Gesellschaft für Anlagen-
und Reaktorsicherheit
(GRS) mbH

SMiRT21
12th International Seminar on FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS

München, Germany
September 13-15, 2011

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Kurzfassung

Im Rahmen des vom Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU) beauftragten Vorhabens 3610R01375 wurde im September 2011 das mittlerweile zwölfte internationale Seminar "Fire Safety in Nuclear Power Plants and Installations" als Pre-Conference Seminar der 21st International Conference on Structural Mechanics In Reactor Technology (SMiRT 21) bei der TÜV SÜD Industrie Service GmbH in München veranstaltet.

Die vorliegenden Proceedings des Seminars enthalten alle sechsundzwanzig Fachbeiträge des zweitägigen Seminars mit insgesamt sechzig Teilnehmern aus vierzehn Ländern.

Abstract

In the frame of the project 3610R01375 funded by the German Ministry for the Environment, Nature Conservation an Reactor Safety (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, BMU) the meanwhile twelfth international seminar on "Fire Safety in Nuclear Power Plants and Installations" has been conducted as Pre-Conference Seminar of the 21st International Conference on Structural Mechanics In Reactor Technology (SMiRT 21) at TÜV SÜD Industrie Service GmbH in Munich, Germany in September 2011.

The following seminar proceedings contain the entire twenty-six technical contributions to the two day seminar with in total sixty participants from fourteen countries worldwide.

Contents

1 In Memoriam Prof. Dr. Ulrich Schneider

It is a very sad duty to announce to all participants that one of the original founders and permanent organizers of the SMiRT Conference seminars on “Fire Safety in Nuclear Power Plants and Installations”, Prof. Dr. Ulrich Schneider has passed away in consequence of a severe medical condition on October 23, 2011.

Ulrich Schneider was born on July 17th, 1942 in Köslin. After having finished school and a practical apprenticeship he started studying engineering at Braunschweig University of Technology (Germany) passing the final exam (diploma) receiving an award in 1971. Schneider got a PhD (Dr. techn.) for civil engineering in 1973 and passed the postdoctoral lecture qualification in 1979. In 1974 a special research field on “Fire Behavior of Building Elements” was founded at Braunschweig University of Technology, a project being also chaired by him. In 1981, Ulrich Schneider was appointed to a professorship for structural engineering at University of Kassel (Germany). In 1990 he left Kassel to take a professorship as ordinary professor for building material and construction at the Institute for building materials, building physics, and fire protection at Technical University (TU) Vienna (Austria). Within a period of 20 years Schneider built up an excellent study programme with various students receiving diploma, master and PhD degrees. Since 2002 he also gave lectures as professor at the state owned Open University (MGOU) in Moscow (Russia) and at the Argo-Technical University in Ksyl-Orda (Kazakhstan).

Ulrich Schneider was member of a variety of national as well as international boards and standards committees, such as the German DIN, the Austrian ÖNV, the German Nuclear Safety Standards Commission (KTA) and the Reactor Safety Committee

(RSK), and technical scientific associations, e.g. CIB, International Union of Testing and Research (RILEM), Vereinigung im Brandschutz (VIB).

In 1987, the first SMiRT Post-conference Seminar on “Fire Safety in Nuclear Power Plants and Installations” took place in Anaheim, CA (USA) under the leadership of Ulrich Schneider, Heinz Liemersdorf (GRS), and Klaus Müller (Research Center of Karlsruhe). This seminar was a great success with respect to fire safety in nuclear facilities, therefore the three organizers succeeded in having such a seminar every two years either as post- or pre-conference seminar in conjunction with SMiRT up to the time being. Ulrich Schneider enjoyed being scientific organizer of this seminar and technical chair and served this position until early summer 2011 with dedication putting more and more emphasis on engineering methods for assessing fire safety in nuclear plants.

Well knowing that he could not participate in the 12th seminar due to his medical condition, Ulrich Schneider was still very interested in our seminar. Hence he submitted a paper together with one of his graduate students giving the presentation. Furthermore, he kindly asked me in one of our last phone calls to send him the proceedings of the highly scientific and technically interesting SMiRT Pre-conference Fire Seminar. It was one of his special wishes to continue the series of seminars providing fruitful contributions nuclear fire safety.

Personally speaking, I am very thankful for the strong support and the fruitful input to the seminar topics I have received by Ulrich Schneider all the years after having joined the permanent scientific organizing team in 1997. The close and excellent technical cooperation, I enjoyed always with great pleasure.

We all will miss him as expert and scientific organizer, but also as a person with an open minded, helpful and modest personality strongly aiming on a good and fruitful cooperation, having lots of ideas and following clearly defined scientific engineering goals. However, we will try to continue with these seminars following the direction Schneider and his co-organizers always had in mind.

Dr. Marina Röwekamp
- Permanent Organizer -

2 Foreword

The meanwhile 12th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' was held as Pre-Conference Seminar of the 21st International Conference on Structural Mechanics In Reactor Technology (SMiRT 21) hosted by TÜV SÜD Industrie Service GmbH in Munich, Germany in September 2011. In total sixty participants from fourteen different countries with nuclear installations from Asia, Europe, North as well as South America followed the twenty-six presentations in the different scientific sessions and a short panel discussion at the end of the seminar.

Most of the participants could also participate in a fire related technical study tour of the Gundremmingen nuclear power plants' site.

While the first presentations gave insights in general aspects of fire safety concepts, recent developments in fire protection systems and international activities with regard to assessing fire risk the general focus of the first seminar day was mainly on actual insights from experimental and analytical fire research activities in the nuclear field. This included also validation and verification of fire simulation tools for nuclear applications.

Essential progress has been made with respect to fire experiments representative for real case fire propagation scenarios within the large experimental program PRISME (*Pr*opagation d'un *i*ncendie pour des *s*cénarios *m*ulti-locaux *é*lémentaires) under the umbrella of OECD Nuclear Energy Agency (NEA. The experiences gained from modeling fires with different types of simulation tools revealed the experience that there is still the need for continuously improving fire models and their verification and validation for those scenarios typical for nuclear installations including importance and sensitivity studies. The results are essential to further increase the levels of confidence of the calculated results. This enables the user to apply fire simulations with an acceptable level of conservatism in the frame of deterministic fire hazard analysis. However, these analytical tools can more and more also be used for probabilistic fire risk analysis (so-called Fire PSA). It turned out that there are still challenges, in particular with respect to modeling the pyrolysis rate or fire suppression in an adequate manner.

The last session of the first seminar day covered regulation, standards and guidelines with respect to fire safety in the design and operation of nuclear installations. In this context, fire protection standards and guidelines have to be adapted to the state-of-the-art in light of actual insights from the operating experience, the intended extension of nuclear plants´ operational lifetime and the licensing of new reactors in n several countries.

The presentations during the second seminar day highlighted approaches for and actual results from fire safety analysis but also provided a variety of insights and lessons learned from the operating experience of the existing plants, which were partly designed corresponding to much earlier regulatory requirements. In particular, the significance of high energy arcing fault (HEAF) fire events was demonstrated. This will result in an experimental program by OECD NEA in the near future for investigating this failure mechanism in more detail. Furthermore, the value of the international fire events database OECD FIRE for analyzing the fire events specific operating experience of nuclear power plants was highlighted indicating that consistent experience feedback from the nuclear power plant operation has to be extended continuously to provide plant independent but also plant specific data for analytical issues.

In addition, special attention was paid to some more actual results from safety assessment and the respective enhancements following the nuclear accidents at the Japanese nuclear power station Fukushima Daiichi in March 2011 as a result of earthquake and Tsunami. Although combinations of fire and external events are typically included in national standards and analyses have shown that the plants are principally designed against such combinations, there is still the potential for further optimizing the plant safety in this respect.

The focus of the brief final panel discussion with members from regulatory bodies as well as experts from technical safety organizations (TSO) and nuclear power plant operators was also on fire related lessons learned from and actions taken after the Fukushima accident. Fires, even those as occurred as consequential events from external events exceeding the design basis, principally represent hazards against a variety of measures can be and is being taken in a multi-level approach from prevention of the fire up to mitigation of the consequences. This also includes different types of combinations of fires with internal events or with internal or external hazards.

From the first ‘Seminar on ‘Fire Safety in Nuclear Power Plants and Installations’ in 1987, when the significance of fires in nuclear installations had just been recognized after the Browns Ferry incident, up to 2011 fire safety in nuclear power plants and other nuclear installations worldwide has significantly increased. In parallel, the methods and analytical tools for assessing fire safety are being continuously enhanced. However, there are still challenges with respect to fire safety analysis and fire risk assessment. In particular, there is the permanent need for theoretical and experimental research in this field. Even the most recent potential applications of the analytical tools require the validation of data and procedures for specific fire event sequences. In addition, the data base with respect to the experience feedback for fire events in nuclear installations needs further extension and has to cover phenomena observed more recently.

The Seminar was hosted with great hospitality by TÜV SÜD Industrie Service GmbH in Munich, Germany. The organizers are indebted to the invitation and support by the hosts during the two days seminar.

Furthermore, the organizers and the participants are grateful to the management of the Gundremmingen Nuclear Power Plant having inviting all the seminar participants to a technical study tour of the plant mainly focusing mainly on fire protection aspects.

Last but not least I want to thank all participants, speakers, authors and chairpersons for their active and fruitful participation as well as for the many high level contributions during the 12th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' which made this twelfth event within twenty-two year again a highly successful one.

The next, 13th seminar of this series is intended to be held as SMiRT 22 Post-conference Seminar in the United States of America in late summer 2013 .

Dr. Marina Röwekamp
- Scientific Chairperson and Organizer -

3 Seminar Agenda

12th SMIRT Pre-Conference Seminar on
Fire Safety in Nuclear Power Plants and Installations

Hosted by TÜV SÜD, München, Germany; September 13-15, 2011

AGENDA

Tuesday, September 13, 2011

09:30 h	***Introduction by TÜV Hosts***	B. Ernst	TÜV SÜD, Germany
	Welcome	R. Hero	TÜV SÜD, Germany
	General Aspects of Fire Safety Concepts in Nuclear Installations	***Chairperson: Matti Lehto (STUK)***	
09:40 h	Commissioning and Integrated Testing of Fire and Life Safety Systems	C. Kilfoil	Bechtel, USA
10:10 h	The ESReDA Working Group on Fire Safety: Perspectives, Activities and Results	M. Demichela, et al.	Politecnico di Torino, Italy
10:40 h	Fire Protection Concept for Auxiliary Buildings in Nuclear Power Plants – Extinguishing System With Powder Generators	A. Knop	Minimax, Germany
11:05 h	***Coffee Break***		
	Experimental Research and Fire Model Experience and Simulation (continued)	***Chairperson: S. Hostikka (VTT)***	
11:20 h	Overview of the OECD PRISME Project - Main Experimental Results	L. Audouin, et. al.	IRSN, France
11:50 h	Fire Code Benchmark Activities Within the International Research Program PRISME – Discussion on Metrics Used for Validation and on Sensitivity Analysis Study	S. Suard, et. al.	IRSN, France
12:20 h	A Predictive Pyrolysis Model for Liquid Pool Fires Including Radiation Feedback from Hot Soot Layer in COCOSYS	M. Pelzer, W. Klein-Heßling	GRS, Germany
12:50 h	***Lunch Break***		

	Fire Model Experience and Simulation	***Chairperson: L. Audouin (IRSN)***	
13:50 h	Experimental and Numerical Simulations of Liquid Spreading and Fires after Aircraft Impact	T. Sikanen, S. Hostikka, A. Slide	VTT, Finland
14:20 h	Validation and Development of Different Calculation Methods and Software Packages for Fire Safety Assessment in Swedish Nuclear Power Plants	P. van Hees	LTH, Sweden
14:50 h	Modeling of Ignition and Flame Spread in the Initial Phase of A Fire	N. Schjerve, U. Schneider	TU Vienna, Austria
15:20 h	Pyrolysis Modeling of PVC Cable Materials	S. Hostikka, A. Matala	VTT, Finland
15:50 h	***Coffee Break***		
	Regulation, Standards and Guidelines	***Chairperson: S. Kirchner (TÜV)***	
16:05 h	Towards an European Common Approach for Specific Fire Protection Concerns: Containing Fire Scenario Variety by Specific Defense-in-depth Standards for Nuclear Application of Fire Protection Products	E.Maillet, T. Magnusson, A. Niggemeyer, C. Bruynooghe	GdFSUEZ, Belgium Vattenfall, Sweden AREVA, Germany EC JRC
16:35 h	Recent Regulatory Activities on Fire Safety in Finland	M. Lehto, P. Välikangas	STUK, Finland
17:05 h	Enhancements in International Guidelines for Fire PSA	H. P. Berg, et al.	
17:30 h	***Adjourn of the first day***		
19:00 h	***Hosted Dinner*** *in the center of Munich*	All participants enscribed and spouses	

Wednesday, September 14, 2011

	Fire Safety Analysis	***Chairperson: P. van Hees (LTH)***	
09:15 h	Safety Fire Zoning and Fire Vulnerability Analysis of LING-AO Power Station Units 3 & 4	W. Liu, G. Tan	CNPE, China
09:45 h	Fire Dynamic Criteria to Filter out Relevant Compartments for Fire PSA	F. Berchthold, B. Forell	U Magdeburg GRS, Germany
10:15 h	Enhancements in the OECD FIRE Database – Fire Frequencies and Severity of Events	W. Werner, J. S. Hyslop, M. Roewekamp R. Bertrand, A. Huerta	SAC, Germany NRC, USA GRS, Germany IRSN, France OECD/NEA

10:45 h	***Coffee Break***		
	Fire Safety Analysis (contd.)	***Chairperson: H. P. Berg (BfS)***	
11:05 h	Update of Reliability Data for Fire Protection Equipment in German Nuclear Power Plants	B. Forell, S. Einarsson	GRS, Germany
11:35 h	Updating of the Fire PRA of the Olkiluoto Nuclear Power Plant, Units 1 and 2	L. Tunturivuori	TVO, Finland
12:05 h	Determining Relevant Rooms Within Nuclear Power Plant Fire PSA and Their Contribution to Core Damage Frequencies for Optimizing Fire Protection	J. Döring	TÜV SÜD, Germany
12:35 h	***Lunch Break***		
	Operatig Experience and Lessons Learned	***Chairperson: I. Skandera (TÜV SÜD)***	
13:35 h	Recent Advances in High Energy Arcing Faults (HEAF) at Nuclear Power Plants	G. Cherkas	CNSC, Canada
14:05 h	HEAF – Update of the German Operating Experience	S. Katzer, M. Roewekamp	GL, Germany GRS, Germany
14:25 h	First Experiences from International Databases on NPP Fire Brigade Activities	H. P. Berg, N. Fritze	BfS, Germany
14:50 h	Predicting Industrial Fire Brigade Tactical Fire Flow Rates	G. Cherkas	CNSC, Canada
15:20 h	***Coffee Break***		
	Operatig Experience and Lessons Learned (contd.)	***Chairperson: M. Röwekamp (GRS)***	
15:35 h	Fire in the Containment During Pressure Test Causing Great Losses at Ringhals Nuclear Power Plant, Unit 2	T. Magnusson	Vattenfall, Sweden
16:00 h	The Multi^-stage Fire Safety Concept in German Nuclear Power Plants	B. Elsche G. Fischer, S. Kirchner	E.ON, Germany; TÜV SÜD, Germany
16:30 h	Analysis for the Optimization of German Nuclear Power Plants after the Incidents in Fukushima	B. Ernst, M. Beesen	TÜV SÜD, Germany

17:00 h	**Panel Discussion on Fire Related Lessons Learned and Actions after the Fukushima Accident**	***Chairperson: G. Cherkas (CNSC)***
	Panel participants:	G. Cherkas, Canada, S. Kirchner, Germany, M. Lehto, Finland, T. Magnusson, Sweden, M. Roewekamp, Germany
17:30 h	***Seminar Adjourn***	

Thursday, September 15, 2011

07:45 h - 17:30 h	***NPP Study Tour*** **(bus transfer arranged)**	Participants registered

4 List of Participants

21st International Conference on Structural Mechanics in Reactor Technology (SMiRT 21) -
12th International Post Conference Seminar on
“FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS“

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5 Seminar Contributions

In the following, the seminar papers prepared for the 12th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' held as Pre-conference Seminar of the 21st International Conference on Structural Mechanics In Reactor Technology (SMiRT21) are provided in the order of their presentation in the seminar. The slides of the presentations can be found as far as publicly available on the CD in the attachment.

COMMISSIONING AND INTEGRATED TESTING OF FIRE AND LIFE SAFETY SYSTEMS IN NUCLEAR POWER PLANTS

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ABSTRACT

Commissioning and integrated testing of fire and life safety systems in nuclear facilities is presented from the perspective of a complete building process. This process is beyond testing and start-up at the end of the construction phase. Commissioning is considered from the very beginnings of the project and follows through the life of the facility.

Commissioning of the building for passive and active life safety systems is to ensure the Owner's needs for a usable system are met. For nuclear power facilities, regulatory as well as operational requirements will be documented and demonstrated.

Commissioning from an authority having jurisdiction's (AHJ) perspective brings together the integrated testing of the systems not accounted for under individual acceptance testing. End to end performance will demonstrate the anticipated system reliability and document these results.

This paper will highlight and discuss the application of commissioning as a flow down of requirements from upper tier documents to be integrated into the building and systems design. Interconnected as well as individual systems are brought together collectively to provide the facility with documented fire and life safety system performance.

Emphasis of topics covered in this paper includes the following:

- ***Basis of Design*** - The framework for the Design, Construction, Acceptance, and Operation of the Building Fire and Life Safety Systems
- ***Commissioning Plan*** - The living document specific to the Owner's Building
- ***Design Methodology*** - How Commissioning is integrated into Design
- ***Construction*** - How Commissioning is integrated into Construction
- ***Testing Criteria*** - How the Integrated Testing is to occur
- ***Occupancy Phase*** - Commissioning during the life cycle of the building

Commissioning ensures that the Owner's requirements are addressed for the life cycle of the facility. For nuclear power plants the fire protection and life safety systems goals include minimizing the probability of fires and their consequences, ensuring the fires will not affect the performance of necessary safe shutdown features and fires will not significantly increase the risk of radioactive releases to the environment.

NOMENCLATURE

Term	Abbreviation
Basis of Design	BOD
Fire Commissioning Agent	FCxA
National Fire Protection Association	NFPA
Owner's Project Requirements	OPR

Term	Abbreviation
Re-commissioning	Re-Cx
Retro-commissioning	RCx

INTRODUCTION

Commissioning of nuclear power facilities fire and life safety systems is a process that occurs throughout the life cycle of the facility. The term "commissioning" is expanding from traditional systems balancing and testing to include total building commissioning [1]. As national and international codes and standards develop commissioning criteria, fire protection and life safety systems are recognized as part of whole building systems [2]. Active and passive systems are included in this scope. Commissioning for the active and passive fire and life safety systems can be separated into distinct phases:

– The "pre-design phase", which organizes a "commissioning team" to develop project fire protection commissioning as a functional requirements;
– The "Design Phase", which incorporates these commissioning features into the facility specifications and design documents;
– The "Construction Phase", which transfers the designs into the facility systems;
– The "Occupancy Phase", where commissioning ensures the systems continue to perform as intended.

WHAT IS COMMISSIONING?

Commissioning is a systematic process that provides documented confirmation that specific and interconnected fire and life safety systems function according to the intended design criteria set forth in the project documents and satisfy the Owner's operational needs, including compliance requirements of any applicable laws, regulations, codes, and standards requiring fire and life safety system. [3], [4]. Commissioning is initiated in the design phase by documenting the design intent and continues throughout construction, acceptance, and the occupancy period. Fundamental to commissioning is integrated testing. Commissioning does not modify or supplant existing standard requirements for testing, rather it's intended to complement these requirements by stressing integrated testing to ensure reliable functionality [5].

Listed here are some of the documents which are used in the United States that contain testing requirements for individual systems:

- NFPA 72 National Fire Alarm and Signaling Code -SIG-TMS forms and requirements,
- NFPA 25 ITM Water-Based Fire Protection Systems - forms and requirements,
- Other NFPA Extinguishing System Standards - forms and requirements,
- American Society of Heating, Refrigerating and Air-Conditioning Engineers,
- Building Management Procedures.

Integrated Testing

A definition of integrated testing as it pertains to fire protection is, "An assessment of fire protection and life safety systems function and operation using direct observation or other monitoring methods to verify the correct interaction and coordination of multiple integrated systems in conformance with the fire protection and life safety objectives.”

Commissioning

Commissioning applies to the functions of integrated systems, provided for fire protection or life safety, in the design phase, construction phase and occupancy phase of the commissioning process. Commissioning is done primarily to ensure that the building Owner’s operational needs are met.

The “owner's project requirements” (OPR) outline those needs. For nuclear power plants owner's project requirements include [6]:

- Minimizing the probability of fires and their consequences;
- Fires will not affect the performance of necessary safe shutdown features.
- Fires will not significantly increase the risk of radioactive releases to the environment.

It is the fire protection engineers’ responsibility to incorporate these requirements into the facility design. Proper system engineering, Hazards analysis, and system reliability through end to end system commissioning will integrate safety as a facility feature. The fire protection engineer will be responsible for development of the owner's project requirements document as the foundational fire and life safety systems commissioning document.

Owner’s Project Requirements (OPR)

The owner’s project requirements document should form the basis from which all design, construction, acceptance and operational decisions are made. The OPR should be developed with input from the owner and all key facility users and operators. The development of the OPR begins with appointment of a commissioning team.

Commissioning team representation should include as many of the following entities’ as is practical:

- Owner or appointed representative,
- Commissioning authority,
- Fire commissioning agent (FCxA),
- Installation contractor(s),
- Manufacturer’s representatives,
- Registered design professional(s),
- Construction manager / general contractor,
- Owner’s technical support personnel,
- Facility manager or operations personnel,
- Insurance representative,
- Authority having jurisdiction (AHJ).

The commissioning team develops the owner’s project requirements document and initiates the commissioning process.

Key Commissioning Team Roles

There are certain commissioning team roles which are important to successful commissioning. Those roles play a vital function to ensure the owner's needs are met. These roles will be further defined here:

- Fire commissioning agent,
- Registered design professional,
- Installation contractor

Fire Commissioning Agent

The fire commissioning agent should have at a minimum, an advanced understanding of the installation, operation and maintenance of all fire protection and life safety systems proposed to be installed, with particular emphasis on system integrated testing. An expert level of understanding of the regulatory requirements for nuclear power plants is mandatory. A fire commissioning agent should have the ability to read and interpret drawings and specifications for the purpose of understanding system installation, testing, operation and maintenance. Analyze and facilitate resolution of issues related to failures in fire protection and life safety systems. They will also be able to provide good written, verbal, conflict resolution and organizational skills.

The fire commissioning agent represents the owner's interests and has responsibly for organizing and executing the owner's requirements document. These responsibilities will vary according to the facilities but have basic categories of duties. A few of these duties are outlines here.

– Organizing and leading the commissioning team to keep the information focused on the project requirements and goals;
– Coordinating the commissioning team meetings to prioritize issues and facilitate consensus among participants. It is also the responsibility of the fire commissioning agent to facilitate the development of the owner's project requirements to ensure regulatory and operational compliance.
– Verifying the commissioning process scope of work to bound the commissioning team's effort and maintain focus;
– Integrating commissioning into the project schedule to ensure adequate resources and time are allocated;
– Preparing the commissioning plan to identify roles and responsibilities;
– Preparing commissioning process specifications to identify and quantify performance expectations;
– Executing the commissioning process according to the commissioning plan and adjusting the plan when conditions require;
– Reviewing the plans and specifications to ensure commissioning attributes are incorporated as well as regulatory and operational requirements;
– Attending pre-bid meetings to provide technical support to the contract formation process;
– Approving systems components and design to ensure compliance with design specifications. Tracking and documenting issuesto closure;
– Preparing commissioning progress reports to document the items which have been accepted and track those tasks which require completion;
– Witnessing system testing to ensure components and systems function as intended.

- Reviewing installation documents and making corrections or alterations to meet commissioning objectives;
- Recommending acceptance to owner's representatives that fire and life safety systems are as documented;
- Tracking compliance with matrix so that all systems and components are integrated.

Registered Design Professional

The next role to be defined is the registered design professional. The registered design professional should be a registered professional engineer, architect, or other professional with credentials acceptable to the jurisdiction in which the project is taking place. The registered design professional is responsible for the technical aspects of the commissioning process. Responsibilities of the registered design professional shall include all of the listed items:

- Participating and assisting in the development of the owner's project requirements;
- Documenting the basis of design(BOD);
- Preparing contract documents to ensure incorporation of commissioning requirements;
- Responding to the commissioning team design submission review comments;
- Specifying operation and maintenance of systems in the project specifications;
- Reviewing and incorporating the commissioning teams comments, as appropriate;
- Reviewing test procedures submitted by the installation contractors;
- Reviewing and commenting on the commissioning process progress reports and issues log reports;
- Reviewing and accepting record documents as required by the contract documents;
- Reviewing and commenting on the final commissioning record;
- Recommending final acceptance of the systems to the fire commissioning agent and the Owner.

Installation Contractor

The installation contractor is responsible for implementing and incorporating into the systems for which they are responsible for the objectives of the owner's project requirements.

- The installation contractor's responsibilities are to include all commissioning process requirements and activities in the scope of services:
- Attend required commissioning team meetings;
- Include commissioning process milestones in the project schedule;
- Implement the training program as required by the contract documents;
- Provide submittals to the registered design professional, owner and commissioning team;
- Develop individual system test plan, including acceptance and integrated testing;
- Notify the general contractor and fire commissioning agent when systems are ready for testing;
- Demonstrate the performance of the systems, including integration;

– Complete the construction checklists as the work is accomplished;
– Continuously maintain the record drawings as required by the construction documents.

COMMISSIONING AS A PROCESS

Commissioning is a process that continues for the life time of the facility. In order to understand how these activities are interdependent the processes are divided into distinct phases. Those phases are:

- Pre-design phase,
- Design phase,
- Construction phase,
- Occupancy phase.

Pre-Design Phase

During the pre-design phase of the project, the commissioning team should consider all aspects of the facility and use that knowledge to complete specific tasks. Those tasks include:

– Developing the owner's project requirements;
– Selecting the fire commissioning agent;
– Identifying the commissioning scope;
– Developing the preliminary commissioning plan;
– Reviewing the pre-design documents,
– Developing regulatory code analysis;
– Initiating the commissioning plan.

By carefully incorporating commissioning of the fire and life safety systems into the facility design the process of flowing down upper tier requirements is facilitated.

Design Phase

During the design phase the commissioning team should devote their efforts into the preparation of documents that facilitate commissioning. This is accomplished by documenting the scope for commissioning activities:

– Documenting the commissioning procedures and creating a commissioning activities schedule;
– Verifying that the construction documents comply with the basis of design;
– Identifying qualified specialists and their responsibilities;
– Coordinating and documenting commissioning team meetings and progress reports;
– Documenting issues and changes that affect the commissioning process and updating the commissioning plan;
– Creating construction checklists for use by the installing contractors;
– Creating required project testing requirements that facilitate commissioning.

This should include check lists requiring when AHJ's and Commissioning Team members are to be present during acceptance testing. Developing project training requirements to standardize and familiarize those responsible for the commissioning activities.

By participating as a team during the design phase, all aspects of commissioning are reviewed and incorporated into the facility and system designs.

Construction Phase

During the construction phase the fire commissioning team should provide guidance and oversight to ensure the design aspects which facilitate commissioning are correctly implemented. This is accomplished by confirming that the commissioning schedule is still valid:

- Verifying submittals are in conformance with the basis of design;
- Verifying materials, construction and installation comply with the basis of design;
- Confirming qualified specialists are performing commissioning activities per plan;
- Coordinating and documenting commissioning team meetings and progress reports;
- Documenting any issues and changes to the project and updating the plan;
- Performing commissioning quality control construction inspections with plan check-lists;
- Performing required observation procedures or causing them to be performed by the responsible party;
- Recording and adjusting for any revisions and/or changes to plan documents;
- Verifying and documenting testing performed in the construction phase.

Occupancy Phase

The minimum requirements for occupancy phase should be contained in the basis of design and should include but are not be limited to the following tasks and responsibilities:

- Acceptance testing and inspection completion and documentation;
- Conducting tests for modifications made during the construction phase;
- Delivering of system manual, operation and maintenance manuals, and vendor emergency contact lists;
- Training on the use and operation of the systems;
- Implementing document control for record set drawings and documents;
- Implementing document control for test and inspection records for the systems;
- Maintaining a digital copy of site specific software for systems that is current with the installed system;
- Documenting and maintaining warranties for the systems and equipment;
- Implementing recommended preventative maintenance program for systems;
- Maintaining a list of required inspections, testing and maintenance for fire protection and life safety systems.

Passive Systems

It is the intent of total building commissioning to include passive systems.

Commissioning plans should also identify the requirements for passive fire protection systems including:

- Fire and smoke dampers,
- Fire and smoke doors,
- Through penetration fire stops.

Re-Commissioning and Retro-Commissioning

Commissioning plans should also address re-commissioning (Re-Cx) and retro-commissioning (RCx) requirements of active and passive fire protection and life safety systems where installed in existing structures. An important aspect of these items is periodic integrated testing.

Periodic Integrated Testing

Periodic integrated testing (PIT) should verify correct operation of fire protection and life safety systems in accordance with the established design criteria, basis of design, owner's project requirements, equipment performance requirements or applicable codes and standards. Fire protection and life safety systems that have been commissioned upon installation in accordance with the commissioning process should have periodic integrated testing performed at intervals according to the commissioning plan.

Integrated systems in structures and facilities that have not been commissioned in accordance with the commissioning process, should have integrated testing performed according to an acceptable commissioning plan. This should occur:

- Where new component fire protection and life safety systems are installed and interconnected to existing fire protection and life safety systems;
- Where existing fire protection and life safety systems are modified to become component, interconnected systems;
- Where the interconnections or sequence of operations of existing integrated fire protection and life safety systems are modified.

Phased sequencing of periodic integrated testing should be permitted subject to the approval of the AHJ.

Forms

Commissioning documents and forms should be used to record commissioning and integrated testing of fire and life safety systems.

Basis of design documents referenced in installation standards should be utilized. Testing and inspection documents referenced in installation standards should be utilized for individual system testing. Where required by the AHJ, jurisdictional forms should be incorporated. Where no form or checklist exists, the registered design professional should be responsible for developing a form or checklist. The authority having jurisdiction should approve all forms.

CONCLUSIONS

Fire protection of nuclear power plants is an important responsibility of the fire protection engineer. Commissioning of the fire protection and life safety systems is a process that documents and ensures that the requirements of the owner are maintained through the life of the facility. For nuclear power plants, fire and life safety systems requirements are summarized as three specific goals.

- Minimizing the probability of fires and their consequences,
- Fires will not affect the performance of necessary safe shutdown features.
- Fires will not significantly increase the risk of radioactive releases to the environment.

The commissioning process occurs throughout the life of the facility and is categorized into four phases:

- Pre-design phase,
- Design phase,
- Construction phase,
- Occupancy phase.

Responsibility for the development and commissioning of the fire and life safety systems falls to several individuals. Those individuals include:

- Owner or appointed representative,
- Commissioning authority,
- Fire commissioning agent (FCxA),
- Installation contractor(s),
- Manufacturer's representatives,
- Registered design professional(s),
- Construction manager / general contractor,
- Owner's technical support personnel,
- Facility manager or operations personnel,
- Insurance representative,
- Authority having jurisdiction (AHJ).

As part of the whole building commissioning, fire and life safety systems must perform as independent and integrated systems. For nuclear power plants, commissioning will document and demonstrate the reliability and integrated functionality of the fire and life safety systems.

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OVERVIEW OF THE ESREDA WORKING GROUP ON FIRE SAFETY: PERSPECTVES, ACTIVITIES AND RESULTS

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ABSTRACT

The paper illustrates the aim, the activities and the early results of the Working Group on Fire Safety, founded within the European Safety, Reliability and Data Association (ESReDA).

The preliminary objectives of the Working Group were:

- Organizing the methodologies and the data to be used in fire safety analysis,
- Providing a guidance handbook, where main fire safety standards and methodologies for fire risk assessment are discussed and criticized trough practical application,
- Facilitating cultural dissemination through a reference guide both for industry application and education and training,
- Extracting in a technical conclusion or consensus on industry reference practices and recommendations,
- Highlighting the need for research enhancements and developments.

The joint activities of expertise coming from academia, research centers, national authorities and companies, both from nuclear and process areas, allowed to create a unique chance of cross-learning between the two fields of application.

This has resulted in the design of a handbook slightly different from the initial intentions.

The output of the whole activity will be in fact a book showing the approaches for risk analysis in nuclear and process fields, comparing them wherever possible, introducing the standards and regulations, also at a national levels, and, above all, highlighting the data available to support risk assessment, the main research activities in the fields, and the need for further research.

THE ESREDA ASSOCIATION

ESReDA is a European association established in 1992 to promote research, application and training in reliability, availability, maintainability and safety (RAMS). The association provides a forum for the exchange of information, data and current research on safety and reliability and a focus for specialist expertise.

ESReDA was formed from the combined forces of EuReDatA (European Reliability Data Bank Association) and ESRRDA (European Safety and Reliability Research and Development Association). The integration of the two forces provides a strong basis for furthering the understanding, development and dissemination of RAMS research and methods throughout Europe.

ESReDA aims to:

- Promote research and development, and the applications of RAMS techniques,
- Provide a forum to focus the resources and experience in safety and reliability dispersed throughout Europe,
- Foster the development and establishment of RAMS data and databases.

- Harmonies and facilitate European research and development efforts on scientific methods to assess, maintain and improve RAMS in technical systems,
- Provide a source of specialist knowledge and expertise in RAMS to external bodies such as the European Union,
- Provide a centralized and extensive source of RAMS data,
- Further contribute to education in safety and reliability,
- Contribute to the development of European definitions, methods and norms.

ESReDA pursues actively its aims through the establishment of project groups, supported by at least four members organizations, involving not only ESReDA members but also other bodies (companies, authorities, universities) not being members of the association, but being willing to make a fair contribution to the project group operation. The initial project duration, and hence the lifetime of a project group, should not exceed three years. A project extension or a new project may be proposed by the project group not later than six months before the planned end date. The project group activities are usually terminated when the desired targets have been met according to the activity plan. The dissemination of the results may be decided during the project group activity, but seminars, guidebooks, and research reports are strongly encouraged.

The organization of seminars is sometimes derived from the activities of the project groups, sometimes devoted to recent advancements or detailed applications of RAMS techniques, methodologies or specific areas of application.

THE ESREDA WORKING GROUP ON FIRE SAFETY – PERSPECTIVES

The aim of this working group was to organize the methodologies and the data to be used in fire safety analysis.

Most companies have standards for fire protection. These standards as well as national regulations typically are prescriptive in nature and require fire protection to be installed, generally without regard to the actual hazard. Fire protection in principle consists of passive fire protection means, such as structural fire barrier elements, and active fire protection features, such as fire detection systems and equipment, and fire extinguishing means, e.g. water and/or foam based stationary fire extinguishing systems and equipment.

In recent years performance-based criteria and methods are under strong development and are increasingly used in determining appropriate fire protection.

FIRE HAZARD ANALYSIS

The first step, in any performance-based approach a fire hazard analysis (FHA) has to be carried out. The hazard analysis techniques mainly used to identify potential hazards in the process and facility are represented by a check-list on the types of typical fire hazards the conditions under which they occur and the operation and/or adequate function of the available fire protection features.

The main problem associated with the quantification step is, as usual, the appropriateness of the data used in the calculations. Obviously, the use of generic industry data may result in risk numbers that vary widely. It is best if company-specific data can be used.

FHA (Fire Hazards Analysis) based evaluations routinely lead to changes in a particular suppression system, modifications to a mechanical process, replacement of materials with their less flammable or non-combustible counterparts, or installation of location specific engineered controls.

Also, changes in existing inspection, testing, and maintenance programs might come about, based on the identification of significant needs for a change to improve the operational reliability of a particular engineered system. It is likely, however, that fire prevention issues will be identified as well.

CONSEQUENCE ANALYSIS

Consequence analysis is the process to determine the impact of the scenarios on people, equipment, etc. From the description of fire hazards, specific fire scenarios of concern arise. Qualitative or quantitative methods, such as a probabilistic fire risk assessment (PRA) can be used to identify consequences for both mitigated and unmitigated conditions.

The resulting consequences can then be evaluated with respect to probability of occurrence, direct financial losses, indirect business interruption, effects on the health and safety of workers and the surrounding public, and other tangible and intangible consequences. In nuclear industry, the impact by radioactive releases to the public and the environment create the most severe consequences to be prevented with highest priority.

From the analysis, specific issues of fire spread, smoke movement, damage to specific pieces of equipment, and dangers to life safety can be identified.

RISK TOLERANCE

After the risk has been assessed, the results must be compared to either governmental or company criteria to determine if the risk is tolerable and thus to introduce preventive or protective improvements, when needed (see Figure 1).

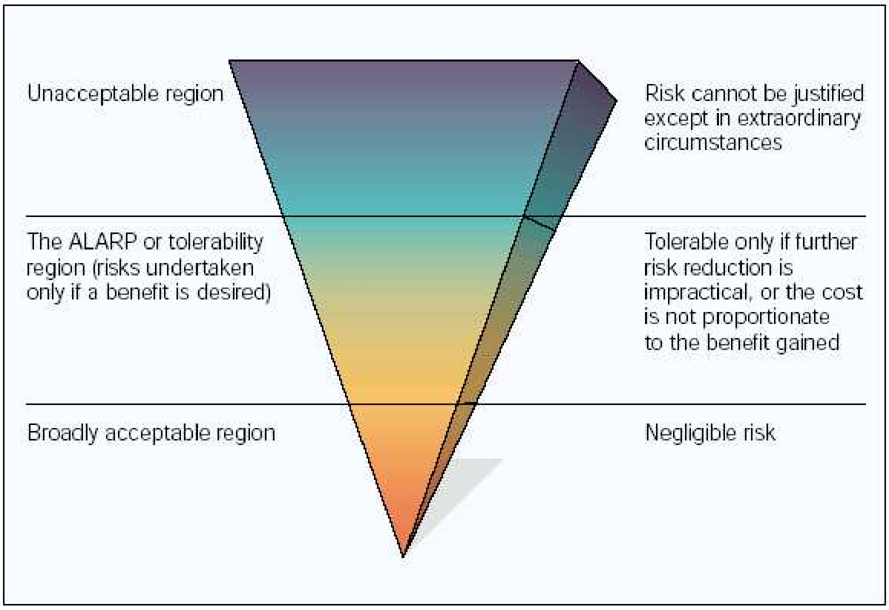

Figure 1 Risk categorization

In Figure 2 a very generic and simplified scheme for risk analysis is outlined.

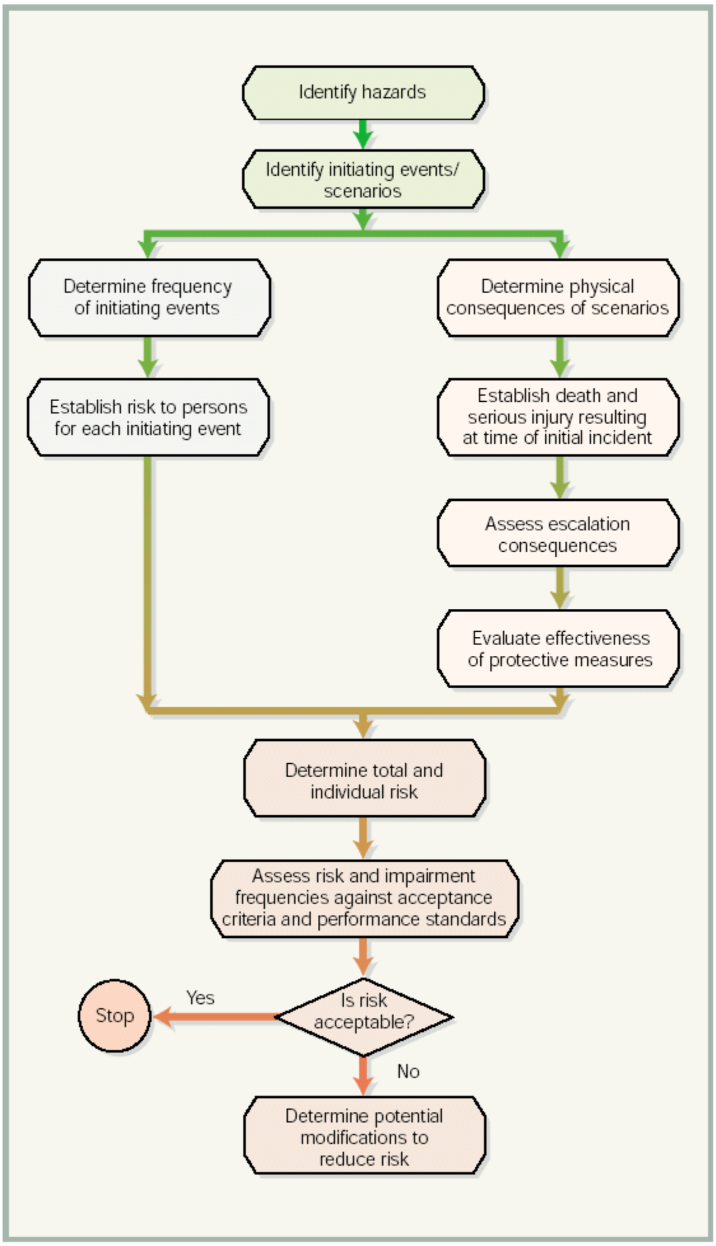

Figure 2 Scheme for risk analysis

Mostly, the performance-based approach relies on experimental data such as tests of building materials, fire spread experiments (e.g. for liquid pool fires, gaseous fires, solid industrial fires forest fires, etc.), fire fighting foam spread tests, smoke spread experiments, and reliability tests of equipment. Thus the research activities in this field are is of primary importance for the ESReDA working group on fire safety.

The major goal of the working group on fire safety therefore is to prepare a handbook where the existing main fire safety standards as well as the state-of-the-art methodologies for fire risk assessment are discussed and criticized trough practical application. This will allow obtaining a reference guide both for application and for education and training.

Participants - from different types of industries, institutional bodies, and universities - are therefore invited to provide practical examples of standards and risk analysis application, with their limits and strengths, results of experimental tests, and data to support fire risk analysis.

The areas of concern for the fire risk working group are described in more detail in the Fire Safety Concepts Tree published by NFAPA [1].

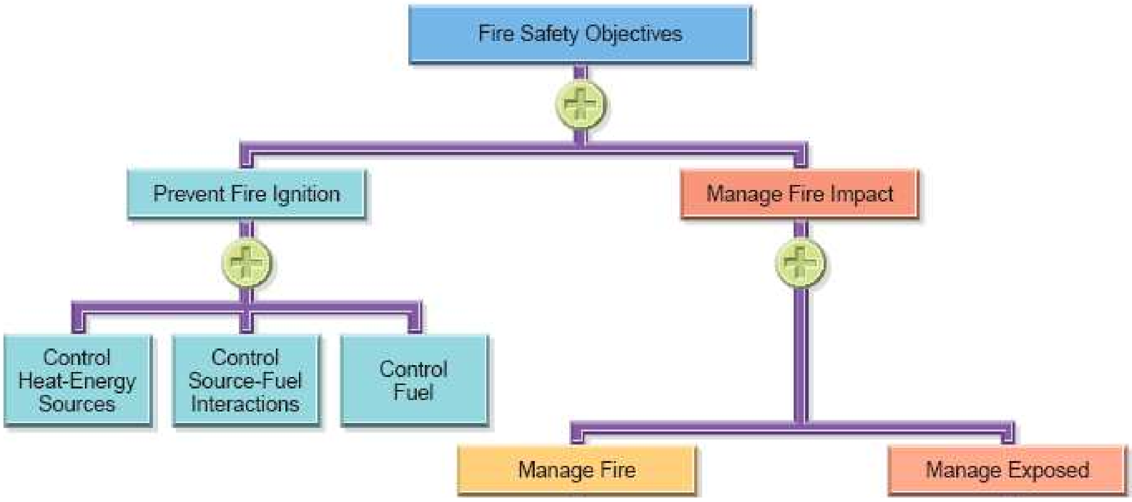

Figure 3 Typical fire safety concepts tree according to [1]

However, in those cases where prescriptive codes supply the minimum requirements for fire protection systems the fire protection is guaranteed with the performance-based approach applying an engineering methodology developed on a scientific basis approach. It allows consideration of a large number of project variables and gives a deeper and often less-expensive engineering solution than the traditional approach. This is true even more, if the special situation requires a tailored engineering and a fit-for-purpose safety approach.

The prescriptive approach is very often used as a pragmatic one which also resolves satisfactorily insurance requirements with a minimum effort. The risk analysis is performed a priori by the legislator, who fixes a safety level and establishes a set of rules able to compensate the existing risk. Thus the fire protection is not guaranteed on the basis of engineering principles and a narrow margin of discretion is left to the fire engineers. In addition, codes usually are written to apply to typical configurations: special situations are very often disregarded or generically treated.

The second approach is performance-based, because it provides solutions based on performance of protection means to established goals rather than on prescriptive requirements with implied goals. The approach is also risk-informed, since the analysis takes into account not only the severity of the events but also the likelihood of the hazard and the probability of failure of any present protection system, providing

- the capability for early identification of weak points in loss prevention and protection systems at the design phase as well as
- the possibility to optimize loss control investments allowing an intelligent allocation of the resources to the area giving rise to the highest risk.

THE ESREDA WORKING GROUP ON FIRE SAFETY – ACTIVITIES

The following experts from different organizations in several European countries are actively participating to the activities of the working group on fire safety at the time being:

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Other contributors for specific subjects and for auditing purposes are also active and expected.

They worked with the following purposes:

- Describe/criticize and review relevant references on:
  - Existing regulatory requirements and7or standards,
  - Existing guidebooks and literature,
  - Existing tutorials and/or training material;
- Provide practical examples in the field of industrial fire safety on:
  - Risk analysis,
  - Hazard analysis,
  - Experimental data,
  - Accident investigations,
  - etc.
- Highlight interdisciplinary peculiarities and/or similitude.

The resulting contributions have been organized as in Table 1.

Table 1 List of contents of the guidebook to be prepared by the ESReDA working group on fire safety

Chapter No.	Contents
1	**Introduction**
2	**Standards and Regulations**
2.1	European and other International Standards and Regulations
2.1.1	*Process Industry*
2.1.2	*Nuclear Industry*
2.2	National Standards and Regulations
2.2.2	*Process Industry*
2.2.2	*Nuclear Industry*
3	**State-of-the-art of Fire Risk Analysis Including Perfor-mance Based Approach and Fire Brigade Strategy**
3.1	Approaches for Fire Risk Analysis
3.1.1	*Process Industry*
3.1.2	*Nuclear Industry*
3.2	Review and assessment of the respective approaches
3.2.1	*Process Industry*
3.2.2	*Nuclear Industry*
3.3	Practical Experiences and Recommendations.
3.3.1	*Process Industry*
3.3.2	*Nuclear Industry*
3.4	Comparison of the Two Approaches
4	**Fire Related Research**
4.1	Fire Experiments
4.1.1	*Process Industry*
4.1.2	*Nuclear Industry*
4.2	Fire Modeling
4.2.1	*Process Industry*
4.2.2	*Nuclear Industry*
4.3	Future Research Needs
5	**Data for Probabilistic Fire Risk Analyses**
6	**Limiting Consequences of Fires**
6.1	Passive Fire Protection
6.2	Active Fire Protection
6.2.1	*Fire Detection*

Chapter No.	Contents
6.2.2	*Fire Fighting*
6.2.3	*Fire Management*
6.2.3.1	Process Industry
6.2.3.2	Nuclear Industry
7	**Conclusions and Recommendations**

THE ESREDA WORKING GROUP ON FIRE SAFETY – COMMENTS AND RESULTS

In the following, some comments about the results obtained so far are provided:

Comments on the Chapter "Standards and Regulations":

Process industry shows a variegate shape, both in production process and materials, as well as in regulations. There is no general rule defining how risk analysis methods shall be adopted in the design of systems. Nevertheless there is a strong trend to move away from prescriptive towards a performance-based design approach, also following the introduction of rules as the ISO/TR13387 [2] or the Regulatory Reform (Fire Safety) Order 200 [3]. The release of a European regulation has forced the national regulation to be aligned, but this applies mainly for the "new" engineering approach, while previous regulations are much differentiated within nations and within industry.

In contrary to the prescriptive approach - which only specifies methods and systems without identifying how these achieve the desired safety goal - performance-based design in the case of fire protection uses an engineering approach based on established fire safety objectives, analysis of fire scenarios and assessment of design alternatives against the objectives. This allows for more design flexibility and innovation in construction techniques and materials, gives equal or better fire safety and maximizes the cost/benefit ratio during design and construction.

Designers of fire fighting systems in process plants adopt either specific company standards (e.g. standard from operators, such as Total, Shell, or standards from the engineering companies, such as Saipem/Snamprogetti, etc.) or they follow the NFPA (mainly) or API standard, or the EN standards where present.

These standards give technical solutions considered to be adequate for the fire protection and generally adopted in process plant firefighting design (e.g. ISO 13702 [4], API RP 2030 [5], NFPA15 [6] provide the minimum specific flow rate to be adopted for cooling of components). In specific cases, they recommend the use of hazard analysis as a tool for defining the requirements; however this is left at a very general level, not recommending any specific approach to be followed.

ISO/TS 16732 [7] and the SFPE Guide to Fire Risk Assessment [8] are guidelines intended to either replace or complement conventional prescriptive codes.

The NFPA 551 code [9] is explicitly designed to assist responsible officials in their duty of confirming (or refuting) the code equivalency of a design proposal justified through a supporting fire risk assessment (FRA); this code provides guidance for those reviewing a fire risk assessment. The International Organization for Standardization (ISO Committee TC 92/SC 4 on "Fire safety engineering" is working on providing fire safety engineering documents for supporting performance-based design and assessment.

The nuclear regulations appear to be less fragmented. In fact, one of the aims of the Western European Nuclear Regulators' Association (WENRA) representing meanwhile the leading regulators of 17 European nuclear regulatory authorities is to develop a harmonized approach to reactor safety. To achieve this objective, the Reactor Harmonization Working Group (RHWG) was set up. The RHWG uses the following understanding of harmonization: No substantial differences between countries from the safety point of view in generic, formally issued, national safety requirements and in their resulting implementation on nuclear power plants.

The safety areas and issues included in the study were selected to cover important aspects of reactor safety where differences in substance between WENRA countries might be expected. They did not seek to cover everything that could have an impact upon safety or to judge the overall level of safety in existing plants. Main task was to develop a set of so-called Reference Levels identifying the main relevant requirements on reactor safety for 18 safety issues. These Reference Levels were primarily based on IAEA (International Atomic Energy Agency) safety standards and one of the safety issues covers the protection against internal fires.

The study indicates that the majority of the Reference Levels are implemented in nuclear power plants in WENRA countries; however, the implementation results need to be further validated. The study also shows that there is a significant amount of work to do to align the national requirements with the Reference Levels, in view of the very strict harmonization definition. It appears that for full harmonization all countries have some work to do both on their regulations and on the implementation of the Reference Levels.

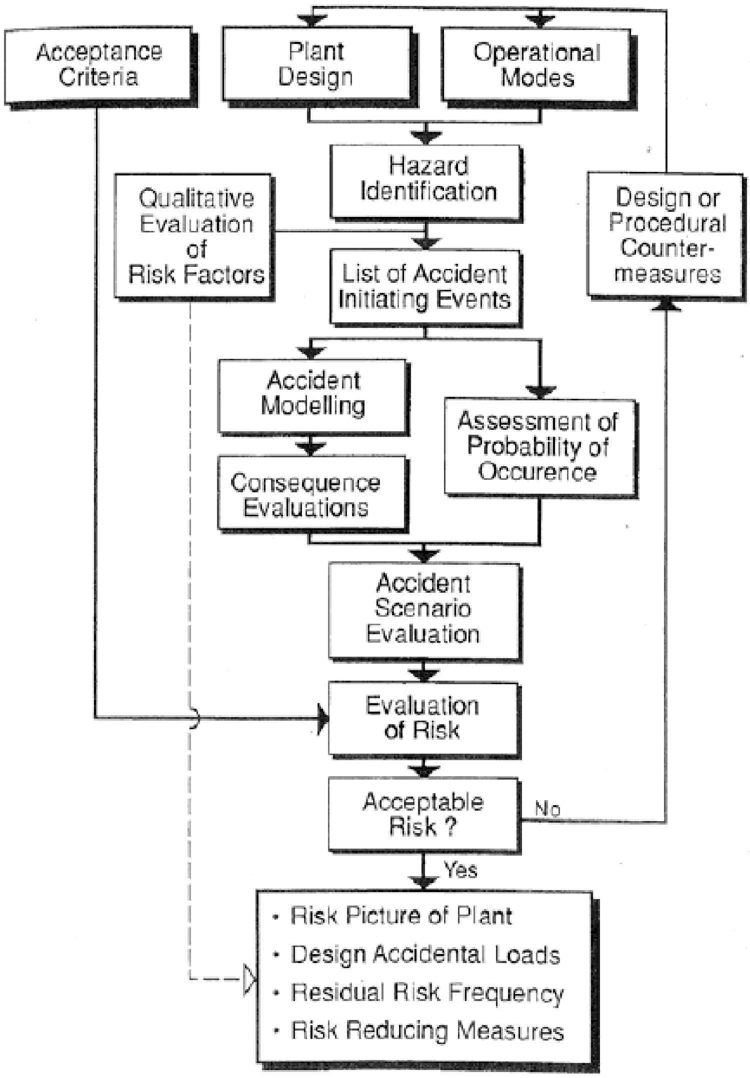
A different area of concern, even if not independent from other ones, is the one of building. The EN Eurocodes are meant to lead to more uniform levels of safety in construction in Europe. They will become the reference design codes. They are currently at the national calibration stage. After publication of the National Standard transposing the Eurocodes and the National Annexes, they have been used in parallel with existing national standards until 2010, when all conflicting standards are withdrawn.

The EN Eurocodes apply for structural design of buildings and other civil engineering works including structural fire design. It is explicitly mentioned that for the design of special construction works (e.g. nuclear installations, dams, etc.) additional or other provisions than those in the EN Eurocodes might be necessary.

Comments on the Chapter "State-of-the-art of Fire Risk Analysis Including Performance-based Approach and Fire Brigade Strategy":

In industrial process plants a generalized fire risk analysis passes through the quantification of the consequences and estimation of the probabilities of the fire hazards identified, the individuation of the hazard control options and the evaluation of their impact on the overall risk, ending with the selection - if necessary - of appropriate further protection means. The systematic steps of a fire risk assessment are:

- Definition of risk assessment objectives;
- Hazards identification;
- Scenarios identification;
- Frequency of occurrence analysis;
- Consequences evaluation;
- Risk assessment;
- Risk-based fire protection analysis and recommendations.



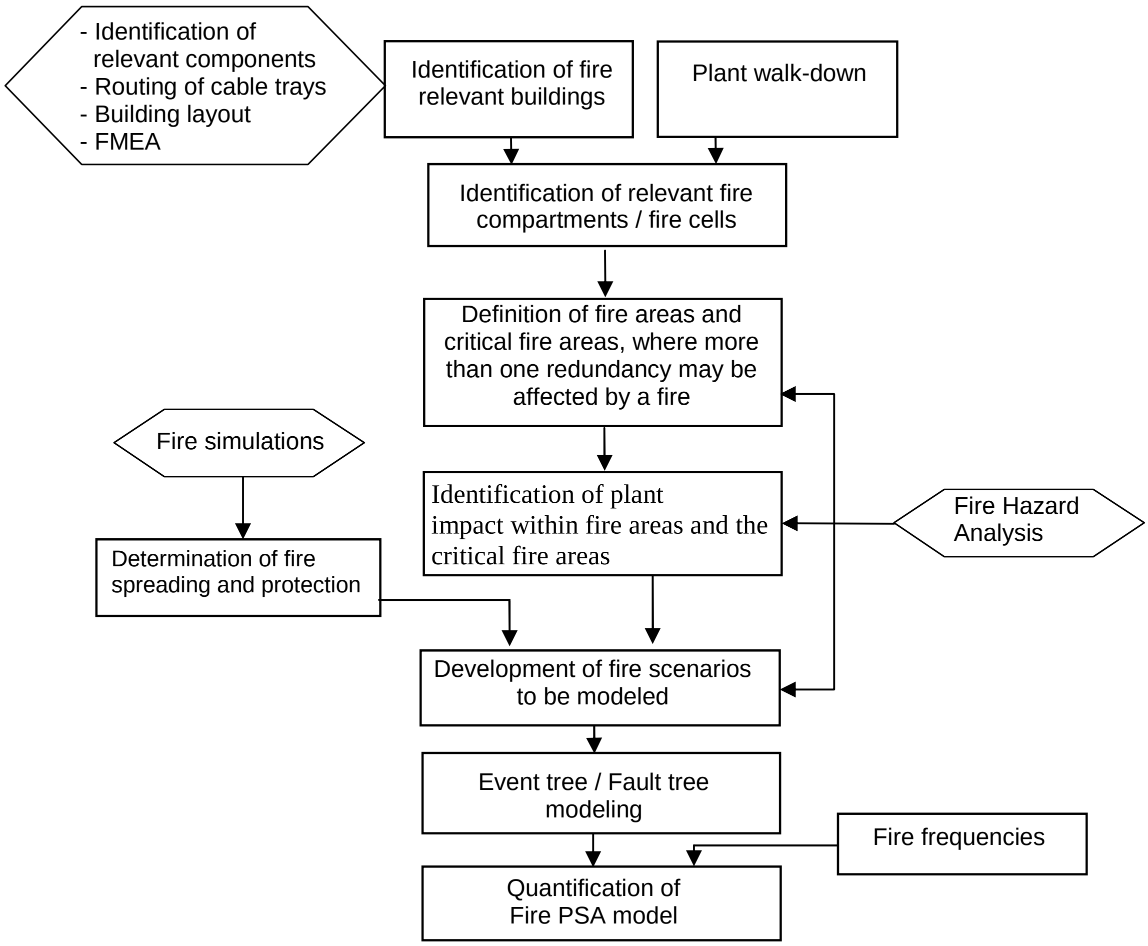
In the nuclear industry, the scope of a probabilistic fire analysis is limited to fires that are initiated from fire sources on-site of a plant (plant internal fires), either inside or outside of buildings. Internal fires are normally analyzed for full power plant operational states as well as for low power and shutdown states considering all areas within the plant boundary where a fire may lead to a core damage sequence. Using probabilistic models, fire PSA takes into account the possibility of a fire at specific plant locations, the propagation, detection and suppression of the fire.

The objective of a fire PSA is to identify fire related event sequences relevant to safety and to quantify their contribution to core damage frequency. This involves e.g. the identification of potential fire hazards, the identification of relevant fire areas, the characterization of fire-induced event sequences, the assessment of fire initiating event frequencies and finally the quantification of the selected fire scenarios. Hence, the contribution of internal fires to the core damage frequency is determined.

A systematic assessment of a fire hazard is one of the important elements in implementing fire protection in plants. When applied at the plant design stage, it permits integration of the proper protection concept into the design and ensures that, throughout all stages of design, construction and commissioning, problems are identified and resolved. For plants in operation it is possible, through a systematic fire hazard assessment, to identify the existing deficiencies in fire protection and to implement practicable and worthwhile improvements in fire safety.

Deterministic and probabilistic techniques are used to assess a fire hazard. The deterministic fire hazard analysis, typically carried out first, is normally required by licensing authorities and other safety assessors. It is usually developed early in the design of new plants, updated before initial loading of the reactor fuel, and then periodically or when relevant operational or plant modifications are proposed. Probabilistic safety analysis (PSA) for fire is undertaken globally to supplement the deterministic fire hazard analysis. It should be noted that a fire PSA is recognized as a tool that can provide valuable insights into plant design and operation, including identification of the dominant risk contributors, comparison of the options for risk reduction and consideration of the cost versus risk benefit.

In the section on practical experiences and recommendations good practices of such analyses in the two areas of application, non-nuclear process plants and nuclear facilities, will be provided as well as some guiding comments and lessons learned from previous experiences to increase the effectiveness and adequateness of fire risk assessment. The approaches within the same area of application and between the two areas of application will be compared and discussed.



Comments on the Chapter "Fire Related Research":

Without any intention of being complete, this section deals with the more recent and/or more relevant fire experiments and modeling to be used as a reference for experimental data and literature, but also to highlight the future research needs, both in process industry as in nuclear. Similar sources of uncertainties have been identified in both areas of application.

From nuclear field experiments, it became obvious that for the simulation of fire scenarios the definition of the fire source presents the biggest uncertainty. Not only the fire dimensions and pyrolysis rate are difficult to specify, but the physical and chemical processes of combustion efficiency, soot and toxic gas yields as well as the radiative fraction also present challenges to the fire modeler. While in theory it is possible to model these phenomena, in practice they generally require 'engineering judgment'. Comparison between blind and open fire simulations show, that the deviation between results and experimental data may be increased by a factor of two, if the pyrolysis rate is not known. Analogous to the pyrolysis rate, parameters such as heat of combustion, combustion efficiency and radiative fraction are also sensitive. The combination of these three parameters controls, to a large extent, the convective power of the fire, and this in turn directly influences the smoke temperature and entrainment rate. The appropriate setting of the convective power is important in obtaining a good match between prediction and measurement for smoke filling cases. For a realistic simulation of fires, the consideration of fire suppression by different types of extinguishing systems (spraywater deluge systems, sprinklers, CO_2 and other gas extinguishing systems) is needed.

In process industry, mainly due to the higher typology of hazardous material manipulated, the situation is even more complex (see Figure 4).

