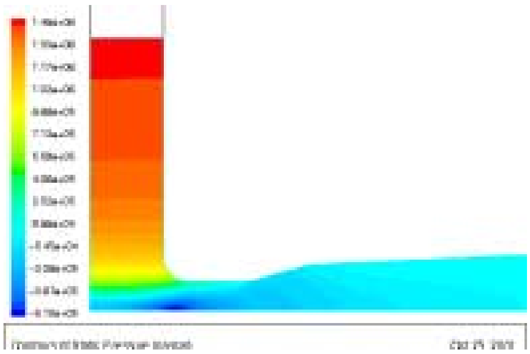




(a) 80 sec (b) 115 sec

Figure 7 Velocity Profile in Fluidic Device





(a) 80 sec (b) 115 sec

Figure 8 Pressure Profile in Fluidic Device

DIRECTION DE L'ENERGIE NUCLEAIRE
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PASSIVITY IN GAS COOLED (FAST) REACTORS

- **The structure of safety analysis**
  - **Defence in Depth**
  - **Classification of accidents**
  - **Farmer's curve and Lines of Defence method**
- **An example of passive system in gas cooled reactor:**
- **heat removal in GT-MHR**
- **A passive heat removal for gas cooled fast reactor ?**
- **Conclusion**

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Safety requirements for future nuclear systems
Defence in Depth

- **1st level : Prevention** : Quality in design and achievement, prevention of nonconformity
- **2nd level: Surveillance, detection and control** : Quality of operation, keeping the facility within authorized limits
- **3rd level: Safety systems and Protection systems design** : Postulate of all plausible incidents and accidents, and implementation of means to limit the effects of these accidents to acceptable levels
- **4th level: Accident management and containment protection, limitation of consequences** : Prevention of deterioration of accidental conditions and limitation of severe accident consequences
- **5th level: Response outside the site** : Limitation of radiological consequences for populations in case of significant releases

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The classification of accidents

- Any plausible plant condition is a coupling of a postulated initiating event (PIE) with an initial plant state :
  - deterministic in the list of PIE (exhaustive?), and in the analysis of the scenario
  - probabilistic in the frequency of PIE, and in the evaluation of the unavailability of the Line of Defence (LOD)

    ➥ **quantify the unavailability of a passive method ?**
- The consequences of an accident are evaluated in terms of :
  - temperature limits: *example of 1600°C for TRISO particles*
  - mechanical limits : level A, B, C, D for stresses
  - and in fine, in terms of radiological doses for workers and public
- ➔ Farmer's curve

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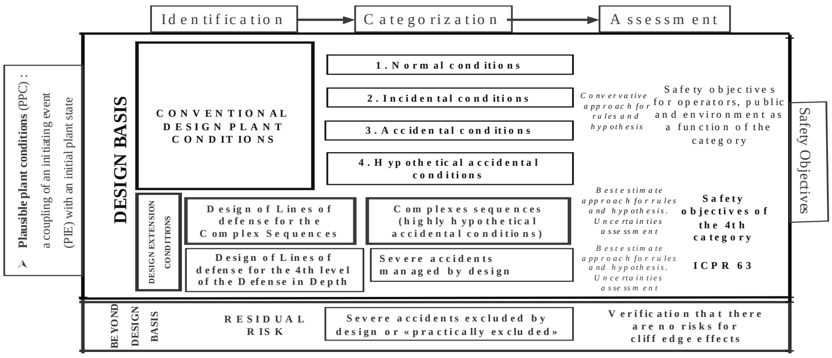
General approach for the safety related design and assessment applicable to all the future nuclear installations



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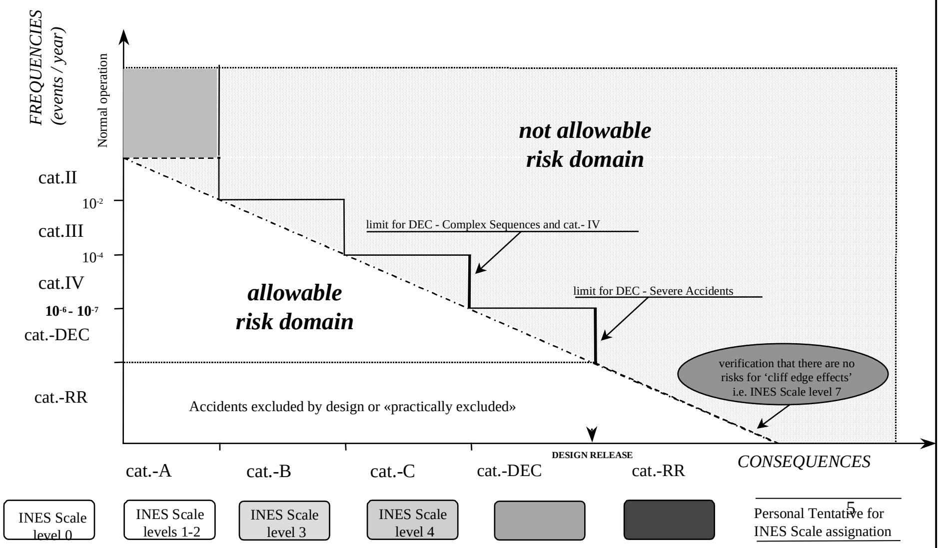
Quantification : Farmer's diagram



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The notion of Line of Defence (LOD)

This term is used for :

- Any **inherent characteristic, equipment, system**, etc., implemented into the safety related plant architecture.
- Any **procedure** foreseen coherently with the General Rules for Plant Operation (e.g. human actions : preventive, protective, etc.), the objective of which is to accomplish a given safety function.
- Two types of LOD are considered :
  - the **strong lines** (called « **a** »), whose probability of failure is of the order of (10^{-3} - 10^{-4}) / year, or the unavailability (10^{-3} - 10^{-4}) / demand;
  - the **medium lines** (called « **b** »), whose probability of failure is of the order of (10^{-1} - 10^{-2}) / year, or the unavailability (10^{-1} - 10^{-2}) / demand.

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The Lines of Defence method

Frequency category of the sequence outcome (resulting operating situation)	Frequency category of the situation			
	I - II	*III*	*IV*	*DEC*
*cat_{freq} – II (10^{-2} < prob. of occurrence / reactor*year < 1)*	**b**			
cat_{freq} – III (10^{-4} < p < 10^{-2})	**b**	**b**		
cat_{freq} – IV (10^{-6} < p < 10^{-4})	**a**	**a**	**a**	
cat_{freq} – Design Extension Conditions	**b**	**b**	**b**	**b**
Total LODs to be implemented to prevent severe accident	**2a+b**	**2a**	**a+b**	**b**
Total LODs to practically exclude a given sequence (i.e. to reject the sequence into the Residual Risk)	**> 2a+b**	**> 2a**	**> a+b**	**> b**

Until now, no significant difference between active and passive system: the rules to be followed are the same

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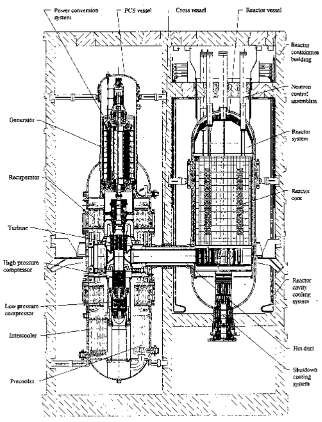
An example of passive system in gas cooled reactor : ultimate heat removal in GT-MHR

- Power Conversion System:
  turbomachine → active

- Shutdown Cooling System:
  circulator + heat exchanger → active

- **Reactor Cavity Cooling System**:
  conduction + (convection) + thermal radiation
  9 fully **passive** 8



GT-MHR module

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Analysis of the transients

- For any plausible plant condition where RCCS is involved, this system seems sufficient to remove heat: the temperature limits for fuel and steel components, corresponding to the category, are respected.
- Even for an hypothetical accident, such a large LOCA without scram. **But** (MHTGR) for some events in intermediate frequencies (category 3), level D conditions could be reached on the vessel ? "deeper analysis required.
- Unavailability of RCCS ?
- Analysis shall take into account:
  - a decrease in core's graphite conductivity, due for example to a fissuration of blocks after a seism
  - a rupture of core support plate, where the core is displaced in front of the RCCS
  - a change of emissivity in the pit, due for example to steam, or to carbone dust

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Passive heat removal for gas cooled **fast** reactor ?

- Parameters involved:
  - power density ∀ "< 10MW / m^3 → unfavourable for fast reactor
  - annular core, to distribute residual power → unfavourable for fast neutron balance
  - heat capacity of the core "difficult to find an equivalent of graphite, transparent to neutrons.
- A preliminary analysis shows that a **passive** system (natural convection and heat exchangers) could be sufficient, but for some accidents (**small** LOCA) and if an equilibrium pressure about 10 bar is possible.
- Alternative solution: passive/active system (injection of gas), waiting for a passive one after sufficient decrease of residual power. *But if active system fails ?*

  → **deep understanding of reliability of active and passive systems**

  → **coherent safety structure, inside defence in depth**

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Conclusion

- For any safety analysis, a method shall be defined at first:
  - →Defence in depth
  - →Lines of defence method
- It requires – among others – an evaluation of the unavailability of safety systems, whether active or passive.
- The conception of a Power Plant is a whole, and it is not obvious to adapt valuable safety systems to any change in the design of the reactor.

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THE REPAS APPROACH TO THE EVALUATION OF PASSIVE SAFETY SYSTEMS RELIABILITY

F.Bianchi[*], L.Burgazzi[*], F.D'Auria[§], M.E.Ricotti[°]

[*]ENEA - Centro Ricerche "Ezio Clementel"
via Martiri di Monte Sole 4, I-40129 Bologna, Italy

[§]University of Pisa - DIMNP
via Diotisalvi 2, I-56126 Pisa, Italy

[°]Polytechnic of Milano – Dept. of Nuclear Engineering
via Ponzio 34/3, I-20133 Milano, Italy

Abstract

Scope of this research, carried out by ENEA in collaboration with University of Pisa and Polytechnic of Milano since 1999, is the identification of a methodology allowing the evaluation of the reliability of passive systems as a whole, in a more physical and phenomenal way. The paper describe the study, named REPAS (Reliability Evaluation of Passive Safety systems), carried out by the partners and finalised to the development and validation of such a procedure.

The strategy of engagement moves from the consideration that a passive system should be theoretically more reliable than an active one. In fact it does not need any external input or energy to operate and it relies only upon natural physical laws (e.g. gravity, natural circulation, internally stored energy, etc.) and/or "intelligent" use of the energy inherently available in the system (e.g. chemical reaction, decay heat, etc.). Nevertheless the passive system may fail its mission not only as a consequence of classical mechanical failure of components, but also for deviation from the expected behaviour, due to physical phenomena mainly related to thermalhydraulics or due to different boundary and initial conditions.

The main sources of physical failure are identified and a probability of occurrence is assigned. The reliability analysis is performed on a passive system which operates in two-phase, natural circulation. The selected system is a loop including a heat source and a heat sink where the condensation occurs. The system behaviour under different configurations has been simulated via best-estimate code (Relap5 mod3.2). The results are shown and can be treated in such a way to give qualitative and quantitative information on the system reliability. Main routes of development of the methodology are also depicted.

Introduction

Since at least a decade, a lot of efforts have been devoted to the design and the evaluation of effectiveness of Passive Safety Systems, within the design strategy for both advanced and innovative reactors. However, main activities have dealt with thermalhydraulic and mechanical aspects of the design. Moreover, as far as the reliability related to the passive safety systems is concerned, the largest amount of work has been employed for the development of methods and tools for the sensitivity and uncertainty analysis, mainly on the models and the codes adopted for the safety assessment.

A wide bibliographic survey has been conducted on six major journals and tens of international conference proceedings, taking into account publications in a ten years period (1990- 2000). The results show that the

State Of Art on methodologies and applications to the reliability of the passive safety systems refers to a not yet consolidated area, relying more on single aspects of the argument than on a global and complete evaluation of the problem. It appears that no systematic studies has been carried out on the passive system reliability and the application of probabilistic methods for the passive system safety assessment, except for the standard ones and in a standard way. This is due mainly to the complexity and wideness of the item.

Notwithstanding a lot of publications were found on the uncertainty of thermalhydraulic codes[1-7] and on the analysis of passive safety systems from the deterministic, phenomenal or probabilistic point of view[8-12], few of them make attempt to face the reliability analysis in a global sight[13-18].

The publications reported in the bibliography are within those considered and found on the subject, containing also a detailed list of references.

Scope of this research, carried out by ENEA in collaboration with University of Pisa and Polytechnic of Milan since 1999, is the preliminary identification of a methodology allowing to the evaluation of the reliability of passive systems as a whole. The paper describe the study, named REPAS (Reliability Evaluation of Passive Safety systems), carried out by the partners and finalised to the development and validation of such a procedure.

Methodology Overview

As a starting step, a nomenclature for the terms adopted in the methodology should be defined. This task is mandatory in order to clarify the procedure and to allow a better comprehension of it. As far as it is possible, terms coming from PRA/PSA should be adopted, or at least a certain parallelism in meaning should be pursued. The following Table 1 resume both the terms and the meanings proposed and should be considered as a first attempt in pursuing this important task.

For evaluating the failure probability of passive systems, the developed methodology moves from the classical methods used for Probabilistic Risk Analysis and considers, in addition to real components (valves, pumps, instrumentation, etc), *virtual components*, that represent the natural mechanism upon which the system operation is based (natural circulation, gravity, internal stored energy, etc.). Therefore the reliability of passive systems may be achieved evaluating the failure probability of all the components (real and *virtual*).

The contribution of real components can be easily assessed by resorting to the reliability databases available, based on fission reactor experience, whereas for evaluating the virtual component contribution it is necessary to develop a procedure which allows to reach this purpose because of the lack of failure data.
The procedure proposed by ENEA, University of Pisa and Polytechnic of Milano foresees several steps, which are depicted in Fig.1 and are briefly listed below, according to nomenclature reported in Table 1:

1. <u>Characterisation of design/operational status for the system.</u> In this task the analyst should point out the mission of the Passive System and the relevant phenomenology involved (e.g. natural circulation, condensation), identify the main links with the Complete System (e.g. the primary system of a reactor) and select those parameters that duly define the functioning and behaviour of the Passive System. These System Parameters are those closely related to the mission of the system. They have been subdivided in *design* and *critical* parameters. The parameters identified as *design* are mainly related to the correct functioning of the system (e.g. pressures, levels). The *critical* ones are physical quantities that may affect the mission of the passive system (e.g. presence of non-condensables). The identified parameters usually refer to boundary and initial conditions for the system. The full characterisation of a thermal-hydraulic system may need a very large, hence unmanageable, number of such quantities.

Therefore, a bounded number of parameters should be selected. Expert Judgement procedures could be adopted in order to identify and select the System Parameters.

