NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

PROCEEDINGS FROM INTERNATIONAL WORKSHOP ON FIRE RISK ASSESSMENT

Helsinki, Finland, 29 June - 2 July 1999
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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.

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The 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 the programme of work. It also reviews the state of knowledge on selected topics on 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 on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects 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 meetings.

The greater part of the CSNI’s current programme is concerned with the 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 nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the 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.

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The opinions expressed and the arguments employed in this document are the responsibility of the authors and do not necessarily represent those of the OECD.

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ABSTRACT

This workshop will give survey on the maturity of different physical fire analysis methods used in fire risk assessments and recommendations for further development and co-operation. This document provides copies of the papers presented at the workshop.
The SESAR reports have emphasised the impact of high temperature of fires and the spreading of smoke on electrical equipment and electronics as important issues in fire risk assessments. The fact that fire risk analysis has become an integral part of PSA and the fires have been recognised as one of the major contributors to risk of nuclear power plants was also well reflected. Accordingly SESAR reports recommended CSNI to tackle the issues in fire risk assessment.

In 1996 PWG5 submitted CSNI a proposal to undertake such a study and the task 97-3 was initiated in 1997. Accordingly, the prime focus of the task was to concentrate on some special issues such as fire simulation, fire spreading, and impact of smoke and heat on instrumentation electronics.

Using responses to a questionnaire and the results of this workshop a state-of-the-art Report was produced [Ref: NEA/CSNI/R(99)27] by the PWG5 task group and issued in February 2000.

One paper is missing from this compilation. The editor wishes to apologise to Mr. Edward Connell regarding this omission. His original and all copies were errantly lost and could not be recovered.

Members of the Workshop Organising Committee and Task Force included:

**MR. REINO VIROLAINEN, STUK (CHAIRMAN)**

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Instrumental in both organising the task work and the workshop was Mr. Virolainen, the Chairman. Also much appreciation is given to the entire staff of STUK, Finnish Centre for Radiation and Nuclear Safety for the excellent arrangements made in hosting the workshop.
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INTRODUCTION

Numerous fire PSAs (probabilistic safety assessments) have shown that fire can be a major contributor to nuclear power plant risk. However, there are considerable uncertainties in the results of these assessments, due to significant gaps in current abilities to perform realistic assessments. These gaps involve multiple aspects of fire PSA, including the estimation of the probability of important fire scenarios, the modelling of fire growth and suppression, the prediction of fire-induced damage to equipment (including the effects of smoke), and the treatment of plant and operator responses to the fire.

In response to recommendations made by the CSNI, PWG5 established a Task Group to review the present status and maturity of current methods used in fire risk assessments for operating nuclear power plants. The Task Group issued a questionnaire in May 1997 to all nuclear power generating OECD countries. The prime focus of the questionnaire (see Appendix A) was on a number of important issues in fire PSA:

- fire PSA methodology and applications
- fire simulation codes
- ignition and damageability data
- modelling of fire spread on cables or other equipment
- modelling of smoke production and spread
- impact of smoke and heat on instrumentation, electronics, or other electrical equipment
- impact of actual cable fires on safety systems.

The questionnaire requested specific information on these topics (e.g., computer codes used in fire PSAs, the physical parameters used to model ignition).

Responses to the questionnaire were provided by Finland, France, Germany, Hungary, Japan, Spain, Switzerland, United Kingdom, and the USA. Responses to the report and this workshop were utilised by the task group to produce a state-of-the-art report. The report entitled “Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat On Instrumentation Electronics”, [Ref: NEA/CSNI/R(99)27] was issued in February 2000 and in part summarises the questionnaire responses and thereby: a) provides a perspective on the current fire PSA state of the art (SOAR) with respect to the issues listed above, and b) provides numerous references for more detailed information regarding these issues.

OBJECTIVE

The main purpose of the Workshop is to exchange information on fire risk analysis in the nuclear industry. The main objectives of the workshop were:

- to provide a forum to review and provide insights on fire risk assessment
- to review state of art of fire tests, simulation and modelling
- to survey future needs and development.
BACKGROUND

The CSNI believes that an essential factor in ensuring the safety of nuclear installations is the continuing exchange and analysis of technical information and data. To facilitate this exchange, the Committee has established Working Groups. PWG5 was established to deal with the technology and methods for identifying contributors to risk and assessing their importance.

Fire risk analysis has become an integral part of PSA and the fires have been recognised as one of the major contributors to risk of nuclear power plants. The main objective of the PWG5 task group is to perform a state-of-the-art review of the most essential methods and practices vital to fire risk analysis. In order to provide a clear picture of the present practices and analysis methods, it is necessary to make a survey of the general framework of fire risk analysis.

The fire risk analysis has not yet achieved the same level of methodological maturity as is typical in some other disciplines of PSA. The fire simulation methods which pose an important role in fire risk analysis are still largely questioned because of large uncertainties typically associated with the quantitative estimates of fire risks. The uncertainties embedded in fire risk analysis are of dual nature. On one hand there is a shortage of basic data on fire ignition and parameters prevailing in fire spreading models, on the other hand there is a lack of knowledge and experimental data on vulnerability of equipment of different sort - electronic, electrical and mechanical.

Hence a lot of effort is still needed to upgrade the fire risk assessment methodology with focus on the physical fire analysis and on the interface between the probabilistic and physical models. This workshop will give survey on the maturity of different physical fire analysis methods used in fire risk assessments and recommendations for further development and co-operation.

SCOPE, CONTENT AND OUTLINE

The Workshop will address the following main topics:

- National fire research programmes
- Fire risk assessment applications
- Experimental fire research
- Fire simulation and fire models
- Future needs and development.

FINAL PANEL AND CLOSING

Prior to closing the workshop a selected group of experts formed a panel and responded to open questions from the floor. This panel was chaired by Mr. Joseph A. Murphy, Chairman of PWG5. Mr. Murphy focused on many of the main elements brought forth in the opening presentation by Mr. Thadani on key issues in Fire Risk Assessment. Participants and panellists had a opportunity to pose and answer important questions regarding the current state-of-the-art. It was apparent that much progress has been achieved, but more work is required.
ORGANISING COMMITTEE

The Organising Committee will examine those abstracts submitted before the deadline, select those best suited to achieve balance and coverage, organise the sessions and draw up the final program for the workshop. It will also, if the number of proposed participants exceeds the capacity of the facilities, allocate the available spaces. Its members are:

Mr. Reino Virolainen, STUK (Chairman)

Jouko Marttila, STUK, Finland
István Kelemen, ETV-ERÖTERV, Hungary
Sándor Czákó, VEIKI, Hungary
David Hamblen, Magnox Electric, UK
Előd Holló, VEIKI, Hungary
Tae Woon Kim, KAERI, Korea
Mamoru Fukuda, NUPEC, Japan

Javier Yllera, CSN, Spain
Francois Bonneval, IPSN, France
Rémy Bertrand, IPSN, France
Marina Röwekamp, GRS, Germany
Nathan Siu, NRC, US
Vijay Raina, Ontario Hydro, Canada
WORKSHOP PROGRAMME

STUK, FINNISH CENTRE FOR RADIATION AND NUCLEAR SAFETY

29 JUNE TO 1 JULY 2000

OPENING SESSION

☐ WELCOMING REMARKS
   Prof. Jukka Laaksonen, Director General, STUK

☐ OPENING REMARKS
   Mr. Barry Kaufer, OECD/NEA

☐ KEY ISSUES IN FIRE RISK ASSESSMENT
   Mr. Ashok Thadani, Director Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (NRC)

SESSION I - NATIONAL AND INTERNATIONAL FIRE RESEARCH AND DEVELOPMENT
   Chair: Mr. Edward O. Connell, Nuclear Regulatory Commission, United States

☐ The U.S. Nuclear Regulatory Commission’s Fire Risk Research Programme - An Overview -
   Dr. Nathan Siu and Mr. Hugh Woods, U.S. Nuclear Regulatory Commission (NRC), United States

☐ A General Overview of the IPSN Fire Research – MM. J.-M. Such, R. Gonzalez, B. Tourniaire,
   L. Audouin, C. Casselman, L. Rigollet, and W. Le Saux, Institut de Protection et de Sûreté
   Nucléaire (IPSN), Département de Recherches en Sécurité, CE de Cadarache, France

☐ Development of Probabilistic Safety Assessment Methodology for Fire Events in Candu Plants -
   Mr. George How Pak Hing and Mr. A. H. Stretch, Atomic Energy of Canada Ltd, Canada

☐ Fire PSA: Applications and Insights - Dr. Mardy Kazarians, Kazarians & Associates, United States

☐ Fire Risk Assessment in Germany: Regulatory Guidance and Applications - Dr. Heinz-Peter Berg,
   Bundesamt fur Strahlenschutz (BfS), Germany and Dr. Marina Röwekamp, Gesellschaft für
   Anlagen- und Reaktorsicherheit mbH (GRS), Germany

SESSION II - FIRE RISK ASSESSMENT AND APPLICATION I
   Chair: Dr. Jean-Marie Lanore, Institut de Protection et de Sûreté Nucléaire (IPSN), France

☐ Probabilistic Study of Fire Scenarios - Mme. Myriam Chausnard, R. Bertrand, F. Bonneval and
   J.M. Mattéi, Institut de Protection et de Sûreté Nucléaire (IPSN), France
SESSION III - FIRE RISK ASSESSMENT AND APPLICATION II
Chair: Mr. David J. Hamblen, Magnox Electric plc, United Kingdom

- Probabilistic Risk Assessment of Fire Safety Design Alternatives - Ms. Lotta Andersson, Sycon Energikonsult AB, Sweden
- Preliminary Study of Fire Event PSA for BWR Plants - Mr. Toshiyuki Zama, Tokyo Electric Power Company (TEPCO), Japan and Mr. Kazunori Hashimoto, (Toshiba Corporation), Japan
- Simulating of Frequency Firing’s on Turbo-generators at Ukrainian NPP - Mrs. Olena Babich and Mr. Sergei Azarov, Ukrainian Academy of Sciences, Ukraine

SESSION IV - EXPERIMENTAL FIRE RESEARCH I
Chair: Stephen P. Nowlen, Sandia National Laboratories, United States

- Full Scale Fire Experiments on Electronic Cabinets - Olavi Keski-Rahkonen, VTT Building Technology, Finland
- Electrical Cable Fire Tests at EdF - M. Bernard Gautier and M. Emmanuel Thibert, Electricté de France (EdF), France
- Comparison of the Burning Behaviour of Electric Cables with Intumescent Coating in Different Test Methods - Dipl.-Phys. Jürgen Will and Dipl.-Phys. Dietmar Hosser, Institut für Baustoffe, Massivbau und Brandschutz (iBMB), Germany

SESSION V - FIRE SIMULATION AND MODELS
Chair: Dr. Marina Röwekamp, Gesellschaft für Anlagen- und Reaktorsicherheit mbH (GRS), Germany

- Modelling of Fire Analysis in Modern Swedish PSAs - Mr. Peter Jacobsson and Mr. Fredrik Jörud, Sycon Energikonsult AB, Sweden
- Spread Simulation of Horizontal Cable Tray Fire - Mr. Risto Huhtanen, VTT Energy, Finland
- Fire Modelling in IPSN Computer Codes - Mme Chantal Casselman, L. Audouin and B. Tourniaire Institut de Protection et de Sûreté Nucléaire (IPSN), Département de Recherches en Sécurité, CE de Cadarache, France
- Development of Fire Severity Factor - Mr. Takahashi Mukae, Mr. Mamoru Fukuda and Mr. Mitsumasa Hirano, Institute of Nuclear Safety (INS), Nuclear Power Engineering Corporation (NUPEC), Japan
- Fire Safety Assessments in Ontario Power Generation - Joan Higgs, Ontario Power Generation, Canada

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**SESSION VI - EXPERIMENTAL FIRE RESEARCH II**

Chair: Dr. Olavi Keski-Rahkonen, VTT Building Technology, Finland

- Results of Experimental Fire Research - Steven P. Nowlen, Sandia National Laboratories (SNL), United States
- Failure Distribution in Instrumentation Cables in Fire - Mr. Johan Mangs, VTT Building Technology, Finland, Dr. Olavi Keski-Rahkonen, VTT Building Technology, Finland, Mr. Hannu Hossi, VTT Automation, Finland and Mr. Arto Salminen, VTT Automation, Finland
- Behaviour of High Efficiency Particulate Air Filters in Case of Fire - M. Jean-Claude Laborde, M. A. Briand and V. M. Mocho Département de Prévention et d’Etude des Accidents, Saclay, IPSN, France

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**SESSION VII - FIRE ANALYSIS, MODELLING**

Chair: Rémy Bertrand, Institut de Protection et de Sûreté Nucléaire (IPSN), France

- Leningrad NPP Unit 3 Deterministic Fire Hazard Analysis Methodology and its Preliminary Results - Mr. Aleksandre Yepikhine, Mr. A. M. Sapozhnikov, Leningrad NPP, Russia and Dr. Antti Norta Fortum Engineering, Ltd., Finland
- Operating Issues for French NPPs In Severe Fire Situations - M. Maurice Kaercher, M. Jean Paul Chatry, Electricité de France (EdF) - SEPTEN, France
SESSION VIII - FIRE RISK ASSESSMENT AND APPLICATIONS III
Chair: Dr. Javier Yllera, Consejo de Seguridad Nuclear (CSN), Spain

- Fire Risk Analysis of United Kingdom Nuclear Chemical Plants A Practical Approach - Mr. Geoffrey Arrowsmith, GMIFE QSFPO, Health & Safety Executive (HSE), United Kingdom
  Mr. Steven Greenwood, Bsc Hon Mech Eng AMI Mech E MIOSH, British Nuclear Fuels Ltd. (BNFL), United Kingdom

- Olkiluoto NPP Fire Risk Analysis - Mr. Mika Yli-Kauhaluoma, Mr. R. Himanen and Mr. K. Taivainen Teollisuuden Voima Oy (TVO), Finland

- PSA Study for An Exemplary Plant Location of a German PWR Built to Earlier Standards - Dr. Marina Röwekamp, and Dr. Heinz Liemersdorf Gesellschaft für Anlagen- und Reaktorsicherheit mbH (GRS), Germany

SESSION IX - FUTURE NEEDS AND DEVELOPMENT
Chair: Professor Matti Kokkala, VTT Building Technology

- Fire PRA Needs - Regulators Perspective - Mr. Edward A. Connell, Nuclear Regulatory Commission, United States. (Missing presentation)

- Fire PRA Needs from the Utility Fire Protection Programme Manager and Engineering Consultant Perspectives - Franklin D. Garrett, P.E., Department Leader, Emergency Services Programs, Palo Verde Nuclear Generating Station, United States and Elizabeth Kleinsorg, P.E., Vice-President, Fire Protection & Hazards Analysis, Duke Engineering & Services, United States

- Outline of a Performance-Based Fire Safety Design Method for Buildings in Japan - Professor Takeyoshi Tanaka, Disaster Prevention Research Institute, Kyoto University, Japan

- A Reference Framework for the Development and Documentation Human Reliability Analyses for Fire PSAs - Dr. Javier Yllera, Consejo de Seguridad Nuclear (CSN), Spain
WORKSHOP PRESENTATIONS

The following sections of this report provide copies of the papers presented at the workshop. One paper is missing from this compilation. The editor wishes to apologise to Mr. Edward Connell regarding this omission. His original and all copies were errantly lost and could not be recovered.
Opening Session

- KEY ISSUES IN FIRE RISK ASSESSMENT - Mr. Ashok Thadani, Director, Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (NRC)
Key Issues in Fire Risk Assessment

A. Thadani, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Introduction

Good morning. I would like to add my personal welcome to you, the participants in this CSNI/PWG5 International Workshop on Fire Risk Assessment. Speaking on behalf of the U.S. Nuclear Regulatory Commission (NRC) as well as CSNI, I can tell you that your participation is greatly appreciated. Without your expert knowledge and your willingness to share this knowledge, the increasing challenges posed by our changing regulatory environments cannot be fully addressed.

One key challenge involves the push for government agencies like NRC to become more efficient, to make better use of available resources. Our strategic planning, budgeting, and performance management processes have to become, and are becoming, increasingly businesslike to meet this challenge. Other challenges arise because of changes in the regulated industry, including deregulation and restructuring in response to economic pressures. We need to be able to anticipate these changes and appropriately adjust our processes to effectively ensure protection of the public health and safety.

To meet these challenges, we need tools to help us make better, more rational, and more effective decisions. Risk assessment, in general, and fire risk assessment, in particular, can provide such tools. I believe that the time you will spend over the next three days discussing the methods and data used in fire risk assessment will tremendously benefit the development of
the technical infrastructure needed to support the increased and improved use of risk information in our decision making.

The title of my talk is “Key Issues in Fire Risk Assessment.” As many of you know, the famous, or perhaps infamous, Browns Ferry cable fire happened in 1975, almost 25 years ago. (I believe that Joe Murphy will give a talk on our first attempts to assess the risk implications of that fire later in this workshop.) That event represented our closest “near miss” to a core damage accident at a commercial nuclear power plant prior to the Three Mile Island accident. For those of you not familiar with the event, summary discussions can be found in a number of places, including a 1976 NRC report [1]. Suffice it to say that it was a major wake-up call for the NRC and industry regarding the nuclear safety hazard posed by fires. Since Browns Ferry, there have been significant changes regarding the design and regulatory approaches to nuclear power plant fire protection, especially regarding the physical separation of safety-related divisions. For example, NRC used the lessons from Browns Ferry to develop a deterministic, prescriptive fire protection regulation known as Appendix R to 10 CFR 50 (“Appendix R” for short). This regulation requires, among other things, that one train of equipment required to achieve hot shutdown be maintained free of fire damage from a single fire. Nevertheless, serious international fires since Browns Ferry (for example, at Vandellos and Narora) remind us that fires still represent an important hazard.

The first detailed studies of fire risk for commercial nuclear power plants were for the Zion and Indian Point plants, completed in 1981 and 1982, respectively. These studies, which were part of larger utility-sponsored PSA studies aimed at determining if additional accident mitigation features (e.g., core-catchers, filtered and vented containments) were needed for these plants, employed a fire risk assessment methodology that looked at all of the fire protection defense-in-
depth barriers: fire prevention, fire detection and suppression, and fire mitigation. The methodology, whose development was sponsored by NRC, combined fire occurrence statistics, fire modeling, fire suppression statistics, and traditional PSA event sequence modeling; it was aimed at identifying potentially significant fire-initiated accident scenarios and deriving estimates of associated plant damage state (including core damage) frequencies including uncertainties [2]. Application of the methodology to Zion and Indian Point led to what, in hindsight, should probably not have been a surprise; fire could be an important contributor to risk. Extensive reviews of the studies identified a number of potential weak points (e.g., the treatment of fire-induced damage outside of the plume, the treatment of certain operator recovery actions), but did not contradict the basic conclusions [3].

This workshop is being held some 17 years after the completion of the Zion and Indian Point studies, after the completion of numerous other studies (a number of U.S. examples are listed in Table 1), and 10 years after a significant NRC effort to identify unaddressed fire risk issues, the “Fire Risk Scoping Study,” which was completed in 1989 [4]. In some ways, things have not changed very much. I observe that fire can apparently still be an important contributor to risk (this observation is reinforced by the results of our recent Individual Plant Examinations of External Events -- IPEEs -- which I will talk about later), but we have some doubts because of the analysis uncertainties. I also observe that the fire risk assessment methodology being used in current studies has essentially the same structure as that used in the Zion and Indian Point Studies, and that many of the methodological gaps and weaknesses raised in the Zion and Indian Point reviews and the Fire Risk Scoping Study have yet to be completely addressed. What has changed in the last few years, at least in the U.S., is the interest level of regulators and industry in the use of fire risk assessment results and insights to support fire protection issue resolution and improvements in nuclear power plant operations and safety. This interest
has increased significantly, despite the recognized methodological gaps and weaknesses:
These observations lead me to the two issues which are the subject of the rest of my talk:

1) the need for improvement in fire risk assessment methods, tools, and data to address identified technical gaps and weaknesses; and

2) the need to identify applications where current fire risk assessment technology, despite its warts and bumps, can be confidently used to support decision making.

I believe that this workshop comes at an opportune moment and, therefore, represents an important opportunity for CSNI and its member organizations to begin the resolution of these issues. Before I explain why, I think it is useful to provide some background regarding NRC's views on how risk information should be used in decision making, both general and fire protection specific. These views directly affect our perspectives on the notions of risk assessment improvements and applications.

**On Risk-Informed Decision Making**

As we have stated in our Probabilistic Risk Assessment (PRA) Policy Statement [5] and many subsequent documents, the NRC is sincerely interested in increasing the use of risk results and insights to the extent warranted by the state of the art. These results and insights will be used in a *risk-informed* manner. In other words, they will be considered *together with other factors* to deal with issues in a manner commensurate with their importance to public health and safety [6]. Let me emphasize the explicit consideration of other factors; this distinguishes the risk-informed approach from a risk-based approach. Of course, we expect that a risk-informed
approach will make better use of available resources; this is the reason for NRC’s re-visiting of our basic approach to regulation. We also expect that the risk-informed approach will lead to real safety improvements.

It is useful to mention that the NRC has developed a risk-informed framework for supporting licensee requests for changes to a plant’s licensing basis. This framework, described in Regulatory Guide (RG) 1.174 [7], provides an acceptable approach for combining risk insights with traditional engineering evaluations in an integrated decision making process. This approach requires consideration of impacts of a proposed change on defense-in-depth and safety margins, as well as on risk. The RG 1.174 framework covers only one regulatory application of risk information, and it is not focused on the needs of any one particular issue (e.g., fire protection). However, its development has provided key groundwork for dealing with a number of issues (for example, what is considered to be a small increase, what impacts besides risk need to be considered) that will need to be addressed in other applications.

**Current NRC Fire Safety Activities**

As I indicated earlier, the NRC believes that fire can be an important contributor to risk. This belief is based upon experience from past events (including but not limited to Browns Ferry), numerous studies conducted by industry and the NRC, and our knowledge of current nuclear power plant fire protection design approaches and programs. As part of our efforts to increase the use of risk results and insights, therefore, the NRC is pursuing a number of activities to better understand plant-specific fire risk contributions and make better use of this understanding. Non-research related activities include: the IPEEE program; the development of methods to assess the risk significance of fire safety inspection findings; and the development
of a risk-informed, performance-based standard for fire protection (NFPA 805). I'll mention our research activities later in this talk.

The IPEEE program involves licensee self-examinations of their plants for external event vulnerabilities. The objectives include the development of an improved understanding of the likelihood of core damage and fission product releases due to external events (including fires) and the identification of cost-effective improvements. The NRC’s Office of Nuclear Regulatory Research (RES) has recently completed the first round of reviews for all 70 licensee submittals (some submittals cover multiple units at a single site), worked with the Electric Power Research Institute (EPRI) on approaches for resolving a number of key fire risk assessment methodological issues identified during the review (see Table 2), and performed a scoping assessment of the fire risk impact of exemptions to specific provisions of Appendix R at nine selected plants.

The latter assessment, which was based largely on available documentation for the IPEEE submittals, showed that the vast majority of the 169 exemptions reviewed are not risk significant. Only five were found to be potentially risk significant. In other words, had these five exemptions not been granted, and had the plant not selected an alternative method to achieve compliance with NRC fire protection requirements, the estimated fire CDF in some or all of the impacted areas would likely have been reduced by at least $10^5$/ry. The impact of 21 exemptions could not be reliably determined given the available information and the current fire risk assessment state of the art. The assessment also showed that a simple count of exemptions at a given plant provides little or no direct insight into the potential risk significance of exemptions at that plant.
Regarding the risk significance of inspection findings, the Office of Nuclear Reactor Regulation is currently developing an analytical tool to assist inspectors in determining if a particular finding is potentially significant. The tool uses a structure that can be explained in terms of a simplified fire event tree (whose top events address fire initiation, detection and suppression, barriers, and safe shutdown) and employs weights that are based on fire risk assessment generic results (e.g., for the effectiveness of automatic suppression). Its use is similar to that for the Accident Sequence Precursor models being developed by the NRC for assessing the conditional core damage probability associated with potentially significant plant conditions.

Regarding NFPA 805, NRC staff are participating in the development of this consensus standard, which is being done under the auspices of the National Fire Protection Association (NFPA). The standard, whose final draft will be completed during September 1999 and issued for public comment on January 21, 2000, describes how deterministic and probabilistic analyses can be used to determine if proposed changes to an existing plant's fire protection program meet specified performance criteria. The standard is currently envisioned to use risk information in a manner consistent with that described in RG 1.174. It is also likely to include a description of requirements for an acceptable fire risk assessment, but not the specific analytical methods or procedures. In other words, it will specify the “what’s” but not the “how’s” for fire risk assessment. The NRC believes that this standard, when completed, will provide flexibility to the licensees when making changes to their existing reactor fire protection programs by allowing a greater use of risk insights; our current plans are to begin associated rulemaking after the final draft is issued [8].
Now that I've outlined NRC's approach to the use of risk information and some of its related activities in the area of fire safety, let me return to why this workshop, whose focus is on the exchange of fire risk analysis information, is important to the NRC and CSNI.

**Workshop Roles**

It is clear that the use of risk insights in regulatory decision making, indeed in any decision making, must be tempered by the limitations and uncertainties in the technology (i.e., the methods, tools, and data) used to develop these insights. For example, NRC's PRA Policy Statement uses the qualification "to the extent warranted by the state of the art." It is also clear that the current risk assessment state of the art for fire-initiated events is not as mature as that for a number of other potential accident initiators, such as most of the so-called internal events. Thus, it is important to improve the methods, tools and data in areas where there are clear gaps or weaknesses. Not only do these gaps and weaknesses represent potential sources of inaccuracy, they also reduce the confidence of the decision makers who wish to use the results of the fire risk assessment. In a risk-informed environment, where risk considerations are integrated with other considerations in a non-prescriptive manner, such a loss of confidence can significantly impact the extent to which fire risk results and insights are actually used.

You will hear an NRC paper later this morning describing the research efforts we are currently undertaking to improve the fire risk assessment technology. These efforts are based on a systematic identification and prioritization of potential research issues and cover a broad range of subjects, including fire-induced circuit failures, data for fire modeling, lessons from historical events, uncertainty analysis, and fire frequency estimation. Of course, this only represents our current views. Other papers in this workshop will provide alternate viewpoints concerning
where improvements are needed. The resulting discussion will certainly be important to all CSNI members. First, they will help identify research areas and topics where redundant efforts are not useful. Second, they should provide valuable information to the NFPA 805 standards development activity. As I mentioned earlier, the final draft version of NFPA 805 will be prepared by the responsible technical committee during September, 1999 and the proposed standard will be issued January 21, 2000. Comments on the proposed standard will be accepted through March 31, 2000. Thus, feedback from this workshop to the technical committee (many of the principals are participating in this workshop) should be especially timely.

I also believe that this workshop can provide something in addition to the identification of gaps and weaknesses. I believe it can initiate the dialog that will eventually tell decision makers where (for what situations or under what conditions) fire risk assessment insights can be confidently used, as well as where they can’t. Not only is this dialog crucial for the successful development of risk-informed fire protection standards, it is extremely important to those of us who are decision makers regarding research directions and activities, as well as those who are decision makers regarding plant design and operations. Note that it addresses an important facet of a question we commonly are asked when starting a new research program: Do we know where we are? Note also that, for a number of reasons, identifying the areas where we can be confident is not necessarily a trivial process.

First of all, it involves more than choosing situations where there are no identified fire risk assessment weaknesses (including gaps) because there can be situations where analytical tools with weaknesses are still “good enough”. Thus, from an applications viewpoint, the notion of “weaknesses” is relative and needs to be addressed in the context of the decision to be
made. Further, it needs to be recognized that, in a risk-informed environment, decisions are not based solely on the results of the risk assessment. The question isn't if there are weaknesses in the fire risk assessment technology -- there certainly are -- but if the weaknesses are such that the decision making process is not improved by the use of fire risk assessment results. In other words, as noted by our Advisory Committee on Reactor Safeguards, the question is not if the technology has weaknesses, but rather if the weaknesses will reduce our confidence “that the use of PRA results and insights will improve on the existing regulatory system for the problem of interest” [9].

Second, the confidence with which we can apply the current fire risk assessment technology depends not only on the type of the decision (e.g., does an inspection finding regarding a degraded fire barrier risk warrant follow-up versus can a barrier be removed) but also on issue- and plant-specific details (e.g., what are the contents of the rooms separated by the barrier, how does degradation or removal affect required plant personnel actions). To my knowledge, no one has as yet formulated any generic rules concerning the applicability of fire risk assessment technology to a particular decision.

Can we resolve these issues in the next three days of the workshop? Probably not. But I do think that we can certainly make a good start towards identifying conditions when fire risk assessment results insights can be used with confidence. Let me reiterate that I see this as a crucial issue, not only to the development of NFPA 805 and any associated rulemaking activities, but also to any of our future efforts related to fire risk assessment. We need to be able to show how fire risk assessment technology can help resolve issues, not just identify issues. Surely, we can use the roughly 20 years of fire risk assessment research, development, and applications experience under our belts to do this.
Workshop Challenge

So these are my two key issues: 1) we need to be able to address identified gaps and weaknesses in current fire risk assessment technology; and 2) we need to be able to identify where fire risk assessment results and insights can be used right now (with an appropriate technical basis).

Let me issue a challenge to you, a challenge I hope you will take up over the next three days of this workshop.

As Director of the Office of Nuclear Regulatory Research, I am one of those decision makers that is supposed to be making decisions supported by the results and insights of fire risk assessments. You know what? I often feel that I'm on shaky ground. I don't feel very confident in fire risk assessment results when I'm told that the current state of the art allows differences in assumptions that can lead to one or more order of magnitude changes in results, and in large variations in the screening of fire areas. I don't feel confident when I see fire risk assessments that don't address the impact of smoke, or studies which address fire effects on operators in a highly rudimentary manner. I don't feel confident when I see the results of highly conservative studies that: a) presume that any fire in any given area will fail all of the equipment in that area, and b) don't do additional analysis when that area turns out to be a potentially important contributor to fire risk. I don't feel confident when I see that guidance has not yet been developed regarding the proper application of fire models, even though it is well within our means to develop such guidance.
I know that our research program is intended to address these and other important issues. I know that others in this workshop are doing this as well. I know that three days is too short of a time to resolve difficult issues. My challenge is this.

Make me (and other decision makers) comfortable that we know what we're doing. That we're knowledgeable about each others' work, and about past efforts. That we know what the problems are. That we aren't using obsolete or inappropriate technologies. That we are being efficient with our available resources. That we know how to effectively use our current technology, as well as criticize it. That we can organize ourselves to resolve our outstanding problems. That we will be successful. (I note that success isn't just about developing an improved technology. It's also getting the technology used to make better decisions. It's not just research and development; it's research and development and deployment and application.)

This workshop has assembled an impressive body of expert participants. We have not only the fire risk assessment practitioners, but also fire modelers and fire protection engineers. We have fire risk assessment technology developers and technology users. We have regulators and regulatees. With ongoing fire protection activities needing the results and insights of fire risk assessments, the time is ripe. I hope you will take advantage of this situation and have a meeting that will significantly impact these activities.

I would like to thank CSNI, the workshop organizing committee, and especially the Chair of the workshop for their efforts in putting this workshop together. And once more, I'd like to thank you, the participants, for spending your time and energy. I wish you success.
References


Table 1 - A Partial List of Fire PRAs for U.S. Nuclear Plants (Not Including IPEEEs)

<table>
<thead>
<tr>
<th>Plant</th>
<th>Sponsor</th>
<th>Date</th>
<th>Fire CDF (lyr)</th>
<th>Total CDF (lyr)</th>
<th>Important Contributors</th>
</tr>
</thead>
<tbody>
<tr>
<td>HTGR (design)</td>
<td>USDOE</td>
<td>1979</td>
<td>1.1E-5</td>
<td>4.1E-5</td>
<td>CSR (only the CSR was analyzed)</td>
</tr>
<tr>
<td>Zion 1/2</td>
<td>Utility</td>
<td>1981</td>
<td>4.6E-6</td>
<td>4.9E-5</td>
<td>Electrical equipment room, CSR</td>
</tr>
<tr>
<td>Big Rock Point</td>
<td>Utility</td>
<td>1981</td>
<td>2.3E-4</td>
<td>9.8E-4</td>
<td>Station power room, cable penetration area</td>
</tr>
<tr>
<td>Indian Point 2</td>
<td>Utility</td>
<td>1982</td>
<td>2.0E-4</td>
<td>4.7E-4</td>
<td>Electrical tunnels, swgr room</td>
</tr>
<tr>
<td>Indian Point 3</td>
<td>Utility</td>
<td>1982</td>
<td>6.3E-5</td>
<td>2.3E-4</td>
<td>Swgr room, electrical tunnel, CSR</td>
</tr>
<tr>
<td>Limerick</td>
<td>Utility</td>
<td>1983</td>
<td>2.3E-5</td>
<td>1.5E-5</td>
<td>Equip. rooms, swgr room, access area, MCR, CSR</td>
</tr>
<tr>
<td>Millstone 3</td>
<td>Utility</td>
<td>1983</td>
<td>4.6E-6</td>
<td>7.2E-6</td>
<td>MCR, instrument rack room, CSR</td>
</tr>
<tr>
<td>Seabrook</td>
<td>Utility</td>
<td>1983</td>
<td>1.7E-5</td>
<td>2.3E-4</td>
<td>MCR, CSR</td>
</tr>
<tr>
<td>Midland</td>
<td>Utility</td>
<td>1984</td>
<td>2.0E-5</td>
<td>3.1E-4</td>
<td>Swgr room</td>
</tr>
<tr>
<td>Oconeec</td>
<td>Utility</td>
<td>1984</td>
<td>1.0E-5</td>
<td>2.5E-4</td>
<td></td>
</tr>
<tr>
<td>TMI-1</td>
<td>Utility</td>
<td>1987</td>
<td>8.6E-5</td>
<td>5.5E-4</td>
<td>MCC area, swgr room, cabinet area</td>
</tr>
<tr>
<td>Sav. River K Rx</td>
<td>USDOE</td>
<td>1989</td>
<td>1.4E-7</td>
<td>3.1E-4</td>
<td>MCR, maint. area, cable shaft, DG rooms</td>
</tr>
<tr>
<td>S. Texas Project</td>
<td>Utility</td>
<td>1989</td>
<td>&lt;1.2E-6</td>
<td>1.7E-4</td>
<td>MCR</td>
</tr>
<tr>
<td>Diablo Canyon 1/2</td>
<td>Utility</td>
<td>1990</td>
<td>2.9E-5</td>
<td>2.0E-4</td>
<td>CSR, MCR</td>
</tr>
<tr>
<td>Peach Bottom 2</td>
<td>USNRC</td>
<td>1990</td>
<td>2.0E-5</td>
<td>2.8E-6</td>
<td>MCR, swgr rooms, CSR</td>
</tr>
<tr>
<td>Surry 1</td>
<td>USNRC</td>
<td>1990</td>
<td>1.1E-5</td>
<td>7.6E-5</td>
<td>Swgr room, MCR, aux bldg, cable vault/tunnel</td>
</tr>
<tr>
<td>La Salle 2</td>
<td>USNRC</td>
<td>1993</td>
<td>3.2E-5</td>
<td>1.0E-4</td>
<td>MCR, swgr rooms, equip rooms, turb bldg, cable shaft</td>
</tr>
<tr>
<td>Grand Gulf 1</td>
<td>USNRC</td>
<td>1994</td>
<td>&lt;1.0E-6</td>
<td>5.7E-5</td>
<td>No areas found to contribute</td>
</tr>
<tr>
<td>Surry 1</td>
<td>USNRC</td>
<td>1994</td>
<td>2.7E-4</td>
<td>4.3E-4</td>
<td>Swgr room, cable vault/tunnel, containment, MCR</td>
</tr>
</tbody>
</table>

a) Area contribution > 1% total fire CDF; contributing areas prioritized by contribution (most important first); MCR = main control room, CSR = cable spreading room  
b) Frequency of core heatup  
c) Prior to plant modifications identified by risk study  
d) Internal events only  
e) Frequency of severe core damage  
f) Total contribution from external events  
g) Seismic contribution calculated using EPRI seismicity curve  
h) Midloop conditions; instantaneous CDF is presented

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**Table 2 - Generic Fire Risk Assessment Guidance Issues Identified During IPEEE Reviews**

<table>
<thead>
<tr>
<th>Issue</th>
</tr>
</thead>
<tbody>
<tr>
<td>Human error probabilities used in screening</td>
</tr>
<tr>
<td>Heat loss factors (used in a FIVE fire model)</td>
</tr>
<tr>
<td>Fire growth modeling using experimental results</td>
</tr>
<tr>
<td>Main control room (MCR) abandonment</td>
</tr>
<tr>
<td>Dependencies in the suppression process</td>
</tr>
<tr>
<td>Seismic-fire interactions</td>
</tr>
<tr>
<td>Control system interactions associated with MCR fires</td>
</tr>
<tr>
<td>Effect of smoke on manual fire fighting effectiveness</td>
</tr>
<tr>
<td>Effect of fire suppressants on safety-related equipment</td>
</tr>
<tr>
<td>Fire-induced special initiators (e.g., loss of service water)</td>
</tr>
<tr>
<td>Scenario screening criteria involving enclosed ignition sources</td>
</tr>
<tr>
<td>Electrical cabinet heat release rates</td>
</tr>
<tr>
<td>Scenario screening criteria involving transient ignition sources</td>
</tr>
<tr>
<td>Scenario screening criteria involving non-combustible shields</td>
</tr>
<tr>
<td>Scenario screening criteria involving multiple targets</td>
</tr>
</tbody>
</table>
Session I
National and International Fire Research and Development

- The U.S. Nuclear Regulatory Commission’s Fire Risk Research Programme - An Overview - Dr. Nathan Siu and Mr. Hugh Woods, U.S. Nuclear Regulatory Commission (NRC), United States

- A General Overview of the IPSN Fire Research - MM. Jean-Marc Such, Richard Gonzalez, B. Tourniaire, L. Audouin, C. Casselman, L. Rigollet, and W. Le Saux, Institut de Protection et de Sûreté Nucléaire (IPSN), Département de Recherches en Sécurité, CE de Cadarache, France

- Development of Probabilistic Safety Assessment Methodology for Fire Events in Candu Plants - Mr. George How-Pak Hing and Mr. A. H. Stretch, Atomic Energy of Canada Ltd, Canada

- Fire PSA: Applications and Insights - Dr. Mardy Kazarians, Kazarians & Associates, United States

- Fire Risk Assessment in Germany: Regulatory Guidance and Applications - Dr. Heinz-Peter Berg, Bundesamt fur Strahlenschutz (BfS), Germany and Dr. Marina Röwekamp, Gesellschaft für Anlagen- und Reaktorsicherheit mbH (GRS), Germany
The U.S. Nuclear Regulatory Commission's Fire Risk Research Program - An Overview

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Washington, DC 20555, USA

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) has initiated a research program aimed at addressing gaps in current capabilities to perform realistic fire risk assessment. The intent of the program is to support an expanded use of risk-informed, performance-based methods for fire protection applications. This paper summarizes the current research plan for the program. The summary includes the program objectives, summary task descriptions, a summary of the overall program schedule and funding, and potential future activities. References are also provided for readers interested in additional details on fire risk assessment, fire research, and NRC's plans.

1. BACKGROUND

As stated in the U.S. Nuclear Regulatory Commission's (NRC's) policy statement on the use of Probabilistic Risk Assessment (PRA) [1], the NRC intends to increase the use of PRA technology in "all regulatory matters to the extent supported by the state of the art in PRA methods and data." Recent activities include the development of a general risk-informed framework for supporting licensee requests for changes to a plant's licensing basis, described in Regulatory Guide (RG) 1.174 [3]; and efforts to make Part 50 of the Code of Federal Regulations more risk-informed.

In the area of fire protection, there is interest from both the NRC and industry in the use of PRA technology to deal with outstanding issues. Specific applications include the identification of plant-specific vulnerabilities, the evaluation of the acceptability of proposed changes to specific parts of a plant's program, the evaluation of the safety significance of certain fire protection issues (e.g., fire-induced circuit failures), and the evaluation of the safety significance of fire protection inspection findings. An industry consensus standard (NFPA 805), which uses risk information in evaluating a plant's fire protection program, is being developed under the auspices of the National Fire Protection Association (NFPA). It is anticipated that the completed standard will use an approach that is compatible with RG 1.174.

When used in a risk-informed decision making framework, fire risk assessment (FRA) is useful in that it provides a systematic, integrated method for evaluating the importance of fire protection issues. However, the current FRA state of the art is not as mature as that for assessing the risk contributions of many other important accident initiators. As shown by a review of Individual Plant Examinations of External Events (IPEEEs) [4], variations in analytical assumptions can lead to orders of magnitude variations in estimates of fire-induced core damage frequency (CDF), and qualitatively different risk insights are possible. Such uncertainties can clearly affect a decision maker's confidence in the results of FRAs and, in hindsight, lead to suboptimal decisions.

1According to Ref. 2, "A risk-informed approach to regulatory decision making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety."
Ref. 5 identifies a number of areas where improvements in FRA methods, tools, and data will improve the ability of FRA to support decision making. To address these areas, the Office of Nuclear Regulatory Research (RES) initiated a fire risk research program in Fiscal Year (FY) 1998.

This paper describes the current research plan for the RES fire risk research program. The paper covers the program objectives, summary task descriptions, a summary of the overall program schedule and funding, and potential future activities. Additional details (e.g., task scope, leads, and resources; interactions with other tasks, fire research programs, and fire safety activities) can be found in Ref. 6.

2. PROGRAM OBJECTIVES AND CHARACTERISTICS

The general objectives of the fire risk research program are as follows.

- Improve qualitative and quantitative understanding of the risk contribution due to fires in nuclear power plants.
- Support ongoing or anticipated NRC fire protection activities, including the development of risk-informed, performance-based approaches to fire protection.
- Develop improved fire risk assessment methods and tools (as needed to support the preceding objectives).

The technical objectives of the program are largely focused on the three elements of fire protection defense in depth (fire prevention, fire detection and suppression, fire mitigation), which have analogous elements in typical FRAs [7-10]. The objectives are, for the most part, aimed at developing an FRA state of the art which is, loosely speaking, comparable in quality to that for current PRA for other internal events. In particular, they are aimed at developing:

- improved estimates of the frequencies of challenging fires;
- improved fire modeling tools for risk significant scenarios, including guidance for proper application (accounting for limitations and uncertainties);
- mode-specific thermal fragilities for cables and other key components;
- guidance for identifying scenarios for which smoke effects may be risk significant;
- improved estimates of the probability of fire and fire effects containment (including active and passive barriers);
- configuration and condition-sensitive fire protection system reliability estimates, including guidance for application;
- improved tools for assessing the risk impact of circuit interactions; and
- improved understanding of the implications of major fire events for FRA.

The program focuses on the development of evolutionary improvements on existing FRA approaches, or improved guidance for using these approaches, as opposed to the development of new methodologies. It emphasizes the improved use of existing information, and generally avoids the performance of new experiments. In cases where the technical issues cannot be adequately dealt with using these approaches, the program employs feasibility or scoping studies to support planning for more detailed studies. The program also takes into account the products and needs of parallel activities (e.g., the NRC ATHEANA program [11], the NFPA 805 standard development).
3. TASK DESCRIPTIONS

The technical tasks included in the fire risk research program are listed in Table 1. This section provides, for each task, a description of the background for the problem being addressed and the technical objectives.

3.1 Tools for Circuit Failure Mode and Likelihood Analysis

When dealing with fire-induced damage to electrical cables, two important effects are the loss of function or spurious actuation of equipment associated with the cables. In FRAs, the latter failure mode is typically assumed to be caused by "hot shorts," i.e., short circuits involving a powered conductor.

Hot shorts can be a significant direct and indirect risk contributor. In one advanced reactor design FRA, hot short scenarios (leading to medium or large LOCAs due to spurious valve operation) contribute over 95% of the predicted fire-induced CDF for that design. Complications in procedures designed to address the potential of equipment damage due to hot shorts contribute to the significant fire risk contribution at another boiling water reactor plant.

Table 1. Fire Risk Research Program Technical Tasks, FY 1998-2000

<table>
<thead>
<tr>
<th>Lead Org.</th>
<th>Task Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>SNL</td>
<td>Tools for circuit failure mode and likelihood analysis</td>
</tr>
<tr>
<td>SNL</td>
<td>Tools for fire detection and suppression analysis</td>
</tr>
<tr>
<td>SNL</td>
<td>IEEE-383 rated cable fire frequency analysis: feasibility study</td>
</tr>
<tr>
<td>SNL</td>
<td>Fire modeling toolbox: input data and assessment</td>
</tr>
<tr>
<td>SNL</td>
<td>Experience from major fires</td>
</tr>
<tr>
<td>SNL</td>
<td>Industrial fire experience</td>
</tr>
<tr>
<td>SNL</td>
<td>Frequency and characteristics of switchgear and transformer fires</td>
</tr>
<tr>
<td>SNL</td>
<td>Fire barrier reliability model development and application</td>
</tr>
<tr>
<td>UMd</td>
<td>Integrated model and parameter uncertainty</td>
</tr>
<tr>
<td>TBD</td>
<td>Frequency of challenging fires</td>
</tr>
<tr>
<td>TBD</td>
<td>Fire model limitations and application guidance</td>
</tr>
<tr>
<td>NRC</td>
<td>Risk significance of turbine building fires</td>
</tr>
<tr>
<td>NRC</td>
<td>Penetration seals</td>
</tr>
<tr>
<td>NRC</td>
<td>Risk significance of multiple unit interactions</td>
</tr>
<tr>
<td>NRC</td>
<td>Use of advanced fire models in fire risk assessment</td>
</tr>
</tbody>
</table>

SNL = Sandia National Laboratories
TBD = to be determined
UMd = University of Maryland
USNRC = U.S. Nuclear Regulatory Commission
From a methodology standpoint, a major concern is that hot short analyses performed for FRAs are generally simplistic. The probability of a single hot short is commonly based on a generic probability distribution derived subjectively in Ref. 12 from a limited amount of information. (The distribution, assumed to be lognormal, has a 5th percentile of 0.01 and a 95th percentile of 0.20; its mean value is 0.07.) The probability of multiple hot shorts is typically obtained by multiplying this probability an appropriate number of times. The latter procedure ignores the potentially significant impact of dependencies, both aleatory and epistemic. Furthermore, both it and the original single hot short distribution do not explicitly reflect such potentially important issues as the circuit design, the function of the cable, and the characteristics of other cables in the vicinity.

The objectives of this task are as follows:

- To develop an improved understanding of the mechanisms linking fire-induced cable damage to potentially risk significant failure modes of power, control, and instrumentation circuits.
- To develop improved methods and data for estimating the conditional probabilities of key circuit faults, given damage to one or more cables.
- To develop representative estimates of the conditional probabilities of key circuit failure modes applicable to currently operating U.S. nuclear power plants. The estimation process will include an identification and quantification of the key uncertainties in the estimates.
- To gain risk insights concerning fire-induced circuit failures, especially those associated with cable hot shorts.
- To identify areas where additional work needs to be done to improve understanding of the risk associated with fire-induced circuit failures.

3.2 Tools for Fire Detection and Suppression Analysis

FRA analyses of the effectiveness of fire detection and suppression efforts require estimates of the reliability of automatic suppression systems. One concern with current approaches involves the use of generic non-nuclear industry estimates for system unreliability. These estimates do not account for variations in such plant- and scenario-specific factors as sprinkler head location relative to the fire, sprinkler system design, and room congestion. The use of generic suppression system reliability estimates may also be optimistic in studies employing severity factors because the estimates are not conditioned on the fire severity.

A suppression analysis also requires estimates of characteristic delay times (e.g., the time to initiate fire suppression, the time to final suppression). More precisely, since these times should be modeled as random variables, estimates of the parameters of the aleatory distributions for these times are required. Event data have been used in the estimation process (e.g., see [10,13]). However, the data are limited and are not always sufficiently defined to support direct estimation of key parameters. Model-based approaches (e.g., [14]) can be used to specialize event-based generic distributions to account for scenario-specific features, but difficulties arise when addressing the uncertainties in the models. (Note that fire models which are conservative with respect to fire damage predictions may be non-conservative with respect to fire suppression.) Expert judgment provides another way to account for plant-specific features (e.g., [15]). To date, however, such approaches have not integrated the results of the expert elicitation with actual event data.

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2 "Severity factors" are commonly used in FRAs to model the fraction of reported fires that have the potential to cause damage to components not involved in the initial fire.
The preceding discussion addresses estimation issues in suppression analysis. Modeling issues which are not quantitatively addressed by most FRAs include: the impact of smoke and loss of lighting on the effectiveness of manual fire fighting and the effectiveness of compensatory measures (e.g., fire watches) for temporary fire protection deficiencies.

The objectives of this task are as follows:

- To develop improved methods and data for estimating the reliabilities of automatic and manual suppression activities.
- To develop estimates of these reliabilities applicable to currently operating U.S. nuclear power plants.
- To identify and quantify key uncertainties in these estimates.

3.3 IEEE-383 Rated Cable Fire Frequency Analysis: Feasibility Study

One key issue in FRA concerns the frequency of self-ignited fires involving cables certified by the IEEE-383-74 flame test. Tests have shown that electrical ignition of fires involving these cables is difficult (e.g., see Ref. 16). A practical FRA question is, for compartments containing only rated cables, what is the frequency of cable fires? Is it sufficiently low that the analysis only need consider transient-fueled fires? As shown by the results of a number of IEEEEs, differences in analysis assumptions can lead to qualitatively different risk insights.

Nuclear power plant data for self-ignited cable fires are sparse; the number of reported events is small and the event descriptions rarely include much detail about the types of cables involved. Information is needed on: a) relevant events from other facilities and industries, and b) scenarios leading to fire initiation.

The objectives of this task are as follows:

- Determine if there is an adequate technical basis for asserting that the frequency of self-ignited fires involving IEEE-383 rated cables is too small to consider in nuclear power plant FRAs.
- Failing the above, determine the feasibility of developing a practical, improved methodology for estimating the frequency of such fires.
- Identify the work needed to develop and implement this methodology.

3.4 Fire Modeling Toolbox: Input Data and Assessment

Some of the key uncertainties in the prediction of the hazardous environment induced by a fire, and of the response of critical equipment to this environment, are due to sparseness of basic data concerning: a) the flammability and damageability characteristics of the equipment under fire conditions, and b) the validity of currently available physical models for predicting the fire-induced environment.

Numerous experiments have been performed to collect various data relevant to the thermal behavior and effects of fires in nuclear power plants. However, there are three problems with these data. First, in some cases, the data have not been processed to allow their use by analysts. Second, the experiments were not usually performed with the needs of fire modeling in mind. (This means that direct measurements of key model parameters may be lacking.) Third, weaknesses in the experimental processes (from an FRA modeling perspective) have not been characterized. The latter two concerns do not mean that the experimental results are useless; they do mean that data processing will require not only transcription of raw data into appropriate media and formats, but also characterization from an FRA perspective.

Some work has been performed on non-thermal effects of fire. This work has led to identification of potential failure modes of electronic equipment due to smoke effects (e.g., [17,18]). It has not yet led to
characterizations of the fragilities of key equipment that can be directly used in FRAs. Additional work is needed to develop these fragilities.

The objectives of this task are as follows:

- Collect and characterize available experimental data potentially relevant to the prediction of electrical cable flammability and thermal fragility.
- Collect and characterize available experimental data potentially relevant to the prediction of the thermal fragility of other potentially risk significant nuclear power plant components.
- Collect and characterize available experimental data potentially relevant to the assessment of model uncertainties in current fire environment models.
- Process and publish the SNL base line fire model validation data (see Ref. 19) in a format suitable for its use by analysts to validate fire models used in FRAs.
- Generate experimental data needed to assess the smoke fragility of potentially risk significant nuclear power plant components.
- Collect and characterize available experimental data potentially relevant to the assessment of fire heat release rates.

3.5 Experience from Major Fires

A number of safety significant fires have occurred in U.S. and international nuclear power plants (e.g., see Refs. 20 and 21). While these events have been studied from a fire protection point of view, current FRAs tend to make limited use of the information obtained from these events. For example, counts of events are used to estimate fire frequencies, but the descriptions of many events have not been seriously studied to determine if changes in the FRA models or even basic FRA structure are warranted.

The objectives of this task are as follows:

- Identify key fire risk and FRA insights from serious U.S. and international nuclear power plant fires.
- Develop recommendations for FRA improvements and areas for further investigation.

3.6 Industrial Fire Experience

Reportable nuclear power plant fires are not frequent events; the average occurrence rate is on the order of 0.3 per plant-year [22]. The frequency of potentially risk significant fires is considerably lower. Thus, current FRA characterizations of the relative likelihood and progression of nuclear power plant fire scenarios are largely model- rather than experience-based. To reduce the uncertainties in these characterizations, it should be useful to review the experience from non-nuclear industrial fires involving equipment and occupancies similar to those found in nuclear power plants. Such a review can provide useful qualitative information (e.g., how well do operators perform in degraded environments) as well as indications of the relative likelihood of different scenarios. As discussed in Ref. 23, it is not expected that the review will necessarily lead to quantitative data that can be directly used in estimates of fire scenario frequencies; the non-nuclear information sources appear to be in such a form that resource requirements for such an effort would be considerable.

The objective of this task is to collect and evaluate industrial data relevant to the analyses of specific nuclear power plant fire scenarios.
3.7 Frequency and Characteristics of Switchgear and Transformer Fires

Fires involving low- to medium-voltage (\( \leq 6.19 \text{kV} \)) electrical switchgear (including motor control centers) are often important contributors to fire risk. However, there is considerable uncertainty as to how switchgear fires should be modeled (as a hazard to other components in the area). Many IEEEs have selected a relatively low heat release rate for their switchgear fires, as compared with the full range of results obtained from Sandia National Laboratories (SNL) electrical panel fire tests [24,25]. The value used (69 kW) represents the burning of a single bundle of IEEE-383 qualified cables; fires involving more fuel will naturally be greater in magnitude. There is also considerable uncertainty concerning the heat release rates of indoor transformer fires.

Besides data uncertainties, a concern with current FRAs is that they treat switchgear and transformer fires essentially as pool fires. They do not account for the events leading up to the fire. In particular, if the fire is started by an electrical fault, the scenario can involve the overheating and ignition of cables far removed from the component. In the case of oil-filled transformer fires, an energetic fault can lead to a spray of burning oil rather than a pool. Furthermore, the blast and missiles from an energetic fault can cause direct mechanical damage to nearby components.

The objectives of this task are as follows:

- Develop frequency-magnitude relationships for switchgear and transformer fires.
- Develop a simple method for addressing the non-thermal effects of switchgear and transformer energetic faults.

3.8 Fire Barrier Reliability Model Development and Application

The treatment of local fire barriers varies in current FRAs. Approaches include: a) fully crediting the barriers if they are included in a fire barrier surveillance program [14]; b) using simple heat transfer models (not a common approach); c) crediting barriers for delaying fire-induced damage and ignition based on experimental results for a limited number of barrier systems [10]. The first approach doesn’t allow for the finite probability of failure of the barrier. The second and third approaches do not account for key factors (e.g., mechanical construction details, material behavior under fire conditions) which affect performance of many current barrier systems. The third approach also uses experimental results in situations not directly covered by the experiments (e.g., different fire severities, geometries).

Intercompartment fire barriers are typically fully credited when the barriers separate fire areas. Some studies employ reliability estimates for specific barrier elements (penetration seals, dampers, doors); these estimates are quoted in Refs. 9 and 15. Many studies fully credit barriers between fire zones under certain conditions (e.g., see [14]). Again, the first approach doesn’t allow for the barrier to fail. Regarding the second and third approaches, the formal technical bases for the estimates/conditions are unavailable.

The objectives of this task are as follows:

- Develop a screening model for predicting the performance of local fire barriers under exposure fire conditions. The model will address probabilistic issues (e.g., barrier construction and installation) as well as phenomenological issues (e.g., exposure fire severity).
- Estimate the probability of failure (on demand) of fire dampers, fire doors, and penetration seals for challenging fire scenarios.
3.9 Integrated Model and Parameter Uncertainty

In general, methods for estimating "parameter uncertainty," i.e., uncertainty in the model output due to uncertainties in the model parameters, are well known and routinely applied in many situations. On the other hand, there currently is no consensus concerning formal methods for estimating "model uncertainty," i.e., the additional output uncertainty due to modeling approximations. Ref. 26 presents many viewpoints on how model uncertainty should be defined and addressed in general situations.

In the case of fire model prediction, simulation codes are available to predict the dynamic behavior of variables that are, in principle, measurable. Furthermore, limited amounts of experimental data potentially useful for estimating output model uncertainty are also available. However, the experiments do not cover all possible situations to which the model will be applied. This can affect the applicability of any experimentally-derived output model uncertainty distribution. Further, the values of the model parameters needed to simulate the experiments may not be well known. (Note that the experiments are not necessarily performed for the sake of model validation.) It may therefore be unclear as to how much of the difference between model predictions and experimental data is due to the parameter uncertainty and how much is due to the model uncertainty.

A relatively simple approach for quantifying uncertainty in model predictions in the presence of model uncertainty and parameter uncertainty is proposed in Ref. 27. However, this approach has not been fully tested. Furthermore, the relationship between the approach and the fundamental frameworks discussed in Ref. 26 has not been investigated.

Work on this task is being performed as part of a cooperative research agreement with the University of Maryland.

The objectives of this task are as follows:

➢ Evaluate the ability of various methodologies to assess model uncertainty to the same level as parameter uncertainties, and formulate a framework under which their combined uncertainties can be assessed.
➢ Demonstrate how the formulated framework can be applied to address real issues involving combined parameter and model uncertainties.

3.10 Frequency of Challenging Fires

One of the key issues in fire frequency analysis for FRA is the reduction of fire frequencies performed in most detailed FRAs to accommodate the fact that not all fires are risk significant, i.e., that a fire must have the proper location and severity characteristics to be an important contributor to critical equipment damage. Current reduction factors used to address location and severity considerations can reduce the compartment fire frequencies (the \( \lambda_i \)) by one or more orders of magnitude. However, the basis for these reduction factors is not strong. Early studies (e.g., Ref. 28) relied heavily on analyst judgment. Attempts to reduce the influence of judgment have led to: a) the component-based approach to fire frequency, employed in the FIVE methodology [14], and b) event-based estimation of severity fractions (e.g., [10]). The concerns with the event-based treatment of the severity issue include: ambiguity in the data (qualitative event narratives are used to determine if a given fire was severe); possible double-counting of the impact of suppression in the data (effective suppression may be the reason why a particular fire was not reported as being severe, but fire suppression is modeled separately in the FRA); neglect of possibly significant differences between conditions (e.g., fuel bed geometry) of the event, and those of the situation being analyzed in the FRA which can affect the severity of the fire; and scarcity of data for the large, transient-fueled fires that have been predicted to dominate fire risk in a number of studies.
The preceding issues deal with the problem of quantifying the likelihood of fire occurrence. A related issue concerns the establishment of conditions for the next stage of the FRA, the estimation of the likelihood of equipment damage. Current methods for performing this next stage generally rely upon fire environment simulation models which require the specification of the initial scenario conditions. The problem is that current fire frequency analyses provide, at most, the frequency of “small” and “large” fires in a specified compartment or involving a specified component. They do not provide the physical characteristics associated with these “small” and “large” fires needed by the simulation models. This ambiguous interface between the fire frequency and equipment damage analyses allows significant analyst discretion. For example, Ref. 28 assumes that “large” fires have a severity equivalent to a 2-foot diameter oil fire, while Ref. 29 assumes that this is the equivalent severity of “small” fires.

The objectives of this task are to:

- Determine the feasibility of developing a practical, improved methodology for defining, characterizing, and quantifying the frequency of challenging nuclear power plant fire scenarios.
- Develop and demonstrate the methodology.

3.11 Fire Model Limitations and Application Guidance

In FRA, characterization of the fire-induced hazardous environment requires the estimation of the time-dependent temperature and heat fluxes in the neighborhood of the safety equipment of interest (i.e., the “targets”). This requires the treatment of a variety of phenomena as the fire grows in size and severity, including the spread of fire over the initiating component (or fuel bed), the characteristics of the fire plume and ceiling jet, the spread of the fire to non-contiguous components, the development of a hot gas layer, and the propagation of the hot gas layer or fire to neighboring compartments. It also requires an appropriate treatment of uncertainties in the structure and parameters of the models used to perform the analysis.

To date, U.S. nuclear power plant FRAs have used quite simple zone model-based tools, e.g., the correlations provided as part of the FIVE methodology [14] and the COMPBRN IIIe computer code [30], to predict the thermal environment due to a variety of fire sources. However, it is not always recognized in FRAs that these tools have been developed to address specific classes of fire problems and are not applicable to all situations. For example, the inherent modeling assumptions in both FIVE and COMPBRN do not address many practical complexities (e.g., obstructions in the fire plume, complex compartment geometry, complexities in forced ventilation flow, physical movement of fuel, room flashover) which can be important in some analyses. Further, the correlations employed implicitly or explicitly by these models are not appropriate for all situations. Some scenarios of potential concern include very small fires (e.g., single wire electrical insulation fires), very large fires (e.g., very large oil spill fires), or elevated fires. Unfortunately, the limitations of these simple models have not been succinctly characterized to inform FRA analysts, many of whom may not have strong background in fire science, when they should be wary of the model predictions.

The objectives of this task are:

- To identify the areas of uncertainty and limitations associated with fire models which are either: a) currently used in FRAs, or b) might be used in future FRAs.
- To develop improved guidance for using these fire models in FRAs.
3.12 Risk Significance of Turbine Building Fires

Historical turbine building fires (e.g., Narora [21]) and a number of IPEEEs show that severe turbine building fires can be important contributors to risk. Potential sources of uncertainty in the evaluation include the lack of knowledge concerning the frequency-magnitude relationship for turbine building fires and the adequacy of current FRA tools for predicting the environment induced by a severe turbine building fire.

The objectives of this task are to:

➢ Improve the technical basis for fire risk assessments of turbine building fires.
➢ Assess the risk significance of turbine building fires.
➢ Develop recommendations for FRA improvements and areas for further investigation.

3.13 Penetration Seals

Between 1994 and 1998, the NRC staff performed a number of technical assessments of fire penetration seals to address reports of potential problems, to determine if there were any problems of safety significance, and to determine if NRC requirements, review guidance, and inspection procedures were adequate [31]. During the resolution process of this issue, questions were raised regarding the risk significance of the issues and problems, and whether risk-informed approaches to issue resolution were available [32].

The objectives of this task are to:

➢ Determine the extent to which current fire risk assessment methods and data can be confidently used to support prioritization of penetration seals for inspection.
➢ Identify issues (if any) requiring research to improve risk-informed prioritization and/or confidence in such a prioritization.

3.14 Risk Significance of Multiple Unit Interactions

The results of a number of IPEEE reviews show that the risk implications of scenarios where a single fire can induce simultaneous transients in multiple units may be significant. Although the frequencies of such scenarios are expected to be low, their potential consequences are significantly greater than those of scenarios affecting only one unit. A concern is that IPEEEs using scenario screening frequencies of $10^{-9}$/yr may have screened out these scenarios without considering their potential effect.

The objectives of this task are to:

➢ Identify plants where a single, severe fire may simultaneously affect multiple units and assess the risk implications of such fires.
➢ Develop recommendations for additional research.

3.15 Use of Advanced Fire Models in Fire Risk Assessment

As discussed in Section 3.11, the modeling assumptions inherent in the fire models currently used in FRAs do not address many practical issues which can be important in some analyses. A number of these issues, e.g., obstructions in the fire plume, complex compartment geometry, complexities in forced ventilation
flow, are addressed by state of the art "field models" (e.g., the National Institute of Standards and Technology's Large Eddy Simulation code [33]) which explicitly address the computational fluid dynamics aspects of fire. Although these models are currently too resource intensive (including analyst time as well as computation time) for routine use in FRAs, it appears that they should be useful tools for evaluating, and even modifying, the simpler FRA models.

The objectives of this task are to:

- Identify specific FRA areas where field models could be used to improve confidence in FRA results.
- Use a selected field model to model fire experiments of interest to FRA (including the SNL base line validation tests [19]).
- Develop recommendations concerning the appropriate role of current field models in FRA and what work needs to be done to allow such use.

4. PROGRAM SCHEDULE

Figure 1 shows the overall schedule for the fire risk research program tasks.

Figure 1. Overall Task Schedule

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5. POTENTIAL FUTURE ACTIVITIES

It is anticipated that, by the end of FY 2000, the NRC fire risk research program will yield a set of FRA improvements and insights that will be useful in addressing specific fire protection issues. However, the program as currently defined does not provide a summary statement of the overall impact of the FRA improvements, nor does it provide a summary set of guidance for performing improved FRA. Furthermore, it does not complete the integration of advanced fire models (or their results) into FRA. These are important activities for supporting the increased use of risk-informed, performance-based methods in fire protection. The following tasks may therefore be defined and initiated after FY 2000.
Fire risk requantification. This task will apply the results of the fire risk research program in a requantification of the fire risk for a selected plant. The objectives of the requantification will be to determine the risk impact associated with the FRA improvements and to develop insights concerning the application of the improved FRA methods and tools.

FRA guidance development. This task will use the results of the fire risk research program to develop an improved guidance document for performing FRA. This document will support the standardization of FRA at a level of description more detailed than that currently envisioned for the NFPA 805 standard.

Integration of advanced fire models into FRA. This task will use the results of the task “Use of Advanced Fire Models in Fire Risk Assessment” (see Section 3.15) to incorporate advanced fire models (or their results) into FRA.

6. REFERENCES


A GENERAL OVERVIEW OF THE IPSN FIRE RESEARCH

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Abstract

For more than ten years, IPSN has been carrying out experimental and modelling research programs in order to improve the knowledge on the fire consequences in a nuclear installation. The fire is indeed a major risk for the industrial installations, and in the case of a nuclear of reprocessing plant, the consequences of a fire can lead to a possible release of radioactive materials inside the installation and even outside into the environment if the confinement and filtering devices are damaged. This paper provides a general overview of the IPSN fire research, including the main future programs: The main conclusions are that the improvement of our knowledge in this complex domain is very important and that the research results (especially assessed computer codes) are already used for safety evaluations. However, a lot of fire configurations and scenarios remain to be studied and some phenomena exhibited so far in IPSN experiments need additional investigations and a better modelling. Research will continue in two complementary ways: a research program addressing safety concerns, which generally ask for a rather quick and global answer, and a more basic research program, dealing with fundamental aspects of the fire, that will be carried out in a tight collaboration with universities and French and foreign research organisations studying combustion and fire.

1. INTRODUCTION

The damages that a fire can cause to nuclear plants (with possible release of radioactive substances to the environment) make fire safety studies of a primary importance in risks assessment. Generally, nuclear plants (nuclear power plants NPP and nuclear reprocessing plants NRP) are ventilated in order to insure a depression in rooms. This depression associated to the filtering devices located into the ventilation networks must enable to confine radioactive substances inside the plant. The design of nuclear plants makes the study of fire development and propagation very complex since there is a strong interaction between the fire and the ventilation. This survey is made even more difficult by the great number of possible flammable materials that can be found in nuclear plants.

The French Nuclear Safety and Protection Institute has been studying for more than 20 years the problem of fire in nuclear plants. The main purpose of this research is to gain knowledge on this very complex domain in order to make pertinent assessments on the rules set up in NPP and NRP to prevent fire risks. The improvement of the prevention and of the emergency plans is also expected from this research.

In the frame of the safety assessment of Fast Breeder Reactors (SPX1 in particular), a large program on sodium fires was carried out between 1975 and 1990. Large facilities were built in CADARACHE to perform the experimental studies. Since 1980, the studies were progressively extended to the problem of "classical" fires (involving organic materials). This paper is focused on this second phase.
The scientific research is classically divided into two parts: experimental tests and modelling. On the one hand, this work leads to the development of validated computer codes which can be used to perform safety evaluation studies. On the other hand, it improves our knowledge in order to better anticipate and solve safety concerns. This research is performed in the frame of a large collaboration with nuclear partners (EDF, COGEMA) and with other French (INERIS) and foreign research institutes including the universities.

This paper starts with the description of the configurations and the phenomena involved in a compartment fire. Then, the R&D programs, the computer codes and the experimental facilities are presented. The main findings and open issues are briefly explained before giving the planned work for the next years.

2. PHENOMENA AND CONFIGURATIONS INVOLVED IN A FIRE

2.1 CONFIGURATIONS AND CLASSIFICATION OF STUDIES

The objective of the studies is to evaluate the consequences of a fire in a nuclear plant. The extent of such consequences mainly depends on the nature of the fuel, the heat release rate of the fire, the design of the building rooms and the management of the ventilation. Flammable materials are either liquids (pool or jet) or solids (horizontal or vertical positions). Two pictures (Figure 1 and Figure 2) present the main configurations that have to be addressed for a fire in a nuclear plant. The studies can be divided into four items:

- one fire located either in the centre, along a wall or in the corner of a room;
- several fires in the room as a consequence of the propagation of the main fire to flammable targets located into the same room;
- fires in several rooms ("multi-room fires"): this can result from the propagation of the fire from a room to neighbouring rooms through doors and the ventilation network;
- "local effects": this consists in evaluating the consequences of a fire on the operation of materials and devices that are crucial for the safe operation of the plant (cable tray, electronic cabinets, ...).

![Figure 1: One room fire scenario](image-url)
2.2 PHYSICAL PHENOMENA IN AN OPEN ATMOSPHERE FIRE

The pyrolysis (for solids) or the vaporisation (for liquids) of the flammable material, the chemical reactions inside the flame (zone located above the flammable pool) and the hydrodynamics of the plume of smokes and gases (zone located above the flame) are the three main phenomena controlling the fire in an open atmosphere (Figure 3) [1][2][3][4].

The vaporisation of the liquid or the pyrolysis of the solid corresponds to the generation of burnable gases from the flammable material due to its heat up by radiative and convective exchanges with the flame. Each flammable material (solid or liquid) is characterised by several parameters related to the chemical reaction of combustion such as the heat of reaction and the generated gases and soot. The mass flow rate by unit area of pyrolysis/vaporisation increases as a function of the characteristic dimension (for instance, the diameter for a circular pool) of the pool fire and reaches a constant value for a dimension greater than 1 m in an open atmosphere.

The burnable gases are mixed and burned with the oxygen of the air into the reactive zone called flame. This combustion phenomenon is called diffusion flame. For a characteristic dimension of a pool fire greater than 0.05 m, the upwards flow of the reactive gases and of the combustion products is fully turbulent and controlled by buoyancy effects [5]. The temperature and the gas velocity in the axis of the flame can reach a maximum of 1000 °C and 5 to 10 m/s respectively. The chemical heat generated into the flame region is exchanged with the surrounding by convection and radiation.

The hot gases produced by the fire go upwards into a zone called the plume, located above the flame region, and whose diameter increases with the elevation due to the mixing with the entrained air. Therefore, the temperature and the velocity of the gas in the plume decrease with the elevation [4][6].
2.3 ADDITIONAL PHENOMENA DUE TO THE CONFINEMENT OF THE FIRE

During a compartment fire, there is an accumulation of the hot gases coming from the plume to the upper part of the room (Figure 4) [7] [8]. According to the heat release rate of the fire and to the ventilation mass flow rate, this upper hot gas layer can fill more or less rapidly the room. Therefore, the modification of the atmosphere surrounding the fire has a strong influence on the combustion (kinetics and generated products of the chemical reaction) and on the plume dynamics [9].

In the early stages of the fire, the room pressure increases due to the thermal expansion of the gases connected with their heat-up. The increase of the pressure is correlated to the increase of the heat release rate of the fire. The ventilation limits this over-pressure effect. At the burn out of the fire, there is a
pressure shutdown, soften by the ventilation, which is due to the thermal contraction of the gases (Figure 5).

![Diagram of room pressure over time showing overpressure with the fire ignition, underpressure's damping, and underpressure with the fire extinction.]

**Figure 5: Pressure versus time during a compartment fire**

The hot gases and the radiative flux coming from the flame can threaten the correct operation of devices, located into the room or into neighbouring rooms, which are crucial for the safe operation of the plant (electrical cables, electronic cabinets, ...). Moreover, the devices located into the ventilation network, or designed for insuring the confinement, are submitted to strong stresses jeopardising their operability: thermo-mechanical stresses on Very High Efficiency (VHE) filters and valves possibly leading to their destruction, plugging of VHE filters by soot and possibly radioactive aerosols, opening of doors separating two rooms... The loss of confinement can lead to the radioactive contamination of the plant and of the environment if during the fire, or as a consequence of it, radioactive materials are released into the atmosphere of the plant.

The hot gases and possibly the fire can propagate into neighbouring rooms, leading to the same consequences on the targets as previously explained for a fire into a room. Finally, the pressure evolution into the fire place can result in reverse mass flow rates into the ventilation network. At the burn out stage of the fire, the under pressure can result in a flow of fresh air through the ventilation which can lead to a new start of the fire.

3. **THE STUDIES ON FIRES**

3.1 **R&D PROGRAMS**

The axis of research are mainly defined according to the safety analysis concerns. The tight connection between expertise and research allows to clearly fix the priorities and the purposes of the programs. Besides, there is also a strong connection between the experiments and the modelling in order to develop and assess the computer codes aimed at realising fire safety studies. This means that the experimental programs must have two aspects: analytical tests to assess the models and global tests to validate the completeness of the modelling and the right coupling between the models. Up to now, the experimental work has mainly involved semi-analytical and global tests for two reasons. First of all, there are a lot of
high priority safety concerns which ask for a short term and rather global answer. Furthermore the phenomenology involved in a nuclear plant fire is very wide and complex, thus making rather difficult to define a priori what are the more sensitive physical phenomena asking for a deep investigation through analytical experiments.

The program dealing with "classical" fires was therefore defined in 1988. It was based firstly on the inventory of flammable materials (solids and liquids) encountered in nuclear plants, secondly on the most probable fire scenarios. The priority of the safety analysis was considered in defining the planning of the program. Up to now, more than 100 fire tests have been performed with different flammable liquids and solids in various geometries and under various ventilation conditions.

A complementary program, not presented in this paper, is conducted on the transfer of radioactive aerosols inside a plant (including the ventilation network) and on the behaviour of the ventilation and confinement devices submitted to the thermomechanical stresses induced by a fire.

The means to carry out the research on fires, i.e. the computer codes and the experimental facilities, are described below.

3.2 THE COMPUTER CODES
A two-year approach is applied by IPSN for the development of computer codes in the domain of fire studies. On the one hand, a zone model called FLAMME_S is developed for the present safety assessment and allows to perform engineering calculations, with a low CPU time. On the other hand, a 3D code, called ISIS, is under development, in order to provide a research tool able to overcome the limitations of the simple tool and to extend the configurations accessible to the simulation.

These two codes are described in a specific paper within this workshop on Fire Risk Assessment [10].

3.3 EXPERIMENTAL FACILITIES
IPSN is carrying out an experimental program in the facilities of its Department of Safety Research located in the CADARACHE centre. These facilities are described in the following table:

<table>
<thead>
<tr>
<th>Vessel name</th>
<th>Volume</th>
<th>Shape and walls</th>
<th>Containment</th>
</tr>
</thead>
<tbody>
<tr>
<td>VEGA</td>
<td>0.316 m³ (Ø 0.7, h=0.9)</td>
<td>Cylinder, Steel</td>
<td>Closed or vented</td>
</tr>
<tr>
<td>CASTOR</td>
<td>4.4 m³ (Ø 1.6, h=2.2)</td>
<td>Cylinder, Steel</td>
<td>Closed (4.10⁵ Pa) or vented</td>
</tr>
<tr>
<td>POLLUX</td>
<td>4.5 m³ (Ø 1, h=5.2)</td>
<td>Cylinder, Steel</td>
<td>Closed (8 10⁵ Pa) or vented</td>
</tr>
<tr>
<td>MERCURE</td>
<td>22 m³ (Ø 2.5, h=5.2)</td>
<td>Cylinder, Steel</td>
<td>Closed (4.10⁵ Pa) or vented</td>
</tr>
<tr>
<td>PLUTON</td>
<td>400 m³ (l=9, w=6, h=7.5)</td>
<td>Parallelepiped, Concrete</td>
<td>Closed (1.250 10⁵ Pa) or vented</td>
</tr>
<tr>
<td>SATURNE</td>
<td>2000 m³ (l=w=10, h=20)</td>
<td>Parallelepiped, Concrete</td>
<td>Natural convection</td>
</tr>
<tr>
<td>JUPITER</td>
<td>3600 m³ (l=20, w=15, h=12)</td>
<td>Parallelepiped, Concrete</td>
<td>Closed (2 10⁵ Pa) or vented</td>
</tr>
</tbody>
</table>

A specific instrumentation allows to measure temperatures, gas velocities, pressure, heat fluxes, concentrations (gas and aerosols), inlet and outlet flow rates of the various components of the facility.
(fuel, gas, walls, targets, ventilation network ...). Furthermore, the fuel vaporisation rate is also followed using a weighting device. This instrumentation set, recording in real time, gives a large amount of data versus time (for example, about 600 measurements per second are available for the JUPITER facility).

4. THE MAIN FINDINGS

Many tests have been conducted by IPSN on different scales which result in a high standard qualification, due to the complete nature of the instrumentation used. For pool fires, the two fuels mainly used are a flammable liquid, the TPH-TBP mixture (sort of kerosene) and a liquid classified as not highly flammable, the pump cooling oil called DTE Medium.

The following sections present some examples of experimental results concerning pool fires, sometimes showing how they allowed codes improvements and mentioning the limits of the knowledge and the needs of further research.

4.1 The flammable materials

4.1.1 The physical properties

A data base of flammable materials (solids and liquids) has been developed and is being supplied gradually. It gives the physical properties (heat of combustion, flash point, emissivity...) for the main flammable materials encountered in nuclear plants. This data base is connected with the FLAMME_S computer code.

4.1.2 The burning rate

Experiments allowed us to determine the burning rate versus time for several materials under various conditions. The burning rate, involved in the calculation of the heat release rate, depends on the fire boundary conditions, in particular on the oxygen concentration. For instance, in confined fire tests, the burning rate increases from 20 to 45 g.m$^{-2}$.s$^{-1}$ for a same 1 m$^2$ pool fire (TPH-TBP). According to BLINOV and KHUDIAKOV [1], the same pool would burn in free atmosphere with a burning rate which order of magnitude is between 70 and 80 g.m$^{-2}$.s$^{-1}$.

The mean experimental values are used for predictions performed with the FLAMME_S code. This approach being not enough general, it is planed to develop a combustion model.

4.2 The flame and the plume

4.2.1 Flame spread

The heat release rate of the fire depends on the burning area of the pool. That is why the spread of the flame on the pool after ignition is a specific subject of interest of our studies. For example, in a test involving a 20 m$^2$ kerosene pool, the duration of the flame spread on the surface of the pool was about 45 s for a total fire duration of 3.1 minutes.

The spread velocity essentially depends on the nature of the fuel, on the initial temperature and on the pool geometry.
4.2.2 Flame and plume structure

In the frame of risk assessment, the knowledge of the temperature field above the pool fire allow us to foresee the behaviour of targets located into the flame or into the plume. Knowing the thermal stresses, it is possible to determine the malfunction delay of equipments (electrical cables for example).

During the experiments, the axial temperature decrease in the flame zone and in the plume was measured, and allowed us to determine a mean value for the flame height (typically about 3 pool diameter).

The results obtained for a pool fire located far from the walls show that during the first minutes following the ignition, the flame and the plume have the same structure than a flame in a free atmosphere (i.e. outside). However, after a few minutes, the hot layer composed of the combustion products may fill the compartment, and then can modify the flame and plume structure. During this second period, classical models of the literature are not in good agreement with experimental results. Therefore, specific studies must be performed to determine a model which will take into account the evolution of the boundary conditions in the development of the flame and the plume.

For pool fires located close to a vertical wall, the flame height of pool fires whose area is smaller than 1 m² is of the same order of magnitude than for fires in the centre of the room. For larger pool fires, the interaction between the flame and the wall gives higher flame heights.

Velocity measurements are also performed into the flame and the plume zones. They are used to estimate the plume flow rate.

4.2.3 Unexpected behaviour of the flame

In spite of the knowledge improvement, the flame behaviour remains not fully understood and some unexpected phenomena have been observed during some experiments. For instance, in a test called LIC 2.12.2, the flame slowly shifted from a 1 m² pool up to an opening located 8 m further. Figure 12 shows the main features of the facility, and Figure 14 gives the main step of the phenomenon, called "the ghosting flame" [11].