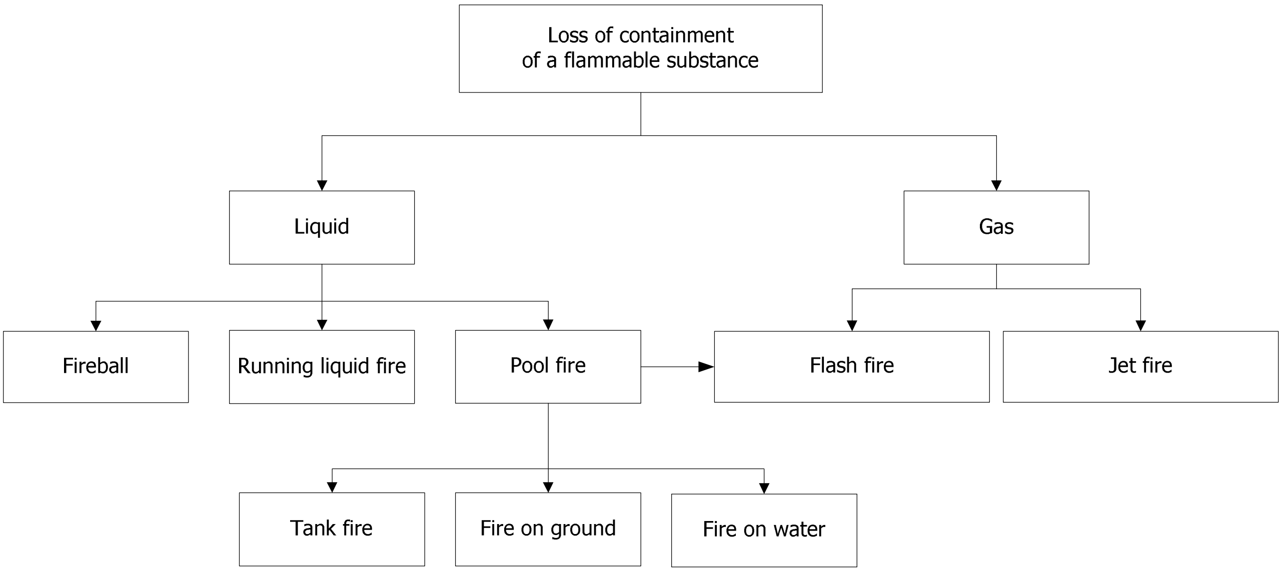


Figure 4 Scheme of different types of fires typically occurring in process industry plants

The value of emissive power as a function of the type of fire and fuel is still poorly known for all types of fire. The same happens with the radiant heat fraction. In the case of pool fires, the expressions to predict the burning rate should also be improved. And, for all types of fire, the size and shape of flames is still modeled in a rather inaccurate way.

The influence of cross wind (inclination and shape of the flames) on large jet fires should also be studied, as well as the behavior of horizontal and inclined sonic jet fires. The expressions used to estimate the main features of a fireball should be improved and, furthermore, the evolution and dynamics of the fireball from its first step are still poorly known; to solve these aspects, the analysis of accidents is essential.

Comparing the different approaches it clearly appeared that, while for fire modeling in process industry CFD (computational fluid dynamics) models are meanwhile state-of-the-art, the experimental tests carried on in nuclear areas have demonstrated that there is still a hard work to do in order to obtain realistic and reliable results from the simulations. This is in part due to the fact that nuclear fires usually occur in enclosed areas of sometimes large dimensions, while in the process industry the fire typically occurs in smaller enclosures or in open fields (even if with obstructions of vessels and piping).

In both areas the main objective of future research is to further reduce the uncertainties of fire simulations with a glance to the reliable prognosis of the failure of safety relevant components in the nuclear field. Furthermore, there is a need for more data obtained from large scale fires to improve mathematical modeling as from fully analyzing the accidents that occur in the process industry.

Comments on the Chapter "Data for Probabilistic Fire Risk Analyses":

A modern assessment of fire effects has to consider all sources of combustible materials and their quantity present, the potential for ignition and the active and passive fire protection features provided to determine frequency and extent of a fire. In order to perform probabilistic fire risk assessment, different types of data are necessary to quantify the fire event tree, such as:

- Fire occurrence frequencies,
- Fire spreading parameters,
- Unavailability of active and passive fire protection features, and
- Failure rates for personnel actions in case of fire extinguishing.

For quantification of potential damage states, data with regard to fire detection, fire enclosure, and fire extinguishing including damages not caused directly by the fire but resulting from the mitigation measures (e.g. the extinguishing media) have also to be provided.

In this nuclear risk analysts are facilitated from the collection of data required by regulations and from the centralized database. This is unfortunately non available for process industry, for which probabilistic analyses rely on data collected on a voluntary basis in single process industry or within large national entities (TNO ion The Netherlands, CCPS **(**Center for Chemical Process Safety)**,** or OREDA, the Offshore REliability DAta) or authorities (HSE (Health and Safety Executive) in the United Kingdom, or the European JRC (Joint Research Center located in The Netherlands).

Comments on the Chapter "Limiting Consequences of Fires":

The section "Limiting Consequences of Fires" will be mainly a reference section about the protection technology and philosophy adopted in both nuclear and process industry.

While passive and active fire protection, fire detection and fire fighting technologies, even if always under development, do not differ in the two areas it was considered worth to analyze the fire management that will probably present some differences.

A specific FMS (fire management system) is a branch of the safety management system of a company and it is built on a similar structure.

The need for fire management in process industry in fact is derived from the introduction of the engineering approach in fire safety, and it is seen as a support to guarantee an acceptable level of risk despite the prescriptive fire safety regulations are not applied in full, even if the technological safety measure have been designed to reach an equivalent safety level (*Italian approach*).

CONCLUSIONS AND OUTLOOK

The final meeting of the working group on fire safety to finalize the work before presenting it to the public is intended to be organized during the ESReDA Seminar in Glasgow, UK in spring 2012.

The results of the working group activities will be presented in a dedicated ESReDA Seminar on this topic to be scheduled, probably in 2013. The Seminar will be devoted not only to the contributions of the working group members but also, and above all, to external contributions to be used as a validation and/or brainstorming on the working group subjects. The final release of the working group results will be published in a handbook.

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FIRE PROTECTION CONCEPT FOR AUXILIARY BUILDINGS IN NUCLEAR POWER PLANTS

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ABSTRACT

Minimax has designed a fire extinguishing system with powder generators for fire protection in auxiliary buildings of nuclear power plants. This fire extinguishing system has been designed for the worst case scenario of a flammable liquid fire in case of an aircraft impact in a nuclear power plant.

MINIMAX AND THE POWER INDUSTRY

In recent years global power requirements have continued to increase. This is not only attributable to industrialisation -particularly in emerging nations- but also to growing electrification. In order to meet the demands the energy sector is currently on the new construction and rehabilitation of power plants. Power plants are characterized by their complex overall systems made up of a range of different operating modules. In addition to this, conditions - such as extremely hot surfaces and lubricating oils -pose huge fire risks. If the beginning of a fire in a power plant is not detected automatically and extinguished immediately, the costs of damage can quickly run up into millions. Even fire damage in a secondary area can cause prolonged shutdown times for the entire power generation process. Equally, false fire alarms and extinguishing system actuation may also lead to power plant shutdown time. In order to protect people, objects and the environment, a sophisticated and made-to-measure fire protection concept is necessary. In power plants, almost the whole spectrum of modern fire protection equipment comes into use. If such equipment comes from one source, fewer interfaces are required, thus ensuring perfect installation and operation.

Minimax has been dedicated to providing power plants with fire protection concepts from one single supplier for over 30 years.

DESCRIPTION OF THE SITUATION

The fire protection of "auxiliary buildings" in nuclear power plants against aircraft impacts or other flying objects with ignition of flammable liquids is not possible with ordinary fire fighting systems.

For the reason of damaged building structures, gas extinguishing systems cannot be used.

Moreover, fire fighting pipework cannot be protected against damage in case of damaged building structures.

THE IDEA – THE USE OF EXTINGUISHING POWDER AS EXTINGUISHING MEDIUM

The installation does not happen as a common facility with piping, but with autarchic powder tanks, which will be activated by destruction.

Common powder extinguishing systems with piping cannot be protected against destruction, e.g. an airplane crash, and therefore cannot be used.

Powder extinguishing systems – as described in this concept – can be repeatedly and redundantly actuated and be combined.

The special is the tripping of the extinguishing system in case of destruction. By this means, the concept presented in the following differs from commonly used extinguishing systems.

GENERAL INFORMATION ON POWDER EXTINGUISHING SYSTEMS

Our powder extinguishing systems are applicable for space and property protection.

Powder extinguishing systems are stationary fire fighting systems. The extinguishing powders used are highly efficient and fast-acting extinguishing agents.

The prompt and three-dimensional extinguishing effect of the powder cloud results from the smothering effect and the anti-catalytic effect, a chemical intervention in the process of combustion.

Extinguishing powders generally consist of atoxic inorganic salts that are mixed with hydrophobing and anti-caking agents. They are called into action of firm, liquid and gaseous fires as well as in metal fires, so in the fire classes A, B, C and D.

POWDER EXTINGUISHING SYSTEM (LARGE)

If several rooms and objects shall be protected by powder extinguishing systems, a central - and sufficient for all protected areas – powder stockage in pressure-resistant steel tanks is to be considered.

In one of these tanks 4,000 kg of extinguishing powders can be stored.

POWDER EXTINGUISHING SYSTEM (SMALL)

For objects with limited scope of protection small extinguishing systems are recommended. These powder extinguishing systems are mostly standardized in structure and dimensioning for a number of objects. Therefore they are minimized in their costs.

Preferred operating areas are cooker hoods, foul-air ducts, laboratory experimental facilities, etc.

APPLICATION AREAS

Typical application areas are:

Chemical factories, fuel depots, compressor and pump stations, transfer stations for oil and gas, gas burner stations at boiler plants, oil basements, rolling mills, test stands, hydraulics facilities, hangars, LNG-, LPG and chemical tankers, laboratory spaces and facilities, special waste facilities.

KEY FEATURES

Key features are:

- Extinguishing module with „fail safe“ conditions
- Release by damage or destruction of components of the powder systems
- Extinguishing powder: COMBI–TROXIN, based on potassium sulphate (suitable for fire classes B and C)
- Propellant: Nitrogen

LAYOUT OF A POWDER GENERATOR

- Powder tank with 100 kg of powder
- Nitrogen cylinder
- Powder nozzle
- Rupture disks
- Release unit
- Test equipment
- Safety valves
- Control line inside the generator (blue line)
- Release line pressurized by compressed air (red line)
- Generator cabinet
- Installation equipment for wall/ceiling mounting

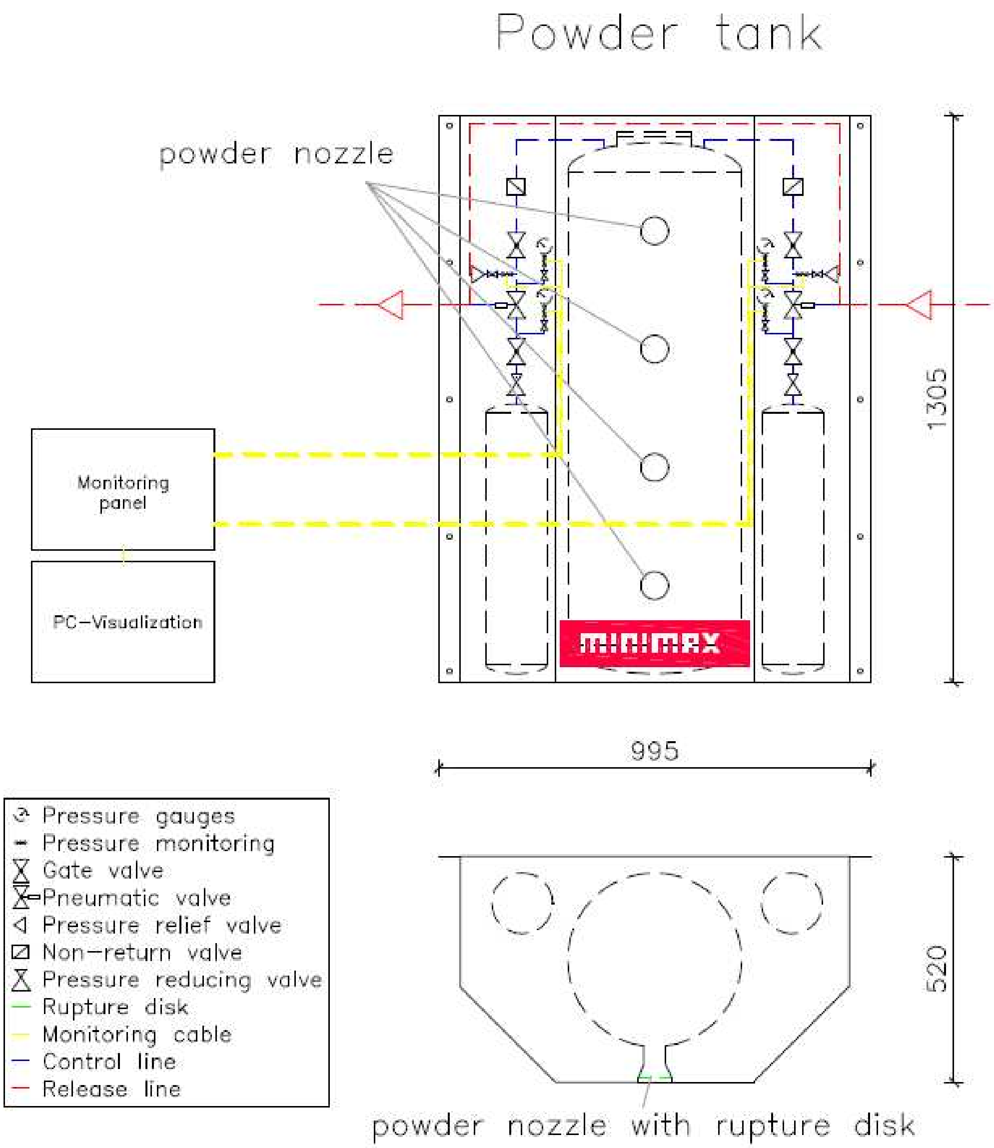

Figure 1 Scheme of the powder extinguishing system

OPERATION DESCRIPTION

The release line is pressurized by compressed air and locks the release valves at the nitrogen cylinders in the modules.

In case of damage or destruction of one or more of the fire modules or release lines, the pressure in the release line drops down and the pneumatic valves at all nitrogen cylinders of the powder generators will be opened.

The nitrogen will pressurize the powder tank. The extinguishing powder will be distributed through the powder nozzles into the protected area.

The flooding time is variable due to the discharge rate of the powder nozzles.

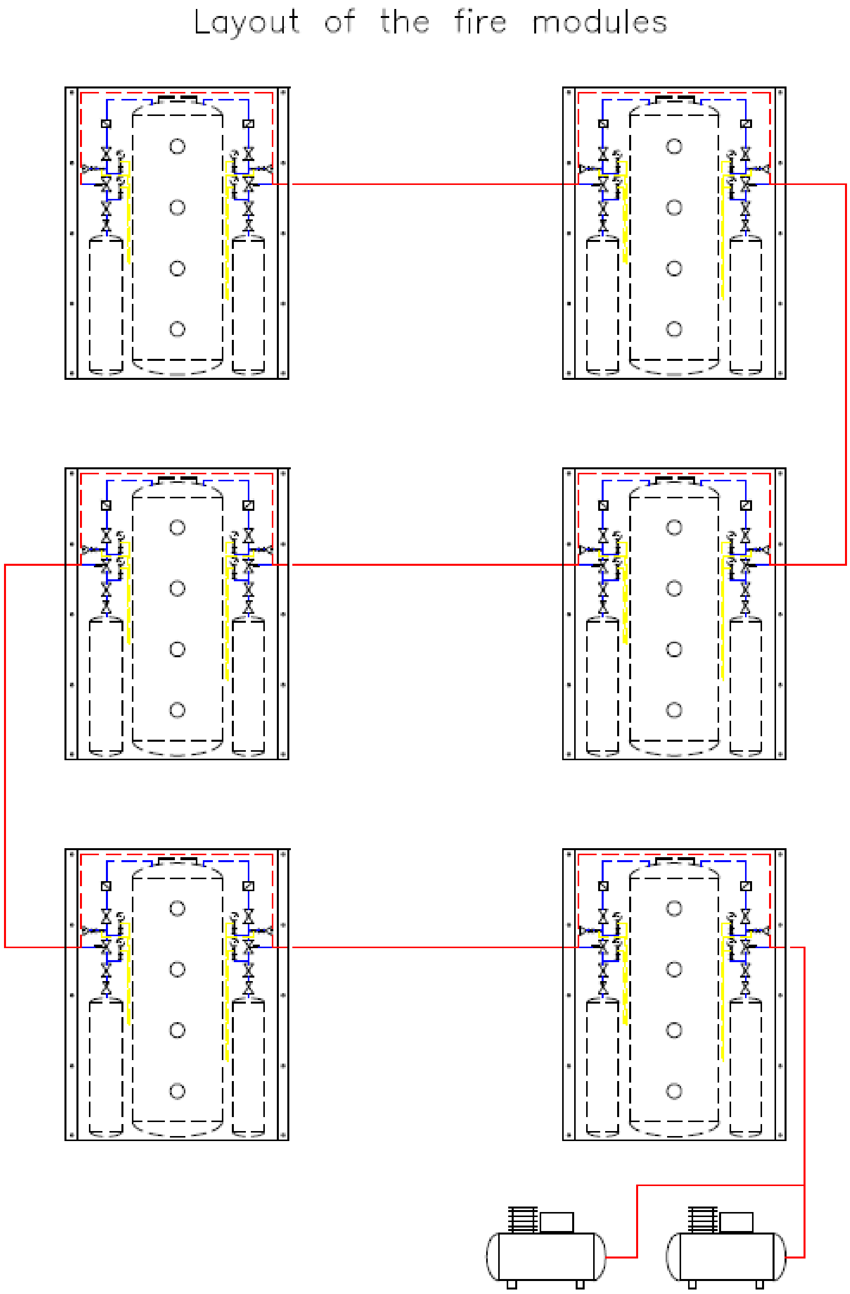

Figure 2 Layout of the fire modules for operation

The required nitrogen propellant cylinders can be arranged in- or outside of the powder tank. Therefore every powder generator is fail safe / inherently safe. The volume of the powder tanks is variable.

Powder tanks in large numbers (as batteries) are mounted inside the building in ceiling height, on the ceiling and/or in various height planes.

All powder generators are assembled in a row by one (or two) control lines on a "release block" per protected area / room.

Through this all-around arrangement of the generators the destruction of all generators in an event of worst-case is unlikely.

This results in a multiple redundancy of the extinguishing system.

EXAMPLE FOR THE DESIGN OF THE EXTINGUISHING SYSTEM

- According to DIN EN 12416 – 2, "Ortsfeste Brandbekämpfungsanlagen Pulverlöschanlagen", September 2007
- Net room volume 700 m^3

➔ Required quantity of extinguishing powder: 500 kg for a closed building structure

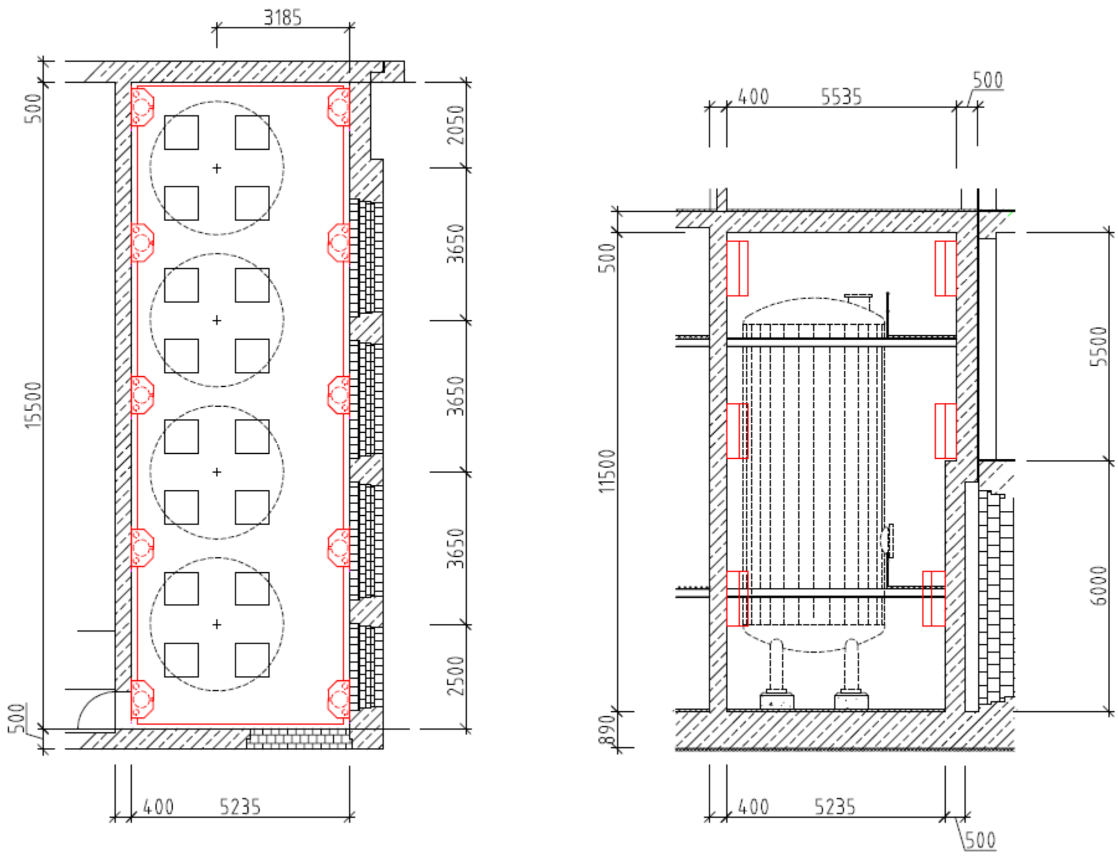

Figure 3 Design example of applying a powder extinguishing system

MINIMAX DESIGN

- 3 levels with 10 powder generators on each level (total 30 pcs.)
- Storage capacity of each powder tank: 100 kg
- Total powder storage: 3,000 kg

SZENARIO OF AN AIRCRAFT IMPACT:

- Loss of 30 % of the extinguishing modules (assumed) in case of an aircraft impact
  - Remaining powder generators: 21 pcs.
  - Equal to 2,100 kg of extinguishing powder for fire fighting
- Required powder quantity: 500 kg in closed rooms (according to DIN)
- According to the above example there will be a safety-factor of the 4.2-fold to compensate the loss of powder inside closed rooms or, in case of disruption of a room, the escape into the atmosphere.

- The design is patent pending by Minimax.

BENEFITS OF POWDER EXTINGUISHING SYSTEMS WITH POWDER GENERATORS

- Independent from foreign energy
- Release of system by damage
- Independent and multiple redundant system
- Modular unit
- Combinable with conventional fire extinguishing and detection systems
- No contaminated fire water
- New fire protection system

ABOUT MINIMAX

Complete Fire Protection Systems from a Single Source

For over 100 years the Minimax Group has been among the world's leading fire protection companies.

Safety through technology: in power plants, paint and wind energy plants, woodworking, logistics warehouses, server rooms and on ships – wherever fire hazards threaten Minimax individual special solutions protect people, machines, buildings and the environment.

As a global full service provider Minimax offers one stop shopping for all fire protection concepts: from simple fire extinguishers all the way to sophisticated automatic extinguishing systems. Our quality is tested regularly, which for us is a matter of course, and for our customers and many licensing authorities a guarantee for fire protection solutions that conform to regulations. Intensive development work in its own research centre also ensures trendsetting innovations for the future. A comprehensive service – from training to maintenance on up to fault senior management – rounds out the Minimax offer. Worldwide around 5,000 employees of the Minimax Group are working for the fire protection safety. You can also rely on fire protection by Minimax.

Minimax Chronology

1902 The company is founded by Wilhelm Graaff and the legendary Minimax conical extinguisher is developed. The trademark Minimax joins.

1906 Minimax is the number 1 worldwide – with foreign companies in Europe and the USA.

1953 In Bad Urach, near Stuttgart, the most modern German factory for fire extinguishers is established.

1967 A new fire research centre for fire protection arises in Bad Oldesloe.

1969 Minimax is taken over by Preussag AG and merged with "Selbsttätige Feuerlöschanlagen Gesellschaft (SFH)".

2001 Preussag concentrates on their main business tourism. Minimax gains a new shareholder with Barclays Private Equity Deutschland.

2003 Investcorp acquires Minimax. Minimax GmbH and Minimax Holding GmbH merge into the Minimax GmbH & Co. KG.

2005 The Minimax GmbH & Co. KG bundle their activities in the range of the mobile fire protection in a discrete association – the Minimax Mobile Services GmbH & Co. KG.

2006 Industri Kapital acquires Minimax from Investcorp.

2007 Minimax expands business in the USA with the acquisition of Consolidated Fire Protection (CFP).

2009 Inauguration of the new and expanded fire protection research centre in Bad Oldesloe.

2009 Minimax merges with Viking.

OVERVIEW OF THE OECD PRISME PROJECT – MAIN EXPERIMENTAL RESULTS

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ABSTRACT

For more than five years (2006 - 2011), the OECD/NEA/CSNI PRISME fire research program has been conducted in an international framework and it dealt mainly with smoke and heat propagation mechanisms in multi-compartment fire scenarios and with the consequences of fire on targets of interest (thermal stress on electrical cables and their potential malfunction). The PRISME Project included several organizations from 12 OECD/NEA member countries: Belgium (TRACTEBEL-Suez, BEL_V), Canada (AECL), Finland (STUK, VTT), France (IRSN, EdF, DGA), Germany (GRS, iBMB, BfS), Japan (JNES), Korea (KINS), Spain (CSN), Sweden (Vattenfall Ringhals), UK (HSE), The Netherlands (VROM-KFD, NRG), and USA (NRC).

As PRISME Project leader, the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN), carried out many fire experiments in confined and mechanically ventilated compartments representative of fire scenarios in the nuclear industry. These fire tests were performed in a large-scale facility, named DIVA, including five compartments connected to an industrial ventilation network. The design of this experimental facility can quite easily be fitted for various fire scenarios of interest in nuclear area and to comply with fire hazard expertise needs.

During this PRISME Project, five experimental campaigns (more than 35 large-scale fire tests) were performed from early 2006 up to mid-2011, named PRISME Source (one single room), PRISME Door (two or three rooms with doorways), PRISME Leak (two rooms with leakages) and PRISME Integral (three and four rooms with doorways).

This paper presents a general summary of the PRISME Project (description of the experimental facilities, the matrix of experiments, the instrumentation used during the fire tests, the objectives of fire experiments) and focuses on some outstanding results.

The experimental outcomes obtained during this PRISME Project provides a better understanding and an increase of knowledge in fire development in confined and ventilated large-scale compartments representative of nuclear area. Moreover, they also contribute to the improvement of fire modeling and constitute a huge experimental database used to validate fire safety software products (based on zone modeling, lumped parameter approach and CFD).

INTRODUCTION

The objectives of the PRISME-OECD project are to investigate different modes and mechanisms involved in the spread of hot gases and smoke from fire room towards adjacent rooms (from 1 to 3 rooms) via the following elements:

- Open door(s);
- Leakages (through openings, narrow slot and a firebreak door);
- Ventilation network (for example, reverse flow due to the effects of pressure, effect of forced vs. natural flow rate in the doorways);
- Ventilation duct crossing through fire room and blowing out adjacent room.

The fire sources are liquid pool fire and real fire source as PVC cables and electrical cabinet. The fire powers are in the range from about 200 kW to several MW depending of the fuel nature, the fuel surface and the ventilation airflow rate. The fire can be extinguished by lack of fuel or lack of oxygen in the fire room.

The large-scale experiments are carried out in a multi-room facility (named DIVA) for the confined and ventilated fire tests and in a calorimeter facility (named SATURNE) for open atmosphere. These experimental installations are located both at Cadarache (France). All compartments of DIVA are representative of nuclear power plants (NPP) with high confined rooms connected to a ventilation network. During the PRISME Project, five experimental campaigns (more than 35 large-scale fire tests) were performed from early 2006 up to mid-2011:

- PRISME Source fire tests devoted to characterize the fire source and to investigate the fire behavior in a single compartment;
- PRISME Door fire tests devoted to study the smoke and gas propagation between two or three rooms through doorway and the effects of thermal stress on PVC electrical cables (surrogate and real cables);
- PRISME Leak fire tests devoted to investigate the smoke and gas propagation between two rooms through leakages (via two openings, a narrow vertical slot, a firebreak door and a crossing duct inside the fire room) and the effects of thermal stress on real electrical cables for which the electrical malfunction was measured;
- PRISME Integral fire tests devoted to study the heat and mass transfer of hot gases and smoke through doorways considering three and four rooms, and a special focus on smoke and heat propagation due to real fire source (electrical cabinet and cable fire sources), on ventilation driving on heat and smoke propagation (ventilation and damper effect), and on effect of water deluge system (sprinkler).

The objective of this paper is to present a general summary of the PRISME Project (such as description of the experimental facilities, the instrumentation used during the fire tests, the matrix of experiments, and the objectives of fire experiments) and to focus on some outstanding results, especially concerning:

- The effects of the oxygen depletion inside the fire room on the fuel mass loss rate;
- The hot gases propagation from fire room to adjacent room for various mechanisms of heat and mass transfer (doorway, leakages, crossing duct);
- The effects of forced versus natural flows between two neighboring compartments.

The experimental results are discussed and the conclusion summarizes the main outcomes of the PRISME Project.

DESCRIPTION OF THE EXPERIMENTAL FACILITIES AND INSTRUMENTATION

The DIVA Facility

The DIVA facility, representative of nuclear installations, is in a large-scale multi-room facility (Figure 1) including four compartments (tagged from 1 to 4) and a corridor. All the walls are of 0.3 m in thick and were built with reinforced concrete allowing to withstand a gas pressure range from – 100 hPa to 520 hPa. The room 4 is not used during this project and would allow to study the vertical hot gas propagation from a lower (room 3) to an upper room (room 4). All the rooms (length × width × height = 6 × 5 × 4 m^3, see Figure 2) are connected with a mechanical ventilation system by means of inlet and outlet ducts. The corridor (length × width × height = 15 × 2.5 × 4 m^3) is located along the rooms 1 to 3. Each compartment is equipped with one inlet and one exhaust ducts of the ventilation network. These latter can be located in the upper or the lower part of each room depending of the fire scenarios. The rooms can be connected through a single doorway (0.7 m × 2.1 m) or different types of elements (simple openings, fire door, etc.). The possibilities offered by DIVA allows to investigate complex scenarios, as encountered in real situations, involving for instance electrical cabinets and cables as targets to study malfunction and failure of such equipment.

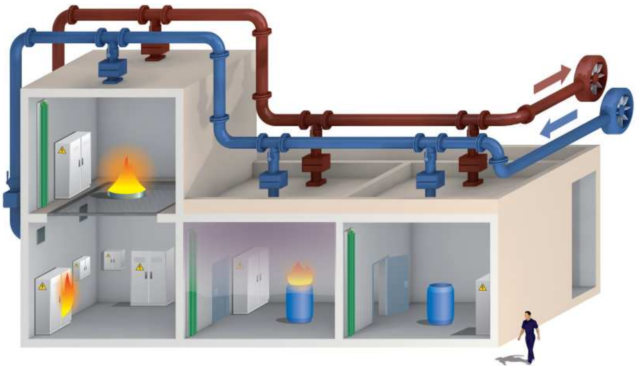


Figure 1 Schematic of the DIVA facility and its ventilation network

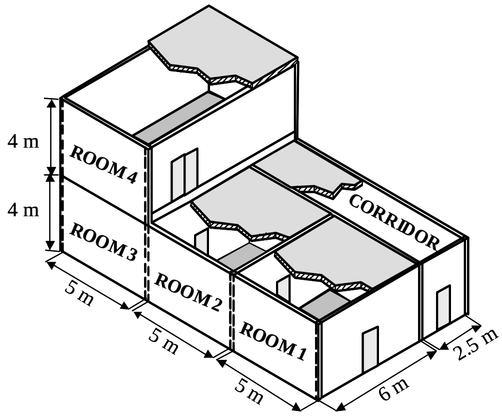


Figure 2 Main geometrical dimensions of the DIVA facility

The DIVA facility is highly instrumented (up to 800 possible measurement channels on the data acquisition system) and its ventilation network allowed it to simulate ventilation configurations representative of NPP as well as nuclear laboratories and power plants.

The SATURNE Facility (Calorimeter)

The SATURNE test facility (see Figure 3) is a large enclosure of 2,000 m^3 (length × width × height = 10 x 10 x 20 m^3), in which is located a large-scale calorimeter.

The fire tests, carried out under the calorimeter hood (see Figure 3 and Figure 4), are devoted to determine the fire behavior in open atmosphere for simple and complex fuels as liquid pool, electrical cabinet and cable trays. They are included in the PRISME Support tests (particularly characterization of HTP pool fire and of cable fire) as described in the Appendix 3. The main characteristics of the calorimeter are summarized in the following:

- Hood: 3 m in diameter,
- Height from floor: for these tests, the height between the floor and the bottom rim of the hood was approx. 4 m.
- The smoke exhaust system is connected to a ventilation network. Its exhaust flow rate can range from 10,000 to 25,000 m^3/h;
- This facility is designed for studying fire source up to nearly 1.5 MW.

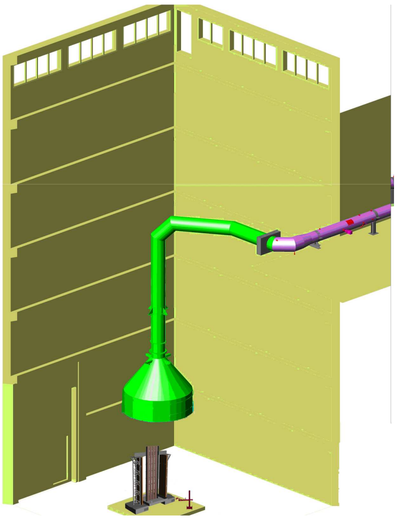


Figure 3 Hood of the SATURNE facility (large-scale calorimeter)

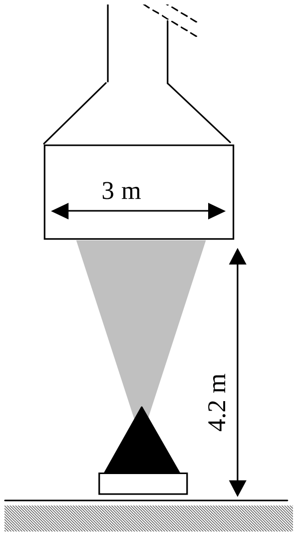


Figure 4 Main geometrical dimensions of the SATURNE facility

Instrumentation

For most of PRISME fire tests, more than 500 measurements are performed in order to fully describe the fire scenarios and to propose a high quality database for code validation. The measurements focus on the following variables: fuel mass loss rate, gas and wall temperatures (or other as inside cables), gas species concentrations (CO, CO_2, O_2 and total hydrocarbons), soot concentrations, radiative and total heat fluxes received by the walls or various targets, pressures and flow rates in all compartments and in ventilation network.

For the SATURNE hood, the measurements available in exhaust duct are mainly those usually in calorimeter system as pressures, gas flow rates, temperatures, gas concentrations of O_2, CO, CO_2 and HCT and soot concentration.

Additional quantities are also determined nearby the fire source such as the fuel mass, temperatures, radiative and total heat fluxes (nearby and far from fire source), and video camera recordings.

DESCRIPTION OF EXPERIMENTS CARRIED OUT IN THE OECD PRISME PROJECT

PRISME Source and Door

As a first stage, the PRISME Source fire tests aims on studying the fire behavior of a HTP pool fire (HTP = Hydrogenated Tetra-Propylene, $C_{12}H_{26}$) being used as fire source in the PRISME Source and Door experimental campaigns. This liquid fuel is tested in open atmosphere (SATURNE calorimeter) for several fuel surfaces and then in confined and ventilated conditions (single room) in order to investigate the effect of oxygen depletion on the fire source. An example of this study will be presented latter in the paper.

In the following of the PRISME Project, the same experimental strategy is used, i.e. to study the fire source in open atmosphere (calorimeter) before to study it in confined and ventilated compartments (DIVA facility) involving the oxygen depletion in fire room. The experimental matrix of PRISME Source including the main parameters of fire tests (pool area, initial mass of fuel, and ventilation flow rate in DIVA) is described in Appendix 1 for the confined and ventilated fires and in Appendix 3 for the fires in open atmosphere. The PRISME Source fire tests in DIVA facility are carried out in the room 2 (see Figure 5).

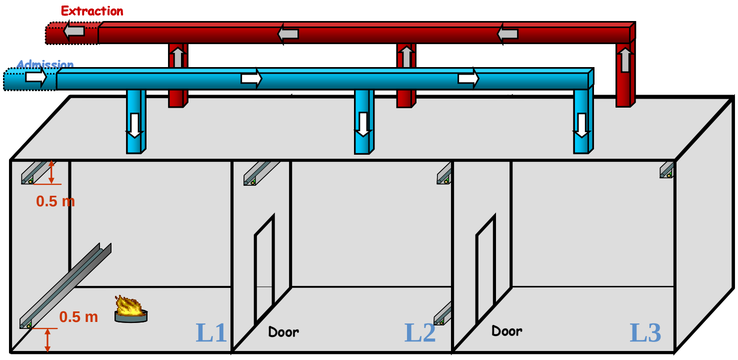


Figure 5 DIVA configurations for PRS Source and Door fire tests (front view)
Source: 1 room (L2) - Door: 2 (L1/L2) or 3 rooms (L1/L2/L3)

The PRISME Door campaign investigates especially the spread of smoke and hot gases through open doors for two and three rooms and also the heat transfer to surrogate and real cables. It used the previous PRISME Source experiments to select the fire test parameters (as the pool area) and their experimental results to study the smoke and hot gas propagation inside the DIVA facility. In Appendix 1, the experimental matrix of PRISME Door is described with details showing the main parameters of fire tests as the pool area, the initial mass of fuel, ventilation flow rate, number of compartments and location of air inlet. The PRISME Door fire tests are carried out in the rooms 1 and 2 for the two-room scenarios and in the rooms 1 to 3 for the three-room scenarios (see Figure 5).

PRISME Leak

The third campaign, PRISME Leak, concerns the propagation of smoke and hot gases through leakages (two openings, narrow slot, firebreak door) between the fire room and the neighboring room (Leak 1 to 3, see Figure 6) and the study of the heat transfer coming from a duct crossing the fire room and flowing in the adjacent room (Leak 4, see Figure 7).

More precisely, PRS_LK1 test concerns the propagation of smoke through two circular holes located in the upper and lower part of the wall separating the source room (fire compartment) and the adjacent room. PRS_LK2 test concerns the propagation of smoke through a vertical slot opening. PRS_LK3 test concerns the propagation of smoke through a real firebreak door. PRS_LK4 test concerns the propagation of smoke through a real firebreak door as well as the propagation of heat through a ventilation duct exposed directly to the fire and blowing heated air in the adjacent room. Some results about hot gas propagation will be discussed latter in the paper. The experimental matrix of PRISME Leak is described in Appendix 2.

A second objective is the cable performance testing of partners' cables (as found in NPP) in order to provide additional data to supplement the knowledge based on cable fire-induced failure modes and effects. This study was carried out with the technical support of Sandia National Laboratories (SNL) and sponsored by the Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research. The equipment and methods used in the PRISME cable performance tests were based directly on those used to support the CAROLFIRE program (cf. [1]).

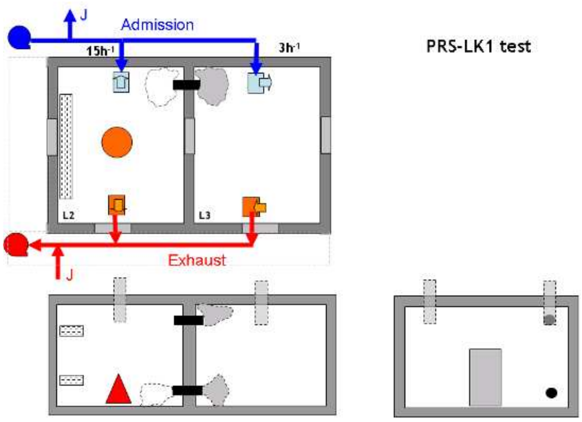


Figure 6 PRS_LK1 fire test (similar for LK2 & 3)(2 rooms; top, front and side view)

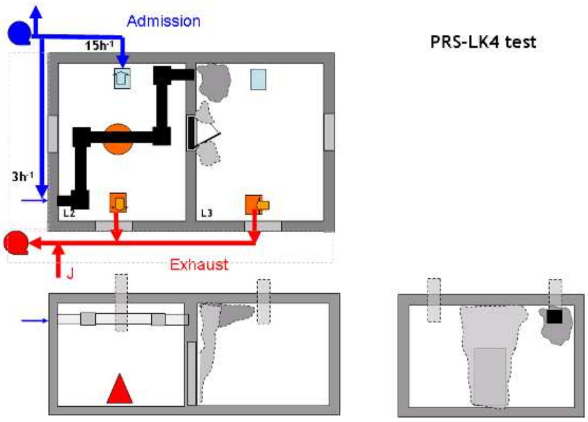


Figure 7 PRS_LK4 fire test (2 rooms; top, front and side view)

PRISME Integral

The last stage, PRISME Integral, aims at studying practical configurations (see Figure 8 and Figure 9) involving various fuels, 3 or 4 rooms connected by doorways, one fire barrier system such as fire dampers and one deluge system such as sprinklers. In these experiments, the air inlet flow goes from room 1 (and corridor) to the exhaust duct in room 3 involving forced vs. natural air flows through doorways (some results will be presented later in the paper). This last experimental campaign involves six fire tests as specified in the Appendix 2. The main objectives of this last campaign are to investigate:

- The propagation of smoke and hot gases through doorways in confined and ventilated rooms;
- The effect of the number of adjacent rooms on the propagation through doorways;
- The effect of sprinkler activation on fire scenario;
- The effect of fire damper closure on fire scenario;
- The behavior of cable fire in confined and ventilated fire scenario;
- The behavior of electrical cabinet fire in confined and ventilated fire scenario.

In Appendix 2, the experimental matrix of PRISME Integral is described with details showing the main parameters of fire tests as the fire source, the number of compartments and other (dampers, sprinkler activation). To determine the fire behavior of cable in open atmosphere, PRISME Support tests (see Appendix 3) are carried out under the SATURNE calorimeter before the fire test (PRS-INT3) inside the DIVA facility.

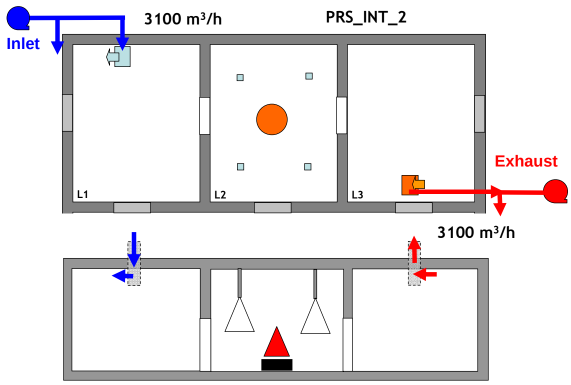


Figure 8 PRS_INT2 fire test (similar for tests 1 & 3) (3 rooms; top and front view)

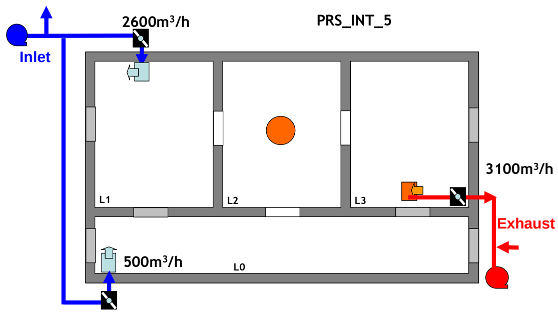


Figure 9 PRS_INT5 (similar for tests 4 & 6) (4 rooms; top view)

OVERVIEW OF THE EXPERIMENTAL OUTCOMES IN THE OECD PRISME PROJECT

Effect of Oxygen Depletion on Fuel Mass Loss Rate

In confined and ventilated fires, the ventilation flow rate is often not high enough compared to the heat release rate (HRR) of the fire source to exhaust the combustion products from the fire room. Consequently, the latter fills up quickly the fire compartment involving the oxygen depletion in it (**[2**], [3], [4], [5], [6], [7], and [13]). As a result, the mass loss rate of the fire source is significantly reduced until fire extinction due to the lack of fuel or of oxygen. In fact, the fire duration can be either shorter because extinction occurs quickly by lack of oxygen, either drastically longer because the decrease of MLR under steady state condition (i.e. without the flame extinction) involved more time to burn all the mass of fuel available in the pan. For example, in the PRISME Source experiments, the fire duration for a renewal rate of Tr = 4.7 is around 2.5 times longer compared to the same pool fire in free atmosphere (see Figure 10 in PRISME Source), similar results being also observed in PRISME Door (as in Figure 11). Of course, the knowledge of the fire duration is of major interest for the analysis of fire hazard and the assessment of fire consequences in nuclear power plants.

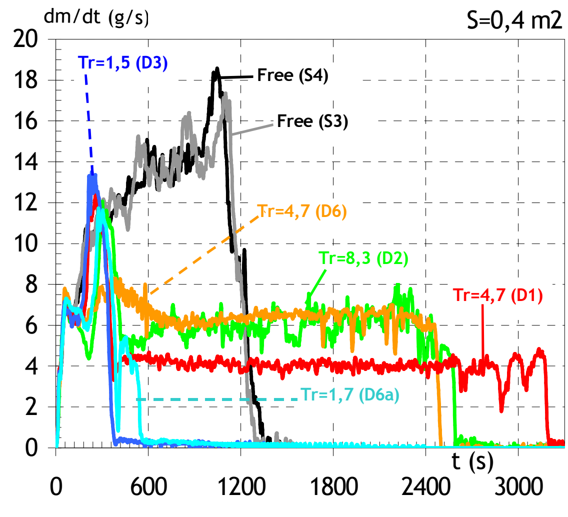


Figure 10 PRISME Source (1 room; S = 0.4 m²) **[4**], **[8**]

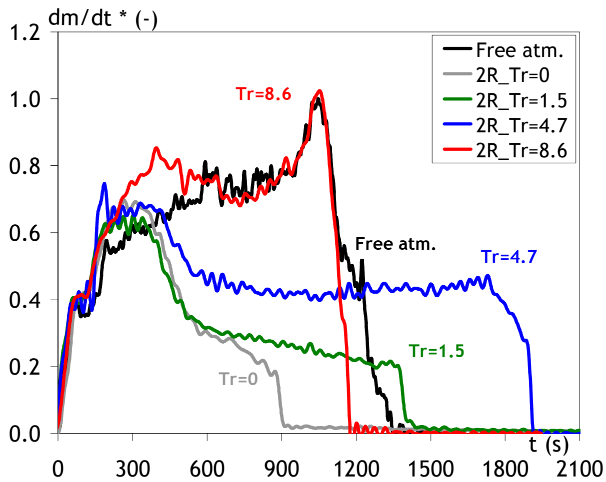
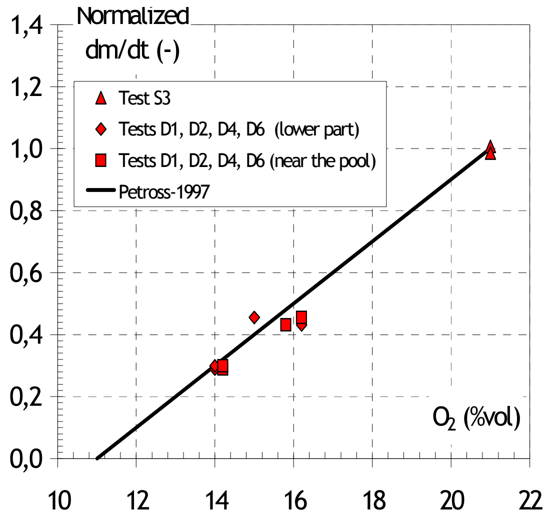


Figure 11 PRISME Door (D1 to D4) (2 rooms; S = 0.4 m²) **[4**], **[8**]

The effect of the air flow rate on the mass loss rate is closely dependent to the oxygen concentration within the fire compartment. From the PRISME Source data, the Figure 12 shows a linear relationship of the fuel mass loss rate versus the oxygen concentration near the pool fire. Moreover, the experimental results are well fitted by the Peatross and Al's correlation [9], [10]. The same behavior is also observed in the PRISME Door test as described in [6].



$$\boxed{\frac{\dot{m}_c}{\dot{m}_c^O} = 0{,}1\,X_{O_2} - 1{,}1}$$

Figure 12 PRISME Source data compared to the Peatross et al. correlation [3], [4], [6], [7]

In PRISME Leak experiments, the fire room reaches mean gas temperature as high as 480 ℃ (so close to flashover conditions obtained with insulated walls of fire room) such as the radiative heat flux from the hot gases increases the mass loss rate of fuel. Consequently, Peatross correlation is no longer valid and a new modeling [5] has been proposed to take account the effect of oxygen depletion and the radiative heat flux from surrounding gases.

Hot Gas Propagation from the Fire Room to Neighboring Rooms

In order to assess the relative effect of heat and mass transfers in PRISME Door and PRISME Leak campaigns, Figure 13 and Figure 14 present the Mass Flow Rate (MFR) and the convective heat flux (CHF) between fire room to adjacent compartment. The propagation of hot gases is obviously larger through doorways than through leakages showing a ratio of nearly 5 to 10 times for MFR and of nearly 3 to 10 times for CHF. This result strengthens in the fact that the high level of confinement remains the best way to limit the propagation of hot gases in nuclear facility and the consequences due to fire (ignition of target, malfunction of electrical components, etc.).

As expected from PRISME Leak fire experiments, the CHF decreases from PRS-LK1 to PRS-LK3 corresponding respectively to openings, narrow slot and firebreak door. Nevertheless, the MFR flowing from fire room to adjacent room is almost similar for these three fire tests and means that the mass transfer of gas depended weakly on the type of leakages in the fire scenarios studied.

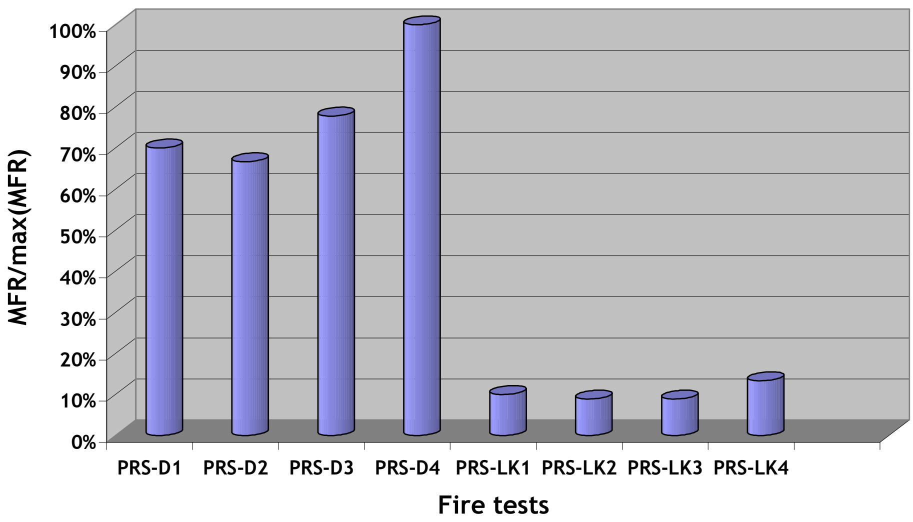


Figure 13 Mass Flow Rate (MFR) from fire room to target room

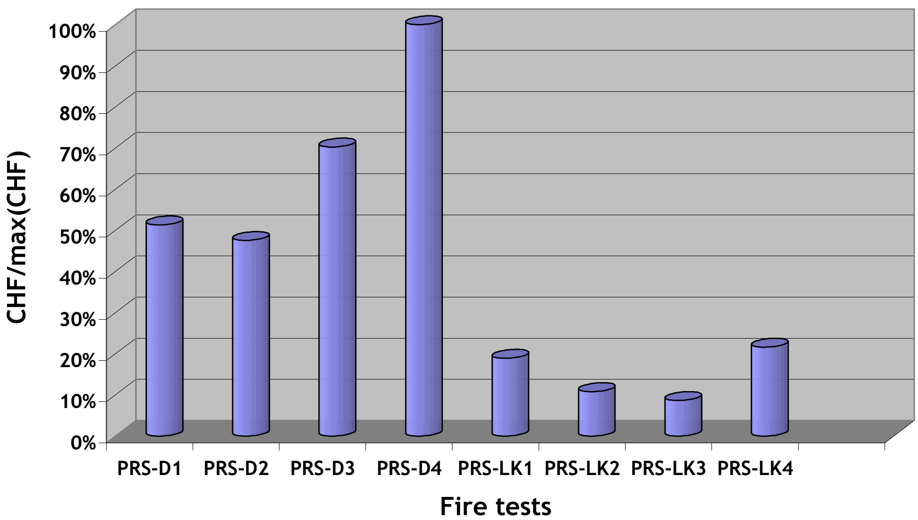


Figure 14 Convective Heat Flux (CHF) from fire room to target room

EFFECT OF VENTILATION FLOW RATE ON THE VELOCITY PROFILES FROM THE FIRE ROOM TO NEIGHBORING COMPARTMENTS

In PRISME Integral fire tests, the effects of the ventilation rate and the fire HRR (obtained from a gas burner, see Appendix 3) on the velocity profiles are analyzed to quantify the natural vs. forced flows during the steady state regime [11]. The forced flow is induced by the ventilation network as initial condition with the air flowing from the room 1 to the room 3 (see Figure 15).This scenario is based on the PRS-INT1 configuration including three rooms connected by doorways.

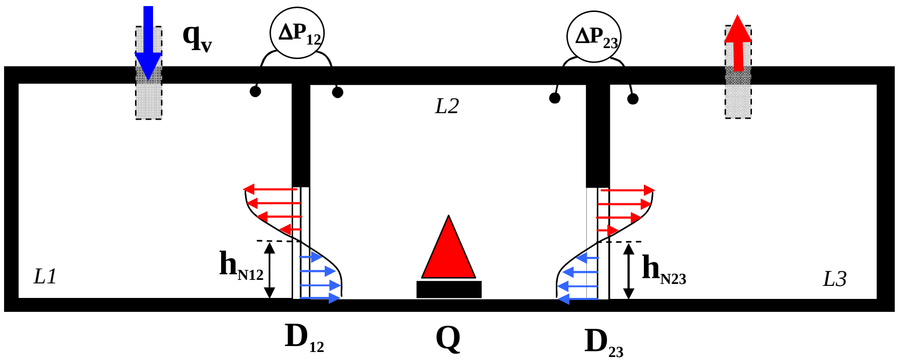
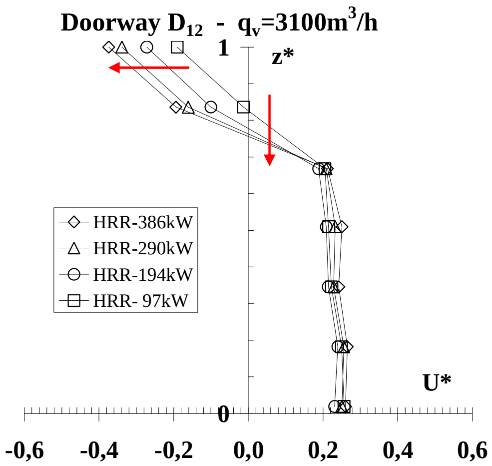


Figure 15 Front view of PRISME Integral configuration for three compartments

Figure 16 and Figure 17 show the change of the velocity profiles due to the fire HRR for fire tests performed with a ventilation flow rate of 3100 m^3/h. The rise of the fire HRR increases the value of the outflow of hot gases at both doorways and the location of the neutral plane becomes lower and lower. The effect of the HRR increase is not significant on the inflow of "fresh" air. In addition, these figures also points out the asymmetry profiles between the D_{12} doorway (close to natural ventilation with two-way flow) and the D_{23} doorway (close to forced ventilation with one-way flow).



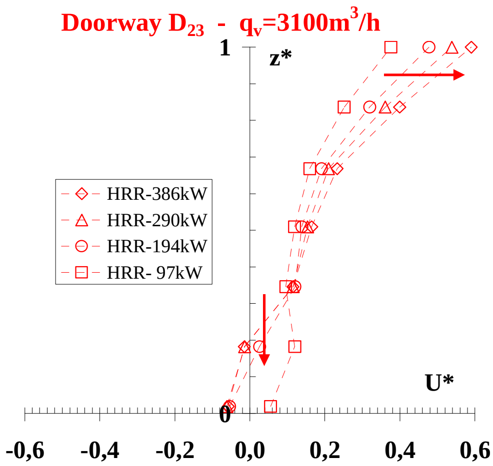


Figure 16 Velocity profile at the doorway L1/L2 for HRR from 100 to 400 kW [11]

Figure 17 Velocity profile at doorway L2/L3 for HRR from 100 to 400 kW [11]

Figure 18 and Figure 19 show the effect of the ventilation flow rate on the doorway flow for a fire HRR of 100 kW. The main effect of the forced ventilation is to shift significantly the value of the inflow. At the upstream doorway (D_{12}), the rise of the ventilation flow rate increases the inflow velocity. Inversely, at the downstream doorway (D_{23}), the flow rate increase contributes to reduce the inflow until completely disappears (forced ventilation with one-way flow).

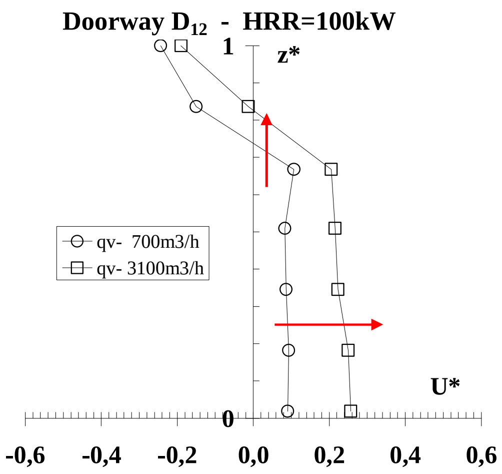


Figure 18 Velocity profile at the doorway L1/L2 for air flow rates of 700 and 3100 m^3/h [11]

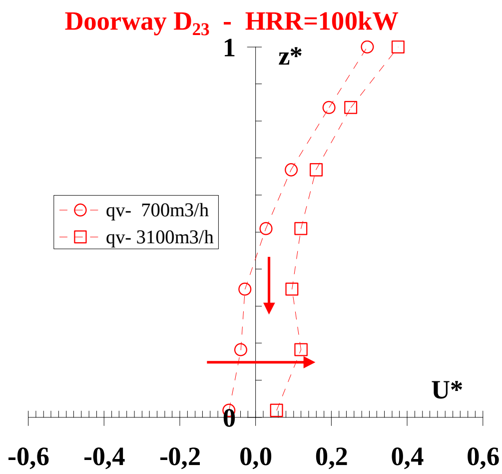


Figure 19 Velocity profile at the doorway L2/L3 for air flow rates of 700 and 3100 m^3/h [11]

Consequently, this experimental result could be important for fire safety in nuclear power plants because the air flow from the ventilation network could promote significantly the propagation of hot gas (and consequently some radioactive materials) to preferential neighboring compartments. This aspect would have to be taken in consideration in fire safety hazard.

CONCLUSIONS

The French Institut de Radioprotection et de Sûreté Nucléaire (IRSN), carried out many fire experiments in confined and mechanically ventilated compartments representative of fire scenarios in the nuclear industry. These fire tests were studied in a large-scale facility, named DIVA, including 5 compartments connected to an industrial ventilation network and a large-scale calorimeter for studying the fire behavior of various fire sources. During this project, five experimental campaigns were performed from early 2006 up to mid-2011, which were PRISME Source (one single room), PRISME Door fire tests (two or three rooms with doorways), PRISME Leak (two rooms with leakages) and PRISME Integral (three and four rooms with doorways). The main experimental results are on the following topics:

- Smoke and hot gas propagation through vertical openings (doorways) between fire room to neighboring rooms and for two modes of convective flows (natural and natural/forced flows);
- Smoke and hot gas propagation through leakages (two openings, narrow slot, fire door) between fire compartments to an adjacent one.
- Study of the heat transfer coming from a duct crossing the fire room and flowing in the adjacent room;
- The effect of sprinkler activation on fire scenario;
- The effect of fire damper closure on fire scenario;
- The behavior of cable fire and electrical cabinet fire sources in confined and ventilated fire scenario.

To illustrate the work carried out in this project this paper presents a focus on some outstanding results from the PRISME experimental campaigns as:

- The effect of oxygen depletion on fuel mass loss rate (from PRISME Source and PRISME Door fire tests);

- The relative effects of heat and mass transfers from fire room to adjacent room(from PRISME Door and PRISME Leak fire tests);
- The effect of ventilation flow rate on the velocity profiles from fire room to neighboring compartments (from PRISME Integral and Support fire tests).

The experimental outcomes obtained during this PRISME Project provides a better understanding and an increase of knowledge in fire development in confined and ventilated large-scale compartments representative of nuclear area. Moreover, they also have as a result the improvement of fire modeling and a huge experimental database used to validate fire safety codes (based zone modeling, lumped parameter approach and CFD).

Nevertheless, as noted by the PRISME partners, some topics in fire safety for nuclear facilities need to be investigated further as:

- Smoke and hot gas propagation through a horizontal opening between two superposed compartments. This type of smoke flow is poorly validated in large-scale facilities controlled by ventilation systems and remains a challenging task for computer codes [8], [12];
- Fire spreading on a real fire source such as cable trays and electrical cabinets and fire propagation from one fire source to another one. These scenarios have been identified by most of the PRISME partners because they currently occur in control panels and switchgear rooms of nuclear power plants;
- Fire extinguishing with studying the performance of various extinguishing systems. Modeling the fire suppression system is still a great challenge and an experimental database of full-scale fire tests representative of typical scenarios in nuclear power plants is needed.

These research topics will be covered by the OECD PRISME-2 Project. This project has started in mid-2011 and will continue for five years.