2. Definition of failure criteria for the system performance. The knowledge of the system mission allows the evaluation of the failure criteria. The most suitable method to perform this second task appears to be the FMEA-FMECA approach, that has to be modified and tailored on the specific argument by introducing the typical phenomenology of the passive system (i.e. natural circulation flow impairing, heat transfer degradation, etc.).The failure criteria can be established as single-value targets (e.g. the system must deliver a specific quantity of liquid within a fixed time) or as a function of time targets or integral values over a mission time (e.g. the system must reject at least a mean value of thermal power all along the system intervention). An example of this second option is depicted in Fig.2: taken the rejected thermal power W as the failure index, we observe the quantity for a time τ_{obs} (e.g. the mission time for the passive system) and define a failure threshold W_{limit} for the thermal power. Hence we consider the passive system in failure state when and for the time period ($\tau_{failure}$) during which the rejected power falls under the limit value.

3. Detailed code Modeling. Due to the lack of suitable experimental databases for passive systems in operation, the characterisation should rely on numerical Modeling, e.g. by means of simulation via best-estimate codes. The system analysis should be carried out with a qualified thermal-hydraulic system code and performing best estimate calculations.

4. Deterministic evaluation ("*nominal cases*"). Once the detailed model is defined, a set of operational and accident scenarios of interest for the passive system are to be identified. The system configuration is defined by adopting *nominal* values for both the design and critical parameters. The main output of the task is the calculation of the time evolution for those parameters under observation, as selected in the failure criteria. They represent the nominal behaviour of the passive system, taken as reference when compared with the other transients coming from the probabilistic configuration set (see step *vi*).

5. Assign Probability Distributions to Design and Critical Parameters. This task is a key step in the procedure: at this level in the path, range of values for the System Parameters must be defined, and Probability Distributions for the occurrence of that values are assigned. This task could be performed via Expert Judgement process, possibly taking into account available data on operation, maintenance, management and operation of passive safety and related systems.

Table 1
Nomenclature for the REPAS methodology

Term	**Meaning (within the methodology)**
Complete (or Complex) System	The whole Plant or System considered, for which the Passive System has been designed; the reliability analysis of the Complete (or Complex) System will be carried out taking into account the characterised Passive System.
Passive System	The specific System, based on passive features (e.g. natural circulation) which is the object of the Characterisation analysis.
Passive System Characterisation	The Passive System analysis (i.e. dynamic simulation), which gives information about the System behaviour, in operation and accident conditions, possibly occurring during the life of the Complete (or Complex) System; the goal is the identification of failure zones/conditions, if they exist.
Virtual Component	Characteristic phenomenology for the Passive System, that are unidentifiable as a classic component in the Fault-Tree analysis.
System Parameters	Typical Parameters for the Passive System, which identify or influence the system behaviour and its functions. They are classified as: Design Parameters Critical Parameters
Design Parameters	Those Parameters coming from the connection between the Passive System and the Complete (or Complex System), into which the Passive System is inserted and by which the Passive System is affected.
Critical Parameters	Passive System Parameters which identify the Passive System behaviour, taken as indicators for the system failure causes or joint causes. Identified by considering the proper missions the Passive System must fulfil.
System Mission	Goal(s) for which the Passive System has been designed and located within the Complete (or Complex) System.
Failure Criteria	Logical and/or numerical relationships which define the failure condition for the Passive System.

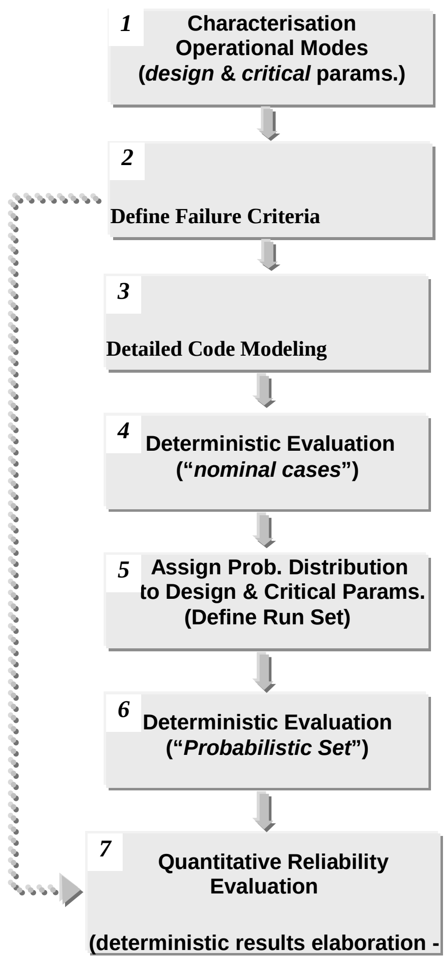


Fig. 1 – REPAS methodology roadmap.

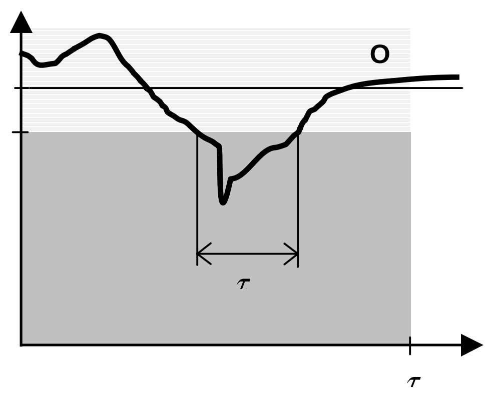


Fig. 2 – Example for a Failure criterion.

Once the ranges and associated probabilities are fixed, a stochastic selection of a limited number of system configurations is performed (e.g. via a Monte Carlo procedure). This group represents a set of possible configurations (the *Probabilistic Set*) for the Passive and Complete Systems, hence represents the input for the accident and operational transients to be analysed with best estimate code.

6. Deterministic evaluation of the *Probabilistic Set*. A set of accident and operational transients is performed with best-estimate codes, being the System Parameters (i.e. the initial and boundary conditions) identified by the Probabilistic Set. The stochastic selection of the configuration set leads to obtain a corresponding output set which is representative of the general, physical behaviour of the Passive System.

7. Quantitative Reliability Evaluation. The output from the code runs performed at the previous level must be quantitatively analysed by means of the failure criteria defined in *ii.* and can be qualitatively compared with the *nominal* behaviour obtained in *iv*. A set of *system performance indicators* could be defined for a qualitatively evaluation of the passive system capability to fulfil its mission (e.g. see Fig. 10). A classical risk curve as in Farmer (Fig. 3) could also be adopted. From the quantitative point of view, a Cumulative Distribution Function obtained from the run set seems to be the most appropriate (see Fig. 11).

It must be pointed out that the development activity is still in progress and improvements of the procedure are envisaged. Among them, for instance the identification the most important parameters with respect to system unreliability, or the identification of the top failure criteria, e.g. via Analytical Hierarchy approach (AHP).

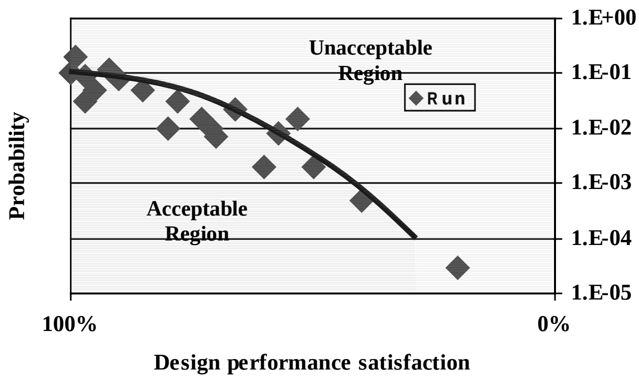


Fig. 3 – Example of qualitative evaluation for the Passive System reliability – acceptability criterion as in “Farmer Curve”

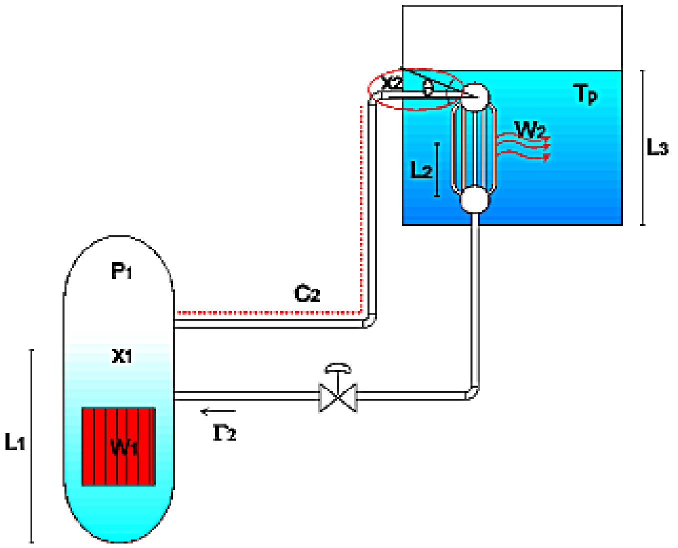


Fig. 4 – Passive System selected for the investigation.

APPLICATION OF THE PROCEDURE

The methodological procedure described in the previous section has been applied to a test case, following the roadmap as in Fig.1.

A Passive System to be investigated has been selected. Then the corresponding design and critical parameters have been identified, the probability distribution functions have been assigned and the failure criteria defined. Once the probabilistic selection of the system configurations has been performed and the code runs carried out, the application of the failure criteria led to a preliminary reliability evaluation of the passive system.

Passive System Test Case

A passive system having a configuration relevant to the technology of currently advanced Light Water Nuclear Reactors has been selected for the application of the procedure. The system is a typical natural circulation, two-phase flow loop. It consists of a heat exchanger immersed in a pool and linked to the pressure vessel by means of piping.

A gate valve, energised by a dedicated supply, activates the passive system. Thus the system belongs to the class C of passivity according to the IAEA[19] suggested classification. The pool and the heat exchanger are at a higher elevation than the power source. The selected configuration and the operating conditions are typical of the Isolation Condenser (e.g. as in most of the current advanced or innovative, passive BWR designs), whereas the operation of this system has been supposed different. A concept scheme of the system is given in Fig. 4. The key mission of the system is to reject the core decay heat to the heat sink by condensing primary fluid into the heat exchanger tube bundle.

Design and Critical Parameters and Probability of Occurrence

At the end of a series of discussion meetings on the relevant phenomenology involved in the passive system behaviour, both the design and critical parameters, the corresponding ranges and the probability distribution functions have been assigned. The main parameters identified as *design* refer all to initial conditions for the system, namely the pressure in the Reactor Pressure Vessel P_1, the liquid level in the Vessel L_1 and in the pool L_3 and the initial water temperature in the pool T_p. They are reported in Table 2, together with the proposed values. The nominal values, the ranges of variation and the selected initial values for the analysis are listed. The probability values for each initial status are reported in italics.
The initial values (last set of column data in Table 2) are based on expert judgement. The definition of these values is necessary in order to associate a probability value of occurrence for each system configuration and thus for each system performance evaluation, that could lead to a system failure.

Table 2
Design Parameters of the passive system.

	Design Parameter	**Unit**	**Nominal Value**	**Range**	**Discrete Initial Values & Probabilities**					
P_1	RPV pressure	MPa	7	0.2-9	0.2 *0.05*	1 *0.1*	3 *0.15*	7 *0.5*	9 *0.2*	value *pdf*
L_1	RPV collapsed level	m	8.7	5-12	5 *0.05*	7 *0.1*	8.7 *0.5*	10 *0.2*	12 *0.15*	
L_3	POOL level	m	4.3	2-5	2 *0.1*		4.3 *0.8*		5 *0.1*	
$T_p(0)$	POOL initial temperature	K	303	280-368	280 *0.1*		303 *0.8*		368 *0.1*	
-	System geometry: layout	-	-	Not assigned			- 1.0			

Table 3
Critical Parameters of the passive system.

	Critical Parameter	**Discrete Values & Probabilities**							
x_1	RPV non-condensable fraction	0. *0.719*	0.01 *0.12*	0.1 *0.07*	0.2 *0.05*	0.5 *0.03*	0.8 *0.01*	1. *0.001*	value *pdf*
x_2	Non-condensable fraction at the Inlet of IC piping	0. *0.71*	0.01 *0.12*	0.1 *0.07*	0.2 *0.05*	0.5 *0.03*	0.8 *0.01*	1. *0.01*	
θ	Inclination of the IC piping on the suction (deg)	0. *0.5*		1. *0.4*		5. *0.08*		10. *0.02*	
C_2	Heat Losses piping – IC Suction (kW)	0. *0.10*		5. *0.7999*		20. *0.10*		100. *0.0001*	
$L_2(0)$	Initial condition liquid level - IC tubes, inner side (%)		0. *0.1*		50. *0.1*		100. *0.8*		
UL	Undetected leakage (m^2)	0. *0.8899*		1.E-5 *0.1*		5.E-5 *0.01*		10.E-5 *0.0001*	
POV	Partially opened valve in the IC discharge line (%)	1. *0.001*		10. *0.01*		50. *0.1*		100. *0.889*	

It is assumed that the system, made up by Reactor Pressure Vessel + Isolation Condenser + piping + valve, may be called into operation or may operate within each value of the foreseen ranges.

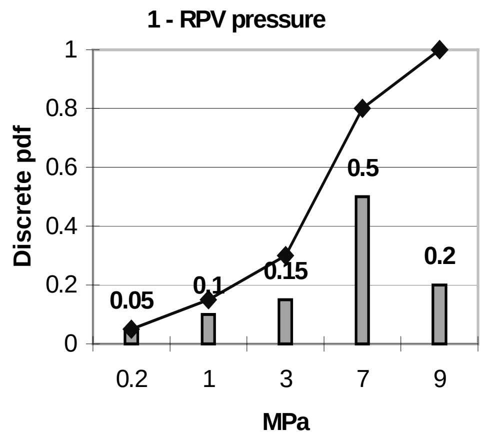
The selected critical parameters, which complete the identification of the system configuration during its mission, are reported in Table 3, together with the proposed discrete ranges of variation. They refer to both initial and boundary conditions for the system, and to those items that could lead to a performance degradation, namely affecting the heat transfer capability (presence of non-condensable gases in the RPV, x_1, or in the piping, x_2, heat losses in the draining pipe, C_2) and the natural circulation flow rate (inclination of the piping in the suction side of the heat exchanger, θ, liquid level in the tube bundle, L_2 , undetected leakage in the piping, *UL*, uncomplete opening of the gate valve, *POV*). As in the previous table, a probability is arbitrarily assigned to each discrete value.