4.3 Gas and aerosols in the compartment

4.3.1 Temperature and pressure

The measurements performed in experimental vessels showed that only the vertical gradient is significant (up to 45°C.m⁻¹).

During the numerous tests performed, the gas pressure evolution was recorded and showed the following trend: a pressure peak at the ignition of the pool, a progressive return to a balanced value during the steady state phase, an under-pressure at the extinction. The amplitude and the duration of the over-pressure and under-pressure depend on the heat release rate of the fire, on the vessel characteristics (dimensions and nature of the walls) and on the ventilation network capacity. For example, during a 20 m² kerosene pool fire in a 3600 m³ concrete vessel connected to a ventilation network (exhaust part only), the over-pressure rose to about 500 hPa within 1 minute (Figure 6).
4.3.2 Gas and aerosols concentrations

The experiments show that in most cases, the combustion is not possible with an oxygen concentration below 12% in the lower layer. This value is used in the numerical simulations achieved with the FLAMME_S code.

Aerosols, composed by soot particles, play an important role in thermal exchanges within the vessel: due to their high concentration (typically between 1 and 2 g.m⁻³), they behave as a thermal screen for the direct radiation from the flame to the targets (walls, equipments...). Because of their emissivity and their temperature, aerosols exchange heat to the surrounding by radiation and convection with an equivalent ratio.

4.4 The walls

During the steady state phase of the fire experiments, about 70% of the heat release rate of the fire is absorbed by the walls. The measurements also allow us to determine convective heat transfer coefficients between the hot gases and the walls, used in the predictions carried out with the FLAMME_S code.

For pool fires located close to a wall, in which there is a feedback effect between a flame and a vertical wall, total and radiative heat fluxes were measured at several elevations above the burning pan. The Figure 7 shows how the maximum wall total heat flux depends on the fire heat release rate. Experimental values are greater than predictions based on BEYLER's model [12] for a fire power greater than 500 kW.
4.5 The ventilation

The knowledge of the behaviour of a ventilation network when a fire occurs in a compartment is one of the main axes of the IPSN research in the field of fires. IPSN studies showed that fire and ventilation are strongly coupled, whatever the ventilation mode (natural or controlled).

Our experimental results on fire under natural ventilation are not sufficient. At the opposite, configurations with a controlled ventilation were much more investigated, especially with large pool fires inside a room whose scale is of the same order of magnitude than in real industrial installations.

Concerning the ventilation network, specific non equilibrium due to pressure rise into the vessel because of the ignition of the pool has been quantified in terms of flow rate and temperature in each pipe and of pressure at each node of the network. The Figure 8 shows the pressure evolution inside the vessel and at several nodes of the network.

Figure 7: Comparison between Beyler’s model predictions and experimental data

Figure 8: Pressure versus time at several locations of the ventilation network
During the pressure peak due to the ignition (see Figure 5), the inlet flow rate decreases (and reverses in some cases), and the outlet flow rate significantly increases (Figure 6). This phenomenon was very well shown during a 3.2 m² fire experiment whose pressure and flowrate curves are given in Figure 14 to Figure 17 (the oscillations after 2100 seconds have not to be considered).

Many points remain not enough investigated and require further research: influence on the fire behavior of the inlet and outlet pipe locations in the vessel, shift of the flame towards and inside a ventilation pipe, ignition of the combustion products into a network pipe, ...

4.6 The exposed materials

The study of the behavior of exposed material concerns only electrical cables. In one experiment, called PEPSI 1 (Figure 9), were involved 4 cables trays, located horizontally within the vessel just above the flame zone (cables tray n°1), in the plume at 0.35 m from the ceiling (n°2), outside the plume also at 0.35 m from the ceiling (n°3), face to the flame at 1 m far from the edge of the pool (n°4), and one vertical cables tray located at a distance of 1 m from the edge of the pool (n°5). Each cable tray was composed by 5 types of Cl cable (answering to the NFC 32.070 specification): 3 x 16 mm², 3 x 6 mm², 2 x 35 mm², 7 x 1.5 mm², 2 x 0.5 mm². These cables were electrically supplied, except concerning cables tray n°5. The fire, a 1 m² oil pool, took place in a vessel of 400 m³ in volume (the PLUTON vessel described before), whose ventilation flowrate was about 2000 m³.h⁻¹.

![Diagram](image)

Figure 9: PEPSI 1 experiment, general view

Concerning the cable trays behavior, we observed that, whatever the type of cable, the origin of the failure is always the same: short circuit between two or several wires. In a second phase only, a short circuit between two or several wires and the sheathing can occur. During this test, only the cables on the tray n°4 did not fail.
Furthermore, during this experiment, the flame remained vertically during 2.8 minutes after the ignition, then slanted until the end of the fire. So, due to this slant of the flame occurring a few minutes after ignition, the cable tray n°3, located at the beginning outside the plume, was damaged before the cable tray n°2.

Figure 10 gives for each type of cable the experimental curve of the failure temperature versus failure time. As a rule, the order of failure of the cable trays was not 1, 2, 3 as foreseen, but 1, 3, 2 as a consequence of the slant of the flame and the deviation of the plume.

![Figure 10: PEPSI 1 experiment, Failure temperature versus failure time, for each type of cable](image)

4.7 Qualification of the FLAMME_S computer code (version A3)

A part of acquired knowledge concerning fire behaviour is integrated into the FLAMME_S computer code which is developed simultaneously to the experiments and their interpretation. But the modelling of some phenomena is not yet satisfactory (flame spread, plume in confined room ...) or does not exist (burning rate, combustion products and radioactive aerosols transfers within the room ...). The improvement and the development of some of these models will require specific accurate measurements during experiments: mass transfer and gas flow into the room (including the plume), heat transfers between the flame and the pool fire ...

This part is described more precisely in the specific paper on fire modelling in IPSN computer codes [10].

5. FUTURE PROGRAMS

The strategy of the IPSN fire research for the future will comply with these two objectives:

- the follow-up of the programs directly related to safety concerns,
- the development or continuation of more basic studies on the fire phenomenology.
The research will continue to involve both code development and experiments. In order to better fulfil the objectives, collaborations with universities and other research organisations working on the combustion or on the fire will be increased. These collaborations will concern universities for basic studies and other French and foreign organisations for safety codes developments and experiments related to safety questions (INERIS (France), STUK (Finland), NUPEC (Japan), ...).

On the experimental point of view, the support to the safety analysis will consist in performing semi-analytical and global experiments answering in a short or medium term some safety questions. This experimental program is composed of several items and mainly split up into two parts. A set of experiments will be performed in the framework of the fire PSA concerning the French PWR 900. Another set of experiments, carried out in collaboration with COGEMA, concerns fire safety questions related to COGEMA's laboratories and reprocessing plants.

A complementary important program concerning the behaviour during a fire of ventilation (Very High Efficiency Filters, ...) and confinement (doors, fire shortcut valves, ...) equipments as well as the resuspension of radioactive materials is carried out by IPSN, but is not described in this paper.

The following paragraphs provide a brief overview of the R&D related to safety concerns and of the basic research.

5.1 The codes in support to safety analysis
The FLAMME_S code will still remain for a long time the code used for safety analysis. The development and qualification of FLAMME_S will be mainly based on the new experimental results obtained from the programs carried out either in support to the safety analysis or in the framework of basic researches.

The future developments concerning IPSN codes are described in detail in a specific paper within this workshop on Fire Risk Assessment [10].

5.2 The experimental programs in support to the Fire PSA
A set of experimental programs are planned in order to assess the answers provided by the Fire PSA level 2 in the following items:

- **Electronic cabinets.** The program will be composed of a first semi-analytical part aimed at qualifying a simple electronic cabinets combustion model which has been already assessed on partial experimental results obtained by VTT. These tests will be performed on a cabinet-like box, submitted to various natural ventilation conditions, containing a well characterised solid combustible (PMMA for instance) in various vertical arrangements and volumetric densities. Few tests involving real electronic cabinets will conclude this program which will be carried out from 1999 to end-2000. A special attention will be put on the determination of the heat release rate since it has been demonstrated that the combustion of this kind of equipment is mainly a transient phenomenon.

- **Fire propagation through several locals.** This program aims at studying first of all the consequences of a fire, located in a room, on neighbouring rooms and on the ventilation, furthermore the interaction between a fire and the ventilation: thermal propagation, pressure effects, smoke and fire spread, consequences of the fire on the ventilation and room equipments, management of the ventilation during a fire. To do this, a new large scale experimental facility will be built in 1999. It will be representative for the LWR rooms and for laboratories and reprocessing plants. It will be composed of three rooms (L = 6 m, l = 5 m,
h = 4 m), a common corridor connecting the three rooms and a room located above the third room and the neighbouring part of the corridor (Figure 11). The rooms will be connected by a ventilation network and by openings such as doors. The experimental program, already defined, will start in 2000.

Figure 11: DIVA "multi-room" facility

- **Cable trays.** A complementary set of experiments planned beyond 2001 will allow to qualify the modelling of the fire propagation on a cable tray.
- **Insulating material either dry or impregnated with oil.** Analytical experiments planned during 2000 and 2001 will allow to determine the auto-ignition of the insulating materials used in the LWR and the combustion properties of such a material.

5.3 **The programs performed in collaboration between IPSN and COGEMA**

The collaboration between IPSN and COGEMA in the framework of the Fire Common Interest Program will continue in 1999 and 2000. The safety questions addressed in this program are related to COGEMA’s nuclear installations.

- **Fire interacting with a wall.** Some room configurations make highly probable that a possible fire will be located either in an angle or in a corner of the room. The purpose of this program is to complete the previous set of experiments carried in 1997 by additional experiments focusing on the case of a fire in a corner. It is expected to improve the plume and flame models, presently applied for a fire located in the centre of a room, to the situation of a fire located near and therefore interacting with a wall.
- **Fire of solid materials.** The goal of this program is to complete the present solid fuel database of the FLAMME_S computer code by including solid materials composing a glove box. A bibliographical study will be possibly complemented by small and medium scale experiments in order to characterise such materials.
5.4 Basic research on fires

As far as basic studies on fires are concerned, both modelling and analytical experiments will be tightly connected and carried out in parallel. The modelling will concern ISIS code. Combustion models and efficient numerical methods (multi-grids, multi-domain approaches, local meshing refinement) will be implemented in ISIS.

The present status of the FLAMMIE_S assessment highlights the main sensitive physico-chemical phenomena which have a strong influence on the fire behaviour. The analytical experimental studies that must be performed in priority concern the burning rate, the characterisation of the flame and the plume as well as the interaction between the fire and the ventilation. The experimental results will allow to develop new models (for instance an accurate plume model for a confined local, a more general combustion model) and better assess or improve present models. A collaboration with the universities will be developed in the frame of these basic studies (fundamental combustion modelling, advanced numerical methods for the solution of turbulent Navier-Stokes equations) in order to benefit from complementary studies and scientific abilities.

It is foreseen to precisely define the basic experimental and modelling researches by the mid of 1999 with the prospect to start the experimental work of the basic researches by 2000.

6. CONCLUSION

The fire is a main industrial risk which can lead to important damages on the installations, the environment and the persons. A special attention has to be put on the analysis of the fire risk in nuclear installations since a fire could lead to the release of radioactive materials inside the installation and even outside, into the environment.

There is so far a large amount of results (both in an experimental and modelling points of view) available from the researches carried out by IPSN for more than ten years in the area of the "classical" fires. These results are already used within the frame of the safety analysis of the nuclear installations (LWR, laboratories and reprocessing plants). It has to be highlighted that the computer code FLAMMIE_S assessed in particular on the IPSN experimental data is used to carried out safety analysis, especially in the framework of the underway fire PSA related to French PWR 900 MW.

Nevertheless, as underlined in this paper, a lot of models still need a complementary qualification and some phenomena exhibited during experiments are either not yet modelled or not well understood or show the limitation of the present modelling. Moreover, a lot of phenomena or configurations related to the fire remain to be studied especially on an experimental point of view. The future programs presented in this paper only cover a small part of the identified necessary research.

The research programs will continue to help in evaluating the risk connected with a fire in the nuclear installations. The coexistence inside the IPSN of safety evaluation and research activities favours a tight dialogue during the definition of the programs and during the code development and validation. This coexistence allows also a quick and efficient feedback between new safety analysis questions and new scientific knowledge gained by the research.

It has to be highlighted that the future research will be also carried out in more fundamental aspects related to the modelling (3-D multi field computer code ISIS) and to the experiments. The development of tight collaborations with universities and other research organisations working on the fire phenomenon will be necessary to succeed in this area.
Finally, since the fire is not a nuclear phenomenon, the main knowledge and developments gained by means of the IPSN research programs can also be used for the analysis of the fire risk in non nuclear installations.

7. FIGURES

Figure 12: Schematic diagram of the fire room inside the JUPITER facility (3600 m$^3$)

Figure 13: The main steps of a real scenario for a ghosting flame:

Step a: $t = 90$ s (ignition of the pool fire) to 300 s.

Step b: $t = 300$ s to 600 s.
Step c: $t = 600\ s$ to $750\ s$.

Step d: $t = 750\ s$ to $1720\ s$.

Step e: $t = 1720\ s$ and so on.
8. **APPENDIX: FLIP 3 experiment**

The main characteristics of the FLIP experiment were the following: a 3.2 m² pool fire involving 160 litres of a TPH/TBP mixture (kind of kerosene) in the "PLUTON" vessel (400 m³), with a ventilation rate of 3 h⁻¹ before the ignition.

Over-pressure (Figure 14), reverse inlet flow rate (Figure 15), outlet flow rate increase (Figure 16) and oscillations at the end of the fire (after 2100 s on the curves) are the outstanding events of this experiment. The origin of these oscillations remains unexplained.

![Figure 14: FLIP 3 experiment, pressure within the vessel](image1)

![Figure 15: FLIP 3 experiment, inlet flow rate](image2)
9. REFERENCES

10. Fire modelling in IPSN computer codes, C. Casselman, L. Audouin, B. Tourniaire, Workshop on Fire Risk Assessment, HELSINKI, 29 June -1 July 1999
DEVELOPMENT OF PROBABILITY SAFETY ASSESSMENT METHODOLOGY FOR FIRE EVENTS IN CANDU PLANTS

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ABSTRACT

Atomic Energy of Canada Limited (AECL) initiated a Generic Probabilistic Safety Assessment (PSA) program in 1998, with the mandate to upgrade the expertise and technology base to support the CANDU products in the future. It expands the scope of earlier PSA work to include external events, detailed human reliability analysis, common cause failure analysis. This paper describes the development of a PSA methodology for fire events in CANDU plants, as part of the overall Generic PSA program, which will be applied to current and future CANDU designs. The methodology was selected to be internationally acceptable and with the flexibility to accommodate changes in design and equipment.

The elements of the Fire PSA include the following:

- Development of fire initiating event frequencies
- Identification of plant characteristic relevant to fire events
- Fire scenario analysis, including qualitative and quantitative screening
- Quantification of the fire risk in terms of severe core damage frequency (SCDF)
- Recommendations for design changes to satisfy PSA objectives.

With the assistance of experienced external consultants, AECL has built a generic experience based CANDU fire events database to estimate initiating event frequencies. The database consists of fire events from US Light Water Reactors, as well as fire events that have occurred in CANDU plants, screened for applicability to the current CANDU design. The data related to plant systems, including the safety related and PSA credited equipment, cables, and their locations, has been assembled in a comprehensive reference CANDU 6 plant characteristics database. As part of the ongoing acquisition of the methodology and technology phase, a one-week fire PSA walkthrough training exercise was conducted at a CANDU plant.

The fire vulnerability analysis for the CANDU 6 design will be started in early 1999, using information gathered up to this point, including the fire initiating event frequencies, the fire zone data, the lists of equipment and fire hazards, and the walkthrough information. This activity involves the identification of fire scenarios for each room and/or area, qualitative and quantitative screening of each area, and the calculation of plant damage and core damage frequencies using modified system models for internal events. The COMPBRN code will be used to model fires in selected areas, and determine their impact on PSA credited systems and components.

1. INTRODUCTION

The probabilistic design approach has long been a feature of CANDU plants, since reliability targets for the Special Safety Systems (two shutdown systems, emergency core cooling, and containment) were set by
the Canadian regulatory authority in the mid-1960s, in conjunction with different dose limits for accidents involving loss of one of the Special Safety Systems. During the 1970s, an analytical technique was developed called “Safety Design Matrices”, which used event trees and fault trees to evaluate safety support systems. This technique was the forerunner of the Probabilistic Safety Assessment (PSA) methodology used for recent CANDU plants, which applies current risk assessment methods and computer codes to an extensive range of internal events.

Last year, Atomic Energy of Canada Ltd (AECL) embarked on a program to upgrade the PSA capability to include the latest methods to address external events (fire, seismic, and flooding), common cause events, human reliability, and severe accident modeling. This program is called the Generic CANDU Probabilistic Safety Assessment Program [1].

This paper outlines the development of the PSA methodology for fire events, as part of the Generic CANDU Probabilistic Safety Assessment Program at AECL.

2. DEVELOPMENT OF CANDU FIRE PSA METHODOLOGY.

The overall objectives for the selection of the methodology for the fire events PSA was that it must be internationally acceptable (as CANDU plants are being marketed globally), able to accommodate the expert judgment of analysts and designers, and flexible enough to easily deal with changes in design and improvements in methodology. In addition, the documentation produced must be easily auditable by clients and regulators, generally following the approach outlined by the International Atomic Energy Agency in Safety Series No. 50-P-4 [2].

The development of the methodology consisted of a planning phase, a methodology development phase, and then a trial application to typical CANDU designs. The initial planning phase included preparation of budgets and schedules for a three year program, the selection of experienced consultants to provide training and expertise, and the staffing of the fire PSA team. The definition of the fire PSA program included a decision to follow a comprehensive risk assessment approach (i.e. based on event trees and fault trees) rather than using a more deterministic approach, such as the EPRI Fire Induced Vulnerability Evaluation (FIVE) technique. To assist in the efficient acquisition of the basic fire PSA techniques and tools, several consultants experienced in the US Individual Plant Examination of External Events (IPEEE) were interviewed. The consultant with the expertise and risk assessment tools and approach felt to be the most compatible with the objective of the CANDU fire PSA program was selected. This consultant provided the initial training for the fire PSA team (as well as for a number of other related resource staff), the fire events database for US plants, ongoing assistance with the application of the tools, and a peer review of the methodology documents.

The methodology was developed in parallel with its trial use on a typical CANDU 6 plant design, which provided a “hands-on” aspect to the development process. The methodology and preliminary design assessment phase, shown in Figure 1, will be followed by a detailed analysis and final design assessment phase. Problems encountered during the application of the methods and computer tools were resolved and factored into the methodology documentation. The methodology development process consisted of the following basic elements:

- Calculation of fire event frequencies for typical plant locations and equipment types
- Identification of plant characteristics relevant to fire events
- Development of a plant walkthrough process to confirm plant characteristics
- Fire scenario analysis, including qualitative and quantitative screening
- Quantification of the fire risk in terms of severe core damage frequency.
3. **FIRE INITIATING EVENT FREQUENCIES**

The frequency with which fires occur in a nuclear power plant is a key parameter for the analysis. To ensure that the methodology is conservative and internationally acceptable, it was decided that the fire initiating event frequencies should include experience beyond that of Canadian operating experience, so a database of US PWR and BWR fire events was acquired. The events were screened for applicability to CANDU plants, and entered into a generic CANDU Fire Initiating Events Database. Events occurring in the CANDU plants operated in Canada were then added to the database, and fire frequencies were calculated.

**Definition of Fire Events**

Fire events were defined as those characterized by the presence of flame, burning, or smoldering which has the potential for growth and propagation to cause a reactor trip and/or damage to safety related or PSA credited equipment. In selecting events for the database, the potential impact of the event on plant safety during reactor operation and the relevance of an event to a CANDU plant were considered. Generally, events with smoke and no fire were not included, as well as those involving arcing, sparking, explosions and other short bursts of energy failing to result in ignition. Such events were carefully considered, however, and if it was obvious that the event could have led to a fire if not detected in time, then it was included. Explosion events involving mechanical effects but no fire were not included.

**Frequency of Fire Induced Initiating Events**

To establish fire initiating event frequencies for PSA purposes, the specific plant operating history (on power and shutdown durations) is required, and the fire initiating events must have been consistently reported throughout that period. This information was available for US and Canadian plants, but was not available to us for off-shore plants, so only North American experience was used to calculate frequencies. The data for CANDU fire events was extracted from the CANDU Owners Group (COG) Significant Event Report Exchange database. Data for the US fire events was purchased from the fire PSA consultant. The US LWR events were considered applicable to CANDU if they involve similar types of equipment in systems providing functions similar to those in CANDU plants. The information was assembled in the CANDU Fire Initiating Events Database.

It was recognized that there is variability among the different plants in areas such as specific design, layout, maintenance procedures and practices, safety culture and fire protection features. However, the general level of technical standards for component manufacturing, safety regulations and requirements, personnel training, plant maintenance practices, etc. are not vastly different for the nuclear industry in North America. Moreover, fire events are evaluated individually at the component/equipment level, minimizing the impact of different plant designs. Therefore it was considered appropriate to use US LWR data for the CANDU fire events database.

**Categories of Fire Initiating Event Sources**

The fire initiating events in the CANDU Events Database are assigned to a number of categories, based on component types (e.g. cables, motors, pumps, etc.), areas in the plant with common characteristics for several plants (e.g. main control room), and transient fires (e.g. fires caused by welding and cutting, transient combustible materials, human errors).

Table 2 shows the list of twenty-five categories chosen to represent fire event sources contributing to fire risk for CANDU plants, and the numbers of events identified for each category, in both the operating and
the shutdown plant state. The definition of these categories is tailored according to specific assumptions, as follows:

a) Category 4 (Main Control Room) relates to all fires occurring in the MCR, regardless of the cause and/or equipment involved (transient fires, fires in control panels, cabinets, etc.). This approach avoids the need to comprehensively list and sum the individual fire initiation frequencies for all fire sources for this area.

b) Category 5 and Category 8 refer to Digital Control Computers (DCC) and to D2O Recovery Dryers respectively, which are specific to CANDU plants and are not present in US LWR plants. Therefore, the calculation of fire frequency considers only the CANDU data and plant operating history.

c) Category 9 (Hydrogen fires) includes fire events associated with hydrogen vessels and supply systems, but excludes turbine-generator hydrogen fires.

d) Category 12 (Pumps) refers to fire events related to pumps only; fires occurring in pump motors are assigned to Category 14 (Motors).

e) Fires in Junction Boxes are included in Category 16 (Power and Control Cables).

f) Category 20 (Turbine-Generator) includes all the fire events related to the T/G group (i.e. T/G exciter, oil, hydrogen fires).

In some cases, classification of an event in one or another category may not be obvious due to limited knowledge of the individual plant and/or limited or incomplete description of the event. This requires judgments and analysis of the event on a case by case basis.

**Screening Criteria**

Since LWR plants may contain systems which do not exist in CANDU plants, the events recorded in LWR plants are screened for applicability to CANDU plants. For example, fires in the following LWR systems are screened out: fires in equipment that does not exist in CANDU plants (e.g. turbine driven pumps), fires in facilities that do not exist in CANDU plants (e.g. fires in the standby gas treatment system in BWR plants, fire associated with the off gas system in the radwaste building).

Events are also screened out based on definitions of fire events. This includes events with smoke and no flame (smoking bearings, belts, unless a detailed examination reveals that open flaming would likely occur if the event is allowed to progress), arcing, sparking, shorts and explosions that are short bursts of heat energy but fail to result in ignition (e.g. crankcase explosions in diesel generator sets, since they are contained and cannot propagate).

Events that involve components and systems that are not modeled in the PSA and are not located in areas that contain safety related or PSA credited equipment are screened out. As a result, the following categories of events are screened out: fires in administrative buildings, main entrance area, guard house, temporary buildings, trailers, and outside the fenced area (e.g. forest or grass fires, even if they result in failure of transmission lines, as these events would be captured in the loss of offsite power frequency).

The raw data obtained from the COG database also contains events that do not involve a fire, and are therefore screened out, such as fire code violations, false fire alarms and failures in fire fighting systems.

During the screening process, an evaluation is made about whether the fire can occur during reactor power operation and/or at shutdown. This is necessary because there may be significant differences between the
plant on power and the plant at shutdown, in areas such as plant configuration, system operation, maintenance activities, personnel access to various areas in the plant. As a result, the frequency of the fire initiating events for the plant with reactor on power may be different than when the reactor is at shutdown.

**Screening of LWR Fire Events**

The US fire events database contained 783 fire event records spanning the period from January 1, 1964 to January 26, 1994. It provided station data on the date of first criticality, date of first commercial operation, average availability, unit years, total unit years for multi-unit stations, unit power years, and total power years for the multi-unit stations.

Following the Browns Ferry fire in 1975, there were a significant number of design upgrades carried out at the US LWRs, such as the use of better materials, improved physical separation and fire protection measures, changes in plant operating philosophy with strict administrative controls, and better defined event reporting requirements. For this reason, the records prior to January 1, 1980 were not considered relevant, and events prior to that date were screened out. For plants commissioned after that date, only events that occurred after the beginning of commercial operation were considered. A cut-off date of December 31, 1992 (or the date of termination of commercial operation, if this happened prior to December 31, 1992) was selected for the fire events considered in the study. Based on the period from 1980 to 1992, the US fire database was reduced to 487 records.

The results of the screening process are presented in Table 1. The breakdown of the 487 events in the US database for the 13 year period ending December 1992 is as follows: 124 are not fire events, 134 are screened out based on the screening criteria, and 229 are fire events.

**Screening of CANDU Fire Events**

The COG Significant Event Report Exchange database contains approximately 15,000 records, and is searchable by many different criteria. The query for fire events resulted in a collection of 337 events, but not all the events were true or actual fire events, since they also included fire code violations and equipment failures in fire protection systems. The CANDU plant sites that were considered were the single-unit CANDU 6 plants, (Point Lepreau and Gentilly 2), and the multi-unit Ontario Hydro stations (Pickering A, Pickering B, Bruce A, Bruce B and Darlington).

The results of the screening process are presented in Table 1. The breakdown of the 337 events from the COG database for the period from first commercial operation of each plant to December 1997 is as follows: 230 are not fire events, 33 are screened out based on the screening criteria, and 74 are fire events.

The calculation of operating years for CANDU plants for the fire frequency calculation was based on the cumulative gross capacity factor since in-service, which provides a conservative value of fire frequencies during on power operation. This will be upgraded later to consider the total inservice period less outage durations.

The fire events from the CANDU plants remaining after screening were added to the US LWR events remaining after screening to form the generic CANDU Fire Initiating Events Database. The database has a combined total of 303 fire events from both LWR plants and CANDU plants. The breakdown of the 303 Fire events is as follows: 20 Power Only events, 28 Shutdown Only events, and 255 Anytime events.
Calculation of Fire Initiating Events Frequencies

The determination of the generic fire frequency due to potential fire sources in a CANDU plant involves statistical processing of the information on fire events included in the CANDU Fire Initiating Events Database. The calculations are performed with data analysis software purchased from the US consultant, using the first stage of a Two Stage Bayesian Update option (the second stage would be used for the fire frequency analysis of a specific plant).

The fire events in the database represent US LWR operating experience from 70 plants with a cumulative service life of 1149 reactor years, and for the CANDU plants it represents 7 plants (22 units) with a cumulative service life of 332.7 reactor years. With the capacity factors or outage durations considered, a total of 1054.8 power operating years (814.4 for US LWR plants and 240.4 for CANDU plants) was used in the fire frequency calculation.

The resulting calculation provides the fire initiating event frequencies for a generic CANDU plant design. The mean frequency of the distribution for each category is given in Table 2. The generic fire frequency represents how often (events per year) a fire caused by each particular category occurs in the plant during one year of operation with the reactor on power.

4. PLANT CHARACTERISTICS DATABASE

The plant characteristics relevant to fire were assembled in a database consisting of linked tables using a modern database software that is widely available. The database includes fire zone data, a list of safety related and PSA credited equipment, and a list of fire hazards. The database tables are structured in such a way that they provide easy collection, organization, grouping, storage, and retrieval of data. The tables are kept to a reasonable size so the data can be easily managed and efficiently used.

Fire Zone Data

The fire characteristics of a CANDU plant are evaluated in terms of “fire zones”, which are small sections of a plant that can be treated as a unit for evaluation purposes. A fire zone usually corresponds to a single room, but can consist of two or more rooms that are spatially linked. A fire zone is not necessarily bounded by physical barriers; spatial separation may be used between fire zones. A “fire area” is one or more fire zones contained within a defined set of fire barriers.

The data tables include the following types of information:

a) Rooms: The data includes the room number and plant building designation, a description of the room function, the location of the room (elevation), and the fire zone and fire area in which the room is located.

b) Zone Exposure: The data includes the zone number, other zones to which the zone is exposed, the fire propagation pathway, and the fire barrier ratings.

c) Fire Detection and Suppression: The data includes a description of the devices in each room, the number of devices in the room, whether the device is automatically initiated, and the hazard category of the device (where the device may also be the source of a fire).

d) Fire Loading: The data includes the combustibles, heat load, fire hazards and ignition sources in each room.

e) Systems: The data includes a list of systems, their identification number (subject index), whether it is safety related or PSA credited, and its function.

f) Equipment: The data includes a list of equipment in each room, its identification number (subject index), tag number, description, hazard category, and a cable termination identifier.
g) Cables: The data includes the cables in each room, the cable identifier, the length of cable in each room, the raceway identifier (conduit, tray, etc.), and hazard category.

The data collection activity for the fire PSA is very time consuming, so existing databases for equipment, cables, electrical devices, etc. are used wherever possible. For future plants, engineering databases assembled during the design and procurement phases will be more compatible with the needs of the fire PSA, allowing linkages directly to the database containing the necessary information. For example, the cable routing database will include the room identification and cable lengths within the room, avoiding the manual collection of this data from cable tray routing drawings.

5. PLANT WALKDOWN

The walkdown is a very important activity to be undertaken by the fire PSA analyst. The walkdown is necessary to confirm the accuracy of the information assembled using design documentation, to obtain additional details about the plant (field installed equipment, maintenance practices, operating procedures, handling of transient combustibles), and to identify interactions between equipment and areas that may affect the propagation of a fire. This past year, a training exercise was held at an operating CANDU 6 plant. The team assembled for the exercise included an experienced trainer from the fire consultant, a team leader, team members including expertise in plant layout, electrical equipment, safety design, process system design, PSA analysts, and plant operations staff. The team visited all areas of the plant (except the reactor building), practicing note taking and discussing the relevant fire hazards and plant characteristics in each area. Notes were taken using a predetermined format, and a digital camera was used to record features of interest in each area, for later reference in the vulnerability analysis phase.

6. FIRE VULNERABILITY ANALYSIS

The fire vulnerability analysis, which has just started for the CANDU 6 generic design and will start later this year for the new CANDU 9 design, provides an understanding of the impact on plant safety from internal fire initiating events, and quantifies the risk in terms of severe core damage frequency. To calculate the core damage frequencies, the fire vulnerability analysis uses information on generic fire initiating event frequencies for the various categories of sources, plant layout (fire areas, fire zones, fire barriers), location and interaction of components and equipment in the plant, as well as modified internal event PSA plant models.

The fire vulnerability analysis consists of the following steps:

a) Determination of fire initiating event frequencies in each fire area.
b) Definition of fire scenarios in each fire area.
c) Qualitative screening of fire areas requiring no further consideration.
d) Quantitative screening of fire areas, and calculation of plant damage and core damage frequency using PSA system models.
e) Refinement of the fire scenarios in areas having a significant impact on the SCDF, using a fire growth analysis computer code to obtain a more realistic evaluation of equipment damaged, and recalculation of plant damage and severe core damage frequency using PSA system models.

The qualitative and quantitative screening of the fire scenarios is performed in order to reduce the number of fire scenarios for which a detailed fire hazard analysis is performed. The screening analyses assume a worst case impact of fire in the areas to which it is applied. Scenarios which are not screened out are retained for detailed fire hazard analysis, which provides a more accurate determination of the impact of the fire on plant safety and severe core damage frequency. The COMBRN IIIe computer code [3] will be used to calculate fire propagation and to determine the time interval between fire initiation and damage to
critical equipment. The PSA plant modeling for both screening analysis and detailed analysis is based on the internal events PSA model, suitably modified to reflect systems and components damaged by the fire.

7. CONCLUSION

The Generic CANDU Fire PSA Program has completed its first phase, the development of the fire PSA methodology. The methodology is flexible, with the capability to be applied to any CANDU plant design, and structured so new information and changes can be easily incorporated. Training materials and documentation have been produced which will ensure that a consistent approach is taken on all plants to which it is applied. The methodology is ready to be applied to future CANDU plant designs.

8. REFERENCES

Table 1
Generic CANDU Fire Events

<table>
<thead>
<tr>
<th>Event</th>
<th>US LWR Plants¹</th>
<th>CANDU Plants²</th>
<th>Generic CANDU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total</td>
<td>487</td>
<td>337</td>
<td>N.A.</td>
</tr>
<tr>
<td>Not a Fire</td>
<td>124</td>
<td>230</td>
<td>N.A.</td>
</tr>
<tr>
<td>Screened Out</td>
<td>134</td>
<td>33</td>
<td>N.A.</td>
</tr>
<tr>
<td>Power Only</td>
<td>17</td>
<td>3</td>
<td>20</td>
</tr>
<tr>
<td>Shutdown Only</td>
<td>21</td>
<td>7</td>
<td>28</td>
</tr>
<tr>
<td>Anytime</td>
<td>191</td>
<td>64</td>
<td>255</td>
</tr>
</tbody>
</table>

Note 1: Based on the US Fire Events Database.
Note 2: For period between January 1, 1980 to December 31, 1992.
Note 3: Based on raw data on search of the COG Database.
Note 4: For period from first commercial operation to December 31, 1997. It is noted that the raw data also contain events that do not involve a fire, e.g. fire code violations, false fire alarms, failure of fire fighting equipment, etc.
<table>
<thead>
<tr>
<th>Category ID</th>
<th>Category Name</th>
<th>Mean Frequency (events / plant / year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Battery</td>
<td>1.29E-03</td>
</tr>
<tr>
<td>2</td>
<td>Battery charger</td>
<td>2.35E-03</td>
</tr>
<tr>
<td>3</td>
<td>Inverters</td>
<td>1.01E-03</td>
</tr>
<tr>
<td>4</td>
<td>Main control room</td>
<td>3.06E-03</td>
</tr>
<tr>
<td>5</td>
<td>Digital control computers</td>
<td>4.15E-03</td>
</tr>
<tr>
<td>6</td>
<td>Diesel generator sets</td>
<td>2.25E-02</td>
</tr>
<tr>
<td>7</td>
<td>HVAC equipment</td>
<td>3.26E-03</td>
</tr>
<tr>
<td>8</td>
<td>Dryers</td>
<td>5.27E-03</td>
</tr>
<tr>
<td>9</td>
<td>Hydrogen fires</td>
<td>7.50E-03</td>
</tr>
<tr>
<td>10</td>
<td>Logic and protection cabinets</td>
<td>1.82E-02</td>
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<tr>
<td>11</td>
<td>PHTS pumps</td>
<td>3.88E-03</td>
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<td>12</td>
<td>Pumps</td>
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<td>13</td>
<td>Motor control center</td>
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<tr>
<td>14</td>
<td>Motors</td>
<td>1.06E-02</td>
</tr>
<tr>
<td>15</td>
<td>Motor generator sets</td>
<td>1.34E-03</td>
</tr>
<tr>
<td>16</td>
<td>Power and control cables</td>
<td>1.26E-02</td>
</tr>
<tr>
<td>17</td>
<td>Low voltage switchgear</td>
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<td>18</td>
<td>High voltage switchgear</td>
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<tr>
<td>19</td>
<td>Standby generators</td>
<td>1.29E-02</td>
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<td>20</td>
<td>Turbine-generator</td>
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<td>21</td>
<td>Main unit transformer</td>
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<td>22</td>
<td>Transformers</td>
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<td>23</td>
<td>Human error</td>
<td>1.89E-02</td>
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<td>24</td>
<td>Cable fires caused by welding and cutting</td>
<td>1.71E-03</td>
</tr>
<tr>
<td>25</td>
<td>Transient fires caused by welding and cutting</td>
<td>2.92E-02</td>
</tr>
</tbody>
</table>
FIRE PSA - APPLICATIONS AND INSIGHTS

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Abstract

Since the early years of probabilistic safety assessments, quantitative methods have been used to assess the contribution to the overall risk from potential fire incidents. Today, fire risk analysis has been completed for a significant portion of the nuclear power plants in the U.S. These studies indicate, despite the variations in plant layout and in methods and data employed for fire risk analysis, that fires are important risk contributors. From a review of a number of these studies, some insights can be noted. The influence on the final conclusions of fire risk studies of selected methodology (i.e., PRA vs. FIVE), plant data, parameter values for fire propagation analysis, human error analysis methodology and several other attributes of a fire analysis has been investigated and insights gained are presented.

1. INTRODUCTION

Fire risk analysis has been conducted for a large number of nuclear power plants in the U.S. Since 1980, several utilities have conducted Probabilistic Risk Assessments (PRAs) that have addressed the contribution of internal fires to plant risk. In addition, in the last few years, as part of their compliance efforts for the Independent Plant Examination for External Events (IPEEE) [1], most utilities have elected, although not specifically required in Reference [1], to use fire risk analysis methodology for that purpose. From a review of all the above studies, a number of insights can be gleaned that may shed some light on the importance of fire events to plant risk and on the methods and data that were employed to analyze the risk. This article is organized around the various aspects of a typical fire risk analysis and includes observations and comments on those issues.

2. RISK SIGNIFICANCE OF FIRE EVENTS

The total core damage frequency for fire events, reported in the studies reviewed or prepared by the author, range between $10^{-6}$ to $10^{-4}$ per reactor year. The total core damage frequency of each plant attributed to fire events can be considered as small. However, in the majority of cases, the contribution of fires to the overall plant risk is significant. The fire occurrence frequencies for individual fire scenarios is typically comparable to many internal initiating events; but unlike the internal events, it can simultaneously disable several pieces of equipment. Furthermore, a fire can influence operator performance and increase the human error probability of recovery actions.
There is considerable variation among plants in terms of fire risk contributors. This is influenced by plant layout and the underlying assumptions employed in conducting the analysis. Some of the variations that may be worth noting include treatment of transient combustibles in areas that are normally closed, screening of cable chases that carry a large number of cables from various systems/trains because of low combustible loading, and the wide variation in parameter values used for fire propagation or suppression modeling.

3. **SIGNIFICANT FIRE SCENARIOS AND RISK CONTRIBUTORS**

Fire risk contributors can be presented in terms of location of the fire, and equipment/systems affected by the fire. The control room and cable spreading room are two most reported significant fire risk contributors. Although there are no clear patterns among the risk studies to draw any clear conclusions, it can be stated that buildings that house non-safety related equipment and cables, with the exception of the turbine building, have been found to be of little or no fire-risk significance.

The dominant sequence for control room fires is typically a fire in a vital control panel that leads to control room evacuation and failure of the operators to successfully use the alternate shutdown panels. The cable spreading and high-voltage switchgear rooms have been mentioned in approximately one-half of the risk studies as dominant fire risk contributors. Since cable spreading rooms contain almost the same set of control and instrumentation circuits as those in the control room, one would expect the cable spreading rooms to be identified as risk significant for as many cases as where the control room was identified as a dominant risk contributor. Differences can be attributed to the fact that several risk studies have screened out the cable spreading room by either concluding that fire scenario frequencies for those rooms are too small, or by using qualitative arguments regarding fire ignition possibility in the room. Only in a small number of plants there are multiple and well separated cable spreading rooms serving the control room. Typically, in such cases, a cable spreading room may only contain one train of the vital circuits, and thus, the fire events in those rooms are found to be insignificant contributors. Operator actions are a key element of fire scenarios associated with control room or cable spreading rooms. This is further discussed below.

The service water and component cooling-related areas have also been reported in several risk studies as important fire risk contributors. Various auxiliary/reactor building and turbine building areas have been included in these lists as well. However, in the case of the latter two buildings, there is no overall pattern, which can be attributed to lack of common patterns among the auxiliary and reactor buildings across the plants licensed in the U.S. In the case of turbine building assessments, in most cases the dominant fire scenarios are attributed to a compartment or a localized area that is part of the turbine building and not a large portion of the building.

4. **LINK WITH INTERNAL EVENTS PSA**

To establish the risk contribution of fire events, the internal events PSA model is needed. In all IPEEE submittals reviewed by the author, the internal events analysis prepared for Independent Plant Evaluation (IPE) is used to establish significant fire scenarios.

One can view the postulated fire scenarios in a fire risk analysis as a set of causes for the equipment failure identified in the internal events analysis. Since the fire scenarios affect equipment and cables from outside (i.e., outside of equipment and cable boundaries), hence fire events are labeled as part of external events or in other words a special case of common cause failures. Furthermore, one can view the internal events PSA
event trees and fault trees as a model of all possible sequences of equipment failures and operator actions that may lead to core damage and fire events are merely one of many causes that can lead to the manifestation of these failures and operator errors.

For an internal events model to be useful to fire risk analysis, it must include certain features. Simplifying assumptions based on likelihood of the event cannot be made in developing the internal events model. An event that may be considered extremely unlikely in the context of internal events, may be found to be significant in case of certain fire scenarios. In practice, the internal events models are developed based on simplifying assumptions, and are later updated based on the needs of specific fire scenarios. For example, simultaneous opening of several steam dump valves in a PWR may be considered in the internal events PSA as very unlikely and no event tree be developed for it. However, when conducting the fire analysis, it may be demonstrated that such an initiating event is possible to occur from a single fire event. Often, the internal events model is not updated to account for the specific needs of the fire analysis. In these cases, the burden lies with the analyst to ensure that a conservative analysis is conducted. However, often the fact that non-conservative assumption (from the stand-point of fire analysis) were used in the internal events analysis is lost to the fire risk analyst.

The possibility of a fire causing an initiating event (as defined in the internal events PSA model) should be addressed explicitly to ensure that the list of initiating events provided in internal events PSA is adequate. Related to the initiating events, the failure modes of control and instrumentation circuits must be investigated and often in great detail. A cable failure under fire conditions may cause a combination of shorts among various wires of a circuit. One set of the shorts may lead to spurious actuation of equipment or damage to equipment in such a way that further recovery of the failure may not be possible. Probabilistic arguments have been used to screen these failure modes. These probabilities are, in all cases, based purely on judgment and are not supported by any field observations.

The reliability of instrumentation circuits are typically not addressed in internal events model. Theoretically, it is possible for all vital information from the reactor and reactor cooling loops to be lost to a fire event. Such an event is very unlikely for a large majority of the plants because the cables for multiple instrumentation channels are routed with some level of separation. The only place in a plant that these cables could be within a short distance of one-another is inside the containment. Given the possibility of a large reactor coolant pump fire, this issue is only a concern for early design PWRs.

5. INFORMATION NEEDS

Fire risk analysis requires the use of considerable amount of information that is not typically addressed in an internal events PSA. It includes information about cable routing, compartment boundary characteristics, ventilation system function and characteristics, connections (intended and unintended) among compartments, electrical control circuits, alternate shutdown system function and design features, fire protection system design and characteristics, fire brigade training and availability, written procedures for dealing with specific plant emergencies and fire events, characteristics of electrical panels, etc. It is typical for some of this information not be available to the fire analysts. Information about cable routing has often proven to be the most difficult to obtain and in almost all cases the analysts have had to make simplifying assumptions to overcome discrepancies or to limit the scope of cable routing identification task.
6. SELECTED METHODOLOGIES

Early risk studies were based primarily on fire PRA methodology [2, 3]. Recent studies have used the FIVE methodology as well. FIVE was developed by Electric Power Research Institute (EPRI) to provide the utilities with a simplified methodology to be used in complying with the IEEE requirements. As it is stated in Ref. [4], FIVE was developed to support a PRA. From a review of FIVE it can be concluded that fundamentally there are many similarities between the FIVE and PRA methodologies, especially in the context of the screening analyses, which is a critical step for a robust and complete fire risk assessment.

Many fire risk studies have used a hybrid of FIVE and PRA methodologies. Areas in which the FIVE methodology was altered, either towards a more or less detailed analysis, include the analyses of fire detection and suppression timing, multi-compartment fire analysis, manual fire fighting, recovery actions, and data input for fire growth and propagation. At the same time, some fire risk studies have used a simplified version of PRA methodology by omitting some aspects of a state-of-the-art PRA. Typical areas of omission or simplification in the PRA-based analyses include: treatment of fire growth, propagation and damage analysis, detection and suppression timing analysis, incorporation of the effectiveness of the fire brigade, plant impact modeling; incorporation of remote shutdown panel in accident scenarios, treatment of control circuit failure modes, and human factors and recovery actions. The influence of the simplifying assumptions on the final conclusions of the fire risk studies could not be made within a reasonable level of confidence.

7. FIRE IGNITION FREQUENCY

The nuclear power industry has experienced a significant number of fires. The average rate of fire occurrence in a typical nuclear power plant is approximately 0.1 per year. From a statistical analysis of these events, the frequency of fire ignition at various parts of a typical power plant has been estimated. For example, Reference [4] includes a set of fire ignition frequencies that in recent fire risk studies have been used extensively. Almost all fire risk studies have adjusted the overall fire occurrence frequencies to establish the fire frequency for individual plant locations. In a few cases the frequency is further adjusted to account for the severity of the fire.

An important aspect of fire initiation frequency is the precise definition of the fire itself. From a review of the events that have been used in the statistical analysis, it can be observed that a wide range of fire severities are present in the data. Fire severity is a multi-dimensional concept. It includes geometric dimensions, duration of fire, heat generation rate, flame temperature, and smoke generation rate among many other attributes. The events that have been used in the statistical analysis were considered by the person who reported it as severe enough to be a reportable fire. From a review of the event descriptions, it can be concluded that the judgment of the persons reporting the fires played an important role in the cases of low severity fires. Of course, the exact definition of a low severity fire is not easy to establish. Thus, when using the fire occurrence frequencies, the analyst has to bear in mind that a certain threshold fire severity exists below which fire ignitions were not reported. This is particularly important when fire propagation analysis is conducted where an initial fire must be postulated.

The fire ignition frequency, despite the relatively large number of reported events, for a specific fire scenario (e.g., fire occurrence in a cable shaft) is often based on very sparse event data. Therefore, the uncertainties in those frequencies is relatively large. Several risk studies have used fire ignition frequency to screen out compartments. The sole content of those compartments were typically cables qualified per IEEE standards.
For example, for a cable chase area, it is argued that since all the cables are qualified per IEEE standards, the area is not visited often and there are no other equipment, the fire frequency must be small enough that the compartment can be screened from further analysis. This conclusion is reached without a review of potential equipment and instrumentation damage possibilities, impact on the plant as a whole and especially the operators’ response to the potential instrumentation and control circuit failures. Given the large uncertainties in fire occurrence frequencies for such compartments, an early screening practice does not allow for plant personnel to gain a clear appreciation of potential accident sequences.

8. SCREENING OF FIRE SCENARIOS

Since a typical power plant may consist of a large number of well defined compartments, detailed analysis of all possible fire scenarios would be resource prohibitive and would yield little useful information. Therefore, screening of fire scenarios is an important step in fire PSA. The purpose of screening is to evaluate fire risk in graduated levels of detail to minimize the level of effort. All fire risk studies have included at least one screening step. Typically a large fraction of the compartments are screened out and a small number are retained for detailed analysis. Qualitative or quantitative methodologies have been used for this purpose. Typically it is assumed that given a fire, the entire contents of the compartment is failed. The qualitative method is often based on the presence of safe shutdown cables and equipment in a compartment. The most common quantitative methodology is based on core damage frequency. If the frequency is above a threshold value, the corresponding fire scenario or compartment is subjected to detailed analysis. The threshold value typically employed by a large number of studies and recommended in Ref. [4] is $10^{-6}$/ry. A minority of risk studies have used $10^{-4}$/ry and $10^{-8}$/ry for this purpose. The benefits of using a low threshold value, because of rare application of such values, could not be readily assessed.

9. FIRE PROPAGATION AND DAMAGE MODELING

Fire risk studies completed prior to the publication of FIVE have either used COMPBRN [5] or have made conservative assumptions to avoid the use of fire growth models. For example, it was assumed that fire would damage all components within a given compartment where the fire originated, and did not model any other possibility. In other studies, it was assumed that fire suppression would be effective without consideration of the relative timing of damage and suppression effectiveness. The risk studies completed in the past few years have utilized FIVE look-up tables extensively [4].

Both the FIVE tables and COMPBRN code must be used with caution, otherwise physically unrealistic fire damage scenarios may be obtained. The fire propagation models include many diverse parameters that must be carefully quantified. For example, the heat loss factor (the fraction of the energy produced by the fire that is lost to walls and other heat sinks) influences the damage time for adjacent cables and equipment. Depending on the value selected, it is possible to conclude that no propagation would take place. Other factors that influence fire propagation include characteristics of transient combustibles, fixed combustibles and cables, and damage thresholds temperature.

FIVE does not include a provision for taking into account the uncertainties in the model. It is possible, as it has been done in at least one case, to screen out fire scenarios based on FIVE results that predict the time for propagation to a redundant train a few seconds shorter than the duration of fire detection and suppression.
Fires originating in electrical cabinets (of all sizes and service types) are found to be important contributors to fire risk. This is in part due to the co-location of these cabinets with electrical cables. Plant-specific details of electrical cabinets are found to be important. At one plant, penetrations where cables exited the top of the switchgear cabinets were not adequately sealed, which provided an exit pathway for the chimney effect. This situation allowed for the fire to be postulated as propagating up and out of the switchgear cabinet. At another plant, control cables in the control room were arranged across the top of control room cabinets with open tops. Again, this led the analysts to postulate cabinet fires that propagate to the overhead cables. The heat release rate and the potential for propagating to an adjacent cubicle are two important factors of cabinet fire modeling. Sandia test results [6] provide a basis for these factors. However, large variations exist in the interpretation of the test results, which has led to optimistic assumptions in some of the fire risk studies. Modeling of electrical cabinet fires in control rooms is also very important since such fires can force abandonment of the control room due to smoke accumulation. This issue is further discussed below.

10. ANALYSIS OF FIRE DETECTION AND SUPPRESSION

There is a large variation among the fire risk studies in their treatment of detection and suppression processes. Only a few fire risk studies have employed a relatively sophisticated model for this purpose. A large majority of the studies have employed simplified models where the unreliability of the detection and suppression is multiplied with the frequency of sequence of events if it can be shown that fire may damage cables and equipment in addition to the source of fire. The failure probability of the suppression system is often gleaned from either FIVE or other industry sources. Many fire risk studies have multiplied this failure probability with the fire occurrence frequency. This makes the assumption that a fire damages the entire contents of the compartment and the suppression system, if it functions properly can prevent all damage. This may represent an optimistic assumption if the layout of cables and equipment within the compartment is not examined. If a critical set of cables and equipment are within a small area inside the compartment, and especially on top of a likely ignition source, this multiplication process is certainly optimistic. Several studies have used generic values for suppression system reliability. This, too, may be optimistic if the fire detection and suppression systems in a plant are not compliant with the fire codes.

Many fire risk studies did not model manual fire fighting (except in the case of control room fires). This can be considered as a conservative approach. However, lack of modeling fire brigade actions is functionally equivalent to assuming that there is a negligibly low conditional probability that the brigade will unintentionally damage equipment which has not been damaged by a fire. Of course, for those studies which have assumed that fires would damage all components within the compartment, secondary damage due to the fire brigade suppression activities is implicitly included.

11. INTER-COMPARTMENT FIRE PROPAGATION

The possibility of inter-compartmental fire propagation has the potential for causing damage to cables and equipment of multiple safety trains. This aspect of fire analysis has been treated in the risk studies with varying degrees of detail and sophistication. In a large number of studies the Fire Compartment Interaction Analysis (FCIA) of FIVE has been employed. In these studies passive fire barriers are assumed to be 100% reliable. Also, this assumption is often extended to active fire barriers.

An important element of this issue is the method by which compartments are defined. Aside from the upper floors of a typical Reactor Building of a BWR, a large majority of the compartments in nuclear power plants are defined by fire barriers that are rated to contain the effects of a fire for one to three hours. However,
practically all the compartments have passages to adjacent compartments via doors, ventilation ducts, etc. An important mechanism for the propagation of the effects of a fire is the escape of hot gases through these openings. A large number of risk studies have ignored the following three potential fire propagation scenarios:

1. Failure of an active fire barrier (e.g., self-closing doors and fire dampers) to close, and propagation of hot gases to adjacent compartments.
2. Failure of fire barrier integrity due to fire-fighting activities (e.g., opening of doors to gain access or route hoses).
3. Failure of a fire barrier from being overwhelmed by an excessive source (e.g., diesel fuel tank fire, or the walls separating the turbine building from the rest of the plant).

In some cases, the possibility exists for the fire-fighting activity to lead to a breach in the integrity of the fire barriers. For example, if trains A and B of a safety system are located in two adjacent rooms that are connected by a door, the possibility exists for the fire brigade personnel to enter the affected room through the unaffected one, and if the door is left open and the fire continues to burn, it is possible for the effects of the fire to propagate through the open door.

12. CONTROL ROOM FIRE MODELING

A significant number of fire risk studies have found the control room as one of the most important fire risk contributors. Other studies have identified very low control room fire core damage frequencies and have screened out the control room as a fire-induced core damage contributor. The differences lie primarily in modeling assumptions than in the physical layout of the control room or training of the operators. Some studies have used a detailed analysis of the fire incidents in the control room and many have used a simplistic approach where a conditional probability is assigned to the failure of controlling the plant from outside the control room. In the simplistic approach no further evaluation of the possible operator action scenarios are attempted. In the detailed approach, every cabinet section is examined for potential fire ignition, failure in the control circuits, and operator response to the specific set of failures in addition to the fire detection and fire-fighting activities. In these studies, the remote shutdown panel (in some plants this may include several separately located panels) is analyzed for potential failures from control room circuit damages and for operator errors in its proper usage.

Ease of fire detection and suppression is the main reason cited by those studies that have concluded low control room fire risk. Control room fire non-suppression conditional probabilities in the range of 1-3% have been used as compared to more typical values of 2-5% used for suppression systems in other parts of the plants. Some studies specifically cited a 15-minute time period before control room abandonment would be required. This value assumes in-cabinet smoke detectors are present. With only general area detectors, the proper interpretation of the Sandia-sponsored tests yields an estimate of seven minutes. A few studies assumed that smoke-forced abandonment of the control room would occur only if multiple cabinets were involved in the fire, which is also inconsistent with Sandia test results [6].

In addition to the above, it may be noted that in at least two cases, the control room is shared between two units. Fire damage in the cabinets in one unit can force the abandonment of the control room for the other unaffected unit as well. The human error coupling between the two units is not addressed in any of the studies. Also, most fire risk studies have not used a systematic method to verify control systems interactions.
Typically, circuit isolation capability, remote location, and procedures are used to ensure that there would be no adverse interactions between the control room and remote shutdown panel.

13. ANALYSIS OF HUMAN ACTIONS

Human actions is found to be an important part of fire risk analyses. Most fire scenarios that lead to core damage (almost regardless of the level of risk significance) include human error probabilities. Often these probabilities are imbedded in the chain of events. Therefore, their proper evaluation is critical to the results of the overall analysis. Most risk significant fire scenarios include human error probabilities for the recovery actions. Internal events human reliability models and data are sometimes used for this purpose. This of course raises very important questions regarding the applicability of internal event human error probabilities to fire scenarios. Another issue that has received little attention, but may be considered as relevant to fire scenarios is errors of commission as a result of a fire effects on control and instrumentation cables. Practically none of the fire studies have addressed the possibility of wrong information on the control board and errors of commission resulting from that.

Fire brigade related human errors may prove to be important as well. Nearly none of the risk studies have included the potential for adverse effects of manual fire-fighting efforts on safe shutdown equipment. There are, however, conditions identified in specific plants that such an error can have a severe effect on plant safety.

14. IMPACT AND PROPAGATION OF SMOKE

Only a handful of fire risk studies have addressed the possibility of smoke propagation, none have considered the possibility of short term effects of smoke on equipment and a few have considered, albeit qualitatively, the possibility of suppression system activation from smoke migration in compartments other than the fire origin. In the latter case, possibility of equipment damage from exposure to fire suppression medium may be of concern. Almost all risk studies have not included the potential for smoke to hinder manual fire-fighting effectiveness or misdirected suppression efforts. It must be noted that FIVE [4] specifically states that degradation of equipment from secondary (non-thermal) fire environmental effects can be ignored. No fire risk study has taken into account the smoke impact phenomena on electronic equipment per the results of Sandia Tests [8].

15. CONCLUSIONS

A large number of fire risk studies have been completed for the nuclear power plants in the USA. These studies indicate that, based on core damage frequency, fire risk, in general, is a significant contributor to overall plant risk. Since only a few risk studies have included level 2 PSA related information, a definitive conclusion regarding significance of fire scenarios cannot be reached. However, for the few available cases, fire scenarios remain important at Level 2 analysis as well. The core damage frequencies reported in the risk studies attributed to fire events span a wide range from $10^6$ to $10^4$ per reactor year. Either PRA or FIVE or a combination of the two methodologies have been used. In many cases the analysts have modified FIVE procedures to match their specific needs. For most plants, the critical fire areas include the control room, cable spreading room, and electrical rooms. In almost all risk studies operator actions are critical to the reduction of fire risk. In none of the fire risk studies have multi-compartment fire scenarios been found to be an important
risk contributor. This is in part based on the assumption made in many fire risk studies that active fire barriers are highly reliable.

None of the fire scenarios identified in the risk studies were found to fail a minimal cutset of equipment leading to core damage. In other words, additional failures, somewhat independent of the fire, have to occur for core damage to be realized. This conclusion confirms the objectives of NRC fire protection requirements (i.e., Appendix R).

16. REFERENCES

FIRE RISK ASSESSMENT IN GERMANY - REGULATORY GUIDANCE AND APPLICATIONS

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Abstract

Methodologies considering quantitative fire risk assessment have been evolving at international level showing that such analyses can serve as an additional tool to assess the safety level of a nuclear power plant (NPP) and to set priorities for fire protection improvement measures.

The recommended approach to be applied within periodic safety reviews of nuclear power plants in Germany starts with a screening process providing critical fire zones in which a fully developed fire has the potential to both cause an initiating event and impair the function of at least one component or system critical to safety. The second step is to perform a quantitative analysis. For that purpose, a standard event tree has been developed with elements for fire initiation, ventilation of the room, fire detection, fire suppression, and fire propagation. This standard event tree has to be adapted to each critical fire zone or room. In a final step, the fire induced frequency of initiating events, the main contributors and the calculated hazard state frequency for the fire event are determined.

A few quantitative fire risk studies have already been performed in Germany and their results are reported.

1. Introduction

Experience has shown that fire can be a safety significant hazard. Thus, the regulators expect the licensees to justify their arrangements for identifying how fires can occur and spread, assessing the vulnerability of plant and structures, determining how the safe operation of a plant is affected, and introducing measures to prevent a fire hazard developing and mitigate against its effects if it should nevertheless develop.

Methods to analyse existing plants systematically regarding the adequacy of their existing fire protection equipment can be deterministic as well as probabilistic. Fire risk assessment has become an integral part of PSA and, at international level, fires have been recognised as one of the major contributors to risk of nuclear power plants, e.g. in the USA [1].
The international operating experiences reveals the indication that both types of fire assessment, deterministic and probabilistic, are complementary regarding the consideration of significance and extension of the potential damage caused by the fire. From the comprehensive joint investigation - combining statistical data, deterministic analysis of fire phenomenology, and engineering judgement - a more complete, detailed and specific perspective of the different situations that can occur in the facility is obtained.

In the past, most of the engineering work in designing fire protection measures in German nuclear power plants has been performed on a deterministic basis. Moreover, the use of deterministic fire risk analysis is current practice in Germany to review the fire protection state of operating NPP. It should be underlined that these reviews have led to comprehensive backfitting and upgrading measures including structural fire protection measures (e.g. fire barriers) as well as the active fire detection, alarm and extinguishing features and administrative fire protection measures (for manual fire fighting) resulting in significant improvements in fire safety, in particular in case of a NPP built to earlier standards.

However, as it can be seen from other areas the probabilistic approach provides different insights into design and availability of systems and components and supplements the results from deterministic analyses. Thus, probabilistic considerations are taken into account for decision making on a case-by-case basis, in particular for the in-depth investigations of a selected fire scenario at a specific safety relevant plant location. One example of such a probabilistic analysis is presented in [2].

Experiences with the application of PSA to selected fire safety related problems and comprehensive activities at international level [e.g. 1, 3, 4] regarding more comprehensive fire risk assessment have supported the decision in Germany to recommend such an assessment in the frame of periodic safety reviews.

Due to the fact that the PSA Guide and the corresponding technical documents are published recently, the fire protection measures have still been assessed on a deterministic basis in most of the periodic safety reviews performed so far for German NPP.

2. Guidance documents for periodic safety reviews

In the course of a longer lasting operating period, the safety related range of information is broadened; the methods and instruments for safety assessments are being further developed. This should lead to a continuous development of the plant's safety status and its operational safety.

Thus, it has been recommended to perform periodic safety reviews (PSR) at time intervals of about 10 years in addition to ongoing routine surveillance activities.
In order to stipulate a uniform procedure within the Federal Republic of Germany and to define a clear frame with regard to objective and scope of the periodic safety review, regulatory guides [5] have been developed supplemented by more detailed technical reports [6 - 8].

These documents also cover the fire specific aspects which have to be assessed within a periodic safety review.

The documentation of a plant's safety status within the scope of the periodic safety review comprises, in particular, the results of the following items:

- a deterministic safety status analysis in form of a review of the plant's safety status oriented to the nuclear protection goals including a description of the operational management as well as an in-depth analysis of the operating experience, and
- a probabilistic safety analysis (PSA).

Moreover, a review of the status of physical protection of the plant has to be performed, which is not further discussed in this paper. Recently, the corresponding guide stipulating kind and scope of this review has been published [9].

According to the parts of the periodic safety review three regulatory documents (see Figure 1) have been filed by a task force of the so-called "Hauptausschuß für Atomkernenergie", the Federal States Committee for Atomic Nuclear Energy. These are the "Basic Principles of Periodic Safety Review", the "Guide on Safety Status Analysis", and the "Guide on PSA" [5]. The basic principles of PSR give fundamental guidance and serve for the overall assessment of the different analyses. The guide on safety status analysis is supplemented by a document describing a protection goal oriented structure of nuclear technical rules and standards [6]. Moreover, this document provides the fundamental requirements for the protection goal oriented analysis required in the frame of the safety status analysis. A further regulatory document included in [5], the regulatory guide on PSA, covers the fundamental requirements concerning the performance of PSA.

The respective responsible supervisory federal states authority assesses the documentation of the safety status of the plant according to §§ 17 and 19 of the German Atomic Energy Act [10]. Consulted experts carry out a review of the PSR submitted by the licensee taking into account the existing PSR regulatory guidance.

The measures to be taken and directives to be given by the responsible supervisory authority in the scope of the overall evaluation of the results are established according to the principle of commensurability.

The analyses of the different items have to be preceded by a description of the current plant status; a proposal for the structure of such a description is given in the guide on the basic principles of PSR including fire safety as a specific topic.
3. Fire risk assessment in Germany

For fire risk assessment in Germany, a qualitative or quantitative screening process is proposed to identify critical fire zones followed by a quantitative event tree analysis in which the fire caused hazard state frequency will be determined. The models proposed have been successfully applied in complete and partial fire risk studies for German nuclear power plants.

3.1 Regulatory guidance

The PSA Guide contains reference listings of initiating events for NPP with PWR and BWR respectively which have to be checked plant specifically with respect to applicability and completeness. Plant internal fires are included in these listings.

Detailed instructions are provided in the technical documents on PSA methods [7] and PSA data [8] which are shortly reported. These technical documents have been developed by a working group of technical experts chaired by the BfS (Bundesamt für Strahlenschutz - Federal Office for Radiation Protection). The German regulatory guidance as it is available now has been restricted in scope to comprise only applications where sufficient practical experience is available and a reasonable consensus between the parties involved is achieved. However, the PSA working group is still in force to further develop and amend the technical documents, also with regard to fire specific aspects.

3.1.1 Screening analysis

The screening process to identify critical fire zones is an important first step within a fire risk assessment. Such a screening analysis should not be so conservative that an unmanageable number of fire scenarios remains for the detailed quantitative
analysis. However, it must be ensured that all relevant areas are investigated within the quantitative analysis.

The systematic check of all rooms' room pairs of the plant can be done in two different ways: The critical fire zones can be identified in a qualitative (qualitative screening) or in a quantitative process (screening by frequency). The qualitative screening allows - due to the introduction of appropriate selection criteria - the determination of critical fire zones with a limited effort. Applying the quantitative screening method, the critical fire zones are identified in a simplified event tree analysis.

The systematic examination of all rooms/room pairs or fire zones in the plant requires detailed knowledge of the plant specific situation.

The determination of critical fire zones starts with the identification of all rooms for which at least one of the following criteria is fulfilled:

1. fire load higher than 25 kWh/m²,
2. room contains safety related equipment or cables of such equipment,
3. room contains operational equipment, or sensing/control equipment of the reactor protection, or power limit control system, or cables of such equipment with the potential that a fire caused damage may lead to a plant transient/initiating event or to a manually operated scram,
4. In case that a fire causes an unintentional opening of a safety valve or of the main steam bypass leading to a loss of coolant accident or a main steam line leak, this fire zone will be classified as a "critical fire zone".

In a first step, those rooms are identified for which the first three criteria (L), (S) and (O) are simultaneously fulfilled. These rooms will be identified as "essential fire zones /rooms". In a next step, for those rooms for which two out of these three criteria are fulfilled, adjacent rooms are checked to identify pairs of rooms that fulfill all three criteria. The "critical fire zones/rooms" and "pairs of rooms" are selected based on the further criterion that the fire leads to a safety related initiating event.

In the case that the quantitative screening process is applied, the hazard state frequency will be calculated in a simple but conservative analysis for each room with PSA related components and components leading to initiating events after the occurrence of a fire. The event tree analysis will be carried out for fire zones/rooms or room pairs with fire loads > 25 kWh/m². Only two elements are taken into account in this event tree analysis: The fire occurrence frequency and the conditional unavailability of the safety related equipment to mitigate the initiating event. All other branches of the event tree, like fire detection, fire suppression and fire spread to adjacent rooms are not considered. All PSA related equipment within the room is assumed to be damaged (probability of damage equal 1.0). Rooms are screened out, if the product of the fire frequency and the conditional unavailability of the safety related equipment is less than 1 % of the total sum of these products. The sum of those neglected contributions shall not exceed 5 %.

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1. Rooms are used in this paper as a synonym for compartments.
2. Operational equipment, such as hoists, lighting, ventilation, fuel storage pool cooling, coolant purification, etc. are not considered.
3.1.2 Quantitative analysis

For the quantitative part of the fire risk assessment a standard event tree has been developed with nodes for fire initiation, ventilation of the room, fire detection and suppression, both in the pilot fire phase and the fully developed fire phase, as well as fire propagation. This standard event tree must be adapted to each critical fire zone or room.

For the assessment of fire spread through walls, fire doors, dampers, cable penetration sealings, etc. into adjacent rooms different methods are recommended in [7]. One methodology applied is a simplified approach for the design of structural fire protection measures in NPP which has been developed based on the estimated overall fire load density and distribution of the fire load inside the fire compartment. The design method needs only a few empirical functions and design features which were derived from systematic fire simulations with an advanced multi-room zone model and which are easily understandable and applicable [11].

For each critical fire zone/room the following results are obtained:

- frequency and nature of fire initiating events,
- list of damaged equipment, categorised corresponding to different damage states, and
- damage frequencies.

If a complete plant specific PSA is available, for each initiator the fire induced frequency will be summed up and specified as input to the corresponding system event tree of the level 1+ PSA. Additionally, the damage state of the equipment has to be introduced into the fault trees. The plant hazard state frequency is calculated for each transient as the sum of the single event core melt frequencies. The total plant hazard state frequency is obtained by summarising the contributions of all transients. The requirement to use only qualified PSA codes has also to be fulfilled for the fire PSA. Moreover, validated fire simulation models and codes have to be used in case of deterministic fire hazard analysis and probabilistic fire risk assessments.

3.1.3 Data base

In order to perform a quantitative fire risk assessment, a basic data base must be established which should, e.g., include initiating frequencies, reliability data for all fire protection measures, fire barriers, etc. Detailed information is needed on ignition sources, detection and extinguishing systems, manual fire fighting, stationary fire suppression systems; further information on secondary effects, safety consequences, analysis of the cause of the event and corrective measures, etc. would be helpful. It should be underlined that plant specific data are to be used as far as possible.

As one contributor to fire specific PSA input data, reliability data for the active fire protection measures are required for the application in the fire specific event tree analysis. These data needed to be estimated are unavailabilities per demand or failure rates per hour of plant operation for those components or systems belonging to the active fire protection means.
Active fire protection measures to be studied include all the fire detection equipment, that means all types of fire detectors including their power supplies, the alarm panels and boards, fire doors and dampers and the stationary fire extinguishing systems including the extinguishing media supplies.

In Germany, for two different types of reactors (PWR and BWR) unavailabilities and failure rates were estimated. These reliability data given in tables 1 and 2 and are updated values of those provided in [12, 13] by processing additional information. The scattering factor \( k \) given in these tables is correlated to the failure rates.

<table>
<thead>
<tr>
<th>Active fire protection feature</th>
<th>Inspection period</th>
<th>Scattering factor ( k )</th>
<th>Failure rate ( \lambda(t) ) [1/h]</th>
<th>Unavailability per demand</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire alarm boards:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>detection drawers</td>
<td>3m, 1a</td>
<td>3.31</td>
<td>6.7( \cdot 10^{-8} )</td>
<td>1.2( \cdot 10^{-4} )</td>
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<tr>
<td>detection lines</td>
<td>3m, 1a</td>
<td>3.29</td>
<td>2.3( \cdot 10^{-8} )</td>
<td>4.0( \cdot 10^{-6} )</td>
</tr>
<tr>
<td>Detectors:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>automatic</td>
<td>1a</td>
<td>1.25</td>
<td>1.4( \cdot 10^{-7} )</td>
<td>1.3( \cdot 10^{-3} )</td>
</tr>
<tr>
<td>press button</td>
<td>1a</td>
<td>2.55</td>
<td>1.1( \cdot 10^{-7} )</td>
<td>9.4( \cdot 10^{-4} )</td>
</tr>
<tr>
<td>Sprinklers</td>
<td>3m, 1a, 3a</td>
<td>1.13</td>
<td>2.3( \cdot 10^{-8} )</td>
<td>6.6( \cdot 10^{-3} )</td>
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<tr>
<td>Doors</td>
<td>1a</td>
<td>1.36</td>
<td>4.9( \cdot 10^{-8} )</td>
<td>4.3( \cdot 10^{-6} )</td>
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<tr>
<td>Dry sprinkler extinguishing</td>
<td>6 m, 1a, 5a</td>
<td>3.14</td>
<td>2.2( \cdot 10^{-7} )</td>
<td>9.9( \cdot 10^{-4} )</td>
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<td>system (total failure)</td>
<td></td>
<td></td>
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<tr>
<td>Dry sprinkler extinguishing</td>
<td>6m, 1a, 5a</td>
<td>1.35</td>
<td>4.1( \cdot 10^{-8} )</td>
<td>1.8( \cdot 10^{-2} )</td>
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<tr>
<td>system: automatic actuation</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>failure only</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Wet sprinkler extinguishing</td>
<td>6m, 1a, 5a</td>
<td>7.62</td>
<td>6.4( \cdot 10^{8} )</td>
<td>3.2( \cdot 10^{-4} )</td>
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<tr>
<td>system</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gas extinguishing systems</td>
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<td>7.64</td>
<td>1.9( \cdot 10^{9} )</td>
<td>9.2( \cdot 10^{-3} )</td>
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<tr>
<td>(CO(_2))</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Stationary fire pumps</td>
<td>1m, 1a</td>
<td>7.64</td>
<td>1.9( \cdot 10^{8} )</td>
<td>1.4( \cdot 10^{-3} )</td>
</tr>
<tr>
<td>Wall hydrants</td>
<td>6m, 1a</td>
<td>7.64</td>
<td>3.9( \cdot 10^{8} )</td>
<td>1.3( \cdot 10^{-4} )</td>
</tr>
</tbody>
</table>

The data on potential failures or unavailabilities per demand of the respective fire protection measures were gained from the plant specific documentation of inspection and maintenance. The assessment whether the detected findings are estimated as failures or only as deficiencies or deteriorations requires a deep insight in the plant specific operating conditions for the fire protection means and needs careful engineering judgement.
Table 2: Plant specific reliability data estimated for active fire protection features in a German PWR reference plant

<table>
<thead>
<tr>
<th>Active fire protection feature</th>
<th>Inspection period</th>
<th>Scattering factor k</th>
<th>Failure rate λ(t) [1/h]</th>
<th>Unavailability per demand</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire alarm boards:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>detection drawers</td>
<td>3m</td>
<td>7.63</td>
<td>3.0·10⁻⁸</td>
<td>7.4·10⁻⁵</td>
</tr>
<tr>
<td>detection lines</td>
<td>3m</td>
<td>7.63</td>
<td>1.7·10⁻⁸</td>
<td>4.3·10⁻⁵</td>
</tr>
<tr>
<td>Fire detectors:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>automatic</td>
<td>1a</td>
<td>1.75</td>
<td>4.8·10⁻⁸</td>
<td>4.2·10⁴</td>
</tr>
<tr>
<td>press button</td>
<td>1a</td>
<td>7.63</td>
<td>3.4·10⁻⁸</td>
<td>3.3·10⁴</td>
</tr>
<tr>
<td>Fire dampers</td>
<td>6m</td>
<td>1.39</td>
<td>6.6·10⁻⁷</td>
<td>2.9·10³</td>
</tr>
<tr>
<td>Fire doors</td>
<td>1a</td>
<td>1.44</td>
<td>7.1·10⁻⁷</td>
<td>6.3·10³</td>
</tr>
<tr>
<td>Dry sprinkler extinguishing system (total failure)</td>
<td>3m, 1a</td>
<td>7.63</td>
<td>1.3·10⁻⁷</td>
<td>3.2·10⁴</td>
</tr>
<tr>
<td>Dry sprinkler extinguishing system: automatic actuation only</td>
<td>3m, 1a</td>
<td>1.26</td>
<td>1.3·10⁻⁵</td>
<td>2.9·10²</td>
</tr>
<tr>
<td>Stationary fire pumps</td>
<td>1m, 1a</td>
<td>7.64</td>
<td>1.3·10⁻⁶</td>
<td>8.5·10⁴</td>
</tr>
<tr>
<td>Wall hydrants</td>
<td>1a</td>
<td>1.75</td>
<td>8.5·10⁻⁷</td>
<td>7.4·10⁻³</td>
</tr>
</tbody>
</table>

Table 3 gives an overview of the generic failure rates for active fire protection features revealed from the plant specific data estimated.

Further detailed information on plant specific and generic estimation of unavailabilities and failure rates for active fire protection measures can be found in [12 - 14].

Most of the estimated values for the technical reliability of active fire protection measures represent realistic data for the failure rates of these systems and components in German NPP. They can be applied in the frame of PSA studies instead of conservative data from past nuclear specific reliability studies as well as instead of generic values available from data of the insurance companies for the reliability of comparable fire protection means in non-nuclear industry, both being available up to the time being. The data have been adapted in the PSA data document [8].

However, it has been seen as a worthwhile task to expand and improve these current data by taking into consideration plant specific data of two other German NPP to replace as far as possible generic data from other non-nuclear or foreign nuclear data bases and sources.
Table 3: Generic reliability data estimated for active fire protection features in German NPP

<table>
<thead>
<tr>
<th>Active fire protection feature</th>
<th>Failure rate $\lambda$ (t) [1/h]</th>
<th>Scattering factor k</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire alarm boards:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>detection drawers</td>
<td>$9.1 \times 10^{-8}$</td>
<td>4.24</td>
</tr>
<tr>
<td>detection lines</td>
<td>$4.0 \times 10^{-8}$</td>
<td>4.34</td>
</tr>
<tr>
<td>Fire detectors:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>automatic</td>
<td>$1.2 \times 10^{-7}$</td>
<td>2.88</td>
</tr>
<tr>
<td>press button</td>
<td>$1.2 \times 10^{-7}$</td>
<td>3.60</td>
</tr>
<tr>
<td>Fire dampers</td>
<td>$1.9 \times 10^{-6}$</td>
<td>2.93</td>
</tr>
<tr>
<td>Fire doors</td>
<td>$7.5 \times 10^{-7}$</td>
<td>2.92</td>
</tr>
<tr>
<td>Dry sprinkler extinguishing systems (total failure)</td>
<td>$3.4 \times 10^{-7}$</td>
<td>4.32</td>
</tr>
<tr>
<td>Dry sprinkler extinguishing system: (automatic actuation failure only)</td>
<td>$1.1 \times 10^{-5}$</td>
<td>2.90</td>
</tr>
<tr>
<td>Wet sprinkler extinguishing systems (total failure)</td>
<td>$1.4 \times 10^{-7}$</td>
<td>5.67</td>
</tr>
<tr>
<td>Gas extinguishing systems (CO$_2$)</td>
<td>$4.5 \times 10^{-6}$</td>
<td>6.60</td>
</tr>
<tr>
<td>Stationary fire pumps</td>
<td>$3.9 \times 10^{-8}$</td>
<td>4.19</td>
</tr>
</tbody>
</table>

3.2 Applications

3.2.1 Fire risk assessment for full-power states

The first German fire PSA was performed by GRS (Gesellschaft für Anlagen- und Reaktorsicherheit - Company for Plant and Reactor Safety) for the Biblis B plant (PWR) as a reference plant [15]. The screening analysis was based on high fire load inventory, the presence of ignition sources, safety related equipment and high inventory of radioactive materials. For the identified 10 relevant fire zones detailed hot gas layer temperature analyses were carried out, and for the two 220 V DC power supply rooms a contribution to the hazard state frequency of $1.7 \times 10^{-7}$/a each was calculated.

A similar, also more generic study for the reference plant Gundremmingen (BWR) has again been performed by GRS [16]. It has to be stated that the estimation of the fire related hazard state frequencies contains large uncertainties. The investigations resulted in frequencies for cable fires of $3 \times 10^{-5}$/a and for oil fires of $< 10^{-5}$/a, with a conditional ignition probability of $< 10^{-2}$ being used. For the failure probabilities of fire fighting measures, ranges were estimated of 1 to 0.1 for oil fires and of $< 10^{-2}$ for cable fires. In case of a failure of the fire fighting measures reactor protection measures are initiated for oil and/or cable fires due to the fire related pressure increase in the containment. Initially, automatic pressure suppression will start; in the further course of the events there will be a pressure increase due to the fire induced failure of the pressure limitation function. Such a sequence is controlled if pressure limitation is functioning and RPV feeding at high pressure is safeguarded. Event sequences and the boundary conditions for their control can only be determined with difficulty and quantified with high uncertainties due to the various failure probabilities.
of the electrical instrumentation inside the containment. The frequency of hazard states is generally estimated to be well below $10^{-9}$/a.

In 1995, a complete fire PSA as part of the level 1+ analysis for the German Unterweser NPP (PWR) has been performed. The procedure outlined in [7] has been applied consistently. From a total of approximately 1100 rooms, 120 essential fire rooms compartiments have been identified and, based on the most important criterion that the fire has to cause an initiating event, 70 critical fire zones were identified using the qualitative screening process. These have been condensed to 13 representative critical rooms for which a quantitative analysis has been performed [17]. The quantification of the fire induced initiating events has resulted in two classes:

- rooms in which a fire directly leads to an initiating event; in this class, 8 rooms are classified. Only 3 of these 8 rooms gave countable contributions to the hazard state frequency,
- rooms where a fire results in a plant transient and where an initiating event is caused due to the stochastic failure of the operational equipment necessary to prevent an incident; in these rooms, no countable hazard state frequencies are calculated ($< 10^{-9}$/a).