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APPENDIX 1: Summary of fire tests (Source, Door) in the PRISME Programme in the DIVA Facility

Test Name	Facility	Fuel	Pool Area [m²]	Initial Fuel [kg]	Fuel Burned [kg]	Fire Extinction	Fire Duration [s]	Air Inlet Location	Ventilation Flow Rate [m^3/h]	Number of Rooms	Comments
PRS-SI-D1	DIVA	HTP [(1)]	0.4	15.0	13.2	O_2	3190	high	560	1	
PRS-SI-D2	DIVA	HTP	0.4	15.7	15.7	fuel	2580	high	1020	1	
PRS-SI-D3	DIVA	HTP	0.4	16.0	2.9	O_2	360	high	180	1	
PRS-SI-D4	DIVA	HTP	0.4	15.7	13.3	O_2	2895	high	565	1	
PRS-SI-D5	DIVA	HTP	0.2	7.2	7.2	fuel	2552	high	555	1	
PRS-SI-D5a	DIVA	HTP	0.2	7.8	4.5	O_2	1978	high	190	1	
PRS-SI-D6	DIVA	HTP	0.4	16.0	12.0	O_2	2495	low	560	1	
PRS-SI-D6a	DIVA	HTP	0.4	15.8	3.4	O_2	575	low	200	1	
PRS-D1	DIVA	HTP	0.4	14.9	6.8	O_2	883	high	0 m	2	One doorway, PVC rods + cables
PRS-D2	DIVA	HTP	0.4	17.7	9.1	O_2	1410	high	180	2	One doorway, PVC rods
PRS-D3	DIVA	HTP	0.4	16.3	16.3	fuel	1910	high	560	2	One doorway, PVC rods
PRS-D4	DIVA	HTP	0.4	15.1	15.1	fuel	1160	high	1030	2	One doorway, PVC rods + cables
PRS-D5	DIVA	HTP	1.0	15.9	15.9	fuel	1310	high	560	2	One doorway PVC rods + cables
PRS-D6 [(2)]	DIVA	HTP	1.0	25.1	13.0	O_2	420	high	560	3	Two doorways, PVC rods + cables

[(1)] HTP = Hydrogenated Tetra-Propylene ($C_{12}H_{26}$); [(2)] PRS-D6: N_2 injection in fire room at 405 s after ignition because of safety reasons

APPENDIX 2: Summary of Fire Tests (Leak, Integral) during the PRISME Programme in the DIVA Facility

Test Name	Facility	Fuel	Pool Area [m²]	Initial Fuel [kg]	Fuel Burned [kg]	Fire Extinction	Fire Duration [s]	Air Inlet Location	Ventilation Flow Rate [m^3/h]	Number of Rooms	Comments
PRS-LK1	DIVA	HTP	0.6	17.5	17.5	fuel	1120	high	1760	2	Two circular ducts
PRS-LK2	DIVA	HTP	0.6	18.1	18.1	fuel	1180	high	1760	2	Narrow vertical slot
PRS-LK3	DIVA	HTP	0.6	17.6	17.6	fuel	1120	high	1760	2	Real fire door
PRS-LK4	DIVA	HTP	0.6	17.7	15.2	O_2	1000	high	1760	2	Real fire door + internal duct
PRS-INT1	DIVA	HTP	1.0	98.9	80.9	O_2	2035	high	3100 (L1)	3	Two doorways
PRS-INT2	DIVA	HTP	1.0	52.3	23.6	O_2	622	high	3100 (L1)	3	Two doorways, sprinkler actuation
PRS-INT3	DIVA	cables	- (1)		4.7	fuel (2)	1500	high	3100 (L1)	3	Two doorways
PRS-INT4	DIVA	HTP	1.0	52.1	52.1	fuel	1610	high	2500 (L1) + 600 (L0)	4	Three doorways
PRS-INT5	DIVA	HTP	1.0	53.5	26.0	O_2	750	high	2500 (L1) + 600 (L0)	4	Three doorways, dampers
PRS-INT6	DIVA	electrical cabinet	- (3)	44.0	35.0	O_2	1950	high	2500 (L1) + 600 (L0)	4	Three doorways, dampers

(1) 4 cable trays of 3 m in length

(2) Self-extinction of fire (limited flame propagation)

(3) Electrical cabinet dimensions: 1.2 m in width, 2.0 m in height, 0.6 m in depth.

APPENDIX 3: Summary of Support Fire Tests (Source, CAB) in the PRISME Programme under SATURNE Calorimeter

Test Name	Facility	Fuel	Pool Area [m²]	Initial Fuel [kg]	Fuel Burned [kg]	Fire Extinc-tion	Fire Dura-tion [s]	Air Inlet Location	Ventilation Flow Rate [m^3/h]	Number of Rooms	Comments
PRS-SI-S1	hood	HTP	0.2	7.8	7.8	fuel	3190	-	open	-	
PRS-SI-S2	hood	HTP	0.2	7.7	7.7	fuel	1510	-	open	-	
PRS-SI-S3	hood	HTP	0.4	14.9	14.9	fuel	1295	-	open	-	
PRS-SI-S4	hood	HTP	0.4	15.1	15.1	fuel	1350	-	open	-	
PRS-SI-S5	hood	HTP	0.1	3.7	3.7	fuel	1945	-	open	-	
PRS-SI-S6	hood	HTP	0.1	3.8	3.8	fuel	1940	-	open	-	
PRS-SI-S7	hood	HTP	0.1	6.1	6.1	fuel	2928	-	open	-	
PRS-CAB-1	hood	cables	- (1)	≈ 47.0 (2)	19.7	fuel	4200	-	open	-	First cable test
PRS-CAB-2	hood	cables	- (1)	≈ 47.0 (2)	28.8	fuel	3300	-	open	-	Improvement of the fire propagation along the cables
PRS-CAB-3	hood	cables	- (1)	≈ 47.0 (2)	27.2	fuel	3390	-	open	-	Repeatability of PRS-CAB-2

(1) 4 cable trays of 3 m in length

(2) In the CAB fire tests, the total mass of PVC power cables was about 173 kg including 104 kg of copper wire and 69 kg of plastic materials. From latter materials, only 47 kg of plastic materials (mainly PVC and Polyethylene including in both additive materials as $CaCO_3$) could be ignited during the fire tests.

In addition to the PRS-INT2 fire test (including activation of sprinklers), a hydrodynamic characterization of the droplets flow was performed for one sprinkler head. These tests measured the water flow rate spatial distribution on the floor for five heights of sprinkler nozzle (1.0, 1.5, 2.0, 2.5 and 3.0 m) for average flow conditions of 42.6 l/min and 2.65 bars.

During the PRISME campaigns, the whole DIVA facility was checked just before the first fire test by carrying out one or more simple fire experiments by mean of a gas burner (PYROS) or a small liquid pool fire. Some of these additional experiments can be used to investigate some research topics (as the forced vs. natural flow through doorway in the PRISME Integral campaign [11])

FIRE CODE BENCHMARK ACTIVITIES WITHIN THE INTERNATIONAL RESEARCH PROJECT PRISME – DISCUSSION ON METRICS USED FOR VALIDATION AND ON SENSITIVITY ANALYSIS

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ABSTRACT

The aim of the present paper is to provide an overview of fire code benchmark activities conducted in connection with the international research project OECD PRISME.

The purpose of the first study was to quantify comparisons between computational results (from field and zone models) and measurements in the case of a full scale pool fire in a confined and mechanically ventilated compartment representative of nuclear plants. Different metrics operators and their ability to perform quantitative comparisons have been studied. The results underline the significance to use more than one metrics for performing an exhaustive validation process of a fire model.

The second study, which deals with a sensitivity analysis of fire models, has quantified the influence of some factors characterizing the fuel, the compartment or the ventilation network on relevant responses for fire safety studies. The main aim of this study was then to define the significant factors on these responses in order to enhance fire simulation accuracy.

INTRODUCTION

The PRISME Project (French acronym for "Fire Propagation in Elementary Multi-room Scenarios") mainly aims at studying smoke and hot gases propagation in full scale, well-confined and mechanically ventilated fire compartments. In particular, the goals of the PRISME program are to understand and quantify, by means of an analytical approach, the propagation mechanisms of smoke and heat from a fire compartment towards one or several adjacent compartments in scenarios representative of nuclear plants. This also covers the feedback effects of vitiated air due to smoke and combustion products on the fire source itself. This work is carried out in the framework of an international OECD (Organization for Economic Co-operation and Development) / NEA (Nuclear Energy Agency) / CSNI (Committee on the Safety of Nuclear Installations) project with partners from twelve member countries: Belgium (BEL V and SUEZ-TRACTEBEL), Canada (AECL), Finland (STUK and VTT), France (IRSN, EdF and DGA), Germany (GRS, iBMB, BfS), Japan (JNES), South Korea (a consortium represented by KINS), the Netherlands (VROM-KFD, NRG), Spain (CSN), Sweden (RINGHALS), UK (HSE) and USA (NRC).

The first step of this research program, called PRISME-SOURCE, aimed at characterizing the burning rate of a hydrogenated tetra-propylene (HTP) pool fire in a large-scale compartment in comparison to the same fire in a free atmosphere. For this purpose, experimental [1], [2] and theoretical investigations [2], [4] on the effects of air vitiation on the pool fire mass loss rate in a confined compartment have been conducted. In connection with this experimental research program, the large amount of measured data is used to perform numerical benchmarks with the aim to validate the fire models used by the participants of this international program. The objective of this paper is thus to present the work performed by the PRISME Benchmarking Group (PBG), which is composed of the partners involved in the OECD PRISME Project.

The next section presents the fire experiment describing briefly the facility, the instrumentation and the pool fire. Then, the two benchmark exercises performed by PBG are presented and the main results are developed. A first presentation of these exercises can be found in [5].

THE PRISME-SOURCE TEST

The PRS-SI-D1 fire experiment was conducted by IRSN in the DIVA facility (Figure 1) which is composed of four rooms and a corridor supplied by an industrial ventilation network. A single room, shown in Figure 2, which is 6 m × 5 m × 4 m in size, was used for this fire test and gave a total fire compartment volume of 120 m^3.

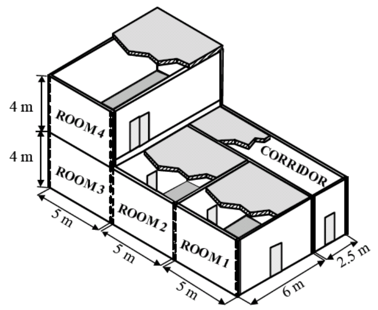

Figure 1 Schematic representation of the PRISME DIVA facility

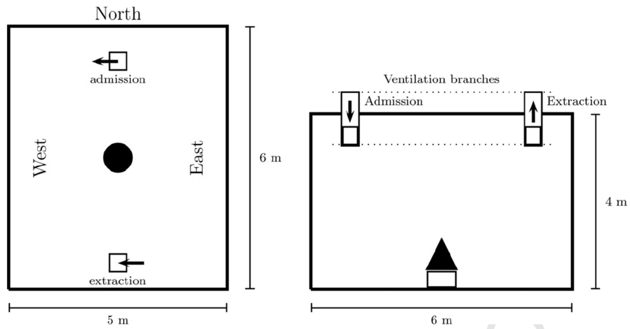

Figure 2 Top view (left) and side view (right) of the fire compartment

The ceiling, the floor and the compartment walls were made in concrete material with a thickness of 30 cm and the ceiling was covered by an insulation material of 5 cm thick. The mechanical ventilation system of the fire compartment was composed of inlet and exhaust branches of cross section 0.18 m^2, located in the upper part of the room. The ventilation flow rate before fire ignition was fixed to 560 m^3/h in the compartment.

The fuel container made in carbon steel was a flat cylinder pool of area 0.4 m^2. It was located in the center of the compartment and placed 40 cm above the floor on a weight scale system. A commercial solvent, hydrogenated tetrapropylene (HTP) was used for the liquid fuel with a total mass of 15 kg. The fire extinguished about 3200 s after ignition due to the lack of oxygen in the fire compartment. The measured mass loss rate is given in Figure 3. A more detailed description of the PRISME-Source program and of the PRS-SI-D1 test (fuel properties, concrete and insulated walls description, etc.) can be found in [1] and [6].

Figure 4 outlines some experimental results which have been selected for the two benchmarks because of their relevance in fire safety engineering: the average gas temperature, determined from the measurement with a vertical thermocouple tree and the oxygen concentration in the compartment. Other safety relevant quantities such as the wall temperature, the radiative wall heat flux or the pressure level inside the fire compartment have also been selected for the benchmark exercises; for more additional details, the reader can refer to [6].

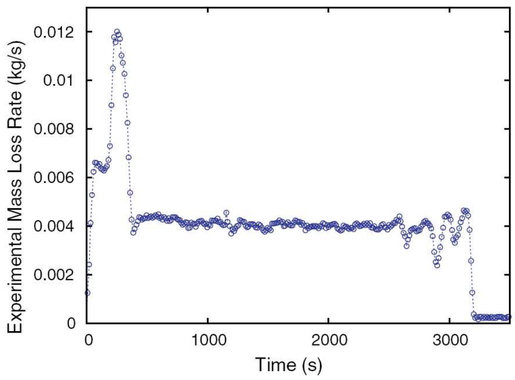

Figure 3 Time evolution of the experimental mass loss rate of PRS-SI-D1, from [6]

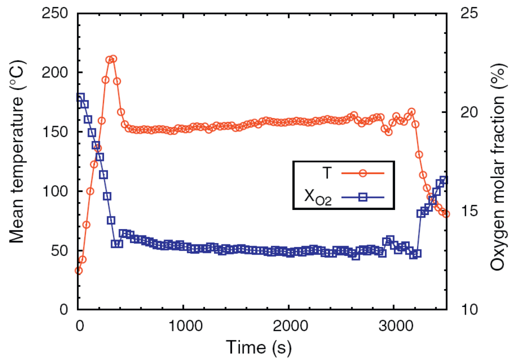

Figure 4 Weighted average value of the thermocouple temperature measurements, from [6]

BENCHMARK EXERCISE #1

Presentation and Objective

The quantification of differences between computational results and measurements was the main objective of this first benchmark. According to [6], the standards of ASTM [7] and ISO [8] provide several methods for the comparison of experimental and numerical results. The choice of the method depends on the characteristics of the data:

– For single-points comparisons (as a peak of temperature or overpressure in a fire compartment), the quantitative comparison may be expressed using the absolute or relative difference.
– For a steady-state or quasi-steady-state regime a comparison between the numerical results and the experimental data may be expressed as the average of the absolute or relative difference.
– For time dependent values, the numerical results may be compared to measured quantities over all the fire scenario duration.

In total, 17 simulations were carried out for this benchmark. 10 simulations were made with fire field models and 7 with zone models. These simulations are, respectively, hereafter abbreviated as CFD and ZC runs. Among the CFD simulations, 8 simulations were carried out with the FDS software developed by the NIST in co-operation with VTT. The two other simulations were conducted with the ISIS code developed by IRSN and the SAFIR code developed by DGA and the French laboratory IUSTI. Fire models used for the zone simulations are more diversified. Three runs were made with the CFAST software developed by the NIST, whereas the other used the COCOSYS, MAGIC, OEIL and SYLVIA software, respectively developed by GRS, EdF, DGA and IRSN. Code names and version numbers of the 17 simulations are given in Table 1.

The main features of the field and zone models are given in [6].

Table 1 Code name and version number of simulations from [6]

Simulation	Code	Version
Field models		
CFD 1 and 3	FDS	4.0.7
CFD 2, 4 and 5	FDS	4
CFD 6	FDS	5.4
CFD 7	FDS	5.1.4
CFD 8	FDS	5.2.0
CFD 9	SAFIR	–
CFD 10	ISIS	1.3.0
Zone models		
ZC 1	CFAST	5
ZC 2 and 3	CFAST	6
ZC 4	OEIL	1.5.1
ZC 5	MAGIC	4.1.2
ZC 6	SYLVIA	1.3.2
ZC 7	COCOSYS	2.4 dev

Each simulation has used the same thermo-physical properties for insulation, concrete walls and fuel. The fuel mass loss rate boundary condition was also imposed to participants (Figure 3) but no indication was given on the ventilation system modeling. Participants may have imposed the experimental mass flow rate at intake, exhaust or both or have performed a complete coupling between the ventilation branches and the compartment pressure without imposing the ventilation flow rates.

Partial Results

The experimental and simulated mean temperatures are shown in Figure 5, field simulations in Figure 5-a and zone simulations in Figure 5-b. Obviously, most of the results predicted by both field and zone models are in good agreement with experimental data. However, most simulations, except two zone runs, underestimate the average measured temperature. The quantitative comparison is shown in Figure 6 where the local and global errors (relative to results in Celsius units) are reported for all the simulations. The local error refers to the peak of temperature and the initial reference state is taken into account in the determination of the two errors. The uncertainty measurement, called U_e, is also plotted in the two figures. A first analysis on metrics' results shows that only few models have a local error significantly greater than the uncertainty measurement whereas more than half of the results show a global error greater than this threshold. More precisely, the local error made by the run ZC 7 could be described as unsatisfactory but an examination of the global error shows a more acceptable result in comparison to the other runs. Conversely, field runs number 2, 4 and 7 as well as zone runs number 3, 4, 5 and 6 gives good local errors of less than the experimental uncertainty while the global errors could be around 15 and 20 %, respectively.

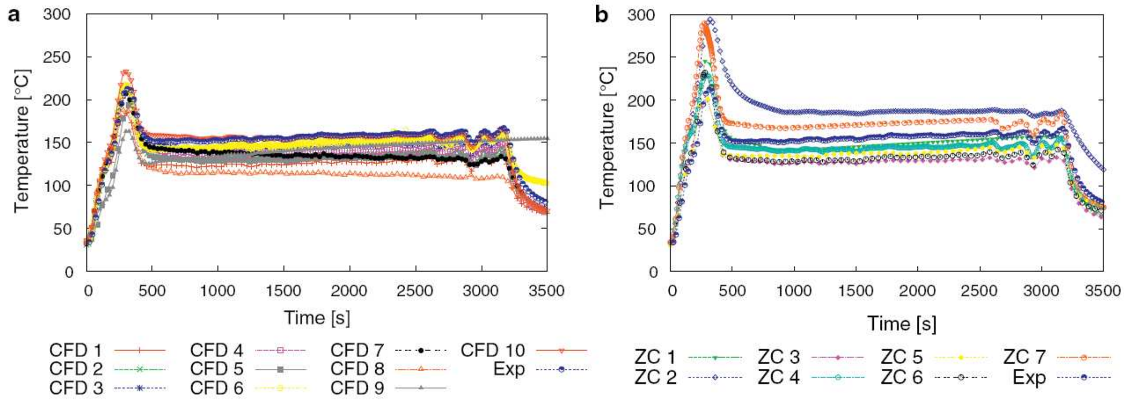

Figure 5 Comparison of measured and predicted mean gas temperature for field (a) and zone (b) simulations, from [6]

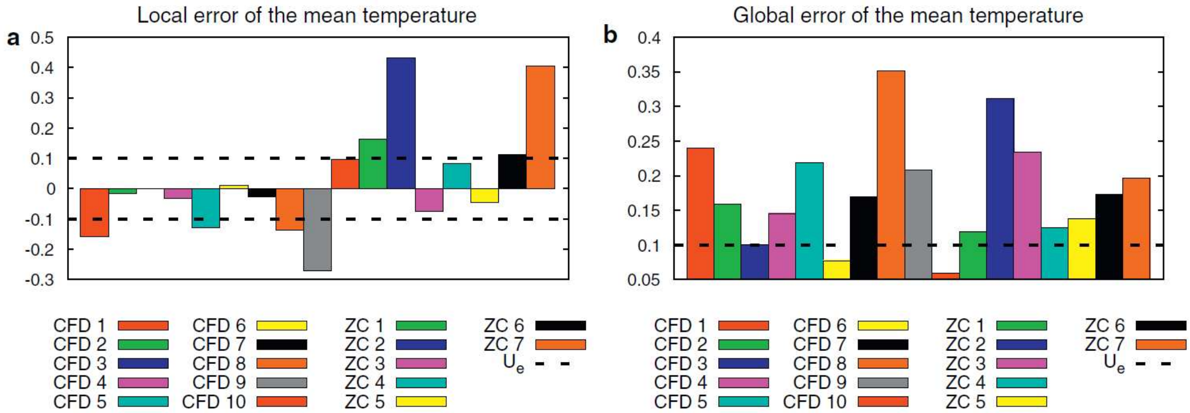

Figure 6 Local (a) and global (b) errors for the mean gas temperature, from [6]

Main Conclusion of the Benchmark Exercise #1

This numerical exercise has involved 17 participants using 8 fire simulation tools (3 CFD or field codes and 5 zone codes). The calculation was qualified as "open" (in opposition with blind calculations), therefore, wall and fuel properties were specified as well as the fuel burning rate, the ventilation conditions and test data. Despite this guidance, the so-called "user-effect" was important for both field and zone models. The main objective of this work was however to investigate the possibility of using metrics in a validation process of a real large scale fire scenario involving several participants with different fire simulation tools.

For the validation process, six quantities were compared during the whole fire duration: the gas temperature, the oxygen concentration, a wall temperature, the total heat flux to a wall, the compartment pressure and the ventilation flow rate at the inlet branch. Compared to the proposals of the literature, two metrics are used for quantifying the evaluation of the models. The first metric, also used by the U.S. NRC (Nuclear Regulatory Commission) and EPRI (Electric Power Research Institute) in the validation work of fire models [9], considers the relative difference of numerical and experimental results expressed in terms of difference between an extreme value and its baseline. The second metric, called the normalized Euclidean distance, considers the differences between computational results and measurements during all the fire duration. This metric behaves as a global error and gives an overview of code capabilities.

From the use of these metrics, applied to the gas temperature time-evolution but also to the other variables, it appears that it is important to consider more than one metric for an exhaustive validation process of computer codes.

BENCHMARK EXERCISE #2

Presentation

The second exercise presents a sensitivity analysis using fractional factorial design (FFD). Several field and zone computer codes have been used to study the influence of some factors characterizing either the fuel, or the compartment or the ventilation network on relevant responses for fire safety studies. More specifically, the effects of these factors on gas and wall temperatures, the concentration of oxygen in the room, total and radiative heat flux to the walls and the total pressure in the compartment were examined.

The purpose of a sensitivity analysis study is to measure the influence of one or more input variables of a mathematical model (such as computer codes) on some selected output variables. It is performed by varying the values of the inputs in order to quantify the effect of these changes on the considered outputs. In this process input variables are called factors and output variables are called responses, respectively noted as X and Y in this document. This kind of analysis needs the specification of the connection between inputs and outputs. In other words, the analyst has to choose a model to translate this connection and that will provide sensitivity measures to quantify the influence of each input. In most cases, a linear regression model is used but a second order or quadratic model is possible for certain specific applications. The unknowns of the model are the regression coefficients called β hereafter. The choice of the simulations, necessary to determine the regression coefficients of the model, is crucial. This choice is often achieved following the theory of Design of Experiment (DoE). It consists in discretizing the variation range of each input in order to collect a large amount of information with a limited number of simulations. Among classical DoE, one can mention Full Factorial Design with all input factors set at two levels each, called "high" and "low". This kind of DoE is composed of these two levels for all k input factors and thus requires 2^k runs. For this reason, when the number of factors is important or when the design is composed of three levels input factors, the full factorial design requires a lot of runs and therefore loses its efficiency. For example, a two-level design with six factors implies $2^6 = 64$ runs. In this case, a fractional factorial design is a more suitable choice and the solution is to only use an appropriate fraction of the full factorial design.

The most simple linear regression model with one predictor variable is expressed as:

$$Y = \beta_0 + \beta X + \varepsilon$$

where X is the input factor or the predictor variable, β_0 gives the value of Y when $X = 0$, β is the regression coefficient and ε represents the residual error of the model which is defined as the difference between the prediction obtained by the linear function and the value of Y observed. The coefficients β_0 and β are determined in such a way as to minimize the root mean square difference between Y and $Y' = \beta_0 + \beta X$.

In the case of two input factors, the linear model can be written as:

$$Y = \beta_0 + \beta_1 X_1 + \beta_2 X_2 + \beta_{12} X_{12} + \varepsilon$$

The relevant responses are related to gas temperature, oxygen concentration, wall temperature, radiative and total heat flux and total pressure. They are listed in Table 2.

Table 2 Output responses

Variable	Location	Comment
Temperature [K]	mean value at (3.75, 4.5, Z)	maximum value over time
	mean value at (3.75, 4.5, Z)	mean value t ∈ [1500 s, 2500 s]
Oxygen molar fraction	upper layer or (3.75, 1.5, 3.3)	minimum value over time
Wall temperature [K]	north wall (4.6, 2.6)	maximum value over time
Rad. heat flux [W/m^2]	north wall (4.6, 2.6)	maximum value over time
Total heat flux [W/m^2]	north wall (4.6, 2.6)	maximum value over time
Relative pressure [Pa]	global variable	mean value t ∈ [1500 s, 2500 s]
	global variable	maximum value over time
	global variable	minimum value over time

The most popular experimental designs are two-level designs and the range of the settings for input factors designates extreme values for concerned quantities. Table 3 describes the six input factors. It concerns the knowledge of the fuel mass loss rate, the radiative fraction of the combustible, wall characteristics as conductivity, heat capacity or emissivity and the ventilation flow rate of the compartment. The reference values generally used are also indicated in Table 3. The lower and upper values define an uncertainty band coming from an uncertainty measurement, a lack of knowledge or a variation of the concerned factor depending on the fire scenarios.

Table 3 Factors for the sensitivity analysis

Input Parameters	Reference Value	Lower Value	Upper Value
Mass loss rate (MLR) [kg/s]	expected value	-10 %	+ 10 %
Radiative fraction	0.35	0.30	0.40
Wall conductivity [W/m/K]	1.50	1.07	1.93
Wall heat capacity [J/kg/K]	1000	800	1200
Wall emissivity	0.7	0.5	0.9
Ventilation flow rate [m^3/h]	560	500	620

Table 4 presents the fractional factorial design (FFD) corresponding to the fire sensitivity analysis. Compared to the full design of 64 runs (2^6), the number of simulations has been reduced by factor 8. Each row of the table indicates the simulation number and the value for each factor.

Table 4 Description of the fractional experimental design involving eight simulations

RUN	Mass Loss Rate (MLR) [kg/s]	Radiative Fraction	Wall Conductivity [W/m/K]	Wall Heat Capacity [J/kg/K]	Wall Emissivity	Ventilation Flow Rate [m^3/h]
	$\dot{m}''$	χ_r	k_w	$C_{p,w}$	ε_w	Q_v
1	- 10 %	0.3	1.93	800	0.9	620
2	+ 10 %	0.3	1.07	800	0.5	620
3	- 10 %	0.4	1.07	800	0.9	500
4	+ 10 %	0.4	1.93	800	0.5	500
5	- 10 %	0.3	1.93	1200	0.5	500

RUN	Mass Loss Rate (MLR) [kg/s]	Radiative Fraction	Wall Conductivity [W/m/K]	Wall Heat Capacity [J/kg/K]	Wall Emissivity	Ventilation Flow Rate [m^3/h]
6	+ 10 %	0.3	1.07	1200	0.9	500
7	- 10 %	0.4	1.07	1200	0.5	620
8	+ 10 %	0.4	1.93	1200	0.9	620

Partial Results

As an example, we present hereafter only the detailed responses for the gas and wall temperatures. The complete study will be available in [10].

The CFD and zone codes which have been used for this benchmark and the organization which performed the simulations are the following:

- FDS: VTT,
- CFAST: JNES,
- COCOSYS: GRS,
- MAGIC: EdF,
- OEIL: DGA,
- SYLVIA: IRSN.

***Maximum of Mean Gas Temperature*:**
The sensitivity analysis of the six fire models concerning the maximum of the mean temperature is presented in Figure 7 (a). The same ranking factor is observed for the two most important factors: the fuel mass loss rate, which has a positive effect and the wall emissivity, with a negative effect. The radiative fraction has a significant negative effect for the zone model CFAST, SYLVIA and COCOSYS.

Average of Mean Gas Temperature:

The average value over the time t = 1500 s, t = 2500 s of the mean temperature is presented in Figure 7 (b). The same ranking factor is again found for this response. The fuel mass loss rate has a positive effect and the wall emissivity, capacity and conductivity are the three non-negligible factors which tend to diminish the gas temperature.

Maximum of Wall Temperature:

The results concerning the wall temperature is shown in Figure 7 (c). The wall properties as the conductivity and heat capacity have a negative effect whereas the fuel mass loss rate has a positive effect. All the fire models give the same trend. The results concerning the wall emissivity are considered not significant because of their low values.

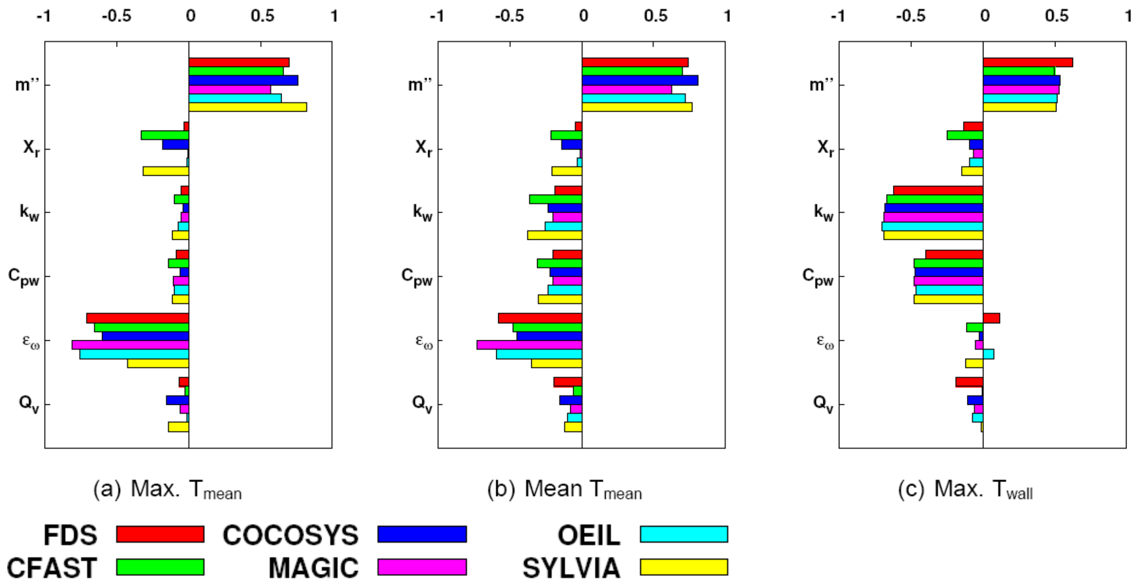

Figure 7 Sensitivity analysis for all field and zone models

Main Conclusions of the Benchmark Exercise #2

A sensitivity analysis was performed on several fire models using a fractional experimental design. The used codes are: FDS, CFAST, MAGIC, OEIL, SYLVIA and COCOSYS. The influence of six factors was tested on 9 responses including gas and wall temperature, oxygen concentration, wall heat flux and over or under pressure peak in the fire compartment. These responses were selected for their importance in fire safety studies. The considering factors are the fuel mass loss rate and radiative fraction, thermo-physical properties of the compartment (conductivity, heat capacity and emissivity of concrete walls) and the ventilation mass flow rate through the ventilation network.

Initially, this study has identified some users code error by comparing the values obtained for each factor of the sensitivity analysis. For this purpose, some issues have already been raised at the analytical working group (AWG) in a previous PRISME meeting. Globally, the ranking factor is identical for most numerical tools. This major result helps to quantify the importance of the different factors for each response with a good confidence. For this purpose, a qualitative three-color coding scheme is used to highlight the most important factors for the considering responses and Table 5 gives an overview of the performed analysis. The results show that the main factor for each response is the fuel mass loss rate. The oxygen concentration seems to be affected by the ventilation mass flow rate, whereas the thermo-physical quantities such as temperature, heat flux or pressure in the room are primarily affected by the wall emissivity and by the fuel radiative fraction. These results are both original and very important for the fire community allowing the fact that they give some orientation for future research with more relevance and thus contribute to the improvement of databases, mandatory for fire models.

Initially, different methods to generate samples were compared. The effects of factors are studied in the case of a Monte Carlo method, a full and a fractional factorial design (FD). For each response, the methods used give similar results with the same ranking factor. This result is also important both for experimental studies but also for numerical simulations performed with fire field models according to the fact that fractional FD with 8 runs provides the same information as a Monte Carlo method with 200 runs or a full FD with 64 runs. Since this drastically reduces the number of runs to perform, fractional

FD makes sensitivity analysis easier for industrial applications. More details will be available in [10] in the near future.

Table 5 Qualitative overview of the most important factors for the selected responses

Responses	Mass Loss Rate (MLR)	Radiative Fraction	Wall Conductivity	Wall Heat Capacity	Wall Emissivity	Ventilation Flow Rate
Maximum mean temperature						
Average mean temperature						
Wall temperature						
Oxygen concentration						
Wall total heat flux						

Red: the factor has a significant influence (> 0.8) for most of the models
Orange: the factor has a relative influence (≈ 0.5) for most or for some models;
Green: the factor has a small influence (< 0.25) for most of the models

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A PREDICTIVE PYROLYSIS MODEL FOR LIQUID POOL FIRES INCLUDING RADIATION FEEDBACK FROM HOT SOOT LAYER IN COCOSYS

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ABSTRACT

Fire simulations for safety analyses of nuclear power plants have gained more and more significance. A crucial point for fire simulation in a confined compartment is the estimation of the pyrolysis rate, which usually deviates from the one measured in open atmosphere conditions. Phenomenologically speaking, the pyrolysis rate in confined compartments depends on the one hand on the availability of oxygen and, on the other hand, on the room temperature, since radiation feedback from the walls and the hot soot layer enhances the evaporation.

For liquid pool fires, the oxygen depletion effect is described by the correlation of Peatross and Beyler. To assess the temperature effect as well, this paper outlines a model for determination of the radiation feedback from the upper soot layer. The radiation from the walls is not considered due to their often comparably low temperatures. Both the Peatross-Beyler-correlation and radiation model were implemented in the lumped parameter code COCOSYS. Thus, it is possible with COCOSYS to predict the pyrolysis rate of a liquid pool fire in a confined compartment if the open environment pyrolysis rate is provided.

The model was validated on three OECD/NEA/CSNI PRISME experiments in the DIVA facility of the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN).

INTRODUCTION

Fires can jeopardize the entire safety of a nuclear power plant. Hence, much effort is used for the further development of fire simulation tools.

A crucial point for fire simulation in a confined compartment is the estimation of the pyrolysis rate (i.e. the rate of vaporized fuel mass) respectively the burning rate and its development in time. For computer code validation on available experimental data, the experimentally obtained pyrolysis rate is usually given as user input into the code. It is evaluated whether the code is capable to manage the combustion product generation and distribution as well as the thermodynamics of the fire compartment and its surrounding area in a suitable way.

Especially for blind pre-test calculations or real applications, it is important to predict pyrolysis rates. Since, related to a large range of uncertainty due to the complexity of the processes taking place during combustion, at least the prediction within some uncertainty boundaries is desired.

Liquid pool fires represent comparably simple fires, but are relevant for many industrial applications, hence they are widely used for fire experiments and investigations, e.g. during the experimental OECD PRISME (French: Propagation d'un incendie pour des scénarios multi-locaux élémentaires*)* program carried out by IRSN (Institut de Radioprotection et de Sûreté Nucléaire) in the DIVA facility in Cadarache [1].

For a liquid pool fire in open atmosphere, Babrauskas [2] derived a correlation for the steady state mass loss rate $\dot{m}_{PYR,o}$:

$$\dot{m}_{PYR,o} = \dot{m}_{MAX} \cdot (1 - e^{-k\beta D}) \qquad \textit{for pool diameter } D > 0.2m \qquad (1)$$

It only requires the maximum burning rate $\dot{m}_{MAX}$ [kg/m²s] in an open atmosphere, the absorption-extinction coefficient k [m^{-1}] and a mean beam length corrector β, which are pre-known material properties of the fuel.

The development of a fire in a confined compartment phenomenologically depends mainly on the oxygen concentration and the room temperature. The lack of oxygen lowers the pyrolysis rate, while high temperatures in the upper soot layer (or of the structure surfaces) can enhance the combustion by radiation to the pool surface.

In this paper, two correlations are given for estimating the effect of oxygen depletion and of the radiation feedback from the hot soot layer below the ceiling on a liquid pool fire. The correlations were implemented in the lumped parameter code COCOSYS (developed by GRS) and validated on three selected OECD PRISME experiments.

OUTLINE OF THE MODEL

The mass of fuel which is pyrolized from the fuel source is determined by the heat balance of the fuel pool. The mass of pyrolized fuel $\dot{m}_{PYR}$ is given by [3]

$$\dot{m}_{PYR} \cdot \Delta i \cdot A_{surf} = \dot{Q}_{conv,fl} + \dot{Q}_{rad,fl} - \dot{Q}_{loss,fuel} + \dot{Q}_{rad,ext} \qquad (2)$$

with Δi being the heat of vaporization of fuel and A_{surf} the fuel surface area. Whether the pyrolized (i.e. vaporized) fuel is burnt further on, depends on the amount and accessibility of oxygen.

The given heat contributions are:

1. Convective heat transfer from the fire flame into the fuel structure $\dot{Q}_{conv,fl}$,
2. Radiative heat transfer from the fire into the fuel structure $\dot{Q}_{rad,fl}$,
3. Heat losses of the fuel structure not contributing to vaporization $\dot{Q}_{loss,fuel}$,
4. External radiation heat flux (e.g. back radiation from hot soot layer) onto the fuel structure $\dot{Q}_{rad,ext}$. This contribution is only significant at very high room temperatures (above the fuel's evaporation temperature) and hence can be mostly neglected.

Nasr et al. [4] give a closed formula for all these four heat flux contributions in (2) using the term of flame temperature given some necessary assumptions and empirical correlations. He successfully validated the resulting formula on the PRISME LEAK and DOOR tests.

PEATROSS-BEYLER-CORRELATION

The effect of the oxygen depletion on the pyrolysis rate in a confined compartment $\dot{m}_{PYR}$ is described by the correlation of Peatross and Beyler [5]

$$\dot{m}_{PYR} = \dot{m}_{PYR,o} \cdot \left[(1+\alpha) \cdot \frac{c_{O2}}{c_{O2,o}} - \alpha \right] \qquad (3)$$

with c_{O2} being the oxygen concentration in vol% in the lower layer and $c_{O2,o}$ = 21 vol% the oxygen concentration in open atmosphere. It was confirmed in the PRISME SOURCE [6] and DOOR experiments [7] for α = 1.1 (which is also the value proposed by [5]).

The Peatross-Beyler-correlation completely accounts for the convective and radiative heat transfer from the flame to the pool surface $\dot{Q}_{conv,fl} + \dot{Q}_{rad,fl}$ in equation (2) [8].

While the Peatross-Beyler-correlation could entirely describe the pyrolysis rate for the "cold" SOURCE and DOOR tests[1], it does not hold true for the LEAK and INTEGRAL tests (Figure 1), which featured very high gas temperatures in the fire compartment - several hundred K higher than the fuel's evaporation temperature. The high gas temperatures induce radiation feedback from the hot upper soot layer onto the pool surface enhancing the pyrolysis of fuel. This process is not accounted for by the Peatross-Beyler-correlation.

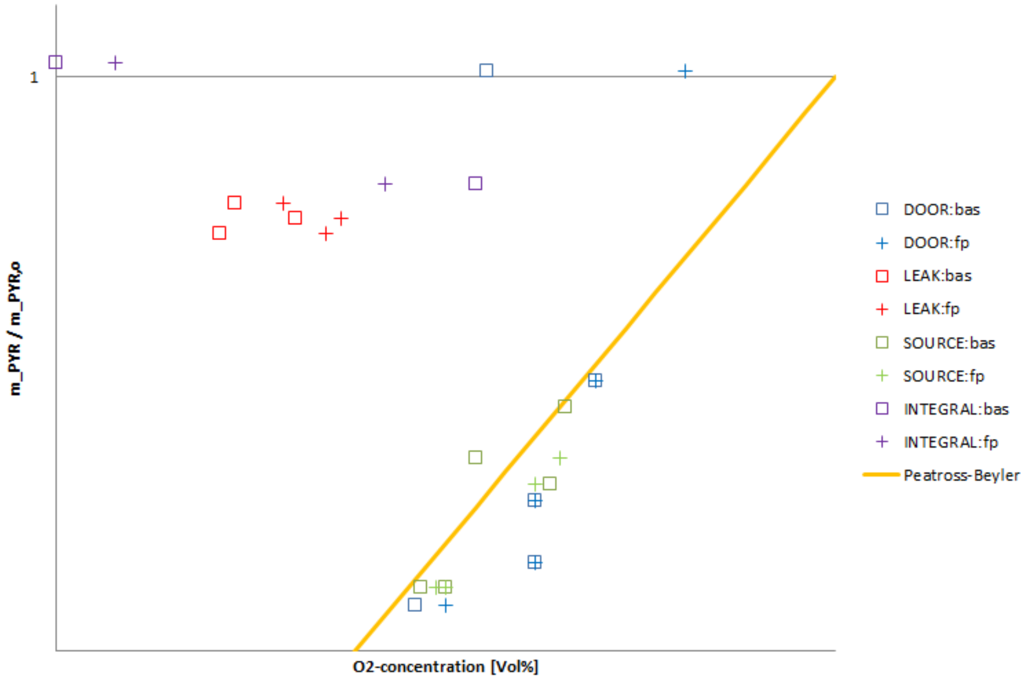


Figure 1 Experimental steady state pyrolysis rates observed in OECD PRISME experiments (normalized by open environment rate) in dependence on oxygen concentration in the lower layer (two measurement positions)

RADIATION FEEDBACK

As a first approximation – when assuming a homogeneous hot soot layer in the upper part of the fire compartment – the radiation feedback from this hot gas layer can be determined by

$$\dot{Q}_{rad,ext} = \Phi \cdot \varepsilon_{upp} \cdot \varepsilon_{fuel} \cdot \sigma \cdot A_{surf} \cdot \left(T_{upp}^4 - T_{evap}^4\right)$$

with

- Emissivity of the fuel surface ε_{fuel} and the upper soot layer ε_{upp}.,
- View factor between fuel surface and upper soot layer Φ,
- Mean temperature of the upper soot layer T_{upp} [K],
- Evaporation temperature T_{evap} [K],
- Stefan-Boltzmann-constant $\sigma = 5.67 \cdot 10^{-8} \frac{W}{m^2 K^4}$.

[1] One PRISME DOOR test showed full open environment rate in spite of some oxygen depletion (Figure 1) due to a blowing effect which seems to have been induced by the high ventilation rate ($8h^{-1}$) [7].

If the wall surface temperatures exceed T_{evap}, their thermal radiation has to be considered as well. This did not happen in the appropriate PRISME experiments and was therefore not further examined.

The presented equation turns out to be inadequate since it triggers a self-supporting process: An increase in gas temperature (e.g. during ignition) automatically leads to a fast increase in the pyrolysis rate (due to the power of 4 in T_{upp}^4) which in turn enhances the gas temperatures. Only the decrease of the pyrolysis rate due to oxygen depletion might control this self-supporting circle, but this process takes time, is linear and does not work below 11 vol% (Peatross-Beyler).

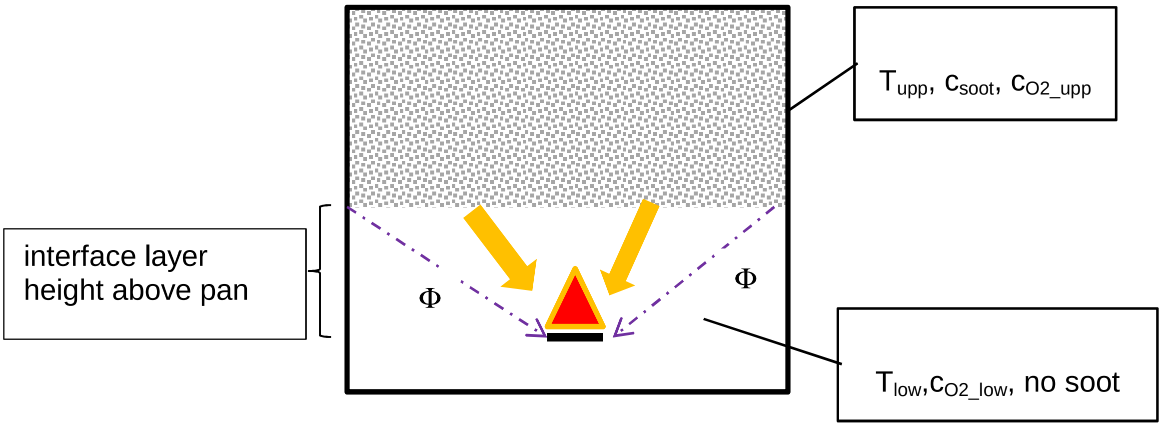


Figure 2 In a first view: Radiative heat flux from upper soot layer

Actually, the radiation is partly adsorbed on its path from the soot layer to the fuel, which limits the final radiative heat flux onto the fuel surface. When dividing the fire compartment in several layers, the emitted radiative heat of the k'th layer towards the fuel surface is (Figure 3)

$$\dot{Q}_{Rad,k} = \Phi_k \cdot \varepsilon_{soot,k} \cdot \varepsilon_{fuel} \cdot \sigma \cdot A_{pan} \cdot T_k^4.$$

There are several possible ways to determine the layer emissivity $\varepsilon_{soot,k}$ which all give – according to Lambert-Beer law - an exponential dependence on the soot concentration $c_{Soot,k}$ and the height Δh_k of the considered layer [9]:

$$\varepsilon_{soot,k} = 1 - \exp(-B \cdot \Delta h_k \cdot c_{Soot,k}) \quad (4)$$

In each layer below, a part of the emitted radiative heat is adsorbed – depending on the path length through the adsorbing layer. The mean path length might be longer than just the vertical height of the layer.

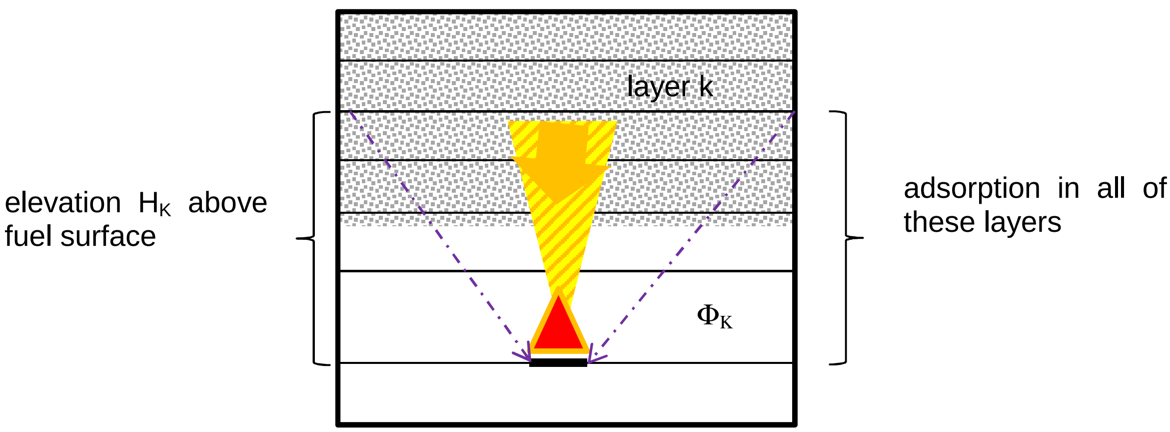


Figure 3 Detailed view: Radiation of layer k is adsorbed in the layers below

Assuming a uniform distribution of weight over the entire surfaces[2], the mean beam length l_H of a path from a square A to a parallel square B (in distance H) is (Figure 4)

$$l_H = \sqrt{H^2 + \tfrac{1}{6}(A+B)}.$$

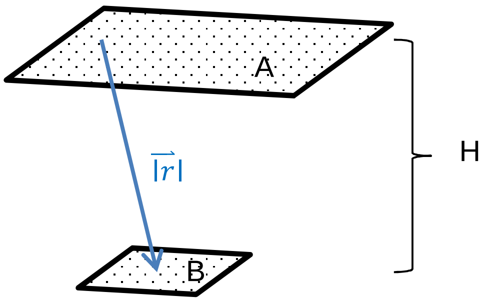


Figure 4 Determination of the mean beam length

Hence from the initially emitted radiation $\dot{Q}_{Rad,k}$, in each lower layer j

$$\dot{Q}_{adsorbed,j} = \left[1 - exp\left(-B \cdot \frac{\Delta h_j}{H_k} \cdot l_{H_k} \cdot c_{Soot,j}\right)\right] \cdot \dot{Q}_{Rad,k}$$

is adsorbed and only the part

$$\left[\prod_{j=1}^{k-1} exp(-B \cdot \tfrac{\Delta h_j}{H_k} \cdot l_{H_k} \cdot c_{Soot,j})\right] \cdot \dot{Q}_{Rad,k} = \left[\prod_{j=1}^{k-1} (1 - \varepsilon_{soot,j})^{\frac{l_{H_k}}{H_k}}\right] \cdot \dot{Q}_{Rad,k}$$

finally reaches the fuel surface (j = 1 is the layer directly above the pan).
In total, the following equation is valid:

$$\dot{Q}_{rad,ext} = \varepsilon_{fuel} \cdot \sigma \cdot A_{pan} \cdot \sum_k \left[\prod_{j=1}^{k-1} (1 - \varepsilon_{soot,j})^{\frac{l_{H_k}}{H_k}}\right] \Phi_k \cdot \varepsilon_{soot,k} \cdot \left(T_k^4 - T_{evap}^4\right) \qquad (5).$$

With increasing mass loss rate, the soot concentration rises and the exponential term $\prod_{j=1}^{k-1} (1 - \varepsilon_{soot,j})^{\frac{l_{H_k}}{H_k}}$ limits the gas temperature enhanced radiation increase.
Nasr et al. [4] determine the radiative heat flux using the flame emissivityε_f:

$$\dot{Q}_{rad,ext} = \sigma \cdot A_{pan} \cdot (1 - \varepsilon_f) \cdot (T_g^4 - T_{Surf}^4).$$

Since the flame emissivity is usually taken from an empirical correlation such as (de Ris [10])

$$\varepsilon_f = 1 - exp(-809 \cdot D \cdot f_v \cdot T_f)$$

with f_v soot volume fraction and T_f flame temperature, Nasr's approach features a similar control of the radiation feedback by the same functional dependencies for soot concentration and gas temperature.

[2] This assumption is not true for the paths from layers which are at low elevation above the fuel surface. The mean beam length is overestimated in these cases, but this overestimation turned out to be of minor importance in the simulation of PRISME experiments.

IMPLEMENTATION OF THE MODEL IN COCOSYS

The containment code system COCOSYS has been developed by GRS for the comprehensive simulation of design basis and severe accidents in light-water reactor containments [11]. Most of the models inside COCOSYS are based on a lumped parameter (LP) concept.

The compartments of the power plant, test facility or other building type to be analyzed have to be subdivided into control volumes, which are connected by so-called junctions. The thermodynamic state of a control volume is defined by its temperature(s) and masses of the specified components. Here the mass and energy balances are solved. The momentum of the flow between the compartments is not balanced. For walls (structures) a one-dimensional heat conduction equation is solved. For simulating oil and cable fires, pyrolysis and burning models have been implemented [12] and, in case of incomplete combustion, the production of soot is calculated according to user given soot factor and chemical soot composition. The soot aerosol particles are grouped in several size classes and transport, agglomeration and deposition are simulated. According to the specified radiative fraction, part of the heat generated by the combustion is transferred directly to the wall surfaces (considering the adsorption factor of walls). The required view factors between control volumes and wall surfaces are pre-calculated by using a Monte Carlo simulation considering the 3D-geometry of the problem.

PEATROSS-BEYLER-CORRELATION

For a calculation of combustion in a confined ventilated compartment using the described pyrolysis model, the user has to provide the time dependent pyrolysis rate of the desired fuel source observed under free environment conditions. This rate can be estimated using the Babrauskas correlation (1).

The actual pyrolysis rate in the simulation is calculated with the Peatross-Beyler-correlation (3) at each time step using the averaged oxygen concentration of the entire lower layer below the interface layer height.

Alternatively, the averaged oxygen concentration of two user given control volumes can be used. These reference control volumes for the oxygen concentration shall be representative for the "fresh oxygen supply" of the lower layer, which reaches the fuel surface and hence should be chosen carefully. In case of a conjectured draught (e.g. through a doorway) towards the fire, these control volumes should be located in there.

The oxygen concentration is furthermore smoothed according to

$$\tilde{c}_{O2} = c^0_{O2} + \left(c_{O2} - c^0_{O2}\right) e^{\frac{t-t_0}{\tau}}$$

with the O_2-concentration c^0_{O2} at the previous time step t_0 and the user given smoothing time τ (recommended: τ = 20 s).

Hence, the user given pyrolysis rate might be reduced due to oxygen depletion leading to a "delay" of mass loss compared to the open fire. The amount of "delayed mass" is accounted for and added whenever possible[3].

[3] There is a user given maximal pyrolysis rate which must not be exceeded (usually open environment steady state rate).

RADIATION FEEDBACK

The vaporized fuel according to the radiation feedback (5) is added to the pyrolysis rate[4]. Only contributions from layers above the interface layer height (Figure 2) and with gas temperatures exceeding the evaporation temperature of the fuel are taken into account. The emissivity of each soot layer is calculated at each time step by summing up all soot size classes i = 1 … m according to [9] by

$$\varepsilon_{soot,k} = 1 - \exp\left(-a_v \cdot \Delta h_k \cdot \tfrac{1}{4}\pi \cdot \textstyle\sum_{i=1}^{m}\left(n_{soot,i} \cdot d_i^2\right)\right) \qquad (6)$$

using the soot particle number concentration n_{soot} and the geometric mean diameter of the respective size class d_i. The emission coefficient a_v for coal and ash "particles" varies from 0.8 to 1 [9].

A list of view factor equations can be found in [13].

As for the oxygen depletion, a smoothing procedure is implemented for $\dot{Q}_{rad,ext}$. Since the soot concentration in the calculation reacts slower than the oxygen concentration (the soot particles have to be distributed in the room mechanically while the oxygen concentration decreases due to fast chemistry), a larger smoothing time (50 – 100 s) is recommended.

APPLICATION TO OECD PRISME EXPERIMENTS

The OECD PRISME experimental series is carried out by IRSN in the multi-compartment DIVA facility in Cadarache (Figure 5, left). The size of the fire compartment is 6 x 5 x 4 m³; the walls are made of concrete with a thickness of 0.3 m. All ceilings and the walls of the fire compartment (usually room 2) are isolated with rockwool panels in order to achieve high gas temperatures. The DIVA facility is equipped with a full ventilation system.

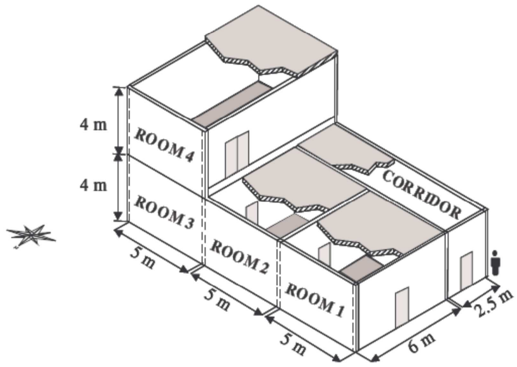
The used fuel is hydrogenated tetra-propylene (TPH), which is similar to dodecane. It was filled into a carbon steel pan, located 0.4 m off the floor in the center of the fire compartment. The pan sizes used range from 0.2 m² to 1 m². The fuel is ignited with a propane gas burner.

Since the validity of the Peatross-Beyler-correlation and its successful implementation into fire simulation codes is described in [6], this paper focuses on the PRISME LEAK and INTEGRAL experiments where radiation feedback from the hot soot layer plays a decisive role due to the high gas temperatures (higher than evaporation temperature of TPH). The pyrolysis rates observed in these experiments do not fulfill the Peatross-Beyler-correlation (Figure 1).

PRISME INTEGRAL test 1 used a three room configuration (Figure 5, right) with a 1 m² fire pool in the central compartment. The doors between the compartments (0.8 x 2 m²) were opened and the ventilation system was set to maximum ventilation rate. There were just one inlet and one outlet branch 0.8 m off the ceilings for the connected three rooms. PRISME INTEGRAL test 4 was very similar with just the corridor added to the configuration.

PRISME LEAK tests took place in just one single room, which was connected by small leakages to an adjacent room. The influence of the small leakages on the thermodynamics of the fire compartment was negligibly small. The pan size was 0.6 m² and the ventilation was adjusted to 15 h^{-1} air renewal rate.

[4] Hence the total pyrolysis rate (taking into account oxygen depletion and radiation feedback) can exceed the user given maximal pyrolysis rate.



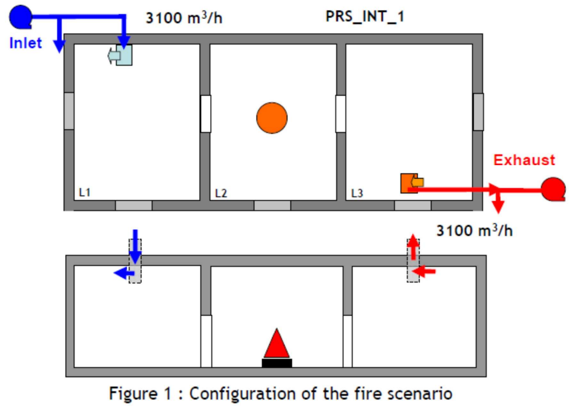


Figure 5 Scheme of the DIVA facility [7] and configuration of the INTEGRAL-D1 test

For the COCOSYS input, the compartments were subdivided in ten vertical layers in order to cope with the high thermal gradients evolving during the fire. The nodalization is shown in Figure 6 and Figure 7. The ventilation ducts are considered by small control volumes indicated in Figure 6. Since COCOSYS does not have a plume model, respective frustum shaped control volumes are used above the fuel surface and behind the doors in order to improve the air entrainment into the plume.

In order to simulate the pressure and volume flow characteristics of the ventilation system, the branch nodes are modeled by separate control volumes [12].

The material properties used for the calculation of the radiation feedback are given in Table 1. The oxygen dependent soot yield (input parameter) was set according to experiences from previous validation calculations of PRISME tests. It was reduced in each calculation after the initial phase of the fire (where combustion was observed to be less effective).

The radiation feedback from the hot soot layer was calculated just for the fire compartment.

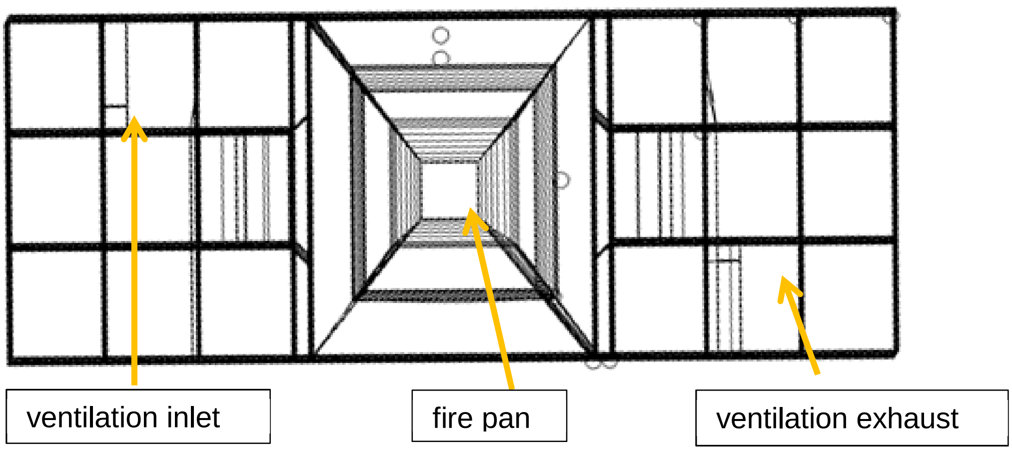


Figure 6 COCOSYS nodalization of the DIVA facility: Top view

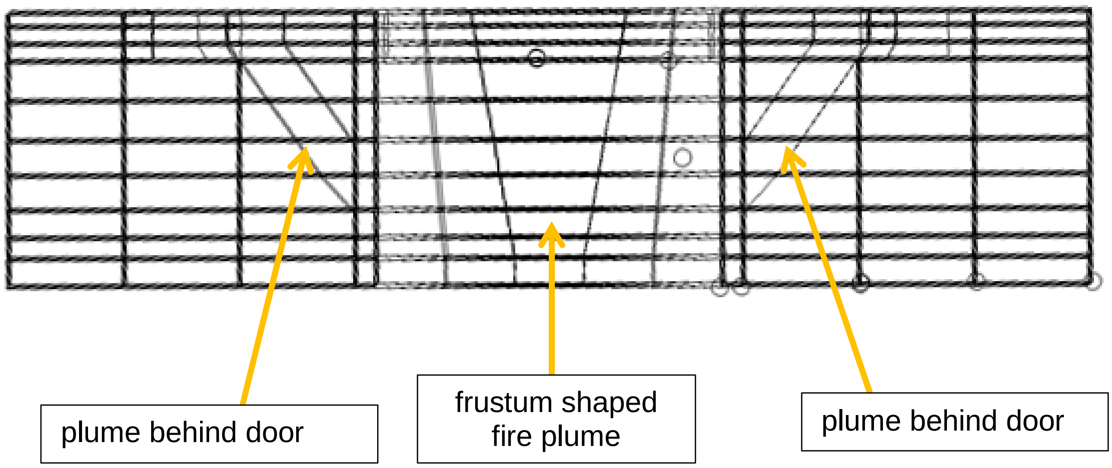


Figure 7 COCOSYS nodalization of the DIVA facility: Side view

Table 1 Material properties for TPH used in COCOSYS

Input variable	Value
Evaporation temperature of fuel T_{evap}	200 °C
Emissivity of fuel surface ε_{TPH}	0.95
Coefficient for emissivity of soot a_v	0.9

EXPERIMENT PRISME LEAK D1

Figure 8 to Figure 11 show the results of the COCOSYS calculation using the predictive pyrolysis model for PRISME LEAK D1 experiment. Since the results of the PRISME project are subject to restricted access up to now, the values in the ordinates are hidden in the presented plots. The user given open environment pyrolysis rate is indicated with orange dotted crosses in Figure 8. The radiation feedback (dashed blue circles) accounts for roughly 1/3 of the calculated pyrolysis rate (red straight squares). The COCOSYS calculation predicts the experimentally observed pyrolysis rate very well for the steady state phase; just the initial phase is mismatched due to the shape of the user given open environment rate.

Since the user given soot yield was adjusted to the experimental order of magnitude, the calculated soot concentration agrees with the experimental values (Figure 10). The oxygen concentrations at different points in the lower part of the fire compartment (calculated values from reference zones for the Peatross-Beyler-correlation and mean concentration) are given in Figure 9 and the gas temperatures at different locations at a height of 3.25 m in Figure 11. The COCOSYS results are in suitable agreement with the experimental values.