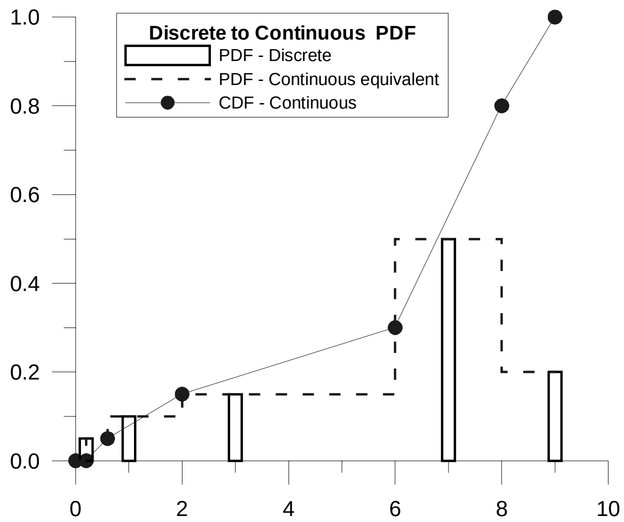
In general, we must point out that the identification of the parameters and associated values (range and probability of occurrences) should be the result of a previous PSA evaluation or a thorough set of Expert Judgement procedures.

As far as the probability of occurrences for the system parameters is concerned, Discrete Probability Distributions have been adopted for the selection of a first Probabilistic Set. This is due mainly to a more straightforward way to identify both the ranges and the corresponding probabilities. In order to investigate possible discrepancies in the reliability evaluation with respect to the type of probability distributions adopted, a second set based on Continuous Probability Distributions has been stochastically selected.

The discrete probability functions have been transformed in continuous functions via a “scaling” procedure as depicted in Fig. 5, thus obtaining cumulative distribution functions. As an example, the discrete and continuous probability distributions for the first design parameter are reported in Fig. 6.

For the selection of the Probabilistic Sets, a Fortran code has been written and IMSL routines with a Monte Carlo procedure have been adopted.





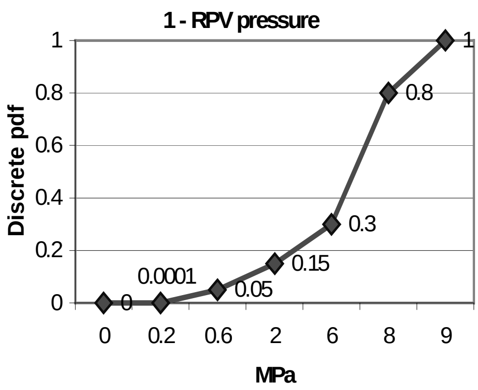


Fig. 6 – Discrete (top) and Continuous (bottom) Probability Distributions for the Pressure in the RPV – Design parameter.

Fig. 5 – Procedure adopted for the transformation from Discrete to Continuous Probability Distributions: parameter #1 - RPV Pressure.

It must be noted that the combination of all discrete values in Tables 2 and 3 brings to a several million possible configuration status for the system. Therefore, a limited but statistically meaningful selection of system configurations is mandatory, in order to make feasible the deterministic evaluation via best-estimate codes. In order to keep meaningful the dimension of the Probabilistic Set for a reliability evaluation, we have to maintain the set stochastically representative of the passive system behaviour in all conditions. The Wilks[20] assumption has been adopted for the aim. According to his equation, a sample of 64 system configurations has been selected, thus obtaining with a confidence coefficient of 0.99 that the stochastic configurations, hence of the outputs, bound at least 90% of the population values, hence of the possible output response of the passive system.

Besides the two Probabilistic Sets, a number of 6 deterministic configurations has been added to the sets (Table 4). The rationale is to assure that system configurations judged of particular interest by the experts are present in the run set. The choice refers to: the nominal conditions, taken as reference behaviour for the system, a low pressure-high level in the RPV, the presence of a significant percentage of non-condensable gas in the system (e.g. due to an incorrect maintenance), an undetected leakage in the valve or in the piping, a partial closure of the gate valve and a sort of “extreme” case with concurrent non-nominal values. The probability of occurrence for the Probabilistic (discrete pdf) and Deterministic Sets are reported in Fig. 7.

Table 4
Deterministic configurations and their probability of occurrence (non-nominal values marked in bold character).

		Design Params. Values					Critical Params. Values							
No.	**Main Parameter considered**	**P_1** MPa	**L_1** m	**L_3** m	**$T_p(0)$** K	**System layout**	**x_1** -	**x_2** -	θ deg	**C_2** kW	**$L_2(0)$** %	**UL** m^2	**POV** %	***Prob.***
1	Nominal conditions	7	8.7	4.3	303	Nominal	0.	0.	0.	5.	100.	0.	100.	*2.06E-2*
2	Pressure & Level	**0.2**	**12**	4.3	303	Nominal	0.	0.	0.	5.	100.	0.	100.	*6.20E-4*
3	Gas	7	8.7	4.3	303	Nominal	**0.01**	**0.5**	0.	5.	**50.**	0.	100.	*1.82E-5*
4	Leakage	7	8.7	4.3	303	Nominal	0.	0.	0.	5.	100.	**5.E-5**	100.	*2.32E-4*
5	Valve	7	8.7	4.3	303	Nominal	0.	0.	0.	5.	100.	0.	**10.**	*2.32E-4*
6	‘Extreme’	**0.2**	**5**	**2**	**368**	Nominal	**0.01**	**0.5**	0.	**20.**	0.	**1.E-5**	**50.**	*4.5E-12*

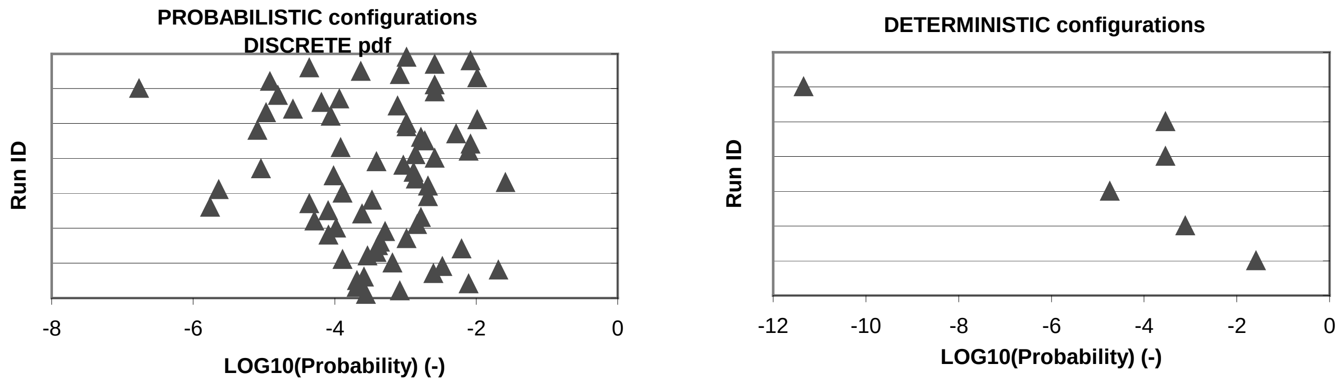


Fig. 7 – Probability of occurrence for each system configuration as selected in the Probabilistic Set (Discrete probability distribution functions) and in the Deterministic set.

Failure Criteria

Acceptability or design limits for the system operation must be known in order to assign failure criteria and to define indicators of system performance. Those limits are specific for the system and connected with its mission. In principle, several acceptability limits and failure criteria can be adopted. For the test case selected, the most suitable parameters to be taken into account for reliability purposes appeared to be those related to its mission, i.e. heat rejection capability.

Thus the following quantities (ref. Fig. 3) are assumed to be strictly connected with the acceptability or design limits of the system and are the output of the best-estimated code calculations:
Thermal power exchanged through the Isolation Condenser (W_2);
Mass flowrate at the IC inlet (Γ_2)

According to step ii. in the methodological procedure, a failure criterion as a function of time should be adopted. This is due mainly to the inherent dynamic behaviour of the system to be characterised and also to the oscillating behaviour that is a general feature for the natural circulation passive system.
A possible failure criterion has been defined as:

$$(Z - Z_{ref}) / Z_{ref} < (-0.2) \qquad (1)$$

where Z is either W_2 or Γ_2 and "ref" is related to the code calculation for the reference or nominal configuration. The condition (1) has to be continuously valid for a time interval greater than 100 s. This criterion is similar to that depicted in Fig.2, except for the reference value (indicated as "nominal") that is not a constant value. In practice, the observed parameter has to follow the reference or nominal trend and it has not to fall below a 20% difference for more than a fixed time period.

Other indicators were defined for the evaluation of the system performance and reliability:

Total Failure Time, that is the sum of the periods during which the condition *(1)* is verified;

Ratio Y/Y_{ref}, where Y is the integral of W_2 or Γ_2 between time 0 (start of the system operation) and the time of observation, or mission time, calculated for each configuration in the Probabilistic and Deterministic Sets; e.g. for the rejected thermal power:

$$\frac{\int_0^{Tmission} W_2(t)\,dt}{\int_0^{Tmission} W_{2,\,ref}(t)\,dt} \qquad (2)$$

This last kind of indicator appears to be the most suitable for quantitative reliability evaluations, because it allows the definition of a continuous, cumulative performance function referred to the probability of occurrence of the run set (Fig.11).

Analysis Results

The RELAP5 mod3.2 was adopted as best-estimate code for the deterministic characterisation of the passive system.

A single accident transient was selected, namely a turbine trip followed by a reactor Scram, with the safety system required to intervene in order to reject the core decay power via an external pool. The transient analysis starts with the reactor Scram and valve opening.

A total of one-hundred-thirty-four (134) system configurations have been selected according to deterministic and stochastic choice and led to the corresponding code runs. Two stochastic selections of 64 configurations each were carried out, the first based on discrete pdf, the second on continuous pdf. It must be recalled that the six system status, shown in Table 4, have been deterministically selected in order to get the sensitivity in relation to the effect of each relevant parameter, and not with the purpose to bound the results expected from the stochastically based selection.

Therefore, 134 code runs have been performed, each one lasting 15000 s, i.e. four hours in the transient (observation time). The real observation period should be equal to the mission time for the safety system, typically in the range of 1-3 days in our case. The limit to 4 hours was necessary for the analysis to be realisable, also being the REPAS methodology at a demonstration level. The requested calculation time was about 20 hrs for each run. Some runs lasting 150000 s, largely beyond one day of simulated transient, were carried out. This allowed a roughly evaluation whether all relevant behaviour features for the passive system were investigated in the reduced observation period. No significant results came out.

Typical outputs are reported in Figs.8 and 9. The time evolution of the thermal power rejected to the pool is shown. Fig. 8 refers to the core decay power and heat transfer capability of the system in nominal conditions, while Fig. 9 shows results of code runs in the Probabilistic Sets.

At the end of the analysis, the integral result set was compared with the failure criteria, as indicated in the previous section. As an example, Fig. 10 represents a *system performance indicator* graph, based on the IC thermal power ratio according to Eq.2. The run configurations in the graph have been clusterised according to their probability of occurrence. Another results representation, suitable for a thorough reliability evaluation process is that depicted in Fig.11. A classic Cumulative Distribution Function for the passive system is obtained, by ordering the Isolation Condenser (IC) power integral ratios (as from Eq.2) with respect to the probability of occurrence of each configuration in the Probabilistic and Deterministic Sets. The total sum of the occurrence probabilities is normalised to 1.

This result, in cumulative (cdf) or in probability distribution function (pdf) form, could be adopted in a classic Fault Tree-Event Tree analysis, provided a "phenomenological" path is created to take into account for the inherent physical features of the passive systems (as suggested in ref.14).

Conclusions

The analysis of the results shows that the procedure is suitable to evaluate the performance of a passive system on a probabilistic / deterministic basis. Important information can also be derived for reliability evaluation purposes, being the final goal to link it with classic PSA methods (e.g. Fault Trees-Event Trees).

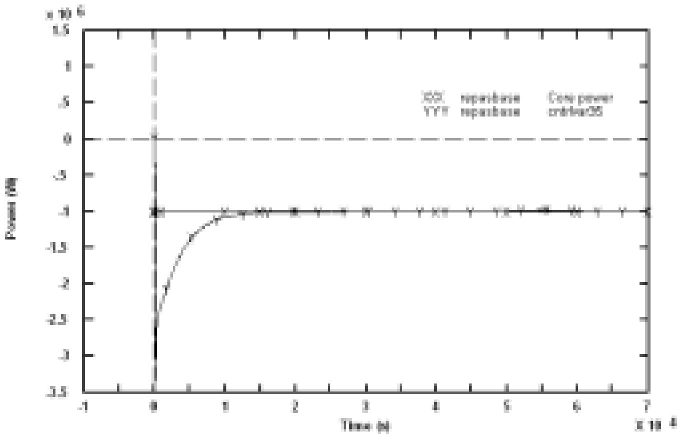


Fig. 8 – Time evolution for the Core (X) and Reject (Y) Thermal Powers for the reference or nominal configuration (i.e. all Design and Critical Parameters with nominal values)

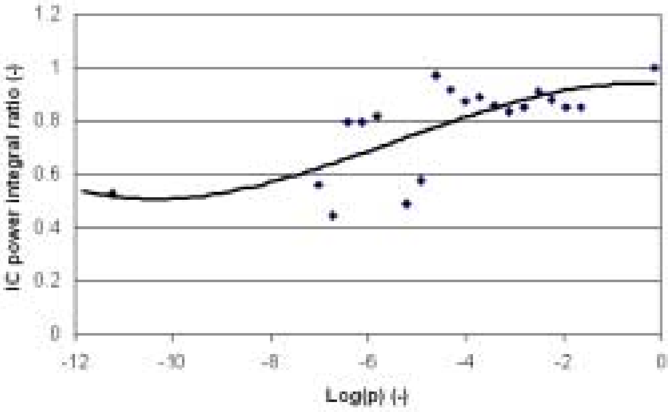


Fig. 10 – Example of a *system performance indicator:* rejected thermal power ratio (single run over nominal) vs. probability of occurrence of the Passive System configuration (Probabilistic Set), and its best fit curve