The fire induced hazard state frequency amounted to a value of about $5 \cdot 10^{-7}$/a for the whole plant [17]; thus, the fire event contributes to about 6% of the total plant hazard state frequency (taking into account internal and external events) which has been recently recalculated to about $8 \cdot 10^{-9}$/a [18]. Main contributors are fires in the electronic rooms of the switchgear buildings.

For the German Isar I nuclear power plant (BWR) a complete fire PSA has been performed, too [19]. The quantitative screening process has been applied to approximately 500 rooms in the reactor building, turbine building, switchgear building, emergency diesel generator room and service water intake structure. 172 critical rooms have been identified and analysed. The relation of local fire frequencies was calculated according to Berry's method, published in [20]. The fire induced hazard state frequency of about $6.3 \cdot 10^{-7}$/a for the whole plant resulted mainly from 14 single rooms and 7 room pairs (calculation including fire spreading analysis). In order to get more information on the consequences on the safety in case of system unavailability the analysis of common cause initiators has been performed for 15 rooms with a contribution of more than 1% to the system unavailability. Main contributors are fires in switchgear buildings.

A fire PSA for the Grafenrheinfeld NPP has recently been elaborated in a comparable manner to the Isar 1 fire risk assessment and is currently in the review process by the competent authority and its experts. Results of this review are expected in the second half of this year. For an actual probabilistic fire safety assessment carried out by GRS for one of the most recent German PWR reactors (more details are provided in [21]) several improvements are intended. Main goal of the activities is the development of a systematic approach for quantifying the fire occurrence frequency of initiating fires taking into consideration the plant specific local and technical conditions. The improvements should also consider the significance of initiating fire events for PSA study, considering reference plant
characteristics. The experiences may be used when updating the technical PSA documents which is foreseen within the next years.

In contrast to an earlier PSA Guide published in 1990, the reference event spectrum is enlarged by the fire event in the PSA Guide [5]. Therefore, after completion of a PSA for a 1300 MW_e PWR plant using the nuclear power plant Isar 2 as a reference plant for seven of the most recent so-called "Vorkonvoi/Konvoi" type plants, a probabilistic fire risk assessment has also been performed on the basis of [7, 8]. In a first step, 204 PSA relevant rooms have been identified and room pairs have been formed. Further considerations have shown that a screening analysis should be carried out for 117 room pairs. Within the screening analysis the event sequences for the room where a fire is assumed are quantified under the boundary condition that the components of the adjacent room also fail with a probability of 1 caused by the fire. Based on the results, those room combinations are selected for further considerations where a relevant contribution (i.e. hazard state frequency of > 10% of the sum of the frequency of the single rooms) is to be expected. As a result, 19 room pairs have to be quantified. For the selected reference plant, a value of \(3.5 \cdot 10^7/a\) for the plant hazard state frequency has been calculated [22] 62% of which resulting from fires without fire propagation. Also this fire PSA shows that the fire risk does not provide a dominant factor (about 11%) to the total occurrence frequency of a plant hazard state with an amount of \(2.5 \cdot 10^{-6}\) for the full power operational state. The successful fire fighting in the initial phase of a fire is one of the most significant factors for the low plant hazard state frequency. A detailed consideration shows that the area of a main feed water pump oil supply where additional cable trays are located is the main contributor to the hazard state frequency [23].

In this section some results of fire PSA performed in Germany have been shown. However, it should be underlined that it is necessary to take care by using these data to assess fire safety levels of plants only by comparing the quantitative fire induced hazard state frequencies. Consideration has to be taken to the problems which are connected with a comparison of fundamentally (PSA for internal events and PSA for fire events) or even partly different analyses (different fire PSAs using specific approaches). A comprehensive comparison has to include a sufficient uncertainty analysis and a reliable comparability requires that all important conditions are identified, presented and evaluated.

3.2.2 Approach for fire PSA considerations to shutdown states

In general, there are three different outage types: refuelling outages, planned maintenance outages and unplanned outages. The outage types differ considerably with respect to the outage duration and the operational modes which need to be taken into account in the shut down PSA. For many unplanned shutdowns, for instance operation can be resumed after a short delay of a few hours. In these cases, it is not necessary to go to cold shutdown or to open the reactor pressure vessel closure head. For an unplanned or a planned outage the operational mode is important, because the unavailability of the mitigating systems is different for the differing modes of operation. During these operational modes plant configuration and operational conditions such as reactor coolant system temperature, primary pressure, primary water level, neutron flux, decay heat level, availability of safety systems and support systems may change considerably.
It is current practice to define a manageable number of so-called plant operational states (POS) which sufficiently represent the different plant configurations and operational conditions. According to the operation manual of the reference plant Unterweser; 15 POS have to be distinguished [24]. For each of them separate analyses are required.

For the fire occurrence frequency a data base has been evaluated, which corresponds in major points to the one of SANDIA National Laboratories, published in [25]. This data base allows to relate the fire frequency to the shutdown periods

- hot shutdown (POS 02 - 05 and POS 13 - 14),
- cold shutdown (POS 06 - 07 and POS 11 - 12),
- refuelling (POS 08 - 10),

for which the analysis is recommended. A plant specific update of these data according to Bayes' theorem is also recommended [8].

The results of the fire propagation analysis of each fire zone are frequencies of fire induced initiating events and other plant conditions as well as lists of damaged equipment, categorised according to the different plant states. These results are implemented in the PSA level 1+ models to calculate the plant hazard state frequencies. For a shutdown PSA, more end states as for a full-power PSA have to be studied such as core of fuel storage pool heat up above a certain point, inadvertent criticality or pressure vessel over pressurisation in cold conditions.

For the execution of the event sequence analyses the following working rules are recommended [26]:

- The results of the full-power PSA are valid for the POS 01 and 15.

- For the event sequences to be analysed, a general cut-off criterion with respect to plant hazard state frequency of $< 10^{-7}$/a for a single event and $< 10^{-6}$/a for the sum of events applies.

- The plant damage frequency has to be calculated for each identified initiating event for the differing POS. For different POS, the plant hazard state frequency may be different, because of different fire frequencies, different numbers of safety trains available and different safety systems affected.

- For the complete fire PSA, the plant hazard state frequencies for the different critical fire zones and for the different initiating events for the different POS have to be summarised.

The recommended procedures are applicable to shutdown and low power operation modes and are demonstrated by means of an example. The plant specific aspects, in particular the collection of the rules and procedures applicable for shutdown operation, require that the investigations are based on a reference plant. Hence, in a qualitative screening analysis for the Unterweser NPP the reactor annulus B0106 was identified as a critical fire zone based on the above mentioned criteria. Under certain circumstances a postulated fire in this plant location can cause - combined
with additional stochastic failures - the loss of the cooling system for the spent fuel storage pool. The calculated plant hazard state frequency for the shutdown period "refuelling" was negligible because of the variety of the redundant supplementary cooling systems. The contribution for the other shutdown periods "hot shutdown" and "cold shutdown" was higher, because of the estimated fire occurrence frequency being higher (one order of magnitude) and because of the number of decay heat removal trains designated for storage pool cooling being less (one train is necessary for the cooling of the fuel assemblies in the core). The sum of the three single contributions to the plant hazard state frequency is about $3 \cdot 10^{-9}$/a.

The calculations were performed using the risk and reliability code "Risk Spectrum" [27], with a detailed modelling of arrangement of piping (parallel trains; additional valves), power supply of pumps and valves, common cause failures, human actions, preventive maintenance and repair.

4. **Concluding remarks**

Due to the fact that fires are complex phenomena resulting not only from ignition and combustion processes, but also from the impact of the fire on safety related equipment and from the necessary appropriate response of the operators, fires have still to be considered also in modern plants. However, main emphasis remains on NPP built to earlier standards, in order to ensure an adequate fire safety level.

In consequence, appropriate deterministic and probabilistic evaluation methods had to be applied to assess the fire safety level which must be fulfilled by the NPP design and operation.

However, the fire risk analysis, as an important part of PSA, has not yet achieved the same level of methodological maturity as being typical for some other disciplines of PSA. Major issues in the fire risk analysis are to a large extent correlated with the physical part of fire analysis and the interface between the deterministic and probabilistic analyses. Thus, more investigations are needed in the areas of determination of fire frequencies, the appropriate modelling of human actions in response to a fire event, and analysis of the dynamic fire development for a better understanding of the fire growth phenomena and the plant response to fire occurrences and recovery actions.

Apart from that, fire risk assessment meanwhile represents an important tool to get a more comprehensive picture of the safety level of a NPP regarding its fire protection arrangements.

The first fire PSA that have been performed for NPP in Germany show that the contribution of plant internal fires to the total plant hazard state frequency is low and that these events do not represent - for the investigated plants - significant contributors to the plant hazard state frequency compared to the results from, e.g., US plants. However, the results of the fire PSA can be used to analyse in detail and assess findings of the deterministic part of the periodic safety review, to determine the necessity and urgency of safety improvements and to set priorities for fire protection improvement measures.
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- Probabilistic Study of Fire Scenarios - Mme Myriam Chaussard, MM. R. Bertrand, F. Bonneval and J.M. Mattéi, Institut de Protection et de Sûreté Nucléaire (IPSN), France

- Fire Risk Analysis for Novovoronezh Nuclear Power Plant in Russia (NVNPP-5) - Ms. Irina Kouzmina and Mr. Artour Liobarski, Science-Engineering & Safety Center of Gosatomnadzor, Russia and Dr. Mardy Kazarians, Kazarians & Associates, United States

- Fire PSA for NPP Paks in Hungary - Dr. Istvan Kelemen, ETV-ERŐTERV Rt., Power Engineering & Contractor Co., Hungary and Mr. Jenő Nigicser, Mr. S. Czakó, Mr. Z. Karsa, and Mr. P. SIKLÓSSY, VEIKI Institute for Electric Power Research Co., Hungary

- Fire Risk Analysis for Loviisa 1 Turbine Hall - Mr. Matti Lehto, Fortum Engineering Ltd, Finland and Mr. Jussi K. Vaurio, Fortum Power and Heat Oy, Loviisa Power Plant, Finland
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"Probabilistic Study of Fire Scenarios"

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APPENDIX 1

APPENDIX 2

APPENDIX 3
A. INTRODUCTION

A fire Probabilistic Safety Assessment - called fire PSA - is being carried out by the French Institute of Nuclear Safety and Protection (IPSN) to be used in the framework of the safety assessment of operating 900 MWe PWRs. The aim of this study is to evaluate the probability of core melt which could result from a fire.

The establishment of fire scenarios is one of the main steps of this study. The fire scenarios are used to assess the conditional probability of damage of the safe shutdown equipment.

The purpose of this communication is to present the methodology used for defining the fire scenarios.

B. QUALITATIVE STEP

The establishment of fire scenarios has to take into account the fire safety feature implemented in the Nuclear Power Plant.

1. Generic events affecting the fire development

The method of establishing fire scenarios is based on the development of fire event trees. Event trees creation is based on an initial analysis of the fire detection and fire extinguishing using a transition graph. The transitions reflect the main means of detecting and extinguishing fires available in a nuclear facility. An example is given in Appendix 1 (for the symbols between parenthesis, see the corresponding graph).

The different phases expressing the development of a fire up to the moment of extinguishing are the following:

- ignition, which corresponds to the appearance of a fire location,
- fire detection and verification by personnel, locally,
- fire extinguishing, corresponding to total control of the fire.

1.1. Ignition

Experience feedback shows that the main ignition sources in Nuclear Power Plants are electrical equipment where short circuits can cause the ignition of combustible materials, such as oil. Furthermore, wood and PVC, representing most of the transient combustibles used during maintenance work, can catch fire during hot work, such as welding, cutting or grinding.

Each fire source thus identified constitutes an initiating event for a fire event tree. The experience feedback shows that in 50 % cases the fire extinguishes itself. This event has to be taken into account in the fire scenarios establishment.

1.2. Fire detection

Three different means of detection can be considered:
Early automatic detection (A)

This is signalled in the control room by a fire detection system, then verified by a field operator dispatched to the zone where the fire broke out.

Local detection (L)

This is the case of detection by a field operator who happens to be in the vicinity of the fire.

Late ("indirect") detection

In the absence of detection of a fire by the above means, the symptoms of fire will alert the plant staff. The symptoms are either noise (crackling, falling objects), or unfamiliar smells propagating via the openings in the compartment or the ventilation network, calling for the dispatch of a field operator to the zone to confirm where the fire has broken out.

1.3 Fire extinguishing

Two types of extinguishing are taken into account.

1.3.1 Automatic extinguishing

Some rooms are equipped with automatic extinguishing systems to fight the fire quickly.

1.3.2 Manual extinguishing - response of the staff

Organisation of response has to be simple and effective to make it applicable to different types of fire and established on the basis of the actions to be performed. It can consist of a number of successive phases:

First-line response (1):

This corresponds to response by the field operator and/or others in the vicinity without additional means of protection.

Second-line response (2):

This relates to fire extinguishing by plant staff using mobile fire extinguishing equipment (extinguishers and fire hoses) while awaiting the arrival of external fire brigades.

The phase consisting of joint fire extinguishing by site and external fire brigades (2P):

This corresponds to fire brigade action.

In the long-term, if fire detection, protection and/or response have failed, the fire can spread. It is considered that in such situations, action at the seat of the fire is no longer possible.

The fire extinguishing strategy therefore consists of preventing the propagation of fire outside the compartment and ensuring that it remains limited to the compartment involved.

2. Fire development

The development of a fire consists of three main phases:
ignition,
fire development inside the compartment,
spreading of fire to contiguous compartments.

2.1. Ignition

Any combustible material in the vicinity of a potential ignition source can be the root cause for an initial fire. For this reason, it is necessary to understand their physical characteristics properly and the fire risk that they represent for the unit.

2.2. Fire development inside the compartment

Knowing the physical characteristics of the combustible, it is possible to model the development of the fire using the fire simulation code FLAMME-S.

Modelling is used to evaluate the damage time of the targets located in the compartment, based on time dependent temperature development in the course of the fire. This also involves knowing all the parameters which can affect the development of the fire, namely the configuration of the compartment analysed, the position of the combustible, the openings to adjacent compartments and the ventilation conditions.

The effect of the air supply on combustion and the temperature of the air in the compartment is indeed important. Therefore, several calculations have to be performed for the quantification of the fire scenarios.

The types of scenarios to consider according to the openings configuration are the following:

- First configuration (1): fire doors and fire dampers closed fifteen minutes after the fire ignition

  The doors of the compartment are closed and the fire dampers close, either automatically or by manual action, by the field operator, when checking fire barrier after verification of fire.

- Second configuration (2): fire doors closed and fire dampers open during all the fire development

  The doors of the compartment are all closed after verification by the field operator but at least one fire damper (an intake one, as this represents the worst case) remains open throughout the duration of the fire (mechanical failure, human error combined with failure of the fusible link...).

- Third configuration (3): fire doors open during all the fire development and fire dampers closed fifteen minutes after the fire ignition

  The dampers are all closed fifteen minutes after the fire ignition but at least one door remains open throughout the duration of the fire.

- Fourth configuration (4): fire doors and fire dampers open during all the fire development

  At least one door and one fire damper remain open after verification by the field operator.

These situations correspond to the main configurations liable to be encountered if fire breaks out in a critical compartments.
2.3. **Spreading of fire to contiguous compartments**

If the compartment does not represent a fire compartment or the fire barrier is deteriorated, allowance must be made for the possibility of the fire not being contained within the compartment. The spreading of fire to adjacent compartments and the consequences depend on the configuration of the compartment (i.e. state of the openings) and the configuration of the ventilation (i.e. fans in operation or not).

With a multi-compartment fire simulation code, it is possible to determine if equipment located in the adjacent compartments are damaged.

### 3. Fire scenarios

Once all the events which can affect the development of fire have been identified, it is possible to construct fire event trees. These must address all the cases represented in the critical compartments selected.

Fire scenarios modelling is performed in two steps.

#### 3.1. First step:

The first step is the construction of the fire barriers and automatic extinguishing event trees. The different events which are taken into account are the following:

- automatic extinguishing by the fixed system if the room is equipped with,
- doors closure,
- fire dampers closure.

This event tree determines the types of scenarios to simulate with the code FLAMME_S to estimate the damage time of the targets.

#### 3.2. Second step:

The second step is the construction of the fire event tree corresponding to the detection and extinguishing phases during the development of the fire. The different events which are taken into account are the following:

- self extinguishing,
- detection,

The events associated with detection are local detection, automatic detection or late detection (sounds or smells noticed at a late stage) are represented in the event tree. Extinguishing by the staff with mobile system.

The events associated with fire extinguishing are first-line response by the field operator or, if this intervention is unsuccessful, the second-line response team. Fire extinguishing by plant fire fighting personnel assisted by the fire brigade is represented by failure of the two previous intervention. For each step an event tree is elaborated. The two event trees are shown in Appendix 2.
C. QUANTITATIVE STEP

Assessment of the probability of generic events is based, insofar as possible, on operational experience feedback or using the fault tree method. Operational experience feedback concerning fires can be used to quantify the initiating event, the fire detection and the fire extinguishing by staff response.

1. Quantification of the fire initiating event

Calculation of the fire frequency for all fire sources in the critical zones selected is based on fire operational experience feedback.

Knowing the fire frequency ($F_{fire \text{, i}}$) for the Type i equipment in all the building studied ignition source family and the total number of items ($N_{total \text{, i}}$) of Type i equipment installed in the reference unit, the fire frequency associated to the source i is:

$$f_i = \frac{F_{fire \text{, i}}}{N_{total \text{, i}}}$$

2. Quantification of the fire barriers and automatic extinguishing events

2.1. Quantification of the reliability of automatic fire extinguishing systems

The reliability of automatic fire extinguishing systems has to be assessed by fault trees using failure rates derived either from analysis of plant operational experience feedback or international database.

The conditional probability of the automatic extinguishing failure is estimated between $10^{-1}$ and $4 \times 10^{-1}$ according to the room studied.

2.2. Quantification of human errors associated with fire barriers

The closure of doors and fire dampers are the main fire barrier actions.

2.2.1. Closure of fire doors

The fire doors of each compartment are normally closed. They are equipped with door closing mechanisms and their closure has to be verified at the first-line response stage.

The basic events quantified are:

- the mechanical failure of the closing mechanism estimated at $10^{-2}$ coming from American literature,
- the absence of manual closure of a door initially open is estimated at $3 \times 10^{-1}$ according to the methodology of human errors quantification presented in appendix 3.

2.2.2. Closure of fire dampers

The fire dampers at the boundaries of the fire compartments are equipped with fusible links which close them in response to a temperature criterion. The field operator has to ascertain, when the presence of fire is confirmed, that the fire dampers are effectively closed and close manually those which are open.

The basic events quantified are:
the mechanical failure of the damper closing mechanism, estimated at $1,4 \times 10^4$ using the operational experience feedback,

the failure of the fusible link, estimated at $7 \times 10^{-5}$ coming from American literature,

the absence of manual closure of the fire damper by the field operator during the prescribed action in the event of fire, estimated at $3 \times 10^4$ according to the methodology of human errors quantification presented in appendix 3.

3. Quantification of the fire detection and extinguishing events

3.1. Quantification of the self extinguishing event

The self extinguishing of the fire corresponds to the damage of the source only. The conditional probability of self extinguishing is evaluated at $4,3 \times 10^4$ using operational feedback on fires.

3.2. Quantification of fire detection and suppression missions

The statistics are based on fires for which the modes of detection and/or suppression are known. The lessons learnt from operational experience feedback on fire can include:

for the fire detection:

- type of detection local, automatic or late,
- the time associated with the fire detection (recorded in the operational event report or assessed by expert judgement),
- the time associated with local verification by a field operator after automatic detection.

for the fire suppression:

- people who extinguished the fire (field operator, plant fire response team or external fire brigade),
- the time associated with the fire suppression (recorded in the report).

An average duration and a probability of success is assigned to fire detection and fire extinguishing related to the detection mode.

The results obtained from the French operating experience feedback (until 1997) are the following:

- for the detection missions:

<table>
<thead>
<tr>
<th></th>
<th>Local detection</th>
<th>Automatic detection</th>
<th>Late detection</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>number of fires</td>
<td>19</td>
<td>27</td>
<td>6</td>
<td>52</td>
</tr>
<tr>
<td>success rate</td>
<td>36.5 %</td>
<td>51.9 %</td>
<td>11.6 %</td>
<td>100 %</td>
</tr>
<tr>
<td>duration</td>
<td>2 minutes</td>
<td>7 minutes</td>
<td>19 minutes</td>
<td></td>
</tr>
</tbody>
</table>

- for the extinguishing missions after a local detection:

<table>
<thead>
<tr>
<th></th>
<th>First intervention</th>
<th>Intervention by internal team fire-fighters</th>
<th>Intervention by external fire brigade</th>
<th>total</th>
</tr>
</thead>
<tbody>
<tr>
<td>number of fires</td>
<td>15</td>
<td>3</td>
<td>1</td>
<td>19</td>
</tr>
<tr>
<td>success rate</td>
<td>78.9 %</td>
<td>15.8 %</td>
<td>5.3 %</td>
<td>100 %</td>
</tr>
<tr>
<td>duration</td>
<td>4 minutes</td>
<td>11 minutes</td>
<td>45 minutes</td>
<td></td>
</tr>
</tbody>
</table>
for the extinguishing missions after an automatic detection:

<table>
<thead>
<tr>
<th></th>
<th>First intervention</th>
<th>Intervention by internal team fire-fighters</th>
<th>Intervention by external fire brigade</th>
<th>total</th>
</tr>
</thead>
<tbody>
<tr>
<td>number of fires</td>
<td>12</td>
<td>9</td>
<td>6</td>
<td>27</td>
</tr>
<tr>
<td>success rate</td>
<td>44.4 %</td>
<td>33.3 %</td>
<td>22.2 %</td>
<td>100 %</td>
</tr>
<tr>
<td>duration</td>
<td>5 minutes</td>
<td>20 minutes</td>
<td>76 minutes</td>
<td></td>
</tr>
</tbody>
</table>

for the extinguishing missions after a late detection:

<table>
<thead>
<tr>
<th></th>
<th>First intervention</th>
<th>Intervention by internal team fire-fighters</th>
<th>Intervention by external fire brigade</th>
<th>total</th>
</tr>
</thead>
<tbody>
<tr>
<td>number of fires</td>
<td>0</td>
<td>1</td>
<td>5</td>
<td>6</td>
</tr>
<tr>
<td>success rate</td>
<td>0 %</td>
<td>16.7 %</td>
<td>83.3 %</td>
<td>100 %</td>
</tr>
<tr>
<td>duration</td>
<td></td>
<td>40 minutes</td>
<td>73 minutes</td>
<td></td>
</tr>
</tbody>
</table>

The total duration of each scenario corresponds to the sum of the associated average duration for the different types of detection and suppression of which it is made up.

The conditional probability of each fire scenario is given in appendix 2.

4. Fire simulation

The simulation of a fire by a computer code makes it possible to estimate the damage time of the equipment necessary to reach and/or maintain a safe shutdown state.

The main data needed for fire scenario simulations are:

- data relative to the compartment geometry, as well as the potential locations of initial fires, items of equipment, cables and openings (dimensions and positions of ventilation inlets and outlets, dimensions of doors, ducts etc.),
- combustible characteristics (nature, combustible products, temperature and/or inflammation rate, concentration of oxygen, proportion of energy radiated by the flame, rate of combustion etc.),
- damage temperature,
- characteristics of the boundaries (thickness, heat exchange coefficients, emissivity, mass per unit volume, heat per unit mass, etc.),
- characteristics of the ventilation system (pressure loss coefficients, inlet and outlet pressures of ventilation ducts, etc.).

The temperature criteria of equipment damage are the following:

- 230 °C for the cables related to the experimental test PEPSI 1,
- 40 °C for the electrical equipment related to the conception rules.

5. Conditional probability of damage of the safe shutdown equipment

The conditional probability of damage of the safe shutdown equipment is obtained by comparison between the duration of the fire scenario and the damage time of the target.

- If the damage time is greater than to the scenario duration, the conditional probability of damage of the safe shutdown equipment is equal to zero,
If the damage time is lower than to the scenario duration, the conditional probability of damage of the safe shutdown equipment is the frequency of the scenario.

D. EXAMPLE OF FIRE SCENARIOS STUDY IN A COMPARTMENT

Study of a fire in the room W403 in the electrical building:

This room is not equipped with automatic extinguishing system. One ignition source in this local is the electrical cabinet LBB (6,6 kV). Its fire frequency is $3,1 \times 10^{-4}$/year reactor.

1. Quantification of the fire scenarios

The first event tree "fire barriers" is quantified to determine the frequency of each configuration of the openings. The event tree quantified is the following:

<table>
<thead>
<tr>
<th>Init.</th>
<th>Doors closure</th>
<th>Fire dampers closure</th>
<th>Frequency (1/year reactor)</th>
<th>Configuration</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Door initially closed</td>
<td>Door closed by the field operator</td>
<td>Failure of the fire damper mechanism</td>
<td>Failure of the fuse</td>
</tr>
<tr>
<td>3.10E-04</td>
<td>0.990</td>
<td>0.861</td>
<td>0.993</td>
<td>0.700</td>
</tr>
<tr>
<td></td>
<td>0.916</td>
<td>0.700</td>
<td>0.007</td>
<td>0.300</td>
</tr>
<tr>
<td></td>
<td>0.139</td>
<td>0.993</td>
<td>0.993</td>
<td>0.700</td>
</tr>
<tr>
<td></td>
<td>0.530</td>
<td>0.139</td>
<td>0.007</td>
<td>0.300</td>
</tr>
<tr>
<td></td>
<td>0.139</td>
<td>0.993</td>
<td>0.993</td>
<td>0.700</td>
</tr>
<tr>
<td></td>
<td>0.861</td>
<td>0.007</td>
<td>0.007</td>
<td>0.300</td>
</tr>
<tr>
<td></td>
<td>0.139</td>
<td>0.993</td>
<td>0.993</td>
<td>0.700</td>
</tr>
<tr>
<td></td>
<td>0.861</td>
<td>0.007</td>
<td>0.007</td>
<td>0.300</td>
</tr>
<tr>
<td></td>
<td>0.139</td>
<td>0.993</td>
<td>0.993</td>
<td>0.700</td>
</tr>
<tr>
<td></td>
<td>0.861</td>
<td>0.007</td>
<td>0.007</td>
<td>0.300</td>
</tr>
</tbody>
</table>

The frequency of the different configurations are the following:

- configuration 1 (fire doors and fire dampers closed fifteen minutes after the fire ignition): $2,66 \times 10^{-4}$,
- configuration 2 (fire doors closed and fire dampers open during all the fire development): $4,35 \times 10^{-5}$,
- configuration 3 (fire doors open during all the fire development and fire dampers closed fifteen minutes after the fire ignition): $7,95 \times 10^{-7}$,
- configuration 4 (fire doors and fire dampers open during all the fire development): $1,35 \times 10^{-7}$.
Only the two first configurations are presented in this example.

The second event tree "fire event tree" is quantified for each configuration studied. The quantification for the configuration 1 is the following:

<table>
<thead>
<tr>
<th>Config.</th>
<th>Self extinguishing</th>
<th>Fire detection</th>
<th>Fire extinguishing</th>
<th>Probability</th>
<th>Time</th>
<th>n° scénario</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.429</td>
<td>0.365</td>
<td>0.789</td>
<td>1.14E-04</td>
<td>0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.211</td>
<td>0.750</td>
<td>4.38E-05</td>
<td>6</td>
<td>1.1</td>
</tr>
<tr>
<td>2,66E-04</td>
<td></td>
<td></td>
<td>0.005</td>
<td>8.76E-06</td>
<td>13</td>
<td>1.2</td>
</tr>
<tr>
<td></td>
<td>0.571</td>
<td>0.444</td>
<td>0.995</td>
<td>2.91E-06</td>
<td>48</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.444</td>
<td>0.971</td>
<td>1.42E-08</td>
<td>160</td>
<td>1.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.029</td>
<td>0.029</td>
<td>3.51E-05</td>
<td>12</td>
<td>1.5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.182</td>
<td>0.182</td>
<td>2.63E-05</td>
<td>27</td>
<td>1.6</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.167</td>
<td>0.167</td>
<td>1.70E-05</td>
<td>83</td>
<td>1.7</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.633</td>
<td>0.976</td>
<td>5.11E-07</td>
<td>160</td>
<td>1.8</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>0.024</td>
<td>2.92E-06</td>
<td>59</td>
<td>1.9</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.182</td>
<td>0.182</td>
<td>1.42E-05</td>
<td>92</td>
<td>1.10</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>0.024</td>
<td>3.55E-07</td>
<td>160</td>
<td>1.11</td>
</tr>
</tbody>
</table>

The quantification for the configuration 2 is realised in the same way.

2. Estimation of the targets damage time

The fire simulation results are the following:

<table>
<thead>
<tr>
<th>Source LBB</th>
<th>Malfunction time of the others electrical cabinets</th>
<th>Malfunction time of the cable tray P1W4481B</th>
<th>Malfunction time of the cable tray P1W4501B</th>
<th>Malfunction time of the cable tray P1W4P51A</th>
</tr>
</thead>
<tbody>
<tr>
<td>Configuration 1</td>
<td>9 minutes</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>Configuration 2</td>
<td>9 minutes</td>
<td>47 minutes</td>
<td>47 minutes</td>
<td>47 minutes</td>
</tr>
</tbody>
</table>
3. Frequency of damage of the safe shutdown equipment
The results are the following:

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Frequency</th>
<th>LBB is lost at $t = 0$</th>
<th>lost of the others electrical cabinets $t_e$ (minutes)</th>
<th>lost of the cable tray P1W4481B $t_e$ (minutes)</th>
<th>lost of the cable tray P1W4501B $t_e$ (minutes)</th>
<th>lost of the cable tray P1W4211A $t_e$ (minutes)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.0</td>
<td>$1.14\ 10^{04}$</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.1</td>
<td>$4.37\ 10^{01}$</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.2</td>
<td>$8.75\ 10^{04}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.3</td>
<td>$2.90\ 10^{02}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.4</td>
<td>$1.42\ 10^{02}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.5</td>
<td>$3.50\ 10^{05}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.6</td>
<td>$2.62\ 10^{05}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.7</td>
<td>$1.70\ 10^{06}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.8</td>
<td>$5.10\ 10^{09}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.9</td>
<td>$2.92\ 10^{09}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.10</td>
<td>$1.42\ 10^{09}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>1.11</td>
<td>$3.54\ 10^{07}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>2.0</td>
<td>$1.87\ 10^{05}$</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>2.1</td>
<td>$7.17\ 10^{06}$</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>2.2</td>
<td>$1.43\ 10^{06}$</td>
<td>9</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
<td>safe</td>
</tr>
<tr>
<td>2.3</td>
<td>$4.76\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.4</td>
<td>$2.32\ 10^{09}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.5</td>
<td>$5.73\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.6</td>
<td>$4.30\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.7</td>
<td>$2.78\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.8</td>
<td>$8.36\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.9</td>
<td>$4.78\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.10</td>
<td>$2.33\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>2.11</td>
<td>$5.81\ 10^{06}$</td>
<td>9</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
</tr>
</tbody>
</table>

The identical scenario are grouped together. The final results are:

- lost of the electrical cabinet LBB only with a frequency of $1.83\ 10^4$/year-reactor,
- lost of the electrical cabinet LBB and the other electrical cabinets in the room, 9 minutes after, with a frequency of $1.19\ 10^7$/year-reactor,
- lost of the electrical cabinet LBB, the other electrical cabinets 9 minutes after and all the cable tray in the room 47 minutes after with a frequency of $6.21\ 10^4$/year-reactor.

REFERENCES:

APPENDIX 1

The transition graph concerning the means of fire extinguishing and detection can be used to focus on the different events to be taken into consideration when establishing fire scenarios.
APPENDIX 2

Fire barriers and automatic extinguishing event tree

<table>
<thead>
<tr>
<th>Initiating event</th>
<th>Automatic extinguishing</th>
<th>Door closure</th>
<th>Fire dampers closure</th>
<th>Probability</th>
<th>Situation</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>0.035</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.00</td>
<td></td>
<td>0.010</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.165</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.090</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.099</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.007</td>
<td></td>
<td>0.300</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.061</td>
<td></td>
<td>0.993</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.993</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.007</td>
<td></td>
<td>0.700</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.061</td>
<td></td>
<td>0.993</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.993</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.007</td>
<td></td>
<td>0.88</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.061</td>
<td></td>
<td>0.993</td>
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Fire event tree

114
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<tr>
<th>Situation (1), (2), (3), Q &gt; (4)</th>
<th>Self extinguishing</th>
<th>Fire detection</th>
<th>Fire extinguishing</th>
<th>Probability</th>
<th>Time</th>
<th>n° scenario</th>
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</thead>
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<tr>
<td></td>
<td></td>
<td>local detection</td>
<td>automatic detection</td>
<td>first intervention</td>
<td>Intervention by Internal team Fire-fighters</td>
<td>Intervention by External fire brigade</td>
</tr>
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<td>1,000</td>
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APPENDIX 3

Quantification of human error for the fire PSA

The model used for the probabilistic quantification of human error depends of the operating procedures at the disposal to the field operator for the fire fighting.

In France, for the detailed study, human reliability related to the field operator has been analysed. The field operator, in case of fire, has to apply FAI-RD (auxiliary systems actions sheets). The purpose of the FAI-RD is:

- confirm to the control-room that there is a fire,
- verify the closure of fire doors,
- close the fire dampers,
- activate the fixed extinguishing system,
- open the valves of the smoke reducing system and ask the control room to activate this system.

Concerning the confirmation of the fire and the activation of the fixed extinguishing system, the human error is estimated from fire experience feedback. For the verification of closure of the fire door and the closure of the fire dampers, a specific model has been elaborated. This model is adapted from the SWAIN model.

Considering that the fire has been identified, the human error related to fire door and damper closure is estimated by:

\[ P = P_b \times K_f \times P_{nr} \]

where:

- \( P_b \) is a basis probability corresponding to the omission of a main stage of FAI-RD (the failure probability adopted is \( 6 \times 10^{-2} \)).
- \( K_f \) is a context factor (it is considered that the actions to be performed by the auxiliary operator are easy but the presence of smoke is an handicap and the quality and the training of the FAI-RD are not considered sufficient). Taking into account these considerations, the \( K_f \) value estimated by a decision tree is 9.
- \( P_{nr} \) is the non-recovery probability (due to the fact that in the case of this kind of error no alarm is emitted, a high non-recovery probability was allocated; \( P_{nr} = 0.6 \)). These values lead to a human error probability of 0.3 for non-closure of the fire dampers and a probability of 0.3 for non-checking of the fire door closure.

Concerning the fire detection and extinguishing, the probability of human error is estimated from operating experience feedback.
Fire Risk Analysis for Novovoronezh Nuclear Power Plant in Russia

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INTRODUCTION

A fire risk analysis has been initiated for the Unit 5 of Novovoronezh Nuclear Power Plant (NVNPP-5), the first Russian VVER-1000 reactor. The analysis is carried out within the framework of the joint Swiss-Russian Probabilistic Safety Assessment (PSA) project (SWISRUS). The main objective of the SWISRUS' project is to assist in the training of the technical staff of the Scientific and Engineering Center for Nuclear and Radiation Safety (SEC NRS), which is the technical support organization of the Federal Nuclear and Radiation Safety Authority of Russia (i.e., Russian Federation Gosatomnadzor [GAN]), for performance and application of PSAs in safety evaluation of Russian nuclear power plants [1]. The PSA project is carried out under the technical direction of the Swiss Federal Nuclear Safety Inspectorate (HSK). Energy Research, Inc. (ERI) assists HSK in project management, training, technical support and review.

The first phase of SWISRUS project, Level 1 PSA for internal initiators of Novovoronezh NPP Unit 5, started in November 1994, and was completed in June 1998 [2]. This PSA was performed by members of SEC NRS and the Novovoronezh plant.

The second phase of the SWISRUS project, comprising of living PSA, external events and level-2 PSA tasks, which is currently ongoing, is scheduled to be completed by July 2000. The fire risk analysis discussed in this paper is part of this second phase. The study is one of a few fire PSAs performed for Russian nuclear power plants. This paper presents the methodology and some of the preliminary results of the study.

1. MAIN FEATURES OF NVNPP-5

The Novovoronezh Unit 5, a nuclear power plant rated at 1000 MW(e), is a water cooled, water moderated VVER Pressurized Water Reactor (PWR) that started commercial operation on May 30, 1980. It is the first VVER-1000 type nuclear power plant that was designed and constructed in the former Soviet Union.

The NVNPP-5 reactor coolant system includes the reactor, a pressurizer, and four coolant loops, each connected to a horizontal steam generator and a main reactor coolant pump. Each coolant loop also includes two valves for isolating the steam generator from the reactor vessel. The radioactive coolant

* The SWISRUS project is the result of cooperation between the Swiss and the Russian Nuclear Regulatory Authorities, and is sponsored by the Swiss Government, Agency for Development and Cooperation, Division of Cooperation with Eastern Europe and the Commonwealth of Independent States (DEZA/AZO).
circuit equipment are enclosed in a concrete containment building which is designed to withstand an internal pressure of 0.45 MPa.

The secondary circuit consists of four steam generators, two steam-driven main feedwater pumps, two turbine-generators and related appurtenances. There are emergency feedwater pumps that are located in the basement of the turbine building. The secondary side is also equipped with Fast Acting Isolation Valves (FAIV) on the main steam lines.

The main elements of plant layout include the containment, the auxiliary building, the turbine building, the diesel generator building and intake structures. The auxiliary building can be envisioned in two parts: the hot (radioactive) area and electrical and control area. The hot area houses pumps, piping, valves and supply tanks. The electrical and control areas are located between the main part of the auxiliary building and turbine hall. It houses electrical power and control buses, batteries, the cable chases and cable spreading rooms. The turbine building houses the two turbines and related pumps and equipment. The fire protection related equipment control panels are located in a compartment that is considered as part of the turbine building.

The hot area consists of several compartments and corridors that are in the majority of cases connected to each other via openings in the walls. Redundant pumps of the same system are often installed inside the same compartment. To control contamination, the floor of the majority of the areas is covered with a special resinous material. The material is suspected to be somewhat combustible and to emanate toxic fumes as a result of combustion.

The Main Control Room (MCR) is a large area where the main control panel is located. There are two adjacent rooms that are generally open to the MCR. These rooms house the relays and the main computer. The MCR has a false ceiling that is made of metal sheets with ventilation holes in them. The smoke detectors of the control room are installed under the false ceiling and there are no detectors under the main ceiling. The fire suppression system of the control room includes CO₂ fire extinguishers, and two hose reels installed on the walls immediately outside the control room doors. The main access to the control room is from the turbine building.

An important fire related feature of NVNPP-5 is the alternate or reserve control room (RCR), which was incorporated into the original design of the plant. Redundant trains of safety related equipment can be controlled from this room. All the control keys in MCR and RCR are operable at the same time. However, the control circuit for regulating valves includes a control switch that overrides the control from the MCR and makes RCR as the main control point of the affected equipment.

2. **FIRE HAZARD ANALYSIS**

Fire hazard analysis, as it is defined in References [3] and [4] has not been conducted for NVNPP-5. Therefore, when starting the fire PSA task, little or no information was available regarding fire zones, fire areas, and routing of the cables. The fire PSA team had to develop an approach for defining fire areas and fire zones (Section 3.3 below) and for identifying cable routing information. In a typical fire hazard analysis conducted per such guidelines as Reference [4], all plant areas are addressed with almost equal level of attention. For each compartment the combustible loading, fire detection and fire suppression systems, accessibility of fire brigade and safety related equipment and cables are identified. Safety related cables are generally defined as those that are needed for safe shutdown. Safe shutdown is defined through a model of plant systems that is focused on identifying system combinations that can be used for safe shutdown. This is different from the PSA approach which is focused on core damage. Fire PSA approach allows the analyst to discriminate among fire areas and fire zones. Only those compartments are analyzed in detail that are suspected to be important to fire risk. The level of analysis varies significantly among different compartments. Combustible loading and fire detection and suppression features, as an example,
are addressed only for those compartments that did not screen out in the initial screening steps. The potential for a very severe fire that may affect safety and non-safety related compartments was addressed. Some areas of the plant were screened out based, simply, on site inspection.

3. OVERVIEW OF FIRE PSA APPROACH AND ANALYSIS DONE

The fire PSA is based on a modified version of the methodology described in References [5] and [6]. The methodology in References [5] and [6] is based on the assumption that some level of information is available for the location of equipment and cables and definition of fire areas and fire zones. For NVNPP-5, the initial tasks were to establish the fire zones and to collect equipment location and cable routing information. Based on that information, a series of screening steps were undertaken with graduating levels of detail. In the initial steps, fire zones were screened out based on an on-site inspection of the area. In later steps, a preliminary core damage analysis was conducted to establish a conservative estimation of the core damage frequency associated with a fire zone. The level-1 PSA model for internal initiators, developed with a large fault tree/small event tree (LFT/SET) approach, was used in the fire PSA to establish the importance of various fire scenarios. A list of those equipment that were identified in the fault trees and event trees were prepared as the "PSA Components" list. The fault trees and event trees were developed and quantified using the IRRAS computer code [7].

Some of the main steps of the fire PSA are discussed below.

3.1. INITIATING EVENTS CAUSED BY A FIRE

The initiating events (IE) identified in the internal events PSA were reviewed and those that can be caused from a fire event were identified. The full list of IEs from the Final Report on Level-1 PSA for Internal IEs was taken as the basis for the analysis of IEs caused by a fire. Some of the IEs were excluded from further analysis as they were identified to be impossible to occur because of a fire (for example, steam generator leaks from primary to secondary circuit, feedwater header pipe rupture, and feedwater pump suction line rupture). Also some of the IEs that were originally excluded from internal events analysis based on negligible probability of occurrence were added back to the list to ensure that the analysis is based on "complete" information. The following is a partial list of initiating events that were added back to the list:

- Simultaneous closure of main isolation valves on several loops
- Simultaneous closure of FAIVs on all steam lines
- Major loss of instrumentation circuits (flying blind syndrome)
- Simultaneous inadvertent opening of several steam relief valves
- Breach in high and low pressure interface.

For a small group of IEs, fault trees were developed to identify the combination of component failures that can lead to the specific initiating event. From this analysis, those components that the failures of which can lead to an initiating event were identified and added to the PSA equipment list. In particular, the high and low pressure interfaces between the main coolant loop of the reactor and other systems were analyzed to make sure that all the pathways connected to the main coolant loop and low pressure systems have been identified. Almost in all cases, more than one valve has to fail inadvertently to initiate an interfacing LOCA. Based on actual NVNPP-5 experience, it was concluded that those paths that include check valves should be included in the high-low pressure interface analysis. Since measures are not implemented for monitoring leak tightness of the check valves and at least in one incident a check valve had leaked through, the reliability of the check valves was considered suspect.
In order to reduce the number of IEs to a manageable level and to facilitate the screening analysis, some of the IEs were grouped together using conservative assumptions. The IEs were grouped based on similarities in plant response and therefore the choice of proper event tree (ET) for modeling the impact of the IE. For each IE group, the most conservative IE was used to represent the group. The following is a sample of IE groups that were used in the analysis:

- ADSH  - Administrative Shutdown
- FAIVALL  - Spurious Closure of All FAIVs
- FWPT  - Loss of Main Feed Water
- LCPS  - Loss of Circulation in Primary Side
- LOOP  - Loss of Off-Site Power for 24 Hours
- etc.

A default IE was defined that would be assigned to all fire scenarios, unless it can be proven that another more severe IE can occur. Use of default IE allowed us to forego identification of all control and instrumentation equipment and circuits that have the potential for causing an initiating event, and thus, somewhat reduced the scope of the effort for collecting cable routing information. Of course at those areas (e.g., some of the pump rooms) where the function of all the cables present in the room can be clearly identified, the default IE is replaced with a reactor trip. The default IE was chosen to be Loss of Main Feedwater. This IE includes loss of main feedwater system and closure of turbine stop valves. Use of the event tree associated with loss of main feedwater in place of the transient event tree is conservative, because it covers more severe accident progression than each of the IEs in the class of IE denoted as transients. The IE groups and associated event trees from internal events PSA model were used for constructing the fire risk model. In addition, several new ETs for those IE groups that were not included in internal events PSA were developed as part of fire risk analysis task.

3.2. CABLE ROUTING AND COMPONENT LOCATION INFORMATION

Prior to the initiation of the fire PSA task, there were no suitable documents or computerized data bases that would provide a catalog of cable and equipment location. A major effort was initiated to create such information for those cables and components that were of interest to the fire PSA effort. The components of interest were labeled as “PSA Components”. To create the list of PSA components, the event trees associated with the IEs discussed in the preceding section and associated system fault trees from the internal events PSA were reviewed and components susceptible to damage from a fire were identified and tabulated as the PSA Components. Also, additional components identified in the initiating event task and a list of key instrumentation circuits was added to the PSA component list. Some of the components added to the list from internal events PSA comprise of (a) power supply system breakers, (b) primary circuit isolation valves at the containment interface, and (c) instrumentation related components.

Special assumptions had to be made to limit the scope of cable routing effort without compromising the results of the fire risk analysis. In particular, because of the definition of the default IE, cable information on equipment related to main feedwater system operation was not collected. This assumption did not lead to excessively conservative results. However, it leads to a considerable reduction in the efforts needed for cable information collection.

Cable routing information was collected for the power, instrumentation and control cables associated with all the PSA components. The documents that were used to identify cable routing included hand-written notes created by the electricians at the time of plant construction regarding cable locations in terms of
cable trays and cable shafts. A former plant operator, who is intimately familiar with the electrical design and layout of the plant, translated the electricians' notes into cable routing information in terms of component serviced, cable type, compartment and cable tray. Power cable related information was collected for all the compartments starting from the component itself all the way to the associated breakers in motor control center (MCC) or switchgear cabinets. For instrumentation and control cables the routing information was collected: (a) from the power source to limit switches (for valves); (b) from the power source to interlock panels in MCR, RCR, safety system compartments, and etc.; and (c) from sensors to control/information panels in MCR, RCR, safety system compartments, and etc. The cable routing information collected was verified with plant layout drawings to ensure that there is no inconsistency in cable paths.

The information was compiled in a specialized data base system, developed using Microsoft Access environment. It included all the forms, tables, print images, search procedures, etc. The data base is referred to as the Data Base on PSA Components and Cables vs. Plant Compartments. It contains 21'000 records of cable location information and 3'000 records on component locations.

The data base allowed us to establish the contents of a compartment in terms of components and cables, and to identify, if needed, the compartments where the cables of a specific component is routed through. The data base possibilities are demonstrated in Fig.1:

3.3. **FIRE ZONES DEFINITION**

Definition of fire zones included three steps:

1) Dividing of the plant into major buildings;
2) Definition of fire zones based on drawings and documents available;
3) Performing plant walkdowns to validate fire zone boundaries defined in Steps 1 and 2.

The plant was divided into reactor building, turbine building, auxiliary building, intake structure, yard area, and diesel generator building, based on such natural boundaries as perimeter walls and substantial structural features (e.g., the containment wall).

In Step 2, the plant was divided into smaller fire zones using a set of criteria specifically defined for this task. Since the rating of the boundaries in Novovoronezh are significantly less than 3-hours and many of the boundaries include openings of varying sizes, it was deemed that the criteria employed by the power plants in the Unites States of America or Western Europe would not lead to a useful set of fire zones. Therefore, the following criteria were developed:

1) If a compartment is defined by 1.5-hour fire rated boundaries (i.e., ceiling, walls, floor and doors) and there are no windows, the compartment is considered as a separate fire zone.

2) If a ventilation duct is equipped with an automatic isolation device (e.g., fire damper), the associated boundary is used in defining fire zones.
3) If a compartment does not have 1.5-hours fire rated boundaries, but does not contain combustible materials or contains restricted amount of fire loads, those compartments are defined as separate fire zones.

4) Stairways and stairwells in buildings are considered as unique fire zones located vertically through the entire height of the stair shaft if they are separated from other plant compartments by a hallways or corridor and their doors are fire rated as 0.5 hour or better.

5) Elevator shafts (passenger and freight) are identified as separate fire zones located vertically through the entire height of the shaft if they are separated from other plant compartments by a hallways or corridor and their doors are fire rated as 0.5 hour or better.

6) Ventilation shafts which traverse vertically from the ventilation room through different elevations of the building are typically defined by concrete walls. Such ventilation shafts often include penetrations for connecting ventilation ducts. These ducts are automatically isolated in case of fire.

7) Cable shafts are considered as separate fire zones. It is assumed that no fire can propagate from other compartments into a cable shaft by breaching the boundaries of the shaft.

Using this criteria, all plant buildings were divided into fire zones. A convenient coding scheme for naming the fire zones was also developed as part of this task.

As a result of first two steps, a preliminary list of fire zones was compiled, and boundaries of the fire zones were drawn on layout drawings. Several plant walk downs were conducted to verify the characteristics of selected boundaries and validity of fire zone definitions. Based on the information gained during these walk downs, the list of fire zones was finalized. In total, 450 different fire zones were
defined. The containment and the turbine hall, given the large, interconnected areas, were treated each as a single fire zone.

3.4. PLANT WALKDOWNS

As it is mentioned above, several plant walkdowns were performed by the fire risk analysis team to ensure that the information used in the analysis reflects the “as-built” conditions. A set of checklists were developed specifically for this purpose to document the observations for each fire zone regarding the amount and type of combustibles, the type and number of ignition sources, fire detection and suppression systems available for the fire zone, the characteristics of the boundaries, etc. From these walkdowns it was discovered that the boundaries of many of the fire zones were not leak-tight. In particular, it was noted that there are holes on doors and gaps between the doors and door frames. In a few cases, openings (large and small) and boles were discovered in fire zone boundaries. For example, the fire zone where make-up pumps, heat exchangers and oil tanks are located has a series of sizable ventilation openings in the walls. These openings are equipped with dampers that close under the weight of a counter weight but would open to allow for air to escape the make-up pump room. Thus, in case of a fire, the dampers would allow hot gases and smoke to propagate outside the compartment of origin.

The information collected in the course of the walkdowns is compiled in a living document that is modified and updated as more information is collected. For example, for those fire zones that do not screen out and detailed analysis is needed, the detailed information regarding equipment and cable configurations are included in this document as well.

3.5. MULTI-COMPARTMENT SCENARIOS

Multi-compartment fire analysis was determined to be the focus of this fire risk analysis. From the information obtained during the walkdowns regarding the conditions of fire boundaries it was concluded that single compartment fire would be the exception and most fire incidents will affect multiple compartments. Therefore, unlike the methodologies prescribed in References [5] and [8], the initial screening of fire zones assuming that the fire is confined to the zone itself was not employed here. The analysis began by postulating a severe fire in each fire zone. All possible propagation paths and mechanisms (i.e., radiative heat, hot gases, and smoke) were considered and documented. A computer program was developed to automate the search for the compartments where the effects of a fire would propagate. The following criteria was developed for this purpose:

<table>
<thead>
<tr>
<th>Type of opening</th>
<th>Propagation Possibility</th>
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</thead>
<tbody>
<tr>
<td>Door, hermetic</td>
<td>No fire propagation</td>
</tr>
<tr>
<td>Door, non-hermetic, unlocked, opens away from the compartment</td>
<td>Hot gases and smoke propagate</td>
</tr>
<tr>
<td>Door, non-hermetic, unlocked, opens into the compartment</td>
<td>Smoke propagates</td>
</tr>
<tr>
<td>Door, non-hermetic, locked</td>
<td>Smoke propagates</td>
</tr>
<tr>
<td>Opening, large (D&gt;150 mm)</td>
<td>Hot gases and smoke propagate</td>
</tr>
<tr>
<td>Opening, small (D&lt;150 mm), located-higher than 1/2 compartment height</td>
<td>Hot gases and smoke propagate</td>
</tr>
<tr>
<td>Opening, small (D&lt;150 mm), located lower than 1/2 compartment height</td>
<td>Smoke propagates</td>
</tr>
</tbody>
</table>
In addition the following assumptions were made:

1) Hot gases propagated through openings may fail cables and equipment in an adjacent compartments.

2) Hot gases having propagated to an adjacent fire zone would not propagate further if the volume of that zone is deemed to be large (this assumes that the fire will be extinguished within a reasonable time).

3) Fire would not propagate through Transportation Corridor. This is a very long corridor with high ceiling that is normally empty and is located at a lower elevation of the plant.

4) Fire in areas near a cable shaft will not fail the shaft boundaries and propagate into the shaft from outside (the penetrations are assumed to be tight).

5) Cable penetration seals are assumed to be perfect fire barriers unless it is otherwise noted in the walkdown observations.

6) The paste applied to cables, for the initial screening, is assumed to be only partly effective.

7) Counter-weighted dampers that close on excessive pressure are considered to be effective barriers for hot gas, but not for smoke propagation.

8) For ventilation ducts and openings:
   - If the fire dampers are not equipped with automatic closure mechanism based on temperature rise or smoke concentration, they can act as a pathway for smoke propagation;
   - If fire dampers are not installed, both hot gases and smoke may propagate through.

9) Hot gases and smoke propagate upwards. The fire zones located at an elevation lower than the fire zone where the fire originates will not be affected.

A set of fire scenarios were identified using the above listed assumptions and criteria. These scenarios were used in the initial screening assuming that all equipment and cables inside the affected fire zones are failed in their worst possible position.

3.6. FIRE INITIATION FREQUENCY ASSESSMENT

Fire initiation frequencies were established using generic and plant specific data, and the two-stage Bayesian approach. The prior distribution is based on the data presented in Reference [9] for different types of components. The following components or fire sources were considered:

- Turbine/generator
- Diesel-generator
- Pump - electric motor
- Cable
- Electrical panel
- Transformer
- Inverter
- Switch.

The fire incidence data from Kalinin and Novovoronezh Units 3 and 4 NPPs were used in the first stage. In the second stage, the data from Novovoronezh NPP Unit 5 were used as evidence. In both cases (i.e., first and second stages) only those fire events (ignitions) that had the potential to grow into a fully developed fire were accounted for. The frequency of fires from transient combustibles were, for initial screening, taken from Reference [9].

The fire initiation frequencies of the fire zones were calculated by summing fire initiation frequencies of the components that are located in the fire zone.
4. INITIAL SCREENING OF FIRE SCENARIOS

The goal of the initial screening task is to screen out fire scenarios that, using conservative methods and assumptions, can be shown as risk insignificant. Level-1 internal event PSA model was used for this purpose. The core damage frequency (CDF) associated with each fire scenario was estimated and those scenarios with CDF less than \(1 \times 10^{-6}\) per reactor year were screened out. For the initial screening, the following assumptions were made:

1. All the equipment susceptible to fire and cables in the affected fire zones where fire started are damaged.

2. All cables that are not covered with fire retardant material and electrical equipment that are exposed to hot gases are damaged.

3. Cables that are covered with fire retardant material may fail only when completely engulfed in flames.

4. A small fraction of cables that are covered with fire retardant material may fail from exposure to hot gases.

5. A fire damages the equipment and electrical circuits (via damage to cables) in their worst failure mode (i.e., faulty signals appear if there is a potential for that, equipment fails in the worst position, etc.). Worst position is determined from the accident sequences of the internal events PSA model.

6. Smoke can only damage electronic equipment.

7. No operator actions can be postulated in fire zones that are affected (either via hot gases or smoke) by the fire.

8. No credit was given to the fire brigade and fire detection and suppression systems.

Preliminary, initial screening analysis has shown that most significant scenarios deal with the following plant areas:

- Turbine hall
- Main control room
- Cable spreading rooms
- Cable shafts
- Containment (loss of all vital instrumentation)
- Make-up system compartment
- Compartment housing high pressure and low pressure pumps.

The fire scenarios associated with these areas are being analyzed using detailed information, less conservative assumptions and more accurate plant impact scenarios than those used in the screening phase. Also, the analysis includes human actions and probabilistic model of fire growth, detection and suppression.

5. TURBINE HALL FIRE

The turbine hall contains the two steam turbines, the main generators of the unit and related appurtenances, which include lubricating oil storage tanks and system, and hydrogen piping. The turbine building is protected by a series of fire monitors (stationary water canons) that are installed at strategic locations on catwalks above the turbine deck. The entire building can be sprayed with water from these monitors. At Novovoronezh 5, the main entrance to the main control room is from the turbine hall at the same elevation where the fire monitors are located. The door of the main entrance is of heavy construction to withstand heat impingement and missile impact. The fire rating of the door is estimated to
be 3-hours. There is a second opening from the control room proper (i.e., the control room and the two adjacent compartments that are normally open into the control room) into the turbine hall that has been closed and sealed. It must be noted that the other door of the control room proper that opens into an area away from the turbine hall is bolted shut. To open that door, the use of a heavy wrench will be necessary.

Above the control room, the air handling units for the control room and other parts of the plant are located. Above the air handling equipment, the deaerators and main steamline piping are located. This is a large compartment that spans the entire length of the Turbine Hall and is open into the Turbine Hall. At the wall, next to the opening into the Turbine Hall, motor control centers are located that control the steamline related equipment, which includes the fast acting isolation valves and steam dump valves.

A large, smoky fire in the Turbine Hall will lead to the failure of normal and emergency feedwater systems. This includes the turbine driven feedwater pumps, that provide normal feedwater to the steam generators. The smoke may propagate into the deaerator area, into the stairwells at the two end of the turbine hall and from there into the air handling equipment area. Smoke or even hot gases entering the deaerator area may lead to failures in the MCCs there and closure of fast acting isolation valves and loss of steam-dump valves. The normal passage into the control room will be blocked by either flames or heavy smoke. Smoke may enter the control room via the air handling system. Also, there is a concern regarding the fire rating of the steel wall separating shift supervisor’s room from the turbine hall, which was not part of the original plant design. It must be noted that the ventilation system serving the control room does not include a recirculation mode and therefore to isolate if from outside effects, the operators switch off the system. Face masks with filter canisters are located within the control room proper. The operators are trained in donning these masks. Thus, some credit can be given to operator actions from the control room.

6. AUXILIARY BUILDING

The auxiliary building includes all the make-up pumps, heat removal pumps and heat exchangers and related piping, valves and control circuits. Many of the redundant trains are located in the same compartment. Also, in addition to normal openings between different parts of the building, several pipe or other penetration openings were discovered during fire PSA walkthroughs. Therefore, a severe fire at one point (especially a low point) of the building has the potential of sending smoke and hot gases to practically the entire building. The auxiliary building, from fire propagation stand-point, can be divided into two parts: north and south of the transportation corridor. Since the transportation corridor is located at a relatively high elevation, a fire in one side of the building may not affect the other side of the building. The upper elevations of the building contain maintenance and service shops that do not include safety related equipment or cables.

In the south of the building, at the lowest elevation, three booster pumps are located for the make-up system that contain large quantities of lubricating oil. A severe fire is possible in these rooms. The smoke and hot gases from these rooms will propagate into the basement area underneath the containment and affect practically all pumps and related electrical panels. In this scenarios hot shutdown related equipment and cables remain available. Therefore, if the plant can be maintained at hot shutdown, core damage can be avoided. However, plant operating requirements (technical specifications) stipulate that when two safety systems are failed operators have to take the plant to cold shutdown. The failure probability of cold shutdown, given the equipment that may be exposed to the effects of this fire scenario, is found to be very large. Therefore, this specific scenario is concluded to be risk significant and will be analyzed in detail.

Another fire scenario in the auxiliary building could not be screened out using conservative assumptions. A fire in the area underneath the containment where all high and low pressure safety related pumps are located, may lead to failure in the ability to achieve cold shutdown. Per our methodology we have assumed that a reactor trip would occur. Hot shutdown can be maintained. However, per plant operating
requirements, operators must initiate cold shutdown procedure which is impossible to implement given the equipment failed because of the fire.

7. CONCLUSIONS

To obtain a complete set of risk contributors for Novovoronezh 5, the fire risk analysis task has been added to the overall PSA effort. The internal events PSA has already been completed and is used as the basis for determining the equipment and cables that may influence the core damage frequency and for defining the equipment failure and human action sequences leading to core damage. A list of components that were identified in the internal events PSA were prepared as the PSA components. However, several other components had to be added to this list. From a thorough review of the assumptions underlying the internal events PSA it was concluded that the list of initiating events had to be expanded to include those that may occur from a fire but were omitted because of negligible likelihood of occurrence under other conditions. Also, for some of the initiating events that were treated statistically in the internal events PSA, fault tree models were developed to identify those components, the combination of which in their failed state may lead to a specific initiating event. Thus from these efforts, further components were identified and added to the PSA component list.

The cable routing information was not readily available. Using the logs of the electrical technicians during cable pulling phase of construction, the cable routing information could be established. This proved to be an extremely time consuming effort and therefore, several simplifying assumptions had to be made to minimize the list of cables. However, for every component the power, control and instrumentation cables were traced. The primary circuit instrumentation related cables were added to this list.

Since a general fire hazard analysis had not been conducted for this plant, fire zones had to be defined as part of this effort. A criteria was developed for this purpose and the plant was partitioned into 5 separate structures and 450 fire zones. Several walkdowns were conducted to verify the partitioning. However, it was soon concluded that many fire compartments have openings among them and therefore, single compartment fire analysis did not seem to be a useful exercise. Therefore, using the information obtained during the walkdowns a series of fire scenarios were identified based on conservative assumptions that start with a fire in a fire zone and propagates (i.e., smoke and hot gas) to other fire zones.

An initial screening of the fire scenarios was conducted using the internal events PSA model and from that a short number of fire scenarios were identified for further analysis. It is interesting to note that despite the interconnection among various compartments in the auxiliary building and presence of redundant trains in the same areas, only a small number of fire scenarios could not be screened out. The Turbine Hall, the Main Control Room and cable spreading rooms were concluded to need further analysis. The Turbine Hall is of particular interest, because a severe fire in this area has the potential for affecting the steam dump system, the main and emergency feedwater systems and perhaps the control room habitability. It is possible for the control room operators to be trapped inside the control room during such a fire.

In summary, the fire risk analysis, using the PSA tool, has succeeded in identifying several important fire risk scenarios. These scenarios are currently being analyzed in further detail. The overall core damage frequency will be estimated and the risk significant fire scenarios will be identified.

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Fire PSA for NPP Paks in Hungary

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ABSTRACT: The paper summarises and gives an overview of the fire PSA procedure performed for unit 1 of NPP Paks, Hungary. It highlights the issues associated with the collection of data and information needed for the fire PSA modelling. The fire PSA has been supported by extensive deterministic analyses. A part of these deterministic results together with other PSA related information and data have been collected and linked in a data base in MS Access environment developed for the purpose of providing direct input to the construction of accident sequence models of fire scenarios. A number of steps within the fire PSA have been supported by this tool. The conclusion from the analysis focuses on components and compartments to which the fire PSA results are most sensitive.

1 INTRODUCTION

One of the analyses have been performed in the frame of the AGNES project (Advanced General and New Evaluation of Safety) was the fire safety analysis. The goal was analysing the adequacy of the fire protection implemented in the NPP Paks. The approach was deterministic. The input to the screening process for systems and compartments to be considered was derived from the Level-1 PSA results. The findings of this analysis directed the attention to the importance of including the fire into the Level-1 PSA model of the plant.

Level 1 PSA studies have been performed for all the four units of the Paks NPP within (1) the national AGNES project started for the re-evaluation of the safety of the plant on the level of the 90's, (2) the Periodic Safety Review of units 1 and 2 and (3) its follow-up analyses. Those studies were concerned with internal initiators occurring during nominal power operational mode. In order to have a more comprehensive evaluation of the safety level an extension of the above studies was necessary with ones covering internal and external hazards. The AGNES project itself has already pointed out the importance of a PSA based evaluation of fire risk. In addition, based on international experience fire in the nuclear power plants is one of the most important hazards. Finally, the fire PSA study for the Paks NPP was required and performed within the framework of the Periodic Safety Review.

2 DETERMINISTIC ANALYSIS OF FIRE SAFETY

The primary goal of the fire safety analysis performed by ETV-ERŐTERV Rt. in the frame of AGNES was to generate information on the adequacy of fire protection of the NPP Paks. Information characterising the fire protection was available earlier only in sparse form. It was a reasonable decision to incorporate this topic into the AGNES project.

The set of components forming the subject of analysis consisted of three sources: technological components, electric supply elements and cabling, that constitutes connection between the first two. Fire zones including these components represented the subject of the fire protection analysis.
The selection of components was based on the results of the Single Failure and PSA analyses.

The cable routes of all the selected components had to be determined i.e. which compartments are passed through by the cables of each component.

It was known from the time when the power station was designed and constructed that in the power station the routes of the three safety electric supplies have common sections even in points where this arrangement is not necessary though.

In the design principles of modern nuclear power stations, safety philosophy and international recommendations, however, a completely independent routing of safety electric power supplies are proposed.

Defined as „meeting points” were the fire zones in which:

a. there are two or more technological components of safety supply
b. two or more safety electric supply routes „meet”
c. a technological component and a cable route belonging to different safety systems „meet”.

Certain further classification in grouping was applied.

Two methods were used during the analysis of the adequacy of separation:

- expert judgement
- calculations to determine if with a particular tray configuration the postulated fire event results in damage to all the three routes or not.

When three safety cable routes met on the same cable tray then it was qualified as inadequate in respect of separation.

The final set of fire zones to be analysed has been generated as a result of this phase.

After determining the compartments to be examined the adequacy of fire fighting was assessed in the following phases:

- Determining the cable routes of components located in the selected fire zones.
- Identifying the meeting points of safety supply routes and assessing the route separation adequacy in these meeting points. A computer programme was used for the assessment.
- The so called redundancy analyses checking independence of the cable routes of redundant components.
- Cable routes inside the electric supply system were determined with a view to verifying the independence of safety routes.
- A survey according to the IAEA 50-SG-D2(Rev.1.) recommendations was performed with a view to assessing the compliance of detecting, fire alarm, extinguishing system, the isolation of redundant elements and the design of ventilation systems.
The main conclusions of the study can be summarised as follows:

- Fire detectability is good in the examined fire zones.
- In case of postulated fire in certain compartments the cable trays supporting redundant cables catch fire, i.e. the degree of isolation is insufficient.
- It could be established that a 90 min. resistance to fire is sufficient to keep the postulated fire within the fire zone even if both the alarming and the extinguishing system fail. This, however, does not exclude the inoperability of certain systems for a long time and the occurrence of significant damage.
- In some instances the independence of redundant components is not fulfilled at several points of the route. On the basis of calculation analyses the degree of separation was inadequate in several compartments.
- Redundant cables run on separate trays but due to inadequate separation by distance each would catch fire in case of the postulated fire. Further cases have been identified during the walk through analysis where the separation is of „inadequate degree” because the cables of components belonging to three different safety systems run on the same trays in these fire zones.

3. FIRE PSA

3.1 Objectives and Scope

The main objective of the Paks fire PSA study was to extend level 1 PSA studies with the evaluation of the effects of fires occurring at the plant. The fire PSA was aimed at the determination of “usual” outcomes of PSA studies like (1) determination of the risk (core damage frequency) originating from fires, (2) identification of the dominant event sequences starting with fires and (3) identification of possible safety enhancement measures based on the above results.

The study was performed for unit 1 of the Paks plant. From the number of internal fires that may occur at the plant during nominal power operational mode the ones causing plant transient (initiating event) and simultaneous degradation/failure of the mitigating systems have been analysed. In accordance with the earlier level 1 PSA studies the endstate of the event sequences was considered to be successful, when removal of the decay heat after the occurrence of the initiating event was ensured for 24 hours (the process of cooling down was not modelled), whilst the only undesirable endstate was core damage. The potential fire spread was accounted for by the use of FIVE (EPRI, 1992) criteria and/or expert opinion.

One of the main features characterising the study was the excessive deterministic analysis performed to determine the consequences of fires followed by probabilistic (event tree - fault tree) modelling of accident sequences. An integrated data base of the deterministic analyses has been developed in MS Access environment, while the probabilistic modelling has been performed using the Risk Spectrum code (RELCON, 1996).

3.2 Analysis Steps

The study can be subdivided into two major phases. After some methodological preparation the first major phase, namely the phase of deterministic analyses was completed mainly in 1997. Based on its results the second major phase, the phase of probabilistic analyses was performed in 1998. This latter one included still some subtasks concerned with deterministic analyses. The flow chart of the Paks fire PSA study is given on Figure 1, illustrating the most important subtasks within the different phases.
3.2.1 Deterministic analyses

This phase was consisted of the following subtasks:

- determination of the initiating events having the potential to occur due to fire events
- definition of safety functions necessary for the mitigation of the above initiating events
- definition of safety systems fulfilling the above safety functions
- identification of the component failure modes endangering the operability of the above safety systems and/or leading directly to an initiating event by system analysis
- preliminary evaluation of fire events.
Figure 1. Major phases and subtasks of the Paks fire PSA study

A fire event was considered to cause an initiating event if it caused either a reactor trip directly or a configuration of unavailable systems that made the reactor shutdown necessary in 8 hours. Initiating events due to fires were results of inadvertent operation of the equipment. The list of initiating events of
the level 1 PSA for the nominal power operational mode has been reviewed, and the combination of equipment failures that may cause them have been defined by system (fault tree) analysis.

Functional event trees have been built up for the identified initiating events with the safety functions in headings necessary for prevention of a severe accident. Safety functions have been defined for the LOCA type initiating events and for transients. Safety functions for the LOCA type initiating events were as follows:

- reactor shutdown, ensuredance of subcriticality
- provision and make-up of primary coolant
- removal of decay heat from the sump to the ultimate heat sink
- removal of decay heat to the ultimate heat sink (in the case of closed loop secondary side heat removal)
- availability of support systems.

Safety functions for the transient type initiating events were as follows:

- reactor shutdown, ensuredance of subcriticality
- maintaining tightness of the primary circuit
- provision and make-up of primary coolant in the case if tightness of the primary circuit cannot be maintained
- removal of decay heat to the ultimate heat sink (in the case of open loop secondary side heat removal)
- availability of support systems.

The next step of the deterministic analysis was the identification of the safety systems fulfilling the above safety functions. The number of normal operational systems and stand-by safety systems as well as their combinations minimally required to fulfil the safety functions have been determined based on process and system analyses taking into account the results of thermohydraulic analyses performed to support the earlier level 1 PSA studies.

By building up fault trees for the above systems (or combination of systems) failure modes endangering the safety functions could be listed. Adding them to the list of equipment failures leading to initiating events a list of important failure modes has been set up. This consisted of 530 failure modes of 450 mechanical components.

The preliminary evaluation of fire events was aimed at the identification of fire compartments and the characterisation of their environment, i.e. description of ignition sources, combustibles, fire barriers, fire spread and fire suppression possibilities in the compartments. As a staring point each plant room was considered as a separate fire compartment. In the next step fire propagation within a compartment and fire spread between compartments were evaluated based on a set of conservative criteria developed on the basis of the FIVE method (EPRI, 1992). Some of the plant rooms were subdivided then into several fire compartments, while others were combined into a single compartment. As a result, about 820 different fire compartments have been identified.

In order to have a correct understanding and model of functional connections (through I&C and power cables) between the pieces of equipment and the fire compartments a data base has been developed in MS Access environment combining and reviewing information from the following sources:

- FONIX filing system of electrical circuit diagrams of Paks NPP
- cable route drawings
• results of computerisation of paper based cable records
• original paper based cable records
• . files of transmitter racks of Paks NPP
• files of hermetic penetrations of Paks NPP
• room lists
• equipment location schemes
• power supply schemes
• updated electrical circuit diagrams of unit 1
• data gathered during walk-throughs
• data of ("Easy Fix") seismic reinforcement
• data gathered during seismic qualification of plant components
• data of safety classification of equipment
• data of the Periodic Safety Review of units 1-2
• consumers data base of Paks NPP
• fire protection plan of unit 1.

The combined data base enables correct identification of all the mechanical, electrical and I&C equipment belonging to a given fire compartment as well as the cable routes and cables located in the given fire compartment. During the integration of the different source data bases a number of contradictions has been found. They were corrected based on walk-throughs and data base verification. After the integration of the different source data bases continuity of the cable routes of the electrical and I&C cables causing the identified failure modes of the mechanical components has been checked and corrected if needed. This resulted in the identification and checking of the routes of more than 6000 cables. The latter data base verification and correction required the largest effort of this phase and the whole study. As a result, the combined data base was able to list the component failure modes having the potential to occur due to a fire event (fire in a fire compartment) including:

• inoperability or spurious operation of the active mechanical, electrical and I/C components located in the given fire compartment
• failure (inoperability or spurious operation) of the components “at the end” of the cables damaged due to the fire in the given fire compartment.

3.2.2 Probabilistic analyses

This phase was consisted of the following subtasks:

• deterministic analysis of the consequences of fire events in order to identify damages to the safe shutdown components
• preliminary deterministic screening of the fire events by evaluating the consequences
• probabilistic modelling of the fire events not screened out deterministically
• probabilistic screening of the fire events
• additional deterministic analyses to refine fire events, i.e. to reduce conservatism involved by (1) taking into account possible corrective actions, (2) further subdivision of fire compartments into fire cells
• probabilistic modelling of the refined fire events, identification of dominant ones.

During the deterministic analysis of the consequences of fires it was assumed that all the components and cables were damaged due to the fire in a fire compartment. In addition, all failure modes were considered with regards to the cable damages. This resulted sometimes in contradictory failure modes of the
components “at the end” of the cables. During the evaluation of the consequences the most conservative situation was intended to be defined, therefore from the above contradictory failure modes the ones were chosen that brought the situation to the worst state. Based on the list of failure modes provided by the integrated data base first the fire induced initiating event was identified. It should be noted, that in a number of cases only a very complex initiating event could be identified. This was followed by the assessment of the mitigation process taking into account the available mitigating systems.

From the large number of fire events the ones were screened out that:

- neither caused an initiating events nor required reactor shutdown in 8 hours due to the regulations in the Technical Specifications
- did not lead to the simultaneous occurrence of an initiating event and degradation of any of the systems required to mitigate the given initiating event (these cases are included in the earlier level 1 PSA studies).

As a result of the deterministic screening 162 fire events remained screened in.

The next step of the analyses was the probabilistic modelling of the screened in fire events. For the cases when particular fire induced initiating events could be identified specific event trees have been built up. For the cases of more complex initiating events a generic event tree has been developed integrating the potential initiating events as well as the necessary mitigating systems. Degradation or failure of the mitigating systems has been modelled on fault tree level by boundary conditions, i.e. house events. For the estimation of the fire frequencies ignition sources, basic frequencies based on the generic nuclear field experience and the appropriate weighting factors have been considered taking into account proposals of the FIVE (EPRI, 1992) method. Reliability data for random equipment failures and probabilities of pre-initiator human errors were taken from the earlier level 1 PSA studies. Post-initiator human actions were modelled in the scope as they were present in the existing level 1 PSA models. No corrective actions were taken into account at this stage of the analyses. By performing the above tasks core damage frequencies have been calculated for all the 162 screened in fire events.

Probabilistic screening of the fire events was performed on the basis of their contribution to the core damage risk. The screening value of the core damage frequency was chosen in such a way that the total core damage frequency originating from the fire events with core damage frequencies below the screening value would be less than 1% of the core damage frequency identified by the earlier level 1 PSA studies on internal initiators occurring during nominal power operational mode. As a result of the probabilistic screening 44 fire events remained screened in.

Additional deterministic analyses were aimed at the refinement of the consequences of the fire events by a review of the conservative assumptions. This was done by walk-throughs in the 44 fire compartments together with members of the Operational Fire Department of the plant. During these walk-throughs information about the fire spread, fire characteristics and fire frequencies of the fire compartments have been reviewed and the possibilities of subdividing the fire compartments into fire cells have been examined. Expected fire spread has also been estimated based on the times necessary for detection, suppression initiation and suppression. As a result of the walk-throughs 34 of the 44 fire compartments were further subdivided into 222 fire cells. (For the purpose of the fire PSA for NPP Paks a fire cell is defined as a space within a fire compartment in which a fire is contained with high probability due to existing non-combustible barriers, measures and means against fire propagation, safety distance, fire detection and suppression capabilities.) Fire frequencies were then re-assessed for the fire cells, and also the event propagation was re-evaluated for each case. In addition, possible corrective actions were identified and the possibilities and effectiveness of the fire suppression were also examined. A simplified
quantification approach has been developed to estimate the probability of non-success of the fire suppression taking into account fire barriers, existence of fire suppression system trains, and existence and redundancy of the fire signalization loops. Results of fire simulation performed by the COMPBRN IIIe code in the framework of a fire hazard analysis (separate from the fire PSA) have also been utilised in this re-evaluation process (OLAJTERV, 1998).

Due to the fact that the fire events had been refined the process of deterministic and probabilistic analyses had to be partly repeated for the newly identified fire cells: the list of failure modes related to the fire cells was generated, event propagation was assessed, initiating event (fire) frequencies were generated for the fire cells and the recovery possibilities were built into the probabilistic model. As a result of the above tasks dominant fire events have been identified.

3.3 Results and Evaluation

The core damage frequency originating from the fires occurring during the nominal power operational mode taking into account the actual state (hardware systems, operational characteristics) of unit 1 is 2.23E-4/year. Contribution of the fires in different types of rooms to the core damage risk is shown in Figure 2, where A - primary circuit rooms, E - secondary circuit rooms and M - turbine hall. The list of dominant rooms together with their contribution is given in Table 1.

![Figure 2. Contribution of the main room categories to the total core damage frequency](image)

Based on the results it can be stated that the core damage risk of the fires occurring on the 15 m level of the turbine hall (room E412/I belongs to the secondary circuit rooms) is the highest. Fires in the turbine hall (including upward cable trays on the wall of the turbine hall) and in the main control room have still high contribution. Core damage frequency originating from the fires in the above three rooms constitutes more than 90% of the total core damage frequency originating from fires. It should be noted that most of the dominant rooms were subdivided into fire cells, so the above values are aggregate ones, gained by summing up core damage frequencies of the particular fire cells of the given room.

Due to the characteristics of the fire PSA study the results of the conventional importance and sensitivity analyses could not be used for the identification of possible safety enhancement measures (the most important factors are the unavailable components that are not part of the model for which an e.g. importance analysis is made). Thus, they were identified by evaluating dominant fire events one by one. The proposed safety enhancement measures were as follows.

- In a major part of the dominant fire events steam generator protections operate spuriously, and isolate all the steam generators terminating in such a way all feedwater supply (thus decay heat
removal). Present EOPs cover only a limited scope of the fire events analysed, and they do not include procedure for the detection and interaction in the cases when all steam generators are isolated simultaneously. EOPs therefore should be supplemented by these cases. (Development of new symptom based EOPs is presently underway, that is expected to solve the problem.)

- Total loss of room E412/I (due to fire spread between fire cells) causes loss of secondary side decay heat removal. Installation of fire detection instrumentation in the room reduces the potential for loosing the whole room.

<table>
<thead>
<tr>
<th>Room</th>
<th>Description</th>
<th>CDF 1/year</th>
<th>Contribution %</th>
</tr>
</thead>
<tbody>
<tr>
<td>E412/I</td>
<td>Piping corridor of feedwater and main steam systems</td>
<td>1.06E-4</td>
<td>47.55</td>
</tr>
<tr>
<td>M01/I</td>
<td>Turbine hall, underground and hall level</td>
<td>6.74E-5</td>
<td>30.23</td>
</tr>
<tr>
<td>E306/I</td>
<td>Main control room</td>
<td>3.12E-5</td>
<td>14.01</td>
</tr>
<tr>
<td>E310/I</td>
<td>Relay room</td>
<td>5.89E-6</td>
<td>2.64</td>
</tr>
<tr>
<td>E212/I</td>
<td>Cable room</td>
<td>1.88E-6</td>
<td>0.84</td>
</tr>
</tbody>
</table>

Table 1. Dominant rooms and their contribution

- Fire in any fire cell of the room E412/I causes a spurious operation of all steam generator protections, the steam generators are isolated simultaneously. This is due to the fact that cables of these protections to all steam generators run on the same cable tray. Thus, they should be made diverse by location. The same is valid for the steam generator protection cables in cable room E310/I and in the main control room (E306/I).
- Fires in the upward cable trays in the turbine hall spread very quickly to the upper located room E412/I and cause the same problem as a fire in any fire cells of that room. Fire barriers should be installed in these cable trays to prevent upward spread of fire.
- Enhancing fire protection of the emergency feedwater pumps decreases the risk of fires in the turbine hall leading to the loss of both normal and emergency feedwater pumps.
- Cabling of the normal, emergency and auxiliary emergency feedwater pumps in the main control room are located behind neighbouring panels. Due to the high probability of fire spread from one panel to the other all the means of feedwater supply (possibility of decay heat removal) can be lost due to a fire on these panels. Thus, they should be better separated.