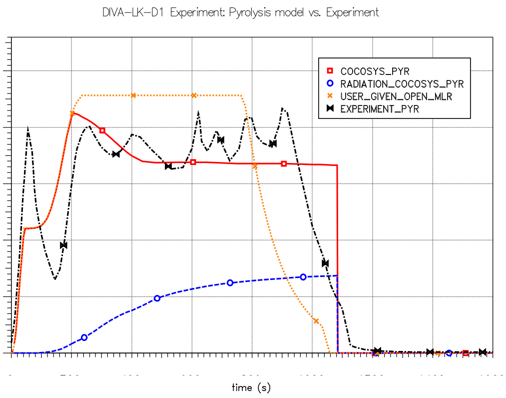


Figure 8 LEAK D1: Pyrolysis rate

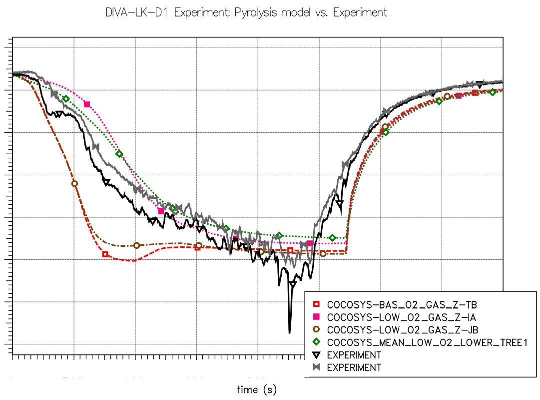


Figure 9 Oxygen conc. in lower layer

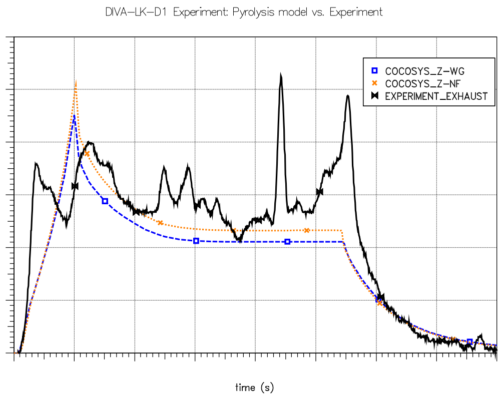


Figure 10 Soot concentration

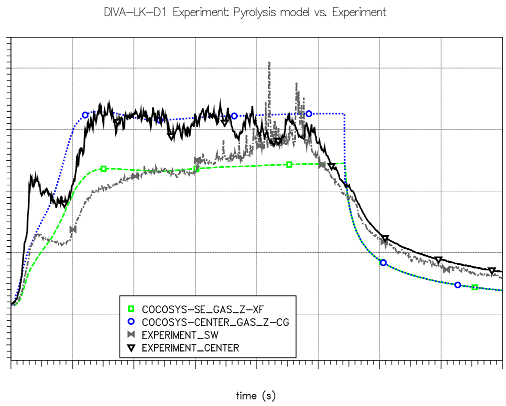


Figure 11 Gas temperature at 3.25 m

EXPERIMENT: PRISME INTEGRAL D4

Similar results were obtained for PRISME INTEGRAL test D4 (Figure 12 to Figure 15). For these, the calculated pyrolysis rate is in good agreement with the experimental data, even for the initial phase (Figure 12). In the initial phase, both the simulated and the measured pyrolysis rate exceed the maximum pyrolysis rate observed in open environment due to the back radiation.

Shortly before extinction, an excursion of the pyrolysis rate took place in the experiment caused by special local or random effects, which cannot be predicted by a code. (The last shallow puddle of oil might have evaporated very quickly due to the hot steel pan and possible impurities or bumps in the pan.)

The calculated oxygen concentrations in almost all parts of the lower layer in the fire compartment are close to each other (Figure 13), just the control volume located in the fresh air inlet flow path from the open door is oxygen enriched (orange circles; no corresponding measurement). This higher concentration should be used for determining the oxygen depletion effect according to Peatross-Beyler, because it represents the supply air, which is sucked in by the fire (and not the used up air).

The gas temperatures are underestimated by more than 100 K (Figure 15).

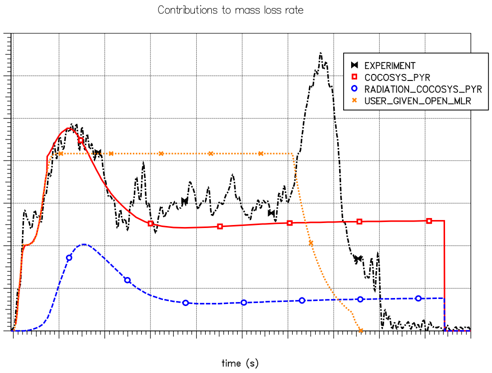


Figure 12 INTEGRAL D4: Pyrolysis rate

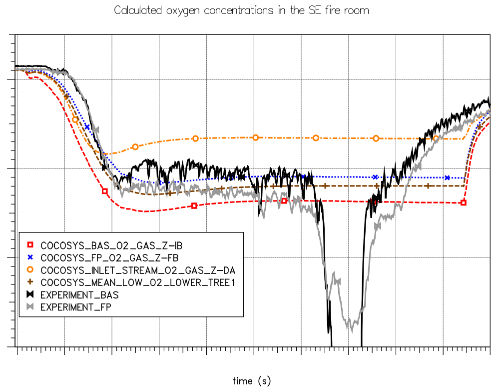


Figure 13 Oxygen concentration in lower layer

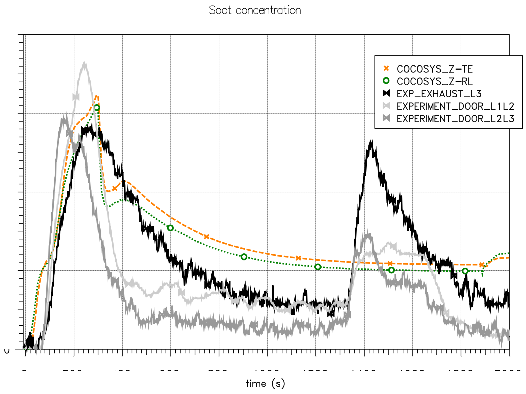


Figure 14 Soot concentration

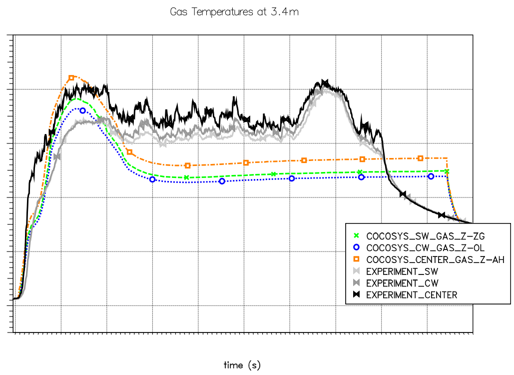


Figure 15 Gas temperature at 3.4 m

Figure 16 and Figure 17 show the results of another COCOSYS calculation with different user given soot yield (plotted with filled squares) compared with the previous calculation (open squares). The sensitivity of the pyrolysis rate prediction of the exact soot yield turns out to be rather small (but, of course, a variation of the order of magnitude of the soot yield will have a larger influence). This is encouraging since the soot yield is usually connected to some uncertainties and difficult to predict exactly.

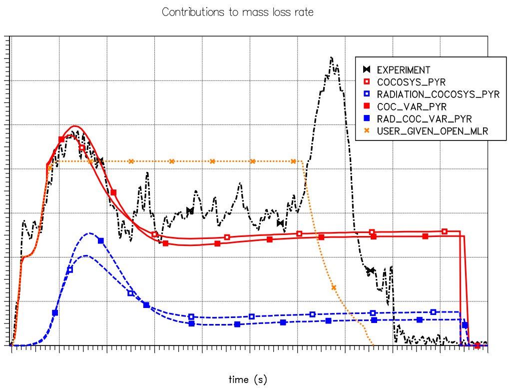


Figure 16 INTEGRAL D4: Pyrolysis rate

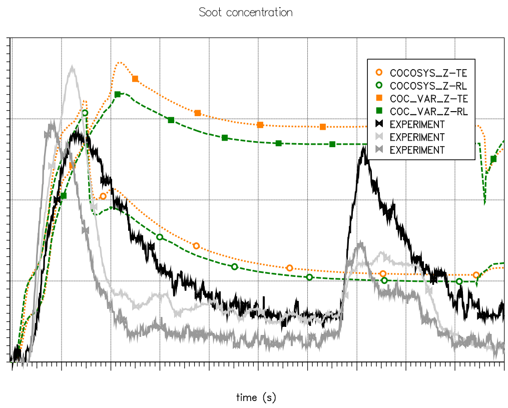


Figure 17 Soot concentration

EXPERIMEN PRISME INTEGRAL D1

In the INTEGRAL test D1, the oxygen concentration falls below 11 vol% (Figure 19). Hence, there is no contribution of the Peatross-Beyler-correlation to the predicted pyrolysis rate in COCOSYS – according to the calculation, the combustion continues to burn just due to the radiation feedback (Figure 18). Similar to INTEGRAL D4, the open environment rate is exceeded in the initial phase and an excursion occurs at the end of combustion.

The initial phase of the experimental pyrolysis rate is not satisfyingly reproduced by COCOSYS. But the experimental measurement of the similar INTEGRAL D2 test illustrates the real experimental scope of possible combustion developments (Figure 22): INTEGRAL D2 test is a reproduction of INTEGRAL D1 (same configuration), just with actuation of sprinklers after establishment of the steady state. The observed initial growth of the fire is much slower than in INTEGRAL D1.

As displayed by the two COCOSYS calculations in Figure 22, the initial course of the calculated pyrolysis rate depends mainly on the initial course of the user given open environmental pyrolysis rate. Its estimation is difficult and uncertain- but the range of possible initial fire evolutions is very broad in reality, too. Furthermore, the initial differences balance out in the simulation after reaching steady state.

Although the configuration of PRISME INTEGRAL D1 and D4 is very similar (INTEGRAL D4 just additionally includes the corridor - Figure 5), the pyrolysis rates evolving are very different. In the configuration *without* the corridor, less burnt hot gas can be delivered into adjacent rooms in comparison to the case *with* corridor. Hence, the gas temperatures in the fire compartment rise more quickly, which triggers even more pyrolysis due to the radiation feedback. Hence, supported by the effect of radiation feedback, the two experiments finally differ during steady state about factor 1.4 in the pyrolysis rate and 150 K in the gas temperatures in the hot gas layer beside the fire plume. Since taking into account the radiation feedback, these evolutions of differences is simulated by COCOSYS as well, but with overestimation of the differences (e.g. compare Figure 15 and Figure 21).

Interestingly, the breakdown of the ventilation inlet flow at ignition is identical in both tests according to the COCOSYS calculation.

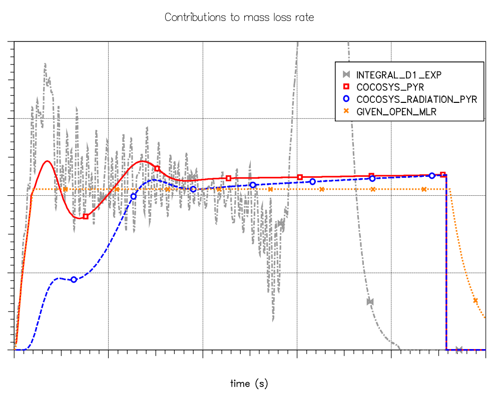


Figure 18 INTEGRAL D1: Pyrolysis rate

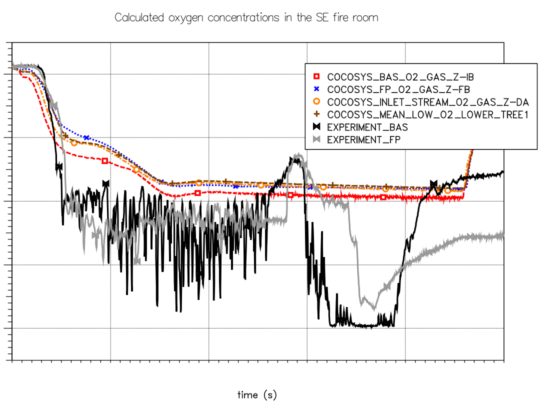


Figure 19 Oxygen concentration in lower layer

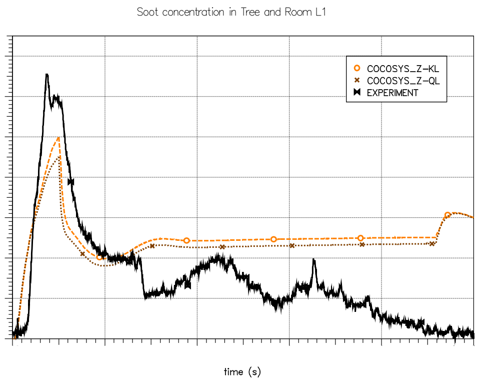


Figure 20 Soot concentration

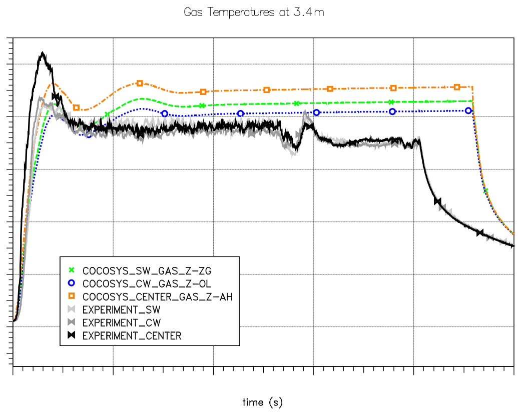


Figure 21 Gas temperature at 3.4 m

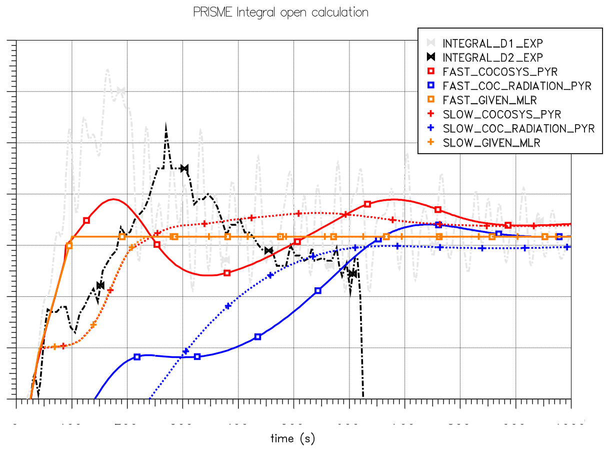


Figure 22 Experimental pyrolysis rates for INTEGRAL D1 and D2 test; two different COCOSYS calculations using two different user given open environment rates

CONCLUSIONS

For liquid pool fires in confined compartments, the Peatross-Beyler-correlation, which accounts for the effect of oxygen depletion, is not sufficient to describe the evolution of the pyrolysis rate in case of high gas temperatures in the fire compartment. The radiation feedback of the hot upper soot layer enforces the pyrolysis even at very low oxygen concentrations. A model for calculation of this radiation feedback has been presented. It is important to include a mechanism which limits the reciprocal enhancement of gas temperature and pyrolysis rate. (In the presented approach this is done by the adsorption coefficient of soot which increases exponentially with soot concentration.)

In addition to the Peatross-Beyler-correlation, this model has been implemented into the lumped parameter code COCOSYS in order to predict the pyrolysis rates of liquid pool fires in confined compartments when the respective rate in the open environment situation is known.

The predictive model succeeds very satisfyingly in simulating the steady state pyrolysis rates observed in several high temperature PRISME tests. The presented calculations underline the significance of considering the radiation feedback in the respective test – e.g. for the difference between INTEGRAL test D1 and D4 as well as for the open environmental pyrolysis rate being exceeded in the initial phase.

The prediction of the initial phase of a fire is still difficult. In the COCOSYS model, the initial fire phase is mainly determined by the open environment pyrolysis rate given by the user. However, in reality the range of possible initial fire evolutions is wide.

Nevertheless, since combustion is an extremely complex process which is sensitive to a variety of single and sometimes random effects, a predictive pyrolysis model can only be expected to give some order of magnitude and its uncertainties must not be neglected. Given these limitations, the presented COCOSYS model performs very well and might prove a useful tool for pre-test calculations and real applications.

ACKNOWLEGEMENTS

The OECD PRISME Project led by IRSN is funded on the German side by the Federal Ministry for Economics and Technology (Bundesministerium für Wirtschaft und Technologie, BMWi). Furthermore, the authors are grateful for the financial support of the participating countries to the joint OECD PRISME Project. The participation in the OECD PRISME Project Review Group and the PRISME Benchmark Group is funded by the Ministry for the Environment, Nature Conservation and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, BMU) within the frame of the project 3610R01375.

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EXPERIMENTAL AND NUMERICAL SIMULATIONS OF LIQUID SPREADING AND FIES AFTER AIRCRAFT IMPACT

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ABSTRACT

Accurate simulation of the fuel spreading and combustion processes resulting from an aircraft impact is an extremely challenging exercise due to the vast range of physical and temporal time scales involved. Some developments in the numerical fire simulations towards a realistic prediction of the fuel spray dynamics and spreading are presented in this work. Experimental results are utilized for the validation of the simulation tools and for the determination of necessary input parameters, such as initial spray velocities and drop size distributions. Applications of the simulations in the prediction of pooling fraction and heat impact in a real-scale but simplified plant geometry are reported.

INTRODUCTION

The spreading and burning of aircraft fuel after an aircraft impact is one of the nuclear power plant accident scenarios that have reached increasing attention during the last ten years. In addition to the mechanical damage caused by an aircraft impact, the impact would most likely be accompanied by a fireball, where fuel carried by the plane burns in explosive manner. All of the fuel is not necessarily burned in this fireball however and some of it may end up in burning pools on the ground. These pools may cause local damages to the buildings and ignite further fires. The smoke emissions from the pools may cause problems for the plant air intakes. The analysis of such scenarios is mostly based on modelling and numerical simulations. In this paper results of the CFD simulations are presented and the most important parameters affecting the pooling of fuel in front of the building are determined. The thermal impact of the fireball is also estimated.

MODEL DESCRIPTION

Simulations were performed using Fire Dynamics Simulator version 5.5.3 **Fehler! Verweisquelle konnte nicht gefunden werden.**. Fluid motion is computed using weakly compressible Large Eddy Simulation (LES)-based solver. This model cannot predict the increase of pressure by detonation. Two-phase flow is computed using Eulerian-Lagrangian concept where the liquid droplets have zero volume in Eulerian space. This means that the liquid-liquid interactions, that may have importance in dense sprays and early phase of the impact, cannot be taken into account.

Ignoring buoyancy, lift and forces arising from fluid acceleration, the motion of single spherical droplet is governed by the equation of motion

$$\frac{dm_d\vec{v}_d}{dt} = m\vec{g} - \tfrac{1}{2}\rho_g C_D A_{eff}\|\vec{v}_{rel}\|\vec{v}_{rel}. \tag{1}$$

Here on the left hand side m_d is the mass of the droplet and $\vec{v}_d$ is the velocity of the droplet. On the right hand side, $\vec{g}$ is the gravitational acceleration, ρ_g is the density of the surrounding gas, $\vec{v}_{rel} = \vec{v}_d - \vec{v}_g$ is the velocity of the droplet relative to the surrounding gas, $A_{eff} = \pi r_d^2$ is the projected surface area of the droplet, r_d is the radius of the droplet, and C_D is the drag coefficient. The drag coefficient is given by

$$C_D = \begin{cases} 24/\mathrm{Re}_p & \mathrm{Re}_p < 1 \\ 24\left(0.85 + 0.15\mathrm{Re}_p^{\;0.687}\right)/\mathrm{Re}_p & 1 < \mathrm{Re}_p < 1000 \\ 0.44 & \mathrm{Re}_p > 1000 \end{cases} \tag{1}$$

where $\mathrm{Re}_d = \rho d_d \|\vec{v}_{rel}\| / \mu_g$ is the droplet Reynolds number. Due to the large number of droplets in a real spray, only a fraction of these droplets is tracked. Instead each droplet in the simulation represents a parcel of droplets with the same properties. By default, the Cumulative Volume Fraction of droplet diameters follows a distribution that is a combination of lognormal and Rosin-Rammler distributions

$$F(d) = \begin{cases} \frac{1}{\sqrt{2\pi}} \int_0^d \frac{1}{\sigma d'} e^{-\frac{[\ln(d'/d_m)]^2}{2\sigma^2}} dd' & d \le d_m \\ 1 - e^{-0.693\left(\frac{d}{d_m}\right)^\gamma} & d_m < d \end{cases} \tag{2}$$

In a configuration where two particles are directly in line, the reduction of hydrodynamic forces to the second (trailing) sphere due to the wake effect was studied by Ramírez-Mũnoz et al.[1]. They developed the following analytical formula for the hydrodynamic force to the second sphere. In our work, this formula is used to compute a reduction factor for drag coefficient

$$C_D = C_{D0} \frac{F}{F_0} \tag{3}$$

where C_{D0} is the single droplet drag coefficient and F/F_0 is the hydrodynamic force ratio of trailing droplet to single droplet.

$$\frac{F}{F_0} = W\left[1 + \frac{\mathrm{Re}_1}{16} \frac{1}{\left(L/d_d - \frac{1}{2}\right)^2} \exp\left(-\frac{\mathrm{Re}_1}{16} \frac{1}{\left(L/d_d - \frac{1}{2}\right)}\right)\right] \tag{4}$$

where Re_1 is the single sphere-Reynolds number and W is the non-dimensional, non-disturbed wake velocity at the center of the trailing sphere:

$$W = 1 - \frac{C_{D0}}{2}\left[1 - \exp\left(-\frac{\mathrm{Re}_1}{16} \frac{1}{\left(L/d_d - \frac{1}{2}\right)}\right)\right] \tag{5}$$

This model assumes that the spheres are travelling directly in line with each other. As such, this provides an upper bound for the strength of the aerodynamic interactions. On the other hand, the results of Prahl et al. [2] indicate that for short droplet separation distances the drag reduction effect is most likely under estimated. Sufficiently accurate predictions of spray propagation can be achieved, at least in the scale of the VTT's Impact tests [[3]], [[4]].

SIMULATION OF PLANE IMPACT ON BUILDING

The accident scenario under investigation is as follows. A plane loaded with 10 t of fuel crashes in to a large building and the fuel carried by it is dispersed into the surroundings. The fuel is ignited immediately on the impact. The fuel may burn in the air but some of it may reach the ground where it will form burning pools. Damage to the building resulting from the mechanical forces involved in the impact is ignored.

The computational domain is 100 m wide, 100 m deep and 150 m high. A Cartesian grid with uniform 1 m discretization interval is used. The building suffering the impact is modelled as a rectangular obstruction 40 m wide, 20 m thick and 50 m tall. Figure 1 shows the dimensions of the computational domain. The liquid droplets are assumed to be heptane with properties listed in Table 1. The combustion reaction of the evaporated heptane is

$$C_7H_{16} + v_{O_2}O_2 \rightarrow v_{CO_2}CO_2 + v_{H_2O}H_2O + v_{CO}CO + v_{Soot}Soot + v_{H_2}H_2.$$

The stoichiometric coefficients ν are given in Table 1. The heat of combustion of heptane is 4.4×10^4 kJ/kg.

Table 1 Physical properties used in the simulations for liquid heptane
Note: density of heptane is 684 kg/m^3 in reality.

Property	Value	Units
Density	1000	kg/m^3
Specific heat C_P	$0.01061T^2 - 2.67961T + 2098$	J/kgK
Heat of vaporization H_V	435867	J/kg
Reference temperature T_{REF}	182.6	K
Freezing temperature T_{MELT}	182.6	K
Boiling temperature T_{BOIL}	371.5	K

Initial and Boundary Conditions

Initially, the air is at rest and both air and structures are at 20 °C temperature. All the mesh boundaries except the bottom are open to the flow.

When a plane or missile hits a wall, any liquids contained in it will be quickly dispersed to the surroundings. We model this "splashing" effect by introducing high speed droplets on a circular band with a 3 m radius. Figure 2 shows the placement of the injection band on the building. The hypothetical impact point is in the middle of the building 15 m above the ground.

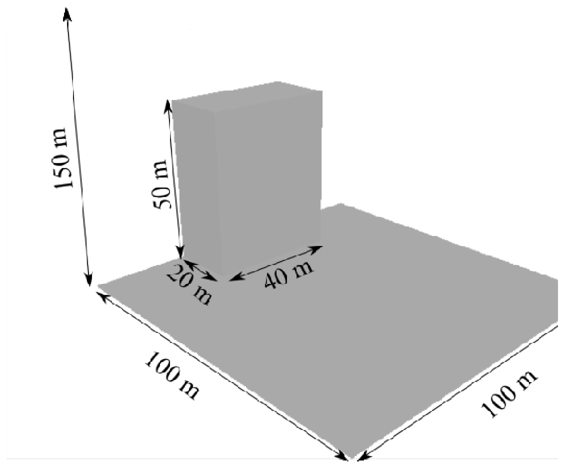

Figure 1 Schematic of the computational domain

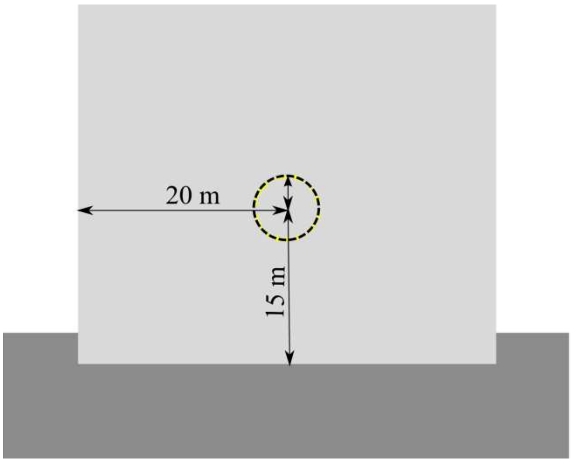

Figure 2 Placement of the initial droplets

The energies involved in a plane crash are very large and thus the liquid will initially have large momentum. In the simulations the droplets are ejected with a radial velocity of 250 m/s. A total of 10000 kg of liquid is injected to the simulation. All the liquid is released during a 0.1 s time period. This corresponds to 10 m wide wing impacting a building with velocity of 100 m/s.

Simulations were performed in two phases. In the first phase, the effects of the droplet size distribution on the simulations were assessed. From these simulations, a droplet diameter was selected for use in further parametric studies. Parametric sensitivity studies for the pooling fraction were conducted for the impact location, amount of fuel and impact time. The parameters of the base simulation case are

Fuel mass	10 000 kg
Splash time	0.1 s

First, an overview of the flame ball properties is given for the base simulation. Next, the results pooling fraction results are given and finally, the thermal exposures are reported for the base case and some alternative ways to prescribe the fuel boundary conditions.

Overview of the Fireball

Figure 3 shows an overview of the flame and smoke plume in a simulation with 10 t of fuel released within a 0.1 s time period as droplets with 30 µm volumetric median diameter. The individual pictures correspond to instantaneous moments of the simulation at times shown at the bottom of each picture. The flame has reached the edge of the simulation domain one second from the beginning. From this time forward, some fraction of the fuel flows out of the computational domain, and the predicted heat release rate is lower than what would be obtained using a larger computational domain. This should have no effect on the fraction of the fuel that is accumulated into the pools because the pools are well within the computational domain. It may have some effect on the predicted thermal exposures though. The current computational domain is clearly too small for the assessment of the smoke entrainment into air intakes.

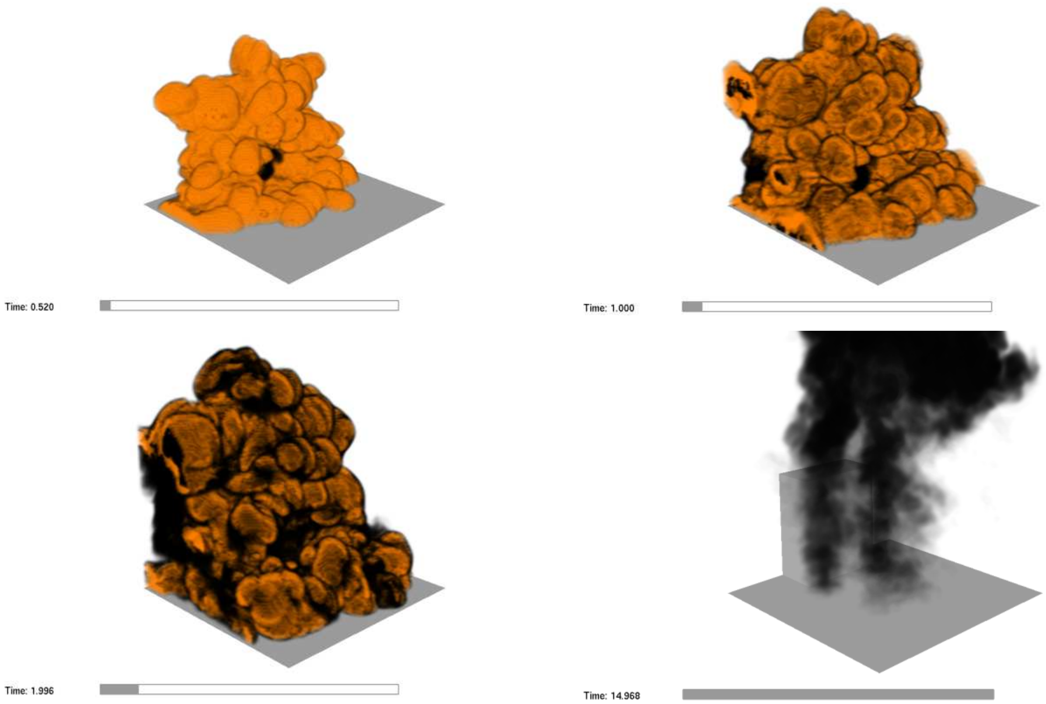

Figure 3 Overview of the flame ball and smoke plume

Figure 4 shows instantaneous gas temperature fields in a vertical plane cutting through the centre of the building, i.e. the impact location. Highest temperatures are observed 2 to 3 s from the impact. However, part of the flame has flown out of the computational domain already at 2 s from the ignition, decreasing the predicted thermal exposure, particularly at the roof of the building. About 6 s from the impact, the flame starts to leave the domain through the top boundary at height of 150 m, and about 12 s from the impact all the flames have left the domain.

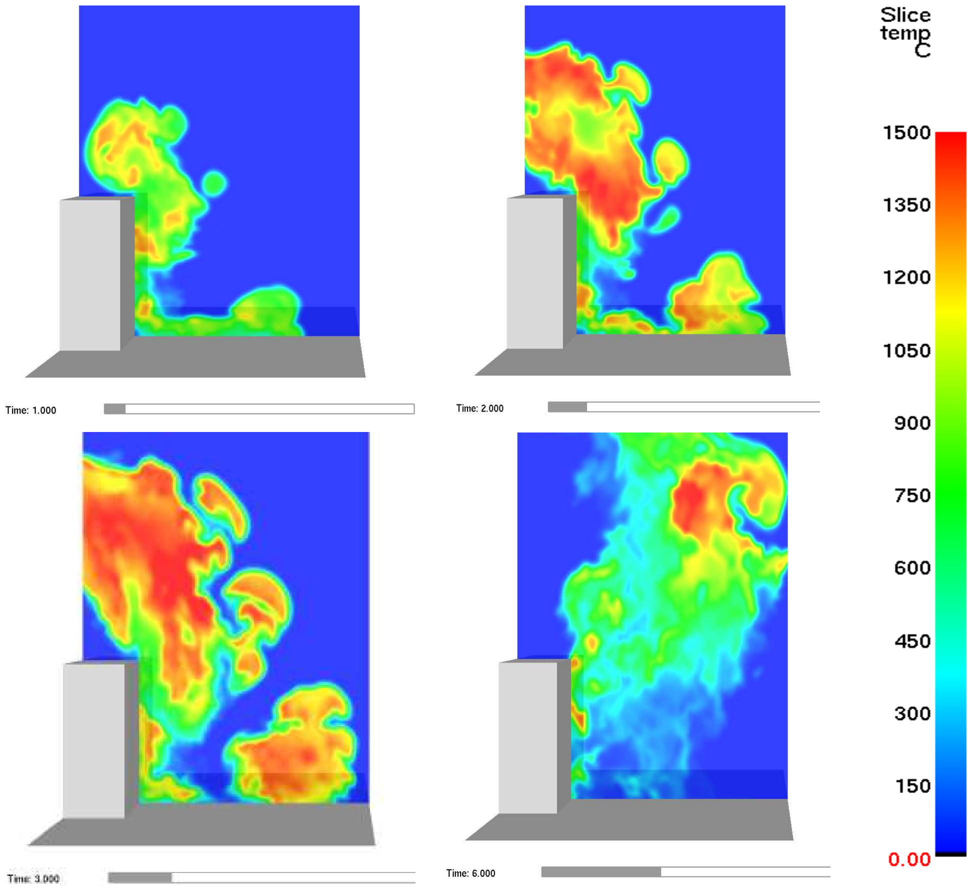

Figure 4 Instantaneous temperature fields in the simulation

Sensitivity to Numerical Parameters

The droplets are inserted in to the simulation at speed that is close to the speed of sound. However their aerodynamic drag slows them down very quickly and in turn gives rise to large accelerations in the gas phase. In the numerical scheme used to solve droplet transport, this sets stringent restrictions for the computational time step. A dynamic time step selection procedure, similar to the CFL-condition used on the gas phase, was therefore used. The time step was selected so that the fastest flying droplet can cross at most a predetermined fraction of a computational cell within a time step. For most of the simulations this fraction was 0.005. This lead to time-steps on the order of $2*10^{-5}$ s. Typically the total 15 s of simulation would take around 400 h of CPU time.

All the simulations were found to be very sensitive to the time step used. Too long a time step would sometimes lead to instability. Increasing the fuel mass or shortening the liquid release time would lead to shorter time steps. The source of the instabilities seems to be in the coupling of the gas phase and the dispersed phase. Turning off the combustion or using water droplets instead of fuel droplets did not have a significant effect on the stability requirements. Efficient computation of this kind of flows in the future would require improvement of the numerical time integration scheme.

The most computationally intensive part of the simulations was the droplet transport. The cost of this part was mostly affected by the number of droplets used to describe the spray. This number can be controlled by varying the rate at which the numerical super drops (or "parcels") are inserted into the simulation. For shorter splash times a larger number should be used. In addition to the droplet insertion rate, the rate of droplet removal, either by evaporation or by hitting the boundaries, affects the number of active droplets in the simulation at any given moment. However the computational requirements grow linearly with the number of droplets used. This meant that due to time restrictions it was not possible to use the exactly same number of droplets in all the simulations.

Actual grid sensitivity studies were not performed. However, the sensitivity of the simulations to the droplet insertion rate was investigated in the base case of the splash time. Insertion rates between $5*10^7$ drops/s and $1*10^8$ drops/s were tested. It had no effect on the stability of the simulations. Based on a visual inspection, it did not have a significant effect on the simulation results either.

Effect of Droplet Size Distribution and Impact Location

Jepsen et al. [5] conducted large scale water slug impact tests at Sandia National Laboratories (SNL) and measure the droplet sizes of 7…12 µm inside the "residual mist", outside the main splash pattern. These experiments were conducted with water as the liquid. For common spray nozzles the droplet size distribution is proportional to the liquid surface tension. Kerosene has a lower surface tension than water and thus it is expected that kerosene would result in much smaller drops. The effect of the median droplet size was studied by turning off the breakup model and using particle size distributions with volumetric median droplet sizes of 300, 150, 75, 30 and 15 µm.

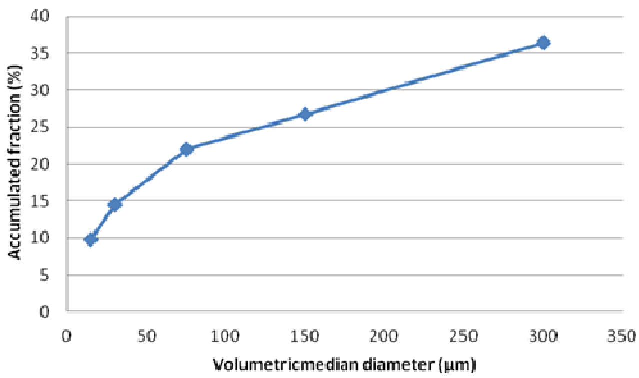

Figure 5 Accumulated fraction of initial fuel mass in the pools as function of the initial volumetric median diameter of the droplets

Figure 5 shows the fraction of the accumulated mass as a function of the droplet size. The accumulated fraction decreases quickly as the median droplet diameter is decreased below 100 µm. At larger diameters, the pooling fraction increases in a linear manner as the drop size is increased. Larger droplets are also easier to handle numerically. Given the previous discussion of experimental and theoretical considerations of the droplet size distribution, it is expected that droplet sizes in real impact scenarios would tend towards the smaller diameters used. For further simulations, the 30 µm median droplet diameter is used.

Figure 6**Fehler! Verweisquelle konnte nicht gefunden werden.** shows the splash patterns for median volumetric diameters of 150 µm and 30 µm. For smaller diameter the splash pattern has "fingers" were the droplets advance faster than the droplets elsewhere in the splash front. These fingers have also been observed experimentally.

a) b)

Figure 6 Effect of droplet size on splashing pattern; median droplet diameter is 150 µm on the left and 30 µm on the right.

Figure 7 shows the fuel accumulation patterns as a function of the volumetric median diameter of the initial droplet size distribution. Clearly smaller droplet sizes lead to pools that have "fingers" while for larger droplets the pools are of a more regular shape. The appearance of the "fingers" in the pool shapes is related to the splash patterns.

Figure 8 demonstrates the effect of height of impact location on the accumulated fraction of fuel. It can be clearly seen that lower impact heights lead to significantly increased pooling fractions. When the impact location is lower in the building, droplets have shorter distance to travel before reaching the ground and thus there is less time for the droplets to evaporate before **Fehler! Verweisquelle konnte nicht gefunden werden.** hitting ground. On the other hand, Figure 9 clearly shows that the lateral offset has no discernible effect on the accumulated fraction. However, there is a very slight increase in the accumulated fraction, when the impact is located very close to the edge of the wall.

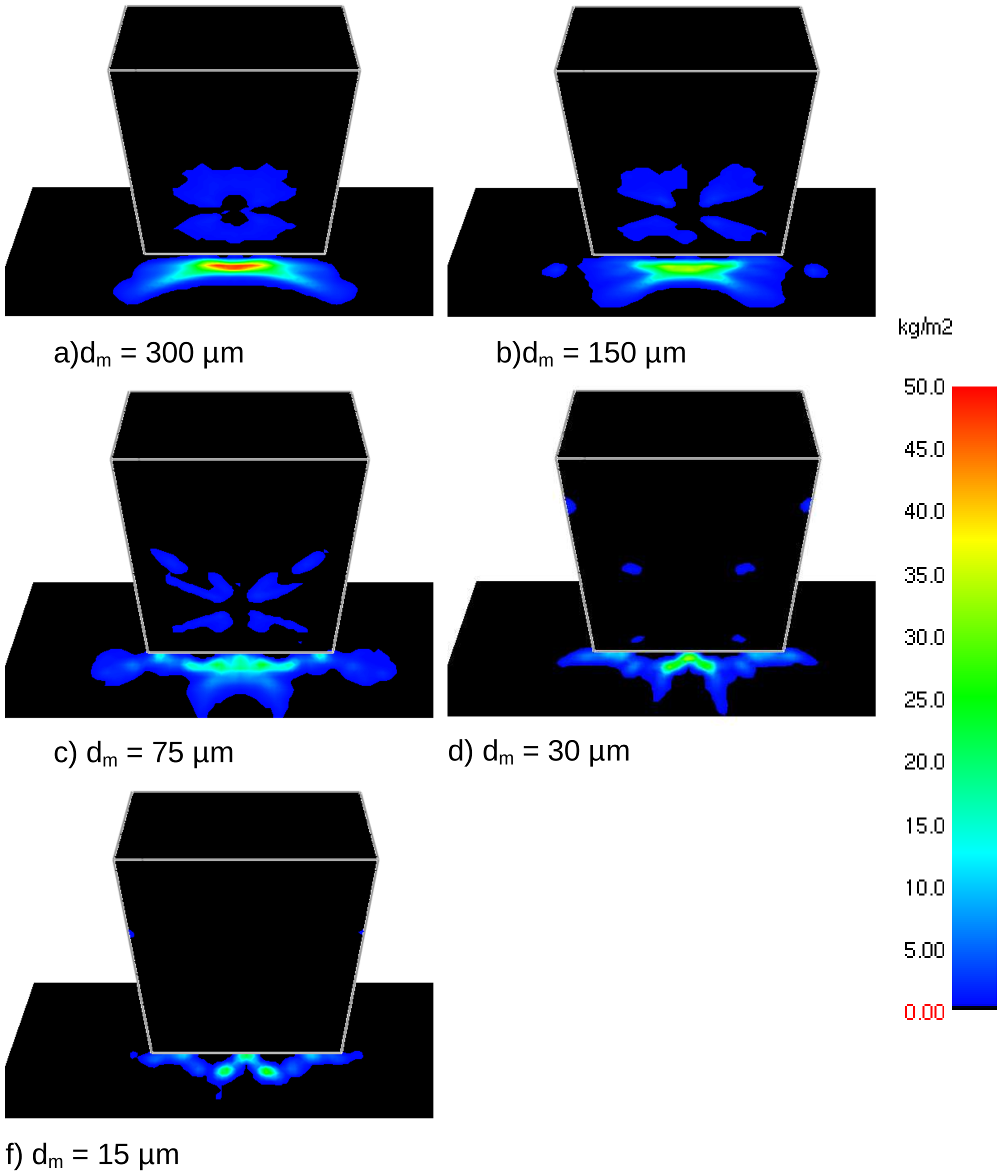

Figure 7 Accumulated mass per unit area for different initial median diameters

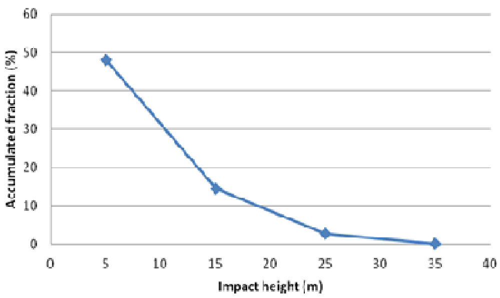

Figure 8 Accumulated fraction of initial fuel mass in the pools as function of impact height

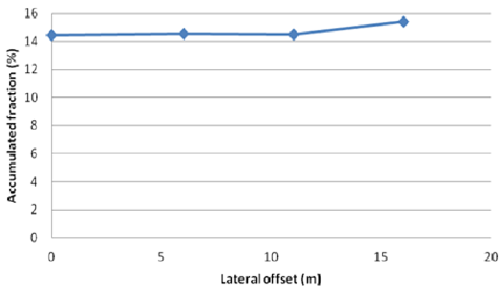

Figure 9 Accumulated fraction of initial fuel mass as function of lateral offset of the impact location

Thermal Exposure

The thermal exposure from the fuel cloud flame on the structures and objects surrounding the impact point is evaluated by plotting the Adiabatic Surface Temperature (AST) at various horizontal distances from the impact point. As these quantities are monitored on the bottom surface of the computational domain, the results at higher distances may not observe the heat flux resulting from the pool fires. One of the measurement locations was placed on the roof of the target building, at horizontal distance of 10 m from the edge of the building. The locations of the measurement sensors are shown in Figure 10.

Fehler! Verweisquelle konnte nicht gefunden werden. shows the recorded AST when 10 t of fuel is released in 0.1 s. The results are plotted in three directions: normal from the wall, to the side from the impact location, and on the diagonal direction between. The peak exposure temperatures are above 1400 °C but the duration of these high peaks is very short. Overall, the thermal exposure from the flame ball lasts about 10 s.

Using the AST results as boundary conditions, the THIEF model was used to compute the cable jacket thicknesses that are needed for electrical cables placed at the corresponding distances to survive the impact. Figure 11 shows the results of the THIEF analysis. The result corresponding to the data in Figure 12 is shown as a black curve, with legend '10 t, 0.1 s, impact'.

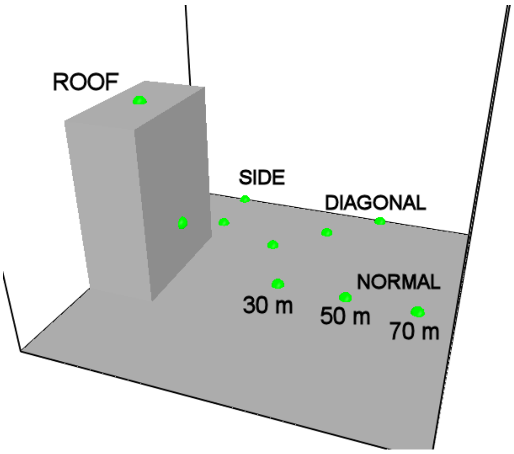

Figure 10 Placement of AST sensors

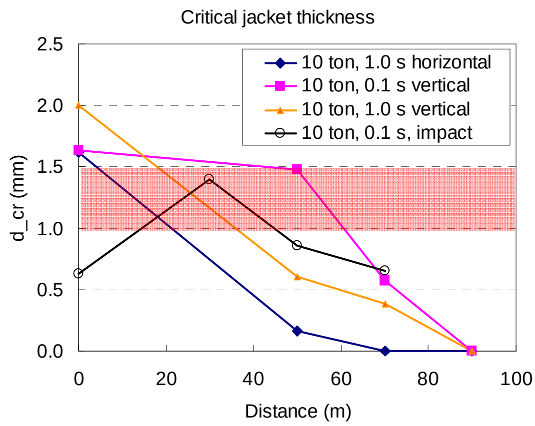

Figure 11 Critical thickness of the cable jacket at different distances from the impact location

The sensitivity of the thermal exposure calculations to the fuel boundary condition was studied by performing three alternative simulations where the fuel source was modelled as a vertical or horizontal burner releasing 10 t of fuel within a 0.1 or 1.0 s release time. This type of boundary condition is similar to the one used by Luther & Müller [6]. In these simulations, the computational domain was 140 m (normal to wall) × 200 m × 200 m (height), and the roof sensor was placed at the edge of the building.

At the roof of the building, the needed jacket thicknesses range from 0.6 to 2.0 mm, depending on the impact scenario. The simulation with spray gives the lowest thermal exposure at this distance, but currently it is not known if this difference is due to the different fuel source or different placement of the sensing element. At higher distances, the spray simulation results are between the two simulations with vertical burner boundary conditions. A general observation from the THIEF results is that horizontal burner boundary condition gives the lowest thermal exposures. If one wants to perform this type of simulations without the actual droplet computations, the recommended practice is to release the gaseous fuel from the vertical surface of the target building within a short release time.

Typical electrical cables used in nuclear power plants have a jacket thickness between 1 and 2 mm. The results in Figure 11 indicate that most cables that are further than 60 m from the impact location would survive the impact.

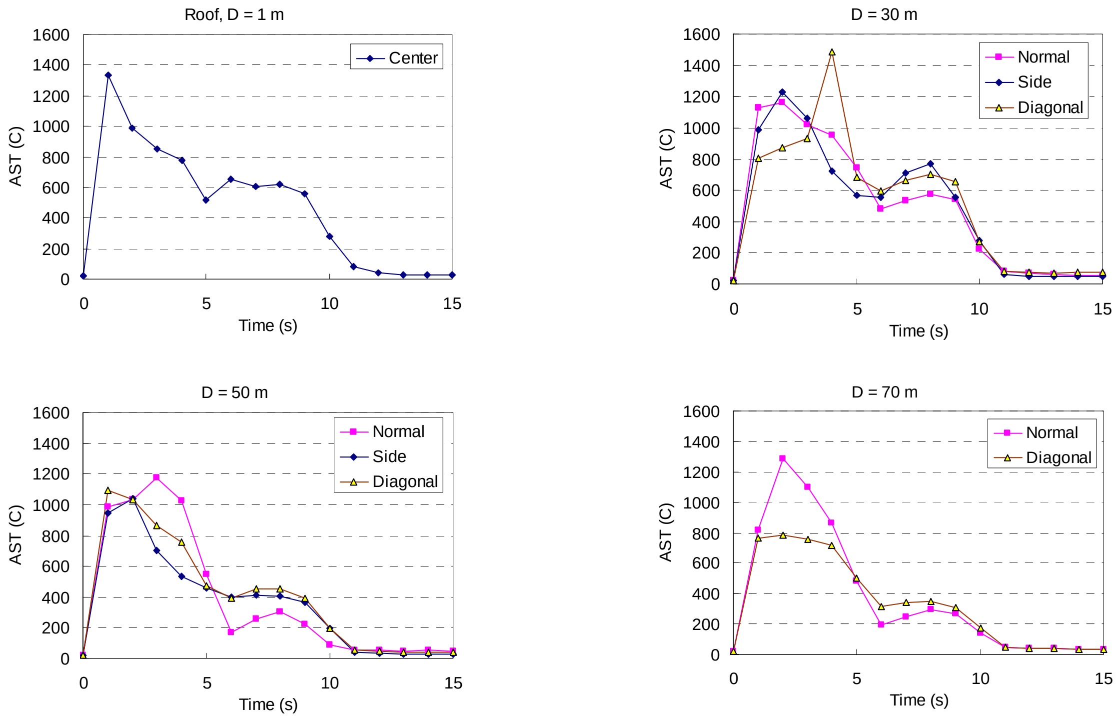

Figure 12 Adiabatic surface temperatures at various distances from the impact point

CONCLUSIONS

CFD simulations of the liquid fuel and the resulting fireball during an aircraft impact were performed using the FDS code. The simulations were used to investigate the fraction of the fuel that is accumulated into the pools on the ground and the thermal exposures from the initial fireball. The results indicate that the fraction of the accumulated fuel depends strongly on the choice of the droplet size distribution and the height of the impact location. The mass of the released fuel had also some effect. At the time when the simulations were performed, the droplet breakup model still produced too small distributions, yielding very small pooling fractions and therefore prescribed initial droplet size distributions were employed.

The thermal exposure from the initial fireball was found to be very intense but short in duration. Using typical electrical cables as an indicator of the severity, the damages could be expected within a 60 m distance from the impact location. For the simulations where the fuel boundary condition is simplified into a burner-like flow, a recommendation is given to place the fuel inflow boundary on the vertical wall of the target building rather than on the ground. The current results highlight the importance of physical modelling the droplet size distribution. To this end, more work is under way on the particle model.

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VALIDATION AND DEVELOPMENT OF DIFFERENT CALCULATION METHODS AND SOFTWARE PACKAGES FOR FIRE SAFETY ASSESSMENT IN SWEDISH NUCLEAR POWER PLANTS

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ABSTRACT

Fire models are used increasingly for fire safety assessment in nuclear power plants. Examples of these fire models are CFD models and zone models. When using these models it is important that they are sufficiently verified and validated. In Sweden the majority of fire safety consultants are using FDS, developed by NIST. The verification and validation of this package is intensively done for a number of scenarios by the developing team. One of the recent modifications in FDS is the implementation of a ventilation module. The first aim of this paper was to validate FDS and ANSYS-CFX within the Swedish part of the OECD/NEA PRISME project using the PRISME SOURCE D1 test. The results are presented in this article and show how powerful the module is for simulation in enclosures with mechanical ventilation. Beside CFD models, fire safety engineers need also simple empirical models for determining temperatures, smoke heights, etc. In the second part of this paper such a model is developed. The model predicts gas temperatures in a room adjacent to a room involved in a pre-flashover fire. The correlation is derived with help of computer simulations and validated by means of a set of fire tests. The results of the correlation model are satisfactory and the correlation formulae will be an additional tool for fire safety engineers.

INTRODUCTION

Validation of CFD Models

The partners participating in the international OECD/NEA project PRISME [1] investigated the use of CFD and zone models for enclosures with mechanical ventilation. This was performed in the benchmarking group. The main PRISME program (French acronym for "Fire Propagation in Elementary Multi-room Scenarios") mainly aims on studying smoke and hot gases propagation in full scale, well-confined and mechanically ventilated fire compartments [2]. In particular, the goals of the PRISME program are to understand and quantify, by means of an analytical approach, the propagation mechanisms of smoke and heat from a fire compartment towards one or several adjacent compartments in scenarios representative for nuclear plants. For this purpose one of the tests was used in an open validation exercise (a posteriori), namely the PRISME SOURCE D1 test. The exercise was performed in the PRISME consortium and reported by Audouin et al. [3] and contained validation of different zone, hybrid and CFD models. The test set-up and overview of the test rig used for the test are given in Figure 1. All the FDS results reported by Audouin et al. were based on the boundary conditions available in FDS [4] [5] at that moment, which did not have the possibility of using the most recent ventilation module developed by Floyd. This paper will report how this ventilation module in FDS have been used and applied on the PRISME SOURCE

D1 tests. Moreover, results with the commercial software ANSYS CFX [6] are also given for the same test set-up. Further details about the test can be found in the publication from the benchmark exercise [3].

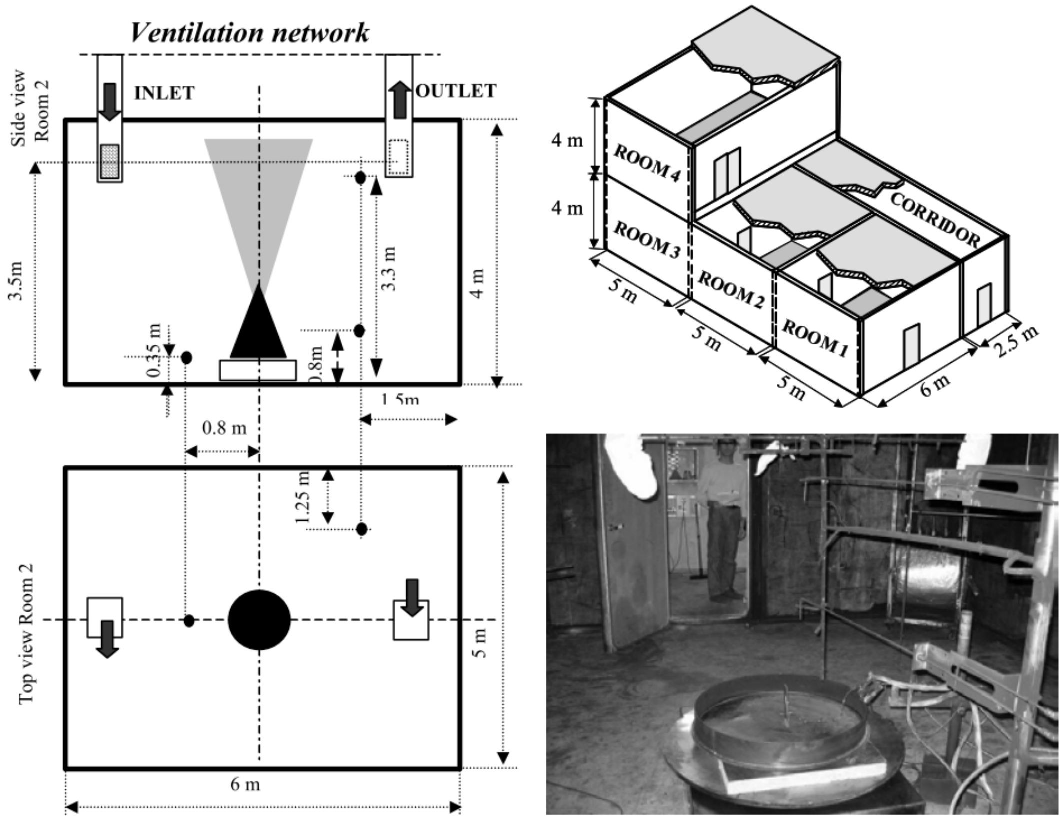

Figure 1 Overview of the experimental setup (courtesy to IRSN)

Use of Empirical Models

Advanced computer modeling software that can predict smoke spread and compartment temperatures has been developed during the last decays. With zone models and computational fluid dynamics (CFD) it is possible to e.g. calculate smoke layer heights, species and temperatures in a multi-room geometry. The programs are generally good tools for fire engineering purposes, but they do not remove the need for simple engineering correlations. Simple correlations can be used for hand-calculations to get a first estimate of e.g. smoke layer temperatures in performance based design of a building and help the fire engineer to determine if it is necessary to perform a detailed CFD calculation. Simple correlations can also be a useful tool to use in sensitivity analysis or in fire risk analysis.

Correlations that predict compartment temperatures for single room enclosures date back to the early eighties [7], [8] and are still used for different purposes by fire engineers. These correlations are rough and less accurate compared to computer simulations but they have the benefit of being simple and giving a good description of the hazard. The method that McCaffrey et al. presented [7] (MQH-correlation) is based on a simple conservation of energy expression. The MQH correlation gives the gas temperature as a function of the heat release rate, ventilation conditions, enclosure geometry and thermal properties of the enclosure. The MQH correlation has a set of limitations, which the user must be aware of, but it has been shown to give good predictions of room fire temperatures [9]. The correlation has even been developed further and modified [8], [9]. Lately new models for predicting compartment fire temperatures have been presented [10], [11]. However there are few correlations that can predict temperatures outside the room of fire origin. Thus such predictions have to be done with the help of zone or CFD models. A simple correlation that would predict temperatures outside a compartment is something that could be useful to get a first estimate when for example evaluating conditions for evacuees in a room next to the room of fire origin or to make a first assessment with respect to functional performance of cables.

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METHOD

Validation of CFD Models

The experimental scenario (Figure 1) was conducted at the French “Institut de Radioprotection et de Sûreté Nucléaire” (IRSN). The quantitative comparisons between measurements and numerical results obtained from “open” calculations concerned six important quantities from a fire safety viewpoint: gas temperature, oxygen concentration, wall temperature, total heat flux, compartment pressure and ventilation flow rate during the whole fire duration. The fire source [12] consisted of a 0.4 m^2 steel pan filled with hydrogenated tetra propylene (TPH), an isomer of n-dodecane. The walls, ceiling and floor of the room were 30 cm thick and made out of concrete. During the experiment, rock wool (THERMIPAN) with a thickness of 5 cm insulated the ceiling to prevent damage to the facility. The ventilation system in the fire room included an inlet branch and an exhaust branch, the relative static pressures and volume flow rates was recorded before the fire was ignited and was later used as input data in FDS simulations [13] [4] and for ANSYS-CFX [6].

Use of Empirical Models

The work presented in this paper has been performed in three steps. In the first step numerous CFD simulations with the computer software FDS 5 [4] have been conducted. Input files to FDS, with randomly sized two-room configurations, were created with a Matlab script. Approximately 140 FDS files were simulations with different, geometries, openings, wall materials, fuels and heat release rates on the Lund cluster. In all simulations the fire was placed in the center of the fire room as illustrated in Figure 1.

The various inputs are e.g. size of the door-opening, size of the room, HRR, fuel, properties of the wall and wall thickness. The mesh size was determined by following the recommendations of characteristic fire diameter D*, which varied between 0.61 and 1.27 m. In the second step a statistical analysis has been conducted with the statistical software package SPSS (Statistical Package for the Social Sciences) [14]. The smoke layer temperature in the adjacent room was retrieved from the FDS simulations and was used as dependent variable in the statistical analysis. The heat release rate, area of boundary surfaces of both enclosures, ventilation factors for both openings and heat transfer coefficient were used as independent variables. A multiple linear regression analysis of the logarithmic values of the variables were conducted in SPSS.

In the third step the correlation was tested and validated against results from full-scale experiments both found in literature and conducted within the project.

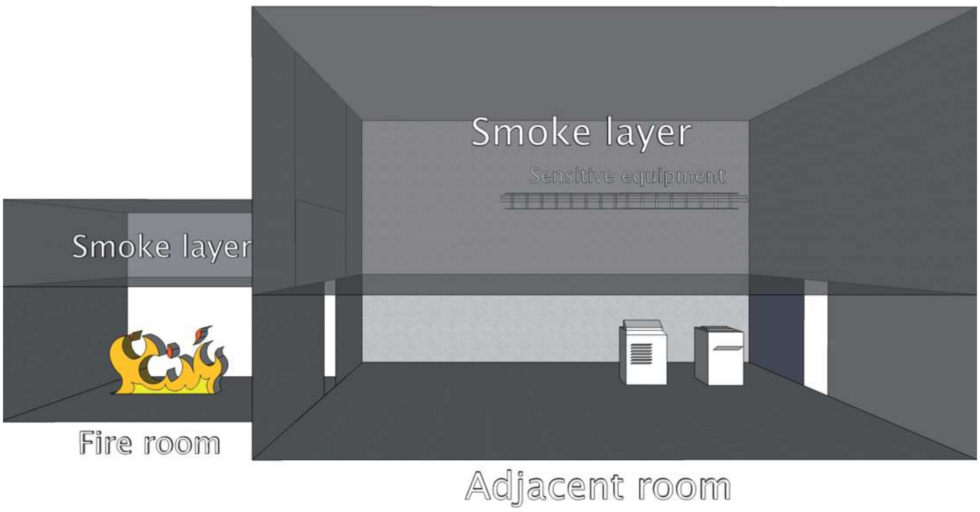

Figure 2 Room set-up for the simulations

RESULTS AND DISCUSSION OF THE VALIDATION OF THE VENTILATION MODULE

The leak area from the fire room to surroundings was calculated using data from PRISME SOURCE – Ventilation Tests. Leakage between the fire room and surroundings was assumed to be a quadratic function of pressure difference. The calculated total leakage area from the fire room was in the order of 4 cm^2. The sensitivity of this parameter was tested by doing two more calculations with FDS, one with zero leakage, and one with 10 cm^2 leakage. As seen in Figure 3, the impact is quite large. When changing the total leakage with 4 – 6 cm^2, the first pressure peak in the experiment changes in the order of 50 Pa.

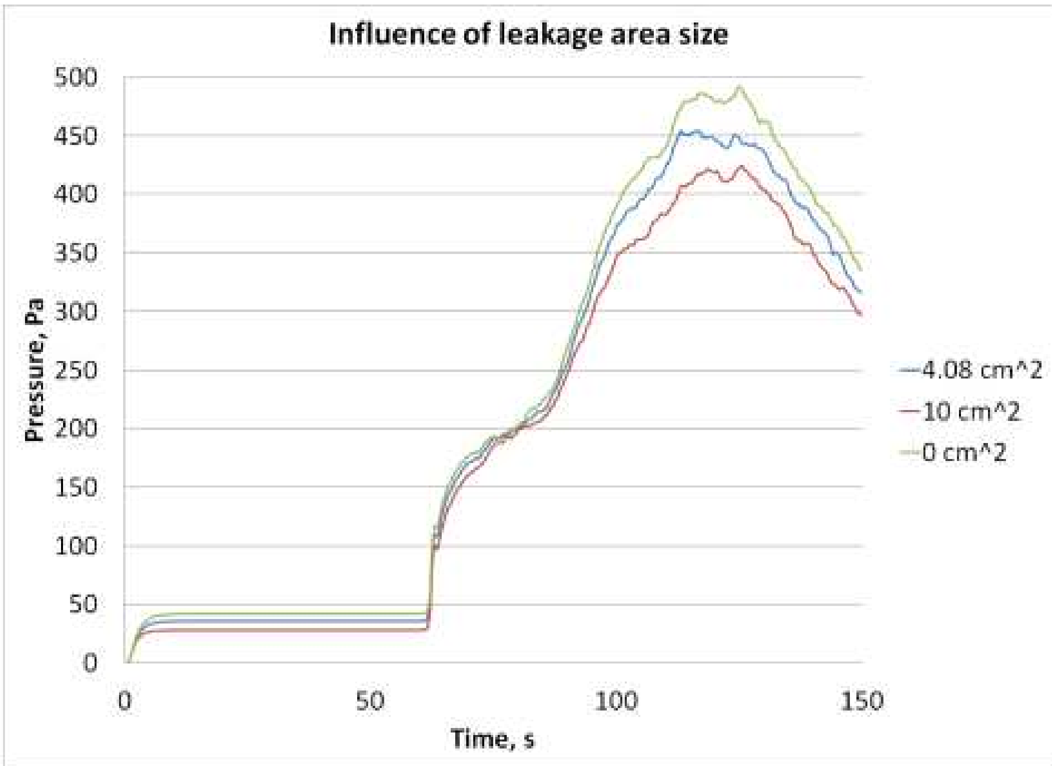

Figure 3 Influence of changing the room leak area in FDS

The geometry used in the simulation can be seen in Figure 4. Since the full ventilation system (Figure 5) was modeled with FDS, it was necessary to compare the experimental data in every node of interest with the data produced with FDS, prior to the fire being ignited. If this proved to give a good prediction, the likelihood of getting good results when compared to the full experiment would be far larger. As can be seen in Table 1, the results agree very well with the experimental data. Only one node shows a relative pressure difference larger than 10%, though the pressure difference is only about 40 Pa.