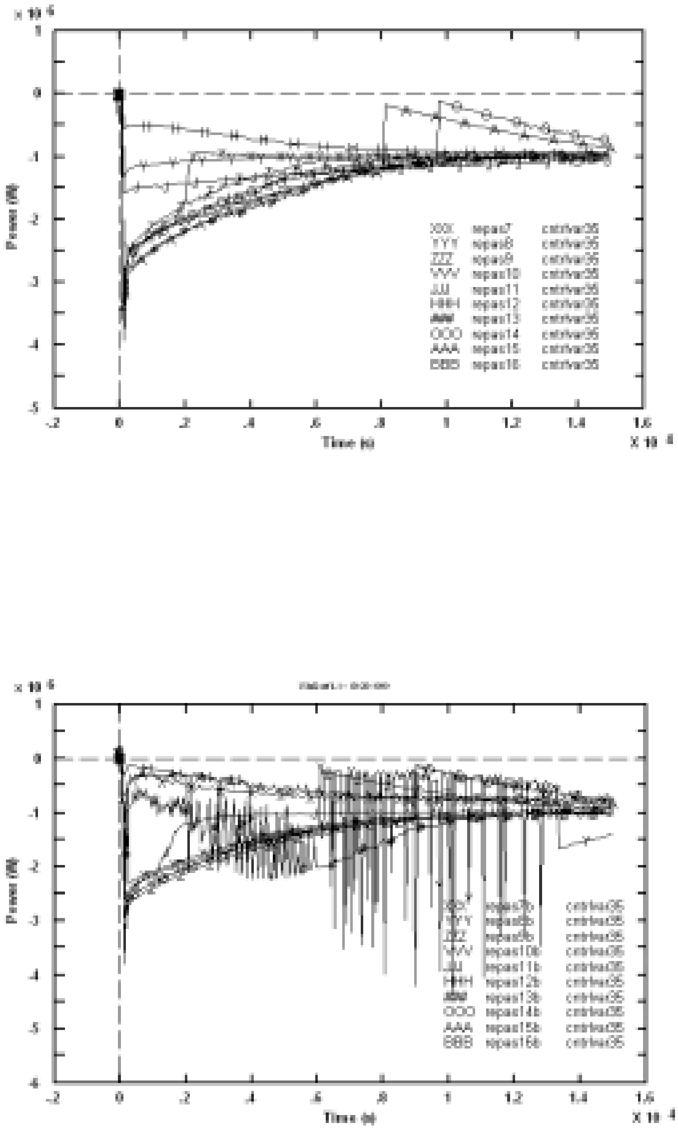


Fig. 9 – Typical results for the Rejected Thermal Power for different system configurations belonging to the Probabilistic Set

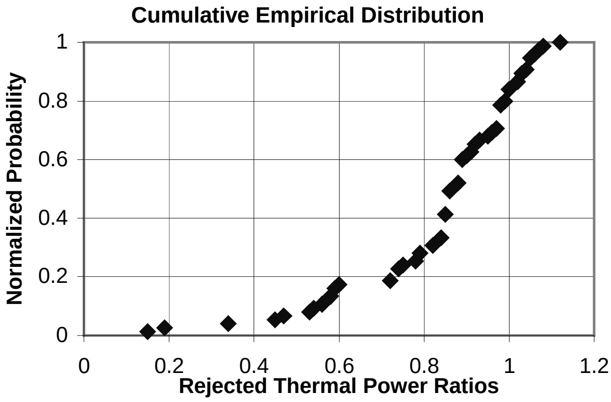


Fig. 11 – Cumulative Distribution Function obtained from IC thermal power ratios by ordering the run sets with respect to the corresponding probabilities (as in Probabilistic and Deterministic Sets)

The REPAS procedure, still in a development and assessment phase, could be applied: i) to evaluate the acceptability of a passive system, specifically when nuclear reactor safety considerations are concerned; ii) to compare two different passive systems having the same mission; and, with additional investigation: iii) to evaluate the performances of an active and a passive system on a common basis.

Limitations of the achieved results and areas for further development and improvement of the procedure have been identified. They are summarised as follows:

- A reference performance for the passive system must be selected as nominal behaviour of the system, corresponding to the ideal fulfilment of its mission. This must be available to the analysts that are user of the procedure. The simulation analysis could be utilised to confirm this target. This implies that the designers of the system are involved in the application of the procedure.

- More rigorous and systematic basis is necessary to select the parameters that characterise the system status (e.g. a complete Expert Judgement procedure). The independence of results upon a minimum number of design and critical parameters should be demonstrated.

- A convergence and stability verification should be performed: the number of combinations (i.e. the number of code runs) for the system parameters should also be defined in such a way to ensure the convergence and stability of the results (i.e. judgement of system performance), when the number of runs is increased.

- In the present context, the thermal-hydraulic system code has been considered as an ideal tool, perfectly reproducing the real physical behaviour of the whole system. Uncertainties in the predictions should be added in the analysis.

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"RELIABILITY METHODS FOR PASSIVE SYSTEMS" (RMPS) STUDY – STRATEGY AND RESULTS

M.E.Ricotti[°], E.Zio[°], F.D'Auria[§], G.Caruso[*]
CIRTEN Consortium

[§]University of Pisa - DIMNP
via Diotisalvi 2, I-56126 Pisa, Italy
[*]University of Roma "La Sapienza" – Dept. of Nuclear Engineering
Corso Vittorio Emanuele II, 244 – I-00186 Roma, Italy
[°]Polytechnic of Milano – Dept. of Nuclear Engineering
via Ponzio 34/3, I-20133 Milano, Italy

Abstract

The RMPS (Reliability Methods for Passive Systems) is a research programme sponsored by EU within the 5th European Framework Programme. Main goal of the activity, begun in the first half of 2001, is the identification of a suitable methodology capable to evaluate both qualitatively and quantitatively the reliability of passive systems. The research development foresees the application of such a methodology to study cases, e.g. passive systems of industrial interest. The working team is leaded by CEA and groups research centres (GRS, JRC, ENEA), universities (CIRTEN consortium) and industries (Technicatome).

The approach moves from a previous study performed by ENEA and some CIRTEN partners. The strategy is based on the identification of all possible ways of departure of the system from the nominal behaviour, defined at the design stage. The passive system can fail its mission not only due to classical mechanical failures, but also because of physical failures, usually related to thermalhydraulic features. Moreover, the failure state of the passive system has to be defined, e.g. connected to physical parameters as natural circulation flow rate or heat transfer rate.

The paper reports a preliminary roadmap proposed for the development of the RMPS method, with a brief description for each step.

Then a sensitivity analysis on the results of the previous study has been performed, showing the most important parameters affecting the behavior of the passive safety system selected as a case study. The analysis revealed also a singular occurrence in the calculations and some considerations have been drew.
Finally, a useful basis for a thorough Expert Judgement process, to be carried out on the identification of the characteristic parameters for the passive safety system, has been set up. Justifications for the selection of the main parameters affecting the system behavior, hence its reliability, for their ranges of variation and their probability of occurrence have been identified.

Introduction

During the last decade, a lot of concepts for advanced and innovative nuclear reactors has been proposed, the larger part of them relying on passive features to implement their strategy on safety. Passive systems has been assumed to increase the degree of safety of the plant, by means of a simplification of the process, the systems and the components. An enhancement in reliability was also claimed for the passive safety systems.

These statements are mainly based on the consideration that usually in passive systems no human intervention is required, no external energy supply is needed and only natural physical phenomena are involved in their functioning, with a very limited number of active devices if not at all.

However, even in these natural-lead conditions, specific physical situations could occur, thus leading to impair the driving forces on which the passive systems is based. As an example, the presence of non-condensable gases could heavily degrades the heat transfer capability into a heat exchanger tube bundle, or could hamper the natural circulation in a loop.

Moreover, no specific methods or procedures are available to thoroughly evaluate the reliability of a passive system, as an absolute or when compared to an active system, corresponding in its mission.

In this frame, the Reliability Methods for Passive Safety functions (RMPS) project aims to propose a specific methodology to assess the passive systems thermal-hydraulic (T-H) reliability. This methodology should be intended to aim at:

- identifying and evaluating the sources of uncertainties and determining the important variables, i.e. those variables whose uncertainty has a significant impact on the uncertainty in the T-H output performance of the passive system under study;

- propagating efficiently such uncertainties through T-H codes and assessing a reliability measure of the T-H passive system with respect to its required performance;

- Insert the passive systems' reliability estimates within an accident sequence so as to be able to assess its relevance.

The RMPS research activity[1] is co-funded by EU within the 5th Framework Research Programme. The working team is leaded by CEA and groups research centres (GRS, JRC, ENEA), universities (CIRTEN consortium) and industries (Technicatome).

Proposal for a Methodology Roadmap

The activity carried out by the CIRTEN consortium in the first phase of the RMPS study, moves from a critical analysis of a previous work performed by ENEA, Polytechnic of Milano (POLIMI) and University of Pisa (UNIPI), identified as REPAS (Reliability Evaluation of PAssive Safety systems) study[2].

In that study, a preliminary path was traced. A case study was selected and a joint deterministic / probabilistic analysis was applied to qualitatively evaluate the reliability of a two-phase, natural circulation passive system.

As a preliminary consideration on the methodological approach, it could be stated that a parallel exists with the reliability studies on mechanical and structural components and systems. To assess the failure probability, a deterministic mechanical model is required and a probabilistic representation of input variables uncertainties is needed. Similarly, the developing RMPS requires a deterministic characterization of the passive safety system, via best-estimate T-H codes, starting from Boundary and Initial Conditions (B&IC) that may range within probability distributions to be identified, thus treated as uncertainty sources for the passive system behavior.

On the contrary, while the failure condition and the utilization of the failure probability within the PSA are quite straightforward for active mechanical systems and components, this is not so for T-H passive systems. However, it appears that the tools and techniques for both uncertainty propagation and sensitivity analysis ought to be implemented in the path that leads to the reliability assessment of T-H passive systems.

The analysis and the conceptual development carried out by the CIRTEN partners led to a proposal for a possible methodological roadmap of the RMPS approach. The steps of the procedure are depicted in Fig.1 and briefly presented in the following key points, having that picture as a reference.

The main objective is the qualitative and quantitative assessment of the reliability of a passive system.

***i.* Characterization of operational modes** for the passive system under evaluation (e.g. start-up, shutdown, etc.): specification of a "nominal phase" as the object of the case study.

This first step is mainly devoted to the characterization of the passive system in terms of its functionality scope and mission, and to the identification of the parameters related to that system configuration and nominal functioning (called "*design* parameters").

Moreover, a preliminary list of those parameters that may affect and hamper the mission of the system should be given (called "*critical* parameters").

During this step and the following characterization & identification phases (steps ***i.*** to ***iv.***), the distinction made in ref.2 between *design* and *critical* parameters, in order to define the scenarios to be analysed should be maintained. Although the subsequent treatment of these parameters, and of their uncertainty, is the same, it seems convenient to keep the distinction, as in the classic PSA when Initiating Events and subsequent Losses of Functionality are dealt with. This helps in keeping systematic and clear the methodology.

***ii.* Define the proper physical failure criteria** for the mode under study (e.g. in terms of the process variables Temperature, Pressure, Flow Rate, Power when exceeding specified threshold limits).

From these parameters, the failure probability and unreliability of the passive system can be defined. FMEA[3] could possibly be the proper tool to perform this task.

The mission of the safety system or component under consideration must be duly considered in the process, in particular the ultimate objective of the mission for the whole system (e.g. maintain peak cladding temperature within design limits, during an accident transient which requires the intervention of the residual heat removal passive system). Not only threshold criteria, but even continuous or integral time criteria could be adopted.

***iii.* Identify related and root causing processes** for each failure criteria, through a structured hierarchy, e.g. via Analytic Hierarchy Process (see next paragraphs), Fault Tree method, Event Tree approach.

This step of decomposition of the problem can also aid the possible utilization of the passive system reliability results within a classic PSA scheme (e.g. Fault Tree/Event Tree) for the whole system or plant into which the passive system is embedded.

***iv.* Continue the structured hierarchy to parameter level,** by decomposing the processes in terms of their relevant system parameters.

This is the final step of the decomposition which breaks down the problem to the level of quantification. Possibly, also steps ***iii.*** and ***iv.*** should most likely rely on structured Expert Judgment Elicitation methods for information acquisition and treatment.

***v.* Ranking of the most important parameters** for the top failure criteria.

At this stage, this task, and the subsequent steps ***vi.*** and ***vii.***, could be carried out utilizing tools such as the Analytic Hierarchy Process pairwise comparisons.

At a later stage, these steps could be repeated for a quantitative evaluation, e.g. via SUSA[4] method application, after the code runs have been executed, thus having detailed information on the passive system behavior.

***vi.* Screening of the least important parameters.**

This phase is to be carried out possibly via Expert Judgment supported by proper Decision Analysis methods. To reduce the number of *design* and *critical* parameters could be mandatory in order to limit the dimensions of the space of characterization (i.e. step ***xiv***.).

***vii.* Identification of parameters relations and dependencies,** both physical and stochastic.

In this phase the physical constraints and the stochastic dependencies among the parameters should also be preliminary pointed out, as done in the development of a classic PSA.

From this level on, three branches should be followed. The third branch in Fig.1 (steps ***x.*** to ***xiii***.) is mainly devoted to a simplification of the process belonging to the second one (steps ***xi.*** to ***xv***.).

***viii.* Detailed modeling (code)** of the effects of the relevant parameters and assignment of their best estimate values; this step is necessary to set up a suitable, detailed nodalization to be used for the best-estimate code analysis, according to the *design* and *critical* parameters selection.

***ix.* Deterministic evaluation** (best estimate calculation).

This step is needed to obtain the “reference” or “nominal” behavior of the system, to be compared with the degraded behavior in “failure” conditions and with the design and regulatory specifications.

Multiple “nominal” analyses could be required, on the basis of the various missions that the passive system is asked to fulfill (different accident scenarios to be evaluated).

***x.* Modeling &Validation:** modeling of the most important processes and parameters and validation of results of the simplified models.

The complementary use of simplified models, in support to the best-estimate codes, should allow to run several scenarios with differing values of those parameters previously identified as important, thus gaining the relevant information on the sensitivity and importance of the system parameters. In many cases, this would be impractical with full scope best-estimate codes.

xi.* Assign the probability distributions** for the parameters identified as important, describing the uncertainty in their values (Expert Judgment). Such distributions may be conditional, in order to account for the dependencies identified in step ***vii.

The definition of the type of probability distribution for the parameters and the number of passive system B&IC scenarios, i.e. the number of code runs to be performed, should be decided with the aid of Expert Judgement and the possible introduction of proper probabilistic criteria (e.g. Wilks rule, random sampling vs. Latin Hypercube, Stratified or Monte Carlo selections, etc.).

***xii.* Probability propagation, uncertainty and sensitivity analysis** with the simplified models for the quantitative identification of the most important combinations of parameters with respect to system unreliability.

Due to the expected time saving in code run performance, the simplified models should allow the evaluation of a wider set of B&IC configurations, i.e. a larger number of code runs, with respect to best-estimate codes. This should, in principle, enable the analysts to study also the interaction effects of the uncertainties on multiple parameters, jointly varying, and not solely the effects of the uncertainty on one parameter at a time.

***xiii.* Elaboration with external methodologies**, e.g. Neural Networks models (NN), Response Surfaces (RS).

These “exotic” methods of investigation demonstrated their capability in finding essential information from multi-parameter data sets pertaining to complex nonlinear system behaviors, without requiring explicit mathematical modeling. They could be adopted as fast, albeit approximate, models within an extensive search of the relevant characteristics of the passive system behavior corresponding to several different scenarios of parameters values.

This step could also be repeated after steps ***xii***. and ***xiv***., to provide useful feedbacks on the method.

***xiv.* Probability propagation and analysis** of the uncertainties in the identified most relevant parameters through the detailed model (code).

A set of methods and software tools to perform this task are available within the RMPS partners: UMAE[5] by CIRTEN-UNIPI and that from GRS[6], for the methods; SUSA and ATHLET by GRS, an ad-hoc CATHARE version by CEA, for the software tools.