It has been estimated that by implementing the above measures the core damage frequency could be reduced by an order of magnitude.

4 CONCLUSIONS

After initial deterministic fire safety analysis performed within the AGNES project a fire PSA study has been carried out for unit 1 of the Paks NPP with the objective of extending earlier level 1 PSA studies with the evaluation of this internal hazard. The study has been carried out in two major phases, deterministic analyses in the first and probabilistic in the second. The most important characteristic of the study was the support of probabilistic modelling by extensive deterministic analyses. After the analysis of fire events a two stage (deterministic and probabilistic) screening process has been applied in order to reduce the large number of preliminary fire events. Screened in fire events underwent a more detailed review that reduced the degree of conservatism involved. Finally, core damage frequency originating from fires occurring during the nominal power operational mode has been identified. It constituted a value of 2.23E-4/year.
Possibilities of risk reduction have been examined and proposals have been prepared for safety enhancement measures. By implementing the proposed safety enhancement measures the core damage risk can be reduced by an order of magnitude. To gain further insights into the characteristics of dominant fire initiated core damage sequences and re-assess some fire scenarios (if needed) fire PSA follow-up analyses have been started before implementing safety enhancement measures at the plant.

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FIRE RISK ANALYSIS FOR LOVIISA I TURBINE HALL

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ABSTRACT

Fortum Power and Heat Oy operates two VVER-440 type nuclear units, Loviisa 1 and 2 (PWR, 488 MWe). The plants are unique combinations of Western and Eastern technology being in commercial operation since 1977 and 1981, respectively.

Level 1 probabilistic safety analysis (PSA) for Loviisa 1 has been completed for internal initiating events (IEs) and external IEs at full power as well as for internal IEs at low power and shutdown modes. Fire-PSA is being merged with the living PSA models to facilitate effective updating and use of PSA as a comprehensive safety management tool.

Fire-PSA was carried out to identify the fire risks, rank them, identify possible weaknesses of the plant and assess potential plant improvements, when necessary. The fire risk consists of the fire frequencies, fire-induced IEs and fire-induced safety system failures combined with internal unavailabilities of safety systems.

Most of the dominating fire risks arise from the turbine building because of feedwater supply failure with high fire frequency. Additionally, the turbine oil fires have potential for failing components and cables located in the rooms next to the turbine hall. Thereafter, loss-of-coolant-accident (LOCA) through primary coolant pump (PCP) seal failures could occur, too. The present core damage frequency due to fires in the Loviisa 1 turbine hall is 9,1E-06 1/a.

Many plant modifications have been done considering the turbine hall fire safety since the beginning of the commercial operation of the units. Currently, improvement design of the residual heat removal system is underway to assure the cold-shutdown cooling reliability.

1 INTRODUCTION

Level 1 PSA for Loviisa 1 has been completed for internal IEs in 1989 as well as for floods (1994), severe weather (1994) and seismic events (1993) considering full-power operation mode. Fire-PSA was completed for the full-power operating mode in 1997. This paper will focus on 1) the methodology and the main tasks of fire-PSA for turbine hall, 2) the highlights on the major contributors to the turbine hall fire risk, and 3) the modifications completed at Loviisa due to fires.

The fire risk analysis includes identification of the fire-induced IEs, estimation of different fire frequencies and estimation of the conditional core damage (CD) probabilities for the fire events. Fire-PSA was carried out in two main phases: (1) screening phase to identify the possibly dominating fire events, and (2) detailed analysis phase. During the second phase the aim was to determine fire-induced damages realistically, and to perform plant improvement need assessment to decrease the dominating fire risks, when considered necessary.

The fire frequencies were estimated with an empirical Bayes method using fire data of nuclear power plants in the United States (US) combined with the Loviisa experience. Additionally, turbine blade failure frequency was estimated using data of turbine failures of nuclear power plants all over the world.

The conditional CD probabilities were determined with the PSA model of internal IEs for individual fire events by taking into account the IEs and possible fire-induced safety system failures combined with internal unavailabilities of safety systems.

The final object is to handle fire-PSA as a part of the "living" PSA models to facilitate updating of the analysis and to be able to use PSA as a comprehensive safety management tool.
2 DESCRIPTION OF THE TURBINE HALL

2.1 Lay-out

The turbine hall is common for Loviisa 1 and 2, but there is a fire wall separating the units below the turbine floor. The turbine hall is about 180 meters in length, 42 meters in width and 34 meters in height. Four turbine-generator combinations are located in the hall.

The following areas are next to the turbine hall:

- control building
- sea water pump station
- demineralization plant
- ventilation control room
- safety related cable tunnels leading through the turbine hall basement

A concrete wall of thickness 100-250 mm separates these rooms from the turbine hall. The following safety-related systems are in the hall:

- five main feedwater pumps and supporting systems
- the emergency feedwater pumps (2 × 100%) and supporting systems; the pumps are located in their own fire zones with the residual heat removal pumps, in the middle of the turbine hall
- feedwater pipelines and many motor-operated valves
- cabling for the above mentioned equipment
- instrument air pipelines and inactive nitrogen pipelines
- service water pipeline needed for cooling of the instrument cabinet rooms (2 redundancies exist)
- the busbars supplying house load switchgears

Next to the turbine hall, above the control room building, are two main feedwater tanks, control valves of the main and emergency feedwater systems, steam generator safety valves, the main steam line isolation valves, the steam relief valves and the turbine bypass valves. This area was originally open to the turbine hall, but it has been later separated from it by a fire wall.

2.2 Fire conditions

Loviisa 1 turbine hall includes several fire loads like two turbine-generator sets, both containing about 56 m³ of oil, the hydrogen used for generator cooling (about 2 * 200 nominal m³), the lubricating oil of the main feedwater pump motors (about 5 * 1 m³), and some plastic cables and electrical equipment.

Fixed fire-suppression systems are located all over the turbine hall and the areas containing oil systems are covered with additional suppression systems. A standby fire pump station with diesel-driven pumps has been built to back up the original electrical pumps. Automatic fire vents are located in the ceiling and windows can be opened by means of hydraulic actuators, too.

3 METHODOLOGY

Fire-PSA was carried out in two phases using a systematic and comprehensive approach to find out the dominating fire risks.¹

3.1 Phase 1

During phase one, the main purposes were:

- to estimate fire frequencies
- to identify possible fire-induced IEs
- to identify possible safety system failures caused by fires
- to find out all fire events possibly affecting plant safety
- to use conservative assumptions

The turbine hall fire events were classified into four categories:

- oil and/or hydrogen fires in the generator area (type G)
- oil fires in the turbine area (type B)
- limited fires in other areas (type T)
- fires arising from turbine blade failures including turbine missiles and other external consequences, also (type M); this fire type was assumed to include very large oil fires, where the turbine oil might burn completely

The fire risk arising from possible fire spreading events outside the ignition area was included in the ignition area’s fire risk. Possible fire spreading routes were identified during plant walk-through according to expert opinion.
Generally, fire spreading outside the turbine hall was assumed to occur via an open fire door. The probability for a fire door being open is 0.07, which was determined during the plant walk-through.

Long-term turbine oil fires could induce temperature rise in the adjacent rooms, which contain, for example, PVC-cables and electric equipment. Thus, the fire risk from the overheating of these rooms without any actual fire spreading event was taken into account if the automatic fire extinguishing systems were unavailable or insufficient: very large fire areas may exceed the design basis water flow rate of the fire-suppression system.

3.2 Phase II

The dominating fire events identified in phase one were analyzed in more detail during the second phase. Mainly, the following recovery actions were taken into account to estimate the fire risk more realistically:

- manual start-up of the diesel-driven fire pumps, if the electric-driven pumps are unavailable; this will decrease the fire spreading probability outside the ignition area
- manual start-up and control of the primary make-up water pumps used for steam generator feedwater supply via the steam generator blowdown lines, if the additional emergency feedwater system is unavailable

3.3 Heat Transfer Code IVOHEAT

Program IVOHEAT is a special heat transfer code developed by Imatran Voima Oy (at present Fortum Engineering Ltd). The code carries out thermal conductivity analyses by a Finite Element Method. It has been used for heat transfer analyses through concrete structures since 1980's.

During a fire, the time delay will be about 1-2 hours until the temperature of the opposite surface of a structure reaches a critical value considering allowed conditions for electric equipment and cables.

During the Loviisa fire-PSA, IVOHEAT was applied to estimate fire-induced temperature rise and failure time of PVC-cables inside different fire protection covers. The input data needed includes properties of the cover structure, surface temperature as a function of time and the properties of the protected cables. The temperature values allowed for emergency operation of the PVC-cables were used as failure criteria. Based on these calculations, the cables maintain operability about 10-20 minutes, if the fire is not put out.

3.4 Main Fire Scenarios

3.4.1 Fire type G

The fires ignited in the generator area could spread for a large area, if the suppression systems were unavailable:

- regardless of fire suppression, all electric components and cables inside the turbine hall were assumed to be damaged because of large oil fires
- at the most 3/4 cable tunnels could be jeopardized because of heat transfer through the walls
- the fire could affect also the sea water pump station or cable tunnels if the corresponding fire door is open and the automatic suppression systems are unavailable
- 12 fire scenarios have been determined for type G fires

3.4.2 Fire type B

The fires ignited in the turbine area were handled like the type G fires. However, certain fires may affect also the control building if the corresponding fire door is open and automatic suppression systems are unavailable. 15 fire scenarios have been determined for type B fires.

3.4.3 Fire type T

This type contains small oil fires of the different pumps (T1) and component or cable fires (T2). These fires could not affect rooms next to the turbine hall. All electric components and cables inside the turbine hall were assumed damaged if the automatic suppression systems were unavailable. Otherwise, the possibly failed components and cables were identified in the vicinity of certain ignition area with rough estimates on the possible damages caused by fire before extinguishment (these estimates include possible failures caused by fire water, too). 14 fire scenarios have been determined for type T fires.

3.4.4 Fire type M

The fires arising from turbine blade failures could be very large or long-term. However, lack of reliable method to estimate spreading and burning of large amount of oil was noticed. Therefore the fire events were classified roughly into three different cases according to the fire area and burning time to reach contribution for the different heat transfer possibilities through the concrete walls and to find out the effect of the suppression systems as a function of the fire area. Additionally, the fourth case contained events with other external consequences of blade failures (without fires).
Fire type M1: "small fire area, long-term oil fire"
- the fire area is less than 300 m², the suppression systems can put out the fire or have it under control preventing fire spreading outside the turbine hall
- regardless of fire suppression, all components and cables inside the turbine hall were assumed to be damaged
- all rooms next to the turbine hall could be jeopardized because of heat transfer through the walls if the suppression systems are unavailable
- 14 fire scenarios have been determined for type M1 fires

Fire type M2: "medium fire area, short-term oil fire"
- the fire area exceeds 300 m² and the suppression systems can not control these kinds of fires definitely
- main part of the oil can be burned out during one hour
- regardless of fire suppression, all components and cables inside the turbine hall were assumed to be damaged
- only the cable tunnels going through the turbine hall could be jeopardized because of heat transfer through the thin walls
- 8 fire scenarios have been determined for type M2 fires

Fire type M3: "large fire area, short-term oil fire"
- the fire area exceeds several hundreds square meters and the suppression systems can not control these kinds of fires definitely
- because of the very large fire area, most of the oil can be burned out during less than 20 minutes
- regardless of fire suppression, all components and cables inside the turbine hall were assumed to be damaged
- the rooms next to the turbine hall can not be jeopardized by heat transfer through the walls
- 13 fire scenarios have been determined for type M3 fires

Type M4: “other external consequences”
- this type contains other risk sequences than fires, for example damages induced by turbine missiles or flooding
- all components and cables inside the turbine hall were assumed to be damaged
- the rooms next to the turbine hall can be jeopardized only because of flooding
- 2 scenarios have been determined for type M4 events

4. FIRE-INDUCED IEs

The IE s under consideration consist of the following items and fire-induced failures:
- Loss of DC Power (LDCP): possible because of fires affecting both redundancies of the control building
- Loss of Instrument Cabinet Room Ventilation (LIRV): possible because of fires inducing inadvertent control signals after cable failures in the control building or because of fires affecting ventilation control room
- Loss of Main Feedwater (LMFW): possible because of fire damages inside the turbine hall
- Loss Of Off-site Power (LOOP): possible because of fires inducing inadvertent control signals after cable failures in the cable tunnels or in the control building, or because of loss of the busbars leading through the turbine hall and supplying the house load switchgears
- Small Loss Of Coolant Accident (SLOCA): possible because of fires inducing PCP seal failures after cable failures in the cable tunnels or in the control building or because of fires affecting the sea water pump station
- Total Loss of Feedwater (TLFW): possible because of fire damages inside the turbine hall (total loss of main feedwater and emergency feedwater)
- Total Loss of Service Water System (TLSW): possible because of fires affecting cable tunnels, control building or the sea water pump station
5  FIRE FREQUENCY ESTIMATION

5.1  Data

Fire frequencies were estimated using fire data from nuclear power plants in the US combined with Loviisa experience. The US data mainly consists of NESC Firedata through 1985, EPRI Fire Data through 1988, plant operation data from IAEA through 1994 and articles in Nucleonics Week since 1986. The gathered data included about 750 fire events in light water reactor plants (LWR) after the beginning of commercial operation.

The gathered data may not be comprehensive since 1989. The IAEA data contains those incidents leading to long-term shutdown or degradation of power supply capacity, thus minor fires may not have been reported. Therefore identified fire events within 1989-1994 were applied only to fires in the generator area.

Turbine blade failure initiated events were studied separately, because of the possibility of a very large turbine oil fire arising from missile damages, vibration-induced pipeline breaks or failures in the generator sealing oil system. Turbine blade failure events were identified from the IAEA plant operation data through 1995, mainly.

5.2  Relevant Events

Fire frequencies were estimated for the power operation mode, only. Fires being ignited by fire hazard works and extinguished before any component failure occurred were ignored as insignificant events.

Electric component fires inside the generator were ignored, because the aim was to estimate oil and/or hydrogen fires in the generator area. Thus, the possible risk arising from electric component fires was determined insignificant.

Considering turbine blade failures, those water cooled reactor plants were studied, which have electrical output more than 100 MW and have shutdown information from the beginning of the commercial operation. Turbine disc failures were excluded because such failures are not applicable to the Loviisa turbines with welded low pressure rotors and forged high pressure rotors.

As a result, 53 relevant fire events were used in estimating the fire frequencies, and 51 turbine blade failure events were used in estimating the turbine blade failure frequency.

5.3  The Frequency Estimation

The fire frequencies were estimated with an empirical Bayes method. During estimation, the US plants and Loviisa were assumed similar but each still having individual fire frequencies. The method determines first the prior distribution from the input data and then individual plant-specific posterior distributions with the mean value, standard deviation and 5%, 95% fractiles.

Turbine blade failure frequency was estimated correspondingly.

The operation time of each plant was calculated from the beginning of the commercial operation, as reactor years. Plant-specific numbers of turbines and generators were multiplied with reactor years to achieve the corresponding component years.

Some US plants have had long-term shutdowns because of regulatory reasons or large modification works. These shutdown years were ignored in the operation times during estimating frequencies for the fire types B, G and T.

The use of US fire data for determining the prior distribution may be misleading, because of the possible differences considering ignition sources, for example electrical equipment and cable material. Therefore, the estimated fire frequencies may not be exact. However, the fire frequencies for Loviisa plant can not be compared with the fires of other VVER-type units; because of Western technology being used in electrical systems. Also, any potential bias was reduced by keeping Loviisa plant as a member of the group when determining the prior distribution.

The plant name was not known for each fire event in the fire data. Therefore, the unknown fire events have been assumed to plants having long operation times without reported fires, considering each fire type. This kind of an assumption assured achieving conservative posterior distributions for Loviisa.

Considering turbine blade failures, only 5/51 events had caused failures outside the turbine. Furthermore, there had been reported fire in 4/5 events. Thus, there were achieved conditional probability 7.8E-02 to occur fire and 2.0E-02 to occur other external damages without fire (type M4) after turbine blade failure. The fire frequency after turbine blade failure was then allocated equally for three parts to achieve the fire frequencies for the different oil fire cases (fire types M1, M2 and M3).

Considering type T fires, the frequency was allocated for the turbine hall and for the steam generator safety valve room. About half of the fire events were reported as oil fires, which refer to Loviisa turbine hall, only. Other fire events refer to the safety valve room, also. Therefore, the safety valve room fire frequency was determined to be 1/6 of the respective fire frequency of the turbine hall, based on the amount of equipment and floor area. As a summary, the fire frequency of the turbine hall is 11/12 of the type T frequency.

The number of relevant fire events and the total observation times of different fire and blade failure types in the generic fire data base are presented in Table 1. The last column indicates the estimated plant-specific frequencies for the Loviisa plant.
Table 1. Frequencies for different fire types and turbine blade failures (type M) in Loviisa turbine hall.

<table>
<thead>
<tr>
<th>Type</th>
<th>Events</th>
<th>Oper. time (a)</th>
<th>Frequency (1/a)</th>
</tr>
</thead>
<tbody>
<tr>
<td>G</td>
<td>10</td>
<td>1844</td>
<td>3.0E-03</td>
</tr>
<tr>
<td>B</td>
<td>11</td>
<td>1224</td>
<td>2.0E-03</td>
</tr>
<tr>
<td>T</td>
<td>32</td>
<td>1190</td>
<td>4.2E-02</td>
</tr>
<tr>
<td>T1</td>
<td>..</td>
<td>....</td>
<td>2.1E-02</td>
</tr>
<tr>
<td>T2</td>
<td>..</td>
<td>....</td>
<td>1.7E-02</td>
</tr>
<tr>
<td>M</td>
<td>51</td>
<td>5014</td>
<td>3.2E-03</td>
</tr>
<tr>
<td>M1</td>
<td>..</td>
<td>....</td>
<td>8.5E-05</td>
</tr>
<tr>
<td>M2</td>
<td>..</td>
<td>....</td>
<td>8.5E-05</td>
</tr>
<tr>
<td>M3</td>
<td>..</td>
<td>....</td>
<td>8.5E-05</td>
</tr>
<tr>
<td>M4</td>
<td>..</td>
<td>....</td>
<td>6.4E-05</td>
</tr>
</tbody>
</table>

6  TURBINE HALL FIRE RISK

6.1 Risk Quantification

Fire event-specific fire risk (RF) was calculated according to the following equation (see notation at the end of the paper):

\[ RF = F_E P_{CDE} \]  \hspace{1cm} (1)

Conditional CD probability was estimated with program CAFTA, when the fire-induced IEs and possible simultaneous safety system failures due to the fire were determined and input to the code.

Fire spreading outside the ignition area increases the fire-induced risk, if the P_{CDE} for the spreading event is increased compared to the corresponding value for the ignition area. This is possible only if new fire-induced safety system failures occur, or a new IE occurs due to fire spreading. Equation (2) was used to calculate the fire risk increase (dRF) for such fire spreading events:

\[ dRF = F_E S_F (P_{CD2} - P_{CD1}) \]  \hspace{1cm} (2)

6.2 Main Results

The IE-specific fire risk values for each fire type are presented in Table 2.

The largest contribution of the turbine hall fire risk is due to the fires inducing feedwater supply failure (loss of the main and the auxiliary feedwater supplies) with a high fire frequency. During these events the operator has to start the additional emergency feedwater system or use the make-up water system for feedwater supply.

13% of the total fire risk comes from the fire spreading events via an open fire door. The probability for a fire door being open (0.07) was determined during the plant walk-through several years ago. Nowadays that value should be lower because of fire door control improvements.

12% of the total fire risk is due to heat transfer through the fire barriers. Major part of this risk comes from the type M2 fires heating up the cable tunnels crossing through the turbine hall basement. These fires were not assumed to be controlled by the suppression systems at all because of the large fire area exceeding the design basis of the suppression systems. However, the inner surfaces of these cable tunnels could be cooled with their own sprinkler systems, which has not been taken into account in the above risk value of type M2. Thus, there is some potential for risk reduction when the study is updated in the future.

The result is conservative considering type T fires, because the fire data being used included one small fire in Loviisa, although the fire did not cause IE or affect safety systems.

6.3 Further Activities

The fire risk models and data are being merged with the "living" PSA models, currently including internal and flood IEs, to facilitate effective updating and use of PSA as a comprehensive safety management tool. However, the first effort is modeling of fire-PSA according to different sequences and corresponding summary fire frequencies without describing individual rooms in the PSA model. Therefore a separate calculation method will be needed for the evaluation of room-specific fire risks and to determine the summary fire frequencies of the sequences needed for the "living" PSA model.

Table 2. The IE-specific fire risk values (1/a) for Loviisa 1 turbine hall according to the fire types.

<table>
<thead>
<tr>
<th>Type</th>
<th>Freq.</th>
<th>LDCP</th>
<th>LIRV</th>
<th>LMFW</th>
<th>LOOP</th>
<th>SLOCA</th>
<th>TLFW</th>
<th>TLSW</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>G</td>
<td>3.0E-03</td>
<td></td>
<td></td>
<td></td>
<td>3.5E-08</td>
<td>1.7E-09</td>
<td>1.1E-06</td>
<td>1.5E-07</td>
<td>1.3E-06</td>
</tr>
<tr>
<td>B</td>
<td>2.0E-03</td>
<td>4.4E-09</td>
<td></td>
<td></td>
<td>2.3E-08</td>
<td>6.7E-08</td>
<td>1.4E-06</td>
<td>9.8E-08</td>
<td>1.6E-06</td>
</tr>
<tr>
<td>T1</td>
<td>2.1E-02</td>
<td></td>
<td>3.8E-07</td>
<td></td>
<td></td>
<td></td>
<td>2.6E-06</td>
<td></td>
<td>3.0E-06</td>
</tr>
<tr>
<td>T2</td>
<td>1.7E-02</td>
<td></td>
<td>2.7E-07</td>
<td></td>
<td></td>
<td></td>
<td>3.5E-07</td>
<td></td>
<td>6.1E-07</td>
</tr>
<tr>
<td>M1</td>
<td>8.5E-05</td>
<td>3.7E-08</td>
<td>3.0E-09</td>
<td></td>
<td>3.7E-07</td>
<td>3.1E-08</td>
<td>7.2E-09</td>
<td>4.5E-07</td>
<td></td>
</tr>
<tr>
<td>M2</td>
<td>8.5E-05</td>
<td>9.0E-09</td>
<td></td>
<td>7.7E-07</td>
<td>3.8E-07</td>
<td>1.7E-07</td>
<td>1.3E-06</td>
<td></td>
<td></td>
</tr>
<tr>
<td>M3</td>
<td>8.5E-05</td>
<td>9.0E-09</td>
<td></td>
<td>6.2E-07</td>
<td>6.2E-09</td>
<td>9.9E-08</td>
<td>7.3E-07</td>
<td></td>
<td></td>
</tr>
<tr>
<td>M4</td>
<td>6.4E-05</td>
<td></td>
<td></td>
<td></td>
<td>6.4E-08</td>
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<td>6.4E-08</td>
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<tr>
<td>total</td>
<td>4.3E-02</td>
<td>3.7E-08</td>
<td>2.5E-08</td>
<td>6.4E-07</td>
<td>1.8E-06</td>
<td>1.0E-07</td>
<td>5.9E-06</td>
<td>5.2E-07</td>
<td>9.1E-06</td>
</tr>
</tbody>
</table>

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PLANT MODIFICATIONS COMPLETED

Many plant modifications have been done in Loviisa over the years considering especially turbine building fires. For example:

- control building steel structures have been covered
- a fire wall has been built to separate the turbine hall and the steam generator safety valve room
- some fire doors have been closed permanently
- suppression system actuators for the cable tunnels have been moved outside the turbine hall
- an additional emergency feedwater supply system has been constructed outside the turbine hall
- an additional fire pump house with diesel-driven pumps has been constructed

At the present, an auxiliary residual heat-removal system is in the planning stage to improve the reliability of long-term cooling in cold-shutdown states.

REFERENCES


5. The Power Reactor Information System (PRIS), International Atomic Energy Agency (IAEA), MicroPRIS package.


Session III
Fire Risk Assessment and Application II

- Probabilistic Risk Assessment of Fire Safety Design Alternatives - Ms. Lotta Andersson, Sycon Energikonsult AB, Sweden

- Preliminary Study of Fire Event PSA for BWR Plants - Mr. Toshiyuki Zama, Tokyo Electric Power Company (TEPCO), Japan and Mr. Kazunori Hashimoto, (Toshiba Corporation), Japan

- Simulating of Frequency Firing's on Turbo-generators at Ukrainian NPP - Mrs. Olena Babich and Mr. Sergei Azarov Ukrainian Academy of Sciences, Ukraine
Probabilistic risk assessment of fire safety design alternatives

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Abstract
To be able to compare different fire safety design alternatives or trade-offs a quantification of the risk level is needed. Making a probabilistic risk assessment of each alternative can do this. In this paper a probabilistic risk assessment method is presented and a case study is done at a nuclear power plant. Two alternative fire safety designs presented in prescriptive regulations are compared using event-tree analysis and uncertainty analysis. The result shows that the two alternatives do not generate an equal safety level.

Introduction
Performance-based fire safety regulations are getting more common. This is generating a need for risk based assessment methods and for acceptance criteria. The objective of this paper is to present a probabilistic risk assessment method and to quantify the safety difference between two alternative fire safety designs presented in a prescriptive regulation. The paper is based upon a report published by the Department of Fire Safety Engineering, Lund University (Andersson, 1999).

Background
Nuclear power plants operating prior to 1979 was not built according to the fire safety regulations in practice today. This has lead to that a lot of upgrading work has to be done in these stations regarding fire safety. To improve the fire safety in an already operating nuclear power plant is difficult because the possibilities are very limited. This is probably the reason to why there are three alternative solutions, of how to separate redundant safety equipment, presented in the “Fire Protection Programme for Nuclear Power Facilities Operating Prior to January 1, 1979” (10 CFR 50 Appendix R, 1987). These three solutions have been used as equal fire safety alternatives for a long time without establishing whether or not they generate an equal safety level.

The transition to more performance-based building regulations has increased the uncertainty as to whether the building is safe or not. According to an evaluation performed by the Swedish Board of Housing, Building and Planning (Boverket, 1997), the uncertainty level does not so much depend on the regulations but one big problem seemed to be that the methods for design had not been completely adopted by the consultants. The reason for that is probably the uncertainty of what is acceptable and how it should be proved.

The three fire safety alternatives presented in Appendix R (10 CFR 50 Appendix R, 1987) are meant to be used as a base for new and more performance-based regulations regarding fire safety at nuclear power plants. These three alternatives will undoubtedly lead to a lot of questions about what the acceptable safety level really is. Is it the safety level generated by this or that alternative, or is it somewhere in between? And how can it be proved that another alternative generates an equal safety level?
Objectives
The main objective of this paper is to present a method on how to quantify safety differences between different fire safety designs and trade-offs, taking both active and passive fire protection features under consideration. The method is applied on a nuclear power plant in Sweden. The result from this application is then used to discuss the fire safety regulations for nuclear power plants of today.

Trade-offs and acceptable safety levels
A trade-off is an alternative solution meant to replace a solution presented in prescriptive regulations. A trade-off, or alternative design, must result in an equal safety level as the original solution or, if established, in an acceptable safety level.

Acceptable safety levels are very seldom established, but, if risk analyses are carried out on buildings that fulfil the regulations, the result can be used to quantify acceptable risk criteria.

The British Standard Institution outlines in their Draft for development No. 240 (BSI, 1997) a framework for an engineering approach to fire safety design in buildings. In this guideline three types of approach are considered of how to decide whether a fire safety alternative is acceptable or not. These three types are:
1. deterministic
2. probabilistic
3. comparative

In a deterministic study the objective is to show that, on the bases of the initial (usually 'reasonable worst case') assumptions, a defined set of conditions will not occur. In a probabilistic study, criteria are set such that criteria are set such that the probability of a certain event is acceptably low. The risk criteria are usually expressed in terms of the annual probability of the unwanted event. In a comparative study the objective is to show that the design provides a level of safety equivalent to that in a building which conforms to more prescriptive codes. But, before it can be demonstrated that a solution offers at least the same level of safety as a prescriptive code, the intent of that code have to be very clear.

In Appendix R (10 CFR 50 Appendix R, 1987) three different alternatives are presented of how to separate cables and equipment and associated non-safety circuits of redundant trains. These three alternatives are:

1. Separation by a fire barrier having a 3-hour rating.
2. Separation by a horizontal distance of 20 feet (6,2 m) with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.
3. Enclosure of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.

Before the fire safety regulations for nuclear power plants are becoming performance-based it is very important to either define an acceptable safety level or at least define which one of the three different design alternatives that are acceptable in what situation. Maybe it is acceptable to have a safety level that is somewhere in between the safety level generated by the first design alternative and the safety level generated by the second or third design alternative. But if this is the case it ought to be clearly presented in the regulations. If this is not done it will be very difficult to use any other alternative design then the three already presented in regulation.
Risk assessment method

General
The base of the risk assessment method presented in this paper is a combination of event tree analysis and uncertainty analysis. The method is meant to be used in all kinds of fire safety design processes, both at nuclear power plants and in ordinary building projects.

The method is built up by following 6 main steps.
1. Identification of problem and objectives
2. Identification of possible fire scenarios
3. Creating a limit state function
4. Identification of fire safety design alternatives
5. Risk assessment of fire protection alternatives
6. Comparison of risk levels

Which one of the three approaches described in BSI (BSI, 1997) to be used is decided in step no. 1. This choice is then influencing the rest of the analysis in terms of how many scenarios shall be considered and if uncertainty analyses shall be done etc.

In the following chapter the method will be demonstrated at nuclear power plant problem.

Case study of a nuclear power plant

Problem and objectives
General objectives
The main objective of this analysis is to compare two different fire safety alternatives with each other and to quantify the safety difference between the two.

Fire safety objectives
The fire safety objective is to prevent redundant safety related equipment to fail.

Acceptance criteria
No acceptance criteria are established, but both analysed alternatives are presented as acceptable solution in Appendix R (10 CFR 50 Appendix R, 1987). The comparative approach according to BSI (BSI, 1997) is therefore used. While the safety differences shall be quantified the comparative analysis have to be probabilistic.

Assumptions and conditions
This risk assessment method starts when fire has already occurred, i.e. the probability of fire is not considered. The calculated risks are therefore relative and not absolute and in what way different fire safety alternatives are influencing the probability of fire is not quantified.
Object characterisation

The studied object is a single room in a nuclear power plant. The room is partly fictive but is representing a type of room that has a high contribute to probability of serious incidents according to a PSA or FRA or has had defective fire protection in comparison with Appendix R (10 CFR 50 Appendix R, 1987). The object is a pumproom and the only equipment in the room are two redundant pumps and their respective power and control cables.

The characteristics of the pumproom are as follows:
- Length: 8 m, width: 8 m, height: 4 m
- Permanent fire load: Crude oil
- Transient fire load: garbage bags
- Ceiling, wall and floor material: concrete
- Number of doors: 1
- Ventilation: 2.0 m$^3$/s
- Safety related equipment: pumps and cables
- Cables are insulated with PVC isolation

The pumps are separated with a distance of 20 feet (6.2 m). The safety related equipment in the room that is most sensitive to fire are the cables.

Fire scenarios

In this example the fire scenarios are chosen deterministically. The only permanent fire load in the room is the oil in the pumps. Probable transient fire load is garbage bags containing plastics and paper. Based on the permanent and transient fire loads in the room the following scenarios were analysed:
- Scenario 1: Oil fire
- Scenario 2: Garbage bag fire
- Scenario 3: Oil and garbage bag fire

Limit state function

The appearance of the limit state function is depending on the fire safety objective. In this case the limit state function are as follows:

$$G = K - D - A - S \geq 0$$

Where
- $K =$ Critical time, i.e. time to critical temperature for the safety related equipment
- $D =$ Detection time, i.e. time to detection
- $A =$ Activation time, i.e. time from detection (alarm) to initiated fire suppression
- $S =$ Suppression time, i.e. time from activation to extinguished fire
- $G =$ Security marginal, i.e. time marginal to component failure
Fire safety design alternatives
In the original report (Andersson, 1999) 17 different fire safety design alternatives were analysed. In this paper only two different alternatives will be discussed. These are:

- Alternative 1: Separation by a fire barrier.
- Alternative 2: Separation by a horizontal distance of 20 feet (6.2 m) with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.

Fire detection is generally installed in every fire compartment, which is why detection is considered also in alternative 1.

Risk assessment
Event trees
For each alternative an event tree is established, see figure 1 and 2. Every event in the tree is occurring with a certain probability and every final event is represented by a probability of component failure. The probabilities in figure 1 and 2 are mean values of the probability distributions used.

Figure 1  Event tree of alternative 1.
Figure 2  Event tree of alternative 2.

Probability of final events
Depending on failure frequencies of every event in the event tree the final events occur with a certain probability. While the failure frequencies can not be described by a single value these are described as probability distributions. The probability distributions are then used as input into the computer program @risk (@risk, 1997), and Monte Carlo simulations are used to produce a total probability distribution for each final event respectively.

Probability of component failure
Every final event in the event trees can be connected to a certain probability of component failure. This is done using the limit state function. The variables in the limit state function are stochastic and are calculated or decided according to the following description.

The critical time, $K$, are determined by simulating the fire scenarios with the computerised two-zone model CFAST (Peacock et al., 1984) and then using radiation and heat transportation equations programmed in @risk to decide the surface temperature of the cable isolation at different times. The critical time is the time when component failure occurs. Component failure are assumed to occur when the surface temperature of the isolation material are 200°C. The input to the radiation and heat transfer equations are defined as distributions. The result from the @risk simulation of surface temperature in terms of 5%-50%- and 95%-fractile are presented in figure 3. The results are then rearranged by creating a triangular distribution using the time values when 200°C is obtained for the different fractals. The triangular distribution can then be used as input to the limit state function.
Figure 3  Surface temperature as a function of time for different probability fractiles.

The detection times are calculated by using the computerised model Detact-T2 (Evans et al., 1985). The activation time of sprinkler systems is estimated by using reference literature. The activation times of manual suppression are calculated by dividing the action into different events according to the event tree in figure 4. The suppression time is estimated by using reference literature and depends for example of fire scenario.

Figure 4  Event tree of manual suppression.

With the inputs explained above the probability of component failure can be calculated. Using @risk this is done by calculating the safety marginal, G, a 1000 times and then calculate the probability that the safety marginal is less than zero, P(G<0).

Consequence
In this example the quantification of consequence is very simple: if G<0 the consequence is 1 and if G>0 the consequence is 0.
Risk calculation
The risk of component failure given a specific fire safety design alternative is calculated by the following equation:

\[
RISK = \sum_{i=1}^{n} \sum_{k=1}^{n} P(\text{scenario}_i) \cdot P(\text{final event}_k) \cdot (P(G < 0)) \cdot \text{Consequence}
\]

Risk comparison
The calculated mean risk values for the two alternatives are presented in figure 5.

![Figure 5](image)

**Figure 5**  
Mean risk values calculated for the two alternatives.

The risk levels can also be compared using risk profiles. Normally a risk profile describes the accumulated probability of a scenario against a quantified consequence. This can not be done in this case because the consequence can just be either 1 or 0 an alternative type of risk profile has therefore been established. The consequence is in this case alternative risk profile described as a probability, namely the probability of component failure P(G<0). In figure 6 risk profiles using mean input values are presented.

![Figure 6](image)

**Figure 6**  
Medium risk profiles for the two alternatives.
Uncertainty
In figure 7 and 8 the uncertainty of the respective risk profiles are presented. The three different lines represent minimum, mean and maximum risk profiles respectively.

![Figure 7](image)

Risk profiles resulting from the uncertainty analysis of alternative 1.

![Figure 8](image)

Risk profiles resulting from the uncertainty analysis of alternative 2.

Conclusion
The mean risk level of alternative 2 is 25 times the mean risk level of alternative 1. This means that the two alternatives do not generate equal safety level. Alternative 2 can, in this specific case, therefore not be seen as an acceptable trade-off for alternative 1.
Result and Discussion

When alternative designs or trade-offs are presented in a regulation it is easy to believe that these designs always generates a safety level that is acceptable. The result from the case study in this paper shows that separation of redundant safety equipment by a fire barrier generates a risk level that is 25 times less than separation by distance in addition with detection and sprinkler system. This makes it very clear that even if a trade-off is presented in a regulation there are no guarantees that this solution really will generate a safety level that is equal with the one generated with the original solution.

The result from the case study does not say that separation by distance in addition with detection and sprinkler system always are worse than separation by a fire barrier. What it does say is that the three alternatives do not generate equal safety levels in all individual cases, and that a comparative analysis has to be done to be sure that they do.

If the three different alternatives in Appendix R (10 CFR 50 Appendix R, 1987) are used as possible solutions also in a performance-based regulation it is necessary to define which of the alternative that is acceptable when and why. If this is not done it will be almost impossible to use any other fire safety design solution than the three alternatives already presented in the regulations. This means that some of the objectives with performance-based design will not be fulfilled, i.e. the objective to provide more flexibility and means of implementation for each individual building.

References


@risk, Guide to using @risk, Palisade Corporation, Newfield, 1997.


Preliminary Study of Fire Event PSA for BWR Plants

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1. Introduction

Probabilistic safety assessment (PSA) of internal event for the individual plant had been performed and reviewed in Japan. After these activities, we started to investigate the fire related features of Japanese BWR plants. As the first step of the fire event PSA study, we investigated fire event experience, plant layout configurations relating to fire in the reference BWR plants, and potential candidates of residual fire risk. We introduce these our recent activities and perspectives upon the fire event PSA.

2. Fire event experience of Japanese BWR plants

This section shows brief explanation about Japanese BWR plants operation and features of fire events in the operation experience.

2.1 Experience of Japanese BWR plants operation

At the present time, 28 BWR plants are being operated in Japan. 2 plants of them are GE type BWR2/3, 8 plants of them are BWR4, and 16 plants are BWR5. Furthermore, two ABWR plants have started their operation recently. Any plant has two or three safety divisions respectively, and principally systems belonging to one safety division, including electricity or equipment cooling, are designed to separate from systems of another safety divisions.

Since the first BWR plant had started it’s operation at 1970 in Japan, we have more than 300 cumulative years in operation experience until 1998. Features of fire events in our operation experience are described below.

2.2 Fire event experience

To analyze features of fire events, we have researched fire experiences of BWR plants based on the trouble reports of Japanese nuclear power plants. Fire events from 1970 to 1993 were researched and classified into fire sources and areas. Total 20 cases of fire events or fire related events were experienced. These fire events include events during plant operation and plant outage.

The fire events that occurred during plant operation contribute 75% of all fire experience, and the fire events that occurred during plant outage contribute 25% of all fire experience. In the events during plant operation, 13% of them resulted in reactor trip, and 20% of them resulted in manual shutdown. The rest of events during plant operation
(67%) were not required plant shutdown. Even in the 33% of events that resulted in plant shutdown, no damage to peripheral equipment from the fire source equipment occurred. So, the plant behaviors at these events were same as that of random equipment failure, and any special influence by fire events was not specified. Based on these facts, the fire events that resulted in plant shutdown have also been incorporated in the initiating events for internal event PSA.

From classification of fire source, 85% of all fire events were initiated in electric equipment. Almost all cases, each fire source equipment was damaged by short circuit, but the fire was extinguished spontaneously and the event resulted in no damage to the equipment except the fire source equipment.

Only one case of oil fire were experienced, and it contributes 5% of all fire events. As the example of another fire events, there were cases of accidental fire by maintenance or repair activities such as welding during plant outage. But these cases did not affect plant safety.

From classification of fire area, the most fire source area was the reactor building, and 55% of all fire events occurred in this area. The second most occurrence area was the turbine building, and it contributes 35% of all fire experience. There were no fire experiences in the control room and the emergency D/G room in which a lot of flammable material exists. Fire events in the reactor building were rather moderate, and fire extinction activities were not required to keep plant safety in these cases.

Only two cases required extinction activities, one was cable fire and the other was oil fire. Both of them occurred at the turbine building, and they contribute 10% of all fire experience.

The case of cable fire occurred during plant outage. In this case, rain penetrated into the cable duct outside of the turbine building, and the water short-circuited the connecting parts of the cables. The cables in the duct and the switch gears connected to the cables burned.

The case of oil fire occurred during plant operation. In this case, stuck oil to the intake filter at ventilation system in the turbine building began to burn spontaneously by oxidation. The fire did not affect plant operation.

2.3 Frequency of fire occurrence as initiating event

As described above, fire event experience in Japanese BWR plants shows that major cases were the fire events or the fire like events of electric equipment in the reactor building. Nevertheless, rather severe events that had possibility of spreading fire, were rare cases in the experience. And in the events that resulted in plant shutdown, there was no damage to the adjacent equipment from the fire source equipment. Compare to the internal events, any peculiarity in the plant condition affected by fire events was not
specified.

From the view point of fire event PSA, the event that has potential hazard to challenge the reactor safety should be selected as the initiating event. Based on this principal, the cases which require fire extinction activities will be the candidates of initiating event. During the period investigated above, 2 fire events in the BWR plants operation were identified to meet the principal, and frequency of fire occurrence as initiating event should be less than 0.01/year per plant. In addition to this, accidental fire events by maintenance or repair activity during plant outage will be another item of study.

3. Plant layout configurations relating to fire event

A BWR4 plant and a BWR5 plant were chosen as reference plants to research into fire related features, since many plants of BWR4 and BWR5 type plants have sufficient operation experience in Japan.

In this section, plant features regarding fire events of the reference BWR plants are summarized.

3.1 Fire area and research items

To arrange fire related information, fire areas were defined in the plant, and information about fire sources and other related condition were put together. Fire related design information of each area was used to research into features of equipment layout or space separation of safety equipment for fire event. Based on the research, plant safety against fire event was investigated. Also, the information collected here was used for the study of finding potential candidates of fire risk that describes below.

The information items researched are as follows.

☐ Equipment in the area (including cables)
☐ Layout of equipment in the area
☐ Safety divisions of equipment
☐ Calorific value of equipment
☐ Structures of walls, floor, ceiling and openings
☐ Fire detection
☐ Fire protection

3.2 Layout configurations of equipment

Primary design of plant layout sets safety related equipment rooms, such as ECCS or RCIC and emergency D/G, to separate from each other. This configuration basically keeps safety divisions separated spatially. So, even if fire occurs in a room (one fire area), there is small probability of spreading fire in the plant beyond safety division.

For the electric equipment, electric cabinets or cables are arranged to separate by safety divisions. If a fire occurs at electric cabinet or cable by overcurrent, it is examined by
experiments or fire analysis that spread of fire to the adjacent cabinet or cable tray will not occur. These features also show that in the case of electric equipment fire, there is quite small probability of spreading fire beyond safety division.

Another fire sources, such as a large amount of oil contained equipment, would be the important fire source. In the case of fire on these equipment, severe damage to the other equipment is suspected. For example, in the reactor building of BWR plant, a Motor-Generator set (non-safety equipment), which generate AC power for Primary Loop Recirculation pump, contains a lot of oil in it. The area of Motor-Generator sets contains the most flammable material in the reactor building. Cable trays exist in this area, but safety related cables do not exist. Also, it is examined that in the case of Motor-Generator set in fire, safety systems of the plant will not be degraded.

However, instead of Motor-Generator set, new plants constructed lately are equipped with inverter system, and quantity of flammable material has decreased.

3.3 Plant layout features from the view point of fire events
As described above, safety related equipment or electric equipment in a safety division is arranged to separate from another safety division, and spatial interaction between safety divisions is restricted to minimum. These features indicate that if a fire occurs in a safety division, it is small probability of spreading fire to the equipment of another safety division. For the non-safety equipment that contains a lot of flammable material, layout of the equipment is basically designed that combustion of the equipment would not affect safety related equipment.

From these results of the research, probability of loss of all safety systems is expected to be small in case of fire, and potential candidates of residual fire risk would be lowered to the exceptional cases that affect all safety divisions.

4. Potential candidates of residual fire risk
The research of the reference BWR plants described in the previous section shows that basically safety related equipment is designed to separate each other in layout, and such event that a fire event causes loss of all safety divisions and leads to core damage would be hard to occur.

However, as the first step of risk study, this section shows potential candidates of fire events that are important to the risk of reference BWR plants.

4.1 Finding fire areas important to the plant risk
Based on the collected information of each fire area, fire areas important to the plant risk were studied. At first, it was studied that function of all safety divisions would be lost or not by assuming fire in each area. Next, for the area that all safety divisions would not lose the function, it was studied whether or not a fire would affect adjacent fire areas. If necessary, fire analysis was carried out in these studies. In the case that there was
possibility to affect adjacent fire areas, degradation of safety functions including adjacent areas was examined.

In any case that a fire affects all safety divisions, the fire area in question is selected as the candidate of important fire area. On the other hand, in the case at least one safety division is intact, the fire event can result in no core damage because of safety functions which belong to the intact safety division. Such cases are less important to the fire risk.

4.2 Candidates of residual fire risk

An example of potential candidate of important fire event is a case in the control room. Electric cabinets or cables in the control room could be the fire sources. Even in the case of fire, there is almost no probability of spreading fire to another cabinets or cable trays, as described above. But if the fire extinction fails, all operators will evacuate the control room. In this case, both non-safety and safety systems, especially for RHR (Residual Heat Removal) system, will be difficult to control. RHR system must start up manually from the control room, so if all operators evacuate the control room, there is some possibility to lose heat removal from the reactor. Nevertheless, it is possible to operate safety systems from RSS (Remote Shutdown System), so that success of RSS operation leads the plant to safe shutdown. To quantify the sequence, both the fire occurrence frequency and the failure probability of extinction or RSS operation will be required to study.

Another example of potential candidate of important fire event is that combustion of a large amount of oil will damage cables in the adjacent fire area. In the case that oil leaks from the pump which contains a large amount of oil, and leaked oil spreads on the floor, ignition of oil will generate high temperature gas in short time. If cables exist in the adjacent area of fire, cables will be damaged by heat of high temperature gas. Moreover, fire initiating pump belongs to a non-safety system (ex. PCS) and cables of all safety divisions exist in the adjacent area, it has possibility to fail in safe plant shutdown. This sequence has many factors in the occurrence of fire, that are leakage of oil, failure of oil leak detection, and ignition of leaked oil. Therefore, study of fire occurrence frequency will be important to quantify the sequence.

As described above, the fire event in the control room or the fire event caused by a large amount of oil that has a potential to damage cables will be important to the risk of reference BWR plants. To quantify the core damage frequency of the events, estimation of fire occurrence frequency and human factors relating to prevention or mitigation of the event will be required. Especially for the estimation of fire occurrence frequency, good assumption will be required, since there was no experience of such kind of fire events in Japanese BWR.

5. Summary
This paper describes recent perspectives on the fire event PSA for Japanese BWR plants. From the fire event experience of BWR plants, it is found that frequency of significant fire event is rather small. Safety equipment is well separated division by division in the reference BWR plants, and potential candidates of fire risk would be lowered to the exceptional cases that affect all safety divisions. Based on these investigation, residual fire risk is expected to be small at the reference BWR plants. Precise estimation of fire occurrence frequency and human factors relating to prevention or mitigation of the event will be required to quantify the residual fire risk.

6. Acknowledgment

This work is based on a joint study of Tokyo Electric Power Co., Tohoku Electric Power Co., Chubu Electric Power Co., Hokuriku Electric Power Co., Chugoku Electric Power Co., Japan Atomic Power Co., Toshiba Corp. and Hitachi Ltd.
SIMULATING OF FREQUENCY FIRING’S ON TURBOGENERATORS AT UKRAINIAN NPP

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Frequency and probability of fire in different premises of NPP’s are important characteristics for evaluation of hazard and risk. For the last five years on Ukrainian Nuclear Power Plants took place 63 fires. Thereupon, that fires on NPP’s rare arises is necessity for creating of model, which use statistical information on fires on Nuclear Power Plants with a probable maximum and support data receiving for probability analysis realization. In this case is necessity to use the model which could ensure representative and significance (assurance) of results, which been received basis on current situation of Nuclear Power Plants fire safety. Significant methods of fire statistical analysis don’t allow makes prediction estimate of degradation (depletion of resources) for equipment which is grounds for accident by reason of fire.

Table 1. Dependence of place (building, room) fire on Ukrainian NPP’s

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>Fire places, %</th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Reactor room</td>
<td>Turbine room</td>
<td>Deaerating Plant</td>
<td>Auxiliary rooms</td>
</tr>
<tr>
<td>WWER-440</td>
<td>39</td>
<td>37</td>
<td>7</td>
<td>17</td>
</tr>
<tr>
<td>WWER-1000</td>
<td>34</td>
<td>36</td>
<td>8</td>
<td>22</td>
</tr>
<tr>
<td>RBMK</td>
<td>38</td>
<td>30</td>
<td>6</td>
<td>26</td>
</tr>
</tbody>
</table>

Table 2. Statistical data on engine room equipment and apparatus fires

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>Type of equipment, %</th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Turbogenerator (TG)</td>
<td>Main Circulating Pump</td>
<td>Feedwater Pump</td>
<td>Cable insulation</td>
<td>Other</td>
</tr>
<tr>
<td>WWER-440</td>
<td>35</td>
<td>17</td>
<td>10</td>
<td>29</td>
<td>9</td>
</tr>
<tr>
<td>WWER-1000</td>
<td>42</td>
<td>19</td>
<td>6</td>
<td>30</td>
<td>3</td>
</tr>
<tr>
<td>RBMK</td>
<td>59</td>
<td>15</td>
<td>15</td>
<td>21</td>
<td>10</td>
</tr>
</tbody>
</table>
Comparative data analysis from table 1.2 shows most quantity of fires takes places in engine room of NPP. In particular it's connected to breakdown of turbogenerators (TG). Windings of TG

\[ \varphi(t) = \left\{ 1 - \int_0^t \frac{dt}{T_0 \left[ a(t) \right]} \right\} T_g(t), \]  

(1)

Where,

- \( T_0 \) — initial resource,
- \( \int_0^t dT(a) \) — exhausted resource;

- \( a(t) \) — vector of loading in the moment of time t.

Denotation of \( T_b(t) \) depends on different operating conditions has different mode for different low of load variations (graded, steady and statistically distributed). Based resource of TG will define by formula:

\[ T_b(t) = \int_0^t P(t)dt, \]  

(2)

where \( P(t) \) — probability of non-failure operation of TG in time of standard conditions of operation.

Give the definition by non-failure operation of TG is possible by two paths:

- Analytical probabilistic apprehension of lows by physical processes and phenomenon.
- Statistical processing of experimental data of initial accelerated fatigue test.

We are proposing a new method for analysis of statistical data on events reduced to fire in engine room and in TG particularly. Fire occurrence for each NPP and for equipment of engine room is described by equation:

\[ E_i = \langle K_i, T_i \rangle, \]  

(3)

where

- \( K_i \) — quantity of fires;
- \( T_i \) — capability of occurrence of event, number is equal of quantity of TG by quantity of years for a NPP.

Let us have prior distribution \( \pi_0 \) of amount \( \mu \) and \( \sigma \). As a consequence of realization \( N \) occurrences \( E_i \) by Bayes' theorem we can define:
\[ \pi(\mu, \sigma \mid E) = K^{-1} \pi_0(\mu, \sigma) \prod_{j=1}^{N} \int_{0}^{\infty} L(E_j \mid \lambda) \Lambda(\lambda \mid \mu, \sigma) d\lambda, \] (4)

Where,

\( \mu, \sigma \) — parameters of log-normal distribution of fire frequency \( \lambda \);

\( \Lambda(\lambda \mid \mu, \sigma) \) — log-normal distribution of fire frequency \( \lambda \);

\( L(E_i \mid \lambda) \) — likelihood of \( E_i \) — event given \( \lambda \);

\( K^{-1} \) — normalizing multiplier.

By equation (4) a lot of new probabilities distribution functions are generated, where expected (average) curve given by:

\[ f(\lambda) = \langle \Lambda(\lambda \mid \mu, \sigma) \rangle = \int_{-\infty}^{\infty} d\mu \int_{-\infty}^{\infty} d\sigma \Lambda(\lambda \mid \mu, \sigma) \cdot \pi(\mu, \sigma) \] (5)

The calculation of non-failure work probability of TG was carried out by processing of statistical data and by gamma-, Poisson-, binomial- and log-normal approximation. It is shown, describe of equation (5) the most possible by equation (6):

\[ P(t) = \frac{1}{t \sigma \sqrt{2\pi}} \exp\left(-\frac{1}{2\sigma^2} (\ln t_0 - \mu)^2\right). \] (6)

if \( t_0 > 0 \), and: average \( \mu=2.15 \), variance \( \sigma=0.43 \), and the best possible value of \( t_0 \) from (7):

\[ \ln t_0 = \mu - \sigma^2. \] (7)

The data source of firing’s on TG is article [S.Azarov, V.Tokarevsky // Fire and safety’94. Conference, December 5-7, 1994, Spain, p.411-423].

Fire occurrences, which took places from 1990 till 1995 on NPP of Ukraine with WWER (class B-213, B-320, B-338) and RBMK-1000, was view in this article.

Life endurance of TG calculated from the initiation to operation, except period of initial testing activities. Necessary data for frequency of fire occurrences on TG calculating have included:

- Information on series of events under each fire;
- Cause of fire;
- Medium for fire spreading;
- Methods and time for fire detection and extinguishing;
- Post-fire state of NPP.

Probabilistic Distribution fire occurrences frequency in Turbogenerators is shown below.
Probabilistic Distribution fire occurrences frequency in Turbogenerators

It can be show that after most possible finishing off of TG probability of fire occurrence is to increase with 6-8 years by factor in 2 in comparison with previous period of time. Thus at a time of operation during pre-accidental state of TG (crisis margin of safety) is necessary to rebuilding exhausted resources by of preventive maintenance repair.

Hereby, proposed model for evaluation of fire risk of TG on margin of safety (resource) let us to prognoses pre-accidental states resulting in fire in real time. This model is universal and can be used for probability of frequency of fire assessment not for Nuclear Power Plants only but for Heat Power Plants and for Hydro Power Plants and for fire-extinguishing equipment and technologies as well.
Session IV
Experimental Fire Research I

- Full Scale Fire Experiments on Electronic Cabinets - Dr. Olavi Keski-Rahkonen, VTT Building Technology, Finland

- Electrical Cable Fire Tests at EdF - M. Bernard Gautier and M. Emmanuel Thibert, Electricité de France (EdF), France

- Comparison of the Burning Behaviour of Electric Cables with Intumescent Coating in Different Test Methods - Dipl.-Phys. Jürgen Will and Dipl.-Phys. Dietmar Hosser Institut für Baustoffe, Massivbau und Brandschutz (iBMB), Germany
Full Scale Fire Experiments on Electronic Cabinets

Olavi Keski-Rahkonen
VTT Building Technology

Abstract

An analytic formula is proposed to express the maximum rate of heat release of a fire inside a closed electronic cabinet of a nuclear power plant. For the maximum rate of heat release burning is oxygen limited. If there is enough fire load in the cabinet, burning is almost independent of fuel, and geometric dimensions are the rate determining variables. Data from full size and model scale experiments of various sizes of cabinets confirm, the proposed formula predicts the maximum rate of heat release for a fire inside a metallic electronic cabinet better than to a factor of two. Another analytic formula is proposed for determination of the minimum RHR to drive the cabinet fire into flashover. The formula is in agreement with available experimental data, although the amount of data is still small.

1. Introduction

The ignition and burning of cables, and other combustible materials in electronic cabinets occurs statistically often enough to have a considerable contribution to all fires on NPPs (Keski-Rahkonen et al. 1999). Electronic circuitry is often placed in relatively well closed metal cabinets. They have small openings for ventilation, and contain much fire load in form of circuit boards, cables and electronic components. In case of fire inside a single cabinet the neighbouring cabinets and the whole room containing cabinets could be involved creating in sensitive parts a severe risk for the safe operation of the whole NPP.

Only small amount of direct information is available in open literature on cabinet fires (Chavez 1987). Therefore, VTT made a survey on cabinet fires, and carried out calorimetric experiments in full (Mangs and Keski-Rahkonen 1994, 1996), and reduced scale (Paananen 1996) to determine the mass loss rate, and the rate of the heat release (RHR) to be used as source terms for computer codes simulating fire development in a big room containing electronic cabinets. The results of RHR determinations of these tests and of earlier tests at Sandia (Chavez 1987) are summarised in this paper using simple closed form models to interpret experimental data. Since the earlier international presentations on the theme have been limited to conferences (Keski-Rahkonen 1994, Keski-Rahkonen and Mangs 1995), much of the earlier results will be repeated here.
2. Model on maximum rate of heat release for a fire in a cabinet

A simple natural ventilation model was written for a fire in an electronic cabinet (Keski-Rahkonen 1994, Mangs and Keski-Rahkonen 1994). In Figure 1 a simplified model of the cabinet is presented. It consists of a single compartment with simple vents in the lower (V1) and upper (V2) sections of the cabinet. The vents are in practice usually small, and mostly unidirectional flow occurs. Uniform inner temperature theory describes the salient features of the problem.

Assuming dimensions of vents V1 and V2 vertically small in comparison with the height $H$ of the cabinet gases flow unidirectional over the whole vent areas. Assuming additionally the burning cabinet to be a well stirred reactor the temperature $T$, density $\rho$, and concentrations of the gases are equal everywhere inside the cabinet. We neglect as small the gas production due to combustion inside leading to a simple mass balance relating the entering and exhausting mass flows in a quasistationary state

$$m_i = m_e$$  \hspace{1cm} (1)

The equation of state of the gas inside is given by

$$\rho = pM / RT$$  \hspace{1cm} (2)

where $p$, $M$ are the pressure and molecular weight of the homogenous gas mixture inside, and $R$ the general gas constant.

The pressure conditions are determined by the outside hydrostatic pressure difference. We take the ambient temperature $T_o$, and the pressure $p_o$ at the level of vent V1.

In a buoyantly driven flow when $T > T_o$ gas will flow in from vent V1 and exhaust from vent V2. Denoting the pressure differences in vents V1 and V2 by $\Delta p_i$ and $\Delta p_e$, the pressure balance at outside points of levels V1 and V2 requires

$$p_o - \Delta p_i - \rho g H - \Delta p_e = p_o - \rho_o g H$$  \hspace{1cm} (3)

The mass flows through vents are (Emmons 1988)

$$m_i = A_i C_i \sqrt{2 \rho_o \Delta p_i}$$  \hspace{1cm} (4)

$$m_e = A_e C_e \sqrt{2 \rho_e \Delta p_e}$$  \hspace{1cm} (5)

where $A_i$, $A_e$, $C_i$, $C_e$ are the areas of vents. $\rho_o$, $\rho_e$ are the densities of the flowing fluid at levels of the corresponding vents, and $g$ the acceleration of the gravity.
Figure 1. Simplified cabinet model with small entrance (V1) and exit (V2) vents, (Mangs and Keski-Rahkonen 1994). Other symbols explained in text.

From the inflow of air the theoretical maximum rate of the heat release $\dot{Q}_{\text{max}}$ is calculated using the oxygen consumption principle observed by Thornton already 1917, but used in fire technology only recently, (Huggett 1980)

$$\dot{Q}_{\text{max}} = \chi C_1 A_1 \rho_o \Delta H_{c,\text{air}} \sqrt{2gH} \sqrt{\frac{1 - T_0/T}{1 + (A_1/A_e)(T/T_0)}}$$

(6)

Energy release per consumed mass of air $\Delta H_{c,\text{air}}$ 20= 2.97 MJ/kg. Difficulties of the theory are collected into a single factor $\chi$, efficiency of oxygen consumption. It is very hard to model, but can be determined experimentally using calorimetry.

3. Fire tests in electronic cabinets

Three series of experiments were carried out at VTT to verify Equation (6). In the first series three tests with a real electronic cabinets furnished with different amounts of combustible material were burned, (Mangs and Keski-Rahkonen 1994). Two of them reached flashover, and the last test is included in table 1 as test number 1. In the second series mock-up chimney-like small cabinets were used to allow variation of the height and ventilation conditions, (Paananen 1996). All four of them (tests 2 -5) reached flashover. In the third series three test with a real electronic cabinets, different in mechanical construction from that used in the first full scale series, furnished with different amounts of combustible material were burned, (Mangs and Keski-Rahkonen 1996). All of them reached flashover, but because of loss of tightness only test 3 is included in table 1. From the extended series of the earlier full scale tests at Sandia (Chavez 1987) data from two tests (ST10 and PCT1) were included, because only they apparently reached flashover, and were carried out in a closed cabinet. Table 1 summarises the geometrical dimensions of all the test cabinets.
Table 1. Dimensions of the test cabinets: sources of data indicated by reference after the test number.

<table>
<thead>
<tr>
<th>Test nr and reference</th>
<th>Height between vents (m)</th>
<th>Horizontal cabinet area (m²)</th>
<th>Vent areas (m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Entrance</td>
</tr>
<tr>
<td>1</td>
<td>1.70</td>
<td>0.348</td>
<td>0.0403</td>
</tr>
<tr>
<td>2</td>
<td>1.69</td>
<td>0.058</td>
<td>0.0050</td>
</tr>
<tr>
<td>3</td>
<td>1.69</td>
<td>0.058</td>
<td>0.0025</td>
</tr>
<tr>
<td>4</td>
<td>1.04</td>
<td>0.088</td>
<td>0.0066</td>
</tr>
<tr>
<td>5</td>
<td>1.04</td>
<td>0.088</td>
<td>0.0033</td>
</tr>
<tr>
<td>ST10</td>
<td>1.74</td>
<td>1.115</td>
<td>0.1054</td>
</tr>
<tr>
<td>PCT1</td>
<td>1.74</td>
<td>1.389</td>
<td>0.1054</td>
</tr>
<tr>
<td>3</td>
<td>2.21</td>
<td>0.293</td>
<td>0.0082</td>
</tr>
</tbody>
</table>

In Equation (6) the dependence of RHR is weak on temperature $T'$22 within wide ranges of practical importance. All other quantities are either dimensions of the cabinet or well known material properties except $\chi$ 23. The burning after flashover is oxygen limited, and very far from the theoretical fuel limited case. Therefore, the amount of fuel is not an important variable for the maximum RHR provided energy release is sufficient to lead to flashover.

In Table 2 the measured values of the measured maximum RHR are given together with the value of $\chi$ 24 calculated from Equation (6). It is observed that values of $\chi$ 25 fall within a reasonable narrow range between 0.5 and 1 allowing determination of maximum RHR within better than a factor of 2.

After initial encouraging results leading to Equation (6), Paananen (1996) analysed in detail his mock-up experiments and available full scale experiments to test rough approximations made in deriving Equation (6) as well as determining numerical parameters used for predictive calculations. His analysis confirmed the major assumptions made earlier.

In deriving Equation (6) for simplicity constant temperature was assumed inside the cabinet. Paananen (1996) measured temperature profiles inside the cabinets, and derived an equation taking into account the variation of temperature in vertical direction. The modified formula becomes
\[ \dot{Q}_{\text{max}} = \chi C_i A_i P_0 \Delta H_{\text{air}} \sqrt{2gH} \sqrt{\frac{1 - T_0/T_r}{1 + (A_i/A_t)(T/T_0)}} \] (6*)

Table 2. Measured values of RHR and \( \chi \) 26.

<table>
<thead>
<tr>
<th>Test nr and reference</th>
<th>Maximum RHR in kW</th>
<th>Efficiency of oxygen consumption</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>180</td>
<td>0.51</td>
</tr>
<tr>
<td>2</td>
<td>30</td>
<td>0.76</td>
</tr>
<tr>
<td>3</td>
<td>21</td>
<td>0.84</td>
</tr>
<tr>
<td>4</td>
<td>36</td>
<td>0.82</td>
</tr>
<tr>
<td>5</td>
<td>22</td>
<td>0.85</td>
</tr>
<tr>
<td>ST10</td>
<td>280</td>
<td>0.66</td>
</tr>
<tr>
<td>PCT1</td>
<td>185</td>
<td>0.47</td>
</tr>
<tr>
<td>3</td>
<td>97</td>
<td>0.81</td>
</tr>
</tbody>
</table>

The only change as compared with Equation (6) is given by average temperature \( T_a \) inside the cabinet calculated using a formula

\[ \frac{1}{T_a} = \frac{1}{H} \int_0^H \frac{dy}{T(y)} \] (7)

where \( T(y) \) is the temperature inside cabinet at height \( y \) in vertical direction. Equation (7) is well known in connection of stack effect, and calculates an average value of the inverse absolute temperature inside the cabinet. In terms of density of ideal gas Equation (7) calculates average density of gas inside the cabinet.

Finally he determined a revised values for a formula for calculating the maximum RHR

\[ \dot{Q}_{\text{max}} = 4.72 \text{ MW} \sqrt{\frac{H}{\frac{2.41}{A_t^2} + \frac{1}{A_i^2}}} \] (8)

where the height is given in m, and areas in m\(^2\). This formula is of the same mathematical form as the earlier proposed formula (Mangs and Keski-Rahkonen 1994), but the coefficients have slightly different numerical values, since more experimental data was available for estimation.
4. Minimum rate of heat release needed for a flashover in a cabinet

The fire tests in closed electronic cabinets have shown in several cases, that after well established ignition of material the fire extinguished by itself without burning all the material inside. This is caused by several reasons, but one of the most apparent is that the fire load consists of several items, which could be at some distance from each other. If the RHR is not sufficient to ignite the next item, fire will remain local and extinguish by itself after the first ignited item has burned out.

The conditions when fire will propagate in the cabinet has a big influence from the fire safety point of view. Describing full necessary conditions is a problem beyond the present capability, but a condition driving the cabinet to complete burnout starting from the flashover, is relatively simple. Here we draft a theory for it, and check the predictions of it against available literature data.

We adopt the model of McCaffrey et al. (1981). In a cabinet shown in Figure 1 the energy balance of the hot gas layer, here presumed for simplicity to fill the whole cabinet above the lower vent $V_1$, is given by

$$Q = m_c c_p (T - T_0) + q_{\text{loss}}$$  \hspace{1cm} (9)

where $Q$ is the rate of energy release from burning inside the whole cabinet, $c_p$ specific heat capacity of air in constant pressure and at ambient temperature. $q_{\text{loss}}$ is a term of other energy losses from the cabinet than convection of the enthalpy with exhaust gases described by the first term on the right hand side of Equation (9). In a metal cabinet the majority of the losses are heat transfer through the walls of the cabinet. It is a non-linear problem, but to describe the flashover condition it can be linearized by using the relationship

$$q_{\text{loss}} = h_t A_w (T - T_0)$$  \hspace{1cm} (10)

where $h_t$ is the effective total heat transfer coefficient at the temperature relevant to flashover. At the temperature of 500 °C it is dominated by radiation. $A_w$ is the total area of the hot walls of the cabinet. Here we assume, that all outer surfaces above vent $V_1$ are hot. Therefore the area becomes

$$A_w = ab + 2(a + b)H$$  \hspace{1cm} (11)

where $a$ is the width and $b$ the depth of the cabinet. The heat transfer through the metal wall is calculated combining the convective and radiative processes, and coupling transfers inside and outside in series with head conduction in the metal wall. The total coefficient is given by

$$\frac{1}{h_t} = \frac{1}{h_{ic} + h_{ir}} + \frac{d}{k} + \frac{1}{h_{ic} + h_{or}}$$  \hspace{1cm} (12)
where the subscripts i and o refer to inner and outer regions, c and r convection and radiation. Finally $d$ is the thickness and $k$ the heat conductivity of the wall material. To estimate the radiative component $h_{or}$ of the heat transfer we assume the wall to be at flashover temperature of 500 °C and have an emissivity of unity, which results to $h_{or} = 41$ W/m$^2$K. Inside the gas is assumed to transfer heat on the wall as a black body. Using for thickness of the wall $d = 1$ mm, and a conductivity $k = 50$ W/Km valid roughly for steel $k/d$ contribute only of the order of 1 per mille to the total resistance and is neglected. These all yield $h_k = 32$ W/m$^2$K, where rough values of 10 and 20 W/m$^2$K are used for the inner and outer convection coefficients.

Solving $m_i$ from Equations (3) to (5) results into

$$m_i = m_e = A_i C_e \rho_0 \sqrt{2gH \frac{1-T_0/T}{(A_i C_e/A_i C_r)^2 + T/T_0}}$$ (13)$$

Substituting this into Equation (7) would allow to estimate minimum RHR needed to flashover. Instead of that we follow McCaffrey et al. (1981), and rework Equation (7) into dimensionless groups of variables, from which results a correlation for the minimum RHR to drive the cabinet into flashover:

$$Q_{min} = \left\{ \frac{\rho c \rho c_0 T^3}{\Delta T (480)} \right\}^{1/2} \left\{ h_k A_w A \sqrt{H} \right\}^{1/2}$$ (14)

which after substitution of $\Delta T = 500$ K and other constants into to first square root simplifies

$$Q_{min} = 19.3 \text{ kW} \sqrt{h_k A_w A \sqrt{H}}$$ (15)

where $A$ is a weighted average of the areas of the exit and entrance vents.

In Table 3 a number of experimental results is collected from literature to check the validity of Equation (15). $Q_{min}$ is the theoretical minimum for flashover calculated using Equation (15). Maximum RHR measured in tests are denoted by $Q_{no}$ if no flashover occurred, and by $Q_{flash}$ when flashover took place. For comparison $Q_{max}$ is the calculated maximum RHR from Equation (6) assuming $x = 1$. In Figure 2 the same data are presented graphically.

On the horizontal axis of Figure 2, the calculated minimum RHR $Q_{min}$ is plotted over the range of available data. The vertical axis is used to plot experimental data. The heavy solid line divides the field into two sections: if the plotted point lays on the lower part, no flashover should occur according to our theory whereas points above the heavy line should represent tests, where flashover occurred. All the data

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presented in Table 3 fulfilled this criterion. For comparison, also data for the maximum RHR $Q_{\text{max}}$ during flashover are plotted using solid circles.

Table 3. RHR in tests of electronic cabinets.

<table>
<thead>
<tr>
<th>$Q_{\text{min}}$ (kW)</th>
<th>$Q_{\text{no}}$ (kW)</th>
<th>$Q_{\text{flash}}$ (kW)</th>
<th>$Q_{\text{max}}$ (kW)</th>
</tr>
</thead>
<tbody>
<tr>
<td>10</td>
<td>22</td>
<td>26</td>
<td></td>
</tr>
<tr>
<td>11</td>
<td>21</td>
<td>25</td>
<td></td>
</tr>
<tr>
<td>11</td>
<td>7</td>
<td>25</td>
<td></td>
</tr>
<tr>
<td>14</td>
<td>36</td>
<td>44</td>
<td></td>
</tr>
<tr>
<td>16</td>
<td>30</td>
<td>39</td>
<td></td>
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<tr>
<td>37</td>
<td>79</td>
<td>97</td>
<td></td>
</tr>
<tr>
<td>72</td>
<td>180</td>
<td>353</td>
<td></td>
</tr>
<tr>
<td>72</td>
<td>9</td>
<td>353</td>
<td></td>
</tr>
<tr>
<td>72</td>
<td>50</td>
<td>353</td>
<td></td>
</tr>
<tr>
<td>163</td>
<td>95</td>
<td>651</td>
<td></td>
</tr>
<tr>
<td>163</td>
<td>93</td>
<td>651</td>
<td></td>
</tr>
<tr>
<td>163</td>
<td>280</td>
<td>651</td>
<td></td>
</tr>
<tr>
<td>175</td>
<td>185</td>
<td>651</td>
<td></td>
</tr>
</tbody>
</table>
Figure 2. Testing the flashover criterion using data from tests (Table 3). For explanations, see text.

5. Discussion

5.1 Maximum rate of heat release

The oxygen consumption efficiencies show a narrow range of values and seem to be insensitive to changes of many variables inside the cabinet provided after flashover there is plenty of fuel available. The proposed model seems to describe the major features of the burning although it is simplified. The temperature inside the cabinet is not uniform, and the vents have finite vertical dimensions. The distribution of the fuel inside the cabinet is also of secondary importance. Therefore Equation (8) is proposed for estimating maximum RHR in electronic cabinets in a conservative way to be used in PSA work of NPPs as source terms for simulation of room fires or estimating fire spread to adjacent cabinets.

Since our experimental material is still very limited, care should be taken not to use Equation (8) outside its validity ranges. We do not know them yet. However, when the dimensions change considerably from those presented here, new phenomena might occur. Likewise, the loss of tightness whether due to distortion of doors, or burning through of parts (aluminium, plastic) increases RHR dramatically as seen in (Mangs and Keski-Rahkonen 1994, 1996).

In these experiments, most of the burning took place inside the cabinet. If the fuel is easily volatile, pyrolyzed, unburned fumes could flow from the cabinet and burn outside. In such a case our estimate in not valid any more. In our experiments during short periods of time
flames emerged from the vents, (Mangs and Keski-Rahkonen 1994, Paananen 1996). The present model is neither able to predict, to what extend the fuel burns inside, nor valid during extensive external flaming. More research is needed to explore the validity of the proposed model for bigger cabinets, different fuels and vent configurations.

5.2 Minimum rate of heat release

In Figure 3 all data plotted fulfil the condition proposed by Equation (15). Therefore criterion like that or any other of similar base could account for a useful flashover criterion. This could be used to assess the fire performance of electronic cabinets without performing real full size tests.

However, full agreement between data and theory might be fortuitous. The amount of data is too small, and the range too narrow to draw final conclusions. Furthermore, the limitations of the theoretical range of Equation (15) should be clarified, because in the limit of very small vents the conditions may not be fulfilled.

In principle, the limit presented by the heavy line is not sharp. If the cabinet is tall or has partitions dividing it into different compartments like in (Mangs and Keski-Rahkonen 1996), flashover could occur locally in them without involving the whole cabinet. There was slight indication of that in experiments of the last full scale series. The partition was not tight, and therefore fire spread quickly in the whole cabinet once high enough temperature was reached. If compartmentation separates parts of cabinet leaving only small openings between different parts, no simple relations could be expected for total behaviour of the system, because then the fluid dynamic coupling between different compartments in the cabin becomes complicated.

6. Conclusions

For fire inside electronic cabinets two analytical formulas are proposed for theoretical calculation of the maximum rate of the heat release after flashover, Equation (8), and of the minimum rate of heat release leading to flashover of the total contents of a cabinet, Equation (15). Theoretical predictions are validated using experimental observations, although the amount of data available is still rather small. These results can be used in estimating fire size, as well as for input data of energy release in calculating fire spread in switch gear rooms and similar places in NPP's using numerical fire simulation, when the origin of fire is inside an electronic cabinet.

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References


Electrical cable tests at EDF

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Keywords: Fire behavior of electrical cable. Real-scale test, Calorimetry, fire model

Abstract: This paper makes a review of several experimental tests carried on at EDF from the eighty's to assess the effect of a cable fire in a Nuclear Power Plant. From the performance tests which normalize the fire resistance requirement to the last real scale fires, a great effort as been done to determine and limit the fire risk associated to this omnipresent component. The Research and Development Division of EDF is working on a fine modeling approach dealing with the fundamentals of solid combustion.

Introduction

The experimental work presented here as been carried on by SEPTEN which is the engineering center of EDF devoted to conception and maintenance of equipment for thermal and Nuclear Plants, and DRD, the EDF research center. The list is not exhaustive but tries to describe briefly the more characteristic campaigns and give a state of the art. The work in progress at EDF/DRD on cable combustion modeling is touched on.

Regulation performance tests

It is necessary to begin this presentation with the performance test that have been used since the seventeen's to select the cables for French NPP. The AFNOR NF C 32-070 test1 (~CEI 3321-1, ~US UL 83) and NF C 32-070 test2 determine respectively the C2 and C1 class for fire resistant cables. They are based on the a fire propagation test in vertical position. The bottom of a few cables is exposed to a flame or a oven, and the distance damaged should not pass a convened distance, the C2 class being more restricting.

![Figure 1: performance tests scheme](image)

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These tests have been quite efficient to qualify the fire resistance of the materials used in the cable sheathes. They assure non-propagation properties, and this kind of cable is indeed quasi-impossible to ignite using a small initial source, such as a lighter for instance. Nevertheless, it does not mean that the cables will not participate, as a combustible, to an existing fire. Significant accidents, have shown that they could take great part of it. For this reason, EDF has realized numerous tests from the last seventies in order to determine and limit the fire risk associated to a component present in huge amounts in a NPP.

**Real scale test series in Fort de Chelles (CNPP 79-86) [1]**

A important number of tests have been realized between 1979 and 1986 in collaboration with CNPP to explore many aspects on the fire behavior of cables in superposed horizontal trays. The principal aim of these test series was to evaluate the efficiency of different extinction systems and choose the more adapted for cables. Halon, CO₂ moss and pulverized water systems were tested in various geometrical configuration with different types of cables: non-halogen, PVC and mix. The effect of ventilation and smoke extraction was also studied. The fires took place in a cellular concrete room of 8.1m×4m, height 2.15m (70m³). Gaz burner (40 kW) were used to ignite the lower cables.

![Diagram of real scale tests at Chelles (79-86)](image)

*Figure 2: Real scale tests at Chelles (79-86)*

A lot of useful information were obtained from these test series : the cables showed no horizontal propagation but flashover were observed for the upper cables after a time laying in the hot upper zone. The ventilation (8 to 20 vol/h) influence on the fire efficiency was observed, some fire were stopped by lack of oxygen. The behavior of PVC cables (resistant 80 ‰) and no-halogen cables (30 ‰) were compared. Many protection rules were reinforced by the experiments: adoption of pulverized water systems, limitation of the number of superposed cable trays, thermal protection of the upper cable trays.
Cable fire in confined compartment (INERIS 95)

A real scale test of cable fire in a confined room was realized in 1995 by INERIS, with aim to analyze the loss of function of exposed cables. This test confirmed that no fire spread was possible without significant airing of the compartment. The fire stopped after 20 minutes with no effects on target cables with an airing of 5 vol/h [2].

Two room test series in CNPP Vernon (96-97)

Four tests were realized in 96-97 by CNPP at Vernon. They aimed to provide information for code validation. Measurement of mass loss, temperature and flux were performed in a two cellular concrete compartment configuration (72 m$^3$ and 48 m$^3$). The ventilation was natural (̃1 m$^3$). It was measured by flowmeter and the opening lintel could be varied. Two of the tests involved a configuration of seven superposed trays [3]. The upper trays were thermally protected the same they are in NPP condition. Cables bunker above the trays were also tested. The initiator for this cable tests was a pool of 12 and 20 l heptane, releasing up to 500 kW during about 20 '. Except for the quantities concerned, this kind of "lighter" corresponds to a realistic initiator (a forgotten bottle of solvent for instance) and has been shown to be quite efficient for cable ignition.

Figure 3: Cable fire tests in Vernon

These test have shown a complete combustion of the cables from the seven superposed trays; due to the sufficient airing and the powerful fire initiator. 240 kg of sheath were burnt in one test, 335 kg in the other. No damage was observed for the protected cables and bunker. The data from these experiment are part of the MAGIC code validation file (figure 4) [4].
**Medium scale tests: CNPP/EDF Calorimetric cell**

These tests have been realized from 1995 by EDF/DRD at CNPP in a cellular concrete cell of 31.5 m³. The exhaust duct is equipped with O₂, CO, CO₂ and C₆H₆ sensors. Together with a Pitot speed measurement, this system enables an estimation of the heat release rate during the combustion of components in the cell (Up to 200 kW). It was calibrated through propane burner tests [5].

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**Figure 5: CNPP/EDF calorimetric cell**
The device provides characteristic scenario of mass and heat release in semi-real fire. Testes were realized on sample of 1m length (1 tray) of different type of cables. The data obtained is reliable and destined to cable combustion modeling [6].

figure 6: data from the calorimetric cell

The PITCARN /CALORIMETRE test bench

This test bench is of the PITCAIRN oven, which is composed principally of a stainless steel pipe of 1.3 m length and 0.21 m diameter, and of the CALORIMETRE device [7], an evolution on the Tewarson calorimeter (1 m length and 0.23 m diameter). PICAIRN enables a pyrolysis in inert atmosphere (Helium) under controlled condition (Up to 360 °C skin temperature, ie 3W/cm²). CALORIMETRE enables the measurement of dynamic Effective Heat of Combustion.

figure 7: The PITCAIRN/CALORIMETRE test bench

Both devices can be coupled [7], and reliable data for modeling is obtained from the separation of pyrolysis and flame (figure 8).
**Loss of function**

Experiment were carried by SEPTEN/ELN in 1996 to estimate the loss of function of cable under fire condition. A temperature controlled oven was used to provide steps of temperature on large periods. Different samples of cable were tested and it was possible to propose a classification of cables in regard to the time they could keep their functionality at a certain level of temperature [9].