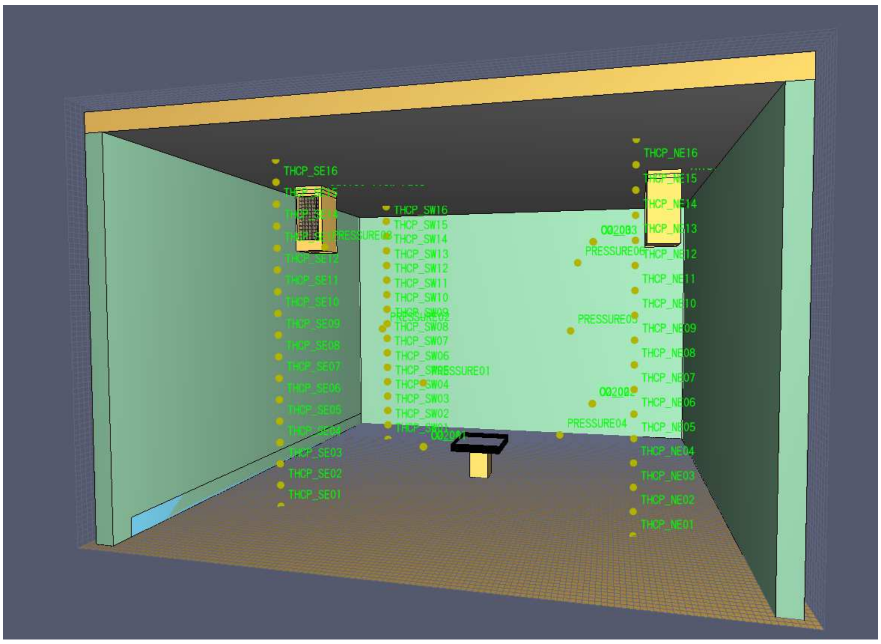

Figure 4 Geometry for the simulations with ventilation module

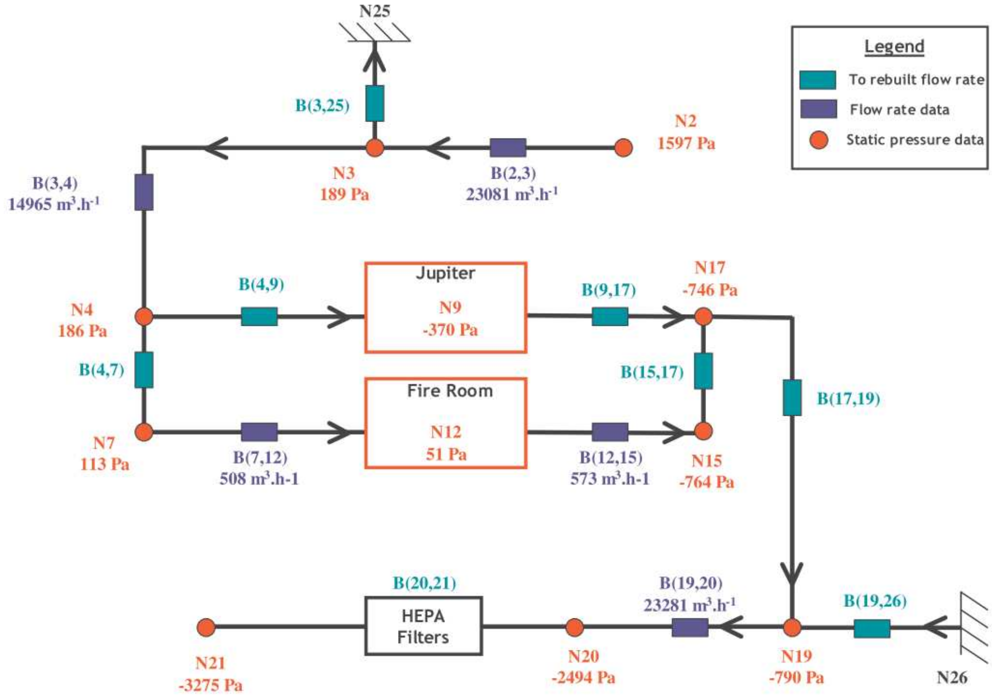

Figure 5 Layout of the ventilation network (courtesy to IRSN) and a comparison between FDS data and experimental data

	N2	N3	N4	N7	Fire room	N15	N17	N19	N20	N21
FDS, Pa	1575.73	191.71	188.81	162.77	35.5	-706.71	-726.09	-769.46	-2446.5	-3228.18
Experiment, Pa	1597	189	186	113	37.9	-764	-746	-790	-2494	-3275
Difference, %	1.33	1.43	1.51	44.04	6.33	7.50	2.67	2.60	1.90	1.43

Table 1 Comparison of FDS5 results and measured pressure in each ventilation node (courtesy to IRSN)

An overview of the temperatures calculated with both CFX and FDS compared to the experimental data can be seen in Figure 6. FDS manages to give a good prediction of the temperatures (within 10 – 15 %) on a relatively coarse grid (10 cm cubes), providing a good basis for evaluating the ventilation system behavior. Unfortunately the same cannot be said about CFX. CFX over-predicts the temperature by far (30 – 50 %), however, it cannot be ruled out that errors made by the software operator influences this deviation. Also, the way CFX handles combustion, for example internally calculating heat of combustion, prevented use of the experimental value obtained. This will likely impact the temperatures in the fire room. Also, heat transfer to the surrounding walls has been taken into account, but it was unclear if it was properly set up even though initial tests were performed.

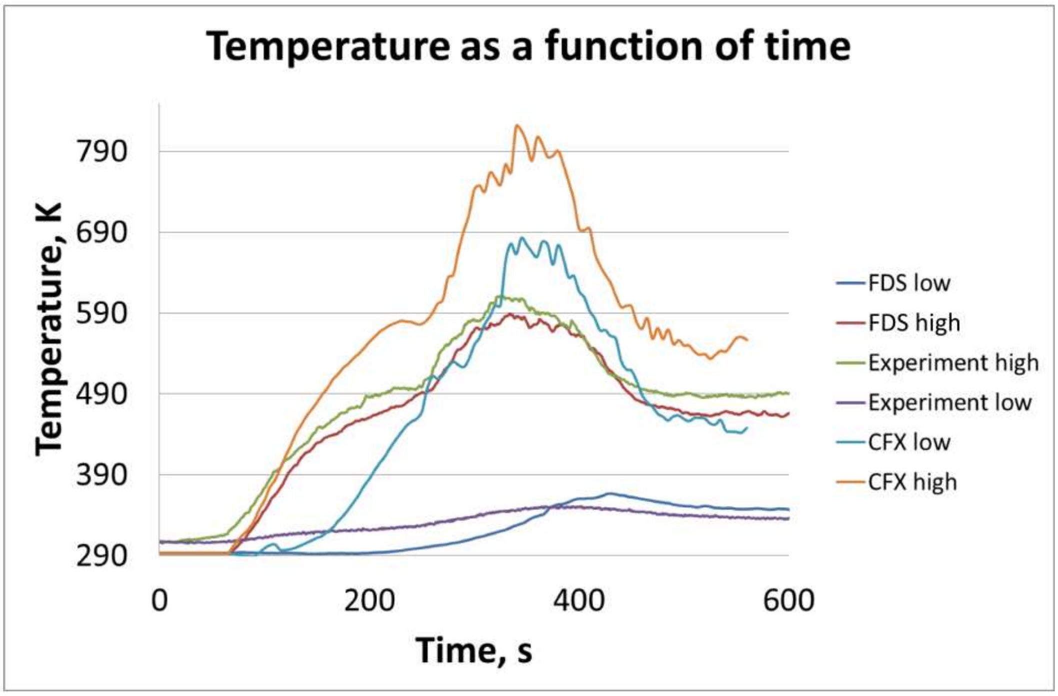

Figure 6 Temperature (highest and lowest measure point) as a function of time for the first 600 seconds

Since full capabilities concerning ventilation system modeling is not present in CFX (simplifications were made at the in- and outlet branch, specifying appropriate boundary conditions to get realistic pressures in the fire room), only results from calculations made with FDS are presented when comparing pressure in fire room and mass flow in the ventilation branches. As seen in Figure 7, the calculated pressure in the fire room is very close to the experimental data. All pressure peaks are fairly well predicted, and this is using only data available prior to the fire being ignited (except for HRR).

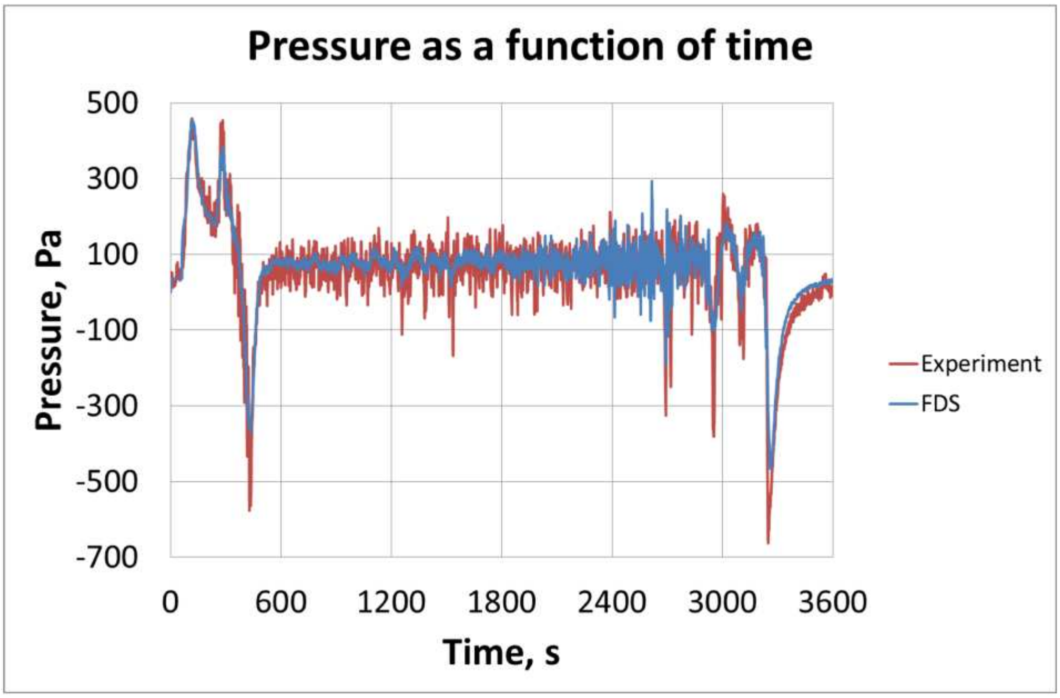

Figure 7 Pressure in the fire room as a function of time

Looking at the inlet and outlet branches (Figure 8) it is shown that FDS manages to predict the backflow in the inlet branch correctly. However, due to differences in the reported data from the experiment (actual measured mass flow not the same as reported in figure 3), the mass flow at the in- and outlet before the fire was ignited does not correspond to the FDS values. This in turn affects the "steady-state" mass flow in the later part of the experiment (after 600 seconds) making the FDS prediction somewhat incorrect. But it can be seen that the difference is constant, indicating that with the right starting values, FDS would give a better prediction.

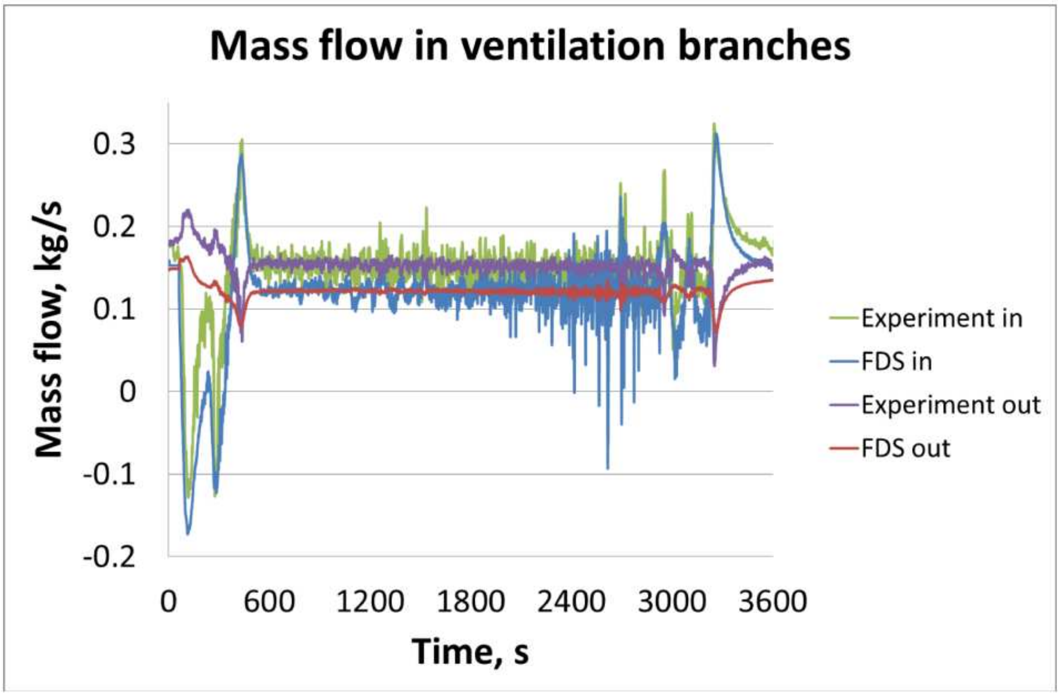

Figure 8 Mass flow in the ventilation branches as a function of time during the experiment

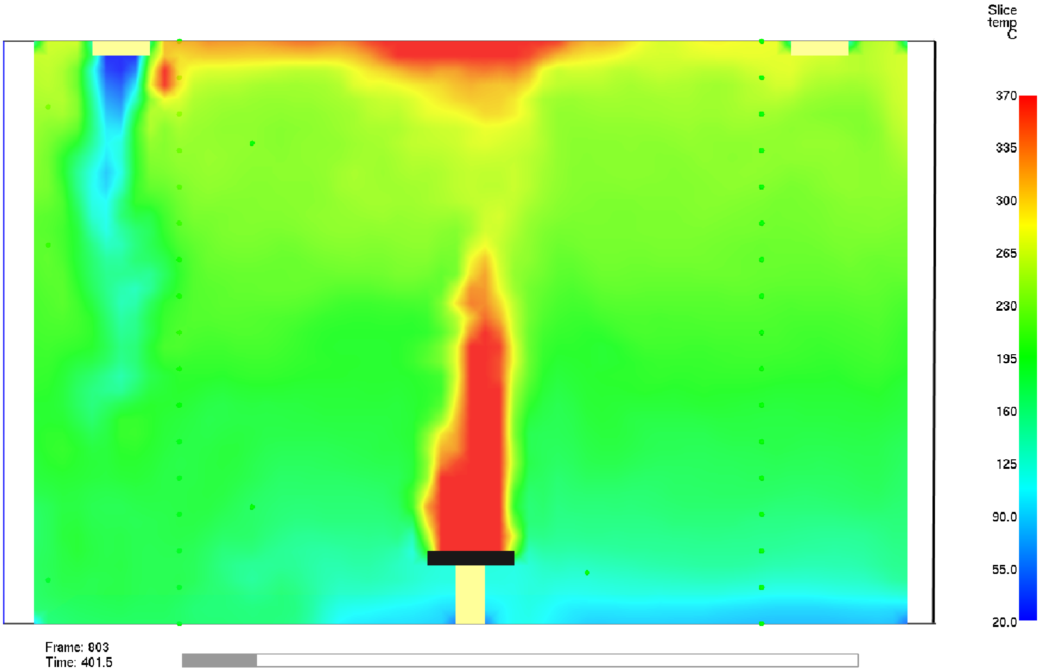

Figure 9 Snapshot of a temperature slice during the simulations done with FDS5. The incoming cold air is clearly visible at the top left corner

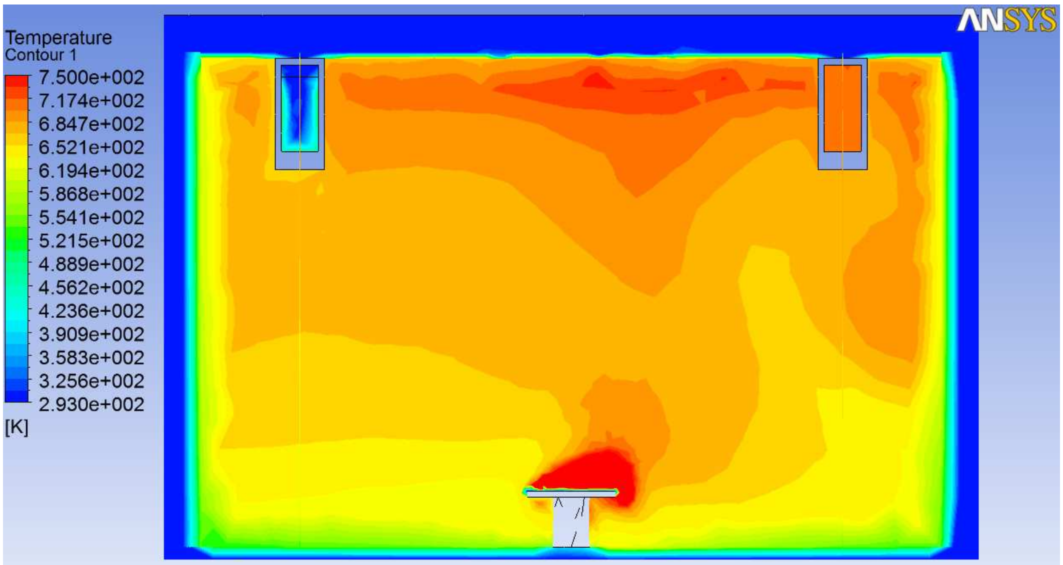

Figure 10 Snapshot of a temperature slice during the simulations done with CFX

In Figure 10, the incoming cold air is clearly visible at the top left corner. It can also be seen that the temperature gradient from ceiling to floor is not as steep as shown with FDS5. The maximum temperature is also overestimated to a quite large degree.

RESULTS AND DISCUSSION OF THE DEVELOPMENT OF THE EMPIRICAL MODEL

All included variables were statistical significant and the correlation had a correlations coefficient, R2-value, of approximately 0.9 with respect to the data from the simulations (Figure 11). The most important variable was the heat release rate. A validity check was performed by studying data from real fire tests [15], [16]. Three sets of experimental data were studied and the result of the validity check can be found in Figure 12. It is considered to be a good agreement between the calculated and measured temperatures since the maximum difference is less than 20 %.

A reliability check was performed by looking at the grid sensitivity of six of the preformed simulations, when decreasing the grid size from 0.1 to 0.05 m. The presented work is based on FDS simulations of well-ventilated pre-flashover fires. Thus the results are only valid for such conditions. This could be seen when adding the PRISME data in the correlation, which turned out to be outliers.

The method used to find a simple correlation for temperatures in the room adjacent to the fire room was very successfully and could possibly be applied to other areas in fire science to be able to find other simple correlations that can be used by engineers in an initial stage of their design. Some more experimental data is necessary to fully validate the developed empirical formulae and it should be investigated how the formulae could be adapted for under-ventilated conditions. This is not the case for the moment as could be seen from the data obtained via the PRISME project.

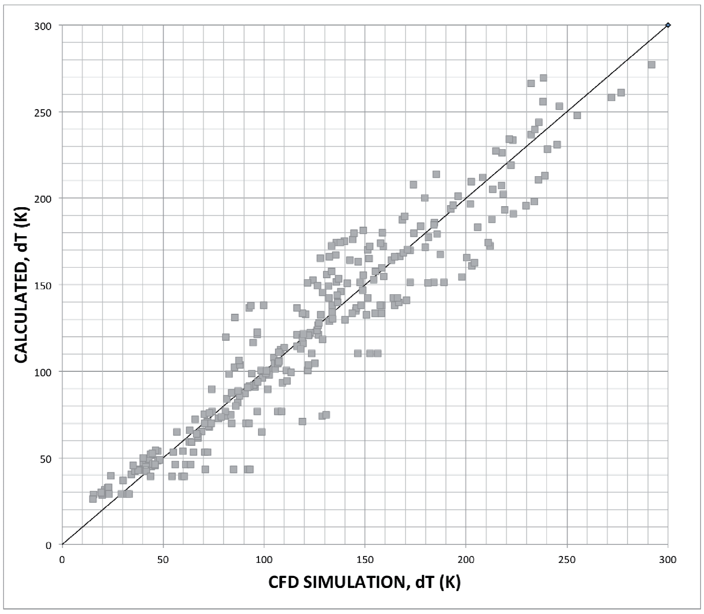

Figure 11 Correlation graph between calculated and simulated temperature increase

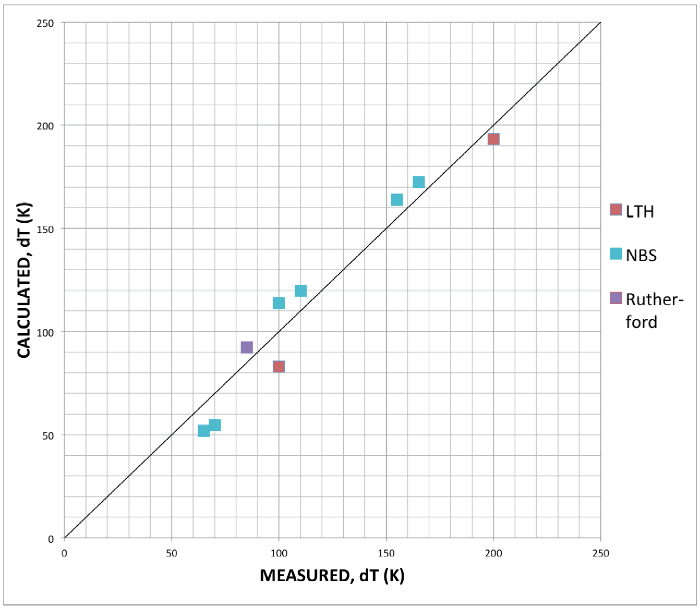

Figure 12 Comparison between calculated and measured temperature increase (only ventilated fires)

CONCLUSIONS

In this paper two activities within the Swedish PRISME project were summarized. One reports on the development of a simple empirical correlation for temperatures in the room adjacent to the fire room. The development was done by numerical experiment technique and validated against a first set of test data. The results are satisfactory and further validation will be done. Application of the models is not only within fire safety design of nuclear power plants but also in traditional buildings. Another activity was the validation of the newly developed ventilation module in FDS against the PRISME Source D1 tests. The results of this validation show that the module is working properly and give satisfactory results. The intention is to validate the model against more data in PRISME project database and to implement the module in a realistic fire safety design of a traditional building where mechanical ventilation is involved.

The activities of the Swedish PRISME project, which were initiated by the nuclear industry and government in Sweden, show clear spin-off to other fire safety areas such as traditional buildings and other industrial applications.

ACKNOWLEDGEMENTS

The work presented in this paper has been conducted within two projects at Lund University. One is the Swedish Part of the PRISME project. The Swedish part of the PRISME project was possible thanks to financial support of Brandforsk (Swedish Board for Fire Research) and NBSG (National Fire safety Group, with representation of SSM - Swedish Radiation Safety Authority, SKB - Swedish Nuclear Fuel and Waste Management Co., Forsmark- Oskarshamn- Ringhals nuclear power plants). Parts of the validation work were also possible thanks to support of SSF (Strategic Research Fund).

The other project is called "Varför blir små brander stora?" (Why become small fires large) and is financed by The Swedish Fire Research Board (Brandforsk) and NBSG (the Swedish NPPs Fire Safety Group). The purpose of the project is to find underlying factors to why some fires grow large.

Finally the authors would like to thank IRSN in order to provide them information about the technical details of the ventilation system, which was used when IRSN performed the fire tests in the PRISME project.

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MODELING OF IGNITION AND FLAME SPREAD IN THE INITIAL PHASE OF A FIRE

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ABSTRACT

Computational Fluid Dynamics (CFD) simulations with the Fire Dynamic Simulator 5 (FDS 5) are carried out to investigate ignition and upward flame spread in the initial phase of a fire. The research includes both thin and thick solids, represented by cellulose-sheets and PMMA samples, the latter of 5 mm thickness. Upward flame spread is examined for different inclined surfaces from horizontal to vertical orientation. It is found possible to simulate ignition for both materials with FDS 5, but flame spread can only be achieved in some PMMA scenarios. The FDS predictions are compared to experimental data for upward flame spread on PMMA samples. The simulations with FDS 5 show limitations in simulating flame spread and ignition for small fires (low HRR) and the influence of the user on modeling and results of the simulations.

INTRODUCTION

It is of common interest in fire safety research to predict the fire development as good as possible. Destructive fires generally develop in several phases from an initial phase to a total fire. In the initial phase the ignition takes place and the fire starts to spread over the ignition point or area, involving an increasing area of the surface. This phase of a fire is characterized by small burning areas, low heat release rates (HRR) and marginal increase in temperature in the area of the fire. Unrestricted further ignition and flame spread can lead from this initial phase to a fully development of a fire.

Ignition of and flame spread over combustible solids are therefore fundamental and highly dangerous phenomena. Ignition and flame spread depend, e.g. on fuel (material), and boundary conditions like ventilations or the surface orientation. Upward flame spread rates are faster than downward or horizontal flame spread rates. Inclination can even increase flame spread rates.

If ignition and flame spread can be predicted and hereby the fire development, the fire protection methods can be adjusted and evidence can be given with respect to meeting protection targets. One method to model fire development is with fire safety engineering methods. Recently, more advanced options of fire modeling have become available for fire safety engineering. Computational Fluid Dynamic (CFD) models, such as the Fire Dynamic Simulator (FDS), can also be applied to predict flame spread and fire growth [1].

The aim of this study was to investigate ignition and flame spread in the initial phase of a fire and to examine the possibilities to predict ignition and propagation of fire under defined conditions for these small fires. CFD simulations with FDS 5.3 and FDS 5.4 are carried out to investigate ignition and flame spread on small samples of cellulose sheets, as thermally thin material, and of 5 mm PMMA samples, as thermally thick material, under different surface orientation (horizontal, vertical, inclined).

COMPUTATIONAL (SIMULATION) SPECIFICATIONS

FDS is a CFD model to solve practical problems in fire safety engineering, but it also provides a tool to study fundamental fire dynamics and combustion [2]. The governing equations are solved in a three dimensional Cartesian coordinate system. FDS requires input parameters to describe a particular scenario, including numerical grid, ambient environment, building geometry, material properties, combustion kinetics and desired output quantities. The following chapters summarize the scenario specifications in this work for modeling ignition and flame spread with FDS 5.

GEOMETRY

All FDS calculations are performed within a domain that is made up of rectangular cells. This domain specifies the space to be modeled, where the governing equations are solved in. The FDS domain used here is constructed close to experimental work in [3]. The size of the three dimensional (3D) domain is 12 cm deep, 120 cm wide and 40 cm high. The 5 cm wide and 30 or 55 cm long material sample is located on the floor in the front of the model (see Figure 1). The inclination of the surface is done through changing the gravitation vectors.

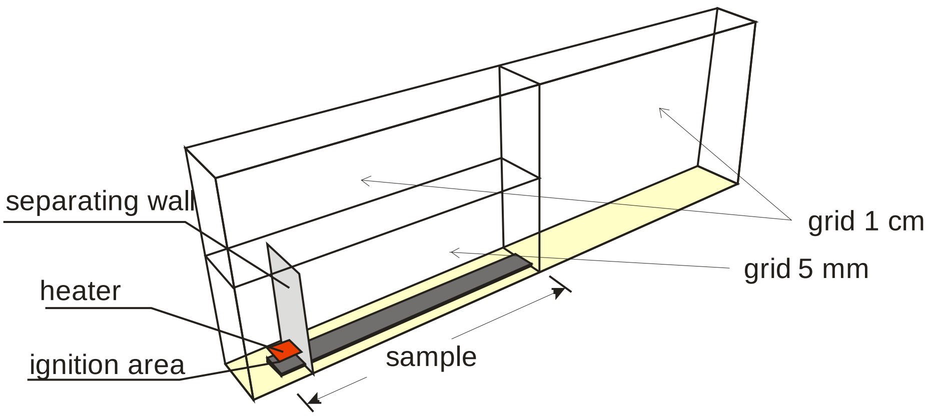


Figure 1 3D simulation model

Further two-dimensional (2D) simulations (Figure 2) allow parameter studies and provide additional information as well as a faster check on correct input data like the applied material properties.

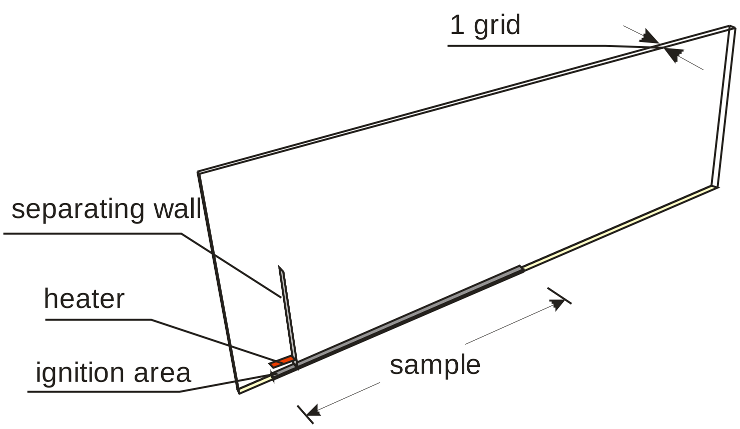


Figure 2 2D simulation model

Grid Resolution

Grid size plays an important role and FDS shows sensitivity to grid size in many applications [4]. A small grid is preferred for better simulations a coarse grid is favored in terms of computational costs.

A grid of 5 mm is chosen for the primary pyrolysis and combustion region (see Figure 1) and of 1 cm for the further domain in 3D scenarios. 2D scenarios are of 5 mm grid only. These grid resolutions result in 158,400 cells in total for the 3D model and in 19,200 cells for the 2D model. It takes approximately 14 days (3D) and 1 - 2 days (2D) to simulate 1.300 s real times for a serial (non-parallel) run.

Materials and Material Properties

Every cell in DFS has to be defined as gas, liquid or solid and boundary conditions have to be applied on these cells. Solids can be treated as multi-layer solids, thus the physical parameters for every layer respectively every material have to be specified. Describing these materials in the input file is most challenging task for the user [1].

In this work, small cellulose and PMMA samples (5/30 or 5/55 cm) are studied. Solid phase reactions (pyrolysis) as well as gas phase reactions (combustion) are computed for these materials. The required material properties are taken from literature and adapted from additional material tests in [3]. These input data are cross-checked by means of additional FDS simulations. Figure 3 illustrates the results of the mass fraction over temperature from thermo-gravimetry analysis (TGA) on cellulose sheet under different heating rates (1 - 5 K/min) in comparison to FDS simulations with cellulose material.

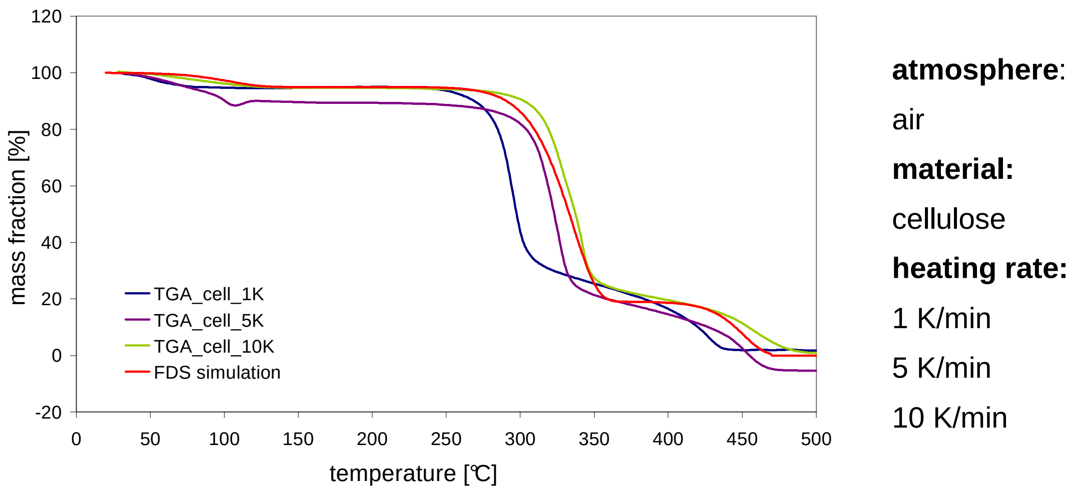


Figure 3 Development of mass fraction over temperature for cellulose, results from TGA [3] and FDS simulation

The results show a good agreement. However material input properties have always to be considered critical because they are still only approximation.

Pyrolysis Model

When solid materials burn each material component of each material layer may undergo several competing reactions and each of these reactions may produce other solids (residue) and gaseous volatiles. For some simulations it is sufficient to define pyrolysis through a specified burning rate or a given heat release [2], but not if ignition and flame spread should be calculated by FDS. In this work the solid phase reactions (pyrolysis) of the materials cellulose or PMMA are computed by FDS.

Combustion

FDS uses per default a mixture fraction model for the gas phase reactions (combustion), requiring the fuel and products of the combustion. There can be many types of combustibles that burn (and pyrolysis) in FDS, but there can only be one gaseous fuel. In this work the gas phase reaction was adapted to the burning fuel since only one material (either cellulose or PMMA) is burning in the simulations.

Ignition Source

For modeling ignition and flame spread the surface has to be set on fire with an appropriate ignition source. Here, a 5 cm deep and 4 cm wide radiant heater is created as ignition source. The distance between heater and surface is 2 or 5 cm. The surface of the heater is set to different temperatures to result in different heat fluxes on the surfaces of approximately 12 - 51 kw/m^2 and different time periods of impingement are set. During the ignition a separating wall prevents preheating of the surface outside the ignition area (see Figure 1).

Definition of the Pyrolysis Front

In FDS there is no parameter defining the ignition of the material. In this work the ignition- respectively the pyrolysis front is therefore specified with a critical burning rate.

The critical burning rate $\dot{m}''_{cr}$ for thermoplastics is mentioned in [5] with 1.3 g/m^{2}s $\leq \dot{m}''_{cr} \leq$ 3.9 g/m^{2}s for natural convection and 2.9 g/m^{2}s $\leq \dot{m}''_{cr} \leq$ 4.5 g/m^{2}s for forced convection, in [6] with 0.8 g/m^{2}s $\leq \dot{m}''_{cr} \leq$ 2.9 g/m^{2}s and in [7] with $\dot{m}''_{cr} \approx 4 \sim 5$ g/m^{2}s for PMMA. In the present study a critical burning rate of 4 g/m^{2}s is specified as pyrolysis front-criterion for PMMA.

Due to the different material properties between PMMA and cellulose a different limit for the latter is defined. Based on a critical burning rate of $\dot{m}''_{cr} \approx 2.5$ g/m^{2}s for wood in [8] the pyrolysis front for cellulose is specified with a critical burning rate of 2.5 g/m^{2}s.

The ignition of the surface is then located with these criterions. The following Figures show for example the burning rate and heat flux on the surface of cellulose (Figure 4) and PMMA (Figure 5) during FDS simulations. An increase of the heat flux can be found after the specified critical burning rate of 2.5 g/m^{2}s (cellulose) or 4 g/m^{2}s (PMMA) is exceeded. This indicates an additional heat source through a burning surface.

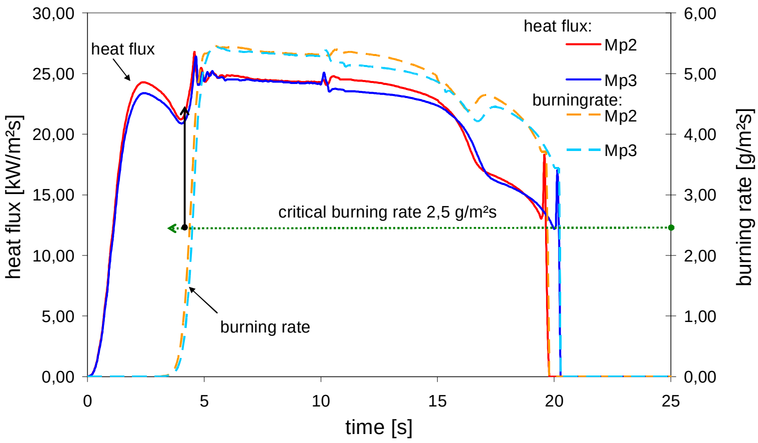


Figure 4 Heat flux and burning rate over time at cellulose sheets simulation

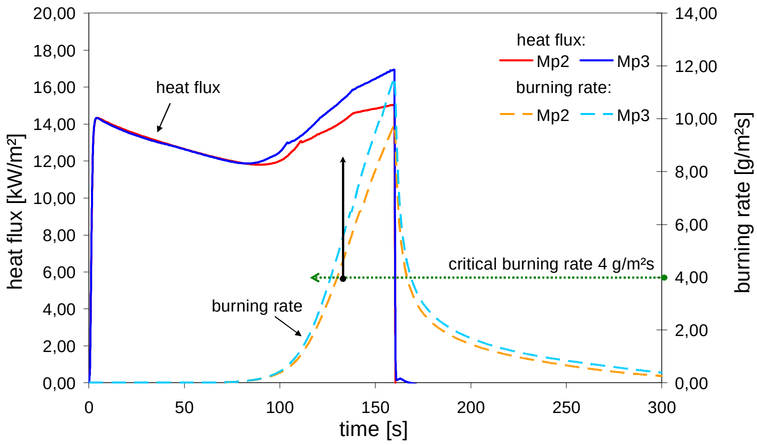


Figure 5 Heat flux and burning rate over time at PMMA simulation

Output Quantities

FDS can compute a lot of quantities. These quantities are computed in each grid cell within each time step. The user has to select which data resp. output-quantity to be saved before the simulation. Output-quantities of the presented simulations are:

- Gas temperature;
- Surface temperature;
- Surface temperature measured with thermocouple;
- Heat flux on the surface;
- Burning rate;
- Material thickness;
- Gas species concentration: O, CO, CO_2, mixture fraction, fuel;
- Total heat release rate (HRR).

Quantities are calculated as devices (DEVC), iso-surfaces, slicefiles and/or boundaries.

EXPERIMENTAL WORK

At TU Vienna (cf. [3]) small scale flame spread experiments were conducted. For the experimental work cellulose sheets and PMMA, cut into rectangular shape of 300 mm length and of 50 mm in width, were selected as the fuels. PMMA was supplied in 5 mm thickness. The materials where tested in a test apparatus which enables to study the flame spread under different, infinitely adjustable, inclinations (see Figure 6).

Figure 6 Experimental configuration, PMMA, inclination of 30 °

The flame spread rate was defined with different pyrolysis front criterions. Besides visual observations, flame spread rate was measured through transient gas phase temperature above the surface on PMMA and through a new developed instrumentation on Cellulose sheets. With the latter it was possible to define the pyrolysis front through a "conductive line measurement", where the pyrolysis front destroys a conductive line and therefore time/location can be measured.

The flame spread rate on cellulose-sheets increased nearly linear with increasing inclination (see Figure 7). Punctual progressions where found at 30 ° and 60 °.

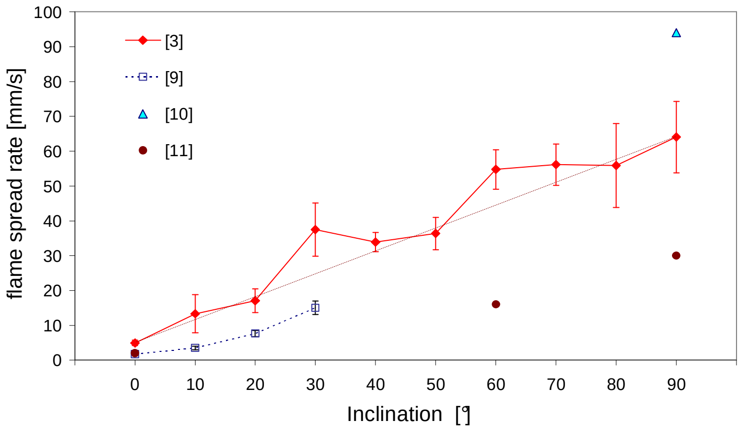


Figure 7 Flame spread rates on cellulose-sheets under different surface inclinations

Figure 8 shows the results for upward flame spread on PMMA. Flame spread was also found to increase with increasing inclinations, exceedingly at inclinations of ≤ 30 ° and > 60 °.

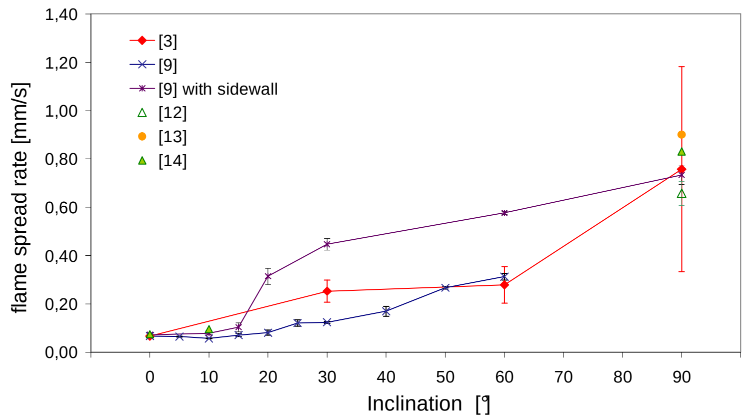


Figure 8 Flame spread rates under different inclinations on the surface of PMMA

The experiments on upward flame spread showed an increase of the flame spread rate with increasing the inclination. The development of the flame spread, for both materials, was strongly influenced by the flame-surface-interaction. The flame interaction to the surface shows a decreasing angle between the flame flow and the surface, with increasing inclination. This angle decreases until the flame start to attach more or less the surface [15]. Critical angles where found at about 30 ° and 60 ° inclination. Flame spread at this inclination was enhanced from air entrainment.

RESULTS AND DISCUSSION

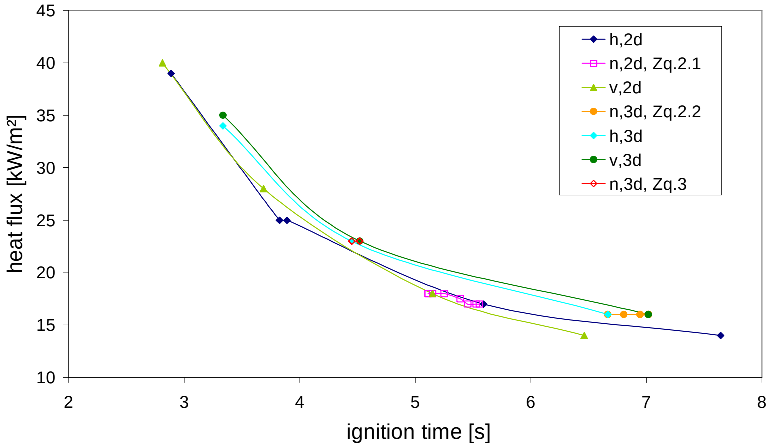
Ignition in FDS

To simulate ignition and flame spread in FDS one problem was, to define an appropriate ignition source which on the one hand ignites the material and on the other hand enables a sustained burning after its removing. To ignite the cellulose and/or PMMA samples in FDS several radiation heater are used. They differ in heat flux and duration of the impingement on the surface. Table 1 shows a brief overview of a few heaters that are used as ignition sources.

Table 1 Ignition sources for the FDS simulations

Ignition Source	Duration of Impingement [s]	Heat Flux on the Surface [kW/m²]
Cellulose		
Zq.2.1	10	17 – 18
Zq.2.2	remains*	17 – 18
Zq.3	10	23 – 28
PMMA		
Zq.A.2.1	160	16 – 18
Zq.A.2.2	remains*	16 – 18
Zq.A.3.1	100	19 – 24
* heater is not removed, but remains during the whole simulation		

Ignition times for cellulose are shown in Figure 9 for different heat fluxes on the surface as well as for different orientations. Ignition of cellulose is computed between 2.8 s for a heat flux of 40 kW/m^2 and 7.7 s for a heat flux of 14 kW/m^2.



Sample orientation:

h: Horizontal

v: vertical

n: inclined (0°–90°)

Model:

2d: 2-D-geometry

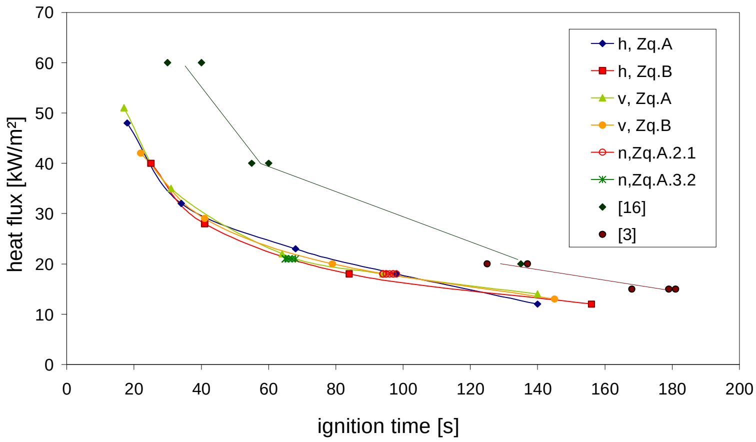
3d 3-D-geometry

Ignition source:

Zq., see Table 1

Figure 9 Ignition results for cellulose-sheets

Heat fluxes of 51 kW/m^2 ignite PMMA after approx. 17 s (Figure 10) whereas 156 s are needed to reach the ignition criterion with heat fluxes of 12 kW/m^2. Ignition times are calculated 10 - 20 s earlier than experimental results in [16] and [3] on PMMA. The reason for this delay is probably due to the different determination of the ignition time. In the simulation a critical burning rate is chosen while in experiments the observation of flames is often chosen to detect ignition on the surface. Still the ignition times in the simulations are in quite a good agreement to the experiments and therefore the chosen critical burning rate is a good quantity to determine ignition.



Sample orientation:

h: horizontal

v: vertical

n: inclined (0 ° - 90 °)

Ignition source:

Zq, see Table 1

Distance to surface:

Zq.A : 20 mm

Zq.B : 50 mm

Figure 10 Ignition results for PMMA

The ignition-simulations with FDS 5 show in general that the time to ignite for both materials is dependent on the intensity of the ignition source but rather independent of the surface orientation.

Sustained Burning in FDS

The ignition source and the separating wall are removed after igniting the materials. The fire should (1) sustain itself and (2) spread out of the ignition area. Unfortunately nearly all computed fires extinguish after a certain period. The time for extinguish is found to be dependent on the material, the ignition source and the inclination of the sample surface. Burning cellulose-sheets extinguish at the latest 14 s after removing of the ignition source and the separating wall. PMMA fires remain a little bit longer. Figure 11 shows the time to extinguish for PMMA scenarios under different surface inclinations. The time to extinguish is defined as time of sustained burning after removing the ignition source.

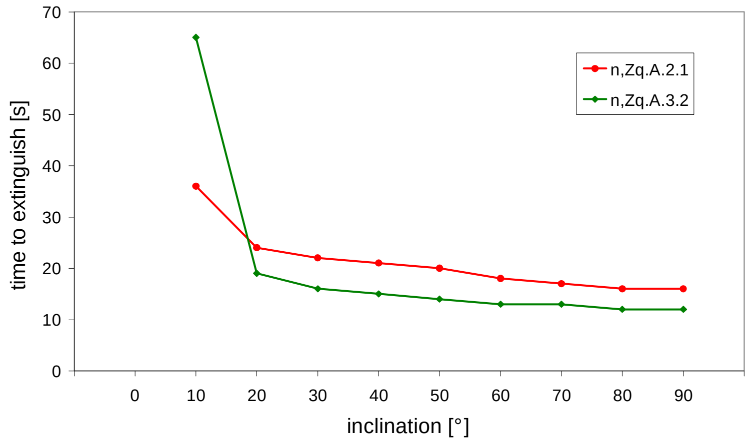


Figure 11 Time to extinguish for simulated PMMA fires, depending on the inclination

Horizontal fires on PMMA sustain themselves for 36 - 65 s depending on the ignition source (Table 1). The fires extinguish even faster on inclined surfaces. Here FDS computes a sustained burning for 12 - 24 s after the removing of the ignition source.

Further simulations on cellulose and PMMA are done also with remaining ignition sources as additional external heater on the surface.

Flame Spread Scenarios for FDS Simulations

Flame spread outside the ignition area could solely be achieved with PMMA samples. Flame spread on cellulose could not be simulated. Table 2 shows the specifications of the PMMA scenarios with which it was found possible to simulate flame spread in FDS on these small samples of PMMA.

Table 2 Scenarios for PMMA flame spread simulations

Orientation / Inclination	Geometry Model	Turbulence Model	Ignition Source
0°	2D	DNS	Zq.A.2.1.
10 °- 80 °	3D	LES	Zq.A.2.2 (remains)
90 °			

Flame Spread on PMMA under Different Inclinations

Figure 12 shows the flame spread rates on PMMA under different inclinations and compares these results with experiments on PMMA in [3] and simulations in [17]. The latter are calculated with the CFD model Safir.

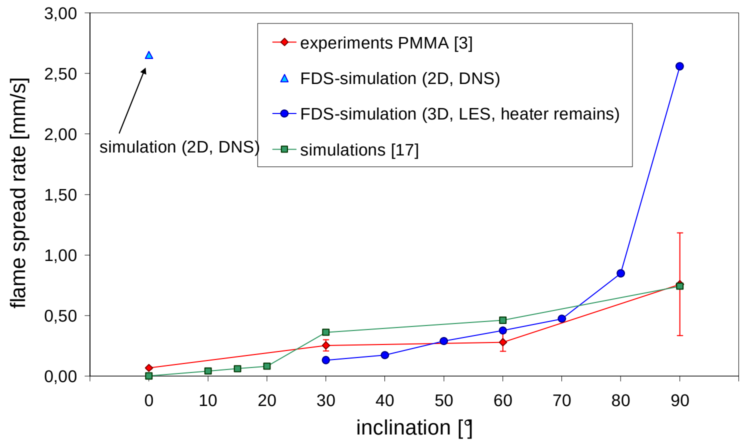


Figure 12 Upward flame spread on PMMA: results from experiments and simulations

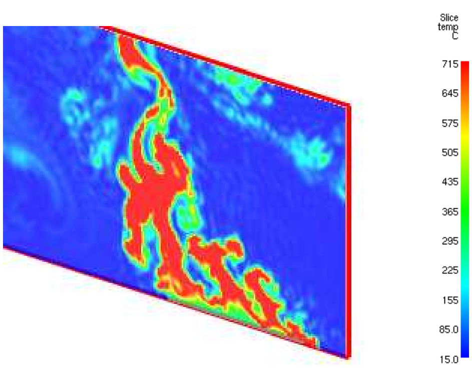
Flame spread outside the ignition area can only be initiated on horizontal and 30 °-90 °inclined PMMA samples. Hereby the horizontal f lame spread can only be achieved in 2D geometry and with a Direct Numerical Simulation (DNS). Flame spread on inclined surface can only be initiated with a remaining heater, in contrary to the experiments where the ignition source was removed after igniting the material. Still FDS simulates similar flame spread rates to the experiments in [3] for 30 – 70 ° inclined PMMA surfaces. The simulated flame spread rates on horizontal PMMA are 38-times faster, the flame spread rate on vertical samples 3 times faster than experimental results in [3]. Flame spread rates in [17] are calculated slower for 0 - 20 °inclinations and faster for ≥ 30 ° inclinations, vertical flame spread rates are in good agreement to the experiments in [3].

The development of the flame spread rate differs between experiments [3] and simulations. The experimental flame spread rates increase for inclinations between 0 – 30 ° and 60 – 90 °, the simulation in [17] show an steep increase of the flame spread rate for 20 – 30 °inclinations and FDS simulations an i ncrease of the flame spread rate with inclining inclination and a steep growth > 70 °inc lination. At inclinations of 30 – 50 °the flames did not spread over the entire length of the sample in FDS but extinguishes earlier, in contrary to the experiment.

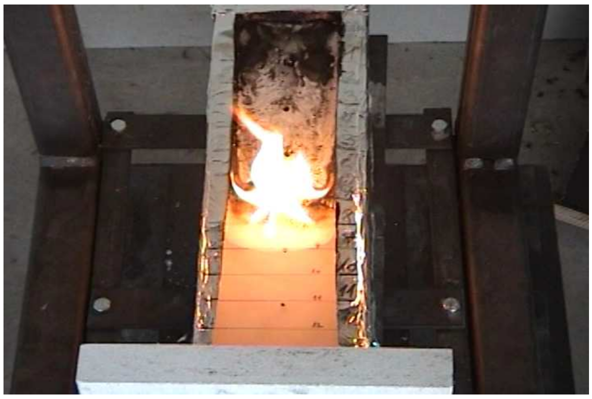
Comparison of Fire Development on PMMA

The following figures show the development of PMMA flame spread in the FDS 5 simulations compared to experiments in [3].

For horizontal flame spread simulation the ignition source and the separating wall are removed after 120 s impingement. For some time afterwards the fire remains on the ignition area before it starts spreading over the entire length of the sample until the whole surface is burning at once (see Figure 13 a). The experimental flame spread is much slower (see Figure 12) and the burning area nearly stays constant small due to a nearly similar flame spread and burn out rate, while the pyrolysis front is wandering over the entire surface (see Figure 13 b).



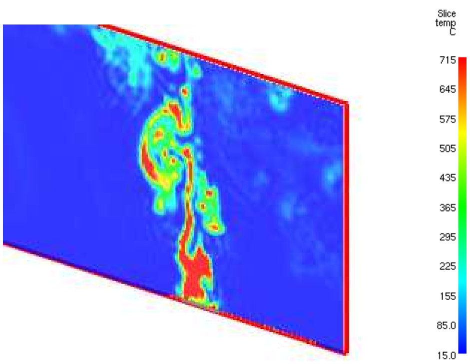
a) t = 323 s



b) t = 2,448.10 s

Figure 13 PMMA, horizontal a) FDS and b) experiments in [3]

Burn out can also be simulated. It starts at the ignition area and migrates over the entire length of the sample (see Figure 14 a). The total burn out of the samples is reached approximately after 500 s of simulation. The experiments needed over an hour until the whole samples was burned (see Figure 14 b)

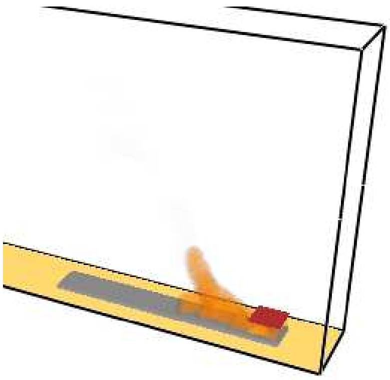


a) t = 470 s

b) t = 3,432.10 s

Figure 14 PMMA, horizontal, burn out a) FDS and b) experiments in [3]

Further simulations on inclined surfaces can only be achieved with remaining ignition sources. Simulations on 30 ° inclined surfaces show a similar development to the experiments for the first 800 s (see Figure 15) but the flames in the simulations do not spread over more than approx. half of the sample, while they go on in the experiments.

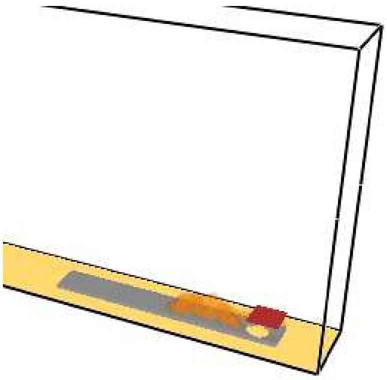


a) t = 817 s

b) t = 818.07 s

Figure 15 PMMA, 30 ° inclination; a) FDS and b) experiments i n [3] ($t \approx$ 820 s)

After 1100 s burn out starts in the ignition area and the flames gradually extinguish (see Figure 16 a). In the experiments, the pyrolysis front as well as the burn out front migrates over the entire length until the whole sample is consumed.

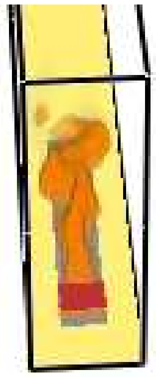


a) t = 1.132 s

b) t = 1.139,02 s

Figure 16 PMMA, 30 ° inclination; a) FDS and b) experiments in [3] (t = 1130 - 1140 s)

For inclinations of 60 °, the flames spread over the entire length and show a similar flame structure as well as flame spread rate as the experiments (see Figure 17). Still they differ in burn out for FDS computes burn out only in the ignition area while in experiments the complete sample is consumed.

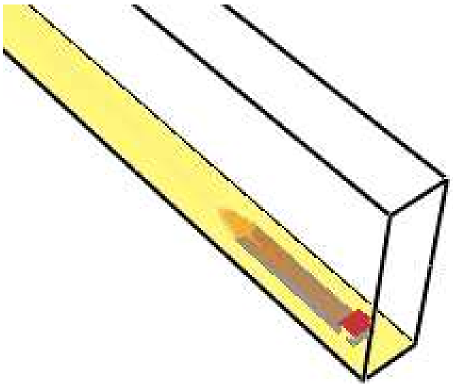


a) t = 752 s

b) t = 751.50 s

Figure 17 PMMA, 60 ° inclination; a) FDS and b) experiments in [3] ($t \approx 750$ s)

Flame spread rates on vertical samples calculated by FDS are three times faster than in experiments (see Figure 12). Burn out starts in the ignition area but also from the top side of the sample (Figure 18 a), in contrary to the experiment, where the samples burn out only from the bottom to the top (Figure 18 b).



a) *t* = 1,440 s

b) *t* = 1,660.0 s

Figure 18 PMMA, 90 ° inclination; a) FDS and b) experiments in [3] (*t* = 1400 – 1700 s)

Influence on FDS Simulations

The outcomes of this work indicate that if a flame spread can be achieved as well as the simulated flame spread rates are dependent on a lot of parameters for e.g. on specific values and boundary conditions. These parameters are either influenced by input specifications through the user or the computational program itself, the latter trough the implemented models and resolutions in FDS. For example, it is found possible, as shown before, to achieve flame spread on PMMA in 2D-DNS simulations but only for horizontal samples and not for inclined surfaces. On the other hand flames do spread on inclined surfaces (≥ 30 °) if the ignition source remains. Still the flames only spread over the entire length for inclinations ≥ 60°. Flame spread on cellulose cannot be achieved however for the thicker PMMA.

Table 3 shows parameters that affect the simulation of ignition and flame spread in FDS and their effects on these simulations.

All these parameters can have direct and indirect influence on the initiating of ignition and flame spread but also on the results (for e.g. ignition time, flame spread rates). They can interact with each other and these interactions can be quite complex (see Figure 19).

Table 3 Effects and influences on simulating ignition and flame spread in FDS 5

Specific Values and Boundary Conditions	**Effects on**	
	Ignition	**Flame Spread**
Input data / user		
Geometry: 2D/3D, surface orientation	○	+
Grid resolution	+	++
Ignition source: ignition area, intensity, duration	++	+
Material: construction, dimension, properties	+	++
Definition pyrolysis front	–	○
Output quantities	–	–

Program / FDS 5		
FDS-version*	○	+
Turbulence-model	–	○
Radiation-model	○	++
Conduction-model	○	○
Calculating pyrolysis	○	+
Calculating combustion	+	+
Heat release rate	○	++
– no effect ○ marginal effect + normal effect ++ significant effect	* using versions FDS 5.3 vs. FDS 5.4	

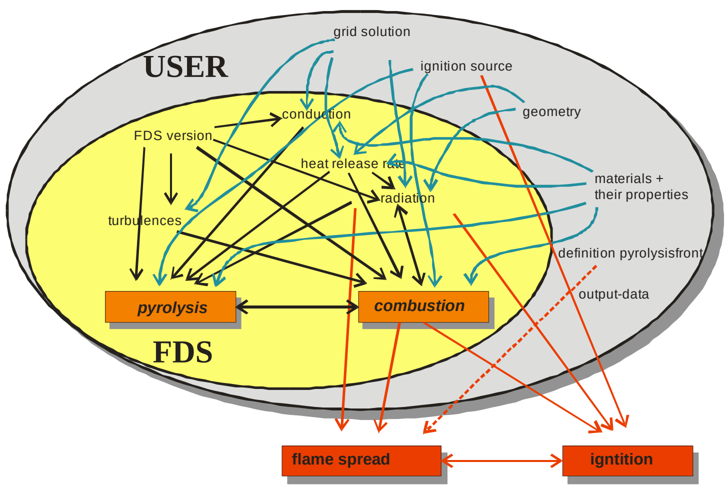


Figure 19 Interaction of influences on modeling ignition and flame spread in FDS 5

Simulations with FDS 5 show the limitations of this CFD model for modeling ignition and flame spread for small fires such as those within thin materials or fire in the initial phase of a fire. However, it has to be kept in mind that numerically determined solutions always represent an approximation and should be viewed accordingly. Even if no flame spread is calculated in the simulations, this does not mean that it corresponds to reality. The latter could be demonstrated in this work by comparing the simulations results with experiments in [3].

CONCLUSIONS

Concerning the FDS simulations with FDS 5 the principal findings of this work are:

- The time to ignite cellulose and PMMA is found to be dependent on the intensity of the ignition source and less on surface orientation.
- The initiation of a self-sustaining burning in the ignition area is dependent on the ignition source (intensity, duration of impingement) and the inclination of the surface.