The uncertainty methods and codes could be utilized to handle both the probability distribution of the characteristic parameters and of the B&IC and models inherent uncertainty.

***xv.* Quantitative Evaluation for Reliability** defined as the probability of the system performing its function, as implicitly specified in the failure criteria, within the assigned mission time.

It seems that the System Performance Indicators reported in ref.7 provide only an integral information on the passive system behavior. It would seem more informative to display the whole information content provided by the output system performances corresponding to all the parameters-values scenarios runs.

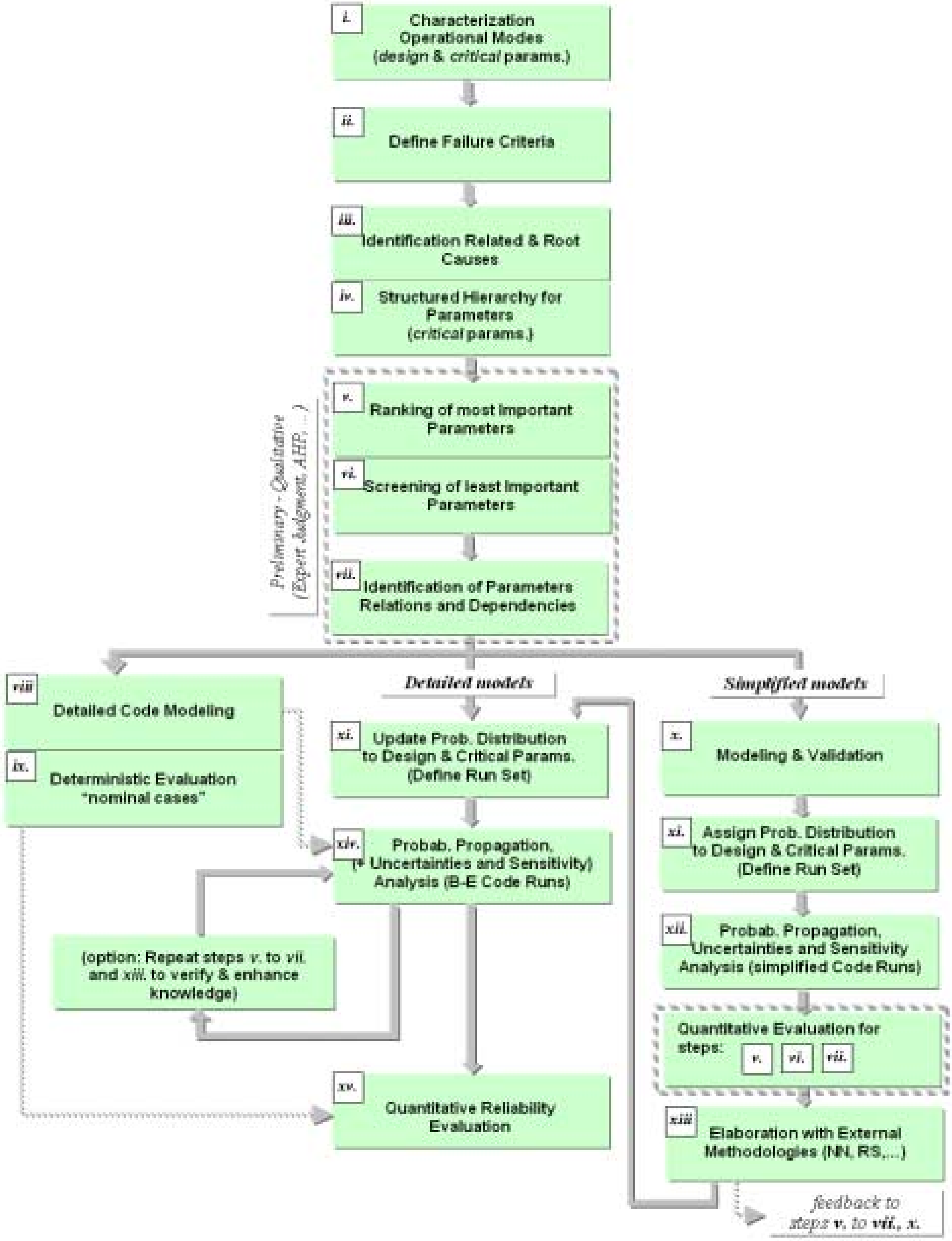

Fig.1
Preliminary methodology roadmap proposed for the RMPS approach.

Illustrative examples of such representations could be those shown in Fig.2 to 5. All the System Model Evaluations (Runs) carried out in the previous step ***xiv***. of uncertainty propagation are displayed in terms of the probability of occurrence of that B&IC configuration/scenario (step ***xi***.) vs. a defined design performance satisfaction level (Fig.2). This kind of representation has more similarity with the classical representation of risk results in terms of probability vs. consequence.

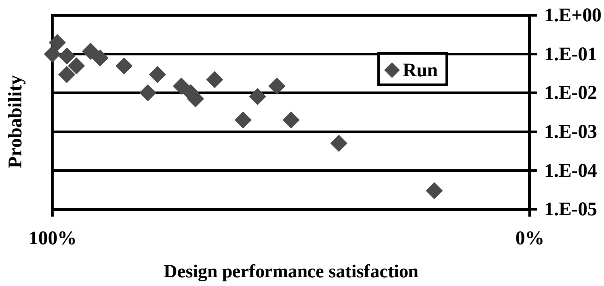

Fig.2
Display of the Deterministic Evaluations (best-estimate Runs) – Degree of satisfaction of the mission criteria (as identified in the Failure Criteria) for the safety system vs. probability of occurrence of the specific system configuration (as set up in the Run).

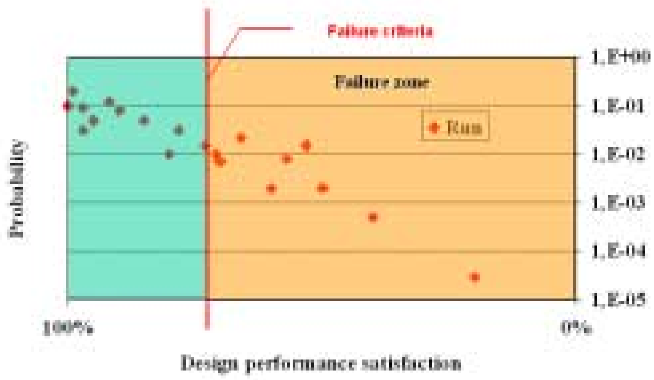

Fig.3
Display of the Deterministic Evaluations (best-estimate Runs) – Application of the Failure Criteria and identification of the Failure Zone.

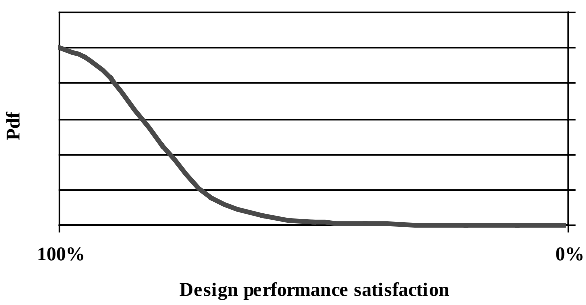

Fig.4
Calculation of equivalent Pdf distribution from Deterministic Evaluation (Runs)

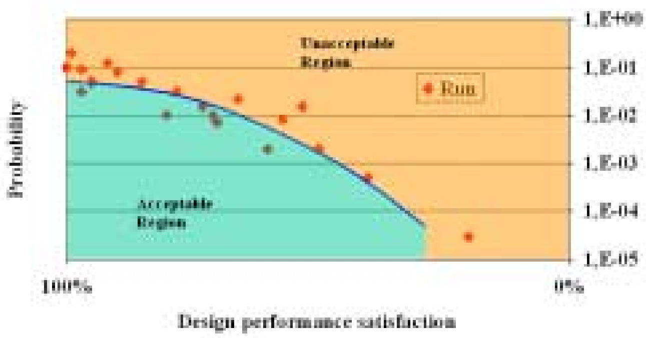

Fig.5
Display of the Deterministic Evaluations (best-estimate Runs) – Acceptability criteria as in "Farmer Curve"

The system reliability could then be evaluated with respect to the selected failure criteria, e.g. "removal of at least 90% of the decay heat during the first 72 hours from the accident". A Failure Zone could be defined and the reliability of the system could be represented by the sum of the probabilities of the runs in the Non-Failure Zone (Fig.3). In principle, a probability function (pdf) curve for the passive system mission can be obtained (Fig.4).

A further development of the results representation could lead to a sort of 'Farmer Curve', as in Figure 5.

Evaluation of the possible use of reliability information on the passive system thus obtained, in classical PSA tools (e.g. with Dynamic Event Trees, Fault Trees) could be investigated.

Analysis of the "REPAS" results

Beyond the proposal of the methodological roadmap, some evaluations of the results obtained in the REPAS study has been carried out.

The case study selected was a two-phase, natural circulation loop (Fig.6), activated by a gate valve and connected to the reactor pressure vessel. A condenser dipped into a water pool is the main component in the path to the heat sink.

The *design* and *critical* parameters, together with their range of variation and the corresponding probabilities of occurrence are reported in Tables 1 and 2.

In order to perform the characterization of the passive system behavior, 64 system configurations were identified via a stochastic selection, according to the discrete probability distributions assigned to the parameters. Hence 64 runs were performed, i.e. 64 transient analysis by RELAP5mod3.2 code.

Two Passive System Performance Indicators were adopted to evaluate the system behavior, based on the ratio between the natural circulation flow rate or the thermal power rejected to the pool, obtained during each analyzed configuration, and the same quantity obtained in the nominal ("reference") behavior of the system, each one integrated over the mission time, i.e.:

Table 1 – **Design Parameters of the passive system.**

Design Parameter		**Unit**	**Nominal Value**	**Range**	**Discrete Initial Values & Probabilities**					
P_1	RPV pressure	MPa	7	0.2-9	0.2	1	3	7	9	value
					0.05	*0.1*	*0.15*	*0.5*	*0.2*	*pdf*
L_1	RPV collapsed level	m	8.7	5-12	5	7	8.7	10	12	
					0.05	*0.1*	*0.5*	*0.2*	*0.15*	
L_3	POOL level	m	4.3	2-5	2		4.3		5	
					0.1		*0.8*		*0.1*	
$T_p(0)$	POOL initial temperature	K	303	280-368	280		303		368	
					0.1		*0.8*		*0.1*	
-	System geometry: layout	-	-	Not assigned	- 1.0					

Table 2 – Critical Parameters of the passive system.

Critical Parameter		**Discrete Values & Probabilities**							
x_1	RPV non-condensable fraction	0.	0.01	0.1	0.2	0.5	0.8	1.	value
		0.719	*0.12*	*0.07*	*0.05*	*0.03*	*0.01*	*0.001*	*pdf*
x_2	Non-condensable fraction at the Inlet of IC piping	0. *0.71*	0.01 *0.12*	0.1 *0.07*	0.2 *0.05*	0.5 *0.03*	0.8 *0.01*	1. *0.01*	
θ	Inclination of the IC piping on the suction (deg)	0. *0.5*		1. *0.4*		5. *0.08*	10. *0.02*		
C_2	Heat Losses piping – IC Suction (kW)	0. *0.10*		5. *0.7999*		20. *0.10*	100. *0.0001*		
$L_2(0)$	Initial condition liquid level - IC tubes, inner side (%)	0. *0.1*		50. *0.1*			100. *0.8*		
UL	Undetected leakage (m^2)	0. *0.8899*		1.E-5 *0.1*		5.E-5 *0.01*	10.E-5 *0.0001*		
POV	Partially opened valve in the IC discharge line (%)	1. *0.001*		10. *0.01*		50. *0.1*	100. *0.889*		

$$\frac{\int_0^{Tmission} \Gamma_2(t)dt}{\int_0^{Tmission} \Gamma_{2,\,ref}(t)dt}\,,\quad \frac{\int_0^{Tmission} W_2(t)dt}{\int_0^{Tmission} W_{2,\,ref}(t)dt}$$

A sensitivity analysis on the results has been carried out by means of Standardised Regression Coefficients (SRC), Partial Correlation Coefficients (PCC) and Correlation Coefficients (CC), in order to evaluate the degree of influence of the parameters on the Performance Indicators. The results are reported in Fig.7 and Fig.8. They shows that the Reactor Vessel Pressure P_1 and the Reactor Pressure Vessel non-condensable fraction X_1 are the most important parameters for the passive system behaviour.

A further analysis on a possible correlation between the two Performance Indicators has been performed, showing that a substantial linear correlation exists (Fig.9). The two Indicators appear to give the same information, hence one is redundant. However, it must be noticed the singular point in figure, far from the linearity trend. It corresponds to the configuration #54 stochastically selected from the discrete probability distributions.

A check on the RELAP input revealed that the transient analysis was carried out with an incorrect value for the collapsed liquid level in the Reactor Pressure Vessel: the Initial Condition was set to 15 m instead of the maximum allowed by the range (12 m).

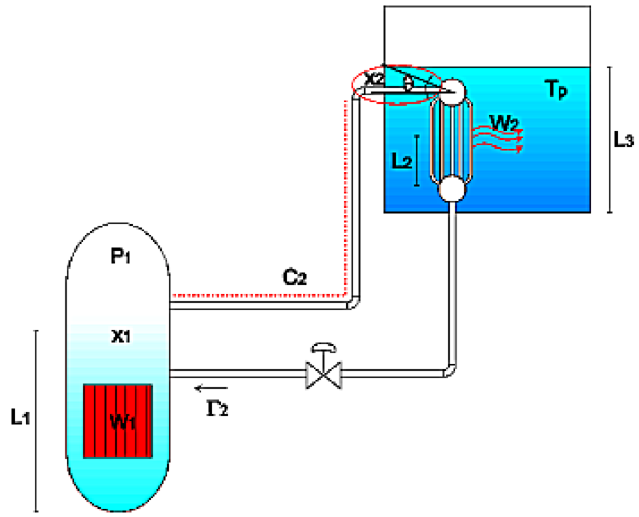

Fig.6
Passive safety system adopted as test case in the study.

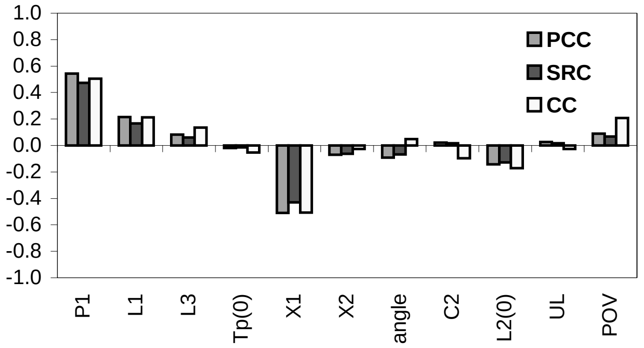

Fig.7
Partial Correlation, Standardized Regression and Correlation Coefficients for the Power Ratio Performance Indicator W_2/W_2ref.