**What did we learn about cables fire behavior?**

The experience has shown that several phases can be generally be observed in a cable fire:

- Function resistance ex: 160 °C 24h (PVC C1)
- Ignition resistance ex: PVC cable: > 250 °C
- fire spread phase : no or slow horizontal propagation is observed, vertical propagation takes place into a cone above the ignition point.
- flashover phase: after staying in upper zone of smokes, the cables ignite tray by tray from the upper zone to the lower.

The fire in sensible to general fire condition effects:

- initial source ( solvent pool seems the worst)
- Effect of ventilation
- Wall proximity and nature
- smoke and hot gases room filling effects

Configuration effects have been observed :

- superposition of horizontal trays must be limited
- type of cable (command and measurement, power), mixing, cable technology : non-halogen, PVC, EPR, others..

**The main objectives of modeling**

The objectives of the cable behavior modeling can be classified as follows:

To predict the loss of function (1)
To predict the ignition (piuted, spontaneous) (2)

To predict the dynamic heat release of a given cable fire

- To predict the spread of a fire initiated in a cable bundle (3)
- To estimate the global mass loss and heat release (4)
- To predict the cable fire efficiency and extinction (5)

The first two lines are matter of cable thermal models, the last one of fire and flame model. A pyrolysis model is necessary to approach points 3 and 4.

**Thermal model of cable**

A program is at work at EDF/DRD to propose a fine approach of cable behavior. First, a thermal model for cable has been developed. It is based on the classical thermal transfer equations, considering a cable cut into section of a few centimeters.

The surface balance of the cable is the following

\[ \Phi_{\text{Total}} = \varepsilon \Phi_{\text{ray}} + \Phi_{\text{conv}} - \varepsilon \sigma T_s^4 \]

with:

- \( \Phi_{\text{Total}} \): Thermal flux (radiation + convection) absorbed by the cable segment (=conductive flux into the first-layer) [W.m\(^{-2}\)]
- \( \Phi_{\text{ray}} \): Incident radiative flux on the cable segment [W.m\(^{-2}\)]
- \( \Phi_{\text{conv}} \): Convective flux received by the cable segment [W.m\(^{-2}\)]
- \( \varepsilon \): Emissivity of the cable external layer
- \( \sigma \): Stephan-Boltzmann Constant \( [5.67 \times 10^{-8} \text{ W/m}^2\text{.K}^4] \)
- \( T_s \): Surface temperature of the cable segment [K]

The convective exchanges of the cable with its surroundings are quantified with a linear convective exchange coefficient:

\[ \Phi_{\text{conv}} = h_{\text{cable}} (T - T_s) \]

with:

- \( h_{\text{cable}} \): Linear convective exchange coefficient with the air \( h=8.5 \) [W.m\(^{-2}\).K\(^{-1}\)]
- \( T \): Temperature of the zone where the barycenter of the cable segment is located [K]
- \( T_s \): Surface temperature of the cable segment [K]

The energy conservation is applied on an elementary ring in cylindrical coordinates:

\[ \frac{\rho C}{\lambda} \frac{\partial T(r)}{\partial t} = \frac{\partial^2 T(r)}{\partial r^2} + \frac{1}{r} \frac{\partial T(r)}{\partial r} \]

with:

- \( T(r) \): Temperature in the cable segment at radius \( r \) [K];
- \( \rho \): Material mass density [kg.m\(^{-3}\)],
- \( C \): Material specific heat [kJ.kg\(^{-1}\).K\(^{-1}\)].
\( \lambda \): Material thermal conductivity [kJ.K\(^{-1}\).m\(^{-1}\)].

This kind of model has been confronted to the PITCAIRN controlled conditions with good agreement (figure 9). It is now implanted in the code MAGIC [10]. The thermal behavior is supposed symmetrical and the concept of sample cable is used, because it results more conservative than that of an "average" cable. The radiative flux in decomposed as follows:

- direct incident radiation from fires located in the same room
- incident radiation from zone and interface walls
- diffuse incident radiation from adjacent rooms through the openings
- incident radiation from the gas of the zone
- incident radiation from fires of unburned gases at the openings
- incident radiation from fires located in other rooms through the openings

![Figure 9: thermal model and experimental data from PITCAIRN](image)

**Pyrolysis model**

The cable pyrolysis approach is based on the Arrhenius formulas.

\[
dm/dt = A e^{-E/RT}
\]

were \( A \) is a pre-exponential factor [s\(^{-1}\)]; \( E \), activation energy [J.mole\(^{-1}\)]; \( n \), order of the pyrolysis reaction. In case of a complex combustible such as a cable, the characterization of each reaction of each protection sheath has to be obtained through thermo-gravimetric analysis [11]. An Arrhenius equation is attached to each of them.

Due to the fire resistant mineral component included in it, cable sheath mostly produces coal residues. The KUNG model was adopted [12]. A mass conservation equation is introduced:

\[
\frac{1}{r_j} \frac{\partial (r_j \dot{m}_j^\prime \prime \gamma)}{\partial r} = -\frac{\partial \rho_j}{\partial t}
\]
with $m_{j,r}$, mass pyrolysis at node $j$ [kg.m$^{-2}$.s$^{-1}$] calculated by Arrhenius formulas, and $r_j$, the distance to the center of the cable at $j$ [m].

This model is complex and won't be exposed furthermore here. Many simplifying hypothesis are taken, among other things:

- no residual humidity in the cable
- no pyrolysis front
- no intumescence
- no changes on surface (axysymetrical problem)
- isotropic heat fluxes
- no crossing time for the gases
- gases and solid at the same temperature
- Constant specific heat of gases
- thermal conductivity is a linear interpolation to material properties to coal properties
- etc..

![Graphs showing calculation/experiment comparison for pyrolysis model](image)

**Figure 10: Calculation/experiment comparison for pyrolysis model**

The results obtained with this model by comparison to PICAIRN/CALORIMETER testes already show that it gives a good idea of the mass loss. Better information, in regard to a one specie model, can be obtained on the heat release prediction, thanks to the evaluation of a medium NHV for each specie emitted [13].

**Conclusion**

The cable tests at EDF represent a large amount of work from which lot of information and teaching have been obtained. The proposal of EDF/DRD in this field is to provide a panel of tools, from practical scenarios available for fire modeling purpose to a fine approach of phenomena coupling. The pyrolysis model developed at present aims at a sufficient level of representation, to understand the mechanism of the cable behavior: fire spread, extinction or flashover. These effects probably depends on the balance between cable thermal comportment, species gazeification reactions and heat release power of the emitted gases.
Références


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COMPARISON OF THE BURNING BEHAVIOUR OF ELECTRIC CABLES WITH INTUMESCENT COATING IN DIFFERENT TEST METHODS

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ABSTRACT

Cables used for electric power and control systems represent a significant potential hazard, particularly in buildings with highly complex wiring. To reduce the risk of ignition and of flame spread and to limit the fire effects, protective intumescent coatings or cables with fire retardant insulation materials have been implemented in safety related areas and locations, particularly of nuclear power plants.

Since 1994, extensive experimental research has been performed by iBMB of the Braunschweig University of Technology on the burning behaviour of unprotected as well as coated PVC cables. The main goal of this investigations was to develop a qualification method and licensing procedure for cable systems with protective coatings based on realistic fire conditions, which is accepted by the building authorities for application in buildings in general and, in particular, in nuclear power plants.

For comparative results of tests with intumescent coatings in different scales - according to the standard IEC 332-3 (large scale), German standard DIN 4102-1 (intermediate scale) and ISO 5660 (small scale) - will be introduced. The setup of the different test procedures and the corresponding phases of natural fires will be discussed. It will be shown that cables with protective coatings are appropriate for fire scenarios up to a developed room fire while only resistance to a local ignition source could be guaranteed by testing according to international or national valid standard. The necessity of testing cables with insulation material, which should guarantee a fire retardant behaviour, by these different methods should be reasonable, because depending on the results of future tests, a more distinguished qualification and application of cables might be possible.
INTRODUCTION

Cables used for electric power and control systems represent a significant potential hazard, particularly in buildings with highly complex wiring. To reduce the risk of ignition and of flame spread and to limit the fire effects, protective intumescent coatings or cables with fire retardant insulation materials have been implemented in safety related areas and locations, particularly of nuclear power plants.

Since 1994, extensive experimental research has been performed by iBMB of the Braunschweig University of Technology on the burning behaviour of unprotected as well as coated PVC cables. The main goal of this investigations was to develop a qualification method and licensing procedure for cable systems with protective coatings based on realistic fire conditions, which is accepted by the building authorities for application in buildings in general and, in particular, for nuclear power plants [1].

Continuing the former work, experiments using the national testing facility for building products, the so-called "Brandschacht" (DIN 4102 Teil 1 [2], Teil 15 [3], Teil 16 [4]), the internationally well know cable fire test according to IEC 332-3 [5] (comparable to DIN VDE 0472 Teil 804 [6], Prüfart C), the Cone-Calorimeter (ISO 5660-1 [7]) and the qualification method developed by iBMB have been planned. In addition to PVC-cables with different types of protective coatings, a number of cables with fire retardant insulation material as well as cables with non-PVC insulation material, which maybe used in EPR, should be tested.

These investigations were proposed in July 1998, but due to the change of the German national federal government, there were indications that there was no more interest on such a program by the Federal State Authority responsible, the Ministry for Environment, Nature conservation and reactor Safety (BMU), giving no permit for the respective project. Therefore, another attempt was started in May 1999 to perform a similar program with main emphasis on fire protection at nuclear power plants. This program will hopefully be supported by the technical association of the power plant licensees (VGB) as well as by the electric power companies.

Since new experimental results are not yet available the burning behaviour of PVC-
cables without or with protective coating using different testing facilities are discussed. It should be explained, for which reason the new qualification method seen best to describe the risk of burning cables in most cases of natural fires.

SET-UP OF TEST PROCEDURES

On one hand, ignition and fire propagation depend on the physical and chemical properties of the material. On the other hand, the course of fire is influenced by boundary conditions given by the fire load itself as well as by the compartment specific conditions. In this case, package density of cables, location and orientation of cable trays are in addition to the cables themselves significant parameters. A supplementary ignition source, other fire loads, dimension of fire compartment, size and location of ventilation openings are characteristics, which are necessary for a quantitative description of the fire risk. A comprehensive physical model of ignition and fire propagation is not available, therefore the classification of the burning behaviour is carried out by testing materials under fixed boundary conditions. These conditions are equivalent to some phases of natural fires. Due to this, the results of testing can be applied to these phases.

Particularly in case of electric cables, four different testing procedures would be considered:

1) "Brandschacht" test, German small scale testing facility for building products according to DIN 4102 Teil 1 [2], Teil 15 [3], Teil 16 [4]
2) Cable fire test, national cable testing procedure according to IEC 332-3 [5], approximately comparable to national DIN VDE 0472 Teil 804 [6] Prüfart C, large scale
3) Cone-Calorimeter test (according to ISO 5660-1 [1]), small scale
4) Room fire test, qualification method developed by iBMB, large scale

A detailed comparison of the above mentioned procedures is given in table 1. Figures 1 - 4 illustrate the general layout of the procedures.

In the "Brandschacht" test or cable fire test a premixed flame hits the bottom of
vertical arranged cables. This flame causes a large heat flux at a small area, which is corresponding to a fire of a waste paper basket. In the large scale arrangement of cable fire test only one surface is exposed to the flame, but in "Brandschacht"-test there are four inner surfaces of a rectangular shaft affected. Classification is based on two criteria: After switching off the burner, the fire of the cables must be self-extinguishing. The fire affected area on cables has to be limited (Figure 1, 2). Outside this area, there are no indications of burning. Adopting this criterion to coated cables, the question arises if also swelling up of an intumescent coating or changing another property will be an indication of fire, as some experts are claiming, or only traces of burning at the cable insulation material have to be taken into account determining the limitation of the affected area.

In contrast to the above mentioned test procedures in Cone-Calorimeter or room fire tests (qualification method of iBMB), the total cable area is exposed to thermal stress: In the Cone Calorimeter a homogenous heat flux is produced by the cone-shaped radiated heater. Coated cables usually have been exposed to a heat flux of 50 kW/m² in the iBMB experiments. If pyrolysis occurs, flammable products are ignited by an electric spark (Figure 3). Considerable criteria in this procedure are the ignition time and the rate of heat release. In the room fire tests, a cable tray is heated up while the temperature in the test chamber is increasing up to 400 °C. A gas burner, running in intervals of 5 minutes with interruption of 5 minutes, is used as a local ignition source (Figure 4). Testing electric cables with intumescent coating, the moments when ignition occurs and flame spread starts or flame propagation reaches the upper end of the cable tray are considerable properties of qualification.

The methods of the first group, "Brandschacht" or cable fire test (IEC 332-3), and those of the second group, Cone-Calorimeter or room fire test, are based on different fire scenarios. In the first case, it is assumed that the cables themselves are the cause of fire or a small ignition source has caused the fire of cables. Thereby, only a small area is affected by the fire at the beginning of the test. There are no other sources of heat release, which lead to additional warming up of the cables. The course of fire is mainly controlled by the cable characteristics on their own.

Testing cables by methods of the second group, the cable material is heated up by
receiving a considerable amount of energy from outside along the total length. Production of pyrolysis gases is mainly controlled by the amount of received energy and the increase of cable temperature, thus by radiant heat flux (Cone-Calorimeter) or environment temperature (room fire test). Due to this, a considerable fire will be assumed, which is supported (fed) by other fire loads.

The boundary conditions of “Brandschacht” or IEC 332-3 are equivalent to the phase of starting fire on cables, because only a small area of cables is affected by a fire. However, on the other hand the Cone-Calorimeter or the room fire test are corresponding to the phase of spreading or fully developed fires, because the whole area of cable is exposed to the fire effects. The latter methods are also including starting phase of natural fires, like “Brandschacht” or IEC 332-3, with the result that the methods of the second group are sufficient for the classification of the burning behaviour.

CLASSIFICATION OF COATED CABLES

PVC-cables will neither meet the requirements of a “Brandschacht” test nor a cable fire test. Maybe PVC-cables of large diameter with high amount of copper show a good burning behaviour and could get a classification. In the Cone-Calorimeter, where unprotected cables have been ignited at an earlier moment, the burning of PVC is assigned to a large value of heat release rate. Vertically positioned cables without protective systems were tested in large scale experiments by pre-heating the chamber up to 400 °C. This cables were ignited directly after starting the additional gas burner. Flames would reach the upper end of the tray within less than two minutes after ignition at the lower part, the value of vertical flame spread velocity was up to 480 cm/min.

PVC-cables with a protective intumescent coating had been tested in the “Brandschacht” in the early eighties. Indications of fire could be found only in the neighbourhood of direct flame application. It can be assumed that an intumescent coating will guarantee that the corresponding classification criteria are met. Different types of intumescent coatings applied to PVC-cables were tested according to IEC
A thickness of the intumescent coating of 0.7 mm was enough to keep the area affected by fire very small and guarantee the classification properties of IEC 332-3.

Comparing the results of testing unprotected as well as protected cables in the Cone-Calorimeter gave a reduction factor of approximately 0.3. This factor is the maximum or average heat release rate ratio of coated in comparison to non-coated PVC cables. A PVC cable with intumescent coating of 1 mm thickness will act in the same way as the same cable without protective coating which is exposed to an 0.3-fold radiant heat flux in Cone-Calorimeter.

Pre-heating environment of cable trays with intumescent coating up to 400°C ignition occurs after 30 minutes, flame spread along the cables can be observed approximately after 45 minutes. Additional measures at the fixing clips of cables are necessary, if this result will be obtained also in case of vertical cable trays. In case of less than 0.7 mm coating thickness, ignition and flame spread along cables with intumescent coating was starting even after 30 minutes. Flame spread also occurs early, if there might be something wrong with regard to the chemical composition of intumescent coating. The qualification method of iBMB seems to be very sensitive on the application (e.g. coating thickness) as well as on the quality of such intumescent coating (e.g. chemical composition).
APPLICATION OF INTUMESCENT COATING

Two types of escape and rescue routes have to be distinguished: necessary ones and low frequented ones. Depending on this assignment, the use of electric cables in escape and rescue routes is more or less restricted by a German national guideline [8]. Up to now, electric cables without any specification of the burning behaviour (e.g. PVC insulation material) could be installed in rescue and escape routes, if the fire load did not exceed 7 kWh/m². If the cable insulation material would not contain halogens and the cables meet the specific test criteria, the fire load would be limited up to 14 kWh/m². The practical application of this guideline has caused a violation of this restriction in many cases, because of exceeding the cable fire load limitation from the beginning or due to changes in utilisation or reconstruction of a building. The use of intumescent coatings or other measures for fire protection of PVC cables is not taken into consideration.

The draft of a new guideline, issued in November 1998 [9], will allow the application of any cables with flammable insulation material in escape and rescue routes, if these routes are low frequented and the cable guarantee the criteria of the cable fire test (DIN VDE 0472 Teil 804 [6]) or the “Brandschacht” test [2]. In other escape and rescue areas cables are not permitted to be installed outside of cable installation channels or below suspended ceilings, which guarantee a specific fire resistance rate. In addition, the experts, working on the draft [9], wanted to issue a specific testing guideline for cable coatings, based on the room fire tests. If an intumescent coating on cables guarantees the criteria of the future guideline (room fire test), it will be permitted to use this protective system on PVC cables in necessary rescue and escape routes without any quantitative restriction of cables. It is recognised that PVC-cables with qualified intumescent coating represent a lower risk of fire than cables, which only guarantee the criteria of “Brandschacht” test or cable fire test. The German Nuclear Safety Committee (RSK) took these characteristics of coated cables into consideration, when already in 1996 recommending the fire protection of PVC cables with intumescent coatings for NPP built to earlier standards.
In addition, the national guideline for fire protection in industrial buildings [10] has to be taken into consideration. Considering cables as fire load, they are described by heat of combustion and mass. In the same way, an intumescent coating could be regarded as a fire load, calculating the product of mass and calorific value. On this way, some critics claim that the risk of coated cables would be higher for coated cables than for unprotected ones because they calculate the total risk as sum of the fire load from the cable itself and the additional coating (summarising both products of mass and calorific value). The same people also prefer the combustion factor m [11] as a suitable measure for describing total risk of fire in industrial buildings. Taking the above mentioned results of testing coated cables into account, it seems reasonable to rate coated cables with a very low combustion factor, if it is possible to measure this value anyway. So there seemed to be a contradiction in the line of reasoning. Even if it is possible to determine a combustion factor of coated cables, this value would not be suitable to calculate the analytically required fire resistance rate by means of national industrial building standard [12], because the participation of coated cables on fire is mainly dependent on the burning behaviour of other fire loads. For this reason, it is also very difficult to fix the value of combustion efficiency and make use of the simplified method for risk oriented design of structural fire protection measures [13] in case of coated cables as main fire load. It seems to be difficult in principle, to apply methods, which are based only on fire load valuation, on materials with improved ignition and flame propagation behaviour, e.g. by using fire retardant compounds.

OUTLOOK

Taking a closer look on to the different classification methods, it is reasonable that the tests based on the phase of starting fire, “Brandschacht” test [2] or cable fire test [5], are not sufficient for a complete estimation of the fire risk in case of cable participation, particularly if the cables are protected by intumescent coatings.

A more sufficient description of cable burning behaviour would be given by testing the material in the Cone-Calorimeter [7] and in large scale room fire test with pre-heating of the environment, because this methods also include the phases of fire.
spreading and fully developed fire. This could be made clear by comparing the results of testing PVC-cables with intumescent coating in different apparatus and by pointing out different applications for qualified cables or cable coatings.

In the same way, cables with insulation material, which should guarantee a fire retardant behaviour, have to be assessed. Depending on the results of testing cables by different methods, a more distinguished qualification and application is possible. For that reason, the investigation program, which is proposed to the VGB, should be permitted as soon as possible. Furthermore, a more detailed description of cable fires is also necessary, if techniques like that given in the industrial building standard [12] or the simplified method for risk oriented design of structural fire protection measures [13] should be applied.
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<td>frame, width 420 mm, height 3600 mm</td>
<td>tray width 600 mm, height 4000 mm</td>
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<td>shaft, vertical</td>
<td>vertical, distance between cables fixed, number of cables fixed</td>
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<td>fixed</td>
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<td>width 2000 mm, length 1000 mm, height 4000 mm</td>
<td>width 3600 mm, length 3600 mm, height 5600 mm</td>
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<td>forced, 5 m³/min incoming air</td>
<td>free, 1 m² area inlet, 3 m² area outlet</td>
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<td>homogenous 40 °C at the beginning</td>
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<td>0 - 20 min: heating up 20 - 400 °C 20 - end of test:</td>
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<td>natural flame 50 kW, 20 - 25 minute, 30 - 35 minute,</td>
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<td>ignition time rate of heat release</td>
<td>non-affected area (&gt; 15 cm)</td>
<td>self-extinguish flame, non-affected area (&gt; 0 cm)</td>
<td>ignition at the earliest 30 min self extinguish flame until 40 min</td>
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Figure 1: "Brandschacht" test, DIN 4102-1 [2]

Figure 2: Cable fire test, IEC 332-3 [5]
Figure 3: Cone-Calorimeter, ISO 5660 [7]

Figure 4: Qualification method developed by iBMB
REFERENCES


[7] ISO 5660 Fire tests - Reaction to fire; Part 1; Rate of heat release from building products (cone-calorimeter method), April 1997


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Modelling of fire analysis in modern Swedish PSA's

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Abstract

In this paper we present an alternative analysis approach for performing fire analysis compared with a traditional conservative modelling approach. The method was originally developed for the upgraded PSA of Barsebäck NPP and it has also been used for the Oskarshamm Unit 2 PSA analysis. The aim of the method is to have a "realistic model of the actual event". The paper will make a discussion about the impact of the new analysis model compared with the original conservative approach. Benefits and drawbacks of the model will be given and insights and lessons learned from the analysis of Barsebäck NPP will be made.

1 Introduction

The use of PSA has become one of the most important tools of reactor safety of a Nuclear Power Plant. In Sweden, the use of PSA has gradually been given a larger importance in the continuous work of reactor safety. All reactors have a level 1 PSA and the process of producing new and upgraded level 1 studies are ongoing for all the NPP. The upgraded level 1 study are focusing on two items; analysis of external events, mainly fire and flooding analysis and detailed modelling of the support system, power- and signal systems. The latter in the context of room location in the power station building.

Initiating events for a fully developed Level 1 PSA include both internal and external events. The former includes transients, LOCA and CCI and the latter includes fire, internal and external flooding, earthquake etc. In Sweden the first generation of PSA were developed covering only the internal events. An example of this was the first version of the PSA level 1 for Barsebäck NPP, which was finalised in 1984 [1] and included five LOCA events and six transients. After that the Barsebäck NPP concentrated on performing a fire analysis as well as upgrading the level 1 PSA. A new version was released 1987 where the fire analysis was presented as a separate study. The fire analysis was performed according to the method suggested by Berry et al [2], i.e. each room was given a ranking according to how large the "fire load" was. The ranking was then used as a divisor for splitting an overall frequency of fire at the NPP. If a fire occurred in the room all objects and components in the room was assumed to be out of order. Since the computer tools at that time were not as developed as today, the fault tree model at the time could not in an easy manner handle the functional dependencies of rooms and components. The fire analysis was done by looking at the cut-sets in the level 1 PSA and comparing them to the list of where the components was placed.

In the beginning of the 1990's both the computer tools for performing PSA and the level of details had evaluated. The revised level 1 PSA gave some interesting findings. One was that there was a larger functional dependency of which room a component belonged to (as well as where the cables of power supply and signal were placed) than previous analyses had showed. The old conservative method of performing analysis of external events; determining in which room the component/its cables was placed, estimating a frequency for the initiating event (fire, flooding etc.) and then assuming that the component function was destroyed given an external event was not good enough. The PSA gave the functional dependencies which was good but the comparison with the ordinary level 1 PSA gave a somewhat lopsided picture of the total result. All measures to be taken of reactor safety were to be done in preventing external events. This was by many analysts felt as untrue and unrealistic. Therefore, in 1994 the Swedish Nuclear Regulatory Body (SKI) together with the Swedish utilities started a project in specifying how to perform analysis of external events in Level 1 PSA [3]. One of the findings in that project was that more realistic models of the fire process must be performed, e.g. examining the fires which have occurred showed in several cases that the fire did not evaluate to a fire which could have any effect on a component.
2 Objective of the analysis

In 1996 the Barsebäck NPP started the fourth revision of their Level 1 PSA. This version would also include "realistic models" of the external event fire. The meaning of "realistic" was not specified in any more sense that a conservative approach in accordance with the former analyses performed was not allowed. Sycon EnergiKonsult (former Sydkraft Konsult) was appointed the mission of performing the realistic analyses with the input from the new level 1 PSA. Sycon EnergiKonsult defined the object of the "realistic analyses" as follows:

- The realistic fire analysis will calculate the temperature, pressure and level of the smoke gas layer in the room by use of deterministic physical models. From that result an estimation of the failure frequency of the different components will be made.
- The realistic analyse will give the time frame of the physical process from the time of the initiating event to the time when components eventually will be out of order.
- The realistic analyse will give a verbal description of what is happening in the NPP when the external event is taking place. This in order to increase the usability of the study.
- Individual component failure frequencies will only be given for those rooms where it is possible to identify the individual components/cables. If this is not possible general failure frequencies will be given for different groups of components.

3 Analysis procedure

Given the objectives of the fire analysis an analysis procedure was developed. This is described in figure 1 and explained in the text below.

![Diagram of the analysis procedure]

**Figure 1: Overview of the analysis procedure**

- As input to the analysis of fire a screening analysis by use of the PSA model is performed for selecting the critical rooms (the rooms to be analysed). Setting the initiating event frequency to 1.0 for each room performs the screening analysis. The rooms, which do not have any barrier (zero-barrier), will come out with the core damage frequency 1.0. The PSA model also gives a list of which components can be found in the room.

- When the rooms are selected the source of fire is identified. Fire is considered as being possible to occur in all the rooms selected.

- Each room is surveyed in order to establish the positions of the components, the geometric data, positions of doors, ventilation etc. This in order to perform the physical modelling of the fire.
process. This is done by studying layouts, process schemes and by inspection of the actual room. Photographs are also used for documentation of the room.

- The initiating event for fire is estimated based on the method suggested by Berry et al. [2]. This method was used for the original fire analysis of Barsebäck and background material could be used in this analysis.

- The physical fire process is modelled separately by use of a computer tool. This is described in more detail below.

- The result from the analysis of fire will give the result as different physical parameters, e.g. pressure and temperature of the room at different time points. The impact/failure intensity of the components in the room will then be estimated. In this analysis this is done by use of different data sources. For the fire analysis the component is assumed to withstand a certain temperature but for the flooding analysis the average scenario is that components are given higher failure intensity due to the new environment.

- When the impact of the components has estimated a transfer of the results is done to the PSA model. For Barsebäck PSA this means transfer to the RISK-SPECTRUM® program and use of attributes exchanging the failure frequency for the basic events.

4 Fire Analysis

The initiating event is estimated by use of the method suggested by Berry et al. [2]. The method was used for the previous PSA of Barsebäck NPP and therefore the background material from those analyses could be used. The method basically gives each room a ranking according to how large the "fire load" is. The ranking is then used as a devisor for splitting an overall frequency of fire at the NPP. The overall frequency of a fire is taken from the Swedish X-book [5], which contains data for external events at the Swedish Nuclear Power Plants. The result of the analysis of the initiating event is given as a failure frequency.

Estimation is given whether the fire is detected or not. All rooms considered in the PSA are covered of the automatic detection system. In the room one or several detectors are placed connected to the detection system which gives an alarm in the central control room. The failure probability of the system has been calculated by Jörud et al. [6]. If the detection system work the automatic sprinkler system (if there is any installed) will start. If the detection system does not work no quenching (automatic or manual) is possible. The result from the analysis of detection is given as a failure probability of the detection system.

If there is an automatic sprinkler system installed in the room this is being considered in the analysis. The analysis of the sprinkler system is a part of the fault tree model in the PSA. This is necessary since there is a functional dependency of the sprinkler system (uses one water tank for several rooms etc.). If the sprinkler system start an estimation of whether the sprinkler system will extinguish the fire or not is made. The estimation is made based on where the spray nozzles are situated in the room and also by expert judgement. It is important to remember that the sprinkler system is designed to limit the fire in the room not to extinguish it. If the automatic sprinkler system works and it extinguishes the fire this is considered as sequence where no components is damaged. The result of the analysis of the automatic sprinkler system is a failure probability of the system (including extinction of the fire).

The analysis of the course of event is performed by use of the computer tool HAZARD [7]. The evolution of fire is modelled in the program and variation of the physical parameters of interest (e.g. volume of ventilation, rate of growth, fire load) gives an overall picture of the course of events. Estimations and assessments is made by the analysis team of the results whether there is a possibility of having a fire which will destroy all components in the room (a critical condition) and at which point of time this will occur. The final result will then be a table giving the fraction probability of fires leading to non-critical conditions and also the fraction probabilities of different time points when the room has gone to a critical condition. This is discussed in detail in the following section.
The analysis of the course of event gives a time frame for when the room will be critical, i.e. it give the time frame for possible manual action. Extinction of the fire by the personnel is modelled in the PSA. The personnel at the NPP's trained for the situation and the modelling of the event is done by estimating the time for the personnel to detect, to collect the right equipment, to find the room and finally to extinguish the fire. Different failure probabilities are used for different points of time. The result of the analysis is then a failure probability of manual extinction.

The impact of the components in the room is estimation based on literature and expert judgement. Since most components are cables and electrical equipment the dominating failure mode is high temperature. For cables the temperature level of 200 °C is used. This has been shown by several references as a proper level (e.g. reference [8]. For electrical components (circuits, breakers etc.) the temperature level of 70 °C is used. This has been shown in several tests, e.g. reference [9]. Note that this is only for short-term periods, e.g. the electrical equipment will not withstand that temperature for long periods of time.

Finally, a verbal description of the whole sequence is done. This describes if the automatic scram system is activated in the NPP and which scram signal is triggered. Also which system are affected by the fire, how the fire is detected, which actions the personnel takes and also what the final impact of the NPP is.

The analysis of fire could be described as below. For each function below a failure frequency or failure probability is given. The event tree in figure 2 illustrates the final result.

<table>
<thead>
<tr>
<th>Initiating event</th>
<th>Detection</th>
<th>Automatic sprinkling</th>
<th>Effect of sprinkling</th>
<th>Sequences leading to critical conditions</th>
<th>Manual Actions</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>OK</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>OK</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>OK</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>CD</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>OK</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>OK</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>CD</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>OK</td>
<td></td>
</tr>
</tbody>
</table>

Figure 2: General event tree of the fire analysis
About 20 to 100 different fire events are modelled for each room. It is wise to vary as few input parameters as possible to keep the amount of calculations to a minimum. The choices of parameters are based on references, expert judgements and/or sensitivity calculations. When we are interested of the temperature in the smoke layer, the parameters are most often chosen to be:

- the fire growth (kW/s²)
- the maximum rate of heat release (kW)
- the ventilation

Also the position of the fire from the floor level is an important parameter.

As an example different growth parameters and maximum rate of heat releases can be categorised dependent of different fire loads in cable spreading rooms or ducts.

<table>
<thead>
<tr>
<th>Fire Load (MJ/m² surrounding area)</th>
<th>Category</th>
</tr>
</thead>
<tbody>
<tr>
<td>0 - 100</td>
<td>0</td>
</tr>
<tr>
<td>100 - 200</td>
<td>I</td>
</tr>
<tr>
<td>200 - 300</td>
<td>II</td>
</tr>
<tr>
<td>300 -</td>
<td>III</td>
</tr>
<tr>
<td>Vertical spread</td>
<td>IV</td>
</tr>
</tbody>
</table>

**Figure 3: Categorisation of the rooms based on the fire load.**

For instance in rooms belonging to category I some lower alpha parameters are chosen. In figure 4 below a comparison between the fire growth for rooms of category I and IV is given.

**Category 0**

<table>
<thead>
<tr>
<th>Fire growth (kW/s²)</th>
<th>Probability</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0003</td>
<td>0.05</td>
</tr>
<tr>
<td>0.0006</td>
<td>0.40</td>
</tr>
<tr>
<td>0.0029</td>
<td>0.30</td>
</tr>
<tr>
<td>0.0117</td>
<td>0.20</td>
</tr>
<tr>
<td>0.047</td>
<td>0.05</td>
</tr>
</tbody>
</table>

**Category IV**

<table>
<thead>
<tr>
<th>Fire growth (kW/s²)</th>
<th>Probability</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0003</td>
<td>0.05</td>
</tr>
<tr>
<td>0.0029</td>
<td>0.15</td>
</tr>
<tr>
<td>0.0080</td>
<td>0.30</td>
</tr>
<tr>
<td>0.020</td>
<td>0.30</td>
</tr>
<tr>
<td>0.047</td>
<td>0.15</td>
</tr>
<tr>
<td>0.070</td>
<td>0.05</td>
</tr>
</tbody>
</table>

**Figure 4: Illustration of comparison between two different categories.**
Figure 5 shows how different parameters are combined. The figures are only taken as an example of a room. The number of calculations needed for this room is at least $3 \times 2 \times 2 = 12$.

![Diagram showing parameters with values for Q1, Q2, Q3, α1, α2, P1, and P2]

**Figure 5: Probability distribution of different parameters, needed when modelling the fire.**

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Probability</th>
<th>Critical or Non Critical</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.030</td>
<td>NC</td>
</tr>
<tr>
<td>2</td>
<td>0.030</td>
<td>NC</td>
</tr>
<tr>
<td>3</td>
<td>0.120</td>
<td>C (600 s)</td>
</tr>
<tr>
<td>4</td>
<td>0.120</td>
<td>NC</td>
</tr>
<tr>
<td>5</td>
<td>0.050</td>
<td>C (780 s)</td>
</tr>
<tr>
<td>6</td>
<td>0.050</td>
<td>C (1320 s)</td>
</tr>
<tr>
<td>7</td>
<td>0.200</td>
<td>NC</td>
</tr>
<tr>
<td>8</td>
<td>0.200</td>
<td>NC</td>
</tr>
<tr>
<td>9</td>
<td>0.050</td>
<td>NC</td>
</tr>
<tr>
<td>10</td>
<td>0.050</td>
<td>NC</td>
</tr>
<tr>
<td>11</td>
<td>0.200</td>
<td>NC</td>
</tr>
<tr>
<td>12</td>
<td>0.200</td>
<td>NC</td>
</tr>
</tbody>
</table>

**Figure 6: Output from the probability calculations**

A certain amount of the calculations will lead to critical temperatures in the room, when using the different combinations shown above. This figure will be used in the event tree item "Sequences leading to critical conditions". If three scenarios will lead to critical condition the probability of 25% will be used in the event tree.

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Critical</th>
<th>Conditional Probability</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>0.120</td>
<td>.55</td>
</tr>
<tr>
<td>5</td>
<td>0.050</td>
<td>0.227</td>
</tr>
<tr>
<td>6</td>
<td>0.050</td>
<td>0.227</td>
</tr>
</tbody>
</table>

**Figure 7: Summarisation of the critical sequences**

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To enable modelling of manual fire fighting ability before critical conditions are reached, the time frame is needed. An example of output from the fire analysis is shown in the following table:

<table>
<thead>
<tr>
<th>Critical time (s)</th>
<th>Probability given a critical fire</th>
<th>Accumulated probability</th>
</tr>
</thead>
<tbody>
<tr>
<td>600</td>
<td>0.55</td>
<td>0.55</td>
</tr>
<tr>
<td>780</td>
<td>0.23</td>
<td>0.78</td>
</tr>
<tr>
<td>1320</td>
<td>0.23</td>
<td>1.0</td>
</tr>
</tbody>
</table>

Figure 8: Input to the human reliability analysis

5 Discussion

One the main objectives in the updated Level 1 PSA for the Barsebäck NPP was to have realistic models of the external events fire and internal flooding. By use of the method described in this paper this has been achieved. The final result of the PSA for Barsebäck NPP still identifies fire for some rooms as being among the dominating sequences of the total core damage frequency. However, the result is not a lopsided picture since there is also several internal events as well.

The experience from the modelling work is that the method is relatively easy to use and easy to understand for different categories of engineers performing reactor safety analyses. The main advantages are:

- No time consuming effort is needed in performing the analyses
- Gives a good platform for discussions between different categories of engineers. It also involves the personnel at the NPP in a natural way in the survey of the rooms.
- Uses experienced models in the analysis process.

There is of course some disadvantages of the model. It is important to remember that the method makes several simplifications in the modelling. The main disadvantages are:

- The impact on the components is considered as a group. It would be and an improvement if the failure probability could be estimated on individual basis.
- Since the main part of the rooms in the NPP is not considered there is a possibility that some rooms which would be among the dominating sequences of the core damage frequency are missing. Therefore, the method requires continuos efforts of analysis of rooms.

The final remark is that analysis of external events such as fire and internal flooding requires use of both deterministic and probabilistic analysis tools. The combination of the two approaches gives the NPP a tool, which combines the straightforward, objective result of the deterministic analysis with the subjective estimation of the probabilistic analysis.
6 References

7. HAZARD: Fire Hazard Assessment Method; Handbook 1 & 2; NIST; Building and Fire Research Laboratory; USA.
SPREAD SIMULATION OF HORIZONTAL CABLE TRAY FIRE

The experimental setup of horizontal cable fire [1] has been simulated by using a general 3D flow code FLUENT. The flame front propagation and rate of heat release have been calculated with a simple model which has been developed at VTT. The experiment is instrumented in order to get some information about temperature, velocity and rate of heat release for comparison.

The aim of this work is to implement the fire spread model to FLUENT-code (originally developed for PHOENICS-code) and to improve the model from an earlier version. This work is connected with the work done in fire safety part of the Finnish RATU research programme (Structural integrity of nuclear power plants).

Estimation of the heat release rate is the most important aspect when we try to make the fire simulations more accurate. A simple fire spread model has been used earlier for a cable tray simulation. It is possible to make this model more accurate by taking into account the environment in which the burning part of the cable tray is located. In this model the flame spread rate and burning rate of the ignited material depend both on the incident radiative heat flux to the surface. This is believed to improve the temporal behaviour of simulated fire. In the early model the burning rate was constant which, although giving qualitatively good results, led to a too rapid fire spread.

The response of the sprinkler heads due to high temperature is also considered on a post-processing basis.

The model development is based on cone calorimeter experiments which have been performed at VTT Fire Technology. Also other sources have been used for estimating ignition of objects and flame front spread.

THE EXPERIMENTAL SETUP AND COMPUTATIONAL GRID

The experimental setup consists of five horizontal cable trays in a 3.6 m long room with a cross section of 2.4 x 2.4 m². Three of the trays are on one side of the room and two on the opposite side. There is an open door at the end of the room. The lowest tray is ignited in the middle from below. After that the flame front proceeds on the lowest tray, further to the upper trays and finally to the trays at the opposite wall.
Flow velocity and (gas) temperature have been measured in several locations in the room. Flow indicators (F1–F4) are located at the doorway. Temperature is measured with unshielded thermocouples at three rakes (6 in each) and in several locations on cable surfaces. Also some sprinkler heads were placed near the ceiling. Some of the measuring points and general room layout is presented in figure 1.

![Diagram of a room with measurement points](image)

Figure 1. General layout of the cable fire test room. Some measurement points are indicated. Flow indicators F1–F4, thermocouple rakes TRx, thermocouple near sprinkler heads TS, cable trays C1–C5, heat radiation probe R.

The horizontal cable tray case is simulated with FLUENT CFD-code version 4.4.8. The computational grid has 41,650 cells of which 26,588 are in the room itself and 9,248 outside the room (in the fluid part, the rest of the cells were used for boundary conditions and structures). The cell distribution is somewhat uneconomical due to the structured computational grid used in this FLUENT version. The volume outside the room is taken into simulation only to get a proper boundary condition at the room doorway. The grid is presented in figure 2.

All the five cable trays in the room have been described in the grid. Both the flow and temperature field in fluid are calculated in the CFD-code. For turbulence, the k-ε model is used with additional source terms for buoyancy. Burning in gas phase is calculated with a single step reaction controlled by mixing of fuel and oxygen. Heat radiation is calculated by discrete transfer method. Local absorption of gas is calculated by weighted sum of grey gas method and soot effects by Tessner two equation model both provided in FLUENT code [2].

**FIRE SPREAD MODEL**

The fire model applied consists of three parts: estimate for ignition, estimate for spread of flame front and estimate for the burning rate of the ignited fuel.
For the simulation, critical incident flux was selected as the ignition criteria. The threshold value 12-15 kW/m² is applied. The surface temperature of cables is calculated by assuming adiabatic surface. Other possibilities are discussed in [3].

Figure 2. Computational grid used for model development. Number of computational (fluid) cells in the room itself is 26,588. Ceiling and wall are removed from figure for clarity.
In this work the flame spread modelling is based on the old data [4] as well as experimental data from the full scale fire test [1]. In figure 3 the flame front velocity is presented as a function of the incident heat flux. The figure is based on [4]. The propagating velocity at the external incident flux of 0 W/m² has been reduced from the earlier value to get the initial flame front velocity closer to the full scale experimental results in [1]. It is not possible to make modifications to the higher incident flux part of the previous data on the basis of these new experiments. The approximation fitted to the data is given below, where the incident flux $\dot{Q}_{\text{INC}}$ is given in [kW/m²]. The fit is used up to the value 160 cm/min. With higher fluxes the velocity is limited to this value.

$$v_\text{Front} = 2.914281 \exp(0.1680530 \dot{Q}_{\text{INC}}) \quad [\text{cm/min}]$$

(1)

The estimate of the burning rate of the ignited surface has been revised from the earlier version [5]. In the present version the fuel release rate depends on the local incident radiative flux at the surface as well as the temporal behaviour. The experiments have been performed with cone calorimeter applying a constant incident flux. In a fire situation the radiative flux to the surface changes over time. In spite of this difference the temporal behaviour is estimated on quasi-steady basis. This discrepancy has to be recognized and hopefully removed in the future.

The burning rate of the cable material (MMJ-cable, diameter 13 mm, $5 \times 2.5$ mm²) at different incident fluxes is presented in figure 4. The approximation of the experimental curve is presented in the same figure. The shape function of the approximation is the same in all three cases. Only the duration and peak values have been scaled. The experimental data is from [6] and additional data sheets from VTT Fire Technology.

A polynomial fit has been drawn on the experimental curves in figure 4. The shape function of the fit is same in all three cases. The shape function has the form

$$RHR_{D}(t_D) = -2.130176 \cdot 10^{3} t_D^{8} + 9.278737 \cdot 10^{3} t_D^{7} - 1.668545 \cdot 10^{4} t_D^{6}$$

$$+ 1.593850 \cdot 10^{2} t_D^{5} - 8.649602 \cdot 10^{3} t_D^{4} + 2.639464 \cdot 10^{3} t_D^{3}$$

$$- 4.225230 \cdot 10^{2} t_D^{2} + 3.103293 \cdot 10^{1} t_D + 1.774149 \cdot 10^{2}$$

(2)

where $t_D$ is the dimensionless time and $RHR_D$ is the non-dimensional rate of heat release. The values were normalized with the total burning time and the peak heat release rate respectively.
The peak value of heat release rate and the total burning time together with total energy release are presented in table 1 as a function of the incident heat flux. The boundary values are used outside the experimental range.

Table 1. Scaling parameters for MMJ-cable heat release rate at different incident fluxes.

<table>
<thead>
<tr>
<th>Incident flux [kW/m²]</th>
<th>RHR_{max} [kW/m²]</th>
<th>Burning time [s]</th>
<th>Total energy [MJ/m²]</th>
</tr>
</thead>
<tbody>
<tr>
<td>15</td>
<td>160</td>
<td>750</td>
<td>70</td>
</tr>
<tr>
<td>25</td>
<td>200</td>
<td>1100</td>
<td>125</td>
</tr>
<tr>
<td>35</td>
<td>425</td>
<td>590</td>
<td>144</td>
</tr>
</tbody>
</table>

![Graph](image1)

![Graph](image2)
Figure 4. Heat release rate of MMJ cable material at three different incident fluxes. Incident flux a) 15 kW/m², b) 25 kW/m², c) 35 kW/m². Experimental values are from [6].

SIMULATION

The initial condition is calculated by setting the temperature to the initial structure value (21–23 °C) and integrating the flow for a while. The boundaries under the hood are taken as free pressure inlets with incoming temperature of 17 °C.

The time dependent simulation is started from initial conditions. The gas burner of 1.9 kW is placed in the middle of the lowest cable tray C1. Simulation proceeds 2–3 minutes with the gas burner as the only fuel source. After that, tray C1 is ignited, i.e. flame fronts are set to the initial locations in the middle of the tray.

Flame propagates slowly about 10 minutes before the next tray C2 is ignited. In the simulation the ignition criteria should take care of the ignition of the next object, but the emissivity of the quite small flame area is low, leading to a too low radiative incident flux on the surfaces. It appears that the used soot model is not suitable for fire applications (the model has been developed mainly for combustion of gas phase fuels). Thus, in some cases tray C2 is ignited by user decision.

The heat release rate is also relatively small after 10 minutes. The average cell size is large compared to the reactive flame volume which leads to a low average temperature in each cell. Thus radiative emission stays low also for this reason. In the later stage the grid size is not so critical, because the reactive volume is larger.

Simulation with MMJ cable data

The first simulation was performed using the material properties of MMJ-type cable, which is commonly used in Finnish nuclear power plants. Tray C1 was ignited after 2 minutes. Then the simulation proceeded by itself to about 10 minutes. At this stage the
incident heat flux on the tray C2 was not high enough to ignite the tray. The soot model does not produce soot in the flame area, which is in controversy with the experiments. This can be a consequence of the coarse computational grid but most obviously the soot forming mechanism of polymer combustion differs from the mechanism taking place in combustion of gaseous fuels. According to the default model parameters soot production starts after temperature is more than 1 100 K. The parameters were altered so that the process is activated at a lower temperature.

In the first run the tray was ignited (with modified soot parameters) according to the model after 12 minutes and 40 seconds. The heat release rate at that time was about 30 kW. After this the two trays interfere with each other which accelerates flame front velocity and fuel release. Heat release rate of fuel is rapidly increasing. The development can be seen in figure 5 together with the experimental values. The heat release rate increases very quickly and becomes soon larger than the measured maximum value in the experiment. The maximum value reached is about 1020 kW which is about twice the experimental value. The duration of fire is also shorter than in the experiments. Together this leads to almost equal total heat release (± 10%).

In the simulation all trays were ignited in the first phase of fire whereas in the experiments the heat release rate was considerably lowered before the trays C4 and C5 were ignited as a consequence of a smoke gas deflagration [1]. This lead to about 200–250 kW increase in the heat release rate. The ignition of trays C4 and C5 does not explain the large difference in the maximum values of rates. It is obvious that the MMJ-cable material properties differ from the MMO-cable which was the actual cable type used in the experiments.
Figure 5. Heat release rate of cable tray experiment and simulations. Simulations made with MMJ cable data and MMO cable data. The cases MMO-v3b and MMO-v3d differ so that in the latter case ignition of trays at the opposite wall is restricted.

Simulation with MMO cable data

The material properties of MMO-cable used in the full scale experiment are known from one cone calorimeter test only. Actually there were three different MMO-cable types used in the experiment [1]. The cables have different number of conductors: MMO-A with 4x1.0 mm² diameter 10 mm, MMO-A with 7x1.0 mm² diameter 12 mm and MMO-A 12x1.0 mm² diameter 16 mm. The cone calorimeter test is performed with incident flux of 50 kW/m² with cable MMO-A 7x1.0 mm². According to this single measurement the MMO-cable has a smaller heat release rate than the MMJ-cable.

Due to the limited cone calorimeter data the model parameters for burning rate are modified from MMJ-version to MMO-version using the existing information as a guideline. Thus the further simulations are not based on solid experimental data. However, it is hoped that the procedure reveals the connections between material properties and heat release rate. The values have to be verified by later experiments. The modified MMO-cable properties are presented in figure 6.

![Graph](image)

Figure 6. Heat release rate as a function of time at different incident fluxes, MMO-cable. The curve of 50 kW/m² is the only experimental curve.

The simulations with the new material data were performed in a similar manner as in the first case. Ignition of tray C1 was defined at 3 minutes. The tray C2 was ignited at 10 minutes (by user definition). The heat transfer from flame was not high enough to ignite the tray. The maximum incident flux on tray C2 was 4.1 kW/m². The heat release rate at 10 minutes was about 28 kW.
After igniting tray C2 by user operation the simulation proceeds by itself. Tray C3 was ignited according to the simulation at 13 minutes 5 seconds. The heat release rate at that time was about 67 kW. Unfortunately the trays C4 and C5 did also ignite according to the ignition criteria. However, in this case the criteria of critical incident flux alone may not be well established. Due to the fire the incident radiative flux and gas temperature at the trays are high, but oxygen mass fraction is only about 7% (from original 23% in normal atmosphere). In the experiment the ignition of trays C4 and C5 took place after a disturbance in stratification and deflagration after that.

Trays C4 and C5 ignite just before 15 minutes when the heat release rate is about 320 kW. The maximum heat release rate is 620 kW (with 5 trays) which is about 100 kW more than the experimental value (with 3 trays). The development of heat release rate is presented in figure 5 labelled as case MMO-v3b.

The case MMO-v3d in figure 5 is otherwise the same as the previous case, but ignition of trays C4 and C5 is hampered. In this case the maximum heat release rate is about 430 kW.
COMPARISON WITH THE EXPERIMENT

The measured velocity and calculated velocity component normal to the doorway are presented in figure 7. The middle points F2 and F3 seem to agree quite well with the experimental values. The too quick rise in the heat release rate can be seen in the calculated velocities after 15 minutes especially in location F3. However, the calculated maximum inflow and outflow velocities in the lower (F1) and upper part (F4) of the doorway are higher than the measured values. It is possible that the overshoot is partly due to the limitations of the k-ε turbulence model. The measurement location F4 is at height 1.8 m when the door height is 2.0 m. The calculated value is from the nearest computational cell which is at the height of 1.805 m. Especially in F4 the velocity differences rise at least partly from the fact that the calculated pressure differences have not been corrected with density differences when velocity has been calculated from the measurement. There was no thermocouple in the same location with the pressure probe. Taking a temperature value from location TR16 we get the modified curve presented in frame F4 of figure 7, which shows the right order of magnitude for velocity.

![Graphs showing velocity measurements and calculations](image)

Figure 7. Measured (thin line) and calculated velocity component normal to the doorway as a function of time. Experimental velocity has been calculated from the measured pressure difference by assuming constant density. Correction to this has been adopted in frame F4 using temperature TR16 from figure 8 for density calculation.
The measured and calculated values of temperature in the thermocouple rake TR1x are presented in figure 8. The calculated values at the upper part of the rake (TR16–TR14) are of the same magnitude as the experimental values. However, the calculated temperature tends to rise more quickly than the experimental value. In the lower part of the rake (TR13–TR11) the maximum calculated values are essentially lower than the experimental ones. The lowest location TR11 is at the height of 0.67 m from the floor and this should be in the fresh inflow area according to the flow measurement. It is possible that the peak seen in the experimental values comes from radiative effect to the unshielded thermocouples.

Figure 8. Measured (thin line) and calculated temperature at locations TR11–TR16 at the symmetry plane of the room.
For sprinkler head response the temperature change is often calculated from equation

\begin{equation}
\frac{dT_e(t)}{dt} = \frac{\sqrt{u(t)}}{RTI} \left( T_{gas}(t) - T_e(t) \right)
\end{equation}

where \( T_{gas} \) is temperature of gas, \( T_e \) is temperature of the sprinkler head sensor, \( u \) is the gas velocity and \( RTI \) is the response time index [(m/s)\(^{1/2}\)]. The \( RTI \) is usually 50-80 (m/s)\(^{1/2}\) for fast sprinkler heads and of the order of 300 (m/s)\(^{1/2}\) for slow heads. The actual properties of the sprinkler heads used in the experiment are not known. Two of the heads are fast and two slow. The details are given in [1]. The estimate of sprinkler head response time is calculated from equation (3) using both the experimental and calculated time history (table 2). The \( RTI \) value 80 (m/s)\(^{1/2}\) is used for fast heads. The value 220 (m/s)\(^{1/2}\) for slow heads gives a response time similar to the experiments when experimental temperature history is used. The effect of the high calculated temperature is seen as the too early response time.

**Table 2:** Response time of sprinkler heads. For fast sprinkler heads the first experimental activation time is taken and for slow heads the last. The actual \( RTI \) values are not known.

<table>
<thead>
<tr>
<th></th>
<th>FAST, ( RTI ) 80 (m/s)(^{1/2})</th>
<th>SLOW, ( RTI ) 220 (m/s)(^{1/2})</th>
</tr>
</thead>
<tbody>
<tr>
<td>experimental</td>
<td>12,2 min</td>
<td>13,8 min</td>
</tr>
<tr>
<td>estimated from measured temperature history</td>
<td>12 min</td>
<td>13,8 min</td>
</tr>
<tr>
<td>estimated from calculated temperature history</td>
<td>7,5 min</td>
<td>9,7 min</td>
</tr>
</tbody>
</table>

The incident radiative heat flux on the floor of the room is presented in figure 9, where the simulated values are from the case MMO-v3d in which ignition of the trays C4 and C5 were hampered. The predicted incident flux is higher than the measured even though the maximum predicted rate of the heat release is lower than the measured one. The difference rises probably from the unsuitable soot modelling and the corresponding differences in the local absorption coefficient. In general the local absorption coefficient is too low in the hot layer thus exposing the flames too well to the environment. There are

![Figure 9. Incident radiative heat flux on the middle of the floor. Simulated values (●) are from the case MMO-v3d.](image)
other soot models available though not yet tested in this simulation. In the standard FLUENT code it is, however, not possible to build an own soot model so that it would co-operate with the existing procedures of absorption calculations.

SUMMARY

The experiment of horizontal cable fire test has been simulated by using a general 3D flow code FLUENT. The flame front propagation and rate of heat release have been calculated with a simple model which has been developed at VTT.

Comparison of the simulated and experimental results shows that the general behaviour of the simulation is similar to the experiment. However, the used model requires specific material properties for input and the material properties known for one cable can’t be generalised for other cable materials due to large differences in the rate of heat release of different cable types.

The first simulation with MMJ-cable data represents the status of ‘blind simulation’. In this case the material properties were deduced from the small scale cone calorimeter data of MMJ-cable. The actual cable used in the experiment was MMO which has different combustion properties. In the first simulation the maximum predicted rate of heat release was about twice the experimentally observed maximum rate.

From the extremely scarce cone calorimeter data of MMO-cable it was obvious that the heat release rate of this cable under the same conditions is lower. The model was changed accordingly and a new simulation gave a better result when the maximum rate of heat release is considered. The proper justification can be done only after the material properties of the MMO-cable are known and included in the simulation.

During the simulation the proper ignition criteria, which would trigger the ignition of a new object, was not found. The working hypothesis of critical incident radiative flux was not practical mainly due to the limited resolution at the early stage when fire intensity is small compared to the computational cells. The low heat rate density leads to low average cell temperature and to low radiative fluxes. The second tray was ignited by user definition in the simulation. The rest of trays were ignited according to the model.

The extremely complex experimental setup revealed also the limitations of the combustion model used in the simulation. The cable material on the trays on the opposite wall was not ignited in the experiment even though the cable insulation was already melting and dropping to the floor. The reason for this is probably the low oxygen concentration in the hot layer. Ignition in the experiment took place only after a small disturbance brought fresh air to the hot layer. At that stage the ignition started as a smoke gas deflagration and lead to a new rise in the rate of heat release. In the simulation the low oxygen combustion is not modelled properly although the oxygen concentration is calculated.
The simulation gives promising results which should be re-generated with the actual material data of MMO-cable. The proper justification can be done only after the material properties of the MMO-cable are known and included in the simulation. The material data has been recently produced and the new simulation is taking place in the near future.

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Fire Modelling in IPSN Computer Codes

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Abstract

One of the main objectives of fire research is to provide safety analysis with computer codes which are well qualified in the field of interest for nuclear plant safety. For that, there is a strong connection between experiments and modelling. This means that experimental programs must have two aspects: analytical tests to assess the models and global tests to validate the completeness of the modelling and the right coupling between the models. A two year approach is classically applied by IPSN for the development of computer codes in the domain of fire studies. The rather simple modelling approach of the FLAMME_S code is mainly designed for the present safety evaluations since it will allow to perform engineering calculations with a low CPU time. The mechanistic modelling of the 3-D ISIS code is in particular aimed at providing a research tool able to overcome the limitations of the simple tool and to extend the configurations accessible to the simulation. In the domain of application of the FLAMME_S code, the use of ISIS will allow to assess or to develop global correlations or models which will be implemented in the simple code.

1. Introduction

The damages that a fire can cause to nuclear plants with possible release of radioactive substances to the environment make fire safety studies of a primary importance in risks assessment. Generally, nuclear plants (nuclear power plants NPP and nuclear reprocessing plants NRP) are ventilated in order to insure a underpressure in rooms. This underpressure associated to the filtering devices located into the ventilation networks must enable to confine radioactive substances inside the plant. The design of nuclear plants makes the study of fire development and propagation very complex since there is a strong interaction between fire and ventilation. This survey is made even more difficult by the great number of possible fuels that can be found in nuclear plants.

One of the main objectives of fire research is to provide safety analysis with computer codes which are well qualified in the field of interest for nuclear plant safety. Fire scenarios which are taken into account involve
one or several different fuels, into one or several rooms which are connected between them or with the outside through doors or ventilation network [1]. In the case of a compartment fire, computer codes have to quantify specific phenomena such as pressure increase at the earlier stage of the fire and pressure shutdown when fire burns out in order to determine the components behaviour (building concrete structure, filters ...).

In principle, computer codes modelling allows the simulation of a large range of scenarios but, performing calculation with a good confidence level requires firstly a knowledge of models limitations, and secondly a rigorous validation work. For that there is a strong connection between experiments and the modelling: analytical tests assess the models and global tests validate the completeness of the modelling and the right coupling between the models.

In this paper, the two computer codes which are involved in the IPSN research on fires are described: the physical models and the state of their development are presented.

A two-year approach is applied by IPSN for the development of computer codes in the domain of fire modelling. The rather simple modelling approach of the FLAMME_S code is mainly designed for the present safety assessment since it allows to perform engineering calculations with a low CPU time. The field modelling of the 3-D ISIS code is in particular aimed at providing a research tool able to overcome the limitations of the simple tool and to extend the configurations accessible to the simulation. In the domain of application of the FLAMME_S code, the use of ISIS will allow to assess or to develop global correlations or models implemented in this simple code. In addition, the more fundamental research carried out with ISIS allows to develop collaborations with universities, especially through PhD Degrees.

2. FLAMME_S

2.1 General features

The FLAMME_S computer code is being developed to be used in the assessment of fire risk in nuclear power plants and reprocessing plants. The main purpose of this program is to predict the development of a fire and the resulting conditions within a compartment in term of gas pressure, species concentrations and temperature (gases, walls, etc.). In a case of a ventilated room, temperature, species concentrations and flow rates of released gases at the openings are calculated. The Figure 1 shows the different kind of problems that should be treated with FLAMME_S: a fire occurs in a compartment that may be connected with other rooms by means of a ventilation network or a door; the elements (electronic cabinets, cable trays, etc.) contained in the enclosure can be damaged by the thermal stress due to the fire.
The development of this code started in 1993 and several new functionalities have enriched the first version of the program (multi-room model, coupling with a ventilation code). FLAMME_S has been validated from many large fire experiments achieved by the IPSN and from experimental data available in the literature.

2.2 Physical models and basic assumptions

FLAMME_S is based on a two zones model [[1], [3]]. The room is divided into two distinct but homogeneous zones. The upper layer contains the hot gases produced by the fire and the air entrained by the plume; these gases are floating over the « cold » gases of the lower layer as a result of the thermal stratification due to buoyancy (figure 2).
\[ m_i^{\text{cold}}, m_i^{\text{hot}} \text{ mass of the specie } i \text{ in the cold and hot zones} \]
\[ T_{\text{cold}}, T_{\text{hot}} \text{ mean temperature in the cold and hot zone} \]
\[ h \text{ height of the interface between the two zones} \]
\[ P \text{ room pressure} \]
\[ T(z) \text{ temperature along the vertical symmetry axis} \]
\[ V(z) \text{ flow rate along the vertical symmetry axis} \]
\[ D(z) \text{ characteristic diameter of the plume above the fire} \]

**Figure 3: Two zones model concept**

Each component (fuel, gaseous zone, walls, ventilation openings...) of the physical system corresponds to a physical and mathematical model and is characterized by its physical phase (solid, liquid or gaseous), its space discretization and the model describing its evolution.

For example, the mathematical model for the two gaseous zones of the room consists in writing the conservation equations of the mass of every chemical species and of the energy for every zone \( \xi \) (\( \xi \) varies from 1 to 2); each of them is considered like an homogeneous mixture with an uniform temperature \( T_\xi(t) \). The main variables are the mass of each chemical specie and the energy in each zone.

Let there be
\[ \xi \]
the set of chemical species in the zone \( \xi \),
\[ m_{\xi, i}(t) \]
the mass of the chemical species \( i \) in the zone \( \xi \) at the time \( t \),
\[ E_{\xi}(t) \]
the energy of the zone \( \xi \) at the time \( t \).
The two equations chosen are written:

\[
\frac{d}{dt} m_{t,t}(t) = \dot{m}_{t,t}(t) + \dot{m}_{t,t}(\text{comb})(t) + \dot{m}_{t,t}(\text{eva})(t) + \sum_{t' \neq t} \dot{m}_{t,t-t'}(t) - \sum_{t' \neq t} \dot{m}_{t,t-t'}(t)
\]

for \( t \in \mathcal{E}_t 

(9)

\[
\frac{d}{dt} E(t) = \dot{E}(t) + \dot{E}(\text{comb})(t) + \dot{E}(\text{eva})(t) - \sum_{t' \neq t} \dot{E}(t, t') + \sum_{t' \neq t} \dot{E}(t, t') + \phi^{\text{net}}(t)
\]

(10)

where

- \( \dot{m}_{t,t}(t) \) : mass flow rate (expressed in kg.s\(^{-1}\)) of the chemical species \( t \) brought by the ventilation at the time \( t \) in the zone \( t \),

- \( \dot{m}_{t,t}(\text{comb})(t) \) : mass flow rate (expressed in kg.s\(^{-1}\)) of the chemical species \( t \) "produced" by the combustion in the zone \( t \) at the time \( t \),

- \( \dot{m}_{t,t}(\text{eva})(t) \) : mass flow rate (expressed in kg.s\(^{-1}\)) of the chemical species \( t \) which appears in the gaseous form in the zone \( t \) at the time \( t \),

- \( \dot{m}_{t,t-t'}(t) \) : mass flow rate (expressed in kg.s\(^{-1}\)) of the chemical species \( t \) carried from the zone \( t \) to the zone \( t' \) of the same "room" at the time \( t \),

- \( \dot{E}(t) \) : energy flow rate (expressed in W) brought by the ventilation into the zone \( t \) at the time \( t \),

- \( \dot{E}(\text{comb})(t) \) : energy flow rate (expressed in W) brought by the combustion products into the zone \( t \) at the time \( t \),

- \( \dot{E}(\text{eva})(t) \) : energy flow rate (expressed in W) brought by the evaporation products into the zone \( t \) at the time \( t \),

- \( \dot{E}(t, t')(t) \) : energy flow rate (expressed in W) carried from the zone \( t \) to the zone \( t' \) of the same "room" at the time \( t \),

- \( \phi^{\text{net}}(t) \) : energy flux (expressed in W) - not linked to a mass transfer - received by the zone \( t \) at the time \( t \).

The mass and energy balance equations together with the perfect gas state equation enable the determination of the two layers temperatures, the pressure of the compartment and the height of the interface between the two zones. The source terms of these equations express the mass and energy exchanges between the two layers, between the compartment and the outside, between the gases and the walls and between the fire and its surroundings.
The thermal plume generated by the fire carries mass, energy and species from the lower layer into the upper one; these transfers and the decrease of the temperature along the plume axis are provided by one of two plume models (Gupta or Heskestad models ([4],[5])).

Flows through openings in natural convection or through the ventilation network in forced convection are simply deduced from the pressure difference between the enclosure and the outside by using Bernoulli's equation. The coupled ventilation code SIMEVENT ([6]) can be used to describe network components behaviour during the fire.

Radiative and convective heat transfers between the gas layers, the walls, the fires and the "targets" are calculated. The radiative energy leaving the flame is a fraction of the total heat release rate and is determined by assuming an energy point source located at the half of the flame height. The radiative fluxes toward the objects are calculated using shape factors and gases radiative characteristics which take into account semi transparent gases and solid particles concentrations. The heat diffusion equation is solved to follow the transient heating of walls and other objects due to the flame (radiation) and to the gases (radiation and convection); the convective term is a function of the temperature difference between the wall and the gases multiplied by a heat transfer coefficient deduced from the experiments.

No combustion model is presently available in FLAMME_S : a constant or transient value based on experimental results is provided in the input data file for the mass flow combustion rate \( \dot{m} \) (kg.s\(^{-1}\).m\(^{-2}\)). Although it is not calculated, this burning rate is limited by the oxygen amount in the air dragged by the plume and the extinction occurs when a given oxygen concentration is reached into the compartment. The combustion reaction is assumed constant during the fire and is defined by the user in terms of molar fraction of produced species.

### 2.3 Zones code limitations

From their basic principle, a zones code provides us only with mean values of gas temperature. For instance, the convective exchange of the gas with a target which is situated at a precise point of the room can not be estimated with satisfying accuracy with regard to damage time.

Other limitations of the zone codes are mainly connected to the validation domain of used correlations for flame height, plume temperature and plume flow rate.... Furthermore, complex and particular geometry (gallery, atrium...), flame/structure and flame/flame interactions, flame and plume evolution (flame tilt coming from ventilation effect) cannot be simulated.

Although much simplification is introduced, zone models as used in FLAMME_S lead to computer programs with low CPU time and reasonable accuracy of predictions in engineering applications. Thus, it allows to study a large number of scenarios for the same problem and to perform a wide sensitivity analysis. As mentioned before, such a code used in conjunction with probabilistic analysis should be able to achieve reliable assessment of fire risk.

### 3. ISIS

The intrinsic limitations of the zones-like code approach lead IPSN to take an interest in an more
detailed approach of the fire simulation which is based on the Navier-Stokes equations applied to turbulent flows with buoyancy effect at which it is added balance equations for chemical species and energy of the gaseous mixture. The 3D, field modelling, computer code, named ISIS is developed. A brief technical description of this code is presented and concluding remarks are proposed.

3.1 Technical Features

The ISIS code is based on the Computational Fluid Dynamics (CFD) theory [7,8]. This type of model solves the mass, momentum and energy balance equations. In order to determine the solution of these equations, the compartment is divided into a three-dimensional numerical grid of typically several ten of thousand of elementary control volumes filling the domain of interest.

Basically, the governing equations [9], [10], [11] are a set of Favre-mean, turbulence-modelled conservation equations for mass, momentum, energy and species concentration, together with transport equations for the turbulence variables k and ε. These equations can be conveniently written in terms of a general time-mean variable \( \Phi \) [12] :

\[
\frac{\partial}{\partial t} \left( \rho \Phi \right) + \frac{\partial}{\partial x_j} \left( \rho u_j \Phi \right) = \frac{\partial}{\partial x_j} \left( \Gamma_\Phi \frac{\partial \Phi}{\partial x_j} \right) + S_\Phi
\]

which \( \Phi \) stands for 1 (mass of the mixture), \( u_i \) (gas velocity), \( h \) (enthalpy), \( k \) and \( \epsilon \) (transport of turbulence kinetic energy and its dissipation rate). The \( \Gamma_\Phi \) term includes both molecular and turbulent diffusion effects and \( S_\Phi \) is a source term (combustion, heat transfer, ...). As for the combustion model, the balances for the mixture fraction \( f \) and its fluctuation intensities are used [10] and these variables obey the same transport equations as above. A special treatment for the flow close to walls is achieved by the well-established logarithmic laws [13]. Buoyancy effects are taken into account into the turbulence model and a simple radiation model is available.

The conservation equations previously described are solved using a discretization process in space and time on the numerical domain of interest. The discretized equations at each node of the numerical grid for each time step are written for a variable \( \Phi \) [12] as :

\[
a_p \Phi_p = \sum_M a_M \Phi_M + b
\]

where \( M \) refers to neighbouring nodes and \( b \) represents the influences of sources or sinks within the control volume around the node \( P \). A resulting set of algebraic equations is then obtained and solved using the usual methods for linear systems (ADI/TDMA, CGS, GMRES ...) [7]. The numerical algorithm is a version of the SIMPLE procedure proposed by Patankar [12,14].

3.2 State of development

A first version of the 3-D ISIS code is available since March 1999. This version will describe an inert turbulent flow with variable density. Standard combustion models are being implemented in 1999. Radiative and convective transfers models will be introduced in 2000. A qualification work will be done firstly with analytical tests and secondly with large scale experimental results.
From a computer and numerical point of view, a new software package, named PELICANS, is developed and used by several teams belonging to IPSN/DRS (CROCO and ISIS teams); the main objectives are to build shared mathematical or computer science libraries and to promote the exchanges between scientific teams working on physical problems asking in particular for the solution of the 2D or 3D Navier-Stokes equations. Furthermore, efficient numerical methods like multi-grids, multi-domain approaches and local meshing refinement will be implemented in ISIS in order to improve the performance of the code (CPU time, precision and robustness).

4. Completeness of the two codes

The two codes described above complete each other with regard to the advantages and disadvantages of their use: global (FLAMME_S) and local (ISIS-3D) approaches, simple (FLAMME_S) and complex (ISIS-3D) geometry, and CPU times very different.

FLAMME_S code allows a lot of calculation for a simple scenario with some parametric studies. ISIS 3D code will treat a complex scenario taking into account local phenomena. Furthermore, ISIS 3D code could propose and validate correlations for FLAMME_S code.

5. Qualification of the FLAMME_S computer code

A part of acquired knowledge concerning fire behaviour is integrated into the FLAMME_S computer code which is developed simultaneously to the experiments and their interpretation. But the modelling of some phenomena is not yet satisfactory (flame spread, plume in confined room ...) or does not exist (burning rate, combustion products and radioactive aerosols transfers within the room ...). The improvement and the development of some of these models will require specific accurate measurements during experiments: mass transfer and gas flow into the room (including the plume), heat transfers between the flame and the pool fire ...

5.1 Single room configurations

The main characteristics of the 17 tests used for the FLAMME_S code qualification are summarised below:

- oil fires surface
  - 0.03 and 0.06 m² in a 5 m³ room either closed or under forced ventilation,
  - 1 to 2 m² in a 400 m³ room under forced ventilation,
  - 5 m² in a 2000 m³ room under natural ventilation;
- solvent fires surface
  - 1 m² in a 100 m³ room under natural ventilation,
  - 1 m² in a 400 m³ room and a 3600 m³ room under forced ventilation,
  - 1 to 5 m² in a 2000 m³ room under natural ventilation,
  - 20 m² in a 3600 m³ room under forced ventilation;
- A cable fire with a tray of four rows of ten cables in a horizontal position in a room of 400 m² under forced ventilation.

For each test, more than forty experimental variables were used to estimate the code ability to calculate the thermal consequences of a given fire. As an example, diagrams are presented in the Figure 3 for the LIC 2.3 test. This test was performed in the PLUTON vessel (400 m³) [[1]] with a 1 m² liquid pool of TPH/TBP (TriButylPhosphat/Hydrogenated TetraPropylene) and an initial ventilation flow rate of 1200 m³.h⁻¹.

The FLAMME_S code assessment carried out up to now allow us to have a high confidence level on the results of the code application on configurations and scenario conditions belonging to the present domain of qualification. The application outside these limits is possible but requires a knowledge of the models used and of their limitations.

It is important to emphasise that the input data of the code such as the fire heat release rate and the thermal exchange coefficients have a significant influence on the numerical results. A user's guide recommends some choices for some of the parameters to be provided.

Figure 3: Qualification of FLAMME_S code - LIC 2.3 test

- Qualification de FLAMME_S (Version A3.1) : Essai LIC 2.3
- Temperature of gas in the PLUTON

\[\text{gas temperature}\]
A: upper zone; B: lower zone; C mean value
\[\text{D E F G H: experimental values}\]

\[\text{Pressure in the PLUTON vessel}\]
A: calculated value; B: experimental value

\[\text{2 3 4}\]
### Oxygen concentration

_upper zone_: A: calculated value; B: experimental value

_lower zone_: C: calculated value; D: experimental value

### CO₂ concentration

_upper zone_: A: calculated value; B: experimental value

_lower zone_: C: calculated value; D: experimental value

---

**East wall temperature**

**Gas outlet flow rate**
5.2 Multi room configurations

The multi room version of FLAMME_S (version 2) has been qualified using the results of the tests performed by Cooper and al. [[15]]. Nineteen tests involved two or three rooms (one or two rooms with a corridor) connected each other by doors. The heat release rate varies from 25 to 300 kW, combustion rating is stationary or transient and the fire duration is equal to or less than 10 minutes.

For these situations, the FLAMME_S code gives the good tendencies and satisfactory levels with regard to the pressure difference between the burning room and the corridor, gas temperature, heat fraction absorbed by walls $\lambda$ and zones interface height $H$ in the three rooms (Figure 4). Furthermore, comparison with Harward VI code results for these cases had shown that the two codes have the same behaviour.

Figure 4 - Full corridor and lobby - HRR = 225 kW

Mean temperature
lobby : ----- flamme_S, ○ exp.

$\lambda$
heat fraction absorbed by walls
— flamme_S, ■ exp.

zones interface height
5.3 Future developments

The next steps for extending the qualification field of the code will be based on IPSN tests performed in 1997 and planned during 1999-2000 [11]:

- tests involving a liquid pool fire close to a wall or in a corner,
- multi-room tests with rooms connected by doors or by a ventilation network,
- electric or electronic cabinets fires,
- solid fuels in a vertical position.

Simultaneously, improvements of physical models will be done, in priority concerning an accurate plume model for a confined room and a combustion model.

6. The codes in support to safety analysis

The FLAMME_S code will still remain for a long time the code used for safety analysis. The development and qualification of FLAMME_S will be mainly based on the new experimental results obtained from the programs carried out either in support to the safety analysis or in the framework of basic researches.

Besides, it is proposed to write a new code based on the same rather simple level of fire modelling and coupled to a ventilation code. This new code will be designed to be flexible, user friendly and low CPU time consuming. It will be based on the present qualified modelling of the FLAMME_S code for the fire behaviour and of the SIMEVENT code for the ventilation behaviour. The assessment matrix of this new code will be very extended.

Although simple, this new fire-ventilation platform has to allow in a rather simple way the
implementation of new physical models and the coupling of other more mechanistic computer codes, such as ISIS for a fire in one room and SOPHAEROS (from the ASTEC code) for the transport of aerosols through the ventilation.

The specifications of this new fire-ventilation platform will be written in 1999 and the development is foreseen to start in 2000.

7. Conclusion

The fire is a main industrial risk which can lead to important damages on the installations, the environment and the persons. A special attention has to be put on the analysis of the fire risk in nuclear installations since a fire could lead to the release of radioactive materials inside the installation and even outside, into the environment.

The large amount of experimental results available from the tests carried out by IPSN for more than ten years in the area of the « classical » fires, had led to significant progress in code development including phenomena understanding and code qualification. Today, the computer code FLAMME_S assessed in particular on the IPSN experimental data is used to carried out safety analysis, especially in the framework of the underway fire PSA related to French PWR 900 MW. Nevertheless, a lot of models still need a complementary qualification and some phenomena exhibited during experiments are either not yet modelled or not well understood or show the limitation of the present modelling.

It has to be highlighted that the future research will be also carried out in more fundamental aspects related to the modelling of the 3-D multi field computer code ISIS. The local approach of 3-D ISIS code will complete the global approach of the zones code FLAMME_S. Furthermore, experiments will be needed with specific measurements to ISIS code qualification. The development of tight collaborations with universities and other research organisations working on the fire phenomenon will be necessary to succeed in this area.

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Development of Fire Severity Factor

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1. Introduction
In consequence of the trail application of fire PSA methodology, which has been used in some of US IPEEE, to a Japanese PWR plant, the influences of the severity factors on the core damage frequency are large, and the severity factors used seem too conservative to evaluate Japanese LWR plants. Therefore, it is necessary to make the Japanese specific severity factors based on fire events at Japanese plants. However, there are not enough fire events at Japanese plants to do it. Therefore, using fire experiments carried out in Japan, $\alpha$-FLOW code that is the Computational Fluid Dynamics code and COMPBRN-IIe code, we are ongoing to make the severity factors analytically, which can be used in fire PSAs for Japanese LWR plants.

2. Background
In NUPEC, a fire PSA methodology introduced from one of US IPEEE methodologies has been developed on consignment of the Ministry of International Trade and Industry since 1992. It is composed of three steps, namely "Spatial interaction analysis", "Screening analysis", and "Detailed analysis". In the spatial interaction analysis, the fire scenarios are made based on plant information. In the screening analysis, safety significant fire scenarios are identified under conservative assumption that all equipment concerning the fire scenario is damaged by the fire. In the detailed analysis, sub-scenarios are made and the core damage frequencies for each sub-scenario are calculated. This methodology preliminarily was applied to 14 specific compartments of a typical Japanese 1,100MWe class four-loop PWR plant at full power operation to confirm the applicability of this methodology to Japanese LWR plants. As a result, the influences of the severity factors on the core damage frequency are large, and the severity factors seem too conservative to evaluate Japanese LWR plants. These factors used in some US IPEEE submittals are employed in detailed analysis as one of risk reduction factors and is defined as probabilities that equipment ignited fires can affect the target equipment as a function of the distance between the fire source equipment and the target equipment. The severity factors are shown in Fig. 1. JEAG-4697, that is a fire protection guideline for the nuclear power plants made by the Japanese industrial side, requires the use of the cables met IEEE-383 and 384 standards for safety graded cables, where the distance of more than 90cm is required between cable trays. However, Fig.1 shows that electrical equipment has the probability of 0.7 to affect the targets at 90cm from the fire source. It suggested that the severity factors seem too conservative to apply to Japanese LWR plants that are designed based on JEAG-4607 and constructed. In order to use fire PSA results, it is necessary to make the severity factors reflecting above Japanese circumstances. However, there are not enough fire events to make the Japanese specific severity factors to be able to apply to Japanese LWR plants. Therefore, using fire experiments carried out in Japan, $\alpha$-FLOW code that is a

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Computational Fluid Dynamics code and COMPBRN-IIIE code, we are ongoing to evaluate the severity factors analytically.

3. Outline of methodology
There are various fire analysis models from the simplified model to the detailed model that uses the CFD technique as the fire simulation technique. A simplified model is suitable for calculating a lot of cases because of short calculation time. However, some important input parameters must be determined by experiments, depending on compartment size, ventilation ratio and so on. The detailed model enables more strict calculation than a simplified model with few assumptions though it needs large calculation time. The CFD calculation has the potential to give input data to simplified model under different boundary conditions without experiments. Therefore, using input data derived from CFD calculation, severity factors can be estimated through uncertainty analysis of simplified model. We use COMPBRN-IIIE code as a simplified model and $\alpha$-FLOW code as a detailed model. Fig. 2 shows the flow chart of this methodology. This methodology consists of four steps of following.

Step 1: Although current $\alpha$-FLOW code has a function for the gas burning analysis, it has no burning process models for oil and the cable covering, that simulates evaporation of the oil and cable covering. Therefore, the burning rate (the evaporation velocity), which is an important parameters of $\alpha$-FLOW code, is an input data. In order to calculate the burning rate by an internal process of $\alpha$-FLOW code, the burning process models for oil and the cable coating are developed and built in $\alpha$-FLOW code. After verification of these models, $\alpha$-FLOW code is verified using existing fire experiments.

Step 2: The sensitivity analysis by which the combustible quantity, the ventilation ratio, and the compartment size, etc. are changed is executed, inputting the parameters set in step 1.