- Flame spread without an additional external heater would require a self-sustained burning. A self-sustained burning can only be achieved here for horizontal PMMA samples, but not for cellulose samples or any inclined surface.
- A self-sufficient flame spread can solely be achieved on horizontal PMMA while only with a 2D geometry and DNS. This flame spread rate is computed 38 times faster than experimental results.
- Flame spread on inclined surfaces can only be initiated for PMMA samples and for inclinations ≥ 30 °, but with a remaining ignition source as additional external heater.
- PMMA flame spread rates between inclinations of 50 – 70 ° are in good agreement with experimental results in [3], yet they differ in the additional external heater needed in the FDS simulations.
- Flame spread over the entire length of the sample is computed only for 60 – 90 ° inclined surfaces. At lower inclinations the fire extinguished earlier.
- For the PMMA simulations the calculated fire developments differ more or less from the experiments.

The results of the simulations with FDS 5 show a great dependency of different influencing factors (flow rate, ignition source) and input data (e.g. material data). This deal can be or is also significantly affected by the user. The level of the user´s skills has a strong influence on the results, in particular for small heat release rates (HRR) such as in scenarios with thin materials or in critical phases like the initial phase of a fire.

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PYROLYSIS MODELING OF PVC CABLE MATERIALS

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ABSTRACT

In this work, the authors have studied the effects of the modeling decisions and parameter estimation methods on the pyrolysis modeling of PVC cables. The kinetic and thermal parameters are estimated from TGA and cone calorimeter experiments. The role of the plasticizers for the early HRR is examined. The results indicate that the current pyrolysis models can be used to predict the burning behavior of PVC cables. The effects of the modeling decisions are not always critical to the model accuracy if a specific set of thermal parameters is estimated for each model.

INTRODUCTION

Flexible Polyvinyl Chloride (PVC) is commonly used in the components of electrical cables, forming a significant fraction of the fire load in existing power plants. If the risk of the cable fires is studied using predictive numerical simulations, the thermal degradation, i.e. pyrolysis, of the PVC materials must be properly modeled. PVC is one of the most versatile thermoplastic materials due to its processability and range of different applications. Flexible PVC is produced by adding 30 – 40 wt. % additives, especially plasticizers to lower the glass-transition temperature [1]. Flexible PVC ignites more easily and burns at higher rate than rigid PVC, because the plasticizers are usually combustible [2].

The numerical simulation of the PVC cable fires requires the modeling of the cable pyrolysis, which is extremely challenging due to the geometrical complexity and the wide range of different PVC compositions and plasticizers. The different approaches for modeling the kinetics of PVC degradation have been studied by Marcilla and Beltrán [3], who concluded that two parallel reactions are needed to describe the first stage of PVC degradation, and a single reaction for the second. The degree of model complexity should be in balance with the amount of experimental evidence and the allowable estimation and computing times. Once the model structure has been fixed, the problem becomes a parameter estimation problem, as explained in Refs. [4] to [8].

This work studies the sensitivity of the pyrolysis model to the decisions concerning the reaction path, reaction order and estimation method. Different modeling decisions are tested in the light of their capability to reproduce the experimentally observed behavior in cone calorimeter. First, models assuming a parallel reaction path are estimated for the two sample materials taking into account the softeners. An alternative reaction path is then created and the results compared to the parallel model keeping the thermal parameters fixed. The sensitivity on the kinetic parameters is studied by using two alternative estimation methods. The significance of the reaction order parameter is examined.

MATERIALS AND METHODS

Experimental

An electrical cable used as an example material was a four conductor power cable (MCMK 4 × 1.5 mm^2) with a diameter of 13 mm. It has a PVC sheath and insulation, and an unknown filler material. The cable dimensions and weights of the cable components are listed in Table 1.

Table 1 Mass fractions of the cable components

	Sheath	Filler	Insulation	Conductor	Other Plastic
Material	PVC	-	PVC	Copper	-
Thickness [mm]	2.5	10	3[a]	15[a]	-
Linear mass [kg/m]	0.0898	0.0321	0.0297	0.0647	0.0009
Linear mass [%]	41.3	14.8	13.7	29.7	0.4
Density [kg/m^3]	1316 ± 25	1745 ± 100	1375 ± 100	-	-

[a] Measured from a photograph

The degradation of each of the material components was studied using simultaneous thermal analysis (STA) including thermo gravimetric analysis (TGA) and differential scanning calorimetry (DSC). A small sample (~ 10 mg) was placed in a furnace and heated at constant rate. The sample mass and energy release were measured during the heating. The experiments were carried out both in air and nitrogen at heating rates between 2 – 20 K/min, using Netzsch STA 449C equipment.

Cone calorimeter experiments were performed for eight 10 cm long samples of the cable placed next to each other to construct a roughly 10 cm × 10 cm exposed area. Radiative heat flux was 50 kW/m^2 and ignition by spark igniter. The heat release rate (HRR) and mass loss rate (MLR) were recorded until all the flames disappeared. Since the experimental mass data was quite noisy, the MLR was determined by fitting a piecewise continuous polynomial to the mass results and taking the MLR as a first derivative [9]. The second-order polynomials had continuous first derivative. The second-derivative discontinuities were allowed in a few locations chosen by visual inspection of the data.

Modeling

All the simulations were made using Fire Dynamics Simulator (FDS), version 5.5.2 [10]. In the model, the reaction rate of the pyrolysis reactions is calculated using Arrhenius equation

$$r_{ij} = A_{ij}\left(\frac{\rho_{s,i}}{\rho_{s0}}\right)^{N_{s,ij}} \exp\left(-\frac{E_{ij}}{RT_s}\right), \tag{1}$$

where A (s^{-1}) is the pre-exponential factor, E (kJ/kmol) is the activation energy and N is the reaction order. Subscript i denotes the i^{th} material component and j the j^{th} reaction. $\rho_{s,i}$ is the solid density of the component, and $\rho_{s,0}$ is the original density of the layer. The solid phase heat conduction is solved in one dimension, according to the heat conduction equation

$$\rho_s c_s \frac{\partial T_s}{\partial t} = \frac{\partial}{\partial x} k_s \frac{\partial T_s}{\partial x} + \dot{q}_s''' , \tag{2}$$

where T is temperature and c and k are the specific heat and thermal conductivity, respectively. The chemical source term $\dot{q}_s'''$ contains the heats of reaction H_r, and is calculated as

$$\dot{q}_s'''(x) = -\rho_{s0} \sum_{i=1}^{N_m} \sum_{j=1}^{N_r} r_{ij}(x) H_{r,ij} \ . \tag{3}$$

A mixture fraction based combustion model was used in the cone calorimeter simulations. The cone calorimeter model had dimensions of 30 × 30 × 40 cm^3. The sample (10 × 10 cm^2) was placed in the middle of the bottom boundary and all the other walls were open. The computational mesh was extremely coarse (10 cm) but a refinement of the mesh was not found to change the results significantly at an external heat flux of 50 kW/m^2. It is clear that a computation with such a coarse mesh cannot capture the details of the flame in the real cone calorimeter experiment. However, it can provide an effective description of the flame heat flux to the sample surface because the combustion model burns most of the fuel within the one or two 10 cm cells above the sample, and because the source term of the gas phase radiation transport equation includes a specified fraction (usually 0.35) of the local heat release rate. The spark igniter was not included in the model because the ignition happens as soon as the fuel meets oxygen.

The cable was modeled as a complete cable, neglecting the small amount of additional plastics. The approximation of the cable structure as a planar surface is illustrated in Figure 1. The first and third layers consist of cable sheath material. The inner layer is a homogenous mixture of the insulation and filler materials. Conductors are not combustible and thus neglected in the model for simplicity and to save computational time in large scale simulations. According to our previous simulations, the effect of the conductor on the model performance is not significant. The properties of the 2 cm thick backing layer were ρ = 800 kg/m^3, k_s = 0.1 W/m·K and c_s = 1 kJ/kg·K.

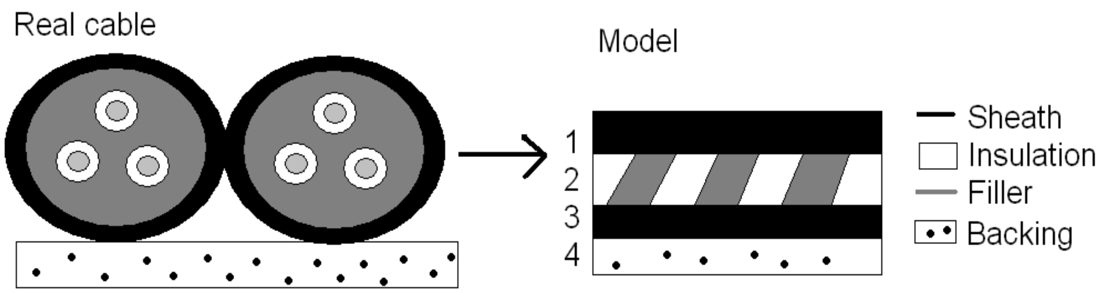


Figure 1 Principle of creating an electrical cable model in cone calorimeter

The kinetic parameters (A, E, and N_s) in Eq. 1 and the material component mass fractions were determined from TGA experiments using optimization, where a model with free parameters was fitted to the experimental results. This method, utilizing genetic algorithm (GA) as an optimization method, has been used in many of the recent works [4] to [7]. Genetic algorithms are based on the idea of survival of the fittest. Originally a random set of parameters is tested against the experiment, and the best fitting sets survive to the next iteration round. The method is effective in non-linear problems with several unknown parameters. The algorithm works well with any kind of reaction paths or parameter ranges. The drawback may be a long estimation time, and the stochastic nature of the algorithm. The number of iterations needed cannot be predicted as all the operations depend on the random numbers and probability distributions. The GA parameters in this work are the same as used in Ref. [6] except for the mutation rate that was set to 0.25.

The thermal parameters, i.e. thermal conductivity k_s, specific heat c_s and the heat of reaction H_r, and the material-specific heats of combustion (H_c) and surface emissivity (ε) were estimated from the cone calorimeter experiments. Material densities were directly measured.

RESULTS

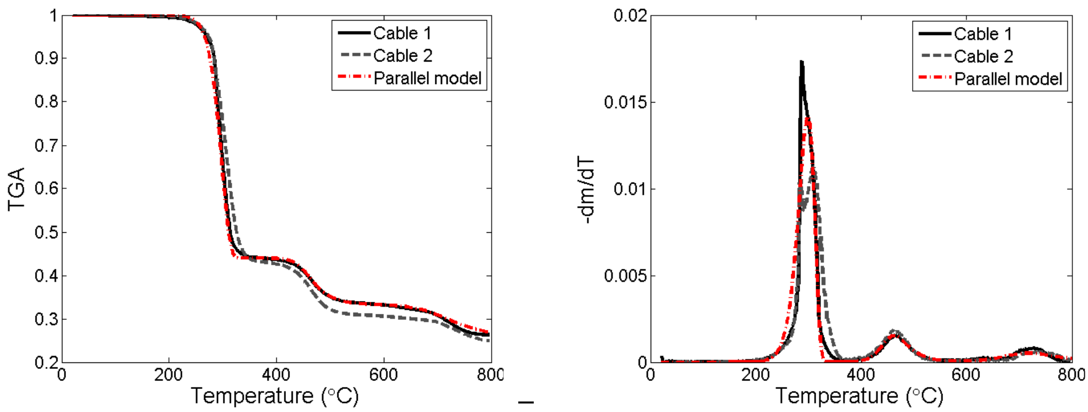
The sheath material was assumed to be a homogenous mixture of three independent components. A parallel reaction path with free reaction orders was used. The reaction path and the kinetic parameters are shown in Figure 2 and the comparison of experimental and simulated TGA results in Figure 3. This figure shows results for two different cables, of which Cable 1 is the one studied in this work. The reaction paths and kinetic parameters of the filler and insulation materials are shown in Figure 4 and Figure 5, respectively.

(56 %) Component 1
$A = 7.56\cdot 10^{13}$, $E = 1.73\cdot 10^{5}$, $N_s = 0.96$ $\longrightarrow$ $\begin{cases}\text{(65\%) HCl}\\ \text{(35 \%) Fuel}\end{cases}$

(33 %) Component 2
$A = 6.50\cdot 10^{19}$, $E = 2.95\cdot 10^{5}$, $N_s = 2.71$ $\longrightarrow$ $\begin{cases}\text{(67 \%) Char}\\ \text{(33 \%) Fuel}\end{cases}$

(11 %) Component 3
$A = 5.43\cdot 10^{11}$, $E = 2.40\cdot 10^{5}$, $N_s = 2.48$ $\longrightarrow$ $\begin{cases}\text{(31 \%) Char}\\ \text{(69 \%) Fuel}\end{cases}$

Figure 2 Parallel model reaction path and kinetic parameters for the cable sheath. *A* in s^{-1} and *E* in kJ/kmol



(a) (b)

Figure 3 TGA results (10 K/min) for cable sheaths and FDS fit: (a) TGA; (b) Gradient of TGA

(49.5 %) Component 1 $\longrightarrow$ HCl
$A = 1.77 \cdot 10^{12}$, $E = 1.51 \cdot 10^{5}$, $N_s = 1.69$

(50.5 %) Component 2 $\longrightarrow$ (75 %) Component 3 $\longrightarrow$; (25 %) Fuel
$A = 3.12 \cdot 10^{14}$, $E = 2.32 \cdot 10^{5}$, $N_s = 3.0$

$\longrightarrow$ Component 3 $\longrightarrow$ (77 %) Char ; (23 %) Fuel
$A = 1.61 \cdot 10^{6}$, $E = 1.59 \cdot 10^{5}$, $N_s = 2.62$

Figure 4 Reaction path and kinetic parameters for the insulating material. *A* in s^{-1} and *E* in kJ/kmol

Component 1 $\longrightarrow$ (75 %) Component 2 $\longrightarrow$; (25 %) Fuel
$A = 5.28 \cdot 10^{9}$, $E = 1.31 \cdot 10^{5}$, $N_s = 0.77$

$\longrightarrow$ Component 2 $\longrightarrow$ (56 %) Char ; (44 %) Fuel
$A = 9.02 \cdot 10^{7}$, $E = 1.96 \cdot 10^{5}$, $N_s = 2.46$

Figure 5 Reaction path and kinetic parameters for the filler. *A* in s^{-1} and *E* in kJ/kmol.

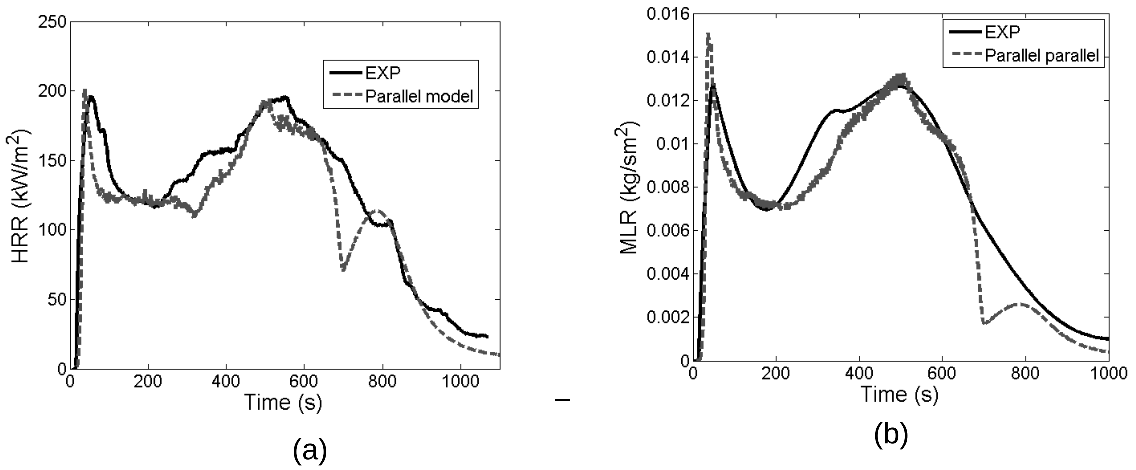


Figure 6 Comparison of experimental and simulated cone calorimeter results at 50 kW/m^2: (a) heat release rate; (b) mass loss rate

The thermal parameters were estimated from the cone calorimeter experiments. The emissivities of the initial material components were set to 1.0, but non-unity emissivities were allowed for the chars because the virgin emissivity only plays a role during the short time before the ignition. As the conductivities and the specific heats were treated as constants over temperature, the estimated values must be treated as averages over corresponding temperature ranges. The middle layer is a mixture of 48.1 % insulator and 51.9 % filler. The layer thicknesses from first to last are 2.73 mm, 3.2 mm and 2.73 mm. A comparison of simulated and measured HRR and MLR is shown in Figure 6. The model predicts very accurately both the ignition time and shapes of the two peaks in the curves. Including data for different radiation levels would probably reduce the model fitness into a single curve but improve the parameter generality.

Table 2 Thermal parameters of the cable model. "Sheath (N = 1)" is related to the effect of the kinetic parameters

	Component 1				Component 2			
	k_s [W/m·K]	c_s [kJ/kg·K]	H_r [kJ/kg]	H_c [MJ/kg]	k_s [W/m·K]	c_s [kJ/kg·K]	H_r [kJ/kg]	H_c [MJ/kg]
Sheath	0.25	2.0	800	40	0.15	2.80	700	45
Insulation	0.77	3.3	450	-	0.40	2.50	300	45[a]
Filler	0.65	2.5	800	30	0.45	0.81	300	40
	Component 3				**Residue**			
	k_s [W/m·K]	c_s [kJ/kg·K]	H_r [kJ/kg]	H_c [MJ/kg]	k_s [W/m·K]	c_s [kJ/kg·K]	ε	
Sheath	0.15	2.09	700	40[b]	0.90	2.0	1.0	
Insulation	0.79	0.80	300	40	0.67	1.3	1.0	
Filler	-	-	-	-	0.25	1.3	1.0	

[a] at upper bound of estimation range

The degradation of rigid PVC could be modeled as a two-stage process: the first releasing HCl and the second producing combustible fuel vapor and residue. However, flexible PVC contains softeners, such as phthalates, with degradation products such as benzene, naphthalene and anthracene that may be combustible and must be considered in the pyrolysis modeling. In the presence of Oxygen and heat these compounds can burn, having a heat of combustion over 40 MJ/kg [11]. In the pyrolysis modeling, the release of combustible products is considered by specifying a non-zero fuel yield for the first reaction of PVC. In the most accurate model for the cable sheaths, the fuel yield of the first reaction was 35 %. The effect of this parameter for the HRR prediction is demonstrated in Figure 7 showing the HRR results for both cables at three different fuel yields 0 %, 15 % and 35 % of the first reaction step. The smaller-than-optimal yield of fuel can, to some extent, be compensated by adjusting the heat of combustion of the corresponding material. The values should however be chosen from a reasonable range, i.e. smaller or equal to 50 MJ/kg.

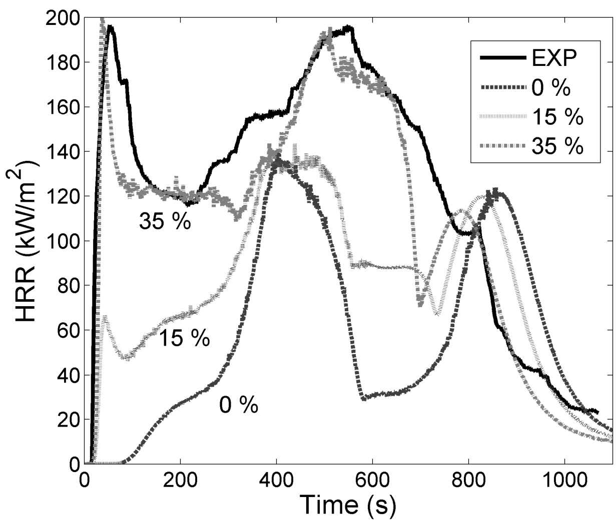


Figure 7 Effect of first reaction fuel yield at 50 K/min heat flux

CONCLUSIONS

The pyrolysis modeling of a PVC cable was studied from the viewpoint of material model parameter estimation. The kinetic and thermal parameters were estimated from the TGA and cone calorimeter experiments. With three structural components describing the structure of the complete cable (sheath, insulation and filler) and two to three material components for each structural components, the estimation algorithm was able to find a set of parameters that accurately reproduced the mass loss and heat release rate curves at one radiation level.

The plasticizers that are used in the production of flexible PVC were found to have a strong effect on the early part of the HRR curve. These additives can make up 30 to 40 wt. % of the flexible PVC, and should therefore be taken into account in the pyrolysis model as a non-zero fuel yield for the first degradation reaction. The exact allocation of the fuel between the components was found to be unimportant, as long as the correct yields of fuel and residue were retained.

The reaction path, estimation method and the parameter sets can be chosen in many ways when estimating the pyrolysis model parameters from the cone calorimeter results. Thermal parameters can, to some extent, compensate the choices made for the kinetic model. Nothing implies that one way to make the choice would be better than another. This is a topic that requires more research and discussion. It is very important to check the model behavior when changing the model details or the simulation code version. Also, the universality of the thermal parameters could be improved by considering a wider set of experimental data during the parameter estimation process, such as sample temperatures, different heat flux levels and different atmospheres.

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TOWARDS A EUROPEAN COMMON APPROACH FOR SPECIFIC FIRE PROTECTION CONCERNS - Containing Fire Scenario Variety by Specific Defense-in-depth Standards for Nuclear Applications of Fire Protection Products

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ABSTRACT

Resulting from the initiative launched at the SMiRT 20 post-conference seminar on "Fire safety in Nuclear Power Plants and Installations" in August 2009, a group of experts composed from the authors started a reflection aiming at enhancing the assessment of the capacity of the components for mitigating or sustaining fire events in nuclear installations without damages. For the assessment of the fire course deterministic as well as probabilistic methods are being applied. For deterministic analyses modeling tools are being applied for simulating development and propagation of a fire under a variety of scenario assumptions. Probabilistic tools, on the other hand, use results of deterministic computations and reliability data fire detection and suppression means.

The authors considered difficulties associated with the two items in the sense that these difficulties may make it impossible to predict with reasonable confidence the fire course and its effects. Three items were more specifically considered: (1) to which extent specific protection measures can be accepted as being capable to reduce the variety of scenario hypotheses, (2) what should be improved to develop earthquake resistance qualification of fire fighting components, and (3) which strategies should be adopted for temporary load management at nuclear power plants.

Progress in all three items should contribute to enhance defense in depth, reduce the amount of different scenarios, hence delivering fire event calculations of higher reliability. The authors believe that demonstrating defense in depth against fire for selected areas could be a good complement, or in some case, an alternative to computational approaches. As an example a lube oil fire at the main cooling pump is outlined.

The authors also consider the need to have a common approach to the qualification of fire protection systems and components. Doing this, the authors expect standards for these usual industrial and public products to be developed for nuclear applications. This would create a sufficient market size to encourage the manufacturers to follow the specific needs of the nuclear domain. The benefit would be appreciable for all the parties concerned with the safety of nuclear installations. A first attempt is given for the fire dampers.

CONSIDERING PROTECTION FEATURERS IN THE FIRE SAFETY DEMONSTRATION

Introduction

Fire is a plant internal hazard with the potential to create common cause failures of structures, systems and components (SSC) important to safety [1] thus the fire protection program of a nuclear power plant shall ensure that the general safety design requirements are adequately met.

Fire protection shall be designed to ensure that in the event of a fire the reactor can be safely shut down and radioactive releases to the environment are minimized. This is achieved by ensuring the following safety performances provided in [2]:

- A method to shut down the reactor safely and to maintain it in safe shutdown condition in operational states as well as during and after accident conditions,
- To remove the residual heat from the core after reactor shutdown, including accident situations, and
- To reduce the potential for a release of radioactive materials and to ensure that any releases are below prescribed limits in operational states and below acceptable limits during accident conditions.

The general approach applied to achieve the nuclear safety objectives with respect to fires, is provided by a sufficient segregation of redundant parts of the safety systems ensuring that a fire affecting one division of a safety system will not prevent performing the safety function by another division.

There are two design approaches used to ensure the operability of redundant safety system equipment, which is based on room arrangements and fire compartments. The fire containment approach isolates, as far as possible, each of the divisional safety systems performing safety functions within individual fire compartments.

In situations where individual fire compartments cannot be used to isolate redundant items important to safety, the fire influence approach provides protection by locating these items in separate fire cells within a fire compartment. Fire cells are separate areas that may not be completely enclosed by fire barriers, but the propagation of flames and other products of combustion between fire cells is strongly limited. The adequacy of this has to be demonstrated.

The reactor building containment is generally such area where special incidences, such as the loss of primary coolant and the resulting pressure rise and release of hydrogen, requires protection measures that prohibits the use of closed fire compartments as ordinary applied for the separation of the redundant safeguard systems. Therefore fire separation inside those areas is mainly provided by the fire influence approach by creating fire cells.

In order to demonstrate the achievement of the safety targets, the potential fire scenarios in such areas are of high interest, whereas the fire scenario and, in particular, the potential fire size and duration are the governing parameters that influence SSC important to safety and therefore constitute parts of such fire safety demonstration. Because the fire risk and the corresponding fire damage shall be as low as reasonably possible, a set of preventive measures shall be provided. These preventive measures can generally be distinguished into three groups of protection features:

- Primary protection features,
- Secondary protection features, and
- Tertiary protection features.

This design strategy is commonly denoted as defense in depth design strategy compensating or correcting failures that may occur without causing harm to individuals or the public at large [3]. A defense is depth strategy does not focus unexceptionally on one unique safety activity; it describes a set of safety features meeting the safety objective if one or more safety features fail.

In the example "fire risk at the reactor coolant pumps (RCP)", primary protection features may be provided by preventing fires by e.g. the substitution of the flammable lube oil by non-flammable or flame retardant lubrication liquids (primary protection). If this type of protection features is limited or even not possible, secondary protection features may come into effect, e.g. by prevention and limitation of leakages that may take part in the fire situation (secondary protection). The tertiary protection features limit the influence on SSC important to safety by e.g. the use of extinguishing systems or fire insulation etc.

In the following, the variety of fire scenarios is outlined by characterizing the performance and potential combination of protection features. For the fire safety demonstration, in general there are three main approaches:

- Fire calculations/simulations without taking into account existing protection systems;
  In this case, there is a large uncertainty on the fire scenario / input data directly affecting the results. Furthermore, the results achieved must be compared to the failure criteria of the SSC important to safety, which may also be not known properly.
- Fire calculations/simulations with consideration of existing protection systems:
  In this case, there is a reduced uncertainty on the fire scenario / input data by diminishing or reducing the variables to be considered. However, even here a comparison to the failure criteria of the SSC important to safety is necessary.
- The existing protection systems are capable to prevent fires or detect failures before fires can occur.

This paper addresses an alternative approach besides fire simulations, which can be selected for the fire safety demonstration at the reactor coolant pump (RCP). It describes a general argumentation concept and potential protection features.

Example

The RCP constitutes a potential fire risk due to the huge lube oil content of the bearings. The reactor containment building is typically an area where the use of closed fire compartments usually applied for the separation of the redundant safeguard systems is limited. As a result the fire separation is mainly provided by the fire influence approach by creating fire cells. Therefore, the safe shut down capability and the capability of the fire separation must be evaluated for these areas. The RCP design meets a high degree of quality and comprises diagnostic and control devices that even prevent and/or indicate upcoming potential oil leakages and thus potential fires. These protection features in combination with the physically/chemically behavior of the lube oil can be used to assess the fire scenarios and to demonstrate the safe shut down capabilities.

The following section gives some examples of safety features and provides the general arguments on fire prevention associated to them:

- Physical and chemical properties of the lube oil,
- Integrated oil systems,
- Lube oil level monitoring,
- Vibration monitoring detectors,

- Lube oil collecting systems,
- Camera monitoring,
- Floor drainage systems,
- Extinguishing systems,
- Motor bearing temperature monitoring,
- etc.

Physical and Chemical Properties of the Lube Oil

There are three main phenomenon that influence fire ignition and fire size on flammable liquids. One is the size of the open liquid surface which has an effect on the heat release rate (fire size); the other phenomena are the oxygen supply and the fact that combustible liquids have to be preheated above their liquid specific fire points where continuous burning is to be observed.

The latter phenomenon provides already a high degree on fire prevention precaution and is part of the primary protection concept. For the safety analysis, the flash point of the particular lube oil shall be assessed with respect to the maximum temperatures in the lube oil systems and in the vicinity of potential leakages.

However, scientific literature describes a phenomenon where high flash point liquids can be preheated above their fire points although the surrounding temperature is lower [4]. This heating is only possible within special boundary conditions using an external heat source and oil film thickness within certain limits, whereas the film thickness is of essential importance (maximum/minimum limit).

Oil collecting pans that may be provided at the reactor coolant pumps can already provide protection against fires by limiting the flammable surface and the delay and/or prevention of locally heating above the flash point where the surrounding temperature is low.

Fires may also occur on porous materials such as thermal insulations. The protection of those insulation materials from sucking flammable oil shall be demonstrated in the fire analysis.

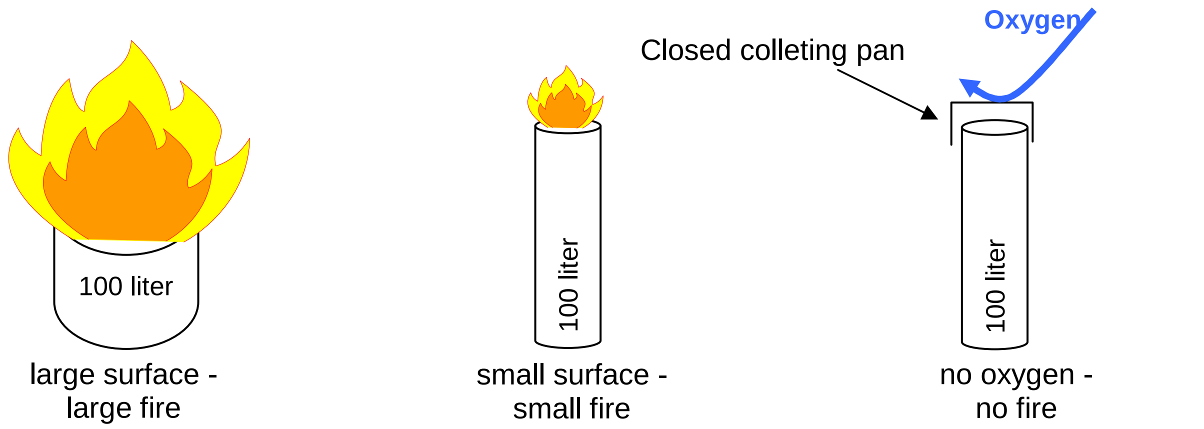
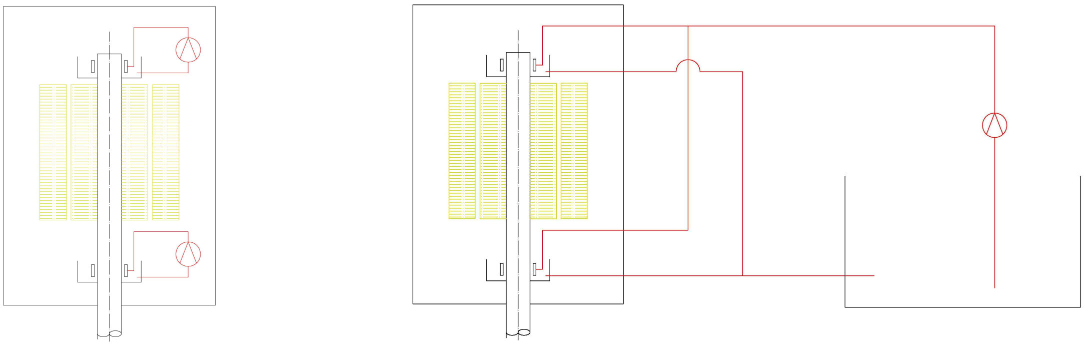


Figure 1 Surface area versus fire size

Integrated Oil System

If the RCP is designed with an integrated oil system the total amount of oil that may participate in a fire situation is significantly reduced. On the contrary, the area for piping and oil storage at external oil supply systems constitutes an area which is potentially jeopardized by oil fires.



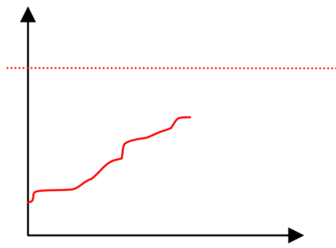
Integrated oil system:
approx. 1000 liter oil

External oil system:
approx. 4000 liter oil

Figure 2 Reactor coolant pump (RCP) oil supply system

Lube Oil Temperature Monitoring

If the RCP lube oil temperature is surveyed by a lube oil temperature monitoring system, the rise of oil temperature may be an indicator of leaking oil by the reduced oil content. Moreover, the oil temperature, in particular the temperature difference to the oil specific flash point is an indicator of the fire safety margin. The following parameters may affect the assessment:

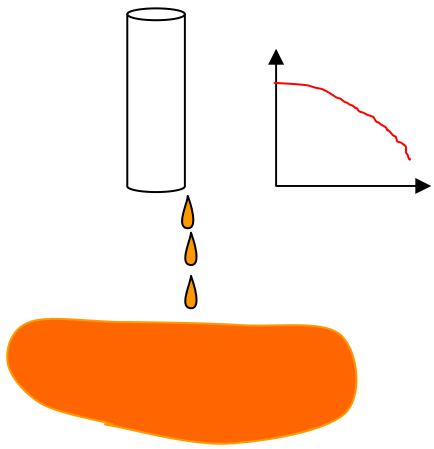


- Detection sensitivity:
  - Steadily,
  - High temperature alarm,
- Indication of alarm,
- Procedures after an alarm has been triggered:
  - Automatically,
  - Manually,
  - Flash point of the lube oil.

Lube Oil Level Monitoring

If the RCP lube oil level is surveyed by a lube oil level monitoring system, lube oil leakages maybe monitored. For the safety assessment the capability/sensitivity of the level measurement systems shall be characterized in order to assess properly detection and to estimate the maximum oil volume until the leakage is detected and precautions can be taken. The following parameters may affect the assessment:

- Detection frequency:
  - Steadily,



    - Frequently,
- Detection sensitivity:
    - Steadily,
    - Low level alarm,
- Oil content until triggering an alarm,
- Indication of alarm,
- Procedures after an alarm has been triggered:
    - Automatically,
    - Manually.

Vibration Monitoring

Vibrations can be considered as pre-stage of upcoming damages where also the lube oil casing may be involved (pipe rupture; leaking flange connections, etc.). If the RCP vibrations are monitored where special actions took place after exceeding a defined threshold value, this monitoring can be used in the safety assessment for damage prevention of the lube oil casing.

Camera Monitoring

Cameras monitoring the reactor coolant pump motors can be used to monitor the general condition at the RCP area, even leakages and fires.

Lube Oil Collecting System

The integrity of the lube oil systems should be primary assessed based on the component quality which may already provide a high level of protection. Flange or similar connections may be a source of leakage. Lube oil collection systems that collect potential leakages provide certain protection against oil fires with respect to the heating (flash point) and limit the surface that may catch fire (influence on the fire size). The following parameters may affect the assessment:

- Potential leakage sources:
    - Flange connections,
    - etc.
- Potential leakage sources covered by a leak oil collection system,
- Is the total oil inventory that can spread out covered?
- Consideration of oil level equalization:
    - Consideration of pressurized systems and/or lifting systems and the oil content that can be sucked by oil pumping systems,
    - Drainage systems from the collecting system.

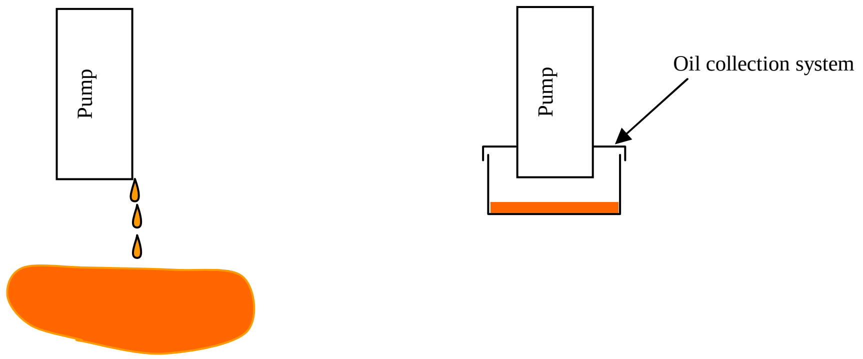


Figure 3 Function of oil collection system

Floor Drainage System

A floor drainage system may limit the total amount of oil participating in a fire situation and therefore may limit the fire duration and almost the fire size (limitation of combustible surface). Furthermore, the limitation of the oil layer thickness may prevent fire spreading through the entire oil surface (cp. physical and chemical properties of the lube oil). The following should be provided:

– Total capacity,
– Location of drains,
– Slopes,
– Floor asperity.

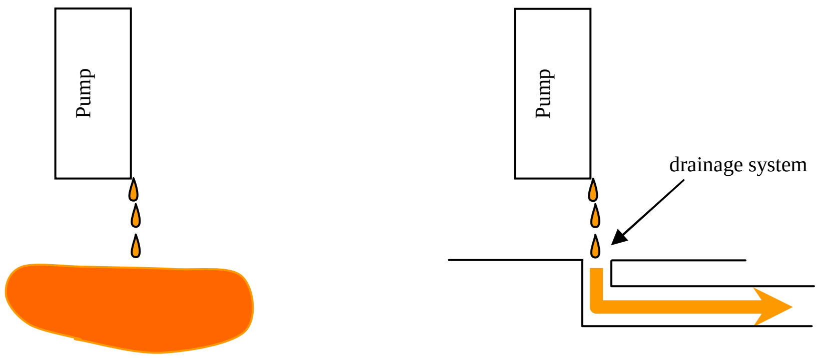


Figure 4 Function of floor drainage system

Extinguishing Systems

Extinguishing systems provide a high degree of fire protection and may be used in order to characterize the general fire protection features. However, if fire extinguishing systems are used for the safety demonstration, these extinguishing systems become safety related.

Conclusions

Reactor coolant pumps are typically equipped with multiple protective installations providing protection against extensive damages. These protective installations may also prevent leakage of flammable lube oil thus contributing to fire prevention. Oil collecting systems, drainage systems, etc. can provide further protection against fires from flammable liquids. The protection systems and safety features outlined above shall be considered as examples for fire prevention precautions and the assessment of real fire situation. For the fire safety demonstration, all protection systems and safety features shall be used and taken into account in the simulations as they are limiting the fire severity directly or indirectly.

APPROACH TO A COMMMON QUALIFICATION SPECIFIC ADDITIONAL REQUIREMENTS FOR NUCLEAR POWER PLANT FIRE PROTECTION EQUIPMENT - EXAMPLE OF SAFETY RELATED FIRE DAMPERS

Introduction

In the framework of nuclear power plants' fire safety, a variety of fire protection equipment is provided. This equipment is basically the same as those installed in industrial buildings and facilities as well as in public buildings. The equipment is compliant with standards that do not cover necessarily the requirements of NPP's fire safety. As a result, this equipment is generally not directly available on the market and, at least, a specific qualification process is needed. Up to now, this happens on a case by case basis according to the licensee needs. For economic reasons, the manufacturers are generally not interested in developing specific equipment for nuclear power plants in such a situation as the equipment is compliant with requirements of a single licensee what limits the market size.

This creates difficulties to the licensee to buy equipment compliant to its requirements at a reasonable cost. A solution could be the development of common (European) requirements/qualifications for fire protection equipment. This would create a sufficient market size to encourage the manufacturers to follow the specific needs of the nuclear domain. The benefit would be appreciable for all the parties concerned with the safety of nuclear installations.

In order to evaluate the feasibility of a common approach at least on the qualifications of equipment, the group started with the case of the fire dampers. In the following, an attempt is made to identify the different conditions that are needed for fire dampers qualification. A matrix is proposed to clarify the different events that can be combined and their timely order of occurrence.

Matrix

The matrix is valid only for fire dampers (FD) that are normally open and will have to close in case of fire. The matrix indicates if the FD have to perform their function – i.e. to close and then satisfying their intended fire compartmentation function (EI xx ho or ve i<->o according to EN 1366-2 [5]; EN 13501-3 [6]; EN-15650 [7] if the two events occur sequentially.

The following situation (see also [1]) is e.g. considered: a fire occurs and after that an earthquake occurs. Considering the fire duration of a few hours, it is extremely improbable that an earthquake would occur during this period. Therefore, the fire dampers do not need to be qualified to maintain their fire rating when they are mechanically solicited by an earthquake. This implies of course that the fire dampers have to be replaced after the fire event.

The following situation can also be accounted for: an earthquake occurs and after a certain time a fire occurs. The fire could occur as a result of the seismic or independently during the time duration necessary to reach and maintain safe shutdown states. Such situation implies that the fire dampers have to be qualified to remain operable after an earthquake and be able to perform their required fire compartmentation function.

If the fire dampers have to close in the case of presence of airflow rate while there a fire event has not yet occurred, the situation is similar to the spurious or faulty actuation of the fire damper before the fire event. Such situation could be considered as very improbable as there is no relationship between the two events in this time sequence.

In contrary, the closure of the fire damper in case of fire can be needed in presence of an airflow rate (this is related to the design of the plant). Generally, the fire dampers would also have to withstand the pressure drop or increase if they are located close to the fans while they still have to fulfill their intended function.

Concerning the environmental conditions (mainly radiation) that could occur before or independently of the fire (for instance due to a LOCA), this case is not considered here.

Table 1 Event matrix

		Second Event				
		Fire	***Earthquake (S)***	***Radiation (G)***	***Flow Rate (A)***	***Pressure Resistance (P)***
First Event	***Fire***	not applicable	no	no	yes	yes
	Earthquake (S)	yes	not applicable	no	yes	yes
	Radiation (G)	no	No		no	no
	Flow Rate (A)	no	No	no	not applicable	yes
	Pressure Resistance (P)	no	No	no	no	not applicable

Fire dampers important for nuclear safety according to the Fire Hazard Analysis (FHA) and probabilistic fire risk analysis (Fire PSA) have to meet some quality requirements depending if they belong to a ventilation system that is or is not safety related. The quality requirements are at least the same of the ventilation system considered but could not be less than material quality requirements, manufacturing process controls, qualification of the technicians involved in the process (welding for instance), etc.

Perspectives

Those minimum requirements still have to be discussed and would belong to the nuclear qualification topic. Using the same logic as the one of EN standards, the authors would expect to achieve a classification such as for instance EI 120 ho i<->o N-S-AP indicating that such a fire damper fulfilling the quality requirements of nuclear (N) remains fully functional after an earthquake (S) can close in presence of airflow rate (A) and withstand to the pressure caused by a fan operation (P).

Next Steps

The qualification tests have to be sufficiently detailed (objectives, criteria, tests features…). E.g., the closure of the fire damper in presence of non-uniform air flow rates could be achieved with the following schematic test arrangement:

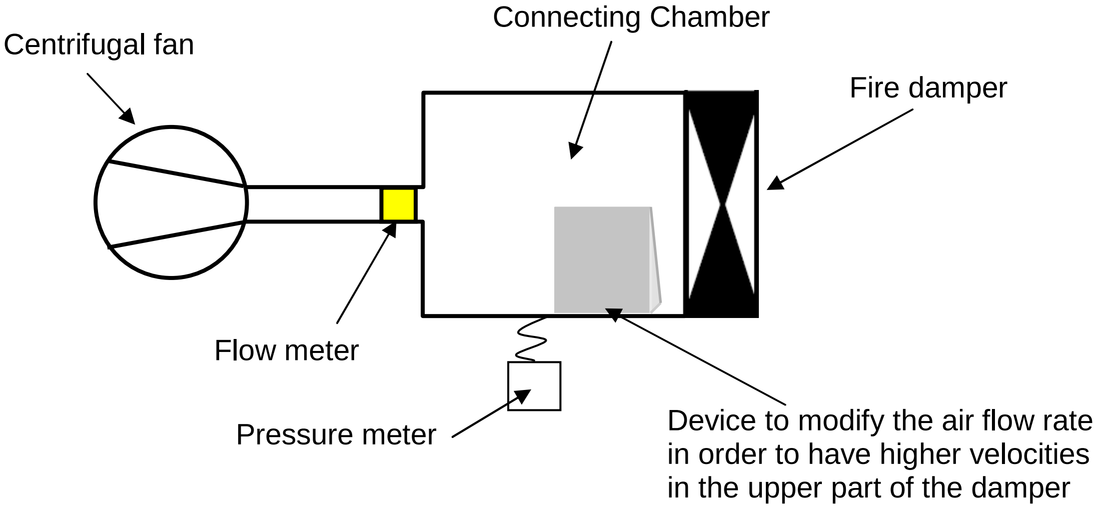


Figure 5 Scheme of a fire damper test arrangement

As soon as the different qualification tests shall be defined, the manufacturers can be consulted to get their opinion and, if needed, the qualification tests shall be adapted.

Conclusion

The attempt to adopt a common approach of qualification of fire dampers for nuclear power plants is encouraging. A matrix of qualification has been proposed. The qualification tests still have to be defined and described before consulting the manufacturers.

The group members are expecting that additional experts will join the group to share the objectives and to develop these common approaches.

STRATEGIES FOR TEMPORARY LOAD MANAGEMENT AT NUCLEAR POWER PLANTS

Background

Dimensional fire scenarios are normally assigned in connection with fire analysis. At loss of another fire load a smaller amount of combustible material is always assumed to be present, e.g. a trash bag.

Purpose

The purpose of this contribution is to develop a dimensional progress of fire for a "trash bag fire" scenario, in order to get a uniform input in connection with future fire analysis. The governing questions that will rise are: "What fire load is acceptable with reference to this trash bag fire?" allowing to have a certain level of fire risk that everybody can understand and we can communicate, and: "What risk is acceptable from transient fire load?"

This is defined in a manner that a "trash bag fire" could be placed at the worst place without affecting redundant systems. If this fire load in the fire risk analysis is acceptable, 25 % of this fire load is the level that the plants do accept.

Method

The maximum effect of fire is developed from reported attempts in reference literature. A simplified calculation is made to develop durability, and thereby obtain a Heat Release Rate (HRR) curve.

Background Together with Existing Attempts

A number of attempts have been documented in regards to the study of the course of a fire in a trash bag. How large the fire becomes as well as the duration of the fire depends mainly on what type of trash the bag contains, the weight of the trash bag as well as how closely packed the contents are, in other words what density the trash bag has.

Some trash bags contain a mix of paper and plastics while others mainly contain plastics or mainly contain paper. The material contents in the trash bag are significant for the development of the course of the fire since different materials have different characteristics during a fire. Plastics have higher energy contents per kg than paper, which results in a course of the fire for plastics lasting a longer time period alternatively with a higher intensity than a course of fire in paper.

Another factor influencing the duration of the course of the fire is how closely the garbage is packed. If the material in the bag is loosely packed, the fire grows more rapidly and a higher maximum effect might eventually be reached. On the other hand, if the material is densely packed, the growth is slower while the course of the fire can last for a longer time period. The speed of the growth is of course influenced by each material's characteristics at fire as well.

The shape of the container does also have an influence on the course of the fire. A non-combustible barrel results in the base of the fire source being the same as the size of the opening to the barrel, that way the maximum heat release rate is limited. A plastic barrel on the other hand will in the beginning direct the heat release rate, but as soon as the barrel itself catches fire, the barrel will contribute to the combustible material and, depending on how the barrel collapses, different maximum effects can be reached even if the original fire sources are the same. Simple trash bags fastened on a steel frame, which ignites relatively fast, also exist.

In exhibit [8] two trash cans with almost identical design have been observed. Each container is made out of plastic and weighs 3.6 kg. The trash in each container consists of paper, sawdust, and cups etc., it weighs 10 kg. In one case, the maximum effect of 300 kW is achieved, while in the other case the maximum effect of 150 kW is achieved. The reason why the two containers reach different maximum effects depends on how each container collapses.

Further on in this instruction only the scenario where the highest effect is reached will be shown. During the attempt the fire grows linear to 300 kW, which is achieved after approximately 880 s. The fire then decreases relatively fast and releases very low effect (between approx. 5 and 30 kW) the last 820 to 900 s.

In [9] there are four different types of courses of fires in trash bags shown. One course of fire that is shown is a fire in a trash bag with a weight of 4.1 kg containing straw and grass etc. The fire reaches maximum effect of 350 kW after approximately 2 min. Maximum effect is maintained during approx. 1 min, and then the fire is decreasing within a period of approx.6 min. Another course of fire that is shown is a fire in a trash bag containing paper with a total weight of 3.51 kg. Maximum effect then amounts to approx. 350 kW, this is reached after approx. 70 s. Maximum effect is maintained during approx. 1 min, then the effect decreases within approx. 7 min. A bag weighing 2.34 kg, also containing paper, reaches a maximum effect of approx. 290 kW after approx. 120 s.

The effect then drops immediately and decreases gradually during approximately 4 min. The curve for heat release rate is also shown for a trash bag containing paper with the mass 1.17 kg that reaches maximum effect 140 kW after approximately 70 s. Even here the effect drops immediately and the decrease continues during approx. 4 min.

Calculation

The appearance of the course of fires in [9] is simplified in this calculation by having the fire in the growth phase follow an α-t^2 curve.

$$Q_{(t)} = \alpha \cdot t_a^{\;2} \tag{1}$$

The cooling is assigned to follow a similar connection where cooling constant β, also this in the unit kW/s^2, is calculated according to below;

$$Q_{(t)} = Q_{\max} - \beta \cdot t_\beta^{\;2} \tag{2}$$

where there is the amount of time [s] that the cooling has been in progress. Table 2 shows the calculation procedure that has been used.

Table 2 Calculation data for fire in a trash bag containing paper

Weight [kg]	Q_{max} [kW]	Time to Q_{max} [s]	Growth Rate α [kW/s^2]	Time with Q_{max} [s]	Time for Cooling [s]	Cooling Constant β [kW/s^2]
4,1	350	120	0.0243	60	360	0.002
3,51	350	70	0.07	60	420	0.002
2,34	290	120	0.02	0	240	0.005
1,17	140	70	0.025	0	240	0.002

An average of the speed of growth, α, is 0.0348 kW/s^2. An average of the speed of cooling, β is 0.00275 kW/s^2.

The level of effect that a fire produces as a maximum depends on the type of combustible material as well as the amount of material the trash bag contains.

The attempts that are shown in [8] are separated from the connection shown in [9]. The fire from [8] is increasing after a more linear connection until maximum effect is reached. After the maximum effect is reached the heat release rate decreases relatively fast which is yet another way to set it apart from the attempts in [9] where the effect decreases gradually during several minutes.

The course of the fire in [8] is therefore illustrated with a linear formula according to below;

$$Q_{(t)} = \omega \cdot t \tag{3}$$

where ω is calculated through 300 kW / 800 s = 0.375 kW/s

In [9] a connection is also shown between the density in the combustible material and the diameter of the combustion surface. At a lower density a higher effect is achieved. A container with a diameter of 700 mm releases 400 kW if the density reaches 30 kg/m^3. Should the density instead reach 100 kg/m^2, a heat release rate of only140 kW is achieved?

Calculation of the Text book Case

With reference to above shown attempts a fire in a trash bag can be set-up according to the following:

The starting point is for the trash bag to grow according to an α-t^2 curve with a growth rate of 0.0348 kW/s^2, which is almost identical to a so-called fast growth (0.047 kW/s^2). The maximum effect being reached is approx. 400 kW. The time period during which the fire continues with a maximum heat release rate depends on the amount of trash the container includes initially which thereby varies from case to case. Calculation of this time is shown below. Furthermore, the cooling is set up to proceed with a speed of 0.00275 kW/s^2.

Below a Heat Release Rate (HRR) curve is outlined showing the time/HRR of a trash bag fire.

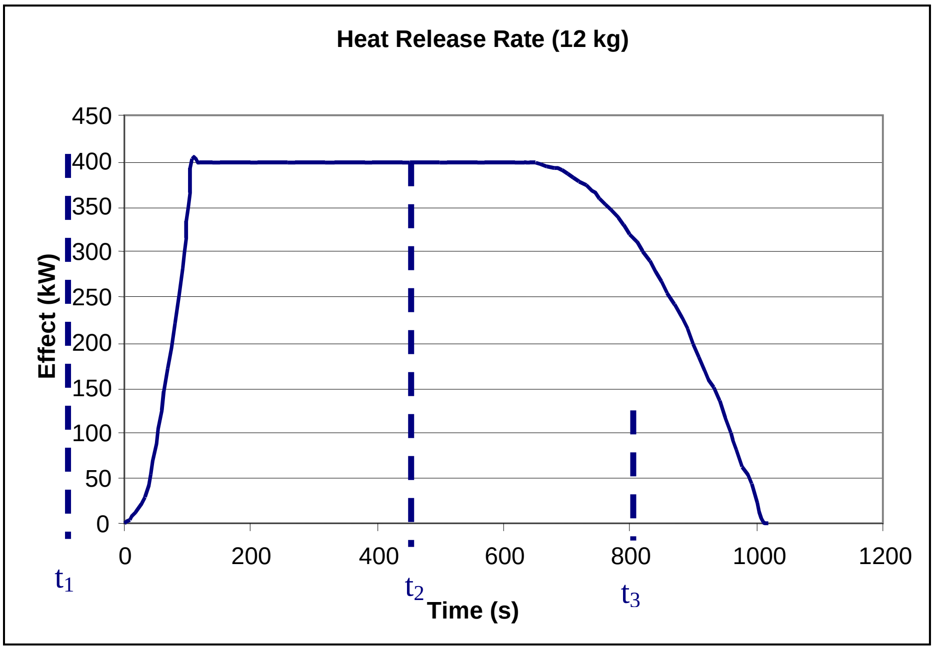


Figure 6 Heat release rate curve for fire in a trash bag containing 50 % paper and 50 % plastics at a total weight of 12 kg

Calculation of the Duration of the Course of the Fire

At a mix of 50 % paper (ΔHc 16.2 MJ/kg) and 50 % plastic (ΔHc ca. 30 MJ/kg) the average is ΔHc 23 MJ/kg. It should be noted that ΔHc for different types of plastic and wood varies.

For this reason an average for the materials is assigned. The total amount of energy being released at a complete combustion is therefore 23 MJ/kg * X kg, where "X" is the total weight of the trash. The energy that is released during the growth phase is calculated through integration of the α-t^2 curve (Eq .1), which gives:

$$Q = \left.\frac{\alpha \cdot t^3}{3}\right|_0^{t_1} \tag{4}$$

The time required to reach 400 kW is 107 s resulting in a Q of 14.3 MJ. The energy that is released during cooling is calculated in the same way. The time from 400 kW to 0 kW is 381 s.

$$\dot{Q}_{cooling} = \beta \cdot t^2 \tag{5}$$

integrated over time, Eq 5 becomes

$$Q_{cooling} = \left.\frac{\beta \cdot t^3}{3}\right|_{t_2}^{t_3} \tag{6}$$

The total energy that is released during cooling equals 50.85 MJ. The energy that remains for, and can be released during, full heat release rate equals 23*X MJ – 14.3 MJ – 50.85 MJ. The time period, during which the fire can release maximum effect, 400 kW is calculated through

$$t_{full\ heatreleaserate} = \frac{Q_{rest}(MJ)}{Q_{\max}(MJ/s)} \tag{7}$$

Assume that the entire trash bag including container weighs 12 kg. This results in full effect to be released under

$$\frac{(23*12-60{,}15MJ)}{0{,}4MJ/s} = 527s \tag{7}$$

The course of the fire will then result in the following heat release curve:

$$Q_{Total} = \int_0^{t1} \alpha_{growth} t^2 + Q_{MAX} * (t_2 - t_1) + \int_{t2}^{t3} \alpha_{cooling} t^2 \tag{8}$$

(t_0 = start, t_1 end of growth, t_2 start cooling,t_3 end of cooling)

Discussion

When choosing the maximum heat release rate a relatively high effect has been assigned in comparison with the shown attempts. The reason for this is due to the fact that a big uncertainty often exists in regards to which materials that are included in the fire as well as how closely packed the material is. The given parameters are obviously in need of adjustment if the fire in question for example is small or if the combustion surface is extremely big. The general ΔHc value should also be adjusted if it is known what type of trash the container holds.

It has in the attempt that is shown in [8], concerning the growth, not been taken into consideration in regards to producing a typical course of fire. The growth in these attempts has been excluded since it is considered to be a more conservative point of view to base the attempts on (which happens in the majority of the documented attempts) where the fire growth is more rapid.

A conservative approach has also been used since all the material was assigned to burn out completely. The combustion efficiency can alternately be used which during normal circumstances, for materials which releases soot, means combustion only corresponds to 60 to 70 % of a theoretical total combustion [10]. The combustion efficiency however varies for different type of materials. Oil which gives off relatively much soot has a somewhat low combustion efficiency while alcohols such as methanol has a high combustion efficiency, close to 100 %.

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RECENT REGULATORY ACTIVITIES ON FIRE SAFETY IN FINLAND

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ABSTRACT

Finnish radiation and nuclear safety authority (STUK) is upgrading YVL- guidelines on fire protection. New principles and requirements of the guideline are discussed. For example, defense-in-depth principle in fire protection has to be described. One possibility is to use event trees considering fire compartments containing big fire loads such as fuel oil storages, lubrication oil storages, cable spreading rooms and big transformers. In addition some first conclusions derived from the Fukushima nuclear power plant (NPP) accident are presented, e.g. requirements for fire water and extinguishing systems in new and operating Finnish NPPs are under consideration. These conclusions are based on the safety evaluation of Finnish NPPs performed before the stress tests were started in June 2011.

Fire properties of some typical flame retardant non-corrosive (FRNC) cable types to be installed in the nuclear power plant Olkiluoto, Unit 3 have been studied by VTT during the recent years in order to verify preliminary safety assessment report (PSAR). The study was finished April 2011. Some results and conclusions of the study will be presented. The study gave input to assess adequacy of fire protection arrangements of Olkiluoto 3 cable rooms.

A new period of Finnish nuclear safety research program started 2011. Recent results of fire safety research program and content of the ongoing program are presented.

INTRODUCTION

STUK issues detailed regulations that apply to the safe use of nuclear energy and to physical protection, emergency preparedness and safeguards. STUK is upgrading the whole detailed nuclear safety regulation, known as YVL- guides. Fire protection guide is under update. Recent YVL 4.3 guide [1] will be replaced by a new guide YVL B.8. The most significant new issues are guidelines for different licensing phases and for defense-in-depth design. Deterministic fire hazard analysis and probabilistic risk assessment (PRA) studies are guided in order to support defense-in-depth design. Further on STUK is developing another guide, specific for PRA, which is referred with fire protection specific explanations. The fire protection guide state also required specifications and documents for different license phases and explain licensees and STUK's responsibilities in the licensing process.

The Fukushima accident launched large global assessment of nuclear safety. Nuclear safety is under verification for conditions, where external natural threats are suggested to overlap existing design basis requirements. Fire protection is also challenged since many scenarios describing external natural events include also consequential fires. Internal fire PSA is referred to in Finland so that external events considered are causing also deficiencies in structural and operational fire protection. Corresponding straightforward additions to requirements have been considered.

STUK ordered first research from VTT concerning typical FRNC cables in 2004. Specific research of cables to be installed in OOL3 (Olkiluoto, Unit 3) was performed during 2008 to 2010. In this paper, some generic results from cable tests are presented.

FIRE PROTECTION GUIDE UNDER UPDATE

Guidance for the Licensing Process

Even though it is not YVL guide's task to cover design standards, it is important to consider new type of technical solutions as well as available design and simulation tools when corresponding requirements as updated. These possibilities are taken into account within the licensing process so that fire protection guides support better than original design and implementation of fire protection and corresponding documentation during licensing process. Figure 1 illustrates licensing steps and corresponding detail level of information.

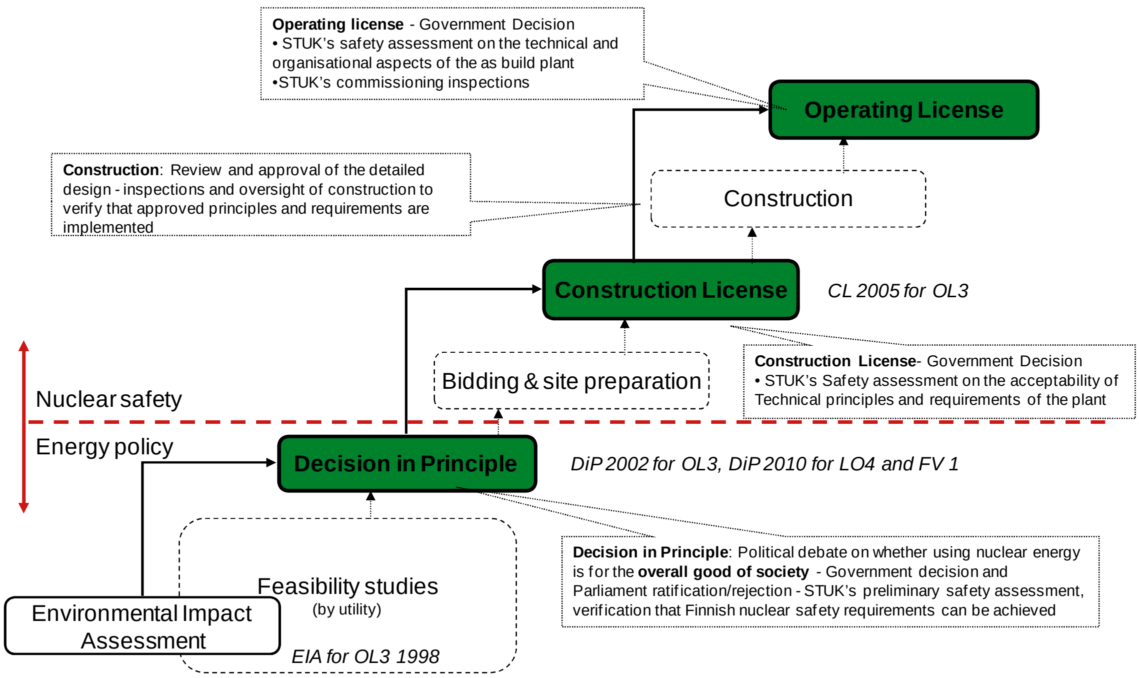

Figure 1 Licensing steps of NPP in Finland

The upgrade of fire protection guide includes more detailed descriptions of STUK's responsibilities during the licensing process. STUK's oversight including inspections during licensing steps is explained in order to understand better actual requirements and documental needs.

STUK's first responsibility in decision in principle phase is to verify that Finnish safety requirements can be achieved. Further on construction license can be approved partly based on preliminary description of fire protection systems so that fire protection principles and constructability can be ensured. Final descriptions of system and design documentation will be inspected during construction period. Final safety assessment report and quality ensuring records will be inspected during commissioning before the operating license.