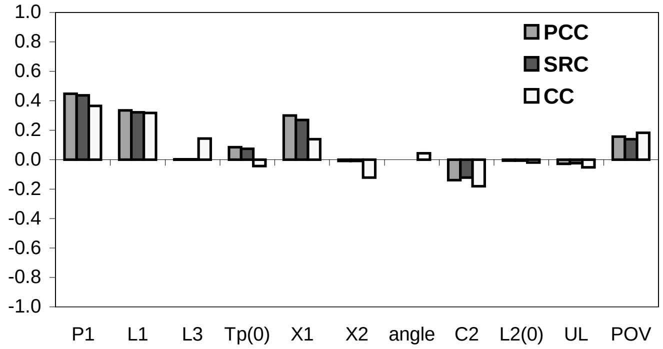

Fig.8
Partial Correlation, Standardized Regression and Correlation Coefficients for the Flow Rate Performance Indicator Γ_2/Γ_2ref.

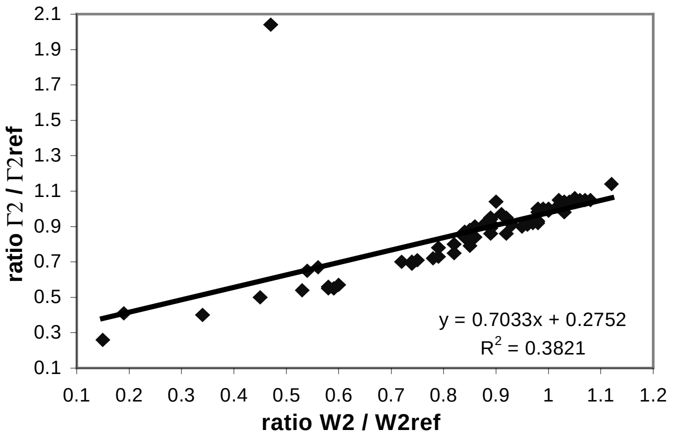

Fig.9
Linear output correlation between the Performance Indicators.

This occurrence showed that i) the correlation analysis could be a useful tool to analyse the regularity of the Performance Indicators and to point out discrepancies, possibly connected to errors in the system characterization; ii) this situation was not foreseen by the "expert" that defined the range of variation for the RPV level, anyway it represents a physical situation that could really occur; iii) being the previous assumption valid, system configurations could exist that do not represent a proportionality between the natural circulation flow rate and the corresponding thermal power rejected to the pool, thus leading to the necessity to investigate possible physical non-linearities in the passive system behavior.

Moreover, the hypothesis of linearity between inputs and outputs is likely to be imprecise, and a non linear analysis tool should be used to obtain more realistic results. The degree of correlation between input parameters should also be taken in account.

The last step of the analysis conducted has been that of drawing complementary cumulative frequency curves as a function of the Performance Indicators. Fig.10 represents those for the Power ratio Performance Indicator. From these curves one can asses the percentage of the scenarios considered that reach a given value of the output variable chosen.

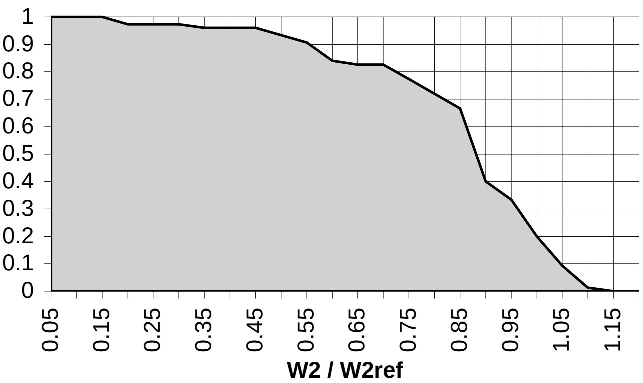

Fig.10
Complementary cumulative failure percentage for the Power ratio Performance Indicator.

Basis for an Expert Judgement Approach

The last activity performed during this first period of work in the RMPS project by the CIRTEN partner, refers to step ***iv***. and step ***xi***. in the proposed methodological roadmap (Fig.1). The objective is a preliminary Expert Judgement approach, as suggested in the procedure, to duly found and justify the assumptions on *design* and *critical* parameters for the Passive System.

The path that lead to the stochastic selection of the system configurations to be analysed by the best-estimate code, is here briefly run through.

According to the general approach adopted in the REPAS study, a set of main characteristic parameters has been identified, among those affecting the behaviour of the Passive System under investigation. Then a range of variation for each parameter has been associated, together with a probability of occurrence of the values. All the process has been performed via a sort of rough, unstructured “expert” or “engineering judgement” procedure.

Selection of the Parameters

Once defined the System Mission, i.e. the rejection of the core decay power to an external heat sink in this case study, a series of discussions on the main phenomena involved in the physical behaviour of the passive system has been carried out. That led to the identification of those quantities having the capability to degrade, to hamper or to prevent the fulfilment of the mission.

Two categories have been defined for the parameters related to those two items: the *Design* and the *Critical* categories.

The *Design Parameters* refer to the quantities directly connected to the design of the system, to the quantities representing the physical links of the passive system to the whole system into which it is inserted (e.g. the primary system), and to the parameters identifying the status or behavior of the passive system during its functioning, in nominal or operational conditions.

The *Critical Parameters* refer to the quantities that could represent a direct source of “failure” for the passive system, i.e. they could leave the mission unfulfilled. Therefore the Critical Parameters are directly connected to the physical phenomenology involved in the passive system behavior, that could impair it.
In practice, the Design and Critical parameters refer to Initial Conditions and Boundary Conditions for the Passive System, thus identifying a specific configuration of the system, at the beginning and all along the mission time.

Range for the Parameters

The procedure for the reliability evaluation foreseen the characterisation of the passive system behaviour in all possible conditions. This task is performed by investigating the passive system response in different system configurations. The configurations are set up by composing different values for the Design and Critical Parameters, to be selected within specific range of values for each parameter.

These ranges identify the bounding values within which the parameters could fall, during nominal, anticipated, abnormal or accident conditions for the whole system.

Probability Distributions for the Parameters and corresponding Ranges

In order to duly compose the system configurations, to be analysed for the passive system characterisation, a stochastic selection has to be performed. For this reason, a probability distribution must be associated to each range of variation for the design and critical parameters. The distribution represents the probability of occurrence of the values, assumed by the parameter when the passive system is on demand (I.C.) or during its operation (B.C.).

The configuration set is made up via a Monte Carlo selection of the parameter values.

Criteria adopted in the Case Study and Justification for the Choices

According to the general procedure previously depicted, the following apply to the case study.

1. The attention has been mostly paid on two items:

    - the heat transfer capability of the passive system,

    - the natural circulation flow rate.

    These two general features have been judged as the most important for the fulfilment of the mission by the passive system. Since the goal is the rejection of the decay heat to an external heat sink, the capability of the system to perform its mission relies mainly on its heat transfer features and on the degree of natural circulation the system could maintain during operation. Therefore the design and critical parameters selected for the analysis are directly linked to these two items.

2. In order to simplify both the identification of the ranges and their corresponding probabilities, discrete values have been selected. As a general rule, a central pivot has been identified, then the range has been extended to higher and lower values, if applicable. The pivot value represents the nominal condition for the parameter. The limits have been chosen in order to exclude unrealistic values or those values representing a limit zone for the operation demand of the passive system.

3. Once the discrete ranges have been set up, discrete probability distributions have been associated. As in step 2., the general rule adopted is that the higher probability of occurrence corresponds to the nominal value for the parameter.

Then lower probabilities have been assigned to the other values, as much low the probability as much wide the distance from the nominal value, as in a sort of Gaussian distribution.

In Table 3, the reasons for the selection of the parameters, their ranges and probability distributions, are reported.

Conclusions

During the first period of activity the CIRTEN consortium, partner in the RMPS project, proposed a methodological approach to the evaluation of the reliability of passive systems. The attempt is to find a systematic approach to the problem and to organize and justify all the assumptions and steps of the research activities.

The proposal is now under evaluation and critics by the RMPS partners.

The proposal moved from a previous work carried out by ENEA, UNIPI and POLIMI. A sensitivity analysis on that results has been performed, leading to the identification and evaluation of the sources of uncertainties in the passive system operation, and the determination of the important variables. A singular occurrence on the calculations has been identified.

Moreover, following the rationale depicted in the proposed methodology, a preliminary analysis of the engineering judgement adopted for the choice of the design and critical parameters, their nominal values, the ranges of variation and the corresponding discrete probability distributions, has been carried out. This step could be a useful basis for a thorough Expert Judgement process.

Table 3 – Justification for the choice of the design and critical parameters, their ranges and their probability distributions.

	Choice justification(+)	Parameter range	Range justification	Probability values justification(*)	
P_1 Reactor Pressure (I.C.)	The vessel pressure determines the operating pressure of the passive system, hence the heat transfer capability of the Isolation Condenser.	0.2 - 9 MPa (nominal: 7 MPa)	0.2 MPa = minimum pressure value below which GDCS actuates. 9 MPa = maximum pressure value above which SRV actuate. Both SRV and GDCS largely affect the IC performance. Eventual IC actuation in combination with SRV and GDCS may constitute an additional design constraint not addressed in the present study.	7 MPa (nominal operation pressure) is the most probable value expected for the operation of the IC: half the total distribution probability has been assigned to this value.	*DESIGN PARAMETERS*
L_1 Reactor Liquid (collapsed) Level (I.C.)	The vessel level characterizes the total coolant mass hence the thermal capacity or thermal inertia for the system.	5 – 12 m (nominal: 8.7 m)	5 m = limit value for core uncovery. 12 m = elevation of the connection between the RPV and the steam line. Level value below 5 m implies unsafe conditions for the core and un-successful mission for the IC. Level value above 12m is challenging the safe operation of the SRV. Both of these conditions imply inadequacy of the overall system design.	see above	
L_3 Pool Level (I.C.)	The pool level characterizes the thermal capacity of the heat sink in the long run and may affect the heat transfer coefficient across the the IC tube bundle.	2 - 5 m (nominal: 4.3 m)	2 m = minimum pool level, resulting in heat exchanger uncovery. 5 m = maximum pool level, resulting in overfilling. Level value below 2 m implies restoration by tools not part of the present study and above 5 m implies malfunction of the water control system.	A significant confidence for the parameter to be kept at nominal value (resulting in a probability equal 0.8) is assumed. This is justified by the assumed operation of a system (the water level control) not considered in the present study.	
T_p Pool Temperature (I.C.)	The liquid temperature of the pool is Initial Condition affects the heat transfer capability of the heat sink.	280 - 368 K (nominal: 303 K)	280 K = minimum water temperature to avoid freezing. 368 K = maximum allowable water temperature at ambient pressure to avoid boiling. Initial temperature below 280 K is assumed non-realistic and temperature above 368 may imply undetected circulation across the IC and failure the water control system.	see above	

(+) *The (initial) value of each of the considered quantities may correspond to the end status of a system transient, when the IC operation is required.*

(*) *Probability values are peaked to the nominal value, and decrease gradually towards the minimum and maximum allowed values. No finalised study has been carried out to propose the relevant probability distributions. A 'fault-tree-analogous' approach can be used to this aim.*

	Choice justification(+)	Parameter range	Range justification	Probability values justification(*)	
X_1 Non-condensables fraction in RPV (I.C.)	The gas presence in the RPV affects the heat transfer capability of the Isolation Condenser, with a significant degradation of the heat transfer coefficient. This situation may occur following various transients from events like pin failure, clad overheating (hydrogen production,), failure in the actuation of sub-systems connected with the primary loop. The total mass of non-condensables in the RPV depends on the steam volume, hence it is connected with the design parameter L_1 .	0 – 1 fraction (nominal: 0)	0 = absence of non-condensables; 1 = presence of 100% of non-condensables in the steam/gas region of the RPV	(**) Probability values decrease as X_1 increase; the lower limit corresponds to the nominal value hence to the Max.prob.value. The presence of non-condensables has been judged generally more unlikely than for the design parameter cases, thus a major probability has been assigned to the lower limit condition ($P_{(X1=0)}$ = 0.719). The upper limit condition has been considered very unlikely to occur, thus a very low value has been assigned to it. Due to a significant sensitivity of the heat transfer coefficient to the non-condensable fraction, 7 discrete values have been considered for the investigation of the parameter effects.	*CRITICAL PARAMETERS*
X_2 Non-condensables fraction at Isolation Condenser tube bundle inlet (I.C.)	The gas presence in the IC line affects the heat transfer capability of the Isolation Condenser in the early stage in the transient, with a significant degradation of the heat transfer coefficient. This situation could occur for the concentration of gases in the upper zone of the draining pipe line, in the passive system loop. The presence of non-condensables could be caused e.g. by an incorrect maintenance of the passive system piping or by gas accumulation during the nominal operational life of the system.	0 – 1 fraction (nominal: 0)	0 = absence of non-condensables; 1 = presence of 100% of non-condensables at the inlet header of the Isolation Condenser tube bundle	Same comment as in (**), except for the upper limit probability, considered more likely to occur if compared with the complete filling of the RPV with non-condensables.	
θ Piping inclination at IC inlet (B.C.)	This Boundary Condition could affect the natural circulation flow rate. It corresponds to a wrongly welded pipe, with a downward inclination that could disturb the mixture flowing towards the IC inlet header. The consequences could be an alteration of the flow regime or a concentration of the gases, if present, in that zone of the circuit.	0 – 10 deg (nominal: 0 deg)	0 deg = corresponds to the design layout, i.e. horizontal piping; 10 deg = this limited value for the inclination angle has been assumed (10 deg) in order to consider it as not easily apparent and recognizable at a preliminary or sight check.	Probability values decrease as θ increase; the lower limit corresponds to the nominal value hence to the Max.prob.value. The occurrence of the nominal condition or even a small pipe inclination have been considered as the most probable conditions, hence most of the probability distribution (90%) lies within these values.	
C_2 Thermal losses at Passive System piping - draining side (B.C.))	This Boundary Condition could affect both the natural circulation flow rate and the heat transfer in the IC. tube bundle. A steam condensation for the mixture flowing upward towards the I.C., increases the system depressurization and decreases the driving density difference between the hot and cold legs.	0 - 100 kW (nominal: 5 kW)	0 kW = corresponds to the ideal case of no heat dissipation; 100 kW = corresponds to the worst condition for the heat losses of the fluid through the piping surface	A complete loss of thermal isolation for the piping has been judged very unlikely, hence a very low probability (P=1.e-4) has been assigned to the thermal losses upper limit. The nominal or pivot value is considered as the most realistic (P=0.8).	