Step 3: The some input parameters of COMPBRN-IIIE code such as combustion efficiency are calculated from the sensitivity analysis results executed in step 2.

Step 4: The uncertainty analysis of COMPBRN-IIIE code input the parameters calculated in step 3 is executed, and the severity factors are calculated.

4. Development of the burning process models
4.1 Burning process model for oil
In this burning process model oil in the oil pan is modeled by a surface layer and a lower layer. The following mass conservation equations and the energy conservation equations are solved assuming that the oil evaporates only in the surface layer. It is assumed that the migration velocity of the oil between layers is equal to the evaporation velocity. And, the thickness of the surface layer is assumed not to change as long as lower oil is not lost. The evaporation velocity is provided as follow. Figure 3 shows the conceptual chart of the burning process model of oil.
Mass conservation equation of the surface layer
Mass conservation equation of the lower layer
\[ \frac{dm_{s1}}{dt} = -q_{\nu} + q_{i} \]

Energy conservation equation of the surface layer
\[ \frac{dH_{s1}}{dt} = -q_{\nu} + q_{1} + Q_{r} + Q_{c} \]

Energy conservation equation of the lower layer
\[ \frac{dH_{s2}}{dt} = -Q_{i} \]

Evaporation velocity
\[ q_{\nu} = \frac{h_{s} \Delta T_{s}}{h_{fg}} \]

4.2 Burning process model for cable covering
Cable is modeled in a metal region and covering region. One dimension mass conservation equation and two dimension energy conservation equation are solved. Mass combustion rates are important input data, which are derived from Ref. 1 in this model.

Figure 4 shows the conceptual chart of the model.

---

**Fig.3 The burning process model for oil**

**Fig.4 The burning process model for cable coating**

\[ 242(a) \]
4.3 Verification of the burning process model for oil
The burning process model developed the above for oil was verified by the analysis of the fire experiment carried out in Japan (2). In the experiment, oil (turbine oil VG32) in the oil pan set up in an infinite space is ignited, and the combustion characteristic such as the burning rate, the height of the flame, and the radiation heat flux are measured. In the analysis, the space was divided in 8x8x9 with 1/4 symmetry boundary. And the boundary condition is assumed the free surface. Fig. 5 shows the calculation model. The burning rates obtained from the calculation and obtained from the examination are shows in Table 1. This table shows that this model reproduces the results of this experiment.

<table>
<thead>
<tr>
<th>Oil pan size (m²)</th>
<th>Burning rate(mm/min)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Experiment</td>
</tr>
<tr>
<td>1.0</td>
<td>1.9</td>
</tr>
<tr>
<td>2.0</td>
<td>2.5</td>
</tr>
<tr>
<td>4.0</td>
<td>2.5</td>
</tr>
</tbody>
</table>

5. The analysis of the full scale compartment fire experiment (3)
5.1 The analysis by α-FLOW code
After verifying the burning process model for oil, we tried to verify the α-FLOW code using a full scale compartment fire experiment. In this experiment, turbine oil VG32 in the oil pan in the compartment (14mx6mx5m) was burned and the temperature distribution and the radiation heat flux, etc. were measured. The ventilation ratio is five times an hour. Fig. 6 shows the experiment system and table 2 summarizes the calculation conditions.

Table 2 Calculation conditions

<table>
<thead>
<tr>
<th>No. of Mesh</th>
<th>54 × 54 × 746 × 746</th>
<th>Temperature of outside air</th>
<th>310K</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oil</td>
<td>Turbine Oil VG32</td>
<td>Ventilation ratio</td>
<td>5 times / hour</td>
</tr>
<tr>
<td>Temp. of ceiling</td>
<td>310K (fixed)</td>
<td>Temp. of walls</td>
<td>310K (fixed)</td>
</tr>
</tbody>
</table>

The temperature distributions obtained from calculation are shown in Fig. 7 comparing with them obtained from the experiment. These calculation results roughly reproduced
the experimental results. However, the temperature distribution in the vicinity of the floor is different between calculation results and the experimental results. This should be caused by the fact that the temperature on the floor is assumed to be 310K and that the mesh division of the height direction near the floor is too rough.

Fig. 7 Temperature distributions in the compartment

5.2 Making of input data of COMPBRN-IIle code from calculation results of α-FLOW code

Some input data of COMPBRN-IIle code such as combustion efficiency etc. were made from the results of the full scale compartment fire experiment analysis by the above α-FLOW code calculation. The input data made are shown below. The value in parentheses denotes a sample input data of COMPBRN-IIle code.

Combustion efficiency: 0.667(0.265)
Ventilation Control Burning Rate Constant: 0.161(0.11)
Buoyant Plume Entrainment Coefficient: 2.323(1.5)

5.3 The analysis by COMPBRN-IIle code

The full scale compartment fire experiment was analyzed by COMPBRN-IIle code, with some input data calculated from analysis of the α-FLOW code. Fig. 8 shows calculation results. Fig. 8 shows that only hot gas layer was formed and that COMPBRN-IIle code underestimates the average temperature in the compartment compared with the experiment. However, it can be judged from the experimental results that two layers were formed. To form two layers, the analysis where the door was set virtually was done. The size of the door was decided that calculation results roughly reproduced the experimental results. Fig. 9 shows calculation results. It is understood to be able to reproduce the experimental results from fig.9 by setting the door. However, when the cases without the experimental results are calculated, the size of the door cannot be adjusted. Therefore, COMPBRN-IIle code should be modified to form two layers even when the case without door will be calculated.
6. Evaluation of severity factors for oil ignited fire by uncertainty analysis of COMPBRN-III code

Four cables were set in the compartment and the severity factors for each cable were evaluated. Fig. 10 shows calculation model. Here, the cable damage temperatures of 645K and oil quantity of 30kg and the suppression time of 42 minutes were assumed. In the analysis, the cables were horizontally shifted and the severity factors for each cable were evaluated as a function of the distance between the center of the oil pan and the center of the cable trays. The calculation results show in Fig. 11, compared with the US severity factor.

7. Sensitivity analysis

As in future the severity factors are intended to be evaluated as functions of compartment size, oil pan size, oil quantity and so on, the influences of the compartment size and the oil pan size on the severity factors were investigated.

7.1 Compartment size

The length of the compartment being made half, the analysis was done. Fig. 12 shows calculation results. It is understood that the severity factors are considerably getting large compared with the base case. Especially, because the distance between the flame and the wall is shorter than the base case, the reflection at the wall increases, and the severity factors increase near the wall.
7.2 Oil pan size

The analysis with the oil pan area set twice (2m²) and 4 times (4m²) were done. Calculation results are shown in Fig. 13 for 2m² and in Fig. 14 for 4m². It is understood that the severity factors increase as a result with the increase of the area of the oil pan.

In both cases, that the compartment size is changed and that the oil pan size is changed, these parameters are large contributor to severity factors.

8. Summary

It was confirmed to be able to evaluate the severity factors with this developing methodology. However, there are some ground for improvement of COMPBRN-III code for making severity factors because two layers are not formed at the case without the door.

References

(1) A.Twewson and R.F. Pion
"Combustion and Frame", 26, 85, (1976)


Fire Safety Assessments in Ontario Power Generation

Joan Higgs, Engineering Planning & Management

The presentation today is on the Fire Safety Assessments that Ontario Power Generation has underway at its operating nuclear units. The presentation was prepared by Wade Lawson of Engineering Planning and Management. The assessments are not full probabilistic studies, but qualitative in nature, and provide means for identifying fire vulnerabilities and establishing the most cost effective responses to prevent or remedy any unacceptable reductions in performance capability.

I am going to discuss today:

- What are we doing
- Why are we doing it
- The methods we are using
- The level of detail
- The insights that we seek
- And share with you some preliminary perspectives and conclusions

Ontario Power Generation has three operating four-unit CANDU reactors: Bruce B, Pickering B and Darlington. Two more sets of four units: Pickering A and Bruce A are currently candidates for restart.

In 1996 Ontario Hydro was requested by the Atomic Energy Control Board (AECB) of Canada to evaluate itself against the applicable sections of the new CSA Standard N293-95, “Fire Protection for CANDU Nuclear Power Plants”. This specified requirements for new CANDU reactors. Our regulator specified sections for comparison focused primarily on the requirements articulated for operating plants. Those requirements in turn required a detailed evaluation of the physical design aspects of our CANDU reactors at a level of detail not previously collected or evaluated.

This activity was also undertaken in response to the results of an Integrated Independent Performance Assessment (IIPA) of the fire protection, fire safety, and post-fire system response capabilities of our units. As a result of the IIPA, Ontario Power Generation (OPG) has initiated an upgrade to fire
protection. To focus that upgrade and logically establish the specific upgrade requirements, OPG has undertaken Fire Safety Assessments at each station.

The Fire Safety Assessment studies are comprised of two major sub-activities:

- the generation of a detailed Fire Hazards Analysis, which includes a cataloguing and listing of all the physical characteristics of the station’s units including combustibles contained therein to determine the fire hazard and fire load in each physical area or volume of the plant, and
- the generation of a Post-Fire System Response Capabilities Assessment. This effort evaluates the impacts of fire on susceptible targets within the plant associated with critical systems required for the necessary cooldown and shutdown safety functions.

The study is unique in that its evaluating the effects of a particular spatially related common mode failure mechanism — fire — assessing the impact on each evaluated volume, and suggesting the response actions to fire in a level of detail not previously applied to OPG units.

The overall process of the fire safety assessment is being done in two steps. The first is the activities associated with the generation of the Fire Hazards Analysis which include the assessment of the fire potential including a cataloguing of ignition sources, fuel load, including fuel magnitude, fuel type, fuel continuity, potential for fire growth. It also includes identifying existing plant mitigating characteristics that will contain, confine, detect, or suppress a fire.

For guidance we are utilizing:

- CSA N293-95, “Fire Protection for CANDU Nuclear Power Plants”

To implement these we are utilizing tools and techniques, including software that has been used elsewhere in similar evaluations of light water reactor system response to fires. Success path models are
developed of each station. We’re modeling systems and components required for the proper achievement of required safety functions including a spatially related model, wires, cables, and circuits to the systems and components and locating them in the associated volumes. Our modeling is at a very detailed level.

The model includes CANDU reactor Group 1 and Group 2 systems (where Group 2 is available within the design). Also modeled are inter-unit connections, cross-connects and power cross-feeds and, in instances where they are present, inter-station connections. The assessment also looks at special system line-ups that may be off-normal relative to normal power generation or normal system alignment but are within the capability of the system and will achieve accomplishment of the safety function either at its full capacity or at a lesser yet adequate capacity.

Simply stated our objective is to meticulously document the specific instances of vulnerability to a spatially-related common mode failure mechanism, i.e., fire. This process includes identifying all available reasonable, achievable ways to use available unaffected system performance capabilities to accomplish the CANDU reactor’s objectives of control, cool, and contained.

The scope of the Fire Safety Assessment includes looking at each “four-pack” and assessing the impacts of fires on these units up to and including the previously assumed design basis fires. Each station is of a different vintage and has slightly different designs. Hence, each station has different sets of strengths and weaknesses that reflect the design characteristics. The success path model that is being constructed for station is a logical model of the four units all presumed operating at full power. The model evaluates an assumed destructive fire in any location that is consistent with the original physical design, the licensing basis including “Level 1” correspondence with the Canadian regulatory authority and the realities of physical plant operation.

The purpose of the Fire Safety Assessment is to pinpoint the discrete physical locations of greatest fire vulnerability to compare the physical design situations at these locations to the guidance documents, to rank these based on the magnitude of potential consequential component and system failures associated with the assumed fire, and then to identify the range of cost effective responses appropriate to the actual situation. (The logical model is evaluated by specific software that has been used extensively for this purpose.)
The analysis models the system safety function to system, to component, to cable, to wire, to circuit, to physical location relationships.

The fire cause can be presumed to be a cable tray fire (which is a design basis at certain Ontario Power Generation units), a consequential cable tray stack fire, a fire resulting from equipment failure, or a hot work/transient combustible fire of a magnitude consistent with the ignition sources appropriate to the area and the magnitude of transients that flow through or are resident in the area.

The analysis is conducted on two levels/in two steps: The first analysis step is “the screen” which arbitrarily assumes full involvement of the combustibles within the fire area. This step assumes damage or loss of function to cables and equipment in the area and determines the residual capability of the unit and station (and/or the overall site) to effect safe control, cool and contained functions, with systems capability independent of the fire-affected zone. If there is no impact or minimal impact on Group 1 or Group 2 systems from the assumed fire in the area, then the analysis steps to the next analysis volume.

Analysis volumes are either defined by the physical fire barriers present in the unit or “potential” fire barriers in the case where the integrity of the wall or ceiling is not currently adequate to qualify as a fire barrier, or the analysis volume is defined by an arbitrary volume of size 30 feet by 30 feet by 30 feet. If the impact of the arbitrarily assumed fire is determined to have significant safety system losses and functional capability losses, then a Level 2 detailed analysis and review of the fire scenario in the fire area or analysis volume is conducted by the systems and the fire protection engineering team.

The second level evaluation looks qualitatively at the physical relationships of the fire sources and targets considering a wide range of factors including potential fire magnitude, target height, ceiling height (if ceilings are present), potential heat flux concentrators, protection system presence and type, suppression system presence and type, suppression system capability and response characteristics, etc. Qualitative assessment judges the target type and location relative to the fire source including whether full involvement or component or wire failure will occur. If that is considered likely and credible, then a palette of potential design options and personnel responses are considered.

Once a credible associated response action or physical plant change is selected to reduce or eliminate the impact of the fire on effected equipment, the specific damage effects in the safe shutdown model are masked and the analysis proceeds. In the end the result of the fire safety assessments will result in
identifying where the potential exists to lose redundancies or total safety function as a result of credible fire scenarios and what design or operational alternatives are most appropriate to remediate the situation.

This effort is still ongoing; the first set of four units results are in the final phases of development and remediation and response strategies are being developed.

What conclusions have been reached thus far from our efforts to date?

1. Upgrades of both detection and suppression will be occurring.
2. "Pre-fire" (to coin a term) or "incipient" fire detection of heated cable combustion products before exothermic reactions are established, coupled with the use of thermographic equipment and well-trained and well-supported fire brigade and emergency response teams, is probably the most cost effective retrofit alternative if good physical access for quick response is present.
3. Ignition source control, transient material control, and effective plant personnel training will always be the most cost effective alternative, if not necessarily the perfect, full-proof totally dispositive alternative.
4. A breakdown in the “fire response system” due to faulty design details, poor maintenance, or the absence of appropriate operator response procedures to unanticipated complex scenarios can overwhelm and radically shift the risk profiles of the plant.
5. Qualitative fire science is still in its infancy while plant modeling of physical plant characteristics is advanced. The latter can be quite instructive in explicitly pinpointing adverse situations.
6. Excellent configuration management information on cable and circuit location and cable/wire routing is necessary and is essential to maintain throughout the life of the units if these types of analyses are to be conducted with confidence.
Session VI
Experimental Fire Research II

- Results of Experimental Fire Research - Steven P. Nowlen, Sandia National Laboratories (SNL), United States

- Failure Distribution in Instrumentation Cables in Fire - Mr. Johan Mangs, VTT Building Technology, Finland, Dr. Olavi Keski-Rahkonen, VTT Building Technology, Finland, Mr. Hannu Hossi, VTT Automation, Finland and Mr. Arto Salminen, VTT Automation, Finland

- Behaviour of High Efficiency Particulate Air Filters in Case of Fire - MM. Jean-Claude Laborde, A. Briand and V. M. Mocho, Département de Prévention et d'Etude des Accidents, Saclay, IPSN, France

Results of Experimental Fire Research

Steven P. Nowlen
Sandia National Laboratories
Accident and Consequence Analysis Department
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Presented at the OECD/CSNI
International Workshop on Fire Risk Assessment
Helsinki, Finland
June 29 - July 1, 1999

- Objective:
  - Present a survey of current experimental and data analysis fire research activities at Sandia National Laboratories (SNL) related to nuclear power plant fire safety.

- Topics:
  - Investigation of smoke damage for digital circuits,
  - Scoping study of smoke and higher voltage circuits
  - Release of the Baseline Validation Test Data
  - Estimation of Heat Loss Factor from test data
  - Benchmarking of simple hot gas layer models
  - Cable failure modes and likelihood data review.

- Sponsors:
  - U.S. Nuclear Regulatory Commission Office of Research
  - Internal SNL funding
Smoke damage and digital circuits

- Efforts began through USNRC sponsorship
- Efforts continue under both USNRC and internal SNL funding
- Substantial interest from the telecommunications industry
- Devices tested previously include:
  - Individual chips and chip packages
  - Optical isolation devices
  - Four classes of active digital circuits
  - Digital trip system mock-up
- Devices currently under testing:
  - Static and dynamic memory chips
  - Hard Disk Drives

Testing Insights:

- Short term damage is a real potential for digital circuits
  - The primary mode of damage is circuit bridging
  - Effects related to both deposition and airborne particulate
  - In many cases performance is recovered when fresh air is introduced
- Moisture is a critical parameter
  - Higher moisture lead to more severe damage
- Fuel type has some impact on results
  - Impact is secondary effect
Testing Insights (cont.)

- The type of digital circuit is also a critical parameter
  - Some types of circuits are less vulnerable (e.g., high current circuits)
  - Chip type also influences vulnerability (e.g., CMOS versus MOSFET)
- For many circuits high smoke exposure is required to cause damage
- Coatings can provide protection, but different coatings provide different levels of protection
- Contact Tina Tanaka, SNL, for further information

Smoke and Higher Voltage Devices

- Risk contribution of smoke damage is poorly understood
- Are switchgear, transformers, breakers, motor control centers vulnerable?
  - Anecdotal reports for switchgear
  - No confirmed reports, no tests
- Scoping study underway
  - Objective: Perform tests to explore fault potential for high voltage circuits
  - Focus on 4kV (ac) level for these tests
- Modes of faulting being explored:
  - Deposition
  - Airborne particulate
Preliminary results - for illustration only

High voltage AC arcing, ~4kV, CSPE/PE cable insulation

The Baseline Validation Tests

- 1984-85: USNRC sponsored 25 large-scale enclosure fire tests
- Objective: Provide validation data for enclosure fire models under NPP conditions.
  - Scale: 18×12×6 m test enclosure
  - Ventilation: supplied by a distributed forced air system
  - Fires: gas burners, liquid fuel pool fires, and electrical control panels
  - Configuration: Open room and control room mock-up
  - Over 300 channels of data generated including:
    - bare bead and aspirated thermocouples for air temperature,
    - calorimeters for heat flux and bulk velocity,
    - smoke optical density devices,
    - gas analysis for CO, CO₂, O₂ and total hydrocarbons, and
    - surface temperature response.
The Baseline Validation Tests (cont.)

- Final processing and release of the data was not completed at the time.
- Previously, only the main data files for three of the 25 tests was available.
- As a part of the current USNRC/RES Fire Risk Methods program, final processing of the data has been completed.
- The full data set from all 25 tests is now publically available!
- If interested, contact the author for more information.

Test Data and Surface Losses

- Fire models must deal with heat losses to enclosure surfaces
  - Wall losses are significant
- One simple approach: assign a pre-defined fraction of the total heat release to surface losses.
  - Heat Loss Factor (HLF)
  - Assigned HLF value substantially impacts estimated hot layer temperature response
- Only a few estimates of the HLF value have been published
- The “correct” HLF value depends on the correlation
Test Data and Surface Losses (cont.)

- Estimation of HLF for fire tests can yield useful information.
  - Provide a benchmark for HLF-based correlations
  - "Reality Check" for the models
- We have recently looked at 2 data sets for this purpose:
  - Twenty Foot Separation Tests (1981)
    - Natural Ventilation Tests
    - Moderate room size (appr. 9 x 4 x 3 m)
  - Baseline Validation Tests (1985)
    - Forced Ventilation
    - Large Enclosure (appr. 18 x 12 x 6 m)

Results for 20-Foot Tests

<table>
<thead>
<tr>
<th>Test ID</th>
<th>Time-Averaged HLF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experiment 2</td>
<td>0.379</td>
</tr>
<tr>
<td>Experiment 3</td>
<td>0.425</td>
</tr>
<tr>
<td>Test 1</td>
<td>0.465</td>
</tr>
<tr>
<td>Test 2</td>
<td>0.465</td>
</tr>
</tbody>
</table>
Results for Baseline Validation Tests

<table>
<thead>
<tr>
<th>Test</th>
<th>Fire</th>
<th>Ventilation rate</th>
<th>Average HLF</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>516 kW (Steady)</td>
<td>10 ACH</td>
<td>0.507</td>
</tr>
<tr>
<td>2</td>
<td>516 kW (Steady)</td>
<td>10 ACH</td>
<td>0.553</td>
</tr>
<tr>
<td>3</td>
<td>2000 kW (Steady)</td>
<td>10 ACH</td>
<td>0.601</td>
</tr>
<tr>
<td>4</td>
<td>516 kW (Growing)</td>
<td>1 ACH</td>
<td>0.619</td>
</tr>
<tr>
<td>5</td>
<td>516 kW (Growing)</td>
<td>10 ACH</td>
<td>0.544</td>
</tr>
<tr>
<td>7</td>
<td>516 kW (Steady)</td>
<td>1 ACH</td>
<td>0.685</td>
</tr>
<tr>
<td>8</td>
<td>1000 kW (Growing)</td>
<td>1 ACH</td>
<td>0.732</td>
</tr>
<tr>
<td>9</td>
<td>1000 kW (Growing)</td>
<td>8 ACH</td>
<td>0.616</td>
</tr>
</tbody>
</table>

ACH = Room Air Changes Per Hour

HLF Insights:

- For 20-foot tests, values appear quite low:
  - Ceiling was insulated - heat loss negligible
  - Un-insulated or concrete ceiling would yield higher HLF
- For Baseline Validation Tests HLF was 0.51 to 0.73
  - Increasing rates of ventilation flow generally led to lower HLF
  - No clear trend associated with fire intensity
  - No clear trend between the transient (growing) and steady-state fires
- Note: Transient (time dependent) HLF estimates also available
Hot Gas Layer Modeling - simple methods

- There are simple models of hot gas layer response
  - Independent engineering correlation
  - Potential fire risk applications - screening methods
    - If simple methods yield appropriate results, why result to more complex models
    - Questions include reliability, limitations, accuracy
      - We need to know when simple models can be expected to yield appropriate results
- Current effort is benchmarking models against data
- Preliminary results for two of the identified models are available.
  - Mowrer/FIVE
  - Foote, et.al.

Mowrer/FIVE Model:

- Model provides a point estimate of the peak hot layer response
- Can be exercised in a “pseudo-transient” mode
- The model itself is expressed as follows (metric version):

\[ T_{\text{final}} = T_0 \exp \left[ \frac{(1 - HLF) \times Q_{\text{total}}}{353 \times V} \right] \]
Mowrer/Five Model Results:

![Graph showing temperature data with annotations indicating test results and model assumptions.]

Foote et al. (LLNL) Model:

- Somewhat more complex than the Mowrer/FIVE model
  - Transient model
  - Includes ventilation rate for forced ventilation fires
  - Not and HLF model, uses heat transfer coefficient approach
- The results that follow assume:
  - Heat transfer only to walls and ceiling (not floor)

\[
\Delta T = \frac{\dot{Q}}{\dot{m} C_p + h_k A}
\]

\[h_k = 860 \times t^{-1/2}\]
**Foote Model Results:**

Comparison of Foote Correlation to Baseline Test 1

Comparison of Foote Correlation to Baseline Test 3
Cable Failure Modes and Likelihood

- The nature or mode of cable failure will determine its impact, for example:
  - Shorts to ground generally lead to:
    - loss of system power/control/indication
    - loss of power busses
    - general unavailability
  - Conductor-to-conductor faults might lead to
    - spurious equipment operations
    - misleading indications on instruments
- For some situations capturing the failure mode might lead to unique risk insights

Cable Failure Moses and Likelihood (cont.)

- The objectives of this task are to:
  - Identify parameters that impact cable failure modes and likelihood
  - To the extent possible, establish the probability that given a cable failure a particular failure mode will be observed
- A search and review of past test data is underway
  - Looking for any tests in which cable fault modes were recorded/monitored
- A review of fire events is also underway
  - May yield qualitative insights into failure mode likelihood
Failure distribution in instrumental cables in fire

Johan Mangs and Olavi Keski-Rahkonen
VTT Building Technology
Hannu Hossi and Arto Salminen
VTT Automation

Abstract
The fire-induced response of a four-conductor automation cable connected to a pressure transmitter has been investigated. Two series of fire experiments with two different cable types exposed to flames from a propane gas burner has been carried out. Times to first disturbance in transmitter voltage supply and transmitter current output was determined as well as times to first disturbance in insulation resistance between pairs of conductors in a non-energised open-circuit cable. Probability distributions have been fitted to the observed cumulative frequencies of time to first disturbance. The function of the system was also studied without fire exposure considering the effects of conductor breaks, different short circuit resistances and voltage supply. The dependence of current output on short circuit resistance and voltage supply has been expressed mathematically.

1. Introduction
Fire-induced changes in automation cable signals may cause critical malfunction in plant control functions (Hasegawa et al. 1992). In the present study, a four-conductor automation cable connected to a pressure transmitter was chosen as an example of a real process control system.

2. Specimen
Two types of polyvinyl chloride jacketed and insulated four-conductor automation cable were used in the fire experiments, MHMS-SI and NOMAK-E. Nominal cable dimensions are given in table 1.

The conductors in NOMAK-E cable are twisted in pairs with an average length of lay 45 mm and the pairs are twisted with an average length of lay of 150 mm according to the manufacturer. The conductors in MHMS-SI are twisted in a similar way with corresponding lengths of lay of the same magnitude, about 70 mm and 170 ... 200 mm measured from a piece of cable. The pairs are surrounded by a plastic tape in both types of cable. A cross-section of the four-conductor cable is presented in section 5, figure 9.
Table 1. Nominal dimensions of automation cables in fire experiments according to the product specifications of the manufacturers.

<table>
<thead>
<tr>
<th>Cable</th>
<th>MHMS-SI</th>
<th>NOMAK-E</th>
</tr>
</thead>
<tbody>
<tr>
<td>Conductor diameter (mm)</td>
<td>0.8</td>
<td>0.8</td>
</tr>
<tr>
<td>Conductor cross-section (mm²)</td>
<td>0.5</td>
<td>0.5</td>
</tr>
<tr>
<td>Jacket thickness (mm)</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Insulation thickness (mm)</td>
<td>0.35</td>
<td>0.35</td>
</tr>
<tr>
<td>Cable outer diameter (mm)</td>
<td>9.5</td>
<td>8.5</td>
</tr>
</tbody>
</table>

The conductors in both cable types are shielded with a plastic faced aluminium tape immediately under the jacket. An earthing conductor without insulation was under the plastic faced aluminium tape.

3. Failure modes in cable connected to pressure transmitter

Automation cable MHMS-SI 2 x 2 x 0.8 + 0.8 mm (conductor diameter) connected to a Siemens Teleperm pressure transmitter was investigated without fire exposure.

The automation cable consists of four conductors and an earthing-conductor (V). The four conductors are numbered as shown in figure 1.

![Pressure transducer diagram]

Figure 1. Numbering of conductors of automation cable connected to pressure transducer.

The following two main failure modes were considered:

- **Break** situation when one or several conductors are in two
- **Short circuit** situation when one or several conductors are in galvanic connection with each other
3.1 Consequences from breaks in and short circuits between conductors

The consequence from a break in any one of the conductors 1...4 is interruption of transmitter current output.

The consequences from short circuits between conductors are presented in table 2.

Table 2. Consequences from short circuits between conductors 1...4 and earthing-conductor V to transmitter current output 0...20 mA.

<table>
<thead>
<tr>
<th>Short circuit between conductors</th>
<th>Current output (mA)</th>
<th>Consequence</th>
</tr>
</thead>
<tbody>
<tr>
<td>1-2</td>
<td>0</td>
<td>The transmitter current output is interrupted</td>
</tr>
<tr>
<td>1-3, 1-4, 1-V</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2-3, 2-4, 2-V</td>
<td>OK</td>
<td>No influence on transmitter function or 0...20 mA current output.</td>
</tr>
<tr>
<td>3-V</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4-V</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3-4</td>
<td>20...0</td>
<td>The transmitter 0...20 mA current output depends very strongly on the short circuit resistance (Figure 2)</td>
</tr>
</tbody>
</table>

If a multiple short circuit contains short 1-2, transmitter current output is interrupted. If the multiple short circuit is 1-3-4 or 2-3-4, transmitter current output is interrupted. Short circuits with the earthing-conductor does not change the current output.

3.2 Pressure transmitter current output with different short circuit resistances

Pressure transmitter current output was measured with pressures 0, 3, 6 and 10 bar. Short circuit resistances between conductors were $\infty$, 0.5, 1, 10, 100, 1000, 10k, and 100kΩ. As presented in table 2, only two short circuit pair combinations have an effect on current output, 1-2 which interrupts the current output and 3-4 when current output depends very strongly on the short circuit resistance.

The consequences of short circuits between a pair of conductors do not change if the pair is in contact with the earthing-conductor or the plastic faced aluminium tape.

The results concerning short circuit between conductors 3-4 are presented in figure 2, where the short circuit current output at different pressures is divided with the corresponding open circuit current (infinite resistance). These normalised currents are almost coincident for the different pressure values making simple curve fitting possible.
The inverse tangent function with fitting parameter $R_0 = 6 \ \Omega$ was found to give a satisfactory fit to the measured values as shown in figure 2. The relation between current output $i$ and pressure $p$ can then be expressed as

$$i(mA) = \frac{2}{\pi} \cdot \frac{p(\text{bar})}{\arctan\left(\frac{R}{R_0}\right)}$$  

with $R_0 = 6 \ \Omega$. It appears from figure 2 that changes in the current output do not occur until short circuit resistances at some hundred $\Omega$.

![Figure 2. Pressure transmitter relative current output $i/i_\infty$ versus short circuit resistance between conductors 3-4, measured values at different pressures and fitted inverse tangent function.](image)

### 3.3 Transmitter current output as a function of voltage supply

Transmitter current output was measured as a function of voltage supply. The relative current output, i.e. current output divided with ordinary current output at ordinary 24 V supply is presented in figure 3.

It appears from figure 3 that the transmitter current output is steady at these pressures down to voltage supply 12 V. When the voltage supply is below 12 V the output signal is disturbed and below 8 V the current output drops to zero. The current output as a function of voltage supply can thus be described by a step function

$$i(mA) = \begin{cases} 
  p(\text{bar}), & u > 12V \\
  0, & u \leq 12V 
\end{cases}$$  

This is in accordance with the transmitter specification which guarantees performance down to 12 V voltage supply.
Figure 3. a) Pressure transmitter relative current output $i(u)/i(u=24\,\text{V})$ versus supply voltage $u$ at different pressures. b) The same data on a smaller scale revealing disturbance in output signal at voltage supply below 12 V.

4. Fire experiments

Two series of fire experiments were carried out under similar circumstances, one series of 9 experiments with cable type MHMS-SI and one series of 24 experiments with cable type NOMAK-E. The cables were exposed to direct flame contact in order to simulate a local fire-induced electrical fault in the cables.

The experimental configuration is presented in figure 4. Two cables were studied in each experiment, one energised cable C1 connected to the pressure transducer P and one non-energised open circuit cable C2 for monitoring short circuiting through insulation resistance measurements. The numbering of conductors in the cables is the same as in section 3.

The cables were attached with steel wire to two adjacent rungs R of a steel cable tray CT (figure 4 b). The distance between the rungs was 250 mm and the inner width of the cable tray was 370 mm. The experiments were carried out in nearly free space with the burner and the part of the cables exposed to fire surrounded by three 15 mm thick mineral wool boards S guiding air flow in order to stabilise flames.

The distance between the upper surface of the gas burner and the cables was chosen so short (85 mm) that the cables above the centre of the burner were in the persistent flame, thus ensuring identical fire exposure conditions during the test series.
Figure 4. The experimental configuration in instrumental cable fire experiments: a) side view, b) top view and c) thermocouple T1, T2 and T3 location, top view. P Siemens Teleperm pressure transmitter, B propane gas burner, CT steel cable tray with steel rungs R, C1 energised pressure transducer cable, C2 open circuit non-energised resistance measurement cable, D data acquisition unit and S mineral wool shield in order to stabilise flames. Dimensions in mm.

The 170 mm x 170 mm propane gas burner was ignited outside the cable set-up and the flame was stabilised to constant 12.4 kW power after which the burner was located beneath the cables with flames rising through the centre of the free space between rungs R as shown in figure 4 a.

Gas temperatures T1, T2 and T3 about 10 mm above the cables were monitored with 0.5 mm K-type thermocouples. Time for exposure to fire was determined from gas temperature measurements as time for distinct temperature rise.
The pressure transmitter was pressurised with compressed air from a cylinder to 6 bar gauge constant pressure which was monitored with a separate pressure transmitter. This pressure corresponds to typical value for the investigated pressure transmitter.

The supply voltage to and current output signal from the pressure transducer was monitored.

Times to short circuits between pairs of conductors in cable C2 were monitored through direct insulation resistance measurements with a voltage divider circuit. With suitable voltage divider circuit parameters a resistance range of magnitude 100 Ω ... 1 MΩ could be studied.

Data were collected with 0.1 s interval in the experiments. Each experiment was continued for a couple of minutes after the observation of specific changes in the electrical signals.

Efforts were made to monitor all possible combinations between pairs of conductors. Difficulties occurred with isolating the combinations, probably due to the properties of the data acquisition system. Thus only independent combinations could be studied. Pairs 1-2 and 3-4 were chosen in accordance with the conclusions in section 3 that only short circuits between these two pairs had consequences for the current output signal. The resistance 1-V between conductor 1 and the earthing conductor V was also monitored because it was noticed in preliminary experiments that the resistance curve behaved in a different way than the resistance between 1-2 or 3-4, showing exponential damping.

5. Results and discussion

5.1 Voltage supply to and current signal from cable C1

Voltage supply

The usual behaviour of the voltage supply curve $U_{\text{supply}}$ in both MHMS-SI and NOMAK-E series was constant voltage until sudden drop to zero voltage. Time to first disturbance in $U_{\text{supply}}$ corresponds thus to first zero voltage measurement. In one experiment no disturbance was detected during the 126.9 s long experiment.

The voltage supply cut off is probably due to functioning of the over-current release of the voltage supply unit as a consequence of short circuiting in conductor pair 1-2. The consequence on $I_{\text{out}}$ as presented in section 3.1 is that the transmitter current output is interrupted.

An example of voltage supply, current signal output and reference pressure curves is presented in figure 5.
Figure 5. NOMAK-E, experiment 5. $U_{\text{supply}}$ in units of V, $I_{\text{out}}$ in units of mA, and reference measurement of pressurisation with compressed air in units of bar. The first disturbance in $U_{\text{supply}}$ and $I_{\text{out}}$ are indicated with arrows.

Current output signal

Different behaviour was noticed in first disturbance in current signal output $I_{\text{out}}$ (figure 6). The first disturbances in current output signal $I_{\text{out}}$ and the corresponding behaviour of voltage supply can be described as follows:

1. $I_{\text{out}}$ dropped to zero or slightly below zero, about 2...3 mA below zero level, without changes in $U_{\text{supply}}$ (3 cases in MHMS-SI series and 15 cases in NOMAK-E series, example in figure 6 a). This can be interpreted as short circuit between conductors 3-4 as presented in section 3.

2. $I_{\text{out}}$ increased to maximum measurement range about 500 mA (3 cases in MHMS-SI series and 3 cases in NOMAK-E series, example in figure 6 b) simultaneously with an 0.5...1.3 V level drop in $U_{\text{supply}}$. In four cases the high $I_{\text{out}}$ level dropped to zero simultaneously with a cut off in supply voltage, in one case $I_{\text{out}}$ dropped to zero 0.1 s before voltage cut off and in one case the ordinary $I_{\text{out}}$ level was recovered for 0.8 s before voltage cut off and $I_{\text{out}}$ drop to zero. This is interpreted as interference with power supply conductor pair 1-2.

3. $I_{\text{out}}$ increased to 9...10 mA (4 cases in NOMAK-E series) simultaneously with $U_{\text{supply}}$ drop to zero, example in figure 6 c. Although the consequence of a voltage cut off should be interruption of current output according to section 4, there was noise in the monitored current output signal after voltage supply cut off in many of the experiments (figure 5). The increase in $I_{\text{out}}$ is thus here interpreted as noise in the data acquisition.

4. $I_{\text{out}}$ dropped to zero or slightly below zero, about 2...3 mA below zero level, simultaneously with or immediately after $U_{\text{supply}}$ drop to zero (3 cases in MHMS-SI...
series and 1 cases in NOMAK-E series, example in figure 6 d). This is interpreted as a consequence of voltage supply cut off as presented in section 3.

5. $I_{out}$ dropped to zero 0.1 s after $U_{supply}$ dropped to zero (one case in NOMAK-E series).

The term simultaneously means here within the 0.1 s data collection time interval.

Additionally a drop of about 1 mA was noticed in the $I_{out}$ level before first disturbance in three NOMAK-E experiments, example in figure 6 d. These drops in current output level were not noticed as first disturbances because the reason of the drop could not be unambiguously interpreted.

In some experiments a recovering of current output signal to normal level was noticed under condition of ordinary voltage supply. This is probably due to temporary opening of short circuit.

No specific correlation was noticed when comparing different types of first disturbance in $I_{out}$ with times to first disturbance.

![Graphs showing different types of disturbance in $I_{out}$](image)

**Figure 6.** Different types of disturbance in $I_{out}$: a) $I_{out}$ goes to zero without changes in $U_{supply}$, b) $I_{out}$ goes to maximum measurement range about 500 mA (out of scale) with corresponding slight $U_{supply}$ drop, c) $I_{out}$ increase to 9...10 mA simultaneously with $U_{supply}$ drop to zero, d) $I_{out}$ and $U_{supply}$ go to zero simultaneously, $I_{out}$ level drop from 5.6 mA to about 4.6 mA (marked with arrow) is also noted.
**Time to first failure**

A comparison between times to first disturbance in current output signal $I_{out}$ and voltage supply $U_{supply}$ gives:

- in MHMS-SI series a disturbance occurred in $I_{out}$ before disturbance in $U_{supply}$ in 3 experiments of 9, and in 6 experiments the same time was recorded for disturbance.

- in NOMAK-E series a disturbance occurred in $I_{out}$ before $U_{supply}$ in 14 experiments of 24, in 9 experiments the same time was recorded for disturbance, and in one experiment $U_{supply}$ went to zero 0.1 s before $I_{out}$.

Current output signal seems to have slightly shorter time to first disturbance than voltage supply in both cable type series, the difference being about 2...4 s according to cumulative failure probabilities presented in figure 9.

### 5.2 Resistance measurements from cable C2

Only independent pairs of conductors could be monitored because of isolating difficulties. Critical combinations 1-2 and 3-4 were studied and additionally pair 1-V. The times to first disturbance in other combinations between pairs of conductors remained thus unexplored.

The first resistance drop was noted as first disturbance, without distinction between magnitude of resistance drop. The times to first disturbance are thus somewhat conservative, because it is possible that a noticed small drop in insulation resistance does not effect the function of the cable, cf. figure 2, section 3.3, where consequences on the current signal output occurs only for insulation resistance smaller than 1 kΩ.

Instantaneous drops occurred at an early stage in NOMAK-E experiments 21 (conductor pair 1-2 and 3-4) and 22 (conductor pair 1-2). It is not clear whether these disturbances are due to fire exposure or electrical effects in the data acquisition system not related to the fire. The latter possibility is not excluded, especially as an instantaneous drop was measured in experiment 21, conductor pair 1-2, even before the burner was placed below the cables.

The overall behaviour of the resistance curves in both MHMS-SI and NOMAK-E series can be described as follows:

1. Intermittent disturbances in periods of no disturbance (figure 7 a).

2. Drop to total short of longer duration (figure 7 b).

3. Subsequent disturbances with exponentially decreasing resistance leading to either total short (figure 7 c) or recovering eventually followed by total short (figure 7 d).

Types 1 and 2 or combinations of them were predominant for pairs 1-2 and 3-4 (figures 7 a and 7 b) with only 3 events showing exponentially decaying resistance before total short.
All resistance curves for pair 1-V showed type 3 shape, examples of which are given in figure 7c and 7d.

Figure 7. a) NOMAK-E cable, experiment 1, conductors 3-4, b) NOMAK-E, experiment 8, conductors 1-2, c) NOMAK-E, experiment 11, conductors 1-V and d) NOMAK-E, experiment 5, conductors 1-V. The first disturbance is indicated with an arrow.

The time to first disturbance in pair 1-V was clearly shorter than times to first disturbance in pairs 1-2 and 3-4 for both cable types. The reason for this is probably the structure of the cable (figure 8). The plastic faced aluminium tape shielding the conductors forms a compact package around the conductors keeping them together also during the fire. The earthing conductor V without insulation is in contact with the aluminium shield after consumption of the plastic tape surrounding the pairs of conductors and the plastic facing of the aluminium tape by fire. The four insulated conductors (numbered 1, 2, 3 and 4 in figure 8) are thus surrounded by the conducting aluminium shield in contact with the earthing conductor. After beginning of reduction of insulation resistance of a conductor, it is assumed to be more probable that the first contact is with the surrounding bare aluminium shield (e.g. path A in figure 8), which in turn is in contact with earthing conductor V, than with one of the other insulated conductors (e.g. path B in figure 8). This argumentation should be applicable on all combinations 1-V, 2-V, 3-V and 4-V, although only one combination was examined in the present work.
Comparing times to first disturbance for conductor pair 1-2 and 3-4 shows that pair 3-4 has slightly shorter times to first failure than 1-2 for cable type MHMS-SI, while the times do not clearly differ for cable type NOMAK-E (figure 9). Definite conclusions are difficult especially for MHMS-SI series because of the small amount of experiments.

In the light of the present experiments and the reflections above, combinations insulated conductor - aluminium shield in contact with earthing conductor seem to be in a special position when compared to combinations insulated conductor - insulated conductor.

The time scale of the “exponential decay” in conductor pair 1-V was compared to the order of magnitude of time constant

\[ \tau = RC \]

for short circuit transient. The short circuit insulation resistance R can be estimated as \( \leq 1 \) k\( \Omega \) and capacitance C was estimated for two cases:

a) two parallel conductors (short circuit between pair of conductor) with

\[
C = \frac{\pi \varepsilon l}{\ln \left[ \frac{a}{2r_c} + \sqrt{\left( \frac{a}{2r_c} \right)^2 - 1} \right]}
\]  

(8)

(Lorrain & Corson 1970) and
b) cylindrical capacitor (short circuit between conductor and shield) with

\[ C = \frac{2\pi \varepsilon l}{\ln \left( \frac{r_u}{r_c} \right)} \]  

(Lorrain & Corson 1970).

Capacitances were calculated using permittivity \( \varepsilon = 5 \cdot 8.85 \) pF/m, cable length \( l = 1 \) m, conductor radius \( r_c = 0.4 \) mm, distance between centres of conductors \( a = 1.5 \) mm (\( = 2 \cdot r_c + 2 \cdot \) insulation thickness) and radius of aluminium tape shield \( r_u = 3.75 \) mm. Both cases yield an order of magnitude for \( C = 0.1 \) nF.

The corresponding time constant is then \( \tau \equiv 0.1 \) \( \mu \)s which is much smaller than the time scale for 1-V disturbances. The observed “exponential decay” in the 1-V resistance curves is thus probably due to heating and reduction of insulation resistance or influence from an outer circuit (data acquisition system) and is not a property of the investigated cable.

### 5.3 Distribution of time to first disturbance

Cumulative failure probabilities for the observed times to first disturbance were estimated using the median ranks suggested by McCormick (1981) and are presented in figure 9.

The observed cumulative frequencies reveal “tails” with higher times to first disturbance at higher cumulative frequencies, with exception for combination 1-V. This appears especially for NOMAK-E cable series and the voltage supply data.

The following cumulative distributions \( F(t) \) were fitted to the observed times (Bunday 1991, McCormick 1981):

- **normal**: \( F(t) = \Phi \left( \frac{t - \mu}{\sigma} \right) \); \[ \Phi(t) = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^{t} \exp \left( -\frac{u^2}{2} \right) du \]  

- **lognormal**: \( F(t) = \Phi \left( \frac{\ln(t) - \mu^*}{\sigma^*} \right) \); \[ \Phi(t) = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^{t} \exp \left( -\frac{u^2}{2} \right) du \]  

- **Weibull**: \( F(t) = 1 - \exp \left( -\left( \frac{t - \tau}{\beta} \right)^{\alpha} \right) \)  

- **Gumbel**: \( F(t) = 1 - \exp \left( -\exp \left( \alpha(t - \beta) \right) \right) \)  

where \( t \) is the observed time to first disturbance, \( \mu \) arithmetic mean and \( \sigma \) standard deviation of the sample (normal distribution), \( \mu^* \) arithmetic mean of \( \ln(t) \) and \( \sigma^* \) standard deviation of \( \ln(t) \) (lognormal distribution), \( \alpha \) shape parameter, \( \beta \) scale parameter and \( \tau \) time-delay parameter (Weibull distribution).
Figure 9. Cumulative failure probabilities for the observed times to first disturbance in a) MHMS-SI cable fire series and b) NOMAK-E cable fire series. In one NOMAK-E case no $U_{supply}$ failure was observed during the 126.9 s long experiment.

All distributions above could be satisfactorily fitted visually to the observed data without time delay. Introducing a time delay to the Weibull distribution gave a still better fit to the observed data as presented below.

The combinations 1-2 and 3-4 in NOMAK-E cable series show three points at exceptionally small times in two successive experiments, 3.7 s for 1-2 and 22.3 s for 3-4 in the same experiment and 22.8 s for 1-2 in the next experiment. The reason for these three disturbances is not clear, but the fact that in the former experiment a disturbance was
noticed in pair 1-2 even before the propane burner was placed beneath the cables may indicate that these exceptionally early disturbances are not due to heating of cables.

If these three possibly spurious disturbances are ignored, there seems to exist threshold values for the distributions. This is reasonable as the burning and consumption of the cable jacket and conductor insulation introduce a delay before insulation resistance has decreased enough for short circuiting to occur. According to this argumentation Weibull distributions with time constant were fitted to observed cumulative distributions giving fairly good results as presented for NOMAK-E cable in figure 10. This effect can be modelled quantitatively, and will be made in a later paper.

The threshold value \( \tau \) from Weibull distributions give thus an estimate of the shortest time to first disturbance in the examined automation cables. The threshold times for power supply \( U_{\text{supply}} \) and current output \( I_{\text{out}} \) in the energised cable C1 and conductor combinations 1-2 and 3-4 in the non-energised cable C2 are in the interval 39.3 ... 41.5 s for MHHI-SI and 36.7 ... 38.0 s for NOMAK-E cable.

6. Conclusions

The present fire experiments give information on times to first disturbance in and the behaviour of the electrical function of a four-conductor instrumental cable connected to a pressure transmitter. Probability functions of disturbance times have been fitted to the observed cumulative frequencies.

The amount of experiments is so small, that no final conclusions on the distributions can be made. However, on rough theoretical grounds Weibull distributions seem preferable, which is not in contradiction with sparsely available experimental information.

The parameters of the distributions have been fitted visually in the present work. There are also computational methods for estimating parameters, but they do not necessarily give more accurate results than the visual fit because of the small amount of observations in the present experimental series.

Both the fire experiments and the failure mode investigation without fire exposure indicate that the transmitter current output signal is not very sensitive to disturbances. Disturbances seem to occur only when the insulation resistance is less than an order of some hundred ohms which can be considered equal to a metal to metal contact. In accordance which this, the experiments show an ON/OFF situation in most of the cases: current signal output is stable until sudden drop to zero. In 6 cases out of 33 the current output exceeded the maximum measurement range 500 mA for a short duration, maximum 9.2 s, after which the current output dropped to zero due to voltage supply cut-off. In three cases a current output level drop of about one mA to 4.6 mA level was noticed. The duration of these level drops were 1.4...12 s.
Figure 10. Weibull distribution fitted to observed cumulative frequencies of time to first disturbance in a) $U_{supply}$ with parameters $\alpha = 1.2$, $\beta = 19.0$, $\tau = 38.0$ s, b) $I_{out}$ with parameters $\alpha = 1.3$, $\beta = 13.0$, $\tau = 37.5$ s, c) conductor pair 1-2 with parameters $\alpha = 1.3$, $\beta = 16.0$, $\tau = 36.8$ s, d) conductor pair 3-4 with parameters $\alpha = 132$, $\beta = 15.0$, $\tau = 36.7$ s and e) conductor pair 1-V with parameters $\alpha = 1.6$, $\beta = 4.3$, $\tau = 17.3$ s.

The present experiments were carried out with a gas burner located 85 mm beneath the centre of the free space between two adjacent cable ladder rungs. They correspond thus to a situation without obstacles such as cable ladder rungs and cable ties, which should be investigated separately.

The present results correspond to a situation with cables in flame contact and are not directly applicable on cable signal behaviour in moderate fire environments. The results
can, however, be used as a conservative estimate. Based on present experimental results fairly reliable estimates of times to first disturbance can be obtained using existing time dependent heat transfer models, and appropriate models for fire exposure other than a direct flame contact.

The results of this study should be considered as merely indicative because of the small amount of experiments in the series.

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References


INTERNATIONAL WORKSHOP ON FIRE RISK ASSESSMENT
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Behaviour of High Efficiency Particulate Air filters in case of fire.

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KEYWORDS:
Fire, HEPA filter, clogging, mechanical strength

ABSTRACT

Among all the confinement equipments found in nuclear plants, several have been the subject of research works at the "Institut de Protection et de Sûreté Nucléaire" (IPSN) in order to characterize their behaviour when they are submitted to stresses (temperature, clogging by combustion aerosols, presence of incandescent particles) resulting from fire. HEPA (High Efficiency Particulate Air) filters require special attention because they are the last confinement barrier of radioactive substances before they are released into the environment.

The proposed paper presents the results specific to the behaviour of HEPA filters in terms of clogging by combustion aerosols and mechanical strength related to effects coupled with temperature and clogging. First, it highlights the need to use experiments for access to realistic data about industrial filter clogging because the theoretical models currently available appear to be limited; fires of solvents used in reprocessing facilities illustrate the influence of the filtration velocity upon the clogging curves. The communication gives subsequently the pressure drop from which an HEPA filter loses efficiency for a given temperature; several types of filters used in French nuclear facilities have been tested.

INTRODUCTION

During the various development phases of a fire, a number of phenomena can occur, liable to affect the static and dynamic confinement functions ensured by the different equipments encountered in the ventilation networks. Among these phenomena, there are the effects of temperature, pressure (overpressure during the ignition of the heart of the fire, negative pressure on extinguishing), increased flow rate, gases and combustion aerosols formed. Within the framework of the safety evaluation of nuclear facilities, it is important to take into consideration the influence of these phenomena on the integrity of the HEPA filters. The constraints evaluated initially at IPSN result from the combined effects of temperature and clogging.

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The implemented experimental program combines two types of study:

- the determination of the clogging power of the combustion aerosols, and more specifically those formed during the combustion of solvents used in the French reprocessing plants;
- the determination of the mechanical strength of the filters; for a given temperature and at the rated flow rate of the filter, the test involves determining the pressure drop from which mechanical breakdown of the filter medium sheets occurs.

These two types of experiments are performed on two complementary test benches:

- a clogging test bench in which small-scale combustible material fires are set;
- the SIMOUN bench on which the clogging is carried out using a test aerosol (sodium chloride) and the thermal constraints produced by the temperature rise of the air passing through the filter; in this way, it is possible to control the coupled "temperature-pressure drop" constraints.

I. EVALUATION OF CONSTRAINTS RELATED TO THE FILTER CLOGGING BY COMBUSTION AEROSOLS

The evaluation of constraints related to the filter clogging by combustion aerosols consists first in determining the pressure drop to which the filter will be exposed in consideration of the deposited aerosol mass. Accordingly, there are two possible approaches:

- the use of phenomenological models if they are suitable;
- recourse to empirical laws specific to a given fuel.

1.1. General information concerning the phenomenological approach

The two major magnitudes in characterizing HEPA filters are their pressure drop (ΔP) and their collection efficiency (E); these quantities depend on the composition of the filter (density, fiber diameter, thickness), on the operating conditions (filtration velocity, temperature) and on the characteristics of the aerosol (type, particle size distribution).

In the case of the dynamic filtration, clogging induces deep modifications in the filter structure causing its efficiency and pressure drop to vary in the course of time. These constraints lead to decreased flow conditions in air networks able of reversing the direction of the flow in some parts of such networks.

During clogging, particles are essentially deposited at the surface and within the porous medium. In addition, the influence of the particle size distribution is a major parameter in clogging; the smaller the diameter of a solid aerosol, the more clogging it will cause; a qualitative illustration is given on Figure 1.
Figure 1: Filter pressure drop as a function of aerosol mass collected for different sizes of solid aerosols

Available phenomenological models

Because of the complexity of the phenomena involved by clogging, there are few authors who propose qualified clogging theoretical models for solid, liquid and fuel aerosols. As far as phenomenological models are concerned, they all derive from laws relative to unclogged filters:

\[ \Delta P = \mu v L_{f} F \]

where:
\( \mu \) is the fluid dynamic viscosity,
\( v \) is the filtration velocity,
\( L_f \) is the length of component fibers of filter per unit of surface,
\( F \) is the friction coefficient.

One of the most widely used models is that of Davies [1], in which the friction coefficient is equal to:

\[ F = 16 \pi \alpha_f \frac{1}{2} (1 + 56 \alpha_f^3) \text{ if } 0.006 < \alpha_f < 0.3 \]
\[ F = 16 \pi \alpha_f \frac{1}{2} \text{ if } \alpha_f < 0.006 \]

where \( \alpha_f \) is the filter medium density.

In the case of clogged filters, two possible approaches are used:

\( \Phi \) Filtration in depth

The filter medium is assimilated to a mixture of two filters: one comprises clean fibers and the other fibers formed from the deposited particles, each having its own parameters (\( F_f \) and \( L_f \) for the clean filter, \( F_p \) and \( L_p \) for the other one); the pressure drop is given by the relation:

\[ \Delta P = \mu v (L_f F_f + L_p F_p) \]
A study by IPSN has revealed the heterogeneous nature of the deposit in its depth; a
differential approach [2] assimilating the filter to a succession of elementary sections with
characteristics varying as a function of time has been developed by combining two models:
- a model giving the efficiency of each elementary section;
- Bergman's model [3] for obtaining the pressure drop of each section.

\[ \Delta P = \Delta P_0 + \Delta P_0 \]

Surface filtration

This approach presupposes that the particles are deposited on the filter surface; the filter
pressure drop is the sum of the clean filter pressure drop (\( \Delta P_0 \)) and that due to the formation
of the cake at the surface of the filter (\( \Delta P_0 \)) [5]:

The current state of knowledge does not enable various authors to specifically indicate the
limits of their model. In addition, to date, all these models have been qualified for plane filters;
when they are applied to industrial filters in which the filter medium is folded, the results are
often very different from reality.

Experimental approach

Several authors have considered the clogging by the combustion aerosols using a purely
experimental approach; to date, the data concerns essentially solid fuels [4], [5]; more recently,
IPSN looked at liquid fuels.

1.2.1. Clogging test bench and associated instrumentation

In order to acquire data about the evolution of HEPA filter clogging by the combustion
aerosols, a test bench has been designed. It consists of a 0.5 m\(^2\) stainless steel enclosure in
which test aerosols are generated. The enclosure is connected to an aeraulic circuit in which
the air flow rate can vary from 50 m\(^3\)/h to 500 m\(^3\)/h. This circuit consists of two parallel
branches which can be used simultaneously or separately. Each of them includes a filtration
housing accommodating a test HEPA filter and a pressure sensor. Downstream of each of
these filters is an orifice plate controlling the air flow rate by differential pressure measurement.
In each of the two circuits, a valve maintains a constant aeraulic flow rate during a test. The
main duct is provided with instrumentation for various sampling and measurement operations:
oxygen concentration, size distribution of the combustion aerosols, aerosols concentration
from sampling on filters, air temperature. In addition, a small branch of the main duct is used
for studying the clogging of plane filters. Finally, all the experimental data are recorded by an
acquisition unit. Figure 2 is a schematic diagram of the test bench and of the associated
instrumentation.
1.2.2. Examples of significant results

Because the existing models are not validated for the cases concerning combustion aerosols, an experimental study was put underway. To present typical results specific to combustion aerosols, we will be investigating the clogging of pleated HEPA filters by aerosols resulting from the combustion of a mixture comprising 30% by volume of TBP (Tributylphosphosphate) and 70% of TPH (dodecane mixture).

Influence of the HEPA filter medium type

First, the study concerns the clogging of plane and pleated filters at various filtration velocities. The experimental parameters of the study are given on table 1.

<table>
<thead>
<tr>
<th>Type of filter medium</th>
<th>Filtration velocity (cm/s)</th>
<th>Typical size distribution</th>
<th>Average concentration (g/m³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plane</td>
<td>0.23</td>
<td>$d_a = 1 \mu m$</td>
<td>0.3</td>
</tr>
<tr>
<td></td>
<td>0.92</td>
<td>$\sigma_g = 4$</td>
<td>0.15</td>
</tr>
<tr>
<td></td>
<td>2.1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pleated</td>
<td>0.23</td>
<td>$d_a = 1 \mu m$</td>
<td>0.3</td>
</tr>
<tr>
<td></td>
<td>0.92</td>
<td>$\sigma_g = 4$</td>
<td>0.15</td>
</tr>
<tr>
<td></td>
<td>2.1</td>
<td></td>
<td>0.1</td>
</tr>
</tbody>
</table>

$d_a$: mass median aerodynamic diameter - $\sigma_g$: geometric standard deviation

Table 1: Experimental parameters
In order to compare the results together, the clogging of the filters is characterized by the function:

\[
\frac{R}{R_0} = f(M_{ae})
\]

where:
- \( R_0 \) is the clean filter resistance,
- \( R \) is the clogged filter resistance for a mass \( M_{ae} \) of deposited aerosols.

The filter resistance is defined by the following relation:

\[
R = \frac{\Delta P \cdot \mu}{Q_v \cdot \mu_0}
\]

where:
- \( \Delta P \) is the filter pressure drop,
- \( Q_v \) is the air volumetric flow rate through the filter,
- \( \mu, \mu_0 \) are the dynamic viscosity of air through the filter at \( T \) and ambient temperatures.

The obtained results, as shown in Figure 3, demonstrate that there is a difference in clogging between the two filter types; the filtration velocity of 2.1 cm/s corresponds to the rated filtration velocity.

![Figure 3: Clogging of plane and pleated filters with respect to combustion aerosols](image)

The trend curves indicating plane filter clogging demonstrate that the influence of the filtration velocity is barely marked. In addition, the \( R/R_0 \) ratio is a relatively linear function of the mass of aerosol deposited on the filters in the range of mass deposited [0 to 16 g/m²].
Conversely, the evolution curves for pleated filter clogging depend considerably on the rate of filtration. For a low filtration velocity (0.23 cm/s), the $R/R_0$ ratio is in the form of an exponential function of the mass of aerosol deposited; at the rated velocity (2.1 cm/s), clogging is similar to that of the plane filter.

The results given also show that even one of the most recent clogging models [3] imperfectly represents the clogging of a pleated filter by combustion aerosols. For all the current clogging models, the $R/R_0$ ratio gives a relationship independent of the filtration velocity. The insufficiency of these models therefore shows the need to continue research into this field and to perform experiments. Complementary experiments have demonstrated that the influence of the filtration velocity on the pleated HEPA filter clogging curves can be generalized to solid aerosols, liquid aerosols and combustion aerosols.

Careful attention to the surface of the various filters used reveals that the deposit of aerosols on the filter medium is not uniform in the case of pleated filters. Indeed, at high velocity, the deposit is homogeneous and corresponds more to filtration in depth. Under these conditions, flow of air is over the whole front surface of the filter, in the same way as for a plane filter. Conversely, at low velocity, the deposit is far less homogeneous and corresponds more to surface filtration (Figure 4). The following hypothesis, which has yet to be confirmed, is that part of the filtering surface located in the depth of the filter pleats is no longer crossed by the filtration air. This decrease in the filtration surface would tend to explain the phenomenon of an exponential increase in the $R/R_0$ ratio as a function of the mass of aerosol deposited. If the hypothesis brought in previously is verified, the development of a pleated filter clogging model with a particularity of a smaller filtration surface as a function of time is liable to give predictive results more representative.

Knowing the particulate source term derived from the fire (mass flow rate of the combustion aerosols formed), the empirical relations $R/R_0 = f(M_a)$ should make it possible to calculate the mechanical constraints affecting the filters in terms of pressure drop.

![Figure 4: Deposit of aerosols on the pleated filter medium](image1)

velocity: 2.1 cm/s

velocity: 0.23 cm/s

Figure 4: Deposit of aerosols on the pleated filter medium
II. MECHANICAL STRENGTH OF FILTERS TO CLOGGING

The mechanical strength tests are carried out on the SIMOUN bench designed to test filtration elements affected by high temperatures air. It operates in negative pressure and is designed to supply 4 000 m³/h of clean air at 400°C.

II.1. Main characteristics of the test bench

AIR

\[
\begin{align*}
\text{quality:} & \quad \text{filtered atmospheric air} \\
\text{temperature:} & \quad \text{included between } 20^\circ \text{C and } 400^\circ \text{C} \\
\text{flow rate:} & \quad \begin{cases} 
\text{up to } 4 000 \text{ m³/h at } 400^\circ \text{C} \\
\text{up to } 10 000 \text{ m³/h at ambient temperature}
\end{cases}
\end{align*}
\]

TEST AEROSOL:

\[
\begin{align*}
\text{ambient temperature:} & \quad \text{soda fluorescein} \\
\text{high temperature:} & \quad \text{sodium chloride (up to } 400^\circ \text{C)}
\end{align*}
\]

The various thermal conditions that can be imposed upon the filter tested are as follows:

1. filter exposed to ambient temperature;
2. bypass of the filter element at high temperature so as to subsequently expose it to thermal shock;
3. gradual rise of air temperature reaching the filter.

The filters are exposed to constraint 2 during the study of mechanical strength.

Sodium chloride (NaCl) is used as aerosol for these tests because it allows operation at high temperatures and gives an instantaneous aerosol concentration value. The sodium chloride production uses a stick thermal generator; a stick of sodium chloride is inserted into an oxygen-propane burner which vaporizes a constant amount of sodium chloride; this vapor then recondenses into a cloud of particles, the mass flow rate of which is defined by the movement velocity of the stick. The apparatus can operate at several different velocities, capable of producing from 0.5 to 5 g/min of sodium chloride aerosols.

Concentration measurements upstream and downstream of the filter are made using a flame photometer able of measuring concentrations ranging from $10^{-2}$ to $10^{-8}$ g/m³.

II.2. Principle of mechanical strength tests by clogging

The mechanical strength test by clogging consists in increasing the pressure drop of the filter at the rated flow rate of the filter by the emission of the NaCl aerosol until the rupture of the filter medium occurs; rupture is confirmed by the discontinuity of the pressure drop and an increase in the concentration of the test aerosol downstream of the filter.

The HEPA filters tested are mini pleats type filters currently used in French nuclear installations; their main characteristics are summarized in the table 2.
Table 2: Characteristics of filters tested

The mechanical strength is determined in the range of temperatures from the ambient temperature to a limit temperature characteristic of the sealant nature, respectively 200°C and 120°C for mineral and polymer sealants.

II.3. Mechanical strength curves

For each type of filter, it is possible to obtain a mechanical strength curve $RM = f(T)$ when the filter is exposed to temperatures less than or equal to its use limit temperature; these curves are shown in Figures 4 and 5.

![Mechanical strength curve](image-url)

Figure 4: Mechanical strength for the filters with mineral sealant
Figure 5: Mechanical strength for the filters with polymer sealant

Whatever the imposed temperature, the filters with the highest pleats height offer better mechanical strength than the others.

In addition, for each test, the B and D filters are exposed to the thermal constraint longer and represent a greater mass of deposited aerosols (because of their higher mechanical strength), thus boosting their performance even more compared to A and C filters.

The mass of sodium chloride aerosol ($\bar{d} = 0.1 \, \mu m$) deposited on a filter type B and D before the rupture of the filter medium is included between 12 g/m² and 25 g/m², and for a filter A and C between 6 g/m² and 22 g/m².

III CONCLUSION

The approach at IPSN used to characterize the behaviour of HEPA filters during a fire consists first in acquiring data about clogging by combustion aerosols, then determining from what "pressure drop - temperature" coupled stresses the filter loses its efficiency. The complexity of the phenomena involved means having recourse to analytical experiments. The experiments carried out today reveal that the clogging of the filters depends on the filtration velocity which will probably decrease in the event of fire because of the process of clogging itself. In addition, the mechanical strength of the tested filters depends on the height of the filter medium pleats.

Work is yet to be done on this subject, in particular that of getting to know the behaviour of filters exposed to the serious effects of overpressure and negative pressure resulting from fire.
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SIMPLIFIED METHOD FOR RISK ORIENTED DESIGN OF
STRUCTURAL FIRE PROTECTION MEASURES

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ABSTRACT

A simplified method for a more risk oriented design of structural fire protection measures in nuclear power plants (NPP) has been developed on behalf of the German Federal Government. This method is based on a calculated fire load density \( q_a \). The burning behaviour of combustibles is described by the efficiency of combustion \( \chi \), which is determined in laboratory scale experiments.

Depending on the ventilation conditions, the room geometry, fire load density and fuel distribution, the heating up of structural elements is calculated by means of a fire simulation zone model in multiple room configuration. The estimated temperature distribution within the structural element is compared with the temperature distribution gained by the fire conditions described by the standard temperature-time curve (ISO 834). As a result of the design calculations, an equivalent time of fire exposure as a function of the fire load density, the geometrical and physical properties of the compartment structures, as well as of the ventilation conditions is revealed. In addition, heat sinks (e.g. large machines or reservoirs/vessels with fluids) can easily be considered by this methodology.

The simplified design method needs only few empirical functions and design curves which are derived from systematic fire simulation calculations and which are easily understandable and applicable. The simplified method for such a risk oriented design of structural fire protection measures yields results which are in good correspondence with results from large-scale room fire tests. This method is therefore superior to most of the competitive design methods, e.g. the method as applied in the German standard DIN 18230-1, with respect to the physical basis as well as to aspects of the practical applicability.
1 INTRODUCTION

On behalf of the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), a basic study on the applicability of simplified methods for the design of structural fire protection features in the nuclear fire protection standard (KTA 2101.2 [1]) was performed [2]. With regard to the burning behaviour, the study included a detailed analysis of fire tests with oil and cable insulation material as fire load. For the application of these experimental data to any other fire load suitable criteria were worked out.

Based on the derived burning models in case of ventilation controlled fires, calculations with a zone model were conducted. The boundary conditions concerning fire load density and ventilation conditions were systematically varied in a realistic range.

Based on these calculations, a methodology for a rough estimate of the equivalent time of fire exposure $t_e$ as a function of the fire load density, the room geometry and the ventilation conditions was developed. Related to an appropriate fire safety concept, this method is applicable to NPP as well to industrial buildings or other buildings with special architecture and use. However, the equivalent time of fire exposure is the period of time, in minutes, during which the same thermal load is attained in a structural element by a standard fire exposure (like ISO 834 standard test fire) as would be attained by a natural fire exposure.

The following presentation gives some fundamental findings of the calculations performed (see [2]). Furthermore, the simplified calculation method alternative to the German DIN 18 230 [3] standard will presented.

2 COMPARTMENT FIRE CALCULATION

2.1 CALCULATION MODEL

Referring to former investigations applying a one-zone model [4] for the development of a simplified design method of structural fire protection features, the study presented here [2] used the multiple room zone model FIGARO [5]. FIGARO (FIre and GAs movements in ROoms) was already applied in connection with real scale fire tests described in [6, 7, 8]. This type of zone model is capable to predict the conditions of a post-flashover fire for a part of a large fire compartment affected. The temperatures calculated with a two-zone multiple room model for the near field of the fire source are plainly higher than the temperatures calculated with the single zone model. The latter may be used only for volumes of smaller size calculating uniform temperatures over the whole compartment.
According to the single zone calculations, the deviations of the average thermal effects depend on the fire scenario. If the fire is limited to a particular area of the building because of the fire load arrangement or the effect of fire fighting activities, temperatures will increase to a high level. In the other areas of the building affected by the fire the effects on the structures will not be significant.

At least for buildings with a floor area larger than 100 m² it can be expected that a higher amount of the fire load is located on a smaller spot and will burn down there. In this case, locally high temperatures can arise from such a ventilation controlled fire lasting for a longer time period. This has a twofold negative effect on the structures in the fire near field.

If the fire load is homogeneously distributed all over the compartment and the fire extinguishing starts late, the fire will spread over the whole room. The fire effects on structural elements are nearly the same all over the compartment. Effects due to the spread of fire will have no practical relevance.

To work out the systematic differences between a locally limited fire and an almost homogeneously distributed fire load, different models of the simulated building have been used. The upper part of figure 1 shows the model for a fire load not distributed all over the fire compartment: The overall fire load is accumulated on approximately 1/3 of the floor area of the fire compartment. The compartment itself is divided into three segments interactively connected by vents. The connections to the outside are modelled by vertical doors. The area of the doors can be varied in a wide range to find out the most unfavourable conditions.

The model for a homogeneously distributed fire load is shown in the lower part of figure 1. In this case, the fire load is located in almost 2/3 of the floor area of the building area. In both cases, it is assumed that the whole fire load is accumulated in the fire compartment.
Figure 1: Multiple room configuration for a homogeneously distributed fire load (lower part) and fire load accumulated in a part of the fire compartment (upper part)

2.2 INFLUENCE OF DIFFERENT COMBUSTIBLES

For the above defined fire scenarios, mass loss rates were assigned which were derived from fire tests with best combustion conditions, in particular sufficient oxygen supply and high combustion temperatures. The rooms analysed in NPP normally have less natural ventilation openings and the mechanical air supply is restricted, too. Thus, the combustion is mainly ventilation controlled. Under these assumptions the theoretically possible maximum mass loss rates of the combustible materials have no relevance because the mass loss rates (and therefore heat release rates) are limited by the lack of air (oxygen). The different velocities for flame spread from different burning materials only influence the heating of structural elements during the pre-flashover fire phase, but not during the post-flashover phase.

It was determined that oil pool fires yield higher equivalent times of fire exposure $t_e$ than cable fires being attributed to a lower combustion efficiency of cables. The relationship between theoretical heat of combustion $\Delta H_c$ and actual heat of combustion $\Delta H_{c,eff}$ due to the conditions of a real fire is called combustion efficiency $\chi$. In case of solid combustibles $\chi$ mainly depends on material properties and fire load density. By use of this matter for heat balance calculations, the combustion efficiency can indirectly be applied by the reduced
actual heat of combustion $\Delta H_{c,\text{eff}} = \Delta H_c \cdot \chi$. Data for $\Delta H_c$ can be taken from e.g. [9].

In view of the difficulties for revealing fixed values for realistic heat release rates and effective heat of combustion for all realistic fire loads and the respective geometric configurations, it seemed to be more useful to calculate the equivalent time of fire exposure for only one representative fire load under optimal combustion conditions. For the well known high combustion efficiency, oil was chosen as reference fire load for the heat balance calculations [2]. Furthermore, oil is well investigated, therefore the uncertainties due to input parameters used for the heat balance calculations were low. For other combustibles, the same equivalent time of fire exposure is achieved in case that $\chi \leq 1.0$ (assuming the same ventilation conditions and therefore the same heat release). The reason is the approximately constant relationship between heat of combustion and stoichiometric fuel to air ratio of combustible (figure 2). Therefore, the different burning behaviour of combustible materials can be described on the one hand by the stoichiometric fuel to air ratio and, on the other hand, experimentally based on realistic configurations by "measuring" the combustion efficiency $\chi$.

![Graph](image)

**Figure 2:** Relationship between heat of combustion and stoichiometric fuel to air ratio

A suitable testing method for the experimental determination of the combustion efficiency, respectively the actual heat release rate, on a realistic scale was developed at iBMB [10]. Data concerning pure materials can be found in [11] gained from investigations by Tewarson with the TEWARSON-Calorimeter. According to these investigations, the combustion efficiency of solids is in the range between 0.4 and 0.7.
2.3 INFLUENCE OF FIRE LOAD DISTRIBUTION ON THE EQUIVALENT TIME OF FIRE EXPOSURE

To estimate the influence of the fire load distribution (fire scenarios) three different room configurations with a unique reference height \( H_{\text{ref}} = 2.5 \text{ m} \) were examined:

a) Fire load homogeneously distributed over the fire compartment; configuration according figure 1 (lower part).

b) Fire load not homogeneously distributed, but accumulated in a larger area of the fire compartment; according to figure 1 (upper part), three rooms of 50 m² each were combined. The fire load of the entire area (here 150 m²) was burnt in the larger part.

c) So-called "point fire source" with locally limited burning area; in this case, five room segments of 50 m² each were combined. Compared with case b) the fire load (placed in the central segment) increased with 2/3 of fire load b) responded with a longer fire duration, followed by higher temperatures of structures.

The results of the heat balance calculations are summarised in figure 7. The following sections will discuss the differences in equivalent time of fire exposure regarding the above mentioned three fire scenarios.

2.4 INFLUENCE OF FIRE COMPARTMENT FLOOR AREA

The influence of the fire load accumulation according to the three fire scenarios of section 2.3 is modelled by burning the entire fire load in the middle room of the configuration. One input parameter is the fire load density \([\text{MJ/m}^2]\). The result is that for rooms with different floor areas the fire loads are different even if the fire load density is the same. But using the same burning area in the middle of the room configuration, results depending on fire compartment floor area can be achieved. It was therefore investigated if this dependency still exists in case that the part of burning area related to the entire floor area is kept constant. The equivalent time of fire exposure was calculated for rooms with 150 m² and 450 m² (height 2.5 m and 5.0 m²) with the assumption 2/3 of entire area as burning area each. For this case, the results in figure 3 show that the equivalent time of fire exposure is almost independent from the floor area considering homogeneously all over the room distributed fire loads. Thereby, the main assumption for the validity of DIN 18 230-1 (German Standard for Structural Fire Protection in Industrial Buildings [3]) can be demonstrated.
2.5 INFLUENCE OF ROOM HEIGHT

The results shown in figure 3 depict the fact of lower equivalent time of fire exposures in case that the room height increases. This is mainly based on the longer distance for the air entrainment into the plume transporting the combustion gases into the hot gas layer. The larger amount of "cold" ambient air causes a lower temperature of the hot gas layer.

It had to be checked whether the influence on the equivalent time of fire exposure for the room height assuming nearly homogeneously distributed fire load is even present in the fire scenario b) (burning area is almost 1/3 of the fire compartment with the entire floor area). For that purpose, three calculations were compared (150 m² fire compartment floor area each):

- $H = 2.5\, \text{m}$  \quad $H_v = 2.0\, \text{m}$ ($H_v$ = height of ventilation opening)
- $H = 5.0\, \text{m}$  \quad $H_v = 3.0\, \text{m}$
- $H = 5.0^*\, \text{m}$  \quad $H_v = 2.0\, \text{m}$

Figure 4 shows the respective calculated equivalent time of fire exposure.

Therefore, the lowest thermal load can be observed in a compartment with a larger room height and a low ventilation opening height $H_v$. In this case, the thickness of the hot gas layer is higher, the temperatures are lower. This comparison validates the results of previous investigations [12] for which the following relation between equivalent fire duration and room height was found:

$$ t_e(H) = t_e(H_{ref}) \cdot \left(\frac{H_{ref}}{H}\right)^{0.3} \quad (1) $$

If $H_{ref}$ is equal to a reference height of 2.5 m a room height of 5.0 m revealed the following $t_e$:

$$ t_e(5.0\, \text{m}) = t_e(2.5\, \text{m}) \cdot 0.81 $$

This conservative approximation, even adopted to DIN 18 230-1 [3] (with a reference height $H_{ref} = 6.0\, \text{m}$) reveals values being approximately 5% higher than the medium curve in figure 4, itself being conservative referring to the influence of the height of the ventilation openings. Therefore, the equivalent time of fire exposure for other room heights than 2.5 m can be conservatively calculated based on the assessment curves for the reference height $H_{ref} = 2.5\, \text{m}$. 

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Figure 3: Equivalent time of fire exposure in rooms; entire floor area between 150 m² and 450 m², room height 2.5 m and 5.0 m

Figure 4: Equivalent time of fire exposure for locally limited fires in a room of 150 m² floor area; room height 2.5 m and 5.0 m; height of ventilation opening \( H_v = 2.0 \) m, 3.0 m and 2.0 m (from top to bottom)

2.6 INFLUENCE OF NATURAL VENTILATION OPENINGS

Ventilation conditions directly affect the mass loss rate. If natural ventilation through openings and mechanical ventilation by HVAC (Heating, Ventilation & Air Conditioning) are available in general, the natural ventilation strongly affects the mass loss rate.

To investigate this important aspect, the total area of ventilation openings \( A_v \) was varied in the range from 1.3 m² to 40 m². The fire compartment floor area was assumed to be 150 m², the room height 2.5 m. Besides the natural ventilation, a mechanical ventilation input air
stream of 3000 m³/h was supplied. This is equivalent to an air exchange rate of 8 times per hour. Example c) is applied as relevant fire scenario of an oil pool fire with a burning area $A_b = 40$ m² in the middle segment and a mass loss rate being nearly constant.

Figure 5 reveals a maximum value for the equivalent time of fire exposure for rooms with ventilation areas between 2.6 m² and 3.9 m² (approx. 2% of the fire compartment floor area). Obviously, an optimum appears among the counter effects

- increasing air supply followed by increase of mass loss rate, and
- increasing ventilation opening causing intensified heat venting.

This fact can be explained by means of figure 6 where the mass loss rate $m_b$ and the equivalent time of fire exposure $t_e$ versus vent area $A_v$ are shown. The calculations are based on a fire load density of 1500 MJ/m² in a fire compartment of 150 m² floor area. The mass loss rate (bottom curve) is continuously increasing with the ventilation area. For more than 4 m² ventilation area (approximately 3% of the floor area) however, the cooling effect of heat venting is dominating. Thus, the curve of the equivalent time of fire exposure decreases; the decrease of the equivalent time of fire exposure is considerably smaller than assumed in DIN 18 230 [3]. In contrary, the mass loss rate was kept constant for the present investigations in the frame of the DIN concept.

![Figure 5: Equivalent time of fire exposure - locally limited fire scenario with fire compartment floor area of 150 m² and room height of 2.5 m - natural ventilation openings were varied](image)

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Figure 6: Mass loss rate (bottom) and equivalent time of fire exposure (top) - locally limited fire scenario with floor area of 150 m² and room height of 2.5 m - natural ventilation openings were varied

3 SIMPLIFIED DESIGN METHOD FOR STRUCTURAL FIRE PROTECTION MEASURES

3.1 GENERAL INFORMATION

Based on heat balance calculations (see Section 2) and documented in detail in [2] with different fire scenarios and fire loads under optimum combustion efficiency, the following practical design method for structural fire protection measures was developed. In this context, a proposal for generalisation to other fire loads of lower combustion efficiency was made. The design method was included in the appendix of the German nuclear fire protection standard KTA 2101.2 "Fire Protection in Nuclear Power Plants; Fire Protection of Structural Elements" [1]. In case that some values of combustion efficiency will be continuously complemented, it might be a useful alternative to DIN 18 230 [3]. The efficiency of this design method compared to the validation method of DIN 18 230 or the draft standard DIN V 18 230, part 1 [13] is outlined in [14].

3.2 INPUT PARAMETERS FOR THE SIMPLIFIED DESIGN METHOD

With regard to the required input parameters, the design method presented here uses a similar approach to DIN 18 230-1. These input parameters are:

- the compartment floor area $A$ [m²]
- the room height $H$ [m]
- the area of vertical ventilation openings $A_v$ [m²]
- the air input flow of mechanical (forced) ventilation $V_{\text{in}}$ [m³/h]
- the mass of unprotected fire loads \( M_i [\text{kg}] \)
- the mass of protected fire loads (in pipe systems, vessels, machines etc.) \( M_\text{p} [\text{kg}] \)

From the masses \( M_i \) of the different fire loads, the heat of combustion \( \Delta H_{ci} \) in combination with the combustion efficiency \( \chi_i \), and the combination factor \( \psi_i \) for protected fire loads the overall fire load \( Q_{\text{ges}} \) will be calculated:

\[
Q_{\text{ges}} = \sum_i M_i \cdot \Delta H_{ci} \cdot \chi_i + \sum_j M_j \cdot \Delta H_{cj} \cdot \chi_j \cdot \psi_j
\]  

(2)

The calculated fire load density is estimated from the ratio total mass of combustibles to fire compartment floor areas \( A \):

\[
q = \frac{Q_{\text{ges}}}{A}.
\]  

(3)

In this context, it has to be mentioned that the values for the heat of combustion \( \Delta H_c \) can be taken from appendix 1 of DIN V 18 230 [13] or from other literature (e.g. [9]).