Defense-in-depth Principle in Fire Protection

STUK is defining requirements for using defense-in-depth (DiD) principle for fire protection. According to DiD, fires are prevented in all stages from ignition to spreading across fire compartments. Consequences are minimized in order to ensure necessary safety functions during and after fires. Final goal is to ensure fire compartments. Technical design requirements for fire protection are not changed, but more guidelines are stated in order to ensure systematic design and safety goals.

DiD principle is a practical way to focus efforts according the safety relevance of nuclear facilities. It is useful to put efforts preventing different phases in fire scenario according to deterministic or/and probabilistic design instead of focusing too much in single details, which could endanger the total control. Design criteria are not valid, if they are concentrating too much in one part in fire scenario. Therefore there is a clear guideline which states that preventing ignitions, detecting fires in early phase, extinguishing and using other means to prevent development and spreading of fire are forming approvable concept for fire protection.

Design solutions basic features are described such as using building materials which fulfill highest property requirements of the Finnish Building Code. Securing and monitoring of fire hazardous machines and instruments such as rotational equipment (turbine generator, diesel generators and large pumps) are as well required. Monitoring concern e.g. vibrations, oil leaks and hydrogen analyzer for transformers. Flash barriers will be required for securing electric distribution boards. Control of temporary burning load and fire hazards must base on clear responsibilities, instructions and training.

Event tree analysis [2], [3] is presented as one possible method to prove out the adequate DiD of fire protection. Countermeasures in different phases of development of fires are analyzed. Important result of these analyses is measured isolation of fires and flooding. Fire hazard analyses and functional fire analyses combined with corresponding calculation model and material parameter testing are the basis for creditability of event tree studies. Important part in event tree studies is to assess sensitivity of countermeasures in the scenarios by assuming corresponding deficiencies. Typical targets are the containment, the annulus area in PWR (pressurized water reactors), control room and cases, where isolation of fire is required inside the fire compartment. Special cases are large fire loads and deficiencies in fire compartments, such as open doors and malfunctioning fire dampers. Important questions in plant safety are cases, where the loss of fire compartment could lead to damaged safety systems in more than one safety division. Such places are e.g. in control rooms and the annulus area.

DiD principle methods are developed also in Finnish nuclear safety research program SAFIR [4], [5]. Fire protection specific research project called LARGO is studying event tree analysis and fire hazard analysis methods and tools for measuring fire protection safety level of nuclear utilities. Main emphasis of STUK has been to follow FRNC cable solutions and corresponding DiD principles.

Examples of the Defense-in-depth Principle in Fire Protection

One practical question in the Olkiluoto 3 case as well as cases in future is that it is better to have FRNC cables without sprinklers compared to the traditional situation of PVC coated cables with sprinklers. FRNC material properties compared to traditional cable coating materials in fire scenarios assure improved fire safety at least in the ignition phase and during fire propagation. The DiD considers from a larger point of view the isolation and extinguishing of fire. For example, sprinklers, shutdown of ventilation and fire dampers are studied as equal from a safety point of view. Moreover, fire fighting and rescue operation can be done under better conditions compared to traditional cable material fires.

Figure 2 shows another kind of DiD case from an outage in 2010. The picture of the big fire load was taken by STUK's resident inspector during his common visit around. The situation was corrected immediately and further on it was checked and ensured that design and work order protocols were in order. Also training, work and quality supervision records of the utility were in order. Fire guarding status was unclear. In spite of all these safety precautions it was possible to bring in enough unallowable flammable liquids and fire loads, which could destroy practically everything in the fire compartment and further on could endanger the safety of the whole containment. DiD studies are useful also for assessing working protocols in order to reject this kind of cases.

Figure 2 Unallowable fire load and flammable liquids during outage 2010 inside containment

RESULTS FROM FIRE SAFETY RESEARCH OF FRNC CABLES

Research of typical FRNC cable types to be installed in OL3 was performed during 2008 to 2010 at VTT in Finland. Five power cable types as well as five instrumentation and control (I&C) cable types were studied using thermogravimetric analysis (TGA), differential scanning calorimetry (DSC), cone calorimeter experiments, heating experiments in a small laboratory furnace and flame spread experiments on 2 m long preheated cable samples in vertical position. The considered cables differed in cable structure and sheath, filler and insulation materials.

TGA is a method to study thermal decomposition as a function of temperature. The experiments were carried out both in nitrogen and air environment, while the mass of the small sample (10 – 20 mg) was monitored during the heating. The experiments showed that meaningful mass loss started around 300 ℃.

DSC measures the rate and degree of heat change as a function of time and temperature. The result shows whether the reaction is endothermic or exothermic and further on, the heat of reaction can be obtained. For example, the results pointed out big endotherm around 370 to 400 ℃.

Cone calorimeter experiments were performed to measure heat release rate, mass loss rate, ignitability and effective heat of combustion. The measured heat release rate of tested I&C cables was clearly higher compared to power cables, but no big differences were found out in the total heat release per total mass loss.

I&C cable samples 100 mm in length were heated in a small laboratory furnace to study softening and melting of the cable material at elevated temperatures. The cables auto-ignited at about 400 to.410 ℃ (see Figure 3)

Figure 3 Softening and ignition of I&C cables in a small laboratory furnace

Flame spread experiments of 2 m long cable samples were performed using a special test rig, where cable samples were pre-heated with hot air. After pre-heating to a desired temperature, the sample was ignited with a small burner. Thereafter, the flame spreading was observed by numerous thermocouples along the sample. The preheated temperature range varied from 22 ℃ to nearl y 300 ℃.

Slow, continuous flame spreading resulted in power cables experiences: the flame spread rate was 1 – 6 mm/min in case of pre-heating temperature varying between 90 and 300 ℃.

Considering I&C cables, the flame spread rate was slower, about ~1 mm/min in case of pre-heating temperature varying 178 and 304 ℃. Com plete burning of I&C cable samples took place only in 3 out of 10 tests (when pre-heated to 226, 239 and 286 ℃). Thus, in these tests the fire behavior of the I&C cables differed from power cables.

Also a typical PVC cable was tested similarly and the flame spread rate varied from 2.5 to 8 mm/min (see Figure 4). However, PVC cable could not be pre-heated to higher temperature than 190 ℃ due to softening.

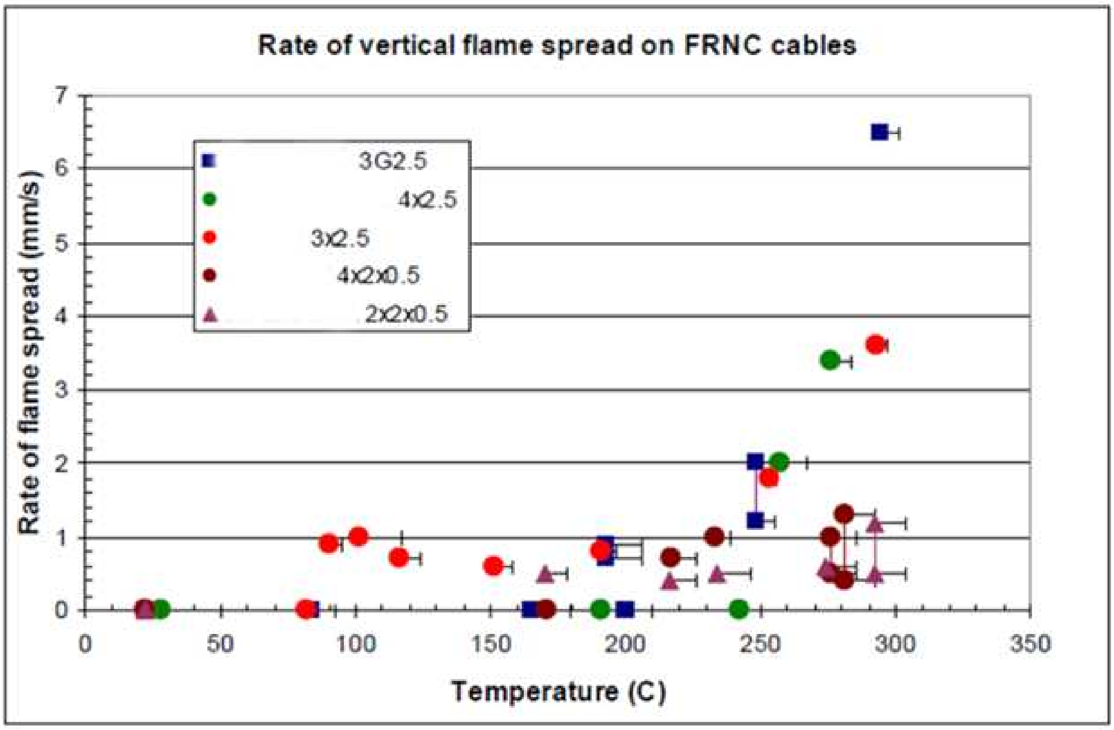

Figure 4 Rate of flame spread of three power cables and two I&C cable types

ACTIONS AFTER THE FUKUSHIMA ACCIDENT

The Fukushima accident in March 2011 caused a global need to update risk assessment of NPPs and related nuclear facilities. STUK and the licensees in Finland started immediately after the Fukushima accident assessments of extreme natural hazards combined with the hypothesis of station black out (SBO) in order to assess protection against loss of ultimate heat sink and core damage. Further on Finland is participating in the European stress tests, which are coordinated by the European Commission (EC) and WENRA (Western European Nuclear Regulators Association). Fire risks are studied as consequences initiated by natural phenomena. Main external hazards in that respect in Finland are earthquake and flooding. Another issue is to consider multi-unit site requirements.

STUK reported first assessment on preparedness of the Finnish nuclear power plants against external events and loss of power supply to the Ministry of Employment and the Economy 16th May 2011. In the domain area of fire protection first actions will be related to ensuring fire protection systems against earthquakes and SBOs. Seismic qualification is not required for all fire fighting tools, but the verification of fire fighting systems and fire compartments integrity require sensitivity studies against beyond design basis earthquake. Seismic design requirements also for fire protection are therefore under consideration.

Same causes which may cause SBO compromise also A.C. power for pumping of fire fighting water. Adequacy of current diesel driven fire water pumps and doubled electric power connections require more evidence.

Natural catastrophe may also injure surrounding infrastructure, which could result into situation where fire brigade and rescue operation could be unavailable. Severe accident related readiness for extra protection of corrective actions, fire fighting and rescue operation are also under consideration.

Finland is participating in the European stress tests and the work for nuclear safety in this forum is continuing. Current status is that first assessments are performed for all operating NPPs, Olkiluoto 3 unit, which is under consideration, and for two units, which have parliament gratified positive decision in principle to apply in time construction license. STUK has asked further questions from licensees in order to complete required assessments. Fire protection has small, but important part in these studies.

STUK is following international development of nuclear safety launched by Fukushima accident. IAEA (International Atomic Energy Agency) and WENRA are in important position from STUK's point of view. IAEA will publish methodology guidelines for design safety margin assessment for the need to continue global stress tests in the near future. The main target will be then to assess design basis based strengths of nuclear utilities against extreme natural hazards. STUK is active in this area. Available information will be used also in Finnish nuclear safety regulation development.

CONCLUSIONS

STUK is developing detailed regulations for safe use of nuclear energy and to physical protection, emergency preparedness and safeguards. The fire protection guide is under update. Recent findings including those from the Fukushima accident have brought into common knowledge that design requirements have to be increased against extreme natural hazards. In the fire protection area most important natural hazards in Finland, which require further study, are earthquake and flooding. Corresponding qualification needs for fire protection are under consideration.

Defense-in-depth principle is presented as a main tool to develop nuclear safety in Finland. It is also important to enhance clearance of fire protection guidelines. In that respect most practical way is to state corresponding safety requirements and STUK's responsibilities in-line with licensing steps for nuclear power plants.

Laboratory test demonstrated continuous burning along vertical FRNC cable samples under certain conditions. Therefore, elimination of "transient fuel fires" is important to prevent critical pre-heating of cables. Procedures to control and limit the amount of transient fuels are important part of DiD in fire protection.

Isolation of fire compartment (e.g. closed fire doors, closure of fire dampers, stopping ventilation systems) is important to limit fire propagation and spreading, if a cable fire starts to propagate. In such cases, fire dampers are seen as most critical components to prevent long-term fires in cable spreading rooms and to prevent spreading of fire impacts to neighboring fire compartments.

To assure adequate fire safety, test interval and technical specifications of certain fire dampers need to be defined taking into account risk-informed measures.

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ENHANCEMENTS IN INTERNATIONAL GUIDELINES FOR FIRE PSA

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ABSTRACT

Probabilistic fire safety assessment (Fire PSA) to investigate the safety level of nuclear installations, in particular of nuclear power plants, with respect to internal fires is to-date recommended or required in many countries, e.g., in the frame of periodic safety reviews and in safety analyses for long-term safety.

Therefore, on national and international level guidelines have been elaborated or extended to cover fire PSA and to set the frame what information is requested and which scope and quality of the fire PSA is expected.

The presentation will describe the requirements and recommendations provided by the Western European Nuclear Regulators' Association and by the International Atomic Energy Agency in different safety guides and safety reports or by the fire protection guidelines issued by the Nuclear Pools' Forum, but it will also address national guides which have a broad application in countries all over the world such as the respective ASME/ANS guide.

Enhancements are also developed to enlarge the German PSA documents with respect to fire PSA for low power and shutdown states which are currently not required in PSA for periodic safety reviews. The changing conditions and additional aspects which have to be taken into account for performing a fire PSA for this plant state is shortly described.

WENRA REFERENCE LEVELS

One of the first aims of the Western European Nuclear Regulators' Association (WENRA) was to develop a harmonized approach to reactor safety. For that purpose, so-called reference levels have been determined which should be fulfilled in all member states of the European Union (EU) and which are also considered outside Europe as one document to be taken into account.

One of the 18 safety issues covers the protection against internal fires [1] stating, among others, that:

"A fire hazard analysis shall be carried out and kept updated to demonstrate that the fire safety objectives are met, that the fire design principles are satisfied, that the fire protection measures are appropriately designed and that any necessary administrative provisions are properly identified.

The fire hazard analysis shall be complemented by probabilistic fire analysis. In PSA level 1, the fires shall be assessed in order to evaluate the fire protection arrangements and to identify risks caused by fires."

A further safety issue is probabilistic safety analysis (PSA) stating under scope and content of a PSA [1]:

"For each plant design, a specific PSA shall be developed for level 1 and level 2 including all modes of operation and all relevant initiating events including internal fire and flooding."

IAEA DOCUMENTS

A series of documents has been provided by the International Atomic Energy Agency as safety guides, safety reports and technical documents.

A Safety Report [2] has been elaborated earlier outlining good practices in conducting fire PSA for plant internal fires at nuclear power plants (NPP) as well as assisting in integrating the threat of a fire into an existing level 1 internal events PSA. Specific details of various aspects of a PSA for the plant internal event fire are globally limited. The report concentrates on the procedural steps for a fire PSA; however the tools needed to implement these steps remain the choice of the analyst.

This Safety Report can be used to assist in implementing a PSA for fire in nuclear power plants on the basis of the current practical experience gained in this area. A particular goal is to promote a standardized framework, terminology and form of documentation for PSA that will facilitate an external review of the results of such studies.

However, internal fire PSA has become in the meantime a mature method which could be applied during the design phase, in particular for the review of operating plants. Currently, a Safety Guide on level 1 PSA has been issued [3]. Part of this Safety Guide is the description on performing internal fire PSA. According to [3], the internal fire PSA process typically includes the tasks shown in Fig. 1.

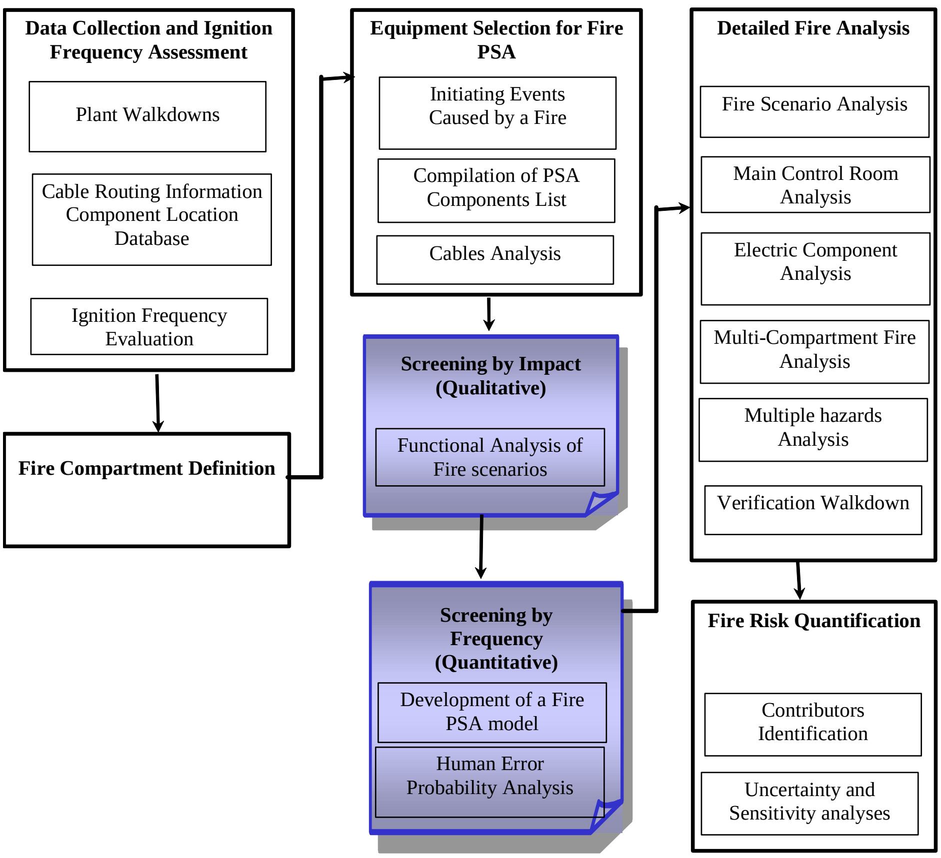

Figure 1 Process for plant internal fire PSA (from [3])

The internal fire PSA should take into account the possibility of a fire at any plant location, fire detection, suppression and confinement; the effects of fire on safety related components and cables, the possibility of damage to these equipment, and, in case of severe fires, to the structural integrity of walls, ceilings, columns, roof beams, etc. Physical separation (i.e. realized by fire barriers) between redundant safety trains may limit the extent of fire damage, therefore quantification of the core damage frequencies (CDF) should generally include those equipment failure probabilities not being affected by the fire, e.g., random failure probabilities, and the likelihood of a maintenance outage.

It is stated that internal fire PSA methods should introduce the likelihood of a fire at any plant location, the effects of the fire on pieces of equipment (components as well as their associated instrumentation and control cables and the supplying power cables), and the impact of equipment failures and human actions coincident with the fire. Deterministic fire hazard analysis (FHA) should provide an important input to internal fire PSA, for example the list of components and cables and their locations, and functional and detailed fire impact analyses performed for designing the fire protection features.

The internal fire PSA approach is based on a systematic analysis of all plant locations. To facilitate this examination, the plant should be subdivided into distinct fire physical units ('fire compartments'), which are then scrutinized individually.

CURRENT STATUS IN THE USA

One acceptable approach for determining the technical adequacy of a PRA (probabilistic risk assessment) is described in [4], providing the technical characteristics and attributes of an internal fire analysis needed.

Information on the EPRI/RES consensus document on fire protection risk assessment is described in [5] and [6]. This document is a detailed compendium of methods and technical bases to estimate risk associated with internal fires in a nuclear power plant, covering a wide range of disciplines, including fire initiation and effects, impact of fire on plant cables and circuits, and plant response to fire generated conditions.

Volume 2 of the EPRI/RES consensus document [5] provides a detailed tool for use in risk informed applications that include estimating changes in risk associated with changes in plant design and/or operational configuration.

In the frame of a current research project, EPRI will, in conjunction with other stakeholders, refine the data, tools, methods, and guidelines needed to support realistic assessments of the risks associated with fire. These efforts will produce new databases and practical guidance for performing fire PRA.

A number of different research tasks are underway to account for the damage caused by fire to be treated in a PRA model. This includes work to understand the types of system responses that might result from damage to electrical cables (both ac and dc), the way in which control room fires might evolve, and the treatment of human reliability. The refined understanding of the potential for damage from the previous research area, coupled with enhanced methods for incorporating this understanding into the PRA models, will result in more realistic risk estimates.

Advances in fire PRA methodology will require progress in several areas. NPP fire research must lead to more refinements of NUREG/CR-6850, including the development of quantitative methods to estimate fire risk during low power and shutdown states [7].

NFPA 551 [8] is another new document "identifying various types of fire risk assessment methods and describing the properties these methods should possess". This guide is intended to provide assistance, primarily to authorities, in evaluating the appropriateness and execution of a fire risk assessment for a given fire safety problem. While this guide primarily addresses regulatory officials, it also is intended for others who are involved in reviewing fire risk assessment such as insurance company representatives.

ANS (American Nuclear Society) has published a fire PRA standard [9], setting forth requirements for fire probabilistic risk assessments (fire PRA) used to support risk-informed decisions for commercial light water reactor (LWR) nuclear power plants and prescribing general requirements for fire PRA practice intended to suit a wide range of applications. This standard covers fires occurring within the plant. This document has been meanwhile incorporated in the standard for level 1 PRA [10].

However, the fire PRA peer review process guide is still under discussion (see [11] and [12]) between the Nuclear Energy Institute and the U.S. Nuclear Regulatory Commission (NRC).

GUIDELINES OF INSURANCE COMPANIES

The fourth edition of the International Guidelines for the Fire Protection of Nuclear Power Plants [13] incorporates new technology and the inspection experience of pool engineers since 1997. Performance based methods to analyze fire risk have been introduced since the third edition, and this technology is now being applied.

Included in these Guidelines are topics and issues that are often not fully considered by national authorities' regulations, but have proven to be important to operators and insurers alike. Nuclear Regulatory Authorities consider fires but, generally, only from the standpoint of their effects on nuclear safety. The Pools' insured experience demonstrates that major fires can occur in the conventional areas of nuclear plants; most do not prejudice nuclear safety, but all can have significant economic impact on the nuclear power plant operator's financial status. Property damage may cost substantial amounts of money, and forced outages of a year or even longer may result in very large loss of generating revenue.

GERMAN PSA GUIDANCE DOCUMENTS

The German PSA Guide contains reference listings of initiating events for nuclear power plants 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 for the analysis of plant internal fires, fire frequencies and unavailability of fire detection and alarm features as well as data, e.g., on the reliability of active and passive fire protection means are provided in the technical documents on PSA methods [14] and PSA data [15].

For fire risk assessment conducted for nuclear power plants in Germany, different screening approaches are applied to identify critical fire zones. The models proposed have been successfully applied in fire risk studies for German nuclear power plants.

The screening methodology has been further improved [16], in particular with respect to the selection of compartments and plant areas relevant to fires. In addition, an uncertainty and sensitivity analysis has been performed for the reference plant fire PSA providing not only mean values for fire induced core damage frequencies but also quantifying major uncertainties increasing the level of confidence of the fire PSA results.

For the detailed quantitative fire risk analysis, a standard event tree has been developed with nodes for fire initiation, ventilation of the fire compartment, fire detection and suppression, both as well for the pilot fire phase as for fully developed fires, and a node for fire propagation.

The standard event tree has to be adapted to every critical fire zone, revealing the following results:

– Frequency and nature of fire initiating events,

Llist of equipment damaged, binned corresponding to different damage states, and

– Damage frequencies.

If a complete plant specific PSA is available, the fire induced hazard state frequencies will be summarized for all initiating events and specified as input to the corresponding event tree of the level 1 PSA. Furthermore, the plant hazard states have to be introduced into the fault trees. The plant hazard state frequencies are estimated for each transient as the sum of the single event core damage frequencies. The total plant hazard state frequency is obtained by adding up the contributions of all transients. Moreover, the core damage frequency has to be calculated.

In this context, it has to be stated that the requirement to use only qualified PSA codes has also to be met for fire PSA. Moreover, validated fire simulation models and codes have to be applied in case of deterministic fire hazard analysis and probabilistic fire risk assessment.

The above mentioned PSA documents describe the approach for a fire PSA to-date only for full power plant operational states.

Meanwhile the approach was adapted from full power plant operational states to low power and shutdown states [16].

The screening for low power and shutdown states has been performed in the same manner as for full power plant operational states. However, particular differences may result from maintenance and repair activities including hot work. In addition, hot work activities correspond to the presence of personnel in the affected compartment resulting in an early manual fire detection and suppression.

Recent German research activities have demonstrated that an evaluation of the hot work permits is particularly needed for observing and considering potential peculiarities during these activities typically performed during low power and shutdown states from the beginning of the analysis and being able to consider those time periods explicitly for the fire occurrence frequency estimation.

One further important finding is that during low power and shutdown states many connections between compartments are pessimistically assumed open due to special activities being performed creating frequently conditions where fire barrier elements are being left open or blocked due to practical reasons for maintenance and repair activities.

On the other hand, the higher presence of personnel during these periods enables an early detection of situations which may create a fire with manual fire detection and suppression during the incipient fire phase.

It is intended to include the approach for performing fire PSA also for low power and shutdown states in the revision of the German PSA guidance documents.

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SAFETY FIRE ZONING AND FIRE VULNERABILITY ANALYSIS OF LING-AO NUCLEAR POWER PLANT UNITS 3 AND 4

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ABSTRACT

Compared to the reference plant LING-AO Nuclear Power Plant, Units 1 and 2 (LING-AO Phase 1), safety fire zoning and fire vulnerability analysis are the significant fire protection design improvements at LING-AO Nuclear Power Plant, Units 3 and 4 (LING-AO Phase 2). The paper gives an overview on the safety fire zoning and fire vulnerability analysis as implemented at LING-AO Phase 2. With regard to problems encountered in the design, the authors give some personal advice.

INTRODUCTION

LING-AO Phase 2, a pressurized water reactor (PWR) type nuclear power plant (NPP) with a capacity of 2×900 MW, is designed based on the reference plant LING-AO Phase 1. Unit 3 has been put in commercial operation in September 2010, while Unit 4 will start commercial operation in August 2011.

Besides the "Design and Construction Rules for Fire Protection in PWR Nuclear Power Plants" (the French RCC-I 83) [1], LING-AO Phase 2 also partly meets the requirements of the "Construction Rules for Fire Protection in PWR Nuclear Power Plants" (RCC-I 97) [2]. According to RCC-I 97 [2], fire protection aims at:

a. ensuring the safety of individuals,
b. guaranteeing the performance of safety functions,
c. limiting damage to equipment which could result in long-term unavailability.

Objective b) is achieved by:

1. Safety fire zoning: separating physically the most important redundant equipment and cable trays,
2. Fire vulnerability analysis (also called common mode analysis): aiming at separating the common modes being able to subsist inside the safety fire volumes (internationally called fire compartments).

In LING-AO Phase 2, all the buildings of the nuclear island are sub-divided into fire zones; a fire vulnerability analysis is performed for each safety fire volume according to the criteria a) to c), d) as defined in RCC-I 97 [2].

SAFETY FIRE ZONING

The safety fire zoning consists of sub-dividing the concerned buildings of the nuclear island into fire volumes (safety fire volumes or basic fire volumes). It is designed to separate as much as possible the most important redundant safety equipment and cable trays in the buildings of the nuclear island.

Safety fire zoning into fire volumes is the basis for the fire safety demonstration as well as for the safety fire vulnerability analysis. Assuming that a fire cannot propagate out of a given fire volume, most of the fire protection studies and the supporting analyses are carried out on a fire volume basis.

Methodology Applied for LING-AO Phase 2 Safety Fire Zoning

The safety fire zoning of LING-AO Phase 2 has been obtained using a repetitive approach.

The following aspects were considered in a first step:

– Existing fire volumes of the reference plant LING-AO Phase 1,
– Boundaries of existing structural elements (walls, ceilings, floors, beams, etc.),
– Potential mechanical common mode failures to be avoided,
– The estimated "majority electrical channel" in order to limit the cable common modes that need fire resistant protections.

Then, in a second step, the following aspects were considered:

– Detailed analysis of structural elements (walls, doors, opening, etc.),
– Detailed analysis of ventilation systems (ducts, dampers, etc.).

This repetitive approach aims at finding a compromise between large safety fire volumes that will reduce boundary requirements, such as fire resistant doors, safety fire dampers, fire resistant penetrations, etc. and small safety fire volumes that would reduce the cable common mode fire resistant protections.

Fire Volume Types of LING-AO Phase 2

The fire volume types defined within the frame of the LING-AO Phase 2 safety fire zoning are the following:

– the SFS, safety fire sector – the ZFS, safety fire zone	constituting safety fire volumes (VFS),
– the ZFA, fire access zone – the ZNS, non-safety zone	constituting basic fire volumes.

Safety Fire Sector (SFS)

A SFS is characterized by:

– Physical separation,
– An obligatory required fire resistance rating of 1.5 h (= 90 min). and
– The following characteristics of rooms with safety related equipment and high fire load density:
  – Physical separation does not eliminate all common mode failures (MC).
  – If any MC remains, an analysis report is provided to define how additional fireproof devices can be used to prevent them from being damaged simultaneously in the case of fire.

Safety Fire Zone (ZFS)

A ZFS is characterized by:

– Non-obligatory fire resistance rating = max [DSdF]* not less than 1.0 h (= 60 min), or,
– Geographical separation (ZFS may contain openings in walls).
– The following characteristics of rooms with safety related equipment and low fire load density (< 400 MJ/m^2[2)]:
    – A fire risk analysis must be carried out for all openings (ZFS support report).

* DSdF: Significant duration of fire
This is the value used to determine the (required) fire resistance rating of walls in a room within a fire volume limit relative to the design basis fire when the value of 90 min is not required.

Fire Access Zone (ZFA)

A ZFA is characterized by:

– Physical separation: obligatory required fire resistance rating of 1 h (= 60 min),
– Typical characteristics: escape routes with no safety related equipment and low fire load density.

Non-Safety Zone (ZNS)

A ZNA is characterized by:

– No requirements for ZNS walls, except those which are adjacent to a VFS or a ZFA,
– Typical characteristics: No items of equipment selected for the fire common mode analysis and no rooms relevant for security reasons.

Table 1Table 1, based on RCCI-97 [2], illustrates the fire resistance rating of each boundary (fire barrier) met within the LING-AO Phase 2 safety fire zoning.

It has to be taken into account in this context that it is assumed that no room significant fire duration [DSdF] will exceed a value of 1.5 h (= 90 min). In case that the calculated fire duration will exceed 90 min (this may possible I a few rooms in the electrical building) the presence of fixed fire fighting systems is required permitting the DSdF to be reduced to 90 min.

Table 1 Fire resistance rating of fire volume boundaries (fire compartment barriers)

	SFS	ZFS	ZFA	ZNS
SFS	90 min	90 min	90 min	90 min
ZFS	90 min	max. DSdF	max. DSdF	max. DSdF
ZFA	90 min	max. DSdF	60 min	60 min
ZNS	90 min	max. DSdF	60 min	not applicable

FIRE VULNERABILITY ANALYSIS (FVA)

The fire vulnerability analysis is the last step in the fire common mode prevention process. It will prove that there is either no remaining fire common mode or it will demonstrate that the remaining fire common modes are acceptable from the safety point of view.

The analysis is performed for each safety fire volume via four analytical steps:

1. **Potential fire common modes detection**

   For each safety fire volume, potential fire common modes are detected using the criteria a) to d) as they defined in RCC-I 97 [2]:

   – Criterion a):
   There is a fire common mode according to the criterion a), if there are safety classified mechanical equipment or electrical cables belonging to two redundant trains of the same system ensuring a safety function installed in the same safety fire volume.

   – Criterion b):
   There is a fire common mode according to the criterion b), if there are safety classified mechanical equipment or electrical cables installed in the same safety fire volume: belonging on the one hand to one redundant train of a system ensuring a safety function or, on the other hand, belonging to systems necessary for the operation of the redundant train (support function).

   – Criterion c):
   There is a fire common mode according to the criterion c), if there are electrical connections in the same safety fire volume which do not fit to the above mentioned categories a) or b), but which are supplied by redundant electrical switchboards and whose number is such that the selectivity of the electrical protections of these switchboards may fail. During the analysis, only those electrical connections present in the same room are taken into account.

   – Criterion d):
   There is a common mode according to the criterion d), if there is equipment installed in the same safety fire volume, whose failure in case of fire may lead to an accidental or complementary operating condition of the equipment required to ensure a safety function necessary for mitigating the event under consideration.

   In LING-AO Phase 2, fire protection features have been installed to eliminate as far as possible the risk of accidents related to a fire in a safety volume, particularly for the inadvertent opening of a SEBIM valve and the inadvertent opening of a GCT atmosphere valve.

2. **Function analysis**

   After having detected all the potential fire common modes, a functional analysis is performed to check that safety functions are still available in the event of a fire.

   Considering assumptions about the potential conditions of a fire to break out, the conclusions of the functional analysis can be:

   – The potential fire common mode is not confirmed. That means that the safety functions are still ensured, because, for example, a functional redundancy is still given due to other equipment are not damaged by the fire, or

   – The safety functions are not ensured and the potential fire common mode is considered as a "confirmed fire common mode". In this case, a fire risk analysis should be performed.

3. Fire risk analysis

A Fire risk analysis (FRA) should be performed, if the common mode is confirmed from the functional point of view. According to physical parameters related to the safety fire volume, the fire risk analysis can justify that, even if two redundant pieces of equipment or cables are installed in the same safety fire volume, at least one piece of equipment will be available, thus the common mode is not confirmed from the fire risk analysis point of view. In this case, no specific treatment is required. Otherwise, the fire common mode must be treated regarding fire effects.

The Fire Risk is assessed considering the following information:

– Characterization of the fire risk in the room and in the safety fire volume,
– Significant fire duration of the room (DSdF),
– Type of equipment concerned by the analysis,
– Location of the pieces of equipment in common mode,
– Fire load locations.

4. Confirmed fire common modes treatment

The treatment of confirmed fire common modes consists mainly of installing qualified passive fire resistant protection means. Fire resistant screens, ceiling transoms, fire resistant cable (tray) wraps, fire resistant enclosures, etc., are considered as passive fire resistant protection means. An example of a fire resistant screen is shown in Figure 1.

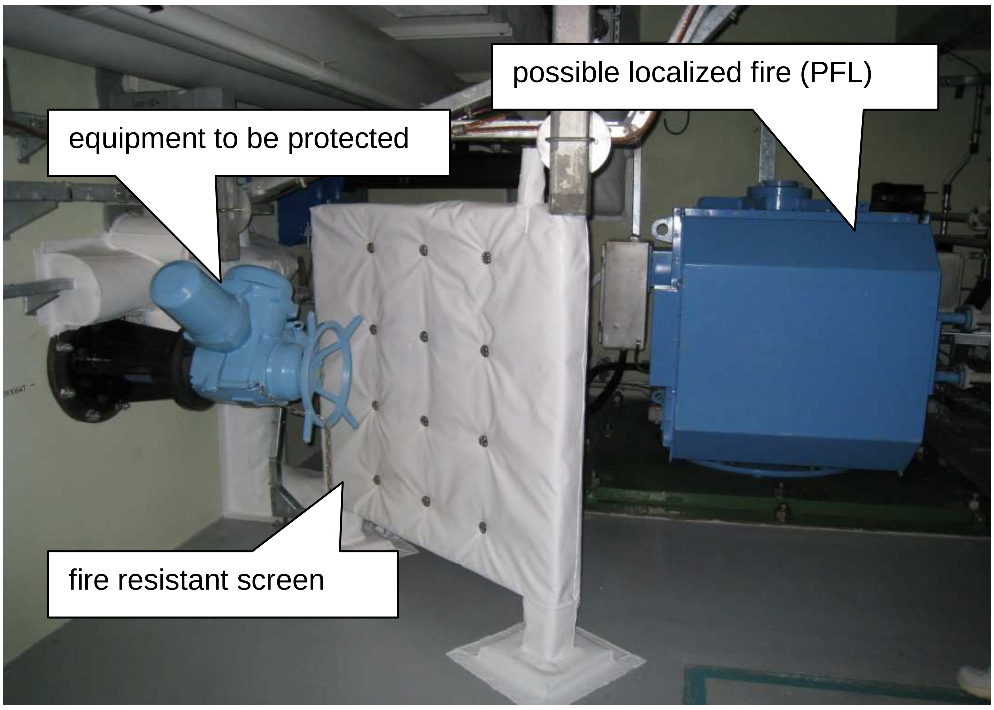

Figure 1 Fire resistant screen as installed in LING-AO Phase 2

The treatment of confirmed fire common modes can also be performed through modifications as for example:

– Modification of the location of the pieces of equipment in fire common modes,
– Modification of the routing of cables in fire common mode.

PROBLEM AND SUGGESTION

Fire Resistance Rating of Fire Volumes

In order to prevent the spread of fire and its effects (e.g. smoke and heat) from one fire volume to another, and thus to prevent the failure of redundant items important to safety, fire should be confined within the fire volume where breaking out initially. A fire volume, internationally also called fire compartment, is one room or a group of rooms that is completely surrounded by qualified fire resistant barriers: all walls, the floor and the ceiling. The fire resistance rating of the barriers should be sufficiently high that total combustion of the fire load in the compartment can occur (i.e. total burnout) without breaching the fire barriers.

According to the nuclear safety Guide "Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants" published by the International Atomic Energy Agency (IAEA, NS-G-1.7) [3], the fire resistance rating of the barriers forming the boundaries of a fire volume (compartment) should be established in the fire hazard analysis. A minimum fire resistance rating of one hour should be adopted. National regulations may require higher values for the minimum resistance rating of the fire volume (fire compartment) boundary.

Fire resistance rating requirements of LING-AO Phase 2 are shown in Table 1. During fire hazard analysis, verification is carried out to check that the duration of the design basis fire in the fire volume is less than the fire resistance rating of the fire volume boundary, using the ISO 834 curve and the DSN 144 curve as shown in Figure 2, where the following is assumed:

- The fire resistance rating is evaluated by normative tests using the ISO 834 curve.
- The fire volume boundary duration is calculated using the DSN 144 curve,
- The fire volume boundary duration is a function of the fire load density.
- The fire resistance rating of a fire volume is deemed appropriate, if the fire resistance rating is larger than the fire duration in the fire volume.

The DSN 144 curve was called into question by the French Nuclear Safety Authority (ASN) requesting to revise it on the basis of our improved understanding of the spread of fire.

In fact, the fire duration is influenced by the chemical nature of the combustible material, and the DSN 144 curve is quite well adapted to rapidly burning fires. 95 % of the fire loads in nuclear power plants produce slowly burning fires (electrical equipment and PVC insulated cables). For this type of fire the temperature of hot gases is lower, but it can last much longer and eventually exceed the duration.

At the time being, investigations to review the DSN 144 curve are going on. A conservative way to manage the situation is to enlarge the margin between the fire resistance rating and fire duration (only if there is no extinguishing system installed in the room). In LING-AO Phase 2, a margin of 10 min is ensured in all safety fire volumes.

The fire resistance rating of the fire volume boundary may be also affected by the physical and geometrical factors of fire volume as well as of the ventilation conditions of fire volume. In order to meet the requirements of safety fire zoning a new method should developed to determine the fire resistance rating of the fire volume taking into account the realistic fire curves in the rooms and the fire volume elements' performance in case of fire.

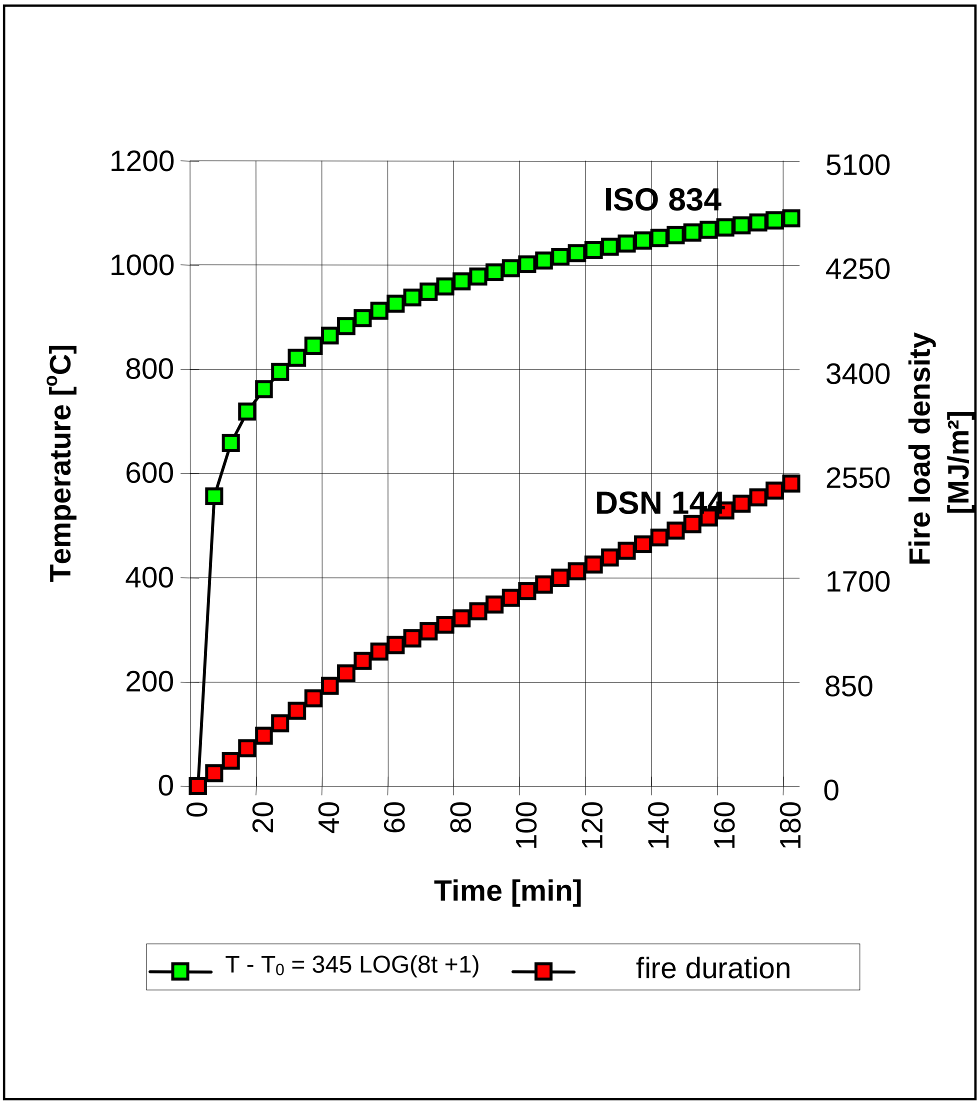

Figure 2 DSN 144 and ISO 834 curves

Invalidity Analysis

In LING-AO Phase 2, the fire risk analysis is carried out mainly in the light of the methodology implemented in the French EdF Fire Action Plan (PAI) for similar PWR power units of the 900 MW_e series, the so called CPY, which is based on the study of thermal radiation, propagation of the fire and hot gas according to current knowledge on the subject.

During LING-AO Phase 2 fire risk analysis a large number of parameters are involved in the demonstration, such as nature of the combustible, geographical distance between the combustible and the target equipment, concentration of the fire load, equipment malfunction criteria, etc. Compared to PAI, a digital control system (DCS) is applied in the LING-AO Phase 2 I&C (instrumentation and control) design, so the influence of DCS has to be taken into account.

If a measurement cable is damaged by the fire, two situations are considered:

– Short circuit on the cable corresponding to a significant overshoot of the high end of the measuring range (e.g. intensity > 25 mA for a 4 – 20 mA sensor),

- The cable is cut off corresponding to a significant overshoot of the low end of the measuring range (e.g. intensity < 3 mA for a 4 – 20 mA sensor).

In order to perform the detailed functional analysis, the management of these situations hereinafter should be clarified:

- The invalidity detection in case of fire, that means to check that the damage initiated by the fire can be detected automatically by DCS through its permanent monitoring,
- The behavior of DCS in case of invalid measurements.

For example, the reactor power range is monitored by RPN 010MA, 020MA, 030MA and 040MA sensors. These sensors can initiate a "power range high neutron flux" reactor trip signal in situations similar to uncontrolled rod accident with logic of 2/4.

If the FVA is performed in a safety fire volume, where RPN 010MA, 020MA, 030MA and 040MA cables are located, it has to be checked first if the DCS can detect invalid measurements in case of fire and if the measurements have to be considered as invalid. If the answer to the first question is yes, what will be the consequence on the RPR logic of invalid measurement?

After confirmation with the DCS supplier, we know that if the fire damaged the RPN power range measurements, the relevant measurement will be considered as invalid by the RPS, and according to the RPS voting logic, the reactor trip order will be initiated automatically as soon as two sensors of RPN are damaged by the fire. So, the aggression of the fire will not affect the safety of the unit and no treatment is required for the common mode.

Regarding to the DCS invalidity character in the case of fire, a case by case study should be performed in the FVA.

Fire and Earthquake Coexistence

There is no particular requirement regarding the coexistence of fire and earthquake in RCCI- 97. Moreover, up to the time being, no study has been performed to check the impact of the coexistence of fire and earthquake on the fire vulnerability analysis.

In LING-AO Phase 2, the earthquake can initiate new unavailability such as:

- RCP pumps,
- Normal spray,
- Decrease of the excess,
- RCP depressurized level (US): RCP 300 MN,
- Seal injection flow-rate: RCV 021 to 023 MD.

The consideration of this unavailability can have an important impact on the functional analysis performed within the fire vulnerability analysis and may lead to more fire protection features installed in a nuclear power plant or result in more seismic qualification requirements for the equipment concerned.

Given the EdF experience feedback about the coexistence of fire and earthquake, LING-AO Phase 2 follows the same strategy as EdF CPY PAI:

- An earthquake is not assumed to be the initiator of a fire.
- The coexistence of an earthquake and a fire is not acknowledged.
- A fire is not assumed to break out after an earthquake (even in the long-term after the earthquake).

Nevertheless, the seismic qualification is required for fire fighting equipment taking part in the safety fire zoning justification. The qualification is just an improvement of the safety in case of earthquake (defense-in-depth) but it is not enough to demonstrate that the plant can be operated until the safe shutdown when a fire breaks out after an earthquake.

According to the U.S. American Standard of the National Fire Protection Association for Fire Protection for Advanced Light Water Reactor Electric Generating Plants (NFPA 804, 2006) [4], a risk assessment demonstrating the potential risk from a seismically induced fire in relationship to the plant's core damage frequency shall be prepared to evaluate the level of safety of the plant. But there is no report about this type of risk assessment up to now.

For the EPR type PWR nuclear power plant, an independent fire is postulated in the post-accident long-term phase not earlier than two weeks after a design basis earthquake [5].

A random combination of events may represent an extremely unlikely scenario and be discounted. But after the Fukushima NPP accident, it is better for us make an introspection about multiple severe events concurrence such as the coexistence of fire and earthquake.

CONCLUSIONS

The fire code applied in LING-AO Phase 1 fire zoning is RCCI-83 [1] (and its application note), which does neither require a safety fire demonstration nor a safety fire vulnerability analysis. The fire zoning of LING-AO Phase 1 is based on an EdF CPY type plant fire zoning before PAI, which was much more a fire load zoning. Indeed, the RCCI-83 concentrates on the actual fire load installed in the plant, however the presence of safety related actuator sensors or cables was not studied in detail. Consequently the methodology applied for LING-AO Phase 1 is appropriate for buildings with high fire loads such as the electrical building, but it is less suitable for buildings with low fire load such as the fuel building or the reactor building.

In LING-AO Phase 2, as in EdF CPY after PAI and according to RCCI-97, all buildings of the nuclear island bare subject to safety fire zoning. Accompanied by the improvements of fire zoning, many corresponding modifications are implemented in LING-AO Phase 2, such as much more fire resistant doors and fire dampers being installed in the fire barriers (boundaries) of the fire volume. Parts of the steel grating floors are replaced by fire resistant ones to ensure the fire retention, etc.

There is no fire vulnerability analysis in LING-AO Phase 1. The fire common mode detection is carried out according to rules based on the train information of the cable and performed only in fire zoned buildings. Moreover, the fire common mode is systematically protected and no analysis of the consequences of the fire common mode on the safety of the unit is performed. While in LING-AO Phase 2 the fire common mode detection is performed in all the safety fire volumes, a functional analysis will confirm the necessity to treat the fire common modes according to the consequences on the safety of the unit. Compared to LING-AO Phase 1, the fire common modes are excluded drastically and less fire protection features are required in LING-AO Phase 2.

In view of the increasing international fire safety requirements, further progress is needed for enhancing fire safety: A much more scientific method has to be developed to determine the fire resistance rating of fire volumes. Further on, the relevant study of fire and accident (including earthquake) coexistence, and a detailed study about the malfunction criteria of equipment with respect to fire have to be carried out.

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FIRE DYNAMIC CRITERIA FOR COMPARTMENT SCREENING IN THE FRAME OF FIRE PSA

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ABSTRACT

In order to reduce the number of compartments to be analyzed in a Probabilistic Fire Safety Analysis (Fire PSA) in Nuclear Power Plants (NPP), a new methodology for a set of filter criteria has been developed. For the definition of these filter criteria, compartment fires with different configurations including fire load, ventilation conditions, and compartment size have been analyzed. An electrical cable located in the hot gas layer (HGL) has been used as being representative of the variety of different appliances in a NPP. An electrical fault occurs and the damage criterion is met, if the maximum cable temperature within the jacket of the cable reaches a specific threshold temperature of approximately 240 °C. If the cable fails, the compartment is relevant for a more specific in-depth analysis in frame of a Fire PSA.

To get the correlation between maximum cable temperature and compartment configuration, the computational fluid dynamics type field model Fire Dynamics Simulator (FDS), Version 5.5.3 and the cable failure model THIEF, which is implemented in FDS, have been used. A validation study has been done regarding the maximum cable temperature in full scale mechanically ventilated compartments. One finding is that the THIEF model under-estimates the maximum cable temperatures with 20 %.

To analyze the effect of compartment configuration and fire dynamics on the maximum cable temperature, a parameter study has been conducted. The growth rate of the fire and the inlet air flow in a compartment with the size of 196 m² and a height of 5 m have been varied. As a result, it has been shown that in compartments with no ventilation and a moderate fire growth rate, cables 1 m below the ceiling and in a horizontal distance of 7 m to the fire source do not fail in case of fire. On a level of 1.4 m below the ceiling cables do not fail in compartment fires with an air exchange rate of up to 2.8 1/h and a moderate fire growth rate.

These first results give an impression of the final set of filter criteria and show that the inlet air flow is more important for maximum cable temperatures than the fire growth rate. In further studies compartment size and height will be varied and all results will be combined in one empirical equation to simplify the new set of criteria for screening purposes in Fire PSA.

INTRODUCTION

A Probabilistic Safety Analysis (PSA) including also a probabilistic fire risk analysis, the so-called Fire PSA, for all plant operational states is mandatory to be performed for nuclear power plants (NPP) in Germany in the frame of the comprehensive Safety Reviews (SR) to be carried out due to legal requirements. In the Fire PSA, the contribution of plant internal fires occurring on-site inside or outside of NPP buildings to the core damage frequency (CDF) is assessed. The fire safety design in NPP basically relies on compartmentation of the whole plant and its buildings by means of structural measures. Following this approach, most methodologies for Fire PSA use a structure by buildings and compartments as starting point. For each compartment it has to be analyzed, if relevant fire scenarios may occur that contribute to the overall CDF. Because of the large number of approximately 700 to 1500 compartments to be considered, it is recommended to apply filter criteria for identifying those compartments for which the contribution of internal fires to the overall NPP CDF is negligible [1], [2].

In a first qualitative analysis, one filter criterion which can be applied according to [1] is a "fire specific" criterion called "fire load criterion". Applying this criterion, compartments with a fire load density (fire load per floor area) of less than 90 MJ/m² can be screened out from further analysis. Technical background of this criterion was the assumption that a fire of that fire load density would not to be able to cause any damage to other components within this compartment or to propagate to other compartments. However, for the fire load criterion neither the justification of this particular value is well documented nor does this criterion take into account varying compartment configurations such as ventilation conditions, fire growth rate, or compartment characteristics [2]. In addition, this criterion is not always valid in case of large fire compartments with locally concentrated fire loads.

To consider these compartment configurations, a new methodology has been developed to screen out those compartments, where a fire of one component neither causes damage to any other component except for that where the fire started, nor can the fire spread to adjacent compartments. According to a methodology for Fire Probabilistic Risk Assessment in U.S. NPP [3], a component may be damaged by fire when it is located in one of the five different zones of influence (ZOI) as shown in Figure 1. The methodology outlined in this paper covers all components located in the upper part of the smoke / hot gas layer (HGL) for conservative reasons. For components being directly exposed to other ZOI with increased thermal impact an additional assessment is necessary.

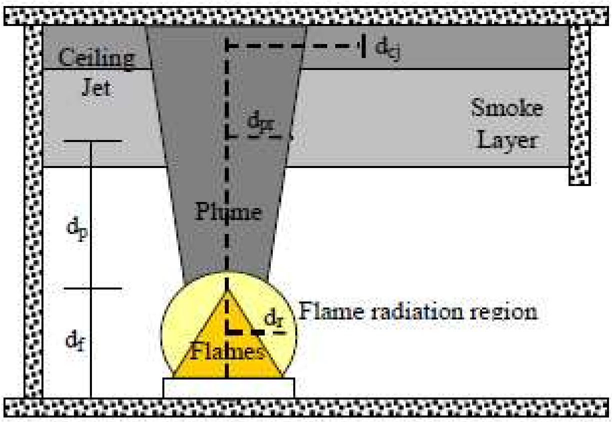

Figure 1 Zones of influence (flames, flame radiation region, plume, ceiling jet, and smoke layer / HGL) in a fire compartment, adopted from [3]

To develop this methodology, first a conservative damage criterion for electrical cables has been defined covering the variety of appliances and components in a NPP. For the damage criterion the model uncertainty was analyzed with a validation and sensitivity study.

After that, significant factors characterizing the compartment configuration have been analyzed concerning their significance on the occurrence of cable damage. In order to receive a quantitative correlation between damage and significant factors, the significant factors were varied in a parameter study and will be taken into account in a set of filter criteria. For the less significant factors, conservative assumptions were taken.

DEFINITION OF A DAMAGE CRITERION

Because of the variety of appliances in a NPP, it is not possible to summarize all the possible damages in one single damage criterion. However, there are several hundreds of kilometers of cable in one NPP supplying all appliances. Moreover, the thermal fragility of cables is important for fire risk evaluation. For these reasons, this methodology focuses on thermal damage of cables [4]. More sensitive components such as electronic devices, which are susceptible to lower thermal impact or smoke damage, are not covered by this methodology.

During the last 30 years, few experimental studies have been performed investigating thermal damage of cables. The results of two studies revealed that concerning their thermal vulnerability, instrumentation and control cables (I&C cables) are less resistant than power cables and that cables constructed with thermoplastic insulation materials are less resistant than cables with thermoset materials [4], [5]. A cable type JE-Y(St)Y 16x2x0.8 represents a typical I&C cable with thermoplastic PVC insulation [5]. In one test, early failure of this cable type was observed. For this reason, this cable was chosen as reference cable for the parameter study to define thermal damage within the compartment.

The reference cable JE-Y(St)Y 16x2x0.8 has an overall diameter of 16 mm including a 2 mm thick PVC jacket to hold together and to physically protect the inner 32 conductors with a diameter of 0.8 mm each. The conductors themselves are electrically insulated by PVC [5].

If the electrical insulation fails, electrical faults such as hot shorts or shorts to ground may occur. Experiments have shown that a specific ambient threshold temperature exists, where no electrical fault occurs. Moreover, it was shown that the inverse time to failure is almost linear depending on the steady ambient temperature [4].

Since the electrical insulation of conductors is crucial for electrical faults, they might be predicted by a threshold temperature at the inner side of the cable jacket [6], in the following simply called cable temperature. In order to identify the failure temperature of the reference cable, two experimental studies were analyzed [4], [5]. The results show that the threshold cable temperature is approximately 240 ± 40 °C. If one cable in the parameter study reaches this temperature, a failure of an appliance is assumed and the compartment cannot be screened out in the Fire PSA.

DEFINITION OF "SIGNIFICANT FACTORS"

For the parameter study, so-called "significant factors" have to be analyzed, by which the compartment configuration can be characterized. To acquire a sensitive set of filter criteria, significant factors were identified that have a strong effect on the maximum cable temperature.

The Fire PSA is performed for compartments in safety related buildings of a NPP. Because most of these compartments do not have any passive vents such as windows and the mechanical ventilation is usually low, fires are commonly under-ventilated [7]. That is the reason for the maximum heat release rate (HRR), which is closely connected to the maximum cable temperature, being limited by the supply of oxygen. Under these conditions, oxygen is provided by two main sources, the compartment volume and the ventilation rate. If the fire grows fast, the HRR can reach higher values than a slow growing fire until the oxygen provided by the compartment is consumed. Because of the continuous supply of oxygen the inlet air flow also has a direct effect on the maximum HRR. The compartment height and size are not only important for oxygen supply, but also for the temperature in the hot gas layer (HGL). The higher a compartment, the more air is entrained in the fire plume cooling the HGL. In larger compartments, the oxygen being available is increased. However, the energy released is more thoroughly distributed and heat loss into the enclosing structures increases, resulting also in lower temperatures [8].

In summary, four factors, namely the fire growth rate, the ventilation rate, the compartment size and the compartment height, are significant factors within this study. The effect of the last two factors is not described in this paper, because preliminary studies showed that these factors are of less influence compared to the other ones. Several other parameters, which might affect the cable temperature, are not varied. For these factors, in principle conservative assumptions were made. For example, it is assumed that in case of fire heat sinks by built-in structures and components are more effective than heat sources such as electro-mechanical equipment installed in the compartment. Therefore, both factors are neglected. In addition to the designed mechanical ventilation, leakages are not assumed. This is due to the conservative assumption that the mechanical ventilation is not switched off in case of fire.

COMPARTMENT AND FIRE MODEL DESCRIPTION

In some real scale fire tests with mechanical ventilation [7], [9], [10] no hot gas layer was developed in the compartment. The gas concentration was well mixed over the whole compartment height and the temperature did not show significant decrease at a certain level but rather a constant linear decrease from the ceiling to the floor. This phenomenon was reported for different ventilation rates and was more likely to occur with inlet air ducts close to the ceiling than close to the floor. According to these reports, it is not clear if the traditional assumption of layer forming in compartment fires is generally true in case of mechanical ventilation.

To account for these experimental results, zone models were not suitable for the simulation task. That is why the computational fluid dynamics field model Fire Dynamics Simulator (FDS), Version 5.5.3 [11] was chosen for the parameter study despite the longer calculation times. FDS has been developed by the U.S. National Institute of Standards and Technology (NIST) and is widely used to analyze fire effects in nuclear power plants [12].

The compartment built-up in FDS is based on several compartments characterized in other fire safety analyses [13], [14], [15] as well as on some conservative assumptions. One assumption made is a square formed compartment floor in order to minimize conductive heat loss over walls. For the parameter study presented here, the compartment has a length of 14 m (196 m²) and a height of 5 m. However, both parameters will be changed in further simulations. Floor, ceiling and walls consist of concrete [16] with a thickness of 0.1 m. The backside of the walls is ambient air (20 °C) representing also the initial condition inside the compartment. Doors are assumed to be closed and are therefore disregarded in the compartment model. Both assumptions, the doors and the wall thickness, have been investigated in a sensitivity study showing minor effects on maximum cable temperatures as outlined below. For mechanical ventilation, there are two inlet air vents with an area of 0.36 m² each and diffusers realized as steel plates 0.2 m in front of the vents. The inlet air flow is constant with ambient temperature during the simulation. The two outlet vents are designed as open vents of 0.36 m² each. The fire source is constant with an area of 1.96 m² and a height of 1.2 m.

To simulate the growth phase of the fire, the t-square approach [8] is used. For combustion FDS applies a mixture fraction model. The user must prescribe the heat release rate (HRR) and FDS calculates the related mass loss rate (MLR) of fuel (propane) that is released into the compartment. The heat release is 13.1 MJ per kg oxygen, a value that is valid for typical organic fuels. For under-ventilated conditions, two extensions of the FDS combustion model come to bear. First, it provides a "two-step reaction" model [17]. In the first reaction step, fuel is burned to carbon monoxide and afterwards burned to carbon dioxide, given that enough oxygen is available. With the further extension, an extinction model is employed, where not the entire oxygen available is consumed because of the usage of a limited oxygen concentration which is linearly dependent on the gas temperature as shown in Figure 2. This concept is derived from the adiabatic flame temperature by Mowrer. It is very sensitive to the amount of heat released within the compartment. One important parameter of this model is the oxygen volume fraction at room temperature called "lower oxygen limit" or lower oxygen index (LOI), which is based on experimental results in small-scale experiments. With the extinction model, there is a steady phase where more fuel is released than burned, so the prescribed HRR as defined in FDS becomes irrelevant for the results of the parameter study. For the sake of simplicity, there is no effect of vitiated air on the MLR as reported in [9] and [18], representing also a conservative assumption [17].

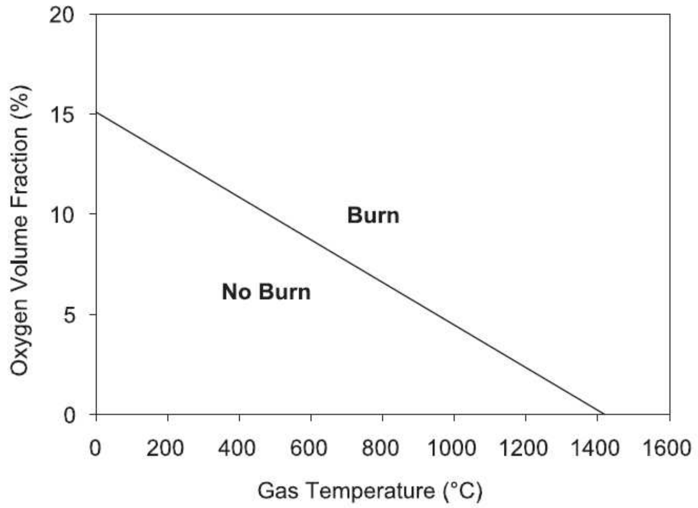

Figure 2 Suppression criterion: LOI plotted against gas temperature, from [17]

Except for the combustion model and the LOI, where sensitivity studies were performed, default settings were used for all other models and parameters in FDS. For the parameter studies, the grid size was 0.2 m according to the "characteristic fire diameter" described in [11].