	Choice justification(+)	Parameter range	Range justification	Probability values justification(*)	
L_2 Fluid Level into IC tube bundle (I.C.)	This Initial Condition affects the natural circulation flow rate in the early period of the transient, because of its direct influence on the driving force.	0 - 100 % (nominal: 100%)	0 % = the lower limit pertains to the worst situation, i.e. the complete lack of liquid into the heat exchanger tube bundle; 100 % = pertains to the nominal condition, i.e. the complete filling of the HX tubes	The largest confidence has been assigned to the nominal condition (P=0.8), i.e. the steam coming from the RPV condenses into the bundle and fill it completely. Marginal and equal probabilities are considered for the other two possible values. A reasonable cause to consider such an abnormal situation is the presence of non-condensables in the piping. Therefore a conditional probability holds for L_2 : a non-nominal value for the liquid level in the IC tube bundle is accepted only in the case a corresponding non-condensable fraction X_2 is present in the bundle inlet, otherwise the liquid fills the bundle completely.	*CRITICAL PARAMETERS (cont.ed)*
UL Undetected Leakage in the Passive System piping (B.C.)	This Boundary Condition directly affects both the flow rate and the global thermal capacity of the primary fluid. Extension of the activity might imply consideration of different positions for the 'undetected leakage'	0 - 10e-5 m^2 (nominal: 0 m^2)	0 m^2 = no leakage 10.e-5 m^2 = maximum area of undetected leakage	Probability values decrease as *UL* increase; the lower limit corresponds to the nominal value hence to the Max.prob.value. The keeping of a leakage undetected for the mission time of the system has been judged very unlikely to occur, thus a major probability has been assigned to the lower limit condition ($P_{(UL=0)}$ = 0.8899) and a very low value has been assigned to the upper limit (P= 1e-4).	
POV Uncomplete Valve opening (B.C.)	This boundary Condition represents the presence of an abnormal form loss in the pressure drops of the passive system, hence a direct degradation of the natural circulation.	1 - 100 % (nominal: 100%)	1 % = minimum percentage of valve .open; 100 % = complete opening of the valve. The considered range of variation may cover a number of uncertainties connected with the definition (in the code input deck) of the pressure loss coefficient. An extension of the activity might imply consideration of different positions for 'incomplete valve opening'.	Probability values increase as *POV* increase; the upper limit corresponds to the nominal value hence to the Max.prob.value. The keeping of a valve partially open has been judged unlikely to occur, thus a major probability has been assigned to the upper limit condition (P= 0.889) and the lowest value has been assigned to the lower limit (P= 1e-3).	

(+) *Same comment as in Table XX.* (*) *Same comment as in Table XX.*

Acknowledgement

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PROBABILISTIC SAFETY ASSESSMENT FOR THE ADVANCED SWR 1000 – EVALUATION OF THE SAFETY CONCEPT WITH ACTIVE AND PASSIVE SYSTEMS

Hartmut Schmaltz
FRAMATOME ANP GmbH (NGES4), 91058 Erlangen, Germany

Abstract

The safety concept of the SWR 1000, a new boiling water reactor product line, consists of passive safety features combined with active systems and aims to reduce releases from the plant to levels even further below internationally required limits.

An in-process probabilistic safety analysis matched continuously to the progress of design work has been performed for the purpose of quantitative follow-up of these aims and to ensure an optimum design. This probabilistic safety analysis was complemented by failure mode and effects analyses as well as sensitivity analyses. The aim of these analyses was to achieve a better understanding of the failure behaviour and the consequences of a failure of the new passive safety systems components and to prove that both the active and the passive systems are able to control an accident by themselves.

The PSA was performed considering internal events from power and shutdown states. The considered events were derived from proven PSA for German reference plants as well as international experiences.

The quantification of the PSA was performed with generic failure rates for the passive components, derived from comparable components. Due to the lack of adequate component data, a conservative CCF approach was taken. The sensitivity analysis was performed explicitly to show the influence of the data-variation of passive components on the PSA-results. The results of the sensitivity analyses prove that both the active and the passive systems are able to cope with accidents each. These results are confirming, that the active and passive systems are redundant as well as diverse. This is an important feature of the advanced SWR1000 concept with a high safety level.

Preface

The development the SWR1000 boiling water reactor (BWR), aimed to integrate and implement alternative solutions in the construction of items important to safety whilst retaining service-proven basic structures of systems and plant engineering approaches, and also optimizing operational and engineering features. The objective was to achieve a new quality in safety engineering and in the safety level of the plant. This is attained by the redundant use of active and passive systems and features, both for the prevention of severe accidents and of the mitigation the consequences thereof. The purpose of using active and passive systems was, within the bounds of cost-effectiveness, to achieve high reliability of the safety features so as to prevent a severe core melt event and to practically rule out releases outside the plant in excess of acceptable limits.

In order to achieve this degree of safety, ongoing probabilistic assessments are performed in the course of development to evaluate the integral safety level and the balance of the design so as to obtain early indications of possible design corrections and obtain confirmation of fulfillment of the ambitious safety objectives.

Probabilistic Safety Analysis

Stages of PSA-work

Since the beginning of the SWR1000 development, PSA was used to confirm the SWR1000 safety concept. In this first phase, the work concentrated on the evaluation of the passive features and their interaction with active systems, which presented a new challenge.

The failure potential and patterns of these passive features was examined and evaluated in the course of a failure mode and effects analysis (FMEA). The FMEA served the purpose of confirming the modeling and the quantitative evaluation of these passive systems.

In order to determine the effect of the passive systems on the quantitative safety level, the SWR 1000 PSA was followed by a sensitivity study to indicate the relevance of the various systems to the results and reveal uncertainties in the assessment.

In addition, supplementary analyses to assess events in shutdown states were performed. This shutdown PSA (SPSA) was performed under the coordination of the Finnish utility TVO, one of the partners in the project.

The purpose of the recently performed PSA-work was to exchange the generic reliability parameter for newer parameters from ZEDB [ZEDB] and T-Book 5 [TB] as well as to change the common cause failure model from Stochastic Reliability Analysis (SRA) Model to Alpha Factor Model which is mainly used in the Nordic countries. The common cause failure parameters for the Alpha factor model were provided by TVO.

Boundary conditions for the probabilistic safety analysis for power states

One focus of the development work on the SWR 1000 was the probabilistic safety analysis (PSA) which was performed to obtain a first assessment of the draft concept for orientation and further development of the safety concept. The associated event sequence evaluation models the interaction of the passive and active accident control features. The primary aim was to obtain a quantitative assessment of the preventive measures of the safety concept to avert damage states (Level 1 PSA) for the following purposes:

- To identify relevant challenges to the safety functions;
- To identify the safety and nonsafety systems for the control of accidents and to establish their thermal hydraulic success criteria;
- To perform a quantitative evaluation of the incidence of a damage state (DS) using event tree and fault tree analysis.

The evaluation of the frequencies of damage states doesn't still consider accident management measures neither recovery actions. Depending on the available period of grace and the plant's engineered features, operator actions (beyond design basis accident management) may be taken to the avert such a damage state. The results of the probabilistic safety assessment therefor can be used to identify such appropriate measures.

The present evaluation is based on in-plant events initiated during full power operation. The reference events for the PSA were derived from German BWR assessments with the following features:

– Major contribution to damage state incidence

– Exacting demands on system performance

– Relatively high event frequencies

By experience, the events, which were divided into three categories, constitute the most important for a PSA and the most challenging for plant design, cf. reference events in Tab. 1.

Table 1: Reference events for the SWR 1000 PSA

Reference events		**Frequencies of occurrence [1/a]**
Transients (T):		
Loss of preferred power	T1	6 E-2
Loss of main feedwater	T2	3 E-1
Loss of main heat sink	T3	5 E-1
Loss of main heat sink and main feedwater	T4	2 E-1
Failure to close of a safety and relief valve (SRV)	T5	1 E-1
Anticipated Transient without SCRAM	ATWS	2.8 E-6
Breaks outside containment (L):		
Break in the feedwater line outside containment	L1	3 E-3
Break in the main steam line outside containment	L2	2 E-3
Break in the reactor water cleanup system outside containment	L3	1 E-3
Loss of coolant accidents (K):		
Small break at the reactor pressure vessel ($A < 60\ cm^2$)	K1	2.7 E-3
Break in reactor pressure vessel bottom head	K2	2 E-4

The frequencies of occurrence of the analysed events were derived from German as well as Finnish operating experience or recent assessments (see Tab. 1).

The reliability data of the components originated from German and Nordic Nuclear Power Plants.

In the case of the passive components, reference data were taken from components fulfilling comparable functions, e.g., heat exchangers, check valves. To verify their failure and reliability data, failure mode and effects analyses (FMEA) were performed for each of the passive items to.

– In the analysis the passive components were assumed to be operable when challenged after testing of a number of experimental facilities had been successfully completed. These experiments involved trials to establish and verify the capacity of the passive components, see [KreM98].

– CCFs were postulated in redundant subsystems and evaluated; in view of the system diversity and different operating principles, cross-system CCFs were not postulated.

– Active initiation functions for the systems were introduced in the form of empirical default values. Components with passive initiation functions were modelled in detail, e.g. reactor scram from low RPV level, in order to capture and evaluate the effect of the new technology.

Failure mode and effects analysis

The failure rates of the components of the active and passive systems have an essentially influence on the PSA results. While it was possible to derive the failure rates of the active systems components from the operational experience at existing plants, the failure rates of the new passive components are subject to great uncertainties. For these components no operational experience is available.

To remedy this situation, a failure mode and effects analysis was performed for all new components of the passive safety. This estimation also served to substantiate the failure rates of these components.
The essential failure modes which were investigated were:

- Leakage within the system;
- Accumulation of inert gases;
- Clogging of tubes.

Leakages within the system are self reporting and normally do not cause a loss of the passive system function.

The accumulation of inert gases in the emergency condenser tubes or in the containment cooling condenser surrounding can affect the heat transfer, which reduces the efficiency of these safety functions or at the worst the function fails. This failure mode was considered in the design of these components and in the arrangement within the building.

The clogging of tubes by corrosion products can be ruled out by of the choice of suitable materials. The entrainment of foreign substances or particles is prevented by design measures.

Suitable test procedures will be planned to check the functional performance of these items of equipment.

Results of the quantitative evaluation

On the above basis an integral damage frequency of F_{DS} ~ 3 E-7/a was obtained in the PSA of the SWR 1000 for full power states. This very low value could be reached despite conservative initial conditions and parameters were used in the model (for example, a large CCF fraction was postulated for redundant passive systems so as to allow for uncertainties.

All in all, this figure reflects the assurance of the availability of a safety function by systems engineering provisions; in general, each safety function is fulfilled by several systems which are redundant and diverse to each other; furthermore, they are based on different, active and passive operating principles.

The integral and event-based damage frequencies are plotted in Fig. 1.

A comparison of the damage frequencies allows the following interpretation:

- All individual event related damage frequencies are $f_{DS} \leq$ 1 E-7/a;
- Loss-of-coolant accidents inside and outside the containment (K1 and K2; L1 – L3) have low occurrence frequencies, and nearly the entire spectrum of safety systems, whether active or passive, is available to provide sufficient core cooling.
- The transients are dominating the damage frequency with a contribution of about 99%.

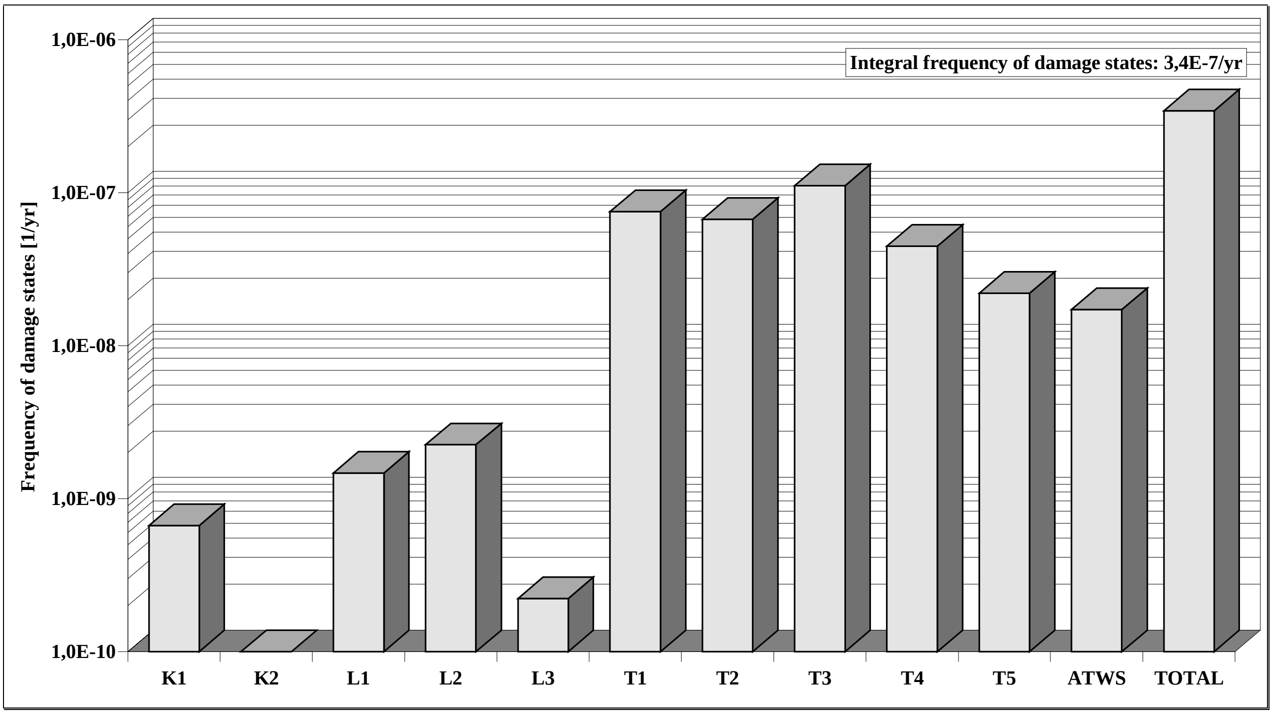

Fig. 1: Integral and event-based damage frequencies (see Tab. 1 for legend)

The events in the category of operational transients (T1 to T4) are all within a range in which differences are practically only attributable to the differences in frequency of occurrence; this is due to their generally relatively high frequencies of occurrence and the more or less identical system availability.

A stuck-open SRV (T5) results in a plant dynamic response consistent with a small-break LOCA but has a higher frequency.

In the case of anticipated transients without scram (ATWS) it is necessary to consider effects resulting from the frequencies of the transients, the high availability of the scram systems and the availability of the pressure limitation function (up to completion of ganged rod insertion or up to the boron injection takes effect). A relatively conservative common cause probability was assumed for the SRVs, although diverse valves are used. Four of the eight safety relief valves (SRV) work on the depressurize to trip principle, the other three on the pressurize to trip principle.

The contributions of the individual events are consistent with other BWRs in their relative distribution but at a very low integral damage frequency. The relative differences are to be assessed in the light of the low overall level and largely result from the different frequencies of occurrence.