Instead of the burning factor \( m \) according to DIN 18 230 which is often criticised the physically more sound combustion efficiency \( \chi \) is used in simplified method developed for NPP, determined from fire tests with representative fire load configurations. The reference value \( \chi = 1 \) belongs to oil pool fires and is approximately valid for wood crib fires, too. For mixed fire loads, if special fire tests are missing, the maximum of the values for each single fire load has to be taken.

For fire loads being fire protected e.g. by encapsulation or protective coatings of the fire loads a combination factor \( \psi \) according to DIN 18230 [3] may be used in equation (2):

\[
\psi_i = \begin{cases} 
0.8 & \text{for the biggest protected fire load and} \\
0.55 & \text{for all other protected fire loads.}
\end{cases}
\]

3.3 APPLICATION OF THE METHOD

The basic value \( t_{e,0} [\text{min}] \) of the equivalent time of fire exposure as a value for the maximum thermal load to be expected on structural elements in the course of natural fires is determined depending on the fire load density according to equations (2) and (3) which is valid for the reference room height \( H_{\text{rel}} = 2.5 \text{ m} \) and the most unfavourable ventilation conditions, \( t_{e,0} \), which can be taken from the assessment curve in figure 7.

The following fire scenarios (according to Section 2.3) can be distinguished:

a) homogeneously distributed fire load; almost the entire floor area of the fire compartment
is involved in the fire,
b) not homogeneously distributed fire load; the fire area taking a larger part of the floor area,
c) fire load as "point source" with locally limited burning area and the overall fire load accumulated there.

The actual room height $H$ and the actual ventilation conditions may be taken into consideration by means of two reduction factors $\leq 1$:

$$ t_c = t_{c,0} \cdot f_H \cdot f_{AV} $$  \hspace{1cm} (4) 

The reduction factor $f_H$ concerning the actual room height $H$ is derived from equation (1) to be:

$$ f_H = \left( \frac{H_{eff}}{H} \right)^{0.3} $$  \hspace{1cm} (5) 

The reduction factor $f_{AV}$ concerning the actual ventilation conditions is available as a function of the effective ventilation area $A_{V,eff}$ shown in figure 8. This curve derived from the calculations in [2] represents the enveloping curve. Despite the fact that the rooms are assumed to be air-tight, there are always small leaks. Due to these leaks, the curve has a lower boundary value of $f_{AV} = 0.5$. 

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Figure 7:  Equivalent time of fire exposure $t_e$ as a function of the fire load density for the three different fire scenarios a) to c)

Based on calculations comparing fire scenarios with and without mechanical (forced) ventilation, the mechanical air supply $V_{aw}$ [m$^3$/h] can be converted into an equivalent vertical ventilation opening $\Delta A_{aw}$ by means of equation (6)

$$\Delta A_{aw} = \frac{V_{aw}}{6000} \quad \text{[in m}^2\text{]}.$$

(6)

Analogue to the former DIN V 18230 [13], the effect of horizontal ventilation openings in the ceiling may be deduced to a roughly equivalent vertical ventilation opening $\Delta A_{aw}$. In this
context, the investigations of [7, 12] should be considered. In that way, a combination of the natural ventilation by vertical and horizontal openings and of the forced ventilation can be taken into account according to equation (7):

\[ A_{v,\text{eff}} = A_v + \Delta A_{h} + \Delta A_{zu} \]  

(7)

![Graph showing reduction factor \(f_{AV}\) considering the ventilation conditions.](image)

Figure 8: Reduction factor \(f_{AV}\) considering the ventilation conditions.

3.4 PARTICULAR PROBLEMS

The simplified design method offers multiple possibilities to solve detailed design tasks. For nuclear power plants, the influence of big solid structural elements or voluminous tanks has to be analysed in many cases. These elements in the near field of the fire may act as heat sinks and reduce the effects of heat release to the structural elements. The heat loss to the walls and through the ventilation openings were investigated. Besides the heat loss to other heat sinks was analysed. It was generally assumed that walls and ceilings consist of homogeneous materials, such as concrete or brick walls. Corresponding to DIN 18 230-1 [3], this case will be rated with the conversion factor \(c = 0.2 \text{ min-m}^2/\text{kWh}\). If the structural elements of walls or ceilings have other thermophysical properties, the equivalent time of fire exposure can be transformed by multiplication with \(c/0.2\), with \(c\) as referenced in DIN 18 230-1 [3].

If several heat sinks have to be considered, the energy released form the overall fire load \(Q_{ges}\) (see equation (2)) will decrease by the heat losses \(\Delta Q_i\) transferred to each of the heat sinks in the room. Due to theoretical investigations, the energy stored in the structural
building construction elements can be calculated by estimations given in [2] including the following input parameters:

- thermophysical parameters of the heat sinks,
- ratio relation of flamed surface to volume of heat sink,
- temperature of ISO 834 curve corresponding to $t_e$ (first iteration step).

Again, the equivalent time of fire exposure $t_{e,0}$ will be determined for the effective fire load

$$q_{\text{eff}} = (Q_{\text{gas}} + \sum \Delta Q_i)/A.$$  

Another peculiarity is the safety concept for safety assessment of structural fire protection measures which uses statistical data for the fire occurrences frequency and functional capability of fire protection measures in contrary to DIN 18 230 [3]. Nevertheless, the safety concept of DIN 18 230 applied to industrial buildings can be adopted without any changes.

Finally, the simplified design method offers the possibility to accompany the course of fire: The fire development starts with a locally limited fire, extending to a fire involving a larger area of the original fire compartment up to a fire covering the entire floor area.

4 CONCLUSIONS

The method outlined in this presentation represents a simplified approach for a risk oriented design of structural fire protection features by calculation of the equivalent time of fire exposure $t_e$. This approach starts with a calculated fire load density similar to the non-nuclear German fire protection standard for building construction DIN 18 230-1 [3].

However, the burning behaviour of combustibles is not modelled with the in that context criticised combustion factor $m$, but with the physically more sound combustion efficiency $\chi$. The combustion efficiency describes the real amount of energy released by combustion; this amount can have a maximum value of 1.0. In practise, under real boundary conditions, $\chi$ will range from 0.4 to 0.7. More detailed investigations have been carried out in a test furnace similar to the so-called Room-Corner-Test [10] only for some materials.

This simplified design method for structural elements needs only few assessment curves derived from extensive heat balance calculations. The input parameters for structural fire protection features are coherent to those applied to DIN 18 230-1 [3]. In contrary to DIN V 18230-1 [13] and DIN 18230-1, this method reveals results in good agreement with large-scale tests (see [12, 14])
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- Leningrad NPP Unit 3 Deterministic Fire Hazard Analysis Methodology and its Preliminary Results - Mr. Aleksandre Yepikhine, Mr. A. M. Sapozhnikov, Leningrad NPP, Russia and Dr. Antti Norta Fortum Engineering, Ltd., Finland

- Operating Issues for French NPPs In Severe Fire Situations - M. Maurice Kaercher, M. Jean Paul Chatry, Electricité de France (EdF) - SEPTEN, France
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Modelling of the interaction between ventilation and fire

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Fire, ventilation, confinement, modelling

ABSTRACT

In the nuclear industry, there is considerable interaction between fire and ventilation because the latter must keep on ensuring its function of radioactive substances confinement without contributing to the spread of fire. To take this interaction into account, the "Institut de Protection et de Sûreté Nucléaire" (IPSN) has been running a program of research for several years, combining the development of calculation codes and the performance of experiments. The present paper starts with an outline of the functionalities of the first version developed code based upon computer coupling of the FLAMME_S code (zones type fire code) with the SIMEVENT code (ventilation network code). It goes on to describe an example of comparison between the results supplied by the code and those obtained from experiment in a ventilated laboratory facility. Finally, it illustrates the potential of the tool so as to permit studies to be carried out into ventilation control best suited to reduce the impact of fire on ventilation network components.
INTRODUCTION

Fire is the most likely accident liable to occur in a nuclear plant. Therefore, it represents a major risk whose consequences may lead to the rupture of the static confinement and to the dispersion of radioactive material through the facilities or even into the environment. In the nuclear industry, the confinement of contamination is handled by a cascade of negative pressure stages between the enclosures containing radioactive materials and the outside. The cascade is obtained by ventilation which, in case of fire, must keep on ensuring confinement without contributing to the spread of the fire. Because of the diversity of the configuration among nuclear facilities (especially among laboratories and plants) and the complexity of the physical phenomena involved, it appears essential to set up qualified simulation tools enabling the consequences of a fire to be evaluated and integrating the strong interaction between ventilation and fire.

To address these requirements, IPSN has coupled the FLAMME_S and SIMEVENT computer codes now used for safety analysis.

The choice of ventilation code naturally went to SIMEVENT, firstly because of its various qualified functionalities, dedicated to ventilation and transfers of species, and secondly because of the many studies that have been carried out using this code in the normal and degraded operating mode of an installation. Indeed, before analysing the consequences of accidental situations in a nuclear installation, it is important to have thorough knowledge of the normal state in terms of the ventilation and confinement of the associated radioactive substances.

Similarly, the choice of fire code went to FLAMME_S, which is the IPSN reference code in terms of fire and has been the subject of developments qualified by many full-scale experiments at the Nuclear Research Centre of Cadarache.

The available coupling version associates the various functionalities of FLAMME_S (version A.3.2) and SIMEVENT (version 5.19); it should therefore make it possible to allow safety studies to be carried out to integrate quantitatively the ventilation-fire interaction, in particular by calculating:

► the quantities characterizing the development of a fire in a room and its impact on the surrounding structures,
► the evolution of temperature, pressure and combustion products in the fire room,
► the concentration of gas and particulate species throughout the network, taking into consideration deposits in the ducts and the air cleaning equipments,
► thermal and mechanical stresses affecting the various ventilation equipments (temperature and pressure drop at the last air cleaning level, for instance).
1. DESCRIPTION OF THE FLAMME_S-SIMEVENT CALCULATION CODE

1.1. General description of the FLAMME_S code

*Main features*

The FLAMME_S [1] computer code is being developed to be used in the assessment of fire risk in nuclear facilities. The main purpose of this program is to predict the development of a fire and the resulting conditions within a compartment in terms of gas pressure, species concentrations and temperature (gases, walls, etc.). The Figure 1 shows the different kinds of problem that should be treated with FLAMME_S: a fire occurs in a compartment that may be connected with other rooms by means of a ventilation network or a door; the elements (electronic cabinets, cable trays, etc.) contained in the enclosure can be damaged by the thermal stress due to the fire.

![Figure 1: Fire scenario](image)

FLAMME_S has been qualified from many large fire experiments achieved by the IPSN [2] and from experimental data available in the literature.

*Physical models and basic assumption*

FLAMME_S is based on a two zones model [3, 4]. The room is divided into two distinct but homogeneous zones. The upper layer contains the hot gases produced by the fire and the air entrained by the plume; these gases are floating over the "cold" gases of the lower layer as a result of the thermal stratification due to buoyancy (Figure 2).

![Figure 2: Two zones model concept](image)
The conditions within each layer are derived from the conservation laws applied to each zone. The mass and energy balance equations together with the perfect gas state equation enable the determination of the two layers temperatures \( (T_u, T_l) \), the pressure of the compartment \( (P) \) and the height of the interface \( (H) \). The source terms of these equations express the mass and energy exchanges between the two layers, between the compartment and the outside, between the gases and the walls and between the fire and its surroundings.

The thermal plume generated by the fire carries mass, energy and species from the lower layer into the upper one; these transfers and the decrease of the temperature along the plume axis can be provided by different plume models \([5, 6]\). Flows through openings (in natural convection) or ventilation network (in forced convection) are simply deduced from the pressure difference between the enclosure and the outside by using Bernoulli’s equation. At last, radiative and convective heat transfers between the gas layers and the walls, fires and "targets" are calculated. The radiative energy leaving the flame is a fraction of the total heat release rate and is determined by assuming an energy point source located at the half of the flame height. The heat diffusion equation is solved to follow the transient heating of walls and other objects due to the flame (radiation) and to the gases (radiation and convection); the convective term is a function of the temperature difference between the wall and the gases and of a classical heat transfer coefficient. No combustion model is presently available in FLAMME\_S: a constant or transient value based on experimental results is provided in the input data file for the rate of combustion. Although it is not calculated, this rate of combustion is limited by the oxygen concentration in the compartment.

**Concluding remarks**

Although much simplifications are introduced, zone models as used in FLAMME\_S lead to computer programs with low CPU time and reasonable accuracy of predictions in engineering applications. Thus, it allows to study a large number of scenarios for the same problem and to perform a wide sensitivity analysis. As mentioned before, such a code used in conjunction with probabilistic analysis should be able to achieve reliable assessment of fire risk.

1.2. General description of the SIMEVENT code

The SIMEVENT calculation code \([7]\) was developed to study the behaviour of the ventilation network in normal or accidental operating conditions. The principle of the modelling is to divide the ventilation network into a series of nodes connected by branches:

- a node is an uniform pressure and temperature zone, even a large capacity room,
- a branch represents any link, generally between two nodes; these links may be resistance elements (ducts, filters), variable resistance elements (control devices), active components (fans), simple or complex thermal components.
The unknown variables of the problem are the pressures, temperatures and concentrations of species at the nodes, and the flow rates in the branches at a given moment \( t \). The equations are the mass and enthalpic balances at the nodes and the laws of behaviour of the branches, which link, from an aeraulic point of view, the pressure drop of the branch at the flow rate passing through it. The models describing the behaviour of the branches are of different natures, depending on they represent either a resistance element which consumes energy or an active component which produces energy. SIMEVENT has a library of aeraulic constitutive equations for most of the equipments ordinarily encountered in ventilation systems (ducts, filters, fans, leaks, controllers, ...) including in accidental operating conditions (filter clogging model, pressure control valve model, inverse flow rate in fan model).

The SIMEVENT program can accommodate steady state and transient operating conditions. This is done by introducing scenarios which may be model variations versus time [8], e.g., fan shutdown and start-up, pressure or temperature variation or changes in the ventilation network configuration (opening of a door, closing of a fire damper). SIMEVENT can then perform the calculations either by iterating equilibrium state phases or by solving differential equations, or by combining both methods.

Models relative to transfers of airborne species (gases and aerosols) have been introduced in the code [9]; the calculations are performed in transient regime to determine the species concentration at a given point in time, as well as the mass deposited at various points in the system, knowing the species source term characterised by an emission flow rate (gaseous or particulate species), the size distribution (particulate species), and the mass density of the species.

A great deal of work has been done on the inclusion of aerosols; two specific models have been developed for computing the deposit of aerosols in two cases: when the branch in the SIMEVENT model is a ventilation duct or when the branch is an HEPA (High Efficiency Particulate Air) filter. Both models are based on elementary wall surface and filter aerosol deposit processes.

Thermal design may be combined with the ventilation design by allowing for heat inputs and losses in the ventilation network and in the structures. The program calculates the temperatures at each node, and the flows in the branches. Pressures and temperatures are calculated in parallel and not simultaneously to avoid convergence problems. Different air temperatures induce dissimilar air densities which can generate motive effects if the ventilation network includes elevation differences; the program takes into account such effects if the elevations of each node are entered.

The SIMEVENT design includes utility programs to generate data and read results. Among them, the Microstation CAD software offers numerous capabilities, including:

- entry of the ventilation network diagram, often using an electrical analogy where each component represents a model type (Figure 3),
- entry of design or measured pressure and flow rate data and the active component characteristics.

The SIMEVENT preprocessor program uses the design or measured data to calculate the elements resistance and verifies the flow rate distribution diagram.
Once the ventilation network has been built, normal operating conditions have to be verified as precisely as possible, before the treatment of an accidental case.

![Diagram of a ventilation network]

Figure 3: Ventilation network diagram

1.3. Principles of FLAMME\textsubscript{S}-SIMEVENT coupling

Computer coupling between SIMEVENT and FLAMME\textsubscript{S} [10] was carried out with the goal of minimizing the algorithmic modifications of the two computer codes. Accordingly, the choice went to the interprocess communication (IPC) technique. This means that two simultaneously operating executables can dialogue through messages or shared common resources. It does not favour any of the two codes and permits use in the coupled or uncoupled mode; each of the codes preserves its functionalities. The hypotheses common to the software are as follows:

- flows are incompressible,
- condensation effects are left out of account,
- gas and particulate species formed during the fire do not affect the physical properties of the bearing fluid (air) in the network.
Calculations carried out using the FLAMME_S-SIMEVENT coupled software are distributed according to the specific aspect of each of the two codes. Thus, the calculation of temperature, pressure and chemical species in the fire room is attributed to FLAMME_S. The temperature, pressure and chemical species passing through the ventilation network (except for the fire room) and all of the flow rates are calculated by SIMEVENT. This results in the following:

➤ SIMEVENT imposes the flow rates in all FLAMME_S room communications,
➤ SIMEVENT imposes the pressures and temperatures of the neighbours to the fire room,
➤ SIMEVENT imposes concentrations of species going out and entering into the fire room,
➤ FLAMME_S imposes upon the SIMEVENT network the pressure and temperatures in the fire room,
➤ FLAMME_S imposes species concentrations inside and at the exhaust duct of the fire room.

There are four different stages in the phases of communication between SIMEVENT and FLAMME_S:

➤ initialisation of the communication protocol (the two executables are identifiable),
➤ mutual data testing (coherence between SIMEVENT and FLAMME_S),
➤ exchanges of values in the computation loop (pressure, temperatures, flow rates, chemical species, etc.),
➤ closing of the communication protocol.

Convergence criteria specific to coupling are used to evaluate the data exchanged by the two codes. These iterations, internal to coupling, are added to the iterations internal to the algorithms specific to SIMEVENT and FLAMME_S.

Coupled code performance has been the subject of an initial evaluation based on a validation and qualification matrix. Validation tests have been carried out to verify the operation of the various code functionalities under serious perturbation conditions. The results were analysed from the standpoint of physical consistency and compliance with the balance equations introduced into the code. First qualification tests were aimed at simulating the experimental conditions of laboratory experiments. The simulated quantities were thus compared with the physical ones registered during the different tests.

2. EXAMPLES OF CALCULATIONS AND APPLICATION OF THE COUPLED CODE

2.1. Comparison of coupled code-experiment

Presentation of the experimental installation

Some preliminary tests have been performed in order to compare the experimental results with the simulation ones. The experimental installation used for testing (see Figure 4) includes:
two rooms (103 and 105), in one of which the fire takes place (105),
* an exhaust network specific to room 105; for this test, the blower fan is stopped,
* a general exhaust network,
* a first HEPA filtration level for room 105,
* a second HEPA filtration level after dilution, upstream of the general exhaust fan,
* various balancing registers.

![Ventilation Diagram](image)

**Figure 4: Ventilation diagram**

Figure 5 below is a SIMEVENT diagram modelling the experimental installation, and including the normal pressure values at the nodes (data in Pa compared to atmospheric pressure) and the flow rates (m³/h) in the branches.

![SIMEVENT Diagram](image)

**Figure 5: SIMEVENT diagram of the experimental facility**

The total volume of the fire room which consists of concrete walls, is 100 m³ and the floor area is 20 m² (Figure 6); at the centre of the test cell, the fuel material (4.5 kg of kerosene) is located in a stainless steel tank having a surface area of 0.16 m². Temperature increase and ignition of the fuel are carried out using a radiant panel and an electric igniter. In the test described here, air supply is in the upper part of the room and exhaust in the lower part.
Results of comparison

Several comparisons between coupled code results and experimental measurements are presented here and concern:

- the pressure in the fire room (Figure 7),
- the various temperatures reached in the upper (represented by exp1_sup and exp2_sup for the experimental results, code_sup for the code results) and lower (exp1_inf and exp2_inf for the experiments, code_inf for the code) parts of the fire room (Figure 8),
- the exhaust flow rate of the fire room (Figure 9),
- the constraints, i.e. the pressure drop (DP_code, DP_exp) and the temperature (temp_code, temp_exp), reached at the filter located on the fire room exhaust duct (Figure 10).

Figure 7: Pressure in the fire room

Figure 8: Temperatures in the fire room
The evolution of the pressure in the room shows a peak during the first seconds of the fire, corresponding to the ignition of the pool. During the fire development, combustion aerosols are generated and the HEPA filter on the exhaust of the fire room clogs, causing a regular increase of the pressure in the room from -130 Pa to -70 Pa. Then, because the room is depleted of oxygen, the fire goes out and generates a high negative pressure. The clogging of the HEPA filter causes its pressure drop to increase (1 800 Pa at the end of the test) and the exhaust flow rate to decrease.

As far as code-experiment comparison is concerned, the following remarks apply:

- the evolution of the pressure in the room is reproduced satisfactorily at the fire ignition and during the fire development; nevertheless, the code overestimates the extinguishing negative pressure peak by 65%; this deviation is probably related to an excessively fast decrease of the air temperature in the room calculated at that moment (figure 8),
- the comparison of the maximum temperatures obtained in the lower and upper zones is satisfying during the fire, even if calculation underestimates the experimental values by 20°C on average. In addition, it is noteworthy that the temperature increase and decrease are slightly faster in the calculations than in the experiments; this is probably due to the modelling of thermal losses by the walls of the room (too fast) because the other parameters, in particular the exhaust flow rate, are accurately reproduced,
- the decrease of the exhaust flow rate is accurately foreseen by the code,
- the filter pressure drop is, however, correctly calculated; this good correlation is linked to accurate acknowledgement of clogging (mass of generated aerosols, then deposited on filter). In addition, the experimental temperature reached at the filter (70°C at the end of the stationary phase) is close to that calculated by the code.
2.2. Application of the coupled code to the study of ventilation control in case of fire

The reference ventilation network chosen to illustrate the potential of the coupled code, is shown in figure 11.

![Diagram of the reference installation](image)

Figure 11: SIMEVENT diagram of the reference installation

The fire room, corresponding to the room identified as S105, has the following characteristics:

- volume: 2 062 m$^3$,
- floor area: 269 m$^2$,
- volume of the solid structure located in the room: 813 m$^3$,
- floor area of the solid structure: 163 m$^2$.

The simulated scenario corresponds to a fire of a 60 kg pool of kerosene (4 m$^3$).

Three different simulations were carried out:

- one "nominal configuration" simulation, without controlling the ventilation
- one consisting in closing the blowing fire damper as soon as ignition takes place ("BFD closing" simulation),
- a final one consisting in closing the blower fire damper 70 s after the beginning of the fire, then the exhaust fire damper 140 s after ignition ("BEFD closing" simulation); leakage flow rates at the fire dampers are imposed.

Figures 12 and 13 show the results of these three simulations in terms of pressure evolution in the fire room and of differential pressure at the filter level.
Figure 12: Pressure in the fire room

Figure 13: Differential pressure at the filter

In the nominal configuration, a major overpressure peak appears within the first few seconds of ignition, reaching approximately 15 000 Pa (in relative pressure compared to the outside). After 15 min or so, the concentration of oxygen in the room is no longer high enough to maintain combustion; the fire stops and the negative pressure drops to -12 000 Pa. Because the room continues to be supplied with fresh air, combustion resumes, generating an overpressure peak of 10 000 Pa before extinguishing again because of a lack of oxygen. The maximum of extinguishing/re-ignition cycles imposed is three and combustion stops approximately 17 min after initial ignition.

The closing of the blowing fire damper on ignition induces higher overpressure and negative pressure levels than in the nominal case because the ventilation network provides higher aeraulic resistance. Final extinguishing of the fire, also linked with an oxygen deficiency, occurs later than in the nominal case (approximately 25 min after ignition) because the air exchange rate in the room has dropped.

In the latter case, the blowing fire damper closes after ignition leading to overpressure on ignition equivalent to that of the nominal case. Since the fire exhaust damper closes 70 s after the blower one, the pressure in the fire reference room increases slightly to 18 000 Pa (combustion continues in a room that is barely ventilated any longer). When the oxygen reaches the extinguishing threshold, the corresponding negative pressure is of the same order of magnitude as the negative pressure reached in the second simulation (-17 000 Pa). The air exchange rate in the room is much reduced and the pressure in the fire room increases very slowly. This leads to very late re-ignition of the fire, approximately 46 min after initial ignition. This re-ignition then induces a higher overpressure level than in other simulations because it takes place in a room with little ventilation (only by the leakage flow rates of the two fire dampers). Nevertheless, it can be hoped that the fire could be brought under control before re-ignition by human intervention means. As far as the constraints imposed on the filter are concerned, it is evident that the constraints are least severe for successive closing of the fire dampers and do not affect the integrity of the filters. Whatever the criteria chosen (pressure in room, mechanical constraints at the filter), the coupled code reveals that the best control strategy in the present case is the latter.
3. CONCLUSION

The development of a code allowing integration of the high interaction between fire and ventilation is important with respect to the safety of nuclear facilities and the protection of operators and the environment; the FLAMME_S-SIMEVENT coupled code is currently operational and meets this requirement.

The initial results of validation and qualification of the code are particularly encouraging; however, qualification on experimental devices still needs to be carried out for more complex ventilation networks and fires using different types of fuel.

The main current code limitations are of two types:

- those requiring developments without any apparent difficulties, such as coupling the fire with the emission of radioactive substances, the development of the existing models (plume model, filter clogging law, fan behaviour on flow reversal in particular),
- those calling for substantial developments such as the changes of phase (e.g. steam condensation), the possible propagation of fire to other rooms, the re-ignition of unburned materials in the ventilation network.

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Leningrad NPP Unit 3
Deterministic Fire Hazard Analysis Methodology
and its Preliminary Results

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The Leningrad Nuclear Power Plant (LNPP) is located 80 km to the West from St. Petersburg, on the Southern coast of the Gulf of Finland.

Figure 1  Location of the Leningrad Nuclear Power Plant

The Leningrad NPP consists of 4 units with RBMK reactors, 1000 MWe each. Units 1 and 2 (Phase I) are about 5 km to the South-West from the town of Sosnovy Bor. Units 3 and 4 (Phase II) are located 2 km further to the West.

Figure 2  Site plan of LNPP first and second stage
The first unit was brought on line in December 1973 and the last one, the fourth unit in February 1981. For so far the electricity generation of the Leningrad NPP has exceeded 500 bln. kWh.

For analysing the operation parameters of the LNPP (Table 1) it is necessary to distinguish three different stages:

- 1973-1981 years, the stage of commissioning and assimilation of design power of the units
- 1982-1988 years, the stage of design operation of the units. This period is featured by high technical and economical parameters with the annual electricity generation over 28 bln. kWh and installed capacity factor from 80 to 90 %.
- Year 1989, present time, the stage of large scale reconstruction at all units.

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<td>Electricity generation (annual average) bln. kWh</td>
<td>11,3</td>
<td>29,0</td>
<td>21,9</td>
</tr>
<tr>
<td>Electricity generation (upward trend) bln. kWh</td>
<td>102,1</td>
<td>304,9</td>
<td>501,9</td>
</tr>
<tr>
<td>Operating stages</td>
<td>Stage I</td>
<td>Stage II</td>
<td>Stage III</td>
</tr>
<tr>
<td>Operating stages</td>
<td>Commissioning, design parameters assimilation</td>
<td>Operation at the design parameter's lever, implementation of immediate measures for improving safety after Chernobyl NPP accident</td>
<td>Implementation of the full-scale safety improving programme (up to 2001 year). Development and implementation of the programme in order to prolong the life time of the RBMK-1000 units</td>
</tr>
</tbody>
</table>

Table 1 Operation parameters of LNPP

The Leningrad NPP was designed and constructed according to the requirements valid in 60's and 70's. The safety requirements of those days do not meet the present ones. During the whole operation period and particularly after the Chernobyl catastrophe at all units, measures aimed to improve plant safety were implemented. At first these measures were implemented in order to increase safety in the following branches:

- Nuclear and technical safety
- Fire safety
- Safety culture
- Radiation safety.
Fire safety concept at Russian NPPs, including LNPP, taken into operation during years 1971-1981 was based on the NPP safety specifications valid at the design and construction stage of these NPPs, which means that the present NPP fire safety requirements were not completely met. Fire hazard of the NPPs with RBMK reactors is due to large amount of combustibles in restricted areas including high voltage electrical cables, oil, hydrogen, plastics and combustible finishing materials. The oil systems of turbine generator, feedwater and condensate pumps, as well as transformer oil coolers are located in the common turbine hall of the turbine plant. At turbine plant there is 928 m³ oil and 36 kg hydrogen. Furthermore safety system related cables are routed normally in common channels.

Fires and accidents at NPP may cause breakdown of process equipment and release of radioactive products into the atmosphere.

After the Chernobyl accident the fire safety level of the operating units was checked and additional measures were developed (CМПБ-88) as well as new specifications for fire safety systems were launched (BCH 01-87).

The introduction of the following major measures to increase the level of fire safety has been of a great positive significance:

1. Equipping of control panels (БЩУ, ЦЩУ, РЩУ, ЦГК, ЦРБ, ЦПБ) and electronic apparatus rooms (СУЗ, СКАЛА) with automatic signaling and fire protection systems.
2. Installation of water fire fighting systems in the main circuit pump rooms and for the oil system of feedwater pumps.
3. Change of gas fire fighting systems of cable rooms into automatic water deluge extinguishing systems.
4. Treatment of cable routes with fire protective materials.
5. Reconstruction of automatic fire signaling systems.
7. Installation of fire safety shields at every NPP unit.
8. Change of foam fire fighting system of turbine oil system into automatic water deluge extinguishing system with installation of diesel pumps.

Within program of EBRD/NSA (items 1 to 4) and some bilateral co-operation programs (items 5 to 8), the following measures to increase the level of fire safety were executed:

1. Replacement of widows by special fire protection walls in between turbine halls and unit transformers as well as reconstruction of special fire protection barriers in between turbine halls and intermediate buildings.
   Fortum Engineering Ltd., Finland
2. Increasing the fire resistance of turbine hall bearing metal structures and roof lattices (0,75 h).
   SVT Brandschutz, Germany
3. Delivery of fire resistance paste KS-2 for cable covering.
   SVT Brandschutz, Germany
4. Reconstruction of an automatic fire detection system for Unit 3.
   Fortum Engineering Ltd., Finland
5. Reconstruction of an automatic fire system of critical areas of Unit 1 and
   an automatic fire system of critical areas of Unit 2.
   Honeywell, USA.
6. Reconstruction of fire fighting system of turbine oil system by replacing foam
   extinguishing with a water deluge system and by installing of diesel driven fire
   pumps.
   Denmark
7. Improvement of operative fire fighting capabilities (delivery of fire hoses,
   branchpipes, hand extinguishers, mobile foam units, extinguishing medium, fire
   suits and electrical handlamps for firemen of the local fire brigade ПЧ-72).
   STUK and Fortum Engineering Ltd., Finland
8. Installation of plant emergency radiotelephone system
   Fortum Engineering Ltd., Finland

Introduction of these measures has greatly increased the fire protection at LNPP
compared to the project stage. However, it is necessary to evaluate more completely
and qualitatively the possibility of a safe shutdown and cooldown of the reactor, as
well as to assure the localization and control of radioactive releases to the atmos-
phere.

For this purpose in 1996 the Russian authorities obliged the operation organizations
to implement fire safety concepts and an analysis of the influence of fires on the
safety at all NPP reactor units. The report with its results will be an obligatory part of
the plant operation license.

At Leningrad NPP these works started in 1997 for the third unit, which was chosen
as a pioneer plant because of its comprehensive overhaul and backfitting work within
the period from 1995 up to 1998.

Prior to the work under the analysis the fire safety concept at Leningrad NPP at the
present stage and method of the implementation of the analysis were developed and
agreed with the Russian authorities.

As a basis of the method was the methodology developed under a programme of the
U.S. Department of Energy for evaluation the safety measures of the RBMK and
VVER reactor cores during a fire (RPEM).

The purpose of the work according to the methodology is to analyse different kinds
of hazards existing during a fire and to develop safety measures of the reactor core
during any kind of fire at NPP. Flow chart of the methodology has been attached as
appendix 3.

The work on the analysis is executed by the technical experts of the Leningrad NPP
involving Russian construction and design organisations (Research and Design In-
stitute of Power Techniques, All-Russian Design and Research Institute of Complex
Power Technology, All-Russian Research Institute of Fire Protection).
Fortum Engineering was incorporated in the work as a technical advisor under the Finnish-Russian co-operation programme. Fortum Engineering gives technical support, organizes joint seminars and delivers means and software for carrying out the analysis. It also executes work on fire simulation in different rooms of the Leningrad NPP.

The following work for the third unit is executed so far:

- Determination of criteria for safe shutdown and cooldown of the reactor as well as functions, ensuring fulfillment of these criteria (Appendix 1);
- Lists of systems and equipment essential for safe shutdown and cooldown of a reactor (Appendix 2) as well as lists of rooms, where these systems and equipment are located;
- Room passports and the overall inspections of rooms;
- A conservative estimate of survival of systems and equipment at the beginning of fires in the rooms;
- Identification of such non-redundant vulnerable systems and equipment which may be harmed during a fire.

For completion of the analysis of the third unit in 1999 it is necessary:

- To analyse the influence of fires in power and control cable lines including the related networks;
- To identify, with a cable tracer, in situ the non-redundant cables;
- To finish the work on simulation of fires in separate rooms;
- To implement a detailed analysis on systems and equipment vulnerable during a fire as well as to develop measures for their elimination;
- To justify a possibility of a safe shutdown and cooldown of a reactor during fires with the localization and control of radioactive releases to the atmosphere;
- To develop measures for strengthening the fire retardant treatment of systems and equipment for safe shutdown and cooldown of a reactor;
Appendix 1

The criteria of safe shutdown and cooldown of the unit

A safe shutdown of the unit is featured by a procedure including an obligatory fulfillment of the following conditions:

1. Transfer the reactor core into subcritical state.
2. Keeping the reactor core in subcritical state.
3. Decay heat removal from the reactor in order to keep the reactor temperature and temperature in KMIΠΙ in given (project) limits.
4. Monitoring and localization of radioactive releases in order not to exceed the main dose limits.

In practice it is carried out two kinds of a safe shutdown.

"Warm" (short-term) shutdown, which is featured by transferring the reactor in subcritical state for a period necessary to pass the "iodine hole". In this case the decay heat removal is carried out in a "steam" mode.

"Cold" (long term) shutdown, which is featured by transferring the reactor in subcritical state and cooldown the reactor to the graphite temperature not above 100\(^\circ\)C and KMIΠΙ to the water temperature not above 80\(^\circ\)C.

The list of functions ensuring the fulfillment of criteria of a safe shutdown and cooldown of the unit

In order to fulfill the criteria of a safe unit shutdown it is necessary to carry out the following functions:
- Duly shutdown (damp) the reactor by inserting all control rods into the core
- Monitoring of neutron flux from starting and working ionization chambers
- Monitoring of power increasing period in starting and working range
- Monitoring of a subcritical state of the reactor
- Monitoring of a coolant amount in KMIΠΙ and its keeping in the design limits
- Monitoring of pressure in drum separators and its keeping in the design limits during cooldown
- Monitoring and localization of radioactive releases in order not to exceed the main dose limits and to keep the radioactive products in design limits
- Management of the systems (elements) of a safe shutdown, cooldown, localization and control of releases to the atmosphere, including the systems ensuring their operation.
Appendix 2

The list of systems necessary for a safe shutdown and cooldown of the reactor, as well as for ensuring the localization and control of radioactive releases in atmosphere

1. Control and safety system of the reactor (СУЗ)
2. Emergency core cooling system (САОР)
3. Blowdown and cooldown system (СПИР)
4. Emergency steam condensing system (САКП)
5. Makeup water supply and delivery system (СПВ)
6. Accident localization system (СЛА)
7. Monitoring system for radioactive releases to the atmosphere (СРК)
8. Systems ensuring the operation of the above mentioned systems
Appendix 3

Leningrad NPP Unit 3 Deterministic Fire Hazard Analysis (FHA) methodology flow chart

1. Collect plant data of NPP (1.2.2)
2. Define safe shutdown for plant (1.2.2.2)
3. Identify systems to achieve safe shutdown (1.2.2.3)
4. Identify equipment/components/cables of selected safe shutdown and support systems (1.2.2.4)
5. Prepare fire compartment drawings (1.2.2.7)
6. Identify location of major equipment and fire hazards (1.2.2.8)
7. Identify locations of electrical and control cables and instrumentation equipment (1.2.2.9)
8. Inspection of fire compartments (1.2.2.11)
9. Determine effect of fire on ability to achieve safe shutdown on a fire compartment (1.2.2.12)
10. List all fire vulnerabilities by fire compartment, zone, system and subject (1.2.2.13)
11. Analyse possible solutions to post-fire safe shutdown vulnerabilities and identify any commonalities (1.2.2.14)
12. Choice of solutions of vulnerabilities to guarantee safe shut down for each fire compartment (1.2.2.16)
13. Prioritize needed fire safety improvements (1.2.2.15)

Potential fire induced failures in safety function logics (1.2.2.10)

Evaluate design data and document existing fire protection features
- water ext.
- elimination
- detecting
- confinement
- fire brigade (1.2.2.6)
ABSTRACT

EDF have verified the sturdiness of the french NPPs to severe fire situations. This analysis has been performed fire zone by fire zone, any unprotected equipment being postulated lost within the fire zone. Wherever necessary, fire protection have been installed around sensitive equipment to avoid common cause failure. In addition to this fire prevention action plan, EDF have elaborated fire operation instructions to complement the post accident operating procedures and thus ensure that the operator can bring the plant in a safe shut down state in an upset environment without worsening the plant state.

This paper recalls the post accident operating procedures already in place on french plants. It describes the rationale for specific fire operation instructions (FAI op) and its interface with the existing operating procedures.

The FAI op cover all plant states. They are not used in accident condition as simultaneous fire and accident are out of design. However a fire initiated during the long term phase of an accident has been considered in the design.

The FAI op is intended to simplify the task of the operator and verify that he will have the necessary information and instruction to operate in an upset condition. It consists in:

- the list of operation actions necessary to limit the transient which will follow the deenergisation plan
- the list of measurements which might be affected by the fire
- the list of alarms which might appear spuriously due to the fire
- the shut down state whenever shut down is requested
- the deenergisation plan of electrical actuators which is implemented to clarify the operating state of the plant and protect the fire fighting team
- the restoration of power supply of equipment, wherever possible, by power sources not affected by the fire.

Once the FAI op has been implemented, operation is performed through normal or incident operating procedures in a situation clarified for the operator. It has been verified that the operating means remaining after the deenergising plan are sufficient to bring the plant to its shut down state.
INTRODUCTION

The fire risk has been well addressed within nuclear power plants. An IAEA safety guide exists on “Fire protection in nuclear power plants” which emphasises the need for prevention of fire generation and acceptable fire fighting capability. In addition to those precautions, Electricité De France have developed a set of operating procedures dedicated to the plant operation during a fire. Those procedure complete the existing operating procedures and ensure that the operators have the capability to carry out the operating actions necessary to bring the plant in a safe shut down state in spite of the degraded state of the plant submitted to a severe fire, in parallel to the fire fighting activity.

1. Fire fighting documentation

Specific Fire action rules (in french FAI op) have been elaborated for the operators. These FAI op are relevant to a fire zone in which the fire can be restricted. A specific document exists for each type of plant personnel involved in the activities during the fire.

- the action sheet for local personnel gives to the local operator send onto the fire zone the instructions for immediate fire fighting actions. It describes the actions necessary to isolate the fire zone when the fire cannot be extinguished and explains how to inform the control room; it then describes the actions needed to prepare the intervention of the fire fighting personnel.

- the action sheet for the "head of the fire fighting team" (the shift supervisor) and the "coordination " action sheet (for the senior shift supervisor) cover the initial phase. They describe the actions of the shift supervisor and the senior shift supervisor after the fire alarm has been given, with regard to the organisation of the fire fighting means and the reception of outside assistance.

- the "operator" action sheet (FAI ops) which precises the plant operation actions. It includes two aspects:

  - fire fighting instructions: it details the type of information that is requested by the control room operator from the local operator and explains when and how to request outside assistance

  - operating instructions: it explains the immediate actions to undertake, the deenergising plan, it lists the unavailabilities subsequent to the deenergising plan or to the fire in the fire zone.

2. Other operating documentation

The FAI op does not give the operating instructions. It only lists the unavailabilities subsequent to the fire or the deenergising plan. The operation will be undertaken with the normal operating procedures or the incident procedures according to the nature of the
unavailabilities. However urgent anticipative actions are ordered within the FAI op. Furthermore to gain some time in selecting the correct operating strategy, the FAI op adresses the right procedure which allows to skip the preliminary orientation tests of the operating procedures. If a fire happened when an incident/accident procedure were in progress, the FAI op would not be used. The incident/accident procedures are based on the state approach, and they are, by construction, capable to deal with the increased constraint of a fire. They make use of the equipment available by order of merit to control the main physical parameters of the plant and bring the plant to the safe shut down state. The physical parameters are the primary water inventory, the control of the removal of the residual heat, the S.G. water inventory and integrity, the containment integrity.

3. FAI op structure

The FAI ops correspond to the fire zones and their respective fire alarms. All the equipment within the fire zone unless equiped with a fire protection will be deenergised. In order to limit the number of FAI op, they may correspond to several fire zones which lead to similar deenergisation plans. Furthermore, to limit the execution time of the deenergisation plan, total switchboards may be deenergised whenever this is functionnally acceptable rather than individual actuators.

The FAI op instructions are dependant on the initial state of the plant (full power, hot shut down etc.).

The FAI op includes:

the presentation which identifies the FAI op. It lists the site, the plant, the building, the fire zone and the functionnal implication (normal or incident operating procedure)

the list of anticipated operation actions to limit the transient subsequent to the D. Plan.

The list of the potential spurious alarms

the fall back state or the technical specification reference

the D. Plan

the restoration actions when possible

Anticipative action

these actions are undertaken from the control room before the implementation of the D. Plan.

They order the lining up of integer redundant systems (the equipment fire protection and fire zoning is such that a unique fire cannot destroy two redundant systems), they anticipate or confirm certain automatic actions, they inhibit the Safety Injection protection whenever the fire could induce a spurious signal.

Some anticipative actions may have to be performed locally.
Informations

The FAI op gives the list of functional unavailabilities by order of importance. It gives first the list of support systems, the loss of which leads to incident operating procedure.

Then it indicates the systems which are surveilled in the incident operating procedures. It lists systems the loss of which do not require immediate use of incident procedures but could require their use later one due to tech.spec.obligation.

finally it lists equipment useful for operation, though not mandatory. Equipment which could be lost through the D.plan, the fire or the anticipative actions.

It lists as well the analog information likely to be affected by the fire so that the operator knows which information is still valid, which is dubious.

It lists all possible phony alarms.

Fall back state

The FAI op cover all the initial plant states. The fall back states are dependant upon the initial operation domain. Whenever the FAI op leads to an incident procedure, the fall back state is indicated and justified according to the systems and equipment still available.

The D. Plan

The aim of the D Plan is to clarify the operational capability of the plant. It allows to know with certainty when the equipment are unavailable. Furthermore, the deenergisation of the cables in the fire zone ensure the safety of the fire-fighters. The breaker openings has been limited to 30 to ensure the execution of the D.plan within one hour. The D. Plan does not deenergise two redundant safety systems.

The D. Plan lists all the equipment the cable of which (power and control) run in the fire zone. It lists the breakers to operate and their localisation.

The restoration

When it is possible to reenergise through other sources, deenergised equipment this is indicated on the FAI op.

4. CONCLUSION

In addition to its fire prevention and fire fighting organisation, Electricité De France have developed operating instructions specific to fire events, consistent with the present operating instructions which allows the operating team to bring the plant to a safe state. This procedure has been successfully tested in Fessenheim 900 MW plant this year in march and will be generalised on all french plants from year 2000.
Session VIII
Fire Risk Assessment and Applications III

- Fire Risk Analysis of United Kingdom Nuclear Chemical Plants A Practical Approach - Mr. Geoffrey Arrowsmith, GMIFE QSFPO, Health & Safety Executive (HSE), United Kingdom, Mr. Steven Greenwood, Bsc Hon Mech Eng AMI Mech E MIOSH, British Nuclear Fuels Ltd. (BNFL), United Kingdom

- Olkiluoto NPP Fire Risk Analysis - Mr. Mika Yli-Kauhaluoma, Mr. R. Himanen and Mr. K. Taivainen Teollisuuden Voima Oy (TVO), Finland

- PSA Study for An Exemplary Plant Location of a German PWR Built to Earlier Standards - Dr. Marina Röwekamp, and Dr. Heinz Liemersdorf Gesellschaft für Anlagen- und Reaktorsicherheit mbH (GRS), Germany
Fire Risk Analysis Of United Kingdom Nuclear Chemical Plants A Practical Approach

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The views expressed within this presentation are those of my colleague and myself, and do not necessarily reflect any statutory policies enforced by the Health and Safety Executive of the United Kingdom or of British Nuclear Fuels PLC.

Introduction
Historically fire risk assessment within the United Kingdom Nuclear Chemical Industry has concentrated on a deterministic approach to fire safety, having regard to the buildings construction, the processes within the facility and the associated services that are installed to facilitate its safe operation. We consider that this approach satisfactorily addresses fire safety in Nuclear Chemical Plants and has been applied successfully over the past years. The approach is being continually improved as more data becomes available with respect to materials used, their combustibility and more up to date means of passive fire protection, detection and suppression.

It is intended to discuss briefly the approach adopted by the United Kingdom for Nuclear Chemical plants and the philosophy for the adequate provision for a fire safe facility.

When considering the philosophy to be adopted, the key elements that would be considered within a fire risk assessment are Building Construction, Compartmentation, Fire Loading, Means of Escape, Ventilation Systems, Detection / Suppression Systems, process operations, fire fighting access and facilities. In addition the effects that a fire can have on radiological safety will be briefly discussed.

One of the elements that can effect the feasibility of a risk assessment is the ventilation systems designed primarily for radiological control and to provide and acceptable working environment for the personnel employed within the complex.

It is our intention to discuss and illustrate the conflicts between the requirements for radiological control and the conventional approach to fire containment. A number of examples will be given where practical solutions have been successful.
Building Construction

As an integral part of the design of nuclear facilities, the choice of construction materials used within the facility is made taking account of primarily the duty to contain radiological inventory, in terms of contamination and radiation. This choice is also made on the basis of its inherent combustibility. Generally UK nuclear chemical plants are essentially constructed from bulk reinforced concrete, structural steel and composite external cladding systems. Within the building envelope there are many other construction details which must all be considered. The combustibility of these systems can significantly effect the overall fire loading within the building. It is important therefore, to select materials of mainly non-combustible construction and preferably, inherently fire resisting.

Therefore the fire safety engineer plays an important role, in assisting with the selection of these materials.

Compartmentation

It is not economically viable to remove all combustible materials from the facilities and there remains a substantial combustible inventory. A large percentage of the remaining combustible inventory tends to consist of electrical cabling, even though the cables are selected with the best properties, with respect to combustibility (i.e. low smoke and fume cabling). However, in sufficient quantities the properties generally change and fire development is possible. In addition, materials being processed may be inherently combustible.

To minimise the ultimate potential growth of a fire, should it occur, the facilities are divided into smaller fire resistant compartments, where higher fire load areas are separated from lower fire load areas, by fire resisting construction. The designated fire resistance of the compartment is determined from evaluating the total combustible inventory for the area and determining the equivalent fire severity, hence the required fire resistance of the compartment boundary. For low fire load areas it is possible to justify lower fire resistance requirements (utilisation of the inherent fire resistance of the structure, at the design stage).

The approach generally adopted for UK Nuclear Chemical plants with respect to fire is the containment approach where the potential fire growth is restricted to a limited area which is bounded by a fire resisting enclosure with a fire resistance in excess of the total combustible inventory within that area. This approach differs from the use of sprinklers, which is a more common approach in conventional industries. The main reason for not using sprinklers within nuclear chemical plants is the potential moderating effect with a possibility of a criticality. Smoke control in radiological areas is not generally deployed as a design option due to the potential for an uncontrolled release of radioactive contamination with extreme potential off site consequences.
Structural Fire Protection

The elements of structure within the building consist, generally, of reinforced concrete and/or structural steelwork. The fire resistance of reinforced structural concrete is determined by the depth of cover to the main reinforcement bars within it. Most concrete structures used within the nuclear/chemical plants have an inherent fire resistance in excess of 1 hours fire resistance. The fire protection requirements for structural steelwork is determined both by an evaluation of the overall combustible inventory in the area (i.e. - an office or an E & I room) and any localised effects from high local combustible inventories. Where a structural member is not threatened by a combustible inventory then lower or no fire protection may be required over and above its inherent fire resistance.

Fire Assessment of Unprotected Structural Steelwork

Fire Engineering Principles

The performance of structural steelwork is normally evaluated experimentally (using single elements with simply supported or pinned connections) in the BS476: Parts 20-22 fire test, which adopts the standard time/temperature heating regime.

In a real fire the temperatures achieved could be significantly different (either higher or lower) than those in the standard fire test. A first principles approach enables a more precise calculation of the fire resistance of the member by considering the significance and severity of a real fire in the building. Estimated fire loads for generic areas in the facilities are based upon historic site inspections of similar existing nuclear chemical facilities.

Collapse during a fire occurs when the imposed load exceeds the load bearing capacity of the structure at elevated temperature. The temperature at which collapse occurs depends upon the steel grade, its elevated temperature strength and the imposed stress. Typically this temperature is taken as 550°C for fully loaded BS4360 structural steel members. However, it should be noted that 550°C is a conservative lower limiting temperature, in practice higher failure temperatures are generally achieved, for example due to the continuity and fixity effects of the building structure.

Published test results for fully loaded steel columns and beams supporting concrete units give failure temperatures of around 580°C and 650°C respectively.

In the BS 476 standard test for beams, failure is considered to have occurred when the deflection exceeds :-

\[
\text{span} = \frac{20}{\text{}}
\]

This coincides with a lower flange temperature of approximately 650°C. In a fire, the beginning of the onset of collapse occurs when the strain (or proof stress) is between 1% and 2%.
Tests on Grade 43A steel (i.e. general structural steelwork) with imposed stresses of 150N/mm² and 50N/mm², produced strains of 1% and 2% at temperatures in excess of 570ºC, 588ºC and 713ºC and 724ºC respectively; thereby drawing a correlation between lower stress levels and higher failure temperatures.

Figure 1 illustrates the relationship between temperatures and proof stress.

The section factor (Hp/A) is used to quantify the heating rate of a steel member in a fire. (Hp is the perimeter of the steel member exposed to fire, A is the cross sectional area of the steel member). A member with a low Hp/A ratio will be heated at a slower rate in a fire than one with a high ratio. Hence members with low ratios will tend to exhibit greater fire resistance.

Fire Research Station tests on fully loaded steelwork have led to the findings that steel members with Hp/A ratios up to 180m⁻¹ can be left unprotected in areas with fire loads not exceeding 15kg wood/m² equivalent (270 MJ/m²), even under the worst ventilation conditions. A fire load of 100MJ/m² is generally used as the value below which structural steel protection is deemed unnecessary other than for certain areas where higher localised fire loadings can lead to local failure.

The fire resistance of unprotected beams and columns may be improved by considering composite action, restraint and continuity effects of the building structure, reduced loading, and the extent of any partial protection or shielding.

**Composite Action**

In the standard fire test for beams, the use of concrete slabs is normally in separate segments to minimise composite action effects. In such instances where no bond exists, the slab and beam are able to deflect individually and the whole of the load tends to be taken by the steel beam.

Where composite action by the use of steel studs welded to the top flange of the beam are used, (also known as shear connectors), the resulting design stresses and fire resistance can be notably improved (i.e. the resulting section behaves as a "T-beam" where most of the compression forces are catered for by the concrete, and the tension forces by the steel). Mean failure temperatures in the lower flange and web will vary depending upon the beam size. For modest sized beams of around 40 kg/m mass and Hp/A values of around 170m⁻¹, the critical temperature is likely to be above 750ºC.

**Restraint**

Generally, in structural design procedures a simply supported (or pin-ended) approach is adopted, based upon a less elaborate but more onerous bending moment value, which in turn provides for a greater margin of safety.

(Pin-ended connections are normally identified by the use of 1 or 2 bolts).
By providing in practice a degree of end restraint and fixity, the fire resistance of the members can be considerably increased.

**Reduced Loading**

Where structural members are only partially loaded, the reduced stress has the effect of raising the critical steel temperature for failure in fire. This principle can be taken advantage of, for example, where main structural members have been designed to withstand seismic loading, which would cause maximum stresses for a relatively short period of time (i.e. less than 7 seconds). It may therefore be assumed that under normal operating conditions, buildings that have been seismically designed will be subjected to stress levels below their design maximum by around 40% to 60%. Hence failure under fire conditions would be expected to occur at appreciably higher temperatures (i.e. around 650°C).

**Partially Protected Steelwork**

Steelwork members are often constructed partially encased in the fabric of the building, e.g. steel columns in blockwork walls. This has the effect of reducing the Hp value, thereby increasing the fire resistance of the steel.

**Means of Escape**

An important aspect in the early stages of the design of a nuclear/chemical plant, is the structural layout to provide adequate access and in particular appropriate egress in the case of emergencies i.e. fire. To ensure that the layout of facilities provide adequate escape, it is important for the fire safety advisor to be involved in the layout process. The objective is to wherever possible, provide alternative means of escape to a place of relative safety, within the appropriate travel distance. The travel distance is determined with reference to National Regulatory guidelines based on an assessment of the fire risk for the area concerned and is related to the potential for a fire to start and its likely speed of development.

There are a number of means by which adequate escape can be provided. In some facilities it is necessary to engineer places of relative safety which are not the final exit. In these cases it is acceptable to use a true compartment boundary, provided that from that point it is not necessary to re-enter the compartment in which the fire is present, to reach the final place of safety. Two typical approaches are shown in fig 1 which shows compartment to compartment escape, where radiological containment considerations are necessary i.e. due to the need to maintain cascade ventilation flows for radiological contamination control and the utilisation of protected routes for providing adequate escape, due to travel distances being excessive without their provision. In addition travel distances are assessed based on the provision of compensatory features which in effect reduces the immediate risk from fire to escaping occupants. These compensatory features include such provisions as automatic detection and alarm which are generally provided in all new UK nuclear chemical plant as standard practice, roof vents where possible if activity release does not override this provision, additional fire resisting constructions over and above the minimum requirements and the use of active suppression systems to reduce the overall risks of fire from building occupants.
Suppression Systems

An important part of the assessment process of the fire risks within a nuclear facility are requirements for fixed and portable fire suppressing systems. There are many occasions when the choosing of a suppression system has to be made very carefully, taking into account the potential reactions that could take place between the suppression agent and the chemicals used within the plant (i.e. water and plut enriched uranium can lead to a meltdown) and reactor metal fires can react violently with fire fighting systems. Not only can this fail to extinguish the fire but actually intensify it and lead to a more severe event.

Suppression systems may also be provided for commercial reasons. In some instances no such suppression systems will be provided will, as the benefits of providing such a system do not justify the cost.

Nuclear Fire Safety

A fire within a nuclear facility may have the potential to lead to a nuclear event in the form of a radiological release or in some cases a radiation excursion. As part of the design, consideration must be given to the effects a fire can have on radiological containment or those protection systems which prevent an event from occurring on demand. A fire which may be remote from the radiological containment but may cause an unsafe failure of safety systems which protect against fault conditions, may ultimately lead to a radiological event.

The means by which these unacceptable consequences can be prevented is by the use of a number of principles. If the containment of a radiological inventory is potentially threatened - directly by a fire, then there are a number of options. The best solution is obviously the removal of the threat by fire, this can be achieved by substituting the combustible materials which presents the threat by non combustible substitutes, or provide additional fire protection to the containment. Another approach will be to provide a suitable detection and suppression system for protection systems, there are a number of methods of providing suitable means of segregation:- Discrepancy checking.

Background

To meet the appropriate Company and legislative safety criteria for the operation of plant, machinery and equipment, companies have developed safety protection systems and safety related systems that have elements of redundancy and diversity built into the engineering system design to increase their reliability. In addition to this, the building structure and compartmentation may provide segregation from fire to some of these systems. However, this has not generally been by design, and without the examination of the safety protection systems and safety related systems and their components, the required fire segregation in a building cannot be demonstrated.
The following methodology provides a framework for the logical evaluation of fire hazard assessments, the direct and indirect consequences of a fire, together with the mitigation of the circumstances to limit the effect upon safety protection systems and safety related systems - including segregation of the potential fire hazards from safety critical systems. This methodology forms a part of a total safety management system on the procedures adopted in the assessment of fire hazards, including a study of plant design criteria, consequence analysis in the event of fire and review of the effect upon safety critical systems, and together with the mitigation of the effects of fire through detection, suppression and protection systems, in order to eliminate, reduce or control the risk of fire.

The complete methodology that follows is appropriate to both the design process and in-service operation of the plant, and which are specifically outlined in Section 3.

**Principles of the Methodology**

The identification of potential fire hazards, due to a lack of integrity of the engineering system, structure and components are determined by the application of a logical screening process.

If it can be demonstrated that there is a potential fire hazard, but there is no remote possibility that the consequences of a single fire event can lead directly or indirectly to a radiological hazard due to the provision of suitable and sufficient segregation of engineered safety protection systems and safety related systems, together with operational safety control measures, which have been identified and implemented in the design and process control systems, to eliminate or reduce the risk of fire (and to control the consequences of a fire) - then no further consideration is required, with respect to the risk of any breach of radiological containment.

However, if there is a possibility of such an event then consideration is required to determine whether a single fire event could lead directly to a radiological hazard due to the loss of a safety protection system, resulting in a breach of radiological containment, such as the loss of: 1) segregation / containment, 2) safety equipment, and/or 3) safety control system.

An example of this could be where a hydrocarbon pool fire could directly impinge on a radiological container which leads to radiological release. In this instance the resultant action would be to eliminate the potential for fire by design or to mitigate the hazard by a fire engineered solution, so reducing the risk.

Where a single fire event could lead indirectly to a breach of radiological containment, from the unacceptable loss of a safety related system in an unrevealed manner, such as a safety monitoring device, which becomes inoperative or fails on demand as a result of the fire.

An example of this would be the loss of a protection system required to force the plant into a safe state on demand, and fails in an unrevealed manner. If this is the case, further consideration is required to eliminate the potential for fire by design or to mitigate the hazard by a fire engineered solution, so reducing the risk. If part of the protection system remains effective during or following the fire, then provided that the potentially reduced

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reliability has been analysed and meets appropriate criteria with its reduced reliability, then segregation or other engineered solution may not be required.

Consideration should be given to whether a fire can cause the loss of a protection system, but not necessarily initiate the unwanted event, or additionally if the fire can be both the initiating event leading to an unacceptable radiological event and the loss of the protection system.

In the first case if the loss of the protection system is revealed and no radiological event is initiated, then it may not be necessary to provide fire protection measures if the plant or equipment capable of leading to the hazard can be shut down safely with out the need of the protection system.

In the second case it will be necessary to segregate the protection systems so as to prevent the simultaneous failure of the protection system and initiate the unwanted radiological event.

An example of such a system is one that requires to remain operational i.e. inerting, where reliance has been placed on the system to eliminate the potential for fire in a given area, the result of which could give rise to a breach of containment. A further example could be a device moving from A to B where the power supply, control and protection circuit cabling share a common route, and a single fire disrupts the control signal (hot short) and eliminates the protection circuits, but leaves the power initially unaffected, such that the device is driven beyond the safe limit, potentially leading to a radiological release.

If it is possible that a fire can lead to both the revealed failure of the protection systems and initiate a revealed radiological event, then consideration should be given to the time period in which the hazard is likely to develop. If there is sufficient time to intervene and allow rectification without developing into an unacceptable event, it may be applicable to justify no segregation of the system. Further consideration is necessary to ascertain whether the hazard will be revealed in an adequate period of time. If the time period is determined to be insufficient, then segregation will be required.

An example of this is a discrepancy checking device, which regularly checks the status of the plant that it is protecting. Such that if it detects a discrepancy in the expected sequence of events, or the range of expected values of a control system, then it shuts down the plant. There is also a time element incorporated, such that if the discrepancy occurs for longer than a predetermined period then the plant is shut down.

If a failure of the protection system is likely to be unrevealed, then the appropriate action would be to either redesign the system so that a fault condition should be revealed, or to provide adequate segregation to the safety systems.

Radiological safety assessors generally determine the required integrity class for protective measures by the application agreed company criteria document detailing the criteria to be applied. This entails determining the Design Basis Class for the public and worker protection, from a combination of the Demand Frequency Class and Hazard Severity Class; and the assignment of a Configuration Class for the minimum number of channels of protection measure required. Subsequently, from the Demand Frequency Class and Hazard
Severity Class, the Required Probability of Failure on Demand for the protective measures, and the Engineered Protection System Integrity Levels from IL.1 to IL.4 can be determined.

For integrity levels 1 to 4, the typical Engineered Protective System configuration requirements, in order to meet the target Probability of Failure on Demand (PFD) state, is that no single failure should result in a total loss of protection. Therefore, during the process of rigorous analysis, checks should be undertaken to ensure that a fire cannot cause a single failure scenario of the protection system. Hazards that have been categorised by quantified risk analysis to have a low demand frequency, but that of a high potential consequence severity, must be considered in isolation, in terms of the effects from fire increasing the frequency or probability of the event.

In addition to examining the above, systems involving high consequence event severity must also be considered, regardless of the frequency, and a rigorous analysis of the effects of a fire should be undertaken for all components of such systems.

The identification of designated radiological fire safety barriers and designated areas that must be free from combustibles throughout the plant buildings can be highlighted on a computerised space management program for the facility, with these areas guarded against the penetration or introduction of services into these areas. In the case of existing operational facilities, identification of radiological fire safety measures may be specified in the building manual, and physically marked on the barriers to warn of their significance.

Where more than one protection system has been identified to meet the safety criteria against a specific hazard, then segregation may not be required for common mode effects of fire. However, if segregation requirements are identified at a sufficiently early stage of the design, then compartmentation can be adapted more readily and easily, thus avoiding excessive sub-compartmentation schemes.

Radiological safety assessments are undertaken to demonstrate the principles of ALARP and to achieve the philosophy of "defence in depth". The vulnerability of systems that have a designated radiological safety function must be addressed to ensure that sufficient measures have been incorporated into the design to prevent common mode failure resulting from a single fire.

Systems that have a designated radiological safety function, are defined as those where failure in response to a demand leads to the potential for:

a) The release of radiological materials

b) The release of radioactivity above the prescribed limits during operational states

c) The release of radioactivity above acceptable limits during accident conditions

d) The release of radioactivity due to fault conditions that could lead to radiation exposure to personnel

Systems that have a designated radiological safety function will operate in the following two ways:
a) **Trips** - These are Systems which are designed to detect a fault condition and remove power from the plant to make it safe; for example, a pump ceasing to operate.

b) **Enables** - These are Systems which are designed to detect a fault condition and energise protective systems to maintain a safe condition; for example, a temperature control and protection circuit on electrical driers for intermediate and high level waste, which if overheated could result in a Magnox fire and potential radiological release. Each drier is therefore provided with partially diverse means of detecting an abnormal temperature rise by the use of a resistance thermometer device (RTD) and thermocouples. The fault scenario considered in the event of fire is a loss of temperature control or an inability to isolate electrical power from the heating elements. Control and protection circuits are therefore assessed for vulnerability to fire, down to component level within the electrical cubicles. Power circuits are not included as their failure would lead to a loss of supply to a heating element which would lead to a safe plant condition.

N.B. Fire can affect these systems in a number of ways, for example, 1) initiate the event and coincidentally disable the protection systems, or 2) cause an unresolved failure of the protection systems. Without detailed examination of the fire effects to such systems and their components, it is not possible to dismiss fire as an insignificant contributor to the event frequency and therefore must be considered. However, there may be a number of circumstances where high frequency, but low consequence events, require safety systems to reduce the frequency of the event to an acceptable level.

For these situations it may be possible to demonstrate that the common mode effects of fire on the safety systems do not significantly alter the event frequency, and that no further consideration of the effects of fire on these systems is required.

Plant should be designed to have “defence in depth” making appropriate use of redundancy, diversity and segregation, commensurate with the safety significance of the potential hazard - ref. Defence in depth, with respect to fire, should be considered for all significant safety systems, the objectives being to:

a) Prevent fires from starting through design integrity

b) Detect and extinguish quickly any fires through the use of comprehensive detection systems, fire suppression for specific risks, and fire fighting facilities through the site fire brigade

c) Prevent the spread of those fires that have not been extinguished by a comprehensive scheme of resisting compartmentation and building containment.

No single “random failure”, assumed to occur anywhere within the safety systems provided to perform a safety function, should prevent that function being performed during any normally permissible state of plant availability. In order to satisfy the objectives with respect to radiological fire safety, a satisfactory level of segregation between independent parts of designated safety systems shall be achieved by the use of fire resisting construction between component parts, or by demonstrating that the design of the system is inherently invulnerable to the effects of a single fire.
The "fire loading" in the plant should be quantified, and together with the justification for the choice of materials, and the fire suppression and fire fighting facilities provided. This is a related safety criteria that is currently addressed as part of the Plant Safety Case submission, and should be addressed for each radiological safety protection system and safety related system, to ensure that an adequate level of fire suppression is provided and structural fire protection is achieved.

**The Basic Principles of a Radiological Containment Cascade Ventilation System**

Within the United Kingdom, radiation risks in nuclear chemical plants are expressed in degrees of exposure to radiation and contamination from a radioactive source. Both exposure risks are generally expressed numerically with representative values from one to five, one being the lowest level of radiation / contamination exposure value.

Within buildings each area is designated as to the degree of radiation risk for exposure and contamination, R2, C3. (Radiation risks 2, Contamination Risk3.). The rating given to the possible contamination of an area also determines the type and level of protection required to be provided for the protection of personnel.

Ventilation systems within buildings do not have any effect on Radiation levels.

One of the principal methods of contamination control is the provision of ventilation systems that cascades air directly from the atmosphere into the building, by supply and extract ducts into C1 areas then by ducts and grills to C2 areas, then C3 areas etc.

Figure 2 is a schematic representation of a cascade ventilation system. The objective is to create airflow from a clean area to the areas of highest possible contamination and then finally extracted through filters before it is released to atmosphere.

The basic principals are, firstly to create an outer zone held at an even atmospheric pressure by input and extract ventilation. The C1/R1 zone, (clean area) is to ensure that any external variations in atmospheric pressure, do not create a lower pressure differential that could result in the reversal of the air flows from the C2 area into the C1 area.

The C2 area is entered from the C1 area by way of a change room, which has input air on the clean side and extract air on the C2 side in order to maintain the air flow.

The C2 area is the next area of protection and so on to the areas of highest possible contamination. This area is provided with both input and extract ventilation, the extract ventilation is designed to create a flow of air from the C1 area to maintain an inward air flow towards the high contamination.

Lastly there is the working area C3 and vessel vents C4and C5 areas. These areas are provided with extract only, all input air is drawn from the C2 area through cascade systems.
One of the main advantages for personnel leaving the C3/C4 areas with possible air born contamination on their protective clothing, is that the induced air flow through the cascade helps to remove any contamination and keep it within the C3/C4 areas.

**Conflicts between Ventilation System for Contamination and Fire Safety**

The ventilation systems within nuclear chemical plant are generally designed for the purposes of contamination control as explained in principal earlier. To maintain the containment barriers, the ventilation system is required to continue operation for as long as possible. To maintain a suitable fire safe building, it is necessary to provide an acceptable system for fire containment, a comprehensive fire compartmentation scheme will be generally provided by design.

To maintain compartmentation, the ventilation systems installed for the purpose of contamination control do not generally recognise the fire boundaries due to the need for cascade flow across the boundaries it is necessary, therefore, to provide fire dampers at the compartment boundaries.

It is clear that the provision of these dampers will lead to conflicts between the need to maintain compartmentation in the event of a fire and to maintain radiological containment for as long as possible. As explained earlier the C1 and C2 systems are essentially for the purpose of providing general ventilation for normally manned areas and to maintain a steady pressure to prevent back flows. In C1 and C2 areas it is normal practice to provide fire dampers that are automatically closed, actuated by an automatic smoke detection system. The extent of closure of dampers is plant specific, but can range from the individual area effected by fire, to the complete system closure of C1 and C2 fire dampers. The exception to this are fire dampers located close to the vent fans or in primary risers which would be place under the control of the plant personnel via a remote manual control from the control room.

For C3 systems, it is necessary to evaluate the potential consequences of a closure of fire dampers in the systems at compartment boundaries. In some cases the C3 area will have a low potential radiological consequence if the dampers were to be closed e.g. the area is only C3 when maintenance is taking place. In this case, it is acceptable to automatically actuate the fire dampers in these areas from the smoke detection system. Another example would be where the fire itself leads to a high radiological consequence. In these cases, it may also be reasonable to automatically actuate the fire dampers, if the fire is likely to be large. In the case where it is not clear when a fire damper should be closed i.e. the consequences of inadvertent closure are significant in terms of the radiological contamination but not immediately fire. It is usually the case that the closure of the dampers would be under the control of the plant personnel. Training is provided to the operators such that they understand the conflicts and can judge the appropriate time to manually close fire dampers remotely from the control centre.

C4 systems and above are not normally fitted with fire dampers as these systems generally only interconnect different in cell areas without any direct connection with out cell areas. The duct work generally then leads directly to the fan room without any connections to
manned areas and enters non combustible HEPA filter banks and finally connects to the stack.

The duct work consists of fully welded pipe work in excess of 3mm up to in excess of 10mm and the fire integrity of the duct is sufficient to resist in excess of 1 hour fire severity. If additional fire protection is required then proprietary fire resisting and insulating materials can be used. If the risk of a fire breaking out of a cell or spreading to an adjacent cell is possible via the duct work connections, it is possible to prevent this by the use of manually operated fire safe valves fitted at the cell boundary. The valves may be either locally operable or remotely operable depending on the risks to the operator and the speed at which the potential fire can develop.

The contamination control system is not generally suitable for the use as a smoke control / extract system in the event of a fire. There are a number of reasons for this:

Firstly the C1 / C2 systems are generally not of a substantial enough construction to withstand the effects of fire without the provision of comprehensive fire damper and fire protection provisions. Also the extract fans are not rated to operate under fire temperatures. The ventilation flow rates from the extract fans are not generally sufficient to cope with a significant fire. Sprinklers are generally used in conjunction with smoke extract systems to ensure that fire growth is limited to the capabilities of the extract system. The ventilation filters would be likely to blind very rapidly under fire conditions and soon the ventilation system if not by passed would become in effective. The main potential concerns with the attempted use of vent systems designed for contamination control for the purposes of smoke extraction are the potential for uncontrolled fire growth, an uncontrolled release of activity or a potential criticality if sprinklers are used in combination with the vent system.
Figure 1: Effect of Temperature on Proof Stress of Structural Steels
Figure 2: Alternative Schemes for Means of Escape

- Cat ladder: escape down
- Protected corridor
- Fire rated ceiling
- Travel distance achieved by escape in either direction to protected corridor
- Protected stair
- Travel distance achieved by primary escape to compartment 'A' or secondary escape to adjacent fire compartment 'B'
Figure 3: Schematic Representation of Cascade Ventilation
OLKILUOTO NPP FIRE RISK ANALYSIS

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ABSTRACT

This paper describes the probabilistic fire safety assessment (Fire PSA) carried out by the utility Teollisuuden Voima Oy (TVO) for two BWR units Olkiluoto 1 and Olkiluoto 2 (OL1 and OL2). Analysing methods and quantitative results are discussed and plant modifications implemented as a result of the fire risk study are described.

The Fire PSA study was done step by step between 1988 and 1998. The study considered only risks relevant to nuclear safety, and economical losses were disregarded. The first fire analysis, started in 1988 and reported in 1991, considered the risks of fires affecting safety system components and their power cables. The analysis was extended between 1992 and 1994 to cover also the fire risk through damages in instrumentation cables. Finally the fire risk during outage was analysed between 1996 and 1998, so that the plant specific fire PSA covers all operational states.

Operational experience of the latest 34 reactor years and 36 outages of units OL1 and OL2 were used to determine the plant specific fire frequencies for power operation and for shutdown conditions, respectively. For the room specific fire frequency analysis Berry’s method was used. The room specific fire parameters needed in Berry’s model for power operation and for outage were assessed separately, too.

The core damage frequency due to fires as a result of the first study was 2·10⁻⁵/ry. Due to improvements in sprinkler systems and in passive fire protection the risk decreased by factor 5 to 4·10⁻⁶/ry. Later plant modifications decreased the core damage frequency due to fires to 3·10⁻⁷/ry¹, although some of them were implemented because of other reasons than fire risk reduction. The core damage frequency due to fires during outage is 2·10⁻⁸/ry, which is almost three orders of magnitude lower than the current total core damage frequency (1.4·10⁻⁵/ry).

Since the contribution of fires to the annual core damage risk is only 2.0 %, and since the risk level is from international point of view relatively low among soberly done PSAs, there is hardly any potential of risk reduction by means of improvement in fire protection and therefore plant modifications are not worthwhile. However, the model is

¹ includes fire risk in shutdown conditions
maintained and updated continuously in order to foresee potential risk increase due to any plant modification or operating experience gained.
1. INTRODUCTION

Teollisuuden Voima Oy (TVO) operates two identical ABB ATOM type BWR units, OL1 and OL2, in Olkiluoto (Finland). The units have been in commercial use since 1979 and 1982, respectively. After the modernisation, that was carried out in connection with normal refuelling and maintenance outages between 1996 and 1998, the net electrical power of each unit is 840 MW.

The level 1 PSA for the units OL1 and OL2 was started in 1984 by the utility TVO\(^1\). At the beginning the analysis was concentrated on internal initiators, whose risks were reported in 1989. The analysis of external events was started in 1988 and reported in 1990 considering floodings and in 1991 considering the first fire risk analysis, which included safety related components and their power cables. Development of living PSA was begun in 1990 and the first version of the level 1 PSA for power operation including internal and external initiators was published four years later. The analyses of external events were carried on and the analysis of weather and earthquake were brought into living PSA in 1994 and 1997, respectively. Also the fire PSA was updated between 1992 and 1994 and the analysis was extended to instrumentation cables, since it was already identified in the first phase that especially a fire in reactor protection system (RPS) can cause tripping or spurious control of safety related systems and components.

The Shutdown PSA analysis, carried out between 1990 and 1992, was the first one in the world for type BWR NPP, so that new methods were to be evolved. The original analysis included loss of residual heat removal and different leakages of coolant, but in 1996 and 1998 it was extended to include also fires. Hence the fire PSA covers all operational states. So far the contribution to fire risk analysis is about six man-years.

The principle of using four half capacity subsystems is applied in both units to all important safety functions. The four redundant circuits of the safety related systems are assigned to two main groups, located in physically separated areas. Within the areas separation by distance or by means of barriers is used between the redundant circuits\(^2\).

The safety design criteria for the Olkiluoto NPPs are based on Swedish codes and standards. Adaptations have been made to Finnish regulations. The units have very wide active and passive fire protection systems and also an on-site fire brigade.
2. FIRE FREQUENCIES

Plant specific fire frequency for power operation was determined by operational fire experience from 1981 to 1997. Similarly, fire frequency for outage was evaluated by outage fire data from 1981 to 1998. The plant specific fire frequency is $1.5 \times 10^{-1}/\text{ry}$ (reactor year) for power operation mode and $1.1 \times 10^{-1}/\text{outage}$ during shutdown for refuelling.

Cable routes of important components were identified room by room from the point of view of fire and safety. Fire frequencies of rooms, in which fire has same kind of effect on safety related components, were grouped, in order to get limited amount of fire initiators to be analysed in event trees.

Berry’s method, that was earlier used for the Swedish NPP Barsebäck, was used to determine room specific fire frequencies both during power operation and outage. The room specific fire parameters needed in Berry’s model were assessed separately, too.

According to Berry’s method a fire can ignite only if an ignition source and a combustible material in a room gets into contact with each other. Potential sources for ignition are human beings as well as mechanical and electrical equipments. Ignition probability is based on the ignition temperature of the fire load in the room. Berry’s method also takes into consideration the probability of self-extinguishment of the fire. In addition the fire can be suppressed by the persons that are present in the room or by fire-watch, which is also taken into account by Berry’s parameters.

The room specific parameters for Berry’s model were evaluated by expert judgement, which required walk-throughs at the plant. All rooms were visited in 1988 to determine the parameters for power operation mode. Parameters were updated and plant modifications were taken into account in 1993. In order to get the best possible understanding concerning the differences in fire ignition risk between maintenance outage and refuelling outage, both units were analysed separately in 1996 by visiting and evaluating all rooms. Finally, in 1998 parameters for maintenance outage were conservatively used to determine room specific fire frequencies during shutdown for refuelling.

3. SAFETY CRITICAL FIRES

The quantitative event tree analysis for power operation covers fires that cause shut down of the plant and at the same time degrade safety functions. During outage fires that cause a loss of decay heat removal and degrade safety functions were regarded as critical to nuclear safety and analysed quantitatively in event trees. Economical losses were disregarded and the turbine building was not included in quantitative analysis, since it was regarded as marginal to nuclear safety.

During the shutting down phase, essential shutdown phase, and start up phase fires are possible inside the containment, since the containment atmosphere is non-inerted during them. Fire in the containment during the shutting down phase and start up phase was
presumed to activate the containment isolation by containment temperature control, because isolation has a stronger impact on residual heat removal than a damage of a single component inside the containment.

Most of the containment isolation signals of the reactor protection system are disconnected during outage, and reconnected shortly before the start up phase, so that the fire risk at the outage consists of power cables only. At the same time with RPS-disconnection the personnel hatches are opened and the inner isolation valves of shut-down cooling system are mechanically locked at open position. From this moment to RPS-reconnection moment fires in containment were presumed not to be able to effect the residual heat removal.

4. QUANTITATIVE FIRE RISK AND PLANT MODIFICATIONS

4.1. Phase one — Safety System Components and their power cables

The principle of using four half capacity subsystems is applied in both units to all important safety functions. The four redundant circuits of the safety related systems are assigned to two main groups, located in physically separated areas. Within the areas separation by distance or by means of barriers is used between the redundant circuits. Figure 1 describes principle of separation in cable tunnel, where cables of two subsystem are located.

![Diagram of cable tunnel separation](image)

Figure 1. Principle of four half capacity sub-system separation at Olkiluoto NPP. Instrumentation cables are located at the lower part of the tunnel inside an iron shelter.

Tunnels containing cables of two redundant sub-system are supplied with sprinkler system, but however, before fire PSA analysis there were some deficiencies in sprinkler system coverage area. Power cables are located at the upper part of the tunnel and instrumentation cables are below them inside an iron shelter.

The first fire risk study, carried out between 1988 and 1991, concentrated on safety system components and their power cables. At the very beginning of the fire PSA the fire propagation was highly overestimated. It was assumed that a fire makes all components in a room unavailable, no matter what kind of fire protection is present. Later, the fire propagation was specified more realistically. Some typical propagation
parameters are in table 1. Parameters are based on experiments and statistic, but more accurate values may be estimated by fire simulation programs in the future.

### Table 1. Fire propagation parameters

<table>
<thead>
<tr>
<th>Parameter value</th>
<th>Application</th>
</tr>
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<tbody>
<tr>
<td>1.0</td>
<td>Fire in a power cable destroys all power cables in the sub-system</td>
</tr>
<tr>
<td>1.0</td>
<td>Fire propagation probability between sub-systems, without sprinkler system</td>
</tr>
<tr>
<td>0.1</td>
<td>Fire propagation probability between sub-systems, with sprinkler system and good separation by distance</td>
</tr>
<tr>
<td>0.1</td>
<td>Fire propagation probability between rows (sub-systems) in relay rooms</td>
</tr>
<tr>
<td>0.1</td>
<td>Fire propagation probability between power cables in pump rooms, where cables are protected passively and separated by distance</td>
</tr>
</tbody>
</table>

The core damage frequency due to fires as a result of the first study was $2 \times 10^{-5}$/ry. During the analysis was also identified that a fire especially in reactor protection system can cause tripping or spurious control of safety related systems and components. On the basis of the results and deterministic fire safety assessment fire protection improvements were started both in active and passive fire protection of the plant.