The relative fractions of the system functions in the damage frequency are presented in Fig. 2 and the following conclusions can be drawn:

Containment heat removal is performed first of all by the two-train residual heat removal system. The flooding pool absorbs the heat input via the SRVs or emergency condensers and transfers it to the pressure suppression pool of the wetwell. Like the RHR system, the cooling chain connected to the wetwell is of two-train configuration and relatively complex on account of its various functions.

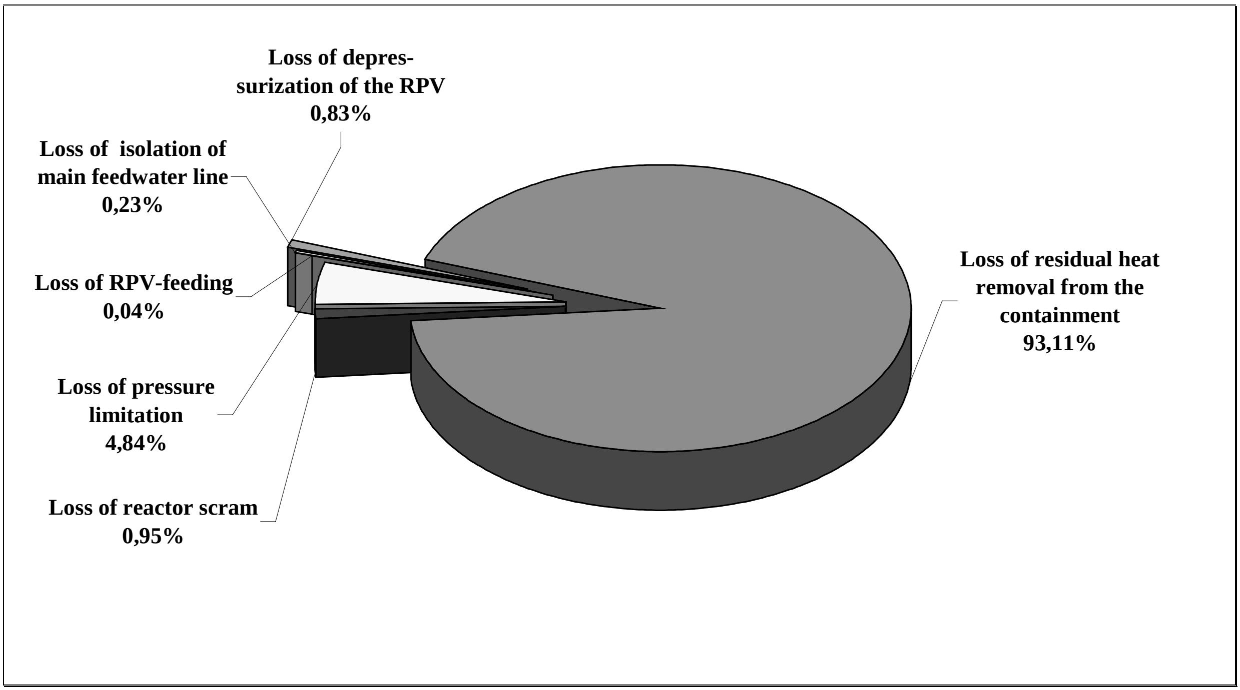

Fig. 2: Relative fractions of the system functions in the damage frequency

On loss of this cooling chain the passive containment cooling condensers provide heat removal, transporting the high from the containment to the dryer-separator storage pool from which it is released to the atmosphere by evaporation. However, in case of the failure of containment heat removal a relatively long grace period (> 2 days) is available for the operator to take remedial action or set up alternative system configurations in order to establish heat removal from the containment.

Compared with the pressure limitation and RPV coolant injection system functions, the degree of redundancy of these systems is not so great and this results as expected in a high relative fraction, dominated by CCF.

The SRVs, which in themselves have a high degree of redundancy and diversity, are backed up by the emergency condensers. Consequently the reliability of the pressure limiting function is high in events in which reactor scram is effected at an early stage.

However, during ATWS events the emergency condensers are not available at first, and the pressure limiting function becomes dominant as a result of the conservative success criterion assigned to it. The fraction for the pressure limiting function is almost exclusively determined by ATWS challenges.

Of minor significance is the fraction for the RPV coolant injection function. On a level reduction the emergency condensers start operating so that no further fluid is lost from the RPV. Additionally the feedwater system is available in many cases to provide injection at high pressures; furthermore, the recirculation pumps of the reactor water cleanup system are capable of providing sufficient injection.

If coolant injection is insufficiently effective in the HP range, the RPV pressure is reduced to allow LP injection by the residual heat removal system or by the passive core flooding system. Based on this large number of systems engineering possibilities it is understandable that the contribution of RPV injection to the damage frequency is small.

Operator actions for accident control are only necessary for initiation of shutdown cooling after loss of containment isolation in the event of breaks outside the containment. The relevance of such actions to the results is small in the light of the grace periods available in this case.

Given the large water inventory inside and outside the containment, a large grace period is available for alternative actions to be taken in the event of loss of heat removal. Initiation of safety functions by active (level monitors) and diverse passive means (pulse transmitters) practically rules out damage states in the short term.

CCFs are theoretically possible in redundant systems and are modeled, separate for active and for passive systems. In the present analysis a relatively conservative model was chosen for passive systems so as to make allowance for uncertainties in the evaluation of the new components. Since the contribution of the passive systems to the result is relatively important, the integral CCF fraction of more than 99 % is absolutely high.

In addition to the PSA for power operation an analysis of events during the shutdown state of the plant was performed under the coordination of the Finnish utility TVO, one of the partners in the SWR 1000 project. The methodical approach was orientated on the basis of the methodology of the shutdown PSA of the Finnish plants Olkiluoto 1 and 2. The damage frequency was obtained as $F_{SPSA,DS}$ ~ 6 E-7/refueling outage. This frequency is a result of optimization measures in systems for initiating coolant injection and surely can be reduced in the course of further development when detailed information on the refueling outage sequence is known and conservative assumptions can be made less penalising. The contribution of the different systems and the CCF on the result yields in a comparable picture like the results of the PSA for power operation.

Sensitivity Analysis for Active and Passive Components

The evaluation capabilities of PSA (for power states) were used to determine the sensitivity of the active and passive components and facilities with respect to effects on the results, as reflected in the damage frequency. For this purpose, the unavailabilities (UA) per challenge (ch) of the active systems under analysis were varied between 1 E-3 < UA < 1/ch and for passive systems between 1 E-6 < UA < 1/ch. This simulates a variation in component availability up to total loss of the component (UA = 1/ch).

This simulation was performed for the active residual heat removal system, the HP injection system, the emergency condenser, the containment cooling condenser and passive core flooding. The result obtained is the damage frequency as a function of the items' unavailability, see Fig. 3 and 4. The base values used in the PSA for the analysed systems are entered as vertical lines.

The most important factor is the containment cooling condenser system. A degradation of system reliability is not reflected directly in the overall result. Above a nonavailability of N > 1 E-3/ch a damage frequency of f_{HS} > 1 E-6/a is predicted. Particular attention is to be devoted to design and operation of this system in order to maintain the high safety level.

However, the result also demonstrates that although the integral damage frequency rises to a value of about f_{HS} > 6 E-4/a on total loss of the containment cooling condenser system, the remaining active systems still provide a basic level of safety.

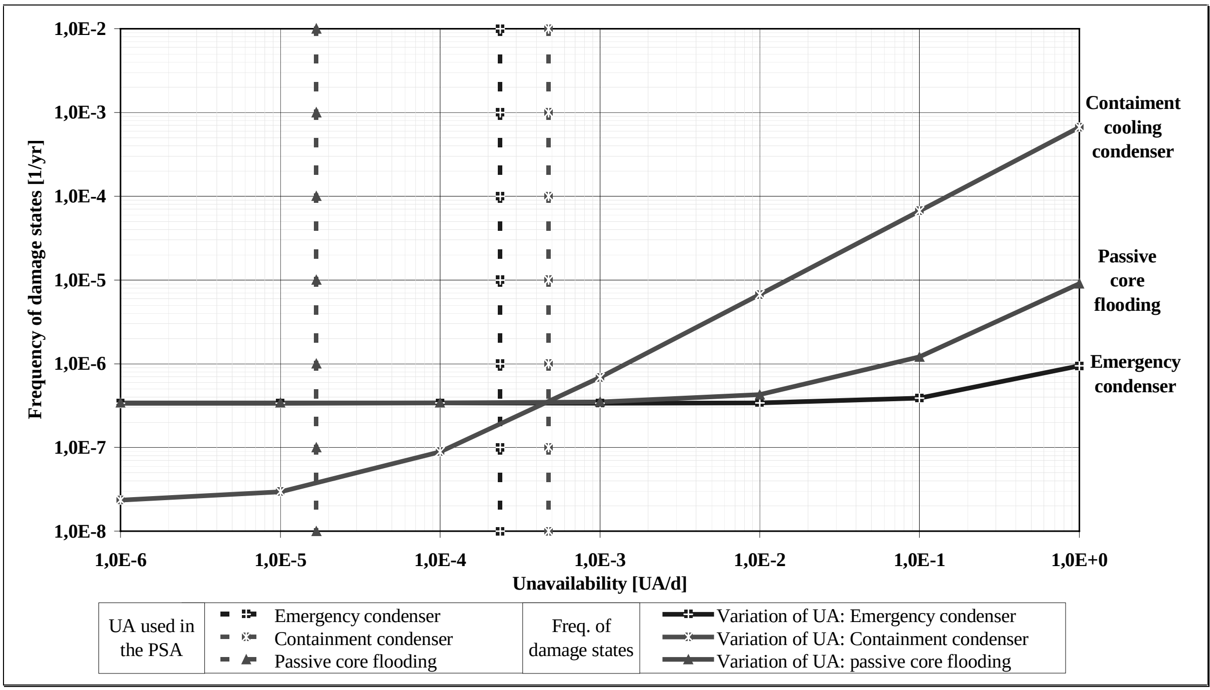

Fig. 3: Damage frequency as a function of the unavailability of the passive systems

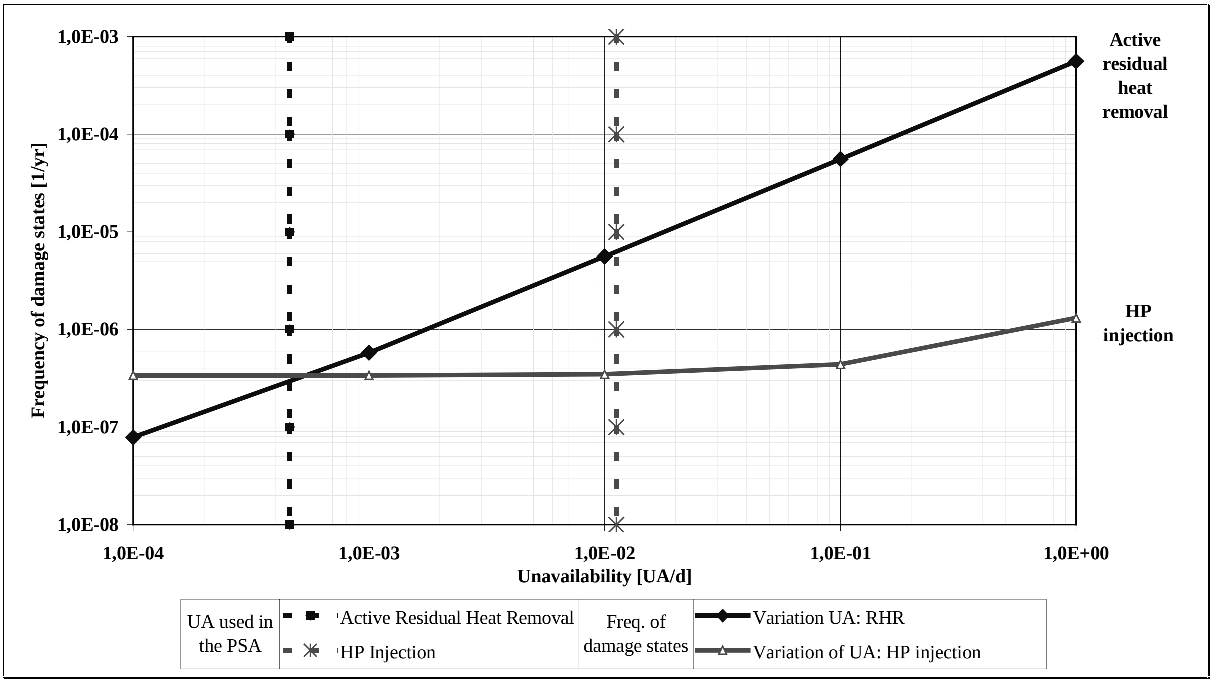

Fig. 4: Damage frequency as a function of the unavailability of active systems

In this case, the damage frequency is essentially determined by the loss of the two-train RHR chain. However, a relatively long grace period (> 2 days) is available for the operator to take remedial action or set up alternative system configurations in order to establish heat removal from the containment.

The passive core flooding system only becomes relevant to the result above a system nonavailability of N > 1 E-2/ch., which is technically unrealistic. At this point the effect becomes more significant; until then other system functions dominate. All in all, the safety concept is not sensitive to variations in this system. The evaluation of this analysis reveals that the safety concept is designed in such a way that a damage state can be prevented by active means (incl. SRVs) or by the passive systems alone.

Safety Concept Assessment

The probabilistic evaluations are performed in an ongoing process concurrent with development. They supply valuable qualitative and quantitative information on the significance of safety-related items, indicate the balance of the design and provide quantitative data for assessing the safety level. Conspicuously dominant elements in the systems engineering were corrected at an early stage.

Experience-based reference events were used in the probabilistic safety analysis (PSA) to obtain an assessment of the design and an estimate of the safety level.

The damage frequency was obtained as F_{DS} ~ 3 E-7/a with the parameters adopted for the analysis. This value is evidence of the excellent level of reliability of the safety system and confirms the quality of the safety concept based on active systems backed up by functionally diverse passive features. It must be noted, though, that the low absolute value is of necessity associated with uncertainties. This is particularly true with respect to the completeness of the event spectrum and the evaluation of the passive systems. However, the functions of the latter have been tested comprehensively and were consciously evaluated very conservatively in the PSA.

The sensitivity analysis conducted specifically for the passive safety features demonstrates their influence on the overall safety level. This analysis confirms the underlying safety concept of the SWR 1000. The active and passive systems are mutually complementary in their functions for accident control; thanks to their functional independence, each is capable of controlling accidents alone. Their interaction results in the high safety level, thus fulfilling the design goal.

The quality of the safety concept is further underlined by the long grace periods after loss of safety features until severe core degradation can occur, let alone until radioactive releases from the plant can take place (several days). These response times allow alternative measures to be taken to restore the plant to a controlled state.

The assessment will be updated and supplemented in the course of further development. In addition, recovery and accident management measures will be integrated in the PSA-model. Current insights suggest that the project goal of practically ruling out severe core degradation can be achieved and that internationally aired quantitative safety objectives for new designs will be met with considerable margin.

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