The aim was to decrease the fire propagation parameters in rooms, where cables or components of two sub-systems are located. Sprinkler system was extended to some new rooms and the location of existing sprinkler heads was improved. Also some of the sprinkler heads were changed to quicker ones.

Passive fire protection was improved by coating the power cables with fire retardant material in service water pump room. Also sub-systems in two large room were divided into separate fire departments by new fire wall. Due to improvements in sprinkler systems and in passive fire protection the original frequency of $2 \times 10^{-5}$/ry was decreased by factor 5 to $4 \times 10^{-6}$/ry. Later plant modifications decreased the core damage frequency due to fires to $3 \times 10^{-7}$/ry, although some of them were implemented because of other reasons than fire risk reduction.

### 4.2. Phase two — Extension of fire PSA to instrumentation systems

During the first fire risk study it was also recognised the importance of instrumentation on safety. Open circuits and short circuits can cause spurious actuation or prevent the function of safety system, so that Fire PSA was extended between 1992 and 1994 to include important safety related instrumentation.

The separation principle of RPS measurements and actuations is illustrated in figure 2. The vertical separation and shielding principle can be seen in figure 1. Fire propagation probability of 0.2 is used in cable tunnels for propagation from power cables to instrumentation.
Figure 2. Physical boundaries of the reactor protection system (RPS). The thicker broken line separates redundant subsystems to the main fire areas whereas the thinner broken line illustrates the separation by distance and sprinkling.

RPS is based on conventional relay technic. Output signal is generated as a result of 2-out-of-4 voting, so that the effects of fire on output is depending on failure mode of the cables and signal energising arrangement. The possible alternatives are on one hand open circuit and short circuit and on the other hand energised and non-energised tripping condition. Most tripping conditions are energised, in which case open circuit causes tripping of condition.

Fire in component control cables can cause miscontrol like spurious valve actuation or unwanted start or stop of pumps. This has been the case in many fires in NPPs. However, malfunction of a single component is less acute than the unwanted tripping of RPS, which can be caused by fire in RPS measurement cables. Spurious tripping requires malfunction of two conditions, but the unavailability of the signal would require three failing conditions, which is very improbable.

The extension of fire PSA to instrumentation increased the core damage frequency due to fires with 10% and the total core damage frequency with 1%. Modifications in active and passive fire protection, that were done as a result of the first fire risk study, decreased also the fire risk concerning instrumentation cables, which explains the small risk increase.

4.3. Phase three — Shutdown Fire PSA

The shutdown fire PSA (1996—1998) was based on existing plant specific PSA studies. In the modelling of outage specific fire events, event trees of shutdown PSA were used. The existing fire initiating events for power operation were analysed qualitatively in
order to discover, which events had an influence on safety during the outage. In addition, new events were identified.

The extension of fire PSA of Olkiluoto NPP to outage increased the core damage frequency due to fires with 8% and the total core damage frequency with 0.17% only. The core damage frequency due to fires during shut down is only $2.16 \times 10^{-5}$, which is almost three orders of magnitude lower than the total core damage frequency. The contribution of fires to the annual core damage risk is 2%. Figure 3 describes the core melt contribution in April 1999.

![Pie chart showing core melt contribution]

**Figure 3. Core melt contribution (CMF=1.4 \times 10^{-5}).** Fires include both power operation and outage.

On the basis of fire PSA for power operation and deterministic fire safety assessment fire protection improvements have been made both in active and passive fire protection of the plant. Improvements were also a good basis concerning fire risk during outage.

Considerable low risks due to fires during the outage are mainly resultant from good physical separation of four redundant trains and very wide active and passive fire protection, but they also ascribe to characteristic of outage. Reactor protection system is disconnected except the very beginning of shut down, so that the loss of heat decay is not possible through fire in instrumentation cables. Shortly after opening the containment air lock, the inner isolation valves of shut-down cooling system are mechanically locked open. After the removal of pressure vessel head both fire fighting water system and distribution system for demineralized water can be used as a source of water in addition to conventional cooling systems. Air has been credited as an ultimate heat sink in the case of boiling the water in reactor pool. Decay heat generation during outage requires only low capacity of water pumping especially in the case of boiling the pool water, which allows a high redundancy. Simple structure of pool water system, its capability and capacity for most of the time during outage to remove residual heat from the core, too, decreases the core damage risk due to loss of residual heat removal. All the valves of the system are manually operated, and no fire can close them. Therefore the system can only be lost in fires that damage power cables of the pumps.
Since the contribution to core damage frequency of the fires during outage was found to be several orders of magnitude below the annual risk, there is hardly any potential of risk reduction by means of improvements in fire protection and the improvements are not worthwhile.

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PSA Study for an Exemplary Plant Location of a German PWR Built to Earlier Standards

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ABSTRACT

For a selected plant location of an exemplary German nuclear power plant (NPP) built to earlier nuclear safety standards GRS has carried out a probabilistic analysis. For the selected relevant fire scenario of a cable fire a detailed analysis was performed with respect to the fire occurrence frequencies, the consequences of fire for the equipment affected, and the incident control by the safety systems required.

The study included simulations of the fire development and the consequential fire and fire related effects under realistic assumptions estimated from the plant specific conditions by advanced fire simulation codes performed by iBMB of the Braunschweig University of Technology.

The analysis furthermore included in-depth investigations on the probability of the screened out fire scenario considering the state-of-the art with respect to research activities on electrical cable and cabinet fires. In addition, the study covered the human influence factor taking into account the plant personnel involved and the fire brigades as well as the plant specific conditions regarding fire protection and control of the accident. A first rough estimate of the uncertainties of those parameters influencing the finally estimated core damage frequency was performed.
INTRODUCTION

In the frame of supervisory activities of the German federal state authority in the years 1995 to 1996 GRS has carried out a probabilistic fire safety analysis for a specific plant location with significance to nuclear safety: the cable spreading rooms. The NPP under consideration was designed to earlier standards. The cable spreading rooms located inside the switchgear building under the main unit control room were identified as a critical zone containing a relatively high fire load from insulation material of electrical cables as well as safety-related equipment. The objective of the probabilistic analysis was to assess the safety significance of deficiencies in fire protection observed in the frame of a deterministic fire hazard analysis.

Applying the so-called defence-in-depth safety concept as explicitly outlined with regard to the plant internal hazard "fire" in [1] and [2], it had to be assessed whether the nuclear protection goals of the different defence-in-depth levels were met. In the frame of the deterministic fire hazard assessment, a cross-check whether the requirements for the defence-in-depth levels 1 and 2 "Prevention of Initiating Events" (see Table 1) were met gave no indication on deficiencies. But the cross-check for meeting the requirements for level 3 "Control of Design Basis Accidents", particularly for fire protection, indicated that the existing measures are not in accordance with the German nuclear fire protection standards. The respective KTA fire protection standards (KTA 2101.1 [3] and 2101.2 [4]) give priority to physical protection by structural fire protection means. In contrary, the fire protection features in the cable spreading rooms are based on a separation of redundant equipment by distance with an additional stationary fire extinguishing system.

Estimating the relevance of a fully developed fire inside the cable spreading rooms as a beyond design basis accident with respect to the defence-in-depth safety level 4 the efficiency and reliability of independent emergency systems and accident management had to be quantified by a probabilistic analysis.
2 Description of the Selected Scenario

Figure 1 gives a schematic overview of the layout of the plant location under consideration.

On the upper level of the switchgear building the unit control room is located adjacent to other rooms containing electronic cabinets for all four redundant trains of the safety instrumentation and control (I&C) system as well as for the operational I&C systems. The cable routing of this equipment is leading to the next lower level compartments (cable spreading rooms) with a specific cable support structure close to the ceiling designed for the cable spreading. This level of the switchgear building with the cable spreading rooms consists of two compartments (see also Figure 2) and contains the so-called marshalling racks. All I&C cables routed from other NPP buildings to the switchgear building are connected to these marshalling racks, where they are further marshalled (spread) to the equipment for processing the respective signals.

These cable spreading rooms furthermore contain also electronic cabinets for the safety related or plant operations I&C systems. Outside the cable spreading rooms, the cable routes of the different redundant trains connecting these rooms with other ones on a lower level or with adjacent buildings are physically separated in principle by structural measures. The design concept in the respective NPP foresees a physical separation of the two redundant trains 1 and 3 (redundancy group A) from the redundant trains 2 and 4 (redundancy group B). However, inside the cable spreading rooms only the spatial separation of the redundant trains 1 and 3 from 2 and 4 is possible revealing the necessity for a detailed analysis of these plant locations. PCV cable insulation material represents the dominating fire load in these compartments.
3 Comprehensive Assessment Approach

The comprehensive deterministic and probabilistic approach applied by GRS, as outlined in more detail in [1] and [2], is based on the principle of the deterministic assessment cross-checking the efficiency of the defence-in-depth levels by quantifying the fire occurrence frequency, the probabilities for fire development as well as for the unavailability of systems affected directly or indirectly by the fire effects.

The basic principle of the assessment was to demonstrate whether the fire protection features are in a well-balanced relation to the other safety features of the NPP. The safety features of nuclear power plants are based on the defence-in-depth safety concept (as outlined in Table 1) requiring that for each of the four safety levels there are measures available which have to be taken to limit the consequences of a safety relevant event to this particular level, in other words, to reduce the probability that the next safety level will be affected. According to this safety concept with its different levels, probabilistic reference values can be assigned.

3.1 Prevention of Pilot Fires

For the specific scenario analysed the safety goals of the defence-in-depth levels 1 and 2 (see Table 1) are covered by the main protection goal "Prevention of Pilot Fires" inside the cable spreading rooms. Various measures (e. g. for the design of electrical equipment to avoid electrical ignition, selective electrical fusing, consequent prevention of combustibles, administrative procedures for prevention of externally available temporary ignition sources, being potentially available) inside the plant have been taken to avoid a fire due to electric or other causes.

Due to the fact that the database for fires in German NPP is much too small for a realistic fire PSA (see also [5]), the fire occurrence frequency for the selected plant location was estimated mainly based on data from US NPP [6] and on additional generic data from the operating experience of French and German PWR. Taking furthermore the above mentioned preventive measures into account, the specific occurrence frequencies for pilot fires starting in the I&C cabinets and for fires starting directly in the cable insulation material were estimated, considering results from fire experiments with
electronic cabinets carried out by VTT Finland [7], [8] on the one hand and human influence factors in case of maintenance on the other hand.

With regard to the occurrence frequency of pilot fires, the analysis revealed the following results:

- for I&C cabinets: \( < 10^3 / \text{reactor year} \), and
- for cable distributions: \( < 10^4 / \text{reactor year} \)

### 3.2 Control of Fires as Design Basis Accidents

A pilot fire cannot be excluded by means of the defence-in-depth level 1 and 2 safety precautions. In case of the occurrence of a pilot fire credit is taken from manual fire fighting as well as from the stationary CO\(_2\)-gas fire extinguishing system (see also Figure 2) to control the fire. Main goal of a successful fire fighting with respect to the plant safety is to avoid any consequences affecting more than one redundancy group. It was therefore necessary to perform in-depth investigations to assess in detail the relevance of the influencing parameters such as the fire development and propagation, steps of the fire fighting, fire effects and consequential failures of systems.

The analysis included the modelling of the fire development and propagation as well as of the consequential fire related effects under realistic assumptions derived from the plant specific conditions.

#### 3.2.1 Fire Modelling

For the assessment of the questions arising in the context of controlling a pilot fire in a cable distribution of the cable spreading rooms the fire event sequence had to be analysed. For the selected scenario of a cable ignition with a fire propagation to adjacent cables the temperature in the fire compartment as well as the hot gas layer size had to be calculated by a fire simulation code. The fire modelling was performed by experts from a German testing institution, the iBMB of the Braunschweig University of Technology, in co-operation with GRS using a multi-zone code based on well known heat balance models. The empirical models applied in this code have been verified generally for the given application (see [9] and [10]). Furthermore, they are suitable in
principle to determine the essential fire effects (e.g. temperature and layer heights of the hot gas and cold gas layer). In this context, it has to be noted that such models are not able to consider the time period of the fire ignition in a realistic manner. Moreover, there is a need of verified input data from fire experiments being applicable to the scenario to be modelled. The zone-model calculations have shown that the hot gas temperature as well as the expansion of the released hot gases strongly depend on the given input parameters, e.g. the speed of the fire spreading on cables. To validate and improve the existing available empirical values for the fire behaviour of PVC cables from former experiments [11] and [12], more recent nuclear specific cable fire experiments have been carried out by iBMB. Table 2 gives insights in the results relevant for the in-depth investigations in the frame of the presented analysis. The results of the experiments are in detail outlined in [13], [14], [15], [16]. Additional information is given in [17].

The results of the fire simulation calculations can be summarised as follows:
Based on the pessimistic assumption that failures of cables will occur as soon as the temperature at the location of cables exceeds 200° C, it could be demonstrated reliably by the simulations that there is enough time for a successful fire extinguishing, mainly by the manually actuated stationary CO₂ gas extinguishing system.

3.2.2 Probabilistic Event Analysis

The probabilistic analysis of the event sequence focused mainly on assessing the unavailability of the fire extinguishing measures during the period from ignition up to the time of failures in more than one of the two redundancy groups.

Two scenarios were analysed in more detail, for which the respective event trees had to be developed, one of these for a fire starting in an electrical I&C cabinet, the other one for a fire starting in a cable distribution. Due to the additional availability of manual fire fighting means (first aid fast response fire fighting shift personnel as well as professional fire brigade) on-site, the unavailability of the fire extinguishing features was estimated to be less than 10⁻² per demand for the first scenario of a fire starting in an electrical I&C cabinet. For the second scenario of a cable fire starting at the cable distribution, only the stationary extinguishing system is of importance. The reliability of this system has been estimated by a simple fault tree. The quantification is based on

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statistical data from German insurance companies taking into consideration plant spe-
cific conditions, e.g. weaknesses in the construction, and, in particular, the manual 
actuation of the extinguishing system. Thus, the unavailability of the fire extinguishing 
measures in case of a cable fire was estimated to be less than \(10^{-7}\) per demand.

For all paths of the event trees without a successful fire extinguishing, a fully de-
veloped fire with a total loss of all safety related functions of the equipment affected inside 
the fire compartment was assumed.

As a first result of the probabilistic analysis, the frequency for a so-called plant hazard 
state (see [18]) represented by a fire with propagation from one redundancy group to 
the other causing a consequential failure of significant safety functions (e.g. failure of 
main feedwater and emergency feedwater supply) was estimated to be less than \(10^{-5}\) 
per reactor year.

3.3 Control of Beyond Design Basis Accident Conditions

In the unlikely case of a total loss of significant safety functions due to a fully de-
veloped fire in one of the cable spreading rooms, additional equipment not affected by the 
fire is available which can be used to ensure the plant safety. This equipment is part of 
an independent emergency system (IES) designed and installed for very unlikely ex-
ternal events, e.g. an impact to the plant resulting from an aircraft crash. The reliability 
of safety functions required in case of a fully developed fire in a cable spreading room 
has been examined in detail for the above mentioned event sequence.

Main emphasis of the probabilistic event sequence analysis was the quantification of 
the unavailability of an additional emergency feedwater supply. In case that the emer-
gency feedwater supply is not available a lot of manual procedures will be needed. The 
probability that this feedwater supply is not available during the time period for ensur-
ing the residual heat removal and core cooling has been evaluated considering in par-
ticular the human influence factor. The quantification of the human factor was based 
on an internationally accepted methodology (so-called THERP method) as outlined in 
[19] and additional data on human reliability analysis from [20]. A maximum period of 
50 or 60 minutes (depending on different results of thermohydraulic calculations) was 
assumed for the actuation of the emergency feedwater supply. In this context, it has to
be stated that the assumption of a maximum time period of 50 min being available for a successful actuation of the emergency feedwater supply is conservative due to the recent expert knowledge.

The probability that faulty plant personnel activities result in a failure of the emergency feedwater supply was thus estimated as follows:

- \(1.0 \cdot 10^2\) per demand in case of 50 min time being available, and
- \(3.4 \cdot 10^3\) per demand in case of 60 min time being available.

Main influencing parameters of these results are the human tasks of correct and punctual diagnostic and decisions. The unavailability of the technical system is less than \(10^6\) per demand.

Considering these values, playing a decisive role for the control of a beyond design basis fire accident, the core damage frequency for a severe accident was determined to be less than \(10^{-7}\) per reactor year.

### 3.4 Results of the Uncertainty Analysis

A first rough estimate of the uncertainties of the results influencing the finally estimated core damage frequency was performed. This estimation showed on the one hand that there are a lot of pessimistic deterministic assumptions which lead to an overestimation of probability values. On the other hand, the fire occurrence frequencies includes several unknown uncertainties.

A detailed uncertainty analysis has been performed only for the quantification of the human influence factor in case of the actuation of the additional emergency feedwater supply.

It can be reliably assumed that the deviations between the results gained from the analysis by GRS (as mean values) and a 95 % upper bound are in the same range as the deviations estimated in the frame of other recent PSA.
Conclusions

The results of the probabilistic fire risk analysis of the selected scenario of a cable fire in a cable spreading room of an operating NPP designed to earlier standards have demonstrated that the preventive safety precautions taken with regard to the fire safety fulfill their intended tasks in the frame of the defence-in-depth safety for safety related events in NPP. Furthermore, it could be shown that the core damage frequencies estimated are considerably lower than the internationally given [21] probabilistic reference values for the core damage frequency of plant internal hazards.

As a consequence, neither an immediate demand for corrective actions nor a fundamental change in the fire safety concept has to be required. In particular, the priority in general given to structural fire protection measures in the German nuclear fire protection standards for the design of NPP should not exclude to take other, additional protection measures in those plants build to earlier standards where a strict physical separation of redundant safety related equipment is not in any case possible.

The final assessment did not exclude the optimisation of particular measures for risk minimisation considering that these have to be well balanced under cost-benefit aspects. The PSA carried out has given indications. It is strongly recommended to equip the stationary fire extinguishing system with automatic actuation devices as well as to protect particular cables in single cases by intumescent coatings. Due to the first measure the reliability of fire fighting will increase, the second measure will reduce the risk of a cable ignition and a fast fire propagation.
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(INSAG); IAEA, Wien, 1995
Table 1: Defence-in-depth concept

<table>
<thead>
<tr>
<th>Defence-in-depth Level</th>
<th>Contributions to Safety</th>
<th>Plant Condition</th>
<th>Strategy</th>
<th>Safety Precautions</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Normal Operation</td>
<td>Planned Operation</td>
<td></td>
<td>Prevention of initiators</td>
<td>Operational systems</td>
</tr>
<tr>
<td>2. Anticipated Operational Occurrences</td>
<td></td>
<td></td>
<td>/control of design basis accidents</td>
<td>Reactor protection system; Safety systems</td>
</tr>
<tr>
<td>3. Design Basis Accidents</td>
<td></td>
<td></td>
<td>Transfer of beyond design basis accidents into safe plant conditions; Damage minimisation</td>
<td>Emergency systems; Accident management (AM)</td>
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<td>4. Severe Accidents</td>
<td></td>
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Table 2: Fire behaviour of unprotected PVC cables

<table>
<thead>
<tr>
<th>Arrangement</th>
<th>Pre-heating Temperature</th>
<th>Ignition Source</th>
<th>Observation</th>
<th>Flame Propagation</th>
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</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Ignition</td>
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<td>Flame Propagation</td>
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<td>yes</td>
<td>Limited</td>
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<td>yes</td>
<td>Limited</td>
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<td></td>
<td></td>
<td>3 - 4 min</td>
<td>3 - 5 cm / min</td>
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<td></td>
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<td>&lt; 1 min</td>
<td>110 - 120 cm / min</td>
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<td></td>
<td></td>
<td>1.5 min</td>
<td>Limited</td>
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<td>(control cables)</td>
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<td>12 min</td>
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<td>(power cables)</td>
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<td>0.5 min</td>
<td>16 - 30 cm / min</td>
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<td>(control cables)</td>
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<td>8 min</td>
<td>60 - 240 cm / min</td>
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<td>(power cables)</td>
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<td>45 s</td>
<td>130 - 160 cm / min</td>
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<td>40 s</td>
<td>360 - 480 cm / min</td>
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</table>
Figure 1: Schematic layout of the switchgear building (cross section)

Figure 2: Fire protection features in the cable spreading room
Session IX
Future Needs and Development

- Fire PRA Needs - Regulators Perspective - Mr. Edward A. Connell, Nuclear Regulatory Commission, United States
  (Paper missing. The editor wishes to apologise to Mr. Edward Connell regarding this omission. His original and all copies were errantly lost and could not be recovered).

- Fire PRA Needs from the Utility Fire Protection Programme Manager and Engineering Consultant Perspectives - Franklin D. Garrett, P.E., Department Leader, Emergency Services Programs, Palo Verde Nuclear Generating Station, United States and Elizabeth Kleinsorg, P.E., Vice-President, Fire Protection & Hazards Analysis, Duke Engineering & Services, United States

- Outline of a Performance-Based Fire Safety Design Method for Buildings in Japan - Professor Takeyoshi Tanaka, Disaster Prevention Research Institute, Kyoto University, Japan

- A Reference Framework for the Development and Documentation Human Reliability Analyses for Fire PSAs - Dr. Javier Yllera, Consejo de Seguridad Nuclear (CSN), Spain

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Fire PRA Needs
From the
Utility Fire Protection Program Manager and Engineering Consultant Perspectives

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Introduction

Fire protection programs at U.S. nuclear generating stations have received significantly increased attention in recent years. In addition, the utility industry and the U.S. Nuclear Regulatory Commission have been pushing forward with application of risk informed and performance based fire protection methods. This combination has created a challenging environment for fire protection program managers and engineering consultants. The purpose of this paper is to provide the program manager and engineering consultant perspectives on the application of Probabilistic Risk Assessment (PRA) methods to nuclear power plant fire protection programs.

The Fire Protection Program at Palo Verde Nuclear Generating Station Units 1, 2, and 3 is built around nine fundamental elements. The following discussion presents a brief description of each element and identification of opportunities and challenges associated with implementing PRA methods. Also provided is a discussion on the fire protection engineering perspectives associated with consultant’s use of PRA methods to justify plant changes.

Fire Protection Program Elements

1. Hazards Analysis and Engineering Design

Background - Each fire area containing structures, systems and components important to safety or representing a potential hazard requires engineering analysis. The analysis should consider potential in situ (cable concentrations) and transient fire hazards (paints and solvents), determine the consequences of a fire in any plant location on the ability to safely shutdown the reactor and maintain it in a shutdown condition and the ability to minimize and control the release of radioactivity to the environment. Based upon the consequences of potential fire scenarios, engineering analyses specify measures for fire prevention, detection, suppression and containment, and considers the effect of fire suppressants on plant systems. These engineering analyses are typically documented in two major engineering design basis documents, which are the fire hazard analysis and fire safe shutdown analysis. These documents are in addition to standard design documentation for fire protection systems and features.

The impact of maintaining fire protection design basis documentation can be significant. All plant modifications and changes to operating practices require evaluation for impact. Procurement documentation and engineering reviews are required to assure material substitutions are acceptable.

Impact of Fire PRA – If use of fire PRA evaluations are adopted, they would result in additional design basis documents requiring configuration management controls. Transitioning to a risk-informed fire protection program would also require additional professional expertise needed for the PRA based evaluations, which would add another interface to the engineering process. However, these burdens could be offset through
simplification and cost reduction in other areas. For example, traditional safe shutdown analysis methods require detailed evaluation of plant areas that may not be risk-significant. Introducing a risk screening may result in fewer evaluations. Benefits also could be realized through improved understanding of risk contributors and optimizing design features.

2. Fire Prevention Activities

**Background** - Fire prevention activities consist of control of transient combustible material, control of ignition sources, and periodic inspections to ensure continued compliance with these administrative controls. Control of combustibles and ignition sources are typically accomplished through the “work control process” utilizing permit authorizations.

Fire prevention activities can have a large overall impact on productivity and cost. For example, stringent combustible and ignition source controls can impede maintenance work and consume operations resources through review and approval of permits. The Fire Protection Program must appropriately balance the level of fire prevention performance required against the cost and productivity impact of the controls.

**Impact of Fire PRA** - A risk informed Fire Prevention Program could provide a better understanding of fire safety contribution on a fire-area by fire-area basis, thereby allowing the development of flexible programs to reduce operational and maintenance costs. Additionally, fire mitigation measures could be improved through identification of risk significant fire areas that would benefit from enhanced controls.

3. Inspection, Testing, and Maintenance of Fire Protection Features

**Background** - At a well-protected nuclear power plant, the collective amount of fire protection features that must be inspected and maintained is extremely large. At last count, Palo Verde Nuclear Generating Station was equipped with 256 automatic fire sprinkler/spray systems, 22 carbon dioxide systems, 18 Halon systems, fire detection and standpipe systems throughout all permanent facilities and a dedicated firewater supply system with redundant 500,000 gallon water tanks. Passive fire barrier features are extensive, including fire doors, fire dampers and approximately 22,000 fire rated penetration seals. In addition, there is a very large amount of manual firefighting equipment and apparatus.

Inspection, testing, and maintenance can be a significant impact on resources. Often the care of fire protection features is distributed among several organizations and the collective impact is difficult to determine. Frequency of activities has historically been based on prescriptive requirements dictated through codes and standards. Often property insurance standards play a key role in determining extent and frequency of activities.

**Impact of Fire PRA** – Due to administrative and oversight burden associated with caring for fire protection features credited for nuclear safety, the cost is high relative to those
installed for industrial fire protection purposes. PRA methods could provide value in identifying fire protection features that should receive the higher level of quality oversight.

4. Impairment and System Status Control of Fire Protection Features

**Background** - System status and control of fire protection (FP) features must be strictly administered. In the event FP features become inoperable, compensatory actions should be implemented. In the U.S. compensatory actions are mandatory for FP features protecting safety related areas and implementation is typically required within 60 minutes. Unique engineering analysis may be required for determining appropriate compensatory actions for inoperable plant components/systems credited for safe shutdown in the event of a fire. System status is important during fire emergencies for supporting rapid incident command decisions.

Typically this element is the responsibility of unit operations staff and can be complex considering the amount of testing and maintenance activities resulting in inoperable conditions. Fire watches are commonly used when passive and active fire protection features are inoperable. A single 24-hour continuous firewatch post typically requires five full time workers dedicated to the function. Inoperable plant components credited for safe shutdown in the event of fire require specialized engineering analysis to determine appropriate compensatory measures.

**Impact of Fire PRA** – Determining compensatory action implementation times would be a good application for PRA techniques. Firewatch costs can be significant over the operating life of a plant. PRA techniques could also be useful for providing insights to determine the type and extent of compensatory actions. Through a better understanding of which fire protection features protect risk significant areas, prioritization of repair resources could be improved.

5. Training

**Background** - Training associated with fire protection involves to some extent every individual at a facility. General topical areas for training include general employee, emergency responders, maintenance technicians, operations, and technical staff. More progressive facilities have integrated fire-induced circuit failures into fire drill exercises utilizing the simulator as the active control room. This has allowed the simulator control room staff to be challenged far beyond the traditional prompting methods when utilizing an operating unit control room.

**Impact of Fire PRA** - PRA insights could improve training through identification risk significant scenarios and high frequency contributors. Over time the fire brigade and operations staff would become very skilled at responding to the most risk significant through focused training.
6. Emergency Fire Response

**Background** - Emergency response typically consists of operations fire brigades trained and equipped in accordance with regulatory requirements. A select number of U.S. facilities have dedicated professional fire departments such as Palo Verde. Although multiple organizations may share responsibility, most facilities have the capability to respond to fire, medical, hazardous materials and rescue events. The more challenging events will involve multiple event responses.

For U.S. plants, the minimum fire brigade complement is five members. Fire brigade members cannot have shift responsibilities that would conflict with the ability to rapidly respond to a fire. For operations fire brigades, most operations personnel require qualification training. In total the implementation cost of emergency response is high.

**Impact of Fire PRA** - Emergency response capability is considered a basic element of fire protection programs and is not a likely candidate for PRA-based cost reductions. However, improved understanding of emergency response performance and reliability assumptions could improve training standards. Pre-fire planning could be improved through incorporation risk insights.

7. Quality Assurance

**Background** - Quality assurance programs ensure that the critical aspects of design, procurement, administrative controls, inspection and maintenance of fire protection features are available and functional. This is typically accomplished through audits, self-assessments, corrective action programs and procedural requirements.

Record keeping can be extensive to demonstrate compliance with expectations. Quality assurance requirements impact resources through personnel cost, adding complexity to processes and increased cost of materials.

**Impact of Fire PRA** - PRA methods could be used to establish a graded quality assurance program for fire protection. Quality assurance requirements could be focused on risk significant features with the remainder being controlled as industrial systems and equipment.

8. Regulatory Compliance

**Background** - Ability to demonstrate compliance with regulations and commitments are a necessary but difficult challenge. The documented licensing basis needs to clearly describe all applicable regulatory requirements and exemptions/deviations. Changes to the facility and the fire protection program require evaluation to assure long-term compliance. When noncompliance issues or events are identified, reportability evaluations are performed which typically include determinations of safety significance.
The impact of maintaining fire protection licensing basis documentation can be significant as well as maintaining the historical knowledge. Currently most reportability evaluations are performed qualitatively.

**Impact of Fire PRA** - PRA techniques could provide technical justification for cost-beneficial changes associated with physical plant modifications and procedural activities. Determination of safety significance of adverse events and noncompliance issues would be of benefit when evaluating regulatory reportability.

9. **Risk Management**

**Background** - The objective of a risk management program is to protect employees, preserve property and protect corporate assets and earnings from accidental loss. Often this is accomplished through compliance with local building and fire codes and insurance. Insurance companies, however, typically establish requirements for all high value locations regardless of nuclear safety. Insurance requirements can be a condition of insurability or recommendations, which result in higher premium payments if not implemented.

Implementation of non-mandatory insurance requirements has historically been a cost/benefit decision based on judgment. The Fire Protection Program typically does not have tools to reasonably predict financial loss over the remaining plant life. Fire-induced plant downtime will justify any and all fire protection modifications. However, convincing senior management of the present value of future fire losses associated with a specific event has proven to be difficult at best.

**Impact of Fire PRA** - A great opportunity exists with the application of PRA techniques to insurance standards and establishment of premiums. In a competitive insurance market, valid loss predictions could create a significant advantage.

**Comments on Use and Acceptance of PRA Methods**

There are many opportunities to use PRA insights to focus Fire Protection Program efforts on those elements or features that have the greatest impact on risk. However, the use of PRA techniques is not without its own inherent "risk." From the users perspective the risk would be associated with the ability to provide economical solutions that would result in a high likelihood of regulatory acceptance.

Economics is a significant factor in utility decision making in part due to the changing competitive market. Often a two-year return on investment is required for facility improvements when nuclear safety is not a consideration. In most situations, compliance with the existing prescriptive regulatory requirements is deemed sufficient to provide an acceptable level of safety. Therefore, PRA methods would need to be competitive with existing qualitative methods for application by fire protection consultants. In the event nuclear safety is in question, freedom to apply more costly methods is increased.
Past applications of PRA methods to fire protection have tended to be generalist exercises. Currently applications are moving much more towards detailed decision tools. This has created controversy in that a large percentage of the industry views PRA applications as “supporting” the answer and not “providing” the answer. Although the future shows promise, the perception is that the methods are not mature enough to independently provide regulatory compliance solutions.

The U.S. regulatory environment is changing to be more accepting of PRA applications to fire protection. Unfortunately there is minimal experience with submittals being accepted. Recent attempts have been met with an extreme questioning response and long duration’s to reach agreement. Until there is a common understanding of what will achieve regulatory acceptance, there will be a reluctance on the part of utility fire protection engineers to utilize PRA as a primary tool.

Foundation data for fire PRA applications, such as fire event frequencies, has been improving over time. However, much more data is needed to gain regulatory acceptance in areas such as component damage thresholds and impact of smoke on component functionality. The thermal response of nuclear facilities to fire environments requires further investigation as it applies to validating fire models.

The final point is consensus within the PRA expert community. Participating in the effort to develop NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, has provided great insights regarding how much agreement exists within the PRA technical community. The experience of the author indicates consensus and agreement is difficult to achieve. As a result, it has been equally difficult for the fire protection engineers to fully embrace the accuracy and acceptability of PRA methods. It would seem that there is much more work to be accomplished prior to PRA methods becoming a routinely used item in the engineer’s toolbox.

**Conclusion**

Application of fire PRA methods to nuclear power facilities has the potential to provide great value in the future. Most of the fire protection elements can be improved both from safety and economic perspectives. However, much work needs to be done to improve maturity of the methods and establish thresholds of acceptance. In the future as fire PRA methods are further developed and standardized, the insight gained through application of these methods will provide value as a support tool for decision making.
OUTLINE OF A PERFORMANCE-BASED FIRE SAFETY DESIGN METHOD OF BUILDINGS IN JAPAN

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1. INTRODUCTION

It can be said that fire safety measures of buildings have been controlled, not only in Japan but also in any other country, basically by the detailed prescriptive standards stipulated in building or fire regulations. Traditionally, these standards have been developed by the discretion of so-called “fire experts”. It should be duly recognized that the safety of buildings against fire have been remarkably improved thanks to such standards. As a result, fires are not considered as a major societal issue at least in most of the developed countries.

However, adverse effects have been also incurred. That is, the multiplication of such prescriptive rules have resulted the tremendous complexities in understanding of the meanings of the rules, the increase of fire protection cost, unnecessary restrictions on building designs and so forth. Such prescriptive standards are convenient in the sense that particular technical expertise is not needed for designers nor for building officials, but its serious disadvantage is that they resist to any in compliance however trivial it seems from the view point of safety, and thereby discourage the use of innovative materials, products, construction technologies and novel designs.

Recognizing such problems and also considering the significant progress achieved in the area of fire science and engineering, Building Research Institute (BRI), Ministry of Construction, Japan undertook a five-year project called “Development of Fire Safety Design Method of Buildings”, 1982 through 1986. In this project, a performance-based fire safety design system was addressed. This system was favorably received among building industries and design firms, which may be proved by the remarkable increase of the fire safety designs applied for the approval by Minister of Construction after the end of the 5 year-project.
Figure 1: Number of Fire Safety Designs Applied for Article 38 of BSL

Unfortunately, however, the design system developed in this project was far from perfect. The most serious drawback is that the system can not yet be independent from the existing regulations, in other words, this can only be used for partially rationalizing the provisions in the Building Standards Law of Japan (BSL). Hence, efforts are still continued by BRI and by The fire safety design committee of Architectural Institute of Japan (AIJ) to improve the design system.

2. PURPOSE AND PHILOSOPHY

Precautions for fire safety in buildings have usually been established as mandatory rules by the authorities responsible for public safety, and enforced through administrative systems for building control. Most of the complaints and pressure from owners or builders of buildings associated with fire precautions arise in connection with the inflexibility of these mandatory regulations. There is no objection on the necessity to assure public safety. What is really needed is a design system which allows more flexibility while assuring the safety equivalent to the existing regulations.

In general, the requirements of a mandatory regulation are supposed to be restricted to what are absolutely indispensable for public safety and welfare since the abuse of mandatory rules may induce the danger of violations of basic human rights. The fire safety design method was intended to be a design system which can be used as an alternative to the existing BSL. Thus the design system should be equivalent to the Law. This means that the objectives and levels of safety of the design system must be basically the same as those required by the BSL, in other words the requirements in the design system are minimum.
On the other hand, the design method must be different in some respect from the BSL to allow more flexibility in building design. This is intended to be achieved by the objective-based structure of the design system and introduction of performance standards.

3. STRUCTURE OF THE DESIGN METHOD

A major part of the rigid nature of prescriptive provisions is caused by that they do not explicitly disclose what their purposes are, what scenarios of fire they assume and what level of safety they intend to be attain. What we had to do was to analyze the BSL, its related government orders, MOC orders and other related documents to identify what the prescriptive provisions mean. The objectives and the functional requirements for fire safety of buildings were thus identified. And the technical standards for verification of compliance were developed in view of equivalence to BSL provisions and flexibility.

3.1 Objectives
According to the analyses of BSL and the other related documents, the objectives of the fire safety provisions of BSL are considered to be as follows:
(1) Fire Safety of Individual Buildings
   (1.1) Prevention of Fire
   (1.2) Exclusion of Highly Hazardous Substance
   (1.3) Safety to Life
   (1.4) Prevention of Damage to Third Parties
   (1.5) Assurance of Fire Fighting Activity
(2) Mitigation of Urban Fire

3.2 Functional Requirements
The functional requirements are the description of the means specifically designated to each of the objectives to achieve their goals. Let’s take an example of the functional requirements for objective (1.3) Safety to Life. Although essential means for life safety may depend on type of hazards, it has been considered as indispensable to provide buildings with adequate means of escape for the safety to life in case of fires. It can be said that the means to assure life safety are virtually specified only to assure safe evacuation. This common practice certainly has a good reason considering the nature of building and building fires.

A number of provisions can be found in the existing regulations regarding the assurance of safe evacuation, which may be interpreted and consolidated into the functional requirements of which items are as follows:
(1.3) Safety to Life
   (1.3.1) Evacuation planning
      (1.3.1.1) Plans prepared in advance
      (1.3.1.2) Plans include all potential occupants
      (1.3.1.3) Plans consider all important building uses
      (1.3.1.4) Plans are practicable
   (1.3.2) Restriction on the use of certain materials
   (1.3.3) Assurance of safe refuge
(1.3.3.1) Adequate refuge(s) provided
(1.3.3.2) Location of refuges
(1.3.3.3) Safety of refuges
(1.3.3.4) Appropriate condition for staying
(1.3.3.5) Alternate refuges depending on fire location
(1.3.4) Assurance of safe path of egress
(1.3.4.1) Assurance of at least one available path of egress
(1.3.4.2) Exits are clear and continuous
(1.3.4.3) Proper capacity and design for egress movements
(1.3.4.4) Exits are safe from dangers due to fire
(1.3.4.5) Special protection for unique circumstances

A definition is given to each of the requirements. These are the verbal manifestation of the principles for fire safety of buildings.

Likewise, functional requirements are given, together with their verbal definitions, to the other objectives. It is these principle statements that provides the grounds for physical provisions to be imposed, so they must be considered as the reflection of the agreement within the society of interest on the fire safety of buildings. It is vital for smooth operation of the design system that whatever is indispensable must be disclosed here and whatever provisions must not be imposed if not based on any of the requirements explicitly stated.

3.3 Technical Standards for Verification of Compliance
With only verbal statements, it is impossible to verify if a specific design of a building meet the requirements. It is necessary to provide each of the requirements with the technical standards which unequivocally indicate what are physically required for the building. These technical standards play important roles to determine the level of fire safety of buildings. There is no denying that the safety requirements, but this does not mean that the requirements are absolute. They are supposed to be more or less relative to the needs of the building. There are different under and the

Objective

Functional Requirement

Technological Standard

Prescriptive Standard

Performance Standard

Design Fires

Safety criteria

Fire Models

Fire Tests

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3.4 Performance Standards
In principle, the technical standards have to be given in terms of measurable or calculable values. In this sense, it can be said that both specification standards and performance standards are eligible as such unambiguous standards. However, specification standards inherently have the defects of stubbornness, so in view of flexibility it is desirable that as most as possible standards are performance standards, although at this moment it is, to some extent, inevitable to use other type of standards such as specification and deemed-to-satisfy.

The key elements of the performance standards consist of design fires, safety criteria and fire models. These respectively correspond to design load, allowable stress or strain and structural calculation methods in structural design system.

3.5 Design Procedure
The fire safety design procedure based on the performance standards will be basically the same for any requirement, which is illustrated in Figure 3, although specific standards and the relevant fire models are different depending on the specific requirement. This procedure is actually the procedure to verify the compliance of a design to the requirements. A building has to clear specified safety criteria under specified fire condition.

![Diagram](image)

Figure 3 Fire Safety design Procedure Based on Performance Standards

4. CONSIDERATION ON SAFETY LEVEL

4.1 Acceptable Safety
It follows that the level of safety, in other words, the level of compliance to the requirements is determined by the combination of the corresponding design fires and safety criteria. Higher level of safety can be attained by imposing severer design fires and/or stricter safety criteria.
The design fire should include not only fire size but also pertinent scenarios: Even though the size of fire is the same, whether doors are open or not, for example, will make a significant difference in fire dynamics, hence safety to life, properties etc. What is important to stress is that probabilistic aspect of fire must be embodied in the design fire so that compliance verification can be performed with a minimum number of deterministic calculations. Otherwise, it will be practically impossible to operate a performance based design system.

The connotation of a design fire is illustrated in Figure 4, where the design fire is expressed by size of fire for simplicity. The solid line stands for a conceptual probability of fire incidences versus the size of fire. Statistically, like many other accidents, small size fires break out fairly frequently, but the larger the size of fires the less frequent their occurrence. The essential role of design fire is to require such fire safety measures that can cope with the hazards represented the design fire. The implicit premise is that the hazards which might be caused by fires smaller than the design fire are removed by the safety measures as a matter of course.

![Figure 4 Connotation of Design Fire](image)

4.2 Equivalence to Existing Code
Naturally, an extent of residual risk remains, in other words, we have to accept that the safety is no longer assured in the event a fire exceeding the design fire happen to break out. The residual risk and the expected loss due to fire may be decreased by setting a severe fire condition as the design fire. But it will in return result the increase of indirect loss such as cost of fire protection and inconvenience in normal use. Theoretically, the desirable design fire will be such that minimize the total of the expected direct and indirect losses. Practically, however, it is next to impossible to estimate the total cost. Also, in reality, safety cost in general tend to be determined based on the risk perceived by people rather than the risk actually exists. On the other hand, people in the countries where fire loss is stable tend to accept both the fire loss and the cost for fire safety measures which are controlled by the current regulations. In fact, nothing is more legitimate as the expression of societal agreements on acceptable fire risk and cost than the existing regulations in the countries. Therefore, the design fires should be so determined that the level of safety attained by the existing fire safety regulations can be
reproduced by the fire safety designs based on the performance standards. Note, however, this does not mean that the same prescriptions as the existing provisions be retained but means the same level of fire safety performance be assured.

4.3. Consistency of level of safety

Accepting a certain degree of residual risk implies that we have the expected loss by fire during the life time of a building $E$ given by

$$ E = P_{fire} P_{fail} L_{damage} Y_{life} $$

where $P_{fire}$ is the probability of fire occurrence in the building per year, $P_{fail}$ is the probability that the safety measure fails in the fire, $L_{damage}$ is the loss caused by fire when the safety measure fails and $Y_{life}$ is the life year of the building.

As an example of application of this concept, let's consider required fire resistance of a structural element on a floor of a multi-story building. For simplicity, we assume that the floors upper than the fire floor must be abandoned when the structural member collapsed by the fire broken out on the floor. In this case, only the fully developed (flashed over) fires become issue, so we can normally expect that

$$ P_{fire} = p_f^* A_{FLR}, \quad P_{fail} = p_{FO} p_{yield}, \quad L_{damage} = NA_{FLR} $$

where $p_f^*$ is fire occurrence probability per unit floor area, $A_{FLR}$ is the floor area, which is assumed to be the same on any floor for simplicity, $p_{FO}$ is the flashover probability, $p_{yield}$ is the probability that the structural member yields by the fire exposure, $N$ is the number of floors above the fire floor.

Analyzing the provisions in the existing regulations, it is probably true that the regulations stand on the premise, although implicitly, that the expected loss of fire should be the same for a building with any height or size. That is, if considering two different buildings here it follows from Eqn.(1) and (2) that

$$ p_f^* p_{FO} N A_{FLR}^2 Y_{life} P_{yield} = p_f^* p_{FO} N A_{FLR}^2 Y_{life} P_{yield} $$

or

$$ P_{yield} = \left( \frac{p_f^*}{p_f^*} \right) \left( \frac{p_{FO}}{p_{FO}} \right) \left( \frac{N}{N} \right) \left( \frac{A_{FLR}}{A_{FLR}} \right) \left( \frac{Y_{life}}{Y_{life}} \right) P_{yield} $$

This equation imply that the probability of yield of the structural members should be smaller for the building having occupancy with higher fire occurrence probability, more stories, larger floor area and longer life time. Incidentally, Canadian statistics indicate flashover probability is reduced to about 1/4 - 1/5, hence $P_{yield}$ for sprinklered buildings can be 4 - 5 times larger than unsprinklered buildings.

Although the provisions on fire resistance in the current regulations seem to empirically reflect the relation of Eqn.(4), clearer recognition of this relationship will be beneficial for establishing a more consistent standards. The right hand side of Eqn.(3) may be considered to correspond to a reference conditions. Therefore, if we can identify the reference conditions of buildings with one hour fire rating, for example, and find $P_{yield}$
for the case, then the allowable yield probability can be rationally determined. In addition, if some statistically-based fire load density distribution is available, the probability can be interpreted to the residual risk when a fire load density is prescribed as the design fire condition.
A REFERENCE FRAMEWORK FOR THE DEVELOPMENT AND DOCUMENTATION OF HUMAN RELIABILITY ANALYSES FOR FIRE PSAs

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INTRODUCTION

Within the framework of PSAs, human reliability analyses are a systematic process for analyzing actions performed, or which might have to be performed, by people working in a nuclear power plant. This process enables potential human errors influencing nuclear safety to be identified, analyzed, described, modelled and quantified. The influence of the human being on risk is due to errors which may lead to the reduction of safety system availability, may give rise to incidents, may lead to incident progression not being mitigated or even to incident conditions worsening. Human error is taken to mean any human action performed outside external tolerance margins deriving from the success criteria established. It must be stressed that the concept of human error in the PSA context explicitly shows the human being as a possible direct cause of failure in a certain action, but this does not imply that the root cause of that failure is also necessarily of human origin. This is why the concept of human error within the PSA framework is not univocally associated with the concept of human being culpability. A detailed analysis of these so called human errors is what determines whether the root cause can be attributed to the nuclear power plant staff or to other factors, e.g., the design of the plant and control room or local panels, the operating procedures, the work share-out between members of the Operating Shift, etc.

Human reliability analyses to be performed in a certain PSA depend on its scope. In this context, human reliability analysis for level 1, level 2, for full power or shutdown operating modes, for fires, floods, etc. may be spoken of. In each case there are features differentiating the scenarios which force particular considerations to be adopted in the analysis methodology. The situation as regards human reliability methodology adequacy and homogeneity becomes more uncertain when human reliability analyses are applied that have been undertaken for other PSA scopes. The reason for preparing this document issues from the latter consideration.

The aim of this document is to formalize or systematize human reliability analyses as included in probabilistic fire analyses. In this document, a reference framework is being set forth from which the human reliability analyses for fire events are developed in a structured and systematic way. This is therefore an effort to rationalize and formalize an activity with which it is intended to define and adopt terminology and establish a useful and valid analysis.
scheme. The scope of this document does not cover the development of an actual "analysis methodology" itself, since this concept involves not only establishing that reference framework but the specific proposal for valid analysis methods. The topic of human error dependencies has not been addressed for the moment.

A fire in a nuclear power station may cause some initiating event, such as reactor and turbine trip, loss of offsite power, etc., an additional damages to safety related equipment. In keeping with this feature, considerations as given in this document have been structured into two main parts.

The first part (A) tackles human actions that should be undertaken to mitigate a fire. That is to say, those actions related to fire fighting tasks which would involve both Operating Shift staff and, very particularly, members of the Fire Brigade.

The second part (B) discusses those human actions which should be undertaken in response to the initiating event which a fire itself might possibly give rise to. Such actions should be taken by the Operating Shift staff and display differences to those as modelled in the internal event PSA. These differences in fact derive from damage caused by the fire which, through reducing the availability of mitigation systems, will complicate the Shift’s work in managing the initiating event. Human actions which, either on their own or in combination with the fire, could give rise to an initiating event are also discussed.

In the first part (A) on "Human actions in response to the fire event", the first four sections include terminology, definitions and considerations on each of the four general phases into which the plant staff’s response to the fire can be subdivided. That is to say, on the fire detection, the fire diagnosis, the selection of the fire fighting strategy and the fire suppression itself. The fifth section considers the time range of the four phases mentioned and, finally, the sixth section discusses the influence the fire may have on successfully performing the previous phases.

The first two sections of the second part (B) regarding "Human actions generating and responding to the initiating event" include terminology, definitions and considerations on various types of human actions modelled in the internal event PSA that should be reviewed when addressing fire events. The third section makes considerations on the influence the fire may have on such actions.

**A) HUMAN ACTIONS IN RESPONSE TO THE FIRE EVENT**

This section addresses human actions which should be undertaken to extinguish the fire. That is to say, those fire extinguishing task related actions which would involve both the Operating Shift staff and, most particularly, members of the Fire Brigade.

Successful fire extinguishing calls for the four general plant staff response phases (detection, diagnosis, strategy selection and suppression) to be successfully carried out. In addition to the correct performance of each phase, they must all be performed fast enough so that the fire is suppressed before damage occurs to certain equipment, i.e., before a certain propagation time estimated by fire simulation codes has elapsed. In short, it must be considered that failure in fire extinguishing would always occur provided that either detection or diagnosis or extinguishing strategy selection or fire suppression were to fail, or provided

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that the time taken in performing the four phases were too long, i.e., longer than the time available.

It could be assumed that sooner or later detection, diagnosis, strategy selection and even fire extinguishing would always be performed since, in the last resort, the fire would extinguish itself when combustible matter were exhausted, whereby the probability of failure for each of the four phases could be nil. This is where the time factor is important, invalidating the previous argument since, when failure in detection or diagnosis, etc. is mentioned, it is being implicitly assumed that it is a matter of estimating the probability of each of these four phases failing before a certain amount of time has elapsed. That is to say, for example, probability of failure in detecting a fire before 20 minutes have elapsed since it began.

In view of the fact that the fire progression in time is divided into several growth stages (associated with progressive equipment damage) in a probabilistic fire analysis, it will be necessary to undertake an analysis of each of the phases of human response to the fire as well as of their performance in the time available, for each of the time intervals deriving from the growth stages laid down.

Moreover, it must be pointed out that human response in detection and hand extinguishing phases may be combined with automatic actions, if there were any, from the Fire Protection Systems (FPS). This fact, which must, of course, be taken into account in the analysis, is not exhaustively described in this document. Nevertheless, some considerations are given inasmuch as automatic actions may determine or influence human response.

1. FIRE DETECTION

Detection represents the plant staff ability to perceive or become aware of an anomaly, i.e., the ability to observe a symptom indicating that something is outside the normal operating conditions. This concept does not therefore include diagnosis or identification of the reasons for the anomaly, which forms part of what has been called the second fire response phase.

There are generally multiple mechanisms which would enable the plant staff to identify a fire related anomaly. Naturally, the applicability of each of them to a specific plant will depend on the particular characteristics of its design and its operating features. Mostly, there are different combinations of detection mechanisms, depending on fire areas. The most usual detection mechanisms are classed and described in the following paragraphs.

1.1 Direct detection

This is defined as those detection mechanisms which enable a plant's staff to become aware, at practically the same time as the symptoms indicating an anomaly are detected, that the most probable reason for such is a fire. That is to say, the symptoms detected are clearly representative of a fire. The subsequent diagnosis phase is simplified with these direct detection mechanisms.
1.1.1 Automatic

Direct automatic detection consists in anomaly symptoms being perceived by the Operating Crew from signals on Control Room panels:

a) Fire Detection Subsystem alarms, belonging to the FPS, such as fire detectors (smoke, thermal, etc.) in a given fire zone.

b) Fire Suppression Subsystem alarms or indications, belonging to the FPS, such as automatic fire water pump start, automatic discharge of CO₂ or Halon, etc.

This is called automatic detection inasmuch as signs automatically generated by plant systems intervene in part of the fire detection process.

Nevertheless, successful detection from the hand extinguishing point of view requires the chain of automatic FPS actions serving for the crew to detect the anomaly (fire detectors, alarms, pump status lights, etc.) to work correctly and the crew to become aware of or perceive such signals. It is easy to foresee that situations may arise wherein, if an automatic fire suppression system does not successfully fulfil its mission due to a failure in the associated fire detection, the automatic detection on the part of the operating crew may also fail through not availing of signs warning of the anomaly. That is to say, total dependency may arise between the automatic fire suppression system actuation and the operating crew response, as both depend on the automatic fire detection equipment.

Moreover, the detection time will depend on whether the plant is stable for a certain time without an initiating event being caused by the fire or whether almost simultaneously to the fire starting a reactor trip is induced. In the first case, detection time would be slightly longer than the time elapsing from the fire starting to the automatic signal being given (when sufficient concentration of smoke in the area is reached or sufficient temperature or the FP pump starts up, etc.). In the second case, detection time for an anomaly would be the same as the time for the initiating event arising.

1.1.2 Human detection

Direct human detection consists in plant staff located in areas close to the fire origin detecting signs of fire (smoke, smell of burning, etc.). It is called human detection inasmuch as the anomaly detection process entirely involves people, with no direct intervention of automatic signals generated by plant systems.

1.1.2.1 Through human presence

Direct human detection through human presence means observation of fire effects or symptoms by plant staff working in areas close to the fire location. There is a certain probability for such areas being occupied by plant staff when the fire occurs.
1.1.2.1.1 Through human presence in the fire’s area of origin.

In this case, the plant staff would be in the room where the fire starts. Although fine differences must be made as a function of the size of the fire area, of the fire origin (cabinet, cables, etc.) and of the fire growth stage, detection time is usually very short by this way.

1.1.2.1.2 Through presence in areas to where the fire effects spread.

In this case, the fire would be detected by plant staff who would not be in the area where the fire breaks out but would perceive some sign of it, such as smoke, noise, etc. The most significant difference compared to the foregoing case is that detection time will be longer. For example, to give credit to smoke detection, it would be necessary to estimate the time elapsing until the smoke spreading from the fire origin reached minimum levels of concentration which a human being could detect in a nearby area.

1.1.2.2 Through patrolling

Direct human detection through patrolling involves plant staff as building attendants, supervisors, etc. perceiving signs of fire when accessing the fire’s area of origin or areas where its effects have spread, as a result of performing their surveillance tasks as laid down in procedures.

1.1.2.2.1 Through patrolling the fire’s area of origin.

In this case, the actual room where the fire broke out would be included in the patrol programme. Although fine differences could be made as a function of the size of the fire area, of the fire origin (cabinet, cables, etc.) and of the fire growth stage, the assumption of the average detection time usually being slightly longer than half the time between patrols could be made.

1.1.2.2.2 Through patrolling areas where the fire’s effects spread.

In this case, the fire would be detected by patrol members who would not be in the fire’s area of origin but who would perceive some effect such as smoke, noise, etc. when passing through propagation areas. Although, like in the foregoing cases, fine differences could be made with regard to the size of areas, the fire origin and growth stage, this detection mechanism would have an associated lag time equal to the time elapsing until smoke spreading from the fire origin reached the minimum levels of concentration in propagation areas that a human being could detect plus half the average time between patrols.

1.2 Indirect automatic detection

This is defined as those detection mechanisms such that the Control Room Crew would observe signs indicating an anomaly in the plant which would not lead them to directly think they were being caused by a fire. These indirect detection mechanisms involve a
significant gap between detection and diagnosis phases due to the anomaly detected not necessarily having to be representative of a fire situation. An immediate or obvious relation between the anomaly detected and a fire cannot be established. Consequently, the subsequent diagnosis phase will be more complex.

1.2.1 Through generation of an initiating event

This indirect detection mechanism involves Control Room Crew detecting an anomaly when an initiating event caused by a fire occurs (e.g., a reactor trip). The anomaly detection time would be the same as the time elapsing from when the fire started to the initiating event occurring.

1.2.2 Through damage to equipment

This indirect detection mechanism involves Control Room Crew detecting an anomaly when the fire begins to damage plant equipment even though an initiating event had still not been generated. The anomaly detection time would be equal to the time from when the fire started to the first equipment failure alarms occurring. Successful detection requires both correct functioning of automatic damaged equipment signals (alarms, status lights, etc.) and the crew to become aware of such signals.

2. DIAGNOSIS

Diagnosis is the second general phase into which the plant staff response to a fire can be divided. The diagnosis phase can be divided in turn into 2 subphases. The first represents the diagnosis of the fire itself and the diagnosis confirmation, and the second the identification of the area affected or the fire front. Naturally, as was indicated when discussing detection mechanisms, there may exist a remarkable dependency between this phase and the previous one, particularly between the use of direct detection mechanisms and the fire diagnosis subphase.

2.1 Diagnosing the existence of a fire.

This represents the plant staff's ability to relate or associate anomalies they have detected during the previous phase with a fire before a certain time has elapsed.

If detection is direct automatic, a final diagnosis (that there is a fire) will be made directly from the Control Room or confirmation of the diagnosis will be requested by sending a person to the area where the fire might foreseeably be located, depending on the accuracy with which alarms activated or Control Room signals enable the event to be diagnosed and the plant operating practices. The diagnosis will take a certain time which shall be increased if making confirmation is also a normal practice.

If detection is direct human (through human presence or through patrolling), the diagnosis subphase is generally contained in the detection phase itself.
If detection is indirect, the diagnosis process will generally be much more complex than in the other cases since an analysis must be made of the mental process leading the control room crew to deduce there is a fire from the signs detected, which, initially, are not directly relatable to a fire. In addition, it is clear that, apart from a preliminary diagnosis which might be made from the Control Room, a local confirmation of the diagnosis will always be necessary in these cases. This will imply that diagnosis confirmation time will have to be added to diagnosis time. In cases of automatic detection, whether direct or indirect, the first signs of anomaly due to a fire could appear in the Control Room in a short time interval, together with signals indicating an initiating event caused by the fire. It would seem logical to assume that in these cases, the control room crew would firstly manage the initiating event for stabilizing the plant, and, after some time, take up the diagnosis subphase of the cause of the anomaly signals (fire) which it had detected in the Control Room. This would be an additional time, generally longer when detection is indirect than direct, to be added into the diagnosis subphase performance time estimate.

2.2 Identification of the fire location

This represents the plant staff ability to identify the area and, more specifically, the fire fronts before a certain time has elapsed. That is to say, in addition to detecting and diagnosing a fire, it will be necessary to identify its location. This identification subphase proves indispensable for selecting the extinguishing means in the following phase.

In some situations accurate identification of the fire fronts may not be necessary, and identifying the area would be enough depending, for instance, on the availability of fixed suppression systems (like spray nozzles) which cover the whole area.

The fire area and specific origin will be located, when possible, by making use of alarms and indications available in the Control Room, but usually local observation will also be done.

Whilst the detection and diagnosis are undertaken mainly by the Control Room Crew (or other plant staff in the case of direct human detection), after the fire location subphase, members of the Fire Brigade would now actively participate in fire fighting, coordinating their activities with Operating Staff members. In fact, for this subphase, they would be expected to receive information from the Operating Staff on the facts of the matter and the foreseeable fire source. As from there, the Brigade would move to the area where the fire is expected to be or to adjacent areas to confirm the fire front and decide on extinguishing means.

For this subphase to be successfully performed from the point of view of Control Room actions (if there were any), it will be necessary for the alarm and indication signals to work correctly (some of which might already have been modelled in the detection phase), as well as for the Operating crew to correctly work out the fire location from them. Naturally, the probability of this deduction being a success or failure will very much depend on the detection mechanism used. In the same way, for this subphase to be successful from the local observation standpoint, the most important thing will be to analyze the feasibility of such observation as a function of the origin, of the fire area and of the fire growth stage. The issue of feasibility in performing the actions as a function of the fire effects (smoke, heat, etc.) begins to have a significant impact, since from this subphase on there may be many local tasks performed by Fire Brigade members.
In analyzing this subphase, it is important to bear in mind the plant operating practices with regard to the staff response to fire situations. Both the written procedures available if used (Fire Protection Manual, Alarm Book, Failure Instructions, etc.) and the training received for facing these situations must be considered here.

In short, the probability of error in this subphase will be determined by the manner of identifying the fire location.

Likewise, this will all involve a certain time used in performing this phase for each fire origin included in the analysis.

Finally, it is stressed that the importance of analyzing this subphase and its consequences for the fire fighting strategy selection phase can be well appreciated in certain scenarios. Thus, in fires producing and spreading abundant smoke, multiple concurrent smoke detector activation may occur which may hinder accurate fire origin location. If the fire front is not accurately identified, suppression agents may well be used in areas where they are not really needed, with potential damage from sprinkling or flooding equipment which, initially, would not be directly threatened by the fire.

3. SELECTING THE FIRE FIGHTING STRATEGY

It represents the ability of the plant staff in charge of fire fighting to decide on the most suitable extinguishing means and manner before a certain time has elapsed.

In general, depending on the fire areas, nuclear power plants avail of automatic and manual fire suppression systems. The scope of this document does not include any discussion on the automatic operation of fire suppression systems but only on hand operated means. Therefore, though logically the impact of automatic fire suppression system operation should be taken into account in the human reliability analysis, this section intends only to discuss how the Operating Shift or Fire Brigade staff would decide on what extinguishing means to hand operate for putting the fire out and how to perform such extinguishing operations.

Amongst hand operated extinguishing equipment, we have automatic systems which, either through the fire detection being unavailable or failed, may not operate automatically but could operate on manual demand. There are also typical hand operated systems, such as water hosepipes and portable extinguishers.

The manner in which fire fighting would be performed with the means available will be influenced by several factors such as type of suppression means in the fire area and surroundings, access routes to the area, availability of protective clothes, breathing equipment, fire growth stage, etc.

In any event, an analysis will be necessary since, depending on the system strategy chosen to suppress the fire, it will be deduced whether the means available are the most suitable, whether all or only some of the fire area equipment would be damaged during extinguishing (for instance, operation of permanent systems covering a whole area when the fire source is located only in one part) and the influence of the fire effects on Fire Brigade members, etc.
In short, the selection of a strategy will be influenced by knowing the type of fuel existing in the area, the extinguishing systems available, equipment in the area not damaged by the fire but which could be damaged by the extinguishing systems, etc. Another important factor in estimating success in choosing adequate fire fighting strategies is whether such strategies are considered in procedures and drills at the plant or not.

Therefore, it is important to analyze and estimate the probability of plant staff choosing the right fire fighting strategy. This process will be influenced by the detection and diagnosis mechanisms used in the two previous phases.

Finally, the average time taken in performing this phase must be estimated for each fire scenario.

4. FIRE SUPPRESSION EXECUTION

Fire suppression execution represents the ability of the plant staff in charge of such mission to extinguish or at least control a fire (stop it spreading) before a certain time has elapsed. It may fail through either failure of the extinguishing equipment or staff errors in their actions, such as wrongly pressing start-up push buttons, impossibility to access components which must be operated, etc.

Finally, the average time necessary for performing this phase must be estimated. Naturally, this average time would be different depending on the development stage of the fire being analyzed (degree of growth reached), on fuel existing, etc.

5. PERFORMING IN THE TIME AVAILABLE

Successfully extinguishing a fire will call for the four general phases discussed in foregoing sections to be successfully fulfilled. This is a necessary but not sufficient condition for ensuring success. The reason lies in the fact that, apart from correctly performing each one, they must all have been overall performed sufficiently swiftly so that the fire is controlled before the available time estimated by fire propagation calculations has elapsed. In short, a correct but too slow response also implies failure in extinguishing a fire.

Consequently, this factor is being used in an attempt to measure the probability of failure in extinguishing a fire on the basis of time considerations only. This probability of failure would basically be conditioned by the quotient between the time available for extinguishing the fire (understood as being the time elapsing from the fire starting until reaching a certain growth or damage stage) and the sum of the average times used in performing each of the four phases described above.

6. FIRE INFLUENCE ON PLANT STAFF FIRE FIGHTING ACTIONS

Up to here, human actions a plant's staff would perform in response to a fire have been discussed and broken down into the phases and jobs comprising them, and errors which would lead to such response failing have been pointed out. Obviously, these human actions will be influenced by performance shaping factors which have been traditionally considered in human reliability studies in PSAs, i.e., through the quality of experience/training, the quality
of the man-machine interfaces, degree of stress, etc. But, in addition, and this is a very significant differentiating factor between human reliability analyses for internal initiating events and hazards, the probability of these human actions being successful will be determined by performance shaping factors or very peculiar considerations of fire scenarios.

In view of their potential impact on risk, these factors or considerations must be analyzed. The following sub-sections describe the three deemed most important.

6.1 Influence of equipment losses

Equipment and components the plant staff would need to adequately respond to a fire itself during some of the four response phases may be left unavailable by the fire or by actions of plant staff in fire fighting.

These are items required by the Operating Staff, members of the Fire Brigade, or both in performing their jobs.

Examples of equipment and components that might be necessary, and which a fire or plant staff might affect either directly or through their associated cables or support systems would be: alarms, instrumentation channels, status lights, controls, lighting system, communications systems, area access control system, ventilation systems, electric supply to equipment like, for instance, Control Room cabinets, FP doors, etc. In differentiating between the four plant staff fire response phases, some of the main problems arising in each one as a result of these damages would be:

**Detection:** Except for the case of direct human detection in which this issue would have no impact, at least losses of alarms, instrumentation channels and component status lights or any other components credited with being a detection mechanism should be taken into account.

**Diagnosis:** Losses of alarms, instrumentation channels and component status lights should be taken into account. If diagnosis confirmation were necessary and in order to identify the fire fronts locally, possible losses of lighting, communications and the access control system should also be taken into account.

**Selecting the fire fighting strategy:** Although the fire would not be envisaged as affecting the FPSs, because of the actual design of the Fire Protection Systems itself, nevertheless, this possibility should not be directly excluded without further analysis. Situations of this kind could lead plant staff to choose fire fighting means which were in fact unavailable. Likewise, possible difficulties in accessing certain areas could lead to wrong fire fighting strategies being chosen.

**Fire suppression execution:** If fire suppression is performed locally (not remote), at least losses of lighting, communications and access control systems which would hinder such task should be considered. The fact that Fire Brigade staff are forced to keep FP doors open during their fire fighting actions may also affect fire extinguishing (and, consequently, the fire spreading). These doors would then effectively become barriers unable to stop smoke spreading and, depending on the cases involved, flames also.
6.2 Influence of the fire effects on people

Fire effects (smoke, rise in temperature, etc.) may have a significant effect on the Operating Staff and Fire Brigade members in charge of fire mitigation. This effect would be considered both from the physical standpoint (breathing problems, reduction of visibility, heat, etc.) and psychological (stress, fear, etc.).

This influence will mainly and significantly affect the diagnosis and fire suppression execution phases when performed locally. Nevertheless, if the fire or its effects were to reach the Control Room, it will also affect the rest of the plant staff response phases.

Thus, for example, problems like smoke spreading through unsuitably insulated ventilation ducts, water stored in areas with live electric equipment, flames and hot surfaces, etc. must be taken into account.
Including these fire effects into the analysis would improve its completeness and quality in explicitly addressing important aspects influencing the risk. This would all favour discussion on the feasibility of actions as postulated and would allow greater accuracy in estimating their probabilities of error.

6.3 Influence of tasks allocation among Operating Staff members to respond to the fire event.

A fire combined with an initiating event (caused by the former) becomes a very heavy work load for the Operating Shift members who must stabilize the plant and simultaneously take some of the actions intended to mitigate the fire. Thus it is usual in plants for the Fire Brigade Chief to be a member of the Control Room Crew, and for supervisors or building attendants to also be members of the Fire Brigade, which represents tasks and responsibilities of a different type concurring in the same people in a certain time interval.

Tasks assigned to Operating Shift members may be known as a function of the management policy as defined in the plant for fire scenarios and, consequently, these aspects could be more realistically considered.

B. HUMAN ACTIONS REGARDING GENERATION OF AND RESPONSE TO A FIRE INDUCED INITIATING EVENT

This second part of the document discusses those human actions which, combined with a fire, would contribute to the generation of an initiating event and those which should be performed in response to such initiating events. These actions should be performed by Operating Shift staff and will display differences with respect to those modelled in the probabilistic internal event PSA. Such differences mainly derive from damage caused by the fire which, by reducing the availability of the safety systems, will complicate the Operating Shift tasks for managing the initiating event.
The first two sections include brief considerations on different types of human actions (speaking in terms of failure, they would be human errors) modelled on PSAs. Specifically, the first section is devoted to type 2 human errors which are those that, either on their own or combined with other failures give rise to an initiating event (for example, a reactor trip). The second section is devoted to types 3, 4 and 5 human errors made during the management of accident sequences after the initiating event has occurred.

Finally, the third section discusses the influence the fire may have on successfully performing the different human actions related to the generation of the initiating event or to the response thereto.

In view of the fact that, in a fire risk analysis, fire progression in time is divided into several stages of growth, an analysis of each of the human actions responding to the initiating event will have to be made for each of the time intervals deriving from the growth stages established.

1. **TYPE 2 HUMAN ACTIONS**

On its own, a fire may cause failures in a nuclear power plant leading to an initiating event. However, depending on the specific design of each plant, a combination of human action and failures caused by the fire is required for an initiating event to occur in certain scenarios. The latter case is addressed here.

What are known as type 2 human errors may be subdivided into three categories as described hereafter, basically depending on such errors being prior or subsequent to the failures caused by the fire and, within the latter, depending on their directly generating the internal initiating event or their being generated during the performance of an action attempting to prevent it.

1.1 **Type 2A human actions**

Type 2A human errors could also be called latent compound human errors. Compound inasmuch as the generation of the initiating event requires an additional failure generated by the fire apart from the human error itself. Latent inasmuch as they are human errors made in routine tasks, for example, testing, maintenance, calibration, etc. prior to the fire, but in which some component is left unavailable and remains unnoticed during the plant operation and continues thus when the routine task has ended. Nevertheless, those human actions involving testing, maintenance, calibration, etc. which, through being performed at the same time when the fire starts, would give rise to the initiating event would also fit into this category.

The importance of these human actions or errors basically lies in their contribution to increase the frequency of fire caused internal initiating events and, therefore, identifying them.

*NOTE: The nomenclature used for classifying human errors in Types 2, 3, 4 and 5 were taken from the EPRI NP-3583: "Systematic Human Action Reliability Procedure (SHARPI)" but not the subdivision of Type 2 errors which is original to this document.
would be of interest. However, such actions would not be influenced by the fire, since they are prior to it by definition.

1.2 Type 2B human actions

Type 2B human actions could also be called pure human errors inasmuch as they generate an initiating event on their own. Naturally, performing an action of this Type is determined by the existence of a fire.

A clear example of this type of action would be the change from mode 1 (power operation) to another operating mode which the control room crew would carry out when seeing that a fire has made unavailable some piece of equipment covered by a Technical Operating Specification (TS) expressly requiring that change of mode in view of such unavailable state.

In short, a fire may not directly lead to an initiating event occurring but, if, for example, it has damaged systems to which a TS requiring a change of plant operation mode has to be applied, this would cause the Operating Staff to initiate that plant transient.

Identifying this kind of error (in fact they are voluntary acts) is deemed important not only inasmuch as they would contribute towards increasing the frequency of fire caused initiating events but because studying these scenarios would serve for proposing and analyzing the convenience of voluntarily provoking these transients as a function of the systems damaged by the fire and its conditions.

Although Type 2B actions caused by the "Action" of a TS are highly representative, they are not the only ones in this category. Thus, for example, manual trips advised by Failure Operating Instructions attempting to respond to failures which, in this case, would have been caused by the fire, would also fit into this Type.

Finally, it is stressed that these actions could be influenced by the fire itself, since they are subsequent to it.

1.3 Type 2C human actions

Type 2C human errors could also be called compound human errors of response. Compound for the same reason as given for the Type 2As. Of response inasmuch as they represent human actions taken to prevent an anomaly or malfunction progressing that would lead to an initiating event.

A typical example of this type of 2C error for a PWR reactor would be the human action of closing the isolation valve of a pressurizer relief valve (PORV) before the reactor trips after such valve has been opened by the fire.

The probability of success in this type of actions could be clearly influenced by the fire effects so they must be identified and analyzed in this context.
2. TYPES 3, 4 AND 5 HUMAN ACTIONS.

In short, it can be said that Type 3 errors relate to human errors in following Operating Procedures, due to failures in detecting the need for action, to failures in implementing same and to failures through ending the action outside the available time limit.

In the same way, Type 4 errors also relate to human errors in following Operating Procedures, but due to failures in diagnosis and to failures in selecting the action strategy.

Finally, Type 5 errors represent human errors in mitigating an initiating event in situations with no procedures.

These three types of human error have in common that they are made in response to an initiating event. This fact sets a difference with respect to Type 2 errors expounded in the foregoing section. Whilst Type 2 human errors have to be identified again in the context of the fire analysis since they is an additional cause for failures which is the fire, Types 3, 4 and 5 human errors may still be the same as those already identified within the framework of the human reliability analysis for the PSA of internal events. This statement is valid provided the fire cannot change the conditions of the accident sequences, which would call for changes to the event trees. A need for such a change could be caused, for example, when the fire affects equipment whose failing in the course of the accident had not been taken into account before through deeming such as a low probability. In these situations, an analysis of further actions would have to be considered.

In general, however, the fire influence on actions in response to an initiating event will basically turn into a modification of its error probability which is reason enough for justifying a re-analysis in this context.

3. INFLUENCE OF FIRES ON OPERATING SHIFT ACTIONS.

The two foregoing sections have briefly described the most representative characteristics of human actions which, in a fire scenario, may generate an initiating event, as well as those which, once the Initiating event has occurred, should be performed for mitigating it. The probability of all these human actions being successful, with the exception of Type 2A, may be significantly determined by performance shaping factors or highly peculiar considerations of fire scenarios. In view of their foreseeable impact on the risk, factors or considerations which may affect human actions must be analyzed. The following sub-sections describe the three deemed most important.

3.1 Influence of the loss of equipment used by the Operating Shift

A fire or the means used in its suppression or the strategies chosen to fight it may leave equipment and components unavailable which the Operating Staff would require to adequately perform their tasks intended to bring the plant to a safe shutdown state.
It seems obvious that during a fire, the first and main cause of equipment loss is, in fact, the direct effect of the fire (flames, heat, smoke). However, there are at least two reasons worth analyzing in each scenario as potential causes of equipment loss. One of them is that the fire suppression agents used for fire fighting (both successfully or not, as a result in the latter case of an error in some fire response phase, i.e., in the phases of diagnosis, execution, etc.) affect equipment installed in the area. Thus, for example, hand (or automatic) operation of water spray systems or water hosepipes may impact equipment which the fire would not initially affect in a certain growth stage. The third cause are strategies which the Operating Shift might adopt in the light of the fire foreseeable progression. For example, tripping equipment or realigning trains for ensuring the plant's more stable operation in foreseeing the fire possibly affecting equipment of a certain train.

Examples of equipment and components that might be necessary and could be affected by the fire either directly or through their associated cables or their support systems, would be alarms, instrumentation channels, status lights, controls, lighting system, communication systems, area access control system, ventilation systems, electric supplies to equipment like, for example, Control Room cabinets, etc.

With respect to cognitive tasks, the effect the fire might have on instrumentation used by the Operating Shift staff for detecting and diagnosing the plant states following an initiating event is highlighted because of its particular importance. In PSAs, it is usual to analyze those instrumentation channels generating automatic equipment actuation signals, and model them only as an integral part of such automatic actuation signals. On the other hand, it is generally assumed that the instrumentation used by the Control Room Crew to manage accident situations is sufficiently numerous, redundant and diverse for its failures not to reduce human action reliability. This assumption, however, is clearly not generally acceptable in fire situations which may cause multiple failures of instrumentation channels or information signals which the control room crew would require to adequately manage the accident.

In relation to local tasks, losses of lighting system equipment, area access system equipment, etc. would be particularly relevant.

In addition, there is a particular circumstance or consideration worth discussing. In general, in a fire analysis, equipment which would be actually damaged by the fire would tend to be identified and the analysis is made by assuming that such equipment is unavailable and that the remainder has a random failure probability. Here more than ever what would happen in a real fire situation has to be broached. The Operating Staff could find the fire location fairly accurately, could have more or less information on the way it is growing and expectations of spreading and could venture hypotheses on what equipment might be damaged by the fire and by the suppression systems used. This would all depend on their training and knowledge of the plant, of the availability of suitable information in the Control Room (like, for instance, lists of equipment which would be affected by the fire in each fire area or specific fire procedures), etc. But, in any event, the Operating Staff cannot accurately predict what equipment will be unavailable nor at what time. This fact implies that uncertainty will be created in the Control Room crew with regard to the means available for mitigating the accident. Such uncertainty will naturally be greater the greater the amount of equipment which might be damaged in the fire area and the lesser the knowledge or information held on the fire area in question. It is here, therefore, where a turning point in the usual trend of fire analyses appears, consisting in the need to identify not only the equipment
which is going to be actually lost but also the equipment the Operating Staff would believe were going to be lost.

Such a need would seem obvious and may be explained with two situations:

The first situation arises from the fact that before taking any action, Emergency Operating Procedures require the Control Room crew to confirm whether a certain system is available or not (a very simplistic example would be that before depressurising the reactor, it is necessary to check whether a low pressure injection system is available or not). The hypothesis which the Operating Shift handles on the current or future availability of mitigation systems will influence the mitigation strategy finally adopted (perhaps depressurization would not be done, at least in the short term, if it is not certain that water can be injected with the low pressure system). It would therefore seem that an Operating Crew action strategy for accident mitigation during a fire could vary not only as a function of the systems which are known to be unavailable, but also as a function of those systems the operating Crew deems could be unavailable, at least in the near future.

A second representative situation would be that of a fire in an area which might affect a large part of Control Room instruments. For example, a fire in the Cable Spreading Room, amongst other things, could make many controls of equipment necessary for accident mitigation unavailable. In a situation like this, the Operating Crew would raise the question of whether it will be able to control the incident from the Control Room or not. A decision it could take, with a certain probability, depending on the size of the fire and the plant general condition would be to abandon the Control Room and manage the accident from the Remote Shutdown Panel.

As a consequence, it proves necessary to analyze the influence of equipment or component losses (the actual and the foreseeable by the crew) in Operating Crew actions related to the mitigation of the initiating event.

3.2 Influence of the fire on the Operating Staff

As was already mentioned when addressing fire effects on people who are to suppressed a fire, the situation is similar when the action of Operating Staff members in charge of managing the initiating event is analyzed. That is to say, the fire effects may have an influence on their actions both from the physical point of view and the psychological standpoint. The only additional consideration in this point is that the exposure of the Control Room Crew to these effects (smoke, high temperatures, etc.) will initially be lower, if not nil, except in two particular circumstances.

The first concerns fire scenarios originating in the Control Room or in areas from which its effects can spread. Fire effects, such as smoke, can reduce the reliability of human actions or even prevent them. Thus, for example, in case of a Control Room cabinet fire, it would be necessary to assess the feasibility of the Control Room crew being able to continue operating the plant from the Control Room. It is questionable whether the smoke generated by the cabinet fire, even though the fire were not to spread outside it, could be extracted by the ventilation system (it would naturally depend on the specific design of each Control Room). In this situation, perhaps the conclusion might be reached that the only alternative for operating the plant would be to do so from the Remote Shut-down Panel.
The second circumstance relates to the fact that performing some human actions related with the mitigation of initiating events requires local actions (actions performed outside the Control Room) which could force the Operating Staff in charge of such work to access or move around areas directly affected by the fire.

In summary, in both cases problems like smoke and its spreading, for example, through unsuitably insulated ventilation ducts, water stored in areas with live electric equipment, flames and hot surfaces, etc. must be taken into account.

To conclude this sub-section, it is deemed that including these fire effects into the analysis would improve its completeness and quality in explicitly addressing important aspects influencing the risk. This would all favour discussion on the feasibility of actions as postulated and would allow greater accuracy in estimating their probabilities of error.

### 3.3 Influence of tasks allocation among Operating Staff for responding to the initiating event.

In analyzing human reliability, it is normal practice for internal initiating events to assume the presence of the four members of the Control Room Crew (Shift Chief, Shift Chief’s Assistant, Reactor Operator and Turbine Operator) as well as the availability of all supervisors or building attendants also forming part of the Operating Shift Staff. In view of the fact, as was mentioned in an earlier section, that some of these people have to participate on specific missions during fire fighting, it is obvious that the starting assumption on staff availability should be reconsidered, and their risk impact for a fire event analyzed.