In order to get worst case boundary conditions for the reference compartment and scenario, the location of the fire source (corner and center position) as well as the location of the inlet air vent (close to the floor, close to the ceiling) and the flow direction were varied. Two major effects were examined. First, with the fire in a corner of the compartment, less air is entrained into the plume which leads to higher HGL temperatures [8]. Second, a fire in the center of the compartment reaches higher maximum heat release rates, because more oxygen can be mixed with the fuel. One scenario combining these two conservative effects is that of the fire source (red surface) in a corner and the inlet air vents (yellow diffusers) at one wall of the fire source close to the ceiling as shown in Figure 3. Inlet air vents close to the bottom would lead to a fast descending hot gas layer which extinguishes the fire earlier than in a completely mixed environment. Although unusual for real NPP compartments, outlet vents were located close to the floor in order to minimize heat losses by drawn out gases. Compared to the other fire scenarios, the boundary conditions described lead to approximately 20 % higher maximum cable temperatures. Hence, it is assumed to be the worst case scenario.

As can be seen in Figure 3, there are three vertical cable trays (grey) arranged in 5 m, 7 m and 9 m distance to the far end of the fire source. The vertical arrangement was determined to be the worst case. Furthermore, simulations showed that the variation of distance along the wall is more sensitive to the maximum cable temperature as the variation in the direction of the center of the compartment. The cable trays are designed as continuous steel plates to represent the physical existence of the cables in the simulation. The front cable trays are arranged as such, so they do not shield the fire from the rear cable tray. At each position of the cable trays, the HGL temperature is determined. Furthermore, the air temperature trees are distributed on five positions over the whole compartment area and gas concentrations are measured at a center position. All measurements are located within a 10 m distance to the fire source. Therefore, the measurements do not have to be rearranged when changing the compartment size.

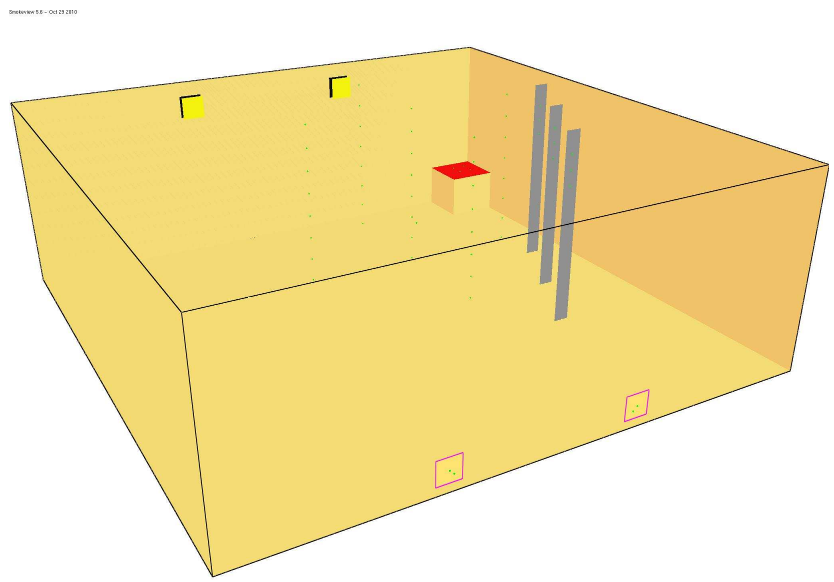

Figure 3 View of the compartment configuration in FDS

DESCRIPTION OF THE CABLE FAILURE MODEL

On each cable tray, at three different levels the JE-Y(St)Y 16 x 2 x 0.8 cable is simulated by the THIEF model (thermally induced electrical failure). It was developed at the Swedish National Testing and Research Institute and is implemented in FDS since Version 5.3. With this model, it is possible to predict electrical failures in cables based on the assumption, that a failure can be represented by a specific threshold temperature within a homogenous cylinder. There are five fundamental assumptions underlying the THIEF model [6]:

1. The heat penetration is in radial direction, the cable is assumed to be completely surrounded by the heat source.
2. The cable is homogenous in composition.
3. The thermal properties of the material are independent of temperature.
4. No decomposition reactions or melting occur within the cable during heating. Ignition and burning are not considered.
5. An electrical failure occurs, when the temperature inside the cable jacket reaches an experimentally determined value.

So, rather than predicting the electrical failure itself, the model is only able to predict a temperature within the cable according to the ambient conditions and the thermal properties of the cable. The thermal properties like thermal conductivity, specific heat and emissivity of the cable in the THIEF model are fixed values independent of cable type [6].

Necessary inputs to this model are the cable diameter, the jacket thickness, the mass per length of the cable and the experimentally determined cable failure temperature. To adjust the fixed thermal properties to the real cables of this study, the density can be determined by the mass per length and the cable diameter. To be able to compare model and real cable, the density of the cable in the model was adjusted to have the same thermal diffusivity as the real cable. The thermal diffusivity is according to Fourier, responsible for thermal penetration in non-steady cases [19].

QUANTIFYING MODEL UNCERTAINTIES

To quantify the uncertainties of FDS simulating confined compartment fires with mechanical ventilation in general and of the THIEF model in particular, a method described by Peacock et al. [20] was applied. This method uses functional analysis which defines operations on vectors allowing the quantitative comparison of two n-dimensional vectors. These n-dimensional vectors represent a series of n measurements in a certain time period, for example a temperature curve with n data points. With this method, time dependent curves can be compared by regarding the difference in the overall magnitude for the vectors of experiment and model as well as by comparison of the shapes of both vectors.

The difference in the overall magnitude of two vectors is calculated by the relative norm of the difference of the two n-dimensional vectors E and M, where E represents the time curve of a measurement in the experiment and M represents the time curve of the same measurement at the same time points in the model. This value is called the Euclidean Distance ε and is calculated with Eq. 1.

$$\varepsilon = \frac{\|E-M\|}{\|E\|} = \sqrt{\frac{\sum_{i=1}^{n}(E_i - M_i)^2}{\sum_{i=1}^{n}(E_i)^2}} \qquad (1)$$

If the Euclidean Distance is zero, both vectors are identical in magnitude. Because of the norms, this equation does not allow the differentiation between under- and over-estimation of a model. For this purpose, the curves have to be further examined. In the following analysis, "+" and "-" give a hint on over- and under-estimation of curves.

To compare the shapes of the two vectors E and M, the cosine of the inner product φ of both vectors is calculated with Eq. 2.

$$\varphi = \frac{\sum_{i=1}^{n} E_i \cdot M_i}{\sqrt{\sum_{i=1}^{n} E_i^2 \cdot \sum_{i=1}^{n} M_i^2}} \qquad (2)$$

When the cosine approaches unity, both vectors have nearly the same shape and both curves differ only by a constant multiplier. This method minimizes the effect of small-scale variations. However, it does not allow quantitative comparison between different measurements.

To compare the maximum values of the model and experiment Eq. 3 is used. For this, the both maximum values are compared as shown in Eq. 3 [10].

$$\delta = \frac{\max(M_i) - \max(E_i)}{\max(E_i)} \qquad (3)$$

For gas temperatures, the values of M and E are averaged over 10 s to reduce the effect of oscillations.

VALIDATION STUDY

To validate the suppression model of FDS, experiments with mechanical under-ventilated conditions and excess of unburned fuel in the fire gases would be needed. Since no suitable experiments could be found, it was not possible to validate the suppression model in FDS. To take this into account, the LOI is also varied in the parameter study because of its significance on the cable temperature. Therefore it can be taken as an additional significant factor for possible adjustments of the results. Nonetheless, to determine whether FDS and THIEF are qualified to simulate cable temperatures in under-ventilated conditions with a prescribed HRR, four real scale fire tests, below named as experiments, were compared to simulations.

Description of Experiments

Three of these experiments were chosen from the PRISME (French acronym for "Fire Propagation in Elementary Multi-room Scenarios") program carried out at the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) Fire Test Laboratory in Cadarache (France). The program, organized in an international OECD Nuclear Energy Agency (NEA) framework, was performed to study the propagation of smoke and hot gases between full-scale, well-confined and mechanically ventilated compartments. In a special facility called DIVA, three adjacent rooms with a size of 5 m x 6 m x 4 m (120 m³) each, additionally connected by one parallel hallway, as well as one upper room were built with 0.3 m thick concrete walls. Inlet and exhaust branches of the mechanical ventilation network are located near the ceiling. The pool fire, a circular pan filled with hydrogenated tetra-propylene (an isomer of dodecane) was installed in the center of one room [10]. Below, the experiments used for validating the model are named as experiments (Exp.) 1 to 3.

The fourth experiment is the so-called NRC (Nuclear Regulatory Commission) test, which is test no. 4 of a series of 15 fire tests conducted by the U.S. NRC and the NIST as part of the International Collaborative Fire Model Program (ICFMP) Benchmarking and Validation Exercise No. 3 to provide data for comparison of fire models with experiments. The compartment was 7.04 m x 21.8 m x 3.82 m in dimension with a total volume of 582 m³. The walls and ceiling were covered with marinite boards, while the floor was covered with gypsum. On both of the long sides of the compartment one air supply vent and one exhaust vent are installed and provide the room with approximately five air changes per hour. The fire was located in the center of the compartment in a 2 m x 1 m fuel pan. The MLR of heptane is controlled by the supply with a spray nozzle directly onto the pan. During the steady phase, the heat release rate is from 1050 to 1200 kW depending on the calculation method by MLR or energy balance. The fire was extinguished by ramping down the supply of fuel after 13:35 min in total. On several cable trays with different locations five control cables and one power cable all with XPE-Insulation were installed. Thermocouples had been placed on the surface and under the jacket of the cables on different locations. In NRC 4, nine measurements were compared to the THIEF model [21].

In the PRISME experiments, the gas volume fractions of O_2, CO_2 and CO were measured on a high and a low position (indices H and L) and one close to the fire. In the NRC No. 4 experiment, O_2 was measured on two levels, CO_2 only in the upper part and CO was not analyzed. Furthermore, several thermocouple trees had been installed in every experiment. Overall, the measurements close to the fire and in the plume region are not considered in the following comparisons because this is not the region of interest in this parameter study.

The uncertainties of measurements in the PRISME tests are 2 % for oxygen volume concentration and 10 % for gas temperatures [10]. In the NRC No. 4 experiment, an uncertainty of 6 °C for gas temperatures, which is around 5 %, was reported [21]. There are no uncertainties discussed for cable temperatures. In total 19 measurements were compared to the THIEF model. Random uncertainties as material properties or the location of the thermocouple are minimized. One systematic source of uncertainty in the experiment is the calculation of HRR. This is considered in two simulations (experiments No, 3 and NRC No. 4) with different HRR signed by the appendix (1) for the lower and (2) for the higher HRR. Furthermore, one unknown systematic uncertainty is the temperature measurement within the cables. Because the thermocouples have contact to a solid surface, this uncertainty is regarded as negligible compared to model uncertainties of THIEF and FDS. For this reason by comparing cable temperatures, the averaged difference between model and experiment is assumed to be the systematic model uncertainty.

All fire compartments of the experiments are modeled in FDS with minor changes regarding the fitting of locations to the grid size. The inlet air flow is modeled according to the experimental measurements. The exhaust ducts are realized with open vents as it is common in mechanical ventilated compartments. The fire was in all cases modeled as a propane pool fire with a HRR as calculated in the experiments. For all real cables, the material properties of the jacket were specified in the test reports, in the model the density was adjusted in order to get the same thermal diffusivity.

Results of the Validation Study

During the simulation of experiment no. 3 (2) the fire simulated by FDS extinguished after 550 s, although the fire in the experiment did not. Compared to the other simulations, the temperatures are over-estimated in this simulation and the oxygen concentration is under-estimated. This is considered as evidence that the HRR calculated via the MLR in the experiment is too high and therefore the failure is not within the model. To take this failure into account for validation, the model is only compared to the experimental data until the time of extinguishment.

In Table 1 the Euclidean Distances ε for different gas species at different levels (H: high, L: low) are shown. Including the differences of measurements, the oxygen concentration is slightly over-estimated in FDS. With average values of 25 and 38 % the carbon dioxide concentration is under-estimated; however, the uncertainties of measurements are unknown and seem to be high because of the low level of concentrations. The carbon monoxide concentration shows severe uncertainties and was therefore not measured in the model except for the Experiment 3 (2) simulation where the fire extinguished in the simulation but not in the experiment.

Table 1 Comparison of gas concentrations for validation

	$\varepsilon(O_{2,H})$ [%]	**$\varepsilon(O_{2,L})$ [%]**	**$\varepsilon(CO_{2,H})$ [%]**	**$\varepsilon(CO_{2,L})$ [%]**	**$\varepsilon(CO_H)$ [%]**	**$\varepsilon(CO_L)$ [%]**
Exp. 1	4 +	6 +	20 -	29 -	---	---
Exp. 2	10 +	12 +	36 -	40 -	100 -	100 -
Exp. 3 (1)	13 +	7 +	35 -	40 -	100 -	100 -
Exp. 3 (2)	6 -	5 -	7 +	44 +	95 -	100 -
NRC 4 (1)	8 +	7 +	32 -	---	---	---
NRC 4 (2)	4 +	5 +	21 -	---	---	---
Average	8	7	25	38	98	100

In summary, the slight over-estimation of the oxygen concentration is conservative, because in the model, more oxygen is available for burning than in reality despite the same HRR. The carbon dioxide concentration shows under-estimation, unless experimental uncertainties cannot be excluded. The Euclidean cosine for all curves is nearly unity, in other words the shapes are comparable between result and experiment. In general, the models to simulate the concentrations of oxygen and carbon dioxide seem to fit for mechanical ventilated fires within a certain uncertainty.

As shown in Table 2 the Euclidean Distance for gas temperatures has an average uncertainty of 19 % for the upper layer. Regarding the Euclidean cosine which is mostly above 0.98 for all gas temperature curves, it is shown that FDS 5.5.3 is reasonably suited to simulate mechanical ventilated fires.

Table 2 Comparison of gas temperatures for validation

	$\varepsilon(T_H)$ [%]	$\varepsilon(T_L)$ [%]	$\delta(T_H)$ [%]	$\delta(T_L)$ [%]
Exp. 1	7 -		- 3	+ 22
Exp. 2	27	24 -	- 28	- 17
Exp. 3 (1)	22 +	20 -	- 3	+ 10
Exp. 3 (2)	14 +	26 +	+ 8	+ 25
NRC 4 (1)	24 -	25 -	- 24	- 27
NRC 4 (2)	20 -	20 -	- 18	- 21
Average	19	20	11	1

The maximum gas temperatures T_H of the upper part of the compartment are approximately 10 % under-estimated, however three simulations showed under-estimation of around 20 %. Regarding the uncertainty of 6 % to 11 % concerning the uncertainties in the experimental HRR as well as the experimental uncertainty of 10 % in the measurement of gas temperatures, an assumed under-estimation of 10 % of the maximum gas temperatures in the upper part of the compartment seems to be a conservative assumption. As there is no clear trend in the lower part of the compartments, 10 % under-estimation seems to be a suitable assumption for T_L as well.

The differences of the cable temperatures are shown in Table 3. In particular, two measurement curves are discussed below to describe the difficulties and capabilities of the THIEF model.

Table 3 Comparison of cable temperatures for validation

	$\varepsilon(T)$ [%]	$\delta(T)$ [%]
Exp. 2	26 -	- 13
Exp. 3 (1)	36 -	- 34
Exp. 3 (2)	17 -	- 14
NRC 4 (1)	25 -	- 22
NRC 4 (2)	20 -	- 12
Average	25	- 19

The reference cable JE-Y(St)Y 16 x 2 x 0.8 was analyzed in experiment no. 2. For this cable, the maximum temperature is under-estimated with 8 %. However, the Euclidean distance is 33 % and the Euclidean cosine is 0.95, which might be due to different locations of the thermocouples within the cable. This is one of the experimental uncertainties mentioned above. The simulated "thermocouple" in the THIEF model seems to be closer to the hot ambient temperature than the real thermocouple. This results in the THIEF model predicting a faster temperature increase until it nearly reaches ambient gas temperature. At time of the maximum HRR, both cables reach nearly the same temperature, and therefore nearly the same maximum temperature despite different curve shapes.

On the other hand, there is the cable temperature curve of the thermocouple B-Tc-15 of the NRC 4 experiment. This thermocouple is located inside an XPE-jacket of a control cable. The experimental measurement and the model prediction both are shown in Figure 4, the Euclidean cosine is unity. As outlined above, it is possible to divide the model prediction by a value of (1- ε) to get a corrected temperature curve of the model. The comparison demonstrates that the model and experimental curve fit very well.

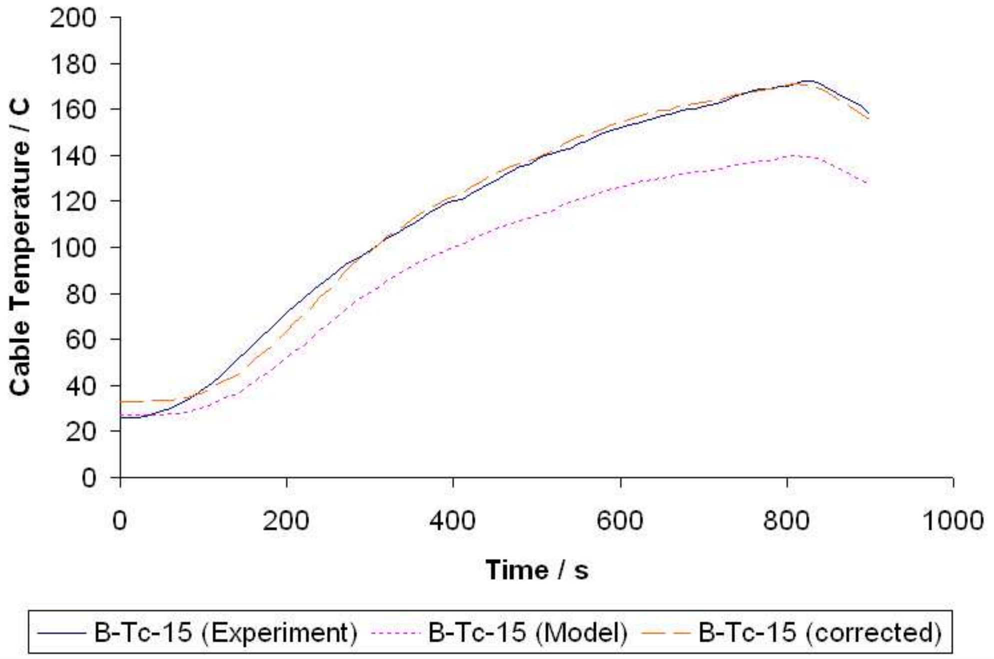

Figure 4 Experimental and modeled cable temperature curve (NRC 4)

In general, as shown in Table 3, the average Euclidean distance is 25 % for all 19 cable temperatures and the Euclidean cosine is mostly above 98 %. In the experiment no. 3, there is an influence of the HRR of 19 % and in the NRC test no. 4 it is 5 %. After all, there is an under-estimation of the THIEF model; however THIEF is capable to predict cable temperatures within a certain uncertainty.

As discussed above, there are two major uncertainties in the results. The uncertainty of the HRR varies from 10 % (NRC 4) to 20 % (experiment no. 3) for maximum cable temperatures. The other uncertainty is the systematic model uncertainty of FDS and THIEF, which is 19 % under-estimation for maximum temperatures. Considering both uncertainties, a value of around 20 % under-estimation (δ = - 0.2) is regarded to be conservative for the following parameter study.

SENSITIVITY ANALYSIS

Another source of uncertainty in the parameter study is caused by model parameters as well as by geometrical assumptions. To receive information about the correlation between the parameters and the results, a sensitivity analysis was carried out. For this sensitivity analysis, the point of interest is the maximum cable temperature averaged over four different assumed cables in seven meter distance to the fire and one meter under the ceiling, in the following named as result. The differences to the default values are calculated by Eq. 3.

One of the most important parameter in field models is the grid size. The default grid was chosen to be 0.2 m in all directions. To find out the effect of grid size, one simulation was performed with 0.1 m and another one with 0.4 m. The results are shown in Table 4.

Table 4 Results of the grid size analysis

Grid Size	δ(T) [%]
0.1 m	- 3
0.2 m	0
0.4 m	- 3

It can be observed that the results do not lead to a definite conclusion because the grid size has no monotonic effect on the maximum cable temperature. So far, the smaller and the coarser grid size show slightly lower maximum temperatures, so the default grid size is assumed to be conservative.

In addition, some other parameters affecting the energy transport in the simulation were varied. Namely, these are the Smagorinsky constant which is responsible for energy dissipation in the "Large Eddie" simulation model, the radiative fraction which is an important factor for the radiative energy source and also the soot yield which has influence on radiation transport. The different values used in the simulations are shown in Table 5.

Table 5 Variation of different model parameters

Parameter	Min	Default	Max
Smagorinsky constant	0.180	0.20	0.220
Radiative fraction	0.300	0.35	0.400
Soot yield	0.005	0.01	0.015

In all cases, the influence of each parameter was between δ = - 0.03 and δ = 0.01. All parameters showed monotonic behavior, the smallest effect had the soot yield. Furthermore, in one simulation the suppression model described above was deactivated. That led to higher heat release rates and to higher maximum cable temperatures with δ = 0.27. This result provides information about the upper limit of cable temperatures if the LOI would be set to zero.

Finally, the effects of some geometric assumptions were analyzed. One assumption is that all doors in the compartment are closed. Therefore, the effect of the connection to open space via two open doors and one small floor was investigated. The results showed significant higher HRR but the maximum cable temperatures changed with less than δ = 0.01. This might be explained by the additional convective heat loss through the open doors and by the entrainment of cold air into the compartment. In another simulation, the thickness of the walls was changed from 0.1 m to 0.2 m which resulted only in a decrease of δ = - 0.01.

Additionally, the area and the height of the fire were varied as shown in Table 6 and Table 7. As can be seen, the effect of the fire area is monotonic and more sensitive than the height of the fire.

Table 6 Results of the fire height analysis

H_{Fire}	$\delta(T)$ [%]
0.6 m	- 2
1.2 m	0
1.8 m	- 1

Table 7 Results of the fire area analysis

A_{Fire}	$\delta(T)$ [%]
1.00 m^2	- 6
1.96 m^2	0
4.00 m^2	+ 1

Summarized, the effects of the parameters assumed in the model as well as in the geometry on the maximum cable temperature are smaller than the uncertainty discovered in the validation study. With these results, the uncertainty of the model on maximum cable temperatures is maintained with δ = - 0.2. With this uncertainty and the experimental results of the cable threshold temperature, the threshold temperature to define a failure within the simulation is T_{FAIL} = 240 (1 - 0.2) = 192 °C.

RESULTS OF THE PARAMETER STUDY

After defining the damage, the significant factors as well as assessing the uncertainties of the model used, the parameter study was conducted to define a set of filter criteria which includes compartment configurations. This paper presents the results of three significant factors, namely the fire growth rate, the ventilation rate, and the LOI of the suppression criterion. A reference compartment with the floor area A_{comp} = 196 m² and the height H_{comp} = 5 m was taken. Therefore, the fire growth rate α was changed between α = 0.00293 kW/s² and α = 0.01172 kW/s² which are values for slow and moderate fire growth. The inlet air flow was varied from 0 m³/s to 4 m³/s which corresponds to air change rates up to 14.7 1/h. The default value of the LOI in FDS is 0.15 mol/mol. At the end of this section, it will be varied from 0.12 mol/mol to 0.18 mol/mol to get an impression of the uncertainties in the results.

Effects on the Maximum Heat Release Rate

To get an impression of the maximum possible fire size within the compartments, the maximum HRR is plotted against the inlet air flow for three different growth rates in Figure 5. The strong dependency of the maximum HRR to the fire growth rate at low inlet air flows can be explained by the initial amount of oxygen which is available in the compartment. The faster the fire growth, the higher the HRR can become until the entire oxygen within the compartment is consumed, except for the rest oxygen concentration in the vitiated air. With higher ventilation rates such as 2 m³/s (7.3 1/h air changes), the difference between the fire growth rates diminishes because initial oxygen provided by the compartment volume becomes less important.

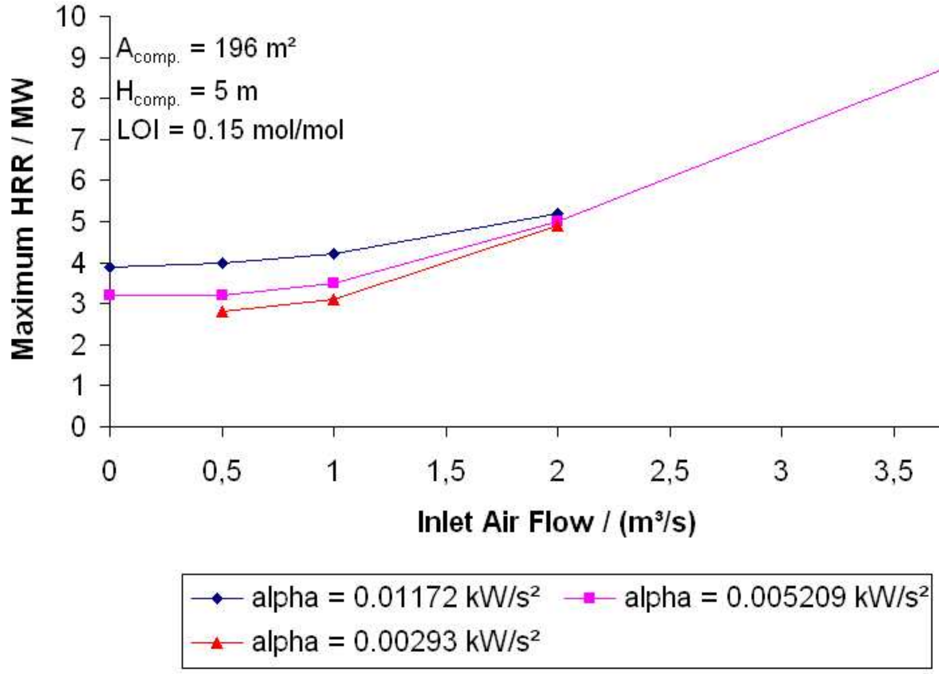

Figure 5 Effects on the maximum heat release rate (HRR)

Effects on the Hot Gas Layer Temperature

The maximum HRR has a direct effect on the maximum temperature of the HGL. However, the inlet air flow which is responsible for the higher HRR also cools the gases and thereby reduces the effects at higher air changes as can be seen in Figure 6 for a HGL temperature measured in a distance d = 7 m of the fire origin.

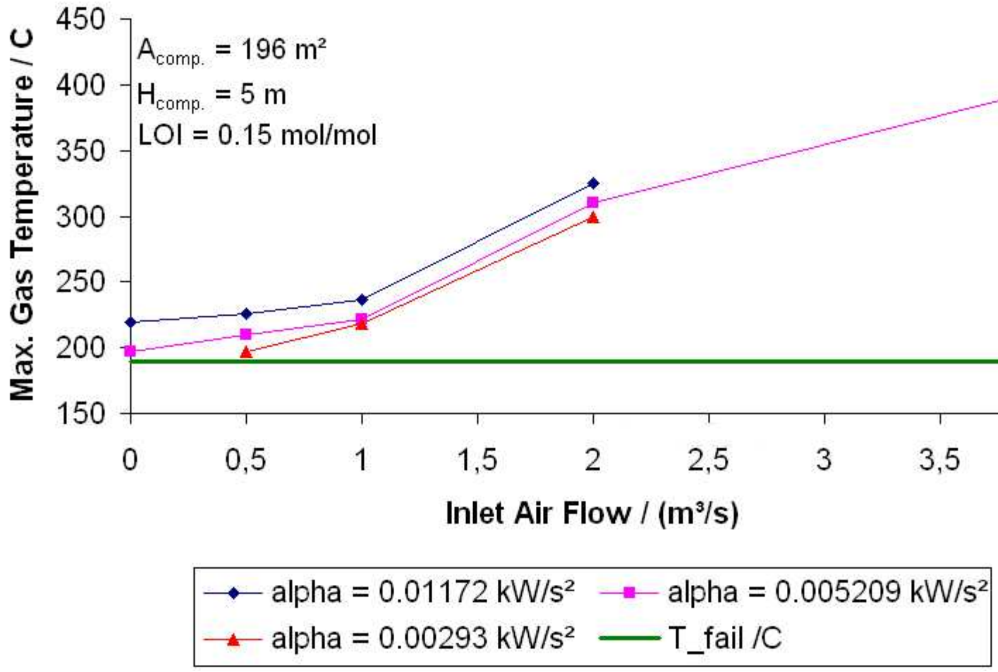

Figure 6 Effects on the hot gas layer (d = 7 m)

With the maximum gas temperature in the HGL and a threshold temperature of 200 °C, a simple first order filter criteria can be defined. With an uncertainty of δ = - 0.1, the threshold temperature in the simulations is T_{fail} = 190 °C at which failure occurs. It can be seen, that in every simulation the failure criteria are true and each compartment would have to be analyzed in Fire PSA. Only compartments with no ventilation and a slow fire growth rate seem to be safe. Since the criteria using the HGL as threshold temperature does not imply the thermal mass of appliances that must be heated until a failure occurs, these criteria are considered to be more conservative but also less realistic as the criteria using the sophisticated THIEF model and the maximum cable temperature as damage criterion.

Effects on the Cable Temperatures and Definition of the Set of Filter Criteria

The maximum cable temperatures of the reference cable calculated by the THIEF model in a distance of 7 m to the fire origin and a height of 4 m are plotted against the inlet air flow for three different fire growth rates in Figure 7.

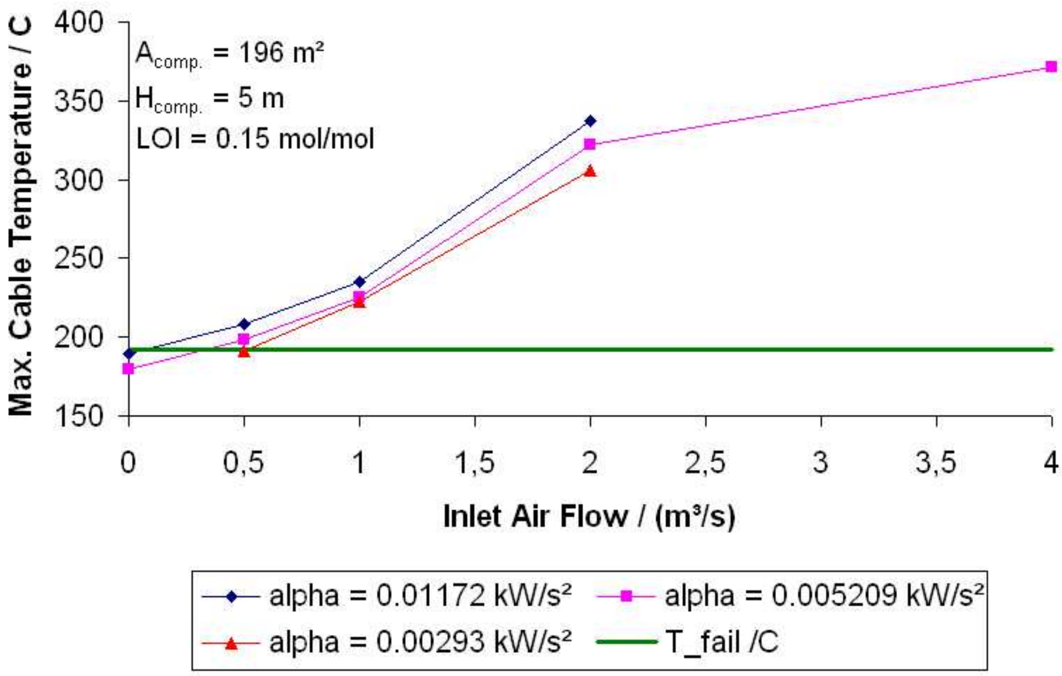

Figure 7 Effects on the reference cable (d = 7 m, h = 4 m)

With a threshold temperature of T_{fail} = 192 °C to failure, it can be seen that in compartments with no ventilation cables will not fail. Additionally, in compartments with low air changes (0.5 m³/s or 1.8 1/h) and a slow fire growth rate (α = 0.00293 kW/s²) no failure occurs, either. In general, the maximum cable temperature is more sensitive to the ventilation rate than to the fire growth rate.

The energy of approximately 1000 MJ to 1300 MJ was released in the compartment until failure of cables occurred. It mainly depends on the fire growth rate because failure occurs mostly in the growing phase of the fire. In this stage the inlet air flow has a minor influence on cable temperature. The amount of energy corresponds to a fire load density of only 5.1 MJ/m² and 6.6 MJ/m². If less energy is available in the compartment, a fire is not able to cause damage to other appliances because it will extinguish before. The critical fire load density is shown to be much lower than the value of 90 MJ/m² given in [1] as “fire-load criterion”.

The dependency of the height under the ceiling and the distance to the fire origin was also analyzed. In Figure 8 the maximum cable temperatures at a height of 3.6 m and a horizontal distance of 7 m to the fire in different compartment configurations are shown.

The temperatures are significantly lower than at a level of 4.0 m. Therewith, compartments with air change rates smaller than 2.8 1/h (0.75 m³/s) and the fire growth rate slower than moderate would be screened out. If cables are on a level of 4.4 m, the temperatures are around 13 % higher than at a level of 4.0 m and in every compartment configuration, the cable would fail. If the horizontal distance of the reference cable is increased to 9 m and the height remains at 4 m, the maximum cable temperatures decrease by 4 %. Additionally, a compartment with an inlet air flow of 0.5 m³/s and a fire growth rate smaller than α = 0.005209 kW/s² would be safe. Generally, the horizontal distance to the fire has minor influence compared to the height of the cables.

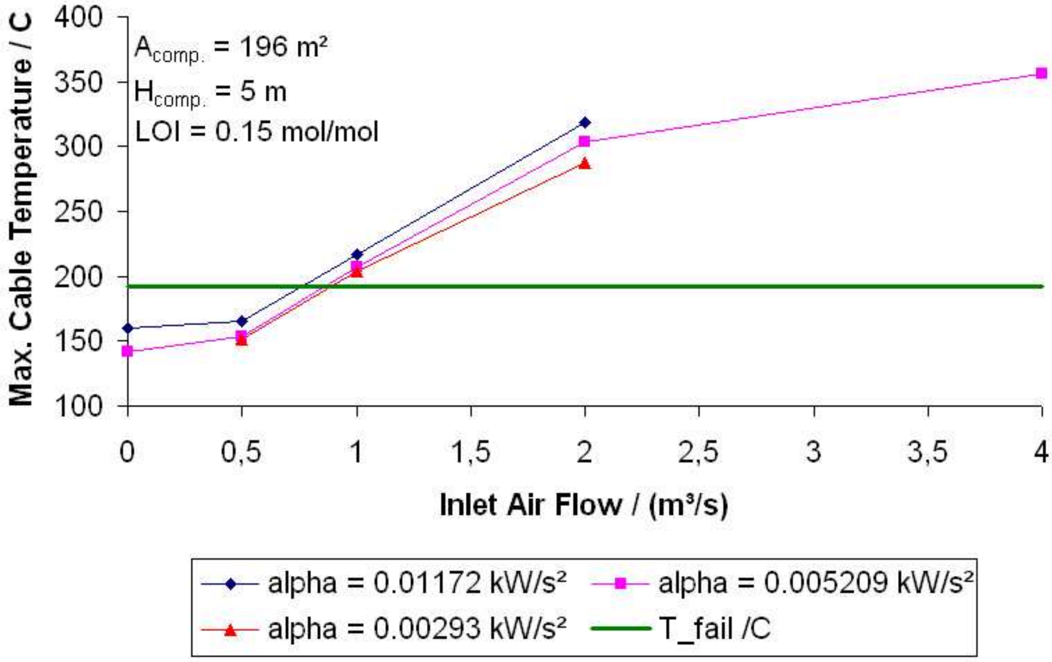

Figure 8 Effects on the reference cable (d = 7 m, h = 3.6 m)

The effect of the LOI is shown in Figure 9 where the maximum cable temperatures are plotted against the LOI for different ventilation rates. As can be seen, the relationship is mostly linear in a broad band of values. This simplifies the implementation of new information on the LOI to the results of this parameter study.

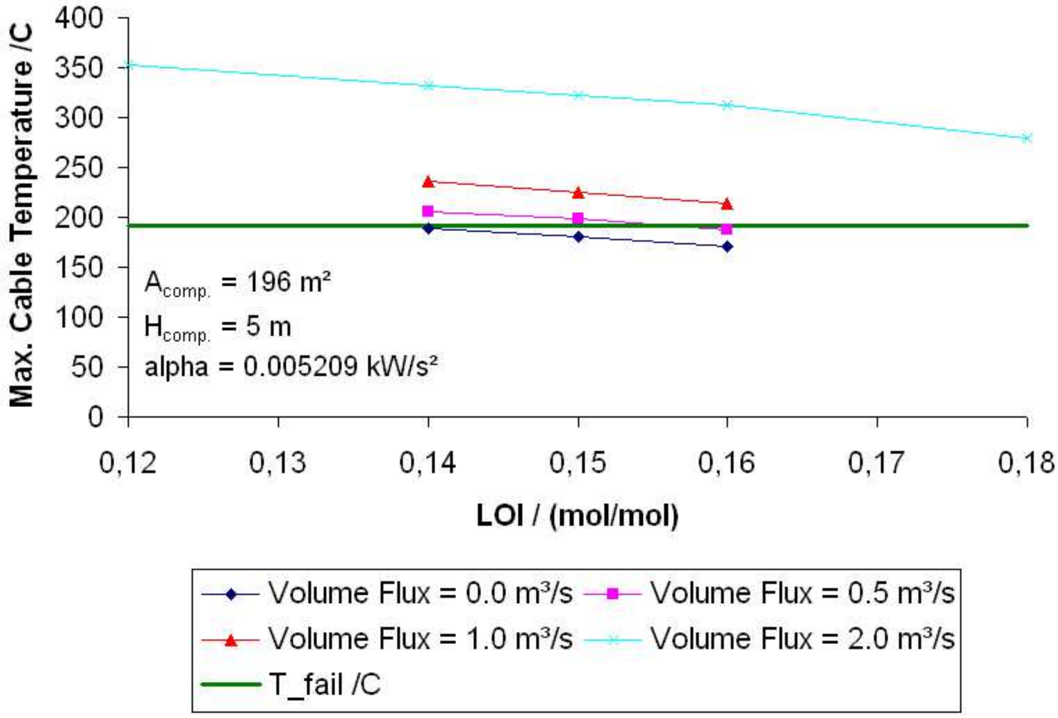

Figure 9 Dependency of the cable temperature on the LOI (d = 7 m, h = 4 m)

CONCLUSION AND OUTLOOK

A new set of filter criteria for Fire PSA has been developed to screen out those compartments where a fire of one component will not damage cables as typical example for vulnerable appliances of the safety systems in the hot gas layer (HGL).

Generally, an electrical fault can be determined by a threshold temperature at the inner side of the cable jacket, for the reference cable this value is 240 °C. The maximum cable temperature is mainly sensitive to the fire growth rate, the inlet air flow, the compartment size, and the compartment height in well-confined mechanically ventilated compartments. For this reason, these factors were defined as significant factors and characterize the compartment configuration in the parameter study. The field model Fire Dynamics Simulator (FDS) Version 5.5.3 was chosen to analyze the effect of different compartment configurations of the significant factors on the maximum cable temperature. The uncertainty of the maximum cable temperature in the model is approximately 20 %. A sensitivity analysis regarding model parameters and geometrical assumptions revealed only negligible uncertainties. The filter criteria showed to be a function of the ventilation rate, the fire growth rate, and the compartment size.

In the parameter study, fires in compartments with a size of 196 m² and a height of 5 m showed no damage to the reference cable at a vertical level of 4 m and a horizontal distance of 7 m to the fire origin, if the ventilation has been switched off and the fire growth rate is less than moderate. Compartments with an air exchange rate of 1.8 1/h and a slow fire growth rate were also safe in this context. Cables at a vertical level of 3.6 m did not fail in case of fires in compartments with air exchange rates lower than 2.8 1/h and slow to moderate fire growth rate. In general, the ventilation rate is a more significant factor than the fire growth rate. This is advantageous, because for a given compartment the knowledge on the ventilation rate is much better than the knowledge on possible fire growth rates.

Furthermore, the energy totally released was calculated until the reference cable fails, but it is too low to define a suitable additional filter criterion.

Further studies will examine the effects of compartment size and height on the set of filter criteria. Preliminary studies have demonstrated that larger rooms and higher ceilings show significantly lower maximum cable temperatures. Therefore, these factors will be taken into consideration as filter criteria. Finally, the data will be transferred into one empirical equation including all four significant factors as well as the lower oxygen index (LOI). The result of this equation is the maximum cable temperature which can be compared to the threshold temperature. For different compartment configurations it can be determined whether the maximum cable temperature is higher than the threshold temperature and whether appliances in the HGL will fail in case of fire or not. With this approach, the two most important uncertainties can be easily diminished as soon as further studies have been carried out.

One of these uncertainties is the modeling of mechanical ventilated compartment fires, particularly the heat release rate in under-ventilated conditions. One important factor is the LOI in FDS, which was not possible to validate. Therefore, it was varied within the parameter study and the results can be adjusted as soon as more precise analysis has been done. At the time being, the heat release rate is considered to be conservative because the effect of vitiated air on the mass loss rate is not modeled. The second important uncertainty is not a model but an experimental uncertainty. The failure temperature of the reference cable was determined by the use of two different experimental references. Anyhow, the failure threshold temperature cannot be determined exactly. Further experiments could ascertain this value more accurately.

With the final results, a new set of filter criteria for Fire PSA has been developed to screen out compartments in which a damage of other components in case of fire is not possible. However, the analysis of damage is limited to thermal damage in the HGL of a compartment fire. Other zones of influence such as the flame, radiation region, plume or ceiling jet are not considered. Non-thermal damage, which could impair electronic devices, is not covered.

In this set of filter criteria different compartment configurations, quantified by four significant factors plus the additional LOI are being applied to predict the damage of a fire in the compartment in one single empirical equation. Therefore, in comparison to the "fire load criterion" with only one parameter as input, the new filter criteria are still easy to apply and represent fire dynamics in NPP compartments much more accurately.

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ENHANCEMENTS IN THE OECD FIRE DATABASE - FIRE FREQUENCIES AND SEVERITY OF EVENTS -

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ABSTRACT

The international fire event database OECD FIRE for collection of event data from fires in nuclear power plants in twelve OECD NEA (Nuclear Energy Agency) member states contains 392 fire event records. While in the first two Project phases the focus of the Project was on building up the database this has meanwhile changed to more analytical work for safety assessment. This also resulted in actual enhancements of the Database.

An in-depth investigation of the records has suggested to make the list of "components where the fire started" more consistent with the operating experience on fires in nuclear plants collected in the Database, particularly for a probabilistic risk analyses. For this purpose, more consistency with already existing approaches, such as the initial component list of fires starting given in NUREG/CR-6850, has been provided resulting in an extended and more precise list of components.

Together with this change the coding of the consequences of the observed fires has been improved by providing multiple choices of potential consequences which could occur such as "fire confined to one room", "total loss of one room", "adjacent rooms affected", "other fire compartments affected", etc. Generic conditional probabilities of more severe fire consequences, given a fire, are presented and provide valuable insights into Fire PSA.

INTRODUCTION

The international fire event database OECD FIRE for collecting event data on fires in commercial nuclear power plants from twelve OECD NEA (Nuclear Energy Agency) member states in its current version [1] contains 392 fire event records from the 1980's to the end of 2010. While in the first two Project phases the focus of the Project was on building up the database this has meanwhile changed to more analytical work for safety assessment. This also resulted in actual enhancements of the FIRE Database, in particular with respect to its capability for providing insights for damage beyond the initiating component based on the data. It should be noted that, in some cases, judgments of the extent of damage are made from the report. Although the database is still not very big, first applications have demonstrated that it can be used to improve both deterministic safety assessment and probabilistic risk analysis with respect to analyzing fire events.

The Database still represents inhomogeneous population, as the criteria for reporting fire events vary among the participating NEA member countries. Many countries participating in this program only supply fires reported at the LER (licensee event report) threshold, while a few countries also provide smaller fires to the database. For statistical analysis and application in the frame of PSA a suitably homogeneous event population is needed. Therefore, for statistical use either incipient (pilot) fires reported only from some countries have to be excluded or event populations only from countries reporting all events regardless of any reporting threshold can be statistically analyzed.

Derivation and quantification of plant specific event trees as a tool for calculating conditional probabilities of fire induced component damages is an item of Fire PSA. One goal of the FIRE Project therefore is to show how data from the OECD FIRE Database may be used to provide information on such probabilities.

The recent work with the FIRE Database, particularly with respect to the topic of the first Topical Report to be provided, gave indications that fire events resulting from high energy arcing faults (HEAF) represent a non-negligible amount of fire events with the potential to impair nuclear safety. HEAF are energetic or explosive electrical equipment faults characterized by a rapid release of energy in the form of heat, light, vaporized metal and pressure increase due to high current arcs between energized electrical conductors or between energized electrical components and neutral or ground. HEAF events may also result in projectiles being ejected from the electrical component or cabinet of origin and result in fire.

ENHANCING THE DATABASE CODING WITH RESPECT TO COMPARTMENTS AND COMPONENTS WHERE THE FIRE ORIGINATED

Estimation of Compartment Specific Fire Frequencies for Boiling Water Reactors

From the existing Database it is already possible to calculate compartment specific fire occurrence frequencies for full power operational states plants for nuclear power plants with boiling water reactor (BWR) from those member countries reporting all events (without any restriction on the reporting level).

Table 1 shows the average number of compartments of typically coded compartment types in buildings to be analyzed in the frame of Fire PSA as provided by the countries reporting all events having occurred at BWR type plants.

Table 1 Average number of compartments in the buildings of BWR plants from member countries reporting all events

Compartments / Buildings	Process Rooms	Switchgear Rooms	Rooms for Electrical Control Equipment	Total
Turbine building	70	8	16	**94**
Diesel generator building	11	12	4	**27**
Auxiliary building	30	16	4	**50**
Reactor building	81	18	21	**120**
Electrical building	10	18	11	**39**
Total	**202**	**72**	**56**	**330**

Table 2 shows the numbers of fire events having occurred in selected relevant buildings and compartments (and associated frequencies per reactor year of operation) in plants with BWR type reactors during power operation as reported to the Database until end of 2010. These frequencies are based on approximately 250 reactor years of power operation for 13 reactors.

Table **3** shows these numbers for events during low power and shutdown plant operational states making up approximately 33 years of operation of the 13 BWR plants. It should be noted that low power fire events do not comprise the full power frequencies, and vice versa.[1]

The underlying populations are reasonably homogeneous. It has to be noted in this context that the statistics in Table 2 and

Table **3** are not exhaustive, because there are further events reported to the OECD FIRE Database which have occurred in buildings and compartments not listed in Table 1.

Table 2 Fire events reported in the OECD FIRE Database [1] for full power operational states for selected buildings/compartments and corresponding compartment specific fire frequencies from BWR reactors in member countries reporting all events

Compartment Type / Building Type	Process Rooms	Switchgear Rooms	Rooms for Electrical Control Equipment (incl. MCR)	Total Number of Events (Frequency per Reactor Year)
Turbine building (frequency per room and ry)	21 (1.2 E-03/a)		2 (5.0 E-04/a)	**23 (9.2 E-02/a)**
Diesel generator building (frequency per room and ry)	*			
Auxiliary building (frequency per room and ry)	4 (5.3 E-04/a)		1 (1.0 E-03/a)	**5 (2.0 E-02/a)**
Reactor building (frequency per room and ry)	2 (9.8 E-05/a)		1 (1.9 E-04/a)	**3 (1.2 E-02/a)**
Electrical building (frequency per room and ry)	1 (4.0 E-04/a)	2 (4.4 E-04/a)	2 (7.3 E-04/a)	**5 (2.0 E-02/a)**
Total amount per room type (frequency per ry)	**28 (1.12 E-01/a)**	**2 (8.0 E-03/a)**	**6 (2.4 E-02/a)**	**36 (1.44 E-01/a)**

* There have been diesel generator fires, but not in those compartment types shown in this table.

The dominant contribution results from process rooms (rooms containing pumps, valves, mostly mechanical equipment; whether or not this equipment is part of a safety system does not affect the classification as "process room") and among them from process rooms in the turbine building. In three events in the turbine building, rooms adjacent to the compartment where the fire originated were directly affected by the fire through the influence of heat and/or hot gases.

In two of these three events, the fire originated in the turbine building at the turbine generator; in the third event it occurred at a fan in a room for ventilation in the waste disposal building (not shown in the tables because the average numbers of rooms are not available).

In one event in the electrical building, rooms adjacent to that where the fire started and other fire compartments were affected by consequential functional effects on components. The fire

[1] NUREG/CR-6850 in many cases includes low power events into its at-power frequencies when the fire is not specific to low power conditions. This decision is made on a frequency bin basis.

originated at a rectifier belonging to a battery train in the electrical building (not shown in Table 1 and Table 2 because average numbers of rooms are not available).

Table 3 Fire events reported in the OECD FIRE Database [1] for low power and shutdown states for selected buildings/compartments and corresponding compartment specific fire frequencies from BWR reactors in member countries reporting all events

Compartment Type / **Building Type**	**Process Rooms**	**Switchgear Rooms**	**Rooms for Electrical Control Equipment (incl. MCR)**	**Total Number of Events (Frequency per Reactor Year)**
Turbine building (frequency per room and ry)	17 (7.3 E-03/a)		1 (1.9 E-03/a)	**18 (5.5 E-01/a)**
Diesel generator building (frequency per room and ry)	1 (2.7 E-03/a)			**1 (3.0 E-01/a)**
Auxiliary building (frequency per room and ry)	2 (2.0 E-03/a)			**2 (6.0 E-02/a)**
Reactor building (frequency per room and ry)	2 (7.5 E-04/a)			**2 (6.0 E-02/a)**
Electrical building (frequency per room and ry)		1 (1.7 E-03/a)	2 (5.5 E-03/a)	**3 (9.1 E-02/a)**
Total amount per compartment type (frequency per ry)	**22 (6.7 E-01/a)**	**1 (3.0 E-02/a)**	**3 (9.1 E-02/a)**	**26 (7.9 E-01/a)**

As for full power plant operational states, the dominating contribution for low power and shutdown states results from process rooms in the turbine building. Events with more severe consequences as discussed in the paragraph on "Improvements with respect to fire severity" have not been identified in the event reports.

The collection of average room numbers is currently under way for nuclear power plants with pressurized water reactor (PWR) in countries reporting all events (without any restriction on the reporting level). Following the completion of the collection, results analogous to those reported here will become available.

Enhancement of the List of Components

The list of components considered in the data collection has been updated and improved in several ways. By the end of 2010 the original component list no longer matched the profile of the collected data in many cases. Some component types coded were not detailed enough to match the collected data, and other components were absent in the component list and therefore had to be coded, e.g. as "others". Another motivation to update the component list was the desire to make it similar to the component list used in NUREG 6850 [2] to facilitate comparisons and to be able to obtain component specific fire occurrence frequencies to be applied in the frame of Fire PSA. The list now in use in the OECD FIRE Database, including the component definitions, is shown in Table 4; the profile of the component counts is presented in Table 5.

Table 4 OECD FIRE Database - List of components where the fire started and definition for coding

Component Types	Component Definitions for Coding
Battery	Each bank of interconnected sets of batteries located in one place (often referred to as "Battery Room") should be counted as one battery set. Cells may not be counted individually.
Boiler	Boilers are generally well-defined items. All ancillary items associated with each boiler may be included as part of the boiler. Control panels that are installed separate from a boiler may be included in the "Electrical Cabinets" code.
Breaker	A breaker is an automatically operated electrical switch designed to protect an electrical circuit from damage caused by overload or short circuit. Its basic function is to detect a fault condition and, by interrupting continuity, to immediately discontinue electrical flow. Circuit breakers are made in varying sizes for different voltage levels.
(Segmented) Bus duct	This category applies to a bus duct where the bus bars are made up of multiple sections bolted together at regular intervals (transition points). Here, the bus bars are contained within open-ended sections of metal covers that are bolted together to form a continuous grounded enclosure running the full distance between termination points. Segmented bus ducts are able to accommodate tap connections to supply multiple equipment termination points. The key parameter for the use of this code is the location where fire is manifested. This code shall be used if the fault is manifested at any transition points along the bus duct length (i.e. bolted connections). Fires which occur at the termination points at the end device shall be treated in accordance with the end device.
Cable (fires caused by welding and cutting)	This code is applied for all exposed cables (i.e., cables that are not in conduits or wrapped by non-combustible materials) which are ignited by welding and cutting activities.
Cable run (self-ignited)	This code applies to all exposed cables (i.e., cables that are not in conduits or wrapped by non-combustible materials) which are self-ignited.
I&C cables	These cables typically include instrumentation and/or control (I&C) cables on a cable tray or other low voltage cables for low power equipment. Add current level (control cable) in milli Amps and voltage level in the narrative description field, if possible.
Power cables	This typically covers 6 kV, 500 V or 200 V power cable on a cable tray. Types of cables should be described in the narrative description field (IEEE/non-qualified, fire retardant, fire retardant cable coating).
Component (other than cable) ignited by hot work	

Component Types	Component Definitions for Coding
Compressor	This code covers the large air compressors that provide plant instrument air included in the Internal Events PRA Model. These compressors are generally well-defined devices. They may include an air receiver, air dryer, and control panel attached to the compressor. These items should be considered part of the air compressor. If portable compressors are part of the model, those compressors should also be included in the equipment count for this code. Note that compressors associated with the ventilation systems are not part of this code. Small air compressors used for specialized functions are also not part of this code.
Diesel generator	Diesel generators are generally well-defined items that include a set of auxiliary subsystems associated with each engine. All diesel generators that are included in the electric power recovery model should be counted here. In addition to the normal safety related diesel generators, this may include the technical support center diesel generators, security diesel generators, etc. It is recommended that each diesel generator and its subsystems be counted as one unit. The subsystems may include diesel generator air start compressors, air receiver, batteries and fuel storage, and delivery system. It is recommended that the electrical cabinets for engine and generator control that stand separate from the diesel generator be included as part of "Electrical Cabinets". Control panels that are attached to engine may be counted as part of the engine.
Dryer	Clothes dryers are generally well-defined units.
Electrical cabinet:	Electrical cabinets represent such items as switchgears, motor control centers, DC distribution panels, relay cabinets, control and switch panels (excluding panels that are part of machinery), fire protection panels, etc. Electrical cabinets in a nuclear power plant vary significantly in size, configuration, and voltage. Size variation range from small-wall mounted units to large walk-through vertical control cabinets, which can be 20' to 30' long. The configuration can vary based on the number of components that contribute to ignition, such as relays and circuit cards, and combustible loading, which also affects the fire frequency. Electrical cabinets shall be separated based on the classification of the fire (HEAF or non-HEAF and by voltage ranges).
High or medium voltage (non-HEAF, $\geq$ 1 kV)	This code shall be used for fires occurring in high or medium voltage electrical cabinets which do not produce a high energy arcing fault. Typically these are cabinets used for 6 kV breakers or 400 V motor breakers. Normally this type of cabinet is located in the switchgear room.
High or medium voltage (HEAF, $\geq$ 1 kV)	This code shall be used for fires occurring in high or medium voltage electrical cabinets which do produce a high energy arcing fault Typically these are cabinets used for 6 kV breakers or 400 V motor breakers. Normally this type of cabinet is located in the switchgear room.

Component Types	Component Definitions for Coding
Low voltage (non-HEAF, < 1 kV)	This code shall be used for fires occurring low voltage electrical cabinets which do not produce a high energy arcing fault Typically these are cabinets used for instrumentation and control, logic build-up, regulation, etc. The type of cabinet can be described in narrative description fields. Normally this type of cabinet is located in relay rooms.
Low voltage (HEAF, < 1 kV)	This code shall be used for fires occurring low voltage electrical cabinets which do produce a high energy arcing fault Typically these are cabinets used for instrumentation and control, logic build-up, regulation, etc. The type of cabinet can be described in narrative description fields. Normally this type of cabinet is located in relay rooms.
Electric motor (not in pump)	This code includes any electric motor with a rating greater than 5 hp. This code does not include electric motors that are attached to equipment already identified and counted in other codes (i.e. reactor coolant pumps, air compressors, dryers, pumps, RPS MG sets, and motors of ventilation subsystems equipment, such as fans or filters). That is, motors associated with a piece of equipment counted as a part of another ignition scores code are not counted separately as motors, but rather, are considered as an integral part of the larger equipment item (the pump., the compressor, etc.).
Equipment (fixed) for illumination	This code includes any fire which occurs as the direct result of fixed illumination equipment. This includes lighting ballast fires and fires caused by fixed lighting failures.
Fan	This code includes components such as air conditioning units, chillers, fan motors, air filters, dampers, etc. A fan motor and compressor housed in the same component are counted as one component. Do not count ventilation fans if the drive motor is 5 hp or less.
Filter	This code applies to all fires in filters for gases, which are mainly part of the ventilation sub-systems. But which van also be installed independently of the ventilation ducts, e.g. in the off-gas system.
Fixed heater	This code applies to all fires caused by fixed heaters installed throughout the plant for various reasons. This includes radiative heaters, convection heaters and fan heaters which are permanently installed in a specific plant area. Portable heaters are captured in a separate code.
Hydrogen containing vessel	Hydrogen storage tanks are generally well-defined items. Multi-tank hydrogen trailers, because they are interconnected, should be counted as one unit.
Iso-phase duct	This code applies to bus ducts where the bus bars for each phase are separately enclosed in their own protective housing. The use of the iso-phase buses is generally limited to the bus work connecting the main generator to the main transformer. The potential effects of the iso-phase faults appear to be

Component Types	Component Definitions for Coding
	unique in comparison to the end device fires (transformer or exciter). Care should be taken to evaluate the fire scenario before coding the event as an iso-phase duct fault or coding as a fire associated with the end device. That is, the fire should be evaluated in conjunction with the definition of a HEAF event and careful consideration as to the initiating component.
Junction box	An electrical junction box is a container for electrical connections, usually intended to conceal them from sight and deter tampering. A small metal or plastic junction box may form part of an electrical conduit wiring system in a building, or may be buried in the plaster of a wall, concealed behind an access panel or cast into concrete with only the lid showing. It sometimes includes terminals for joining wires.
Main control board	A control room typically consists of one or two (depending on the number of units) main control boards as the central element of the room. The main intent was to capture the main "horseshoe" and little else. This scope of this code is sharply limited to the main control boards associated with the direct operation of the plant in the control room. This category should be used for control room systems which meet the following requirements (1) Serve as an integral part of the main plant monitoring and control functions; (2) located in the center of the operators' main work area; and (3) manned on a nearly continuous basis. This code would not include smaller detached panels housing such equipment as computers and the event recording equipment and printers or "back panels" and other detached panels housing items such as balance-of-plant and off-site power controls and indicators. All of these panels should be excluded from the main control board code and treated as general electrical panels. The scope is also limited to the main control room. All auxiliary shut down panels and auxiliary control boards redundant to the main control boards shall be treated as electrical panels.
Main feedwater pump	Main feedwater pumps are generally well-defined entities. If there are ancillary components associated with each pump, it is recommended to include those items as part of the pump.
Miscellaneous hydrogen containing equipment (e.g. piping)	This code includes hydrogen fires in miscellaneous systems other than hydrogen cylinder storage, generator cooling, and battery rooms. Care should be taken to make sure this code is distinguished from the turbine generator hydrogen fires.
Off-gas/hydrogen re-combiner	This code includes all fires which occur in the off-gas systems and hydrogen re-combiner systems.
Oil separator or oil stripper	This code applies to all fires caused by equipment for separating oil (so-called oil strippers or oil separators) installed throughout the plant.
Portable equipment	This code typically applies to e.g., heaters, low power electric

Component Types			Component Definitions for Coding
			equipment, portable lights, etc. This code should take care to distinguish between portable heaters and fixed heaters as well as portable illumination equipment and fixed illumination equipment. The intent of this code is to capture fires which occur as the direct result of the interim use of portable equipment
Pumps			It is assumed that above a certain size, fire ignition is the same for all pumps. Pumps below 5 hp are assumed to have little or no significant contribution to risk. Do not count small sampling pumps. .
	Electrically driven or turbine driven		This code includes motors, pumps and support equipment for cooling, lubrication, etc. This code excludes pumps with a rating of 5 hp or less. Turbine driven pump, such as auxiliary feed water pump (BWR, some PWR)
	Reactor coolant pump (RCP, for PWR)		The reactor coolant pumps (RCPs) are distinct devices in PWRs that vary between two and four, depending on primary loop design.
	Main feedwater pump		Main feedwater pumps are generally well-defined items. All ancillary items associated with each pump should be included in this code.
Rectifier, inverter, or battery charger			These are generally well defined items associated with DC buses.
RPS motor generator sets			In PWRs, the RPS MG sets are well defined devices. The electrical cabinets associated with the MG sets are not included as part of these items.
Transformer:			Care should be taken to evaluate the fire scenario before coding the event as a transformer fire or coding as a fire associated with an iso-phase bus duct. That is, the fire should be evaluated in conjunction with the definition of a HEAF event and careful consideration as to the initiating component.
	High voltage (voltage $\geq$ 50 kV):		High-voltage power transformers typically installed in the yard belong to this code. They include plant output power transformers, auxiliary-shutdown transformers, and startup transformers, etc.
		Oil involved, catastrophic	The catastrophic failure of a large transformer is defined as an energetic failure of the transformer that includes a rupture of the transformer tank, oil spill and burning oil spattered at a distance from the transformer.
		Non-catastrophic	Similar to the "catastrophic" code, this code includes the high-voltage power transformers typically installed in the yard. This code shall be used for fires which do not involve a rupture of the transformer tank, oil spill and burning of oil spattered at a distance from the transformer.
	Medium or low voltage (voltage level < 50 kV):		This code includes all transformers that are not integral parts of another code. Control power transformers and other small

Component Types			Component Definitions for Coding
			transformers, which are sub-components in electrical equipment, should be ignored. They are assumed to be an integral part of the larger component. Examples of transformers accounted for in this code include transformers attached to AC load centers, low voltage regulators, and essential service lighting transformers.
		Dry	Dry medium or low voltage transformers are typically cabinet external transformers with lower fire load.
		Oil filled	Oil filled medium or low voltage transformers are typically cabinet external transformers using oils as coolant.
Transient material			This code should be used to classify transient fires of materials as e.g., trash cans, stored personal protection materials, additional outage load such as temporary scaffolding, etc., which are specifically initiated by hot work activities. This would not cover transient fires where the source of the fire is unknown such as self-ignited rags.
Turbine generator:			
	Exciter		The turbine generator exciter is a well-defined item. Generally, there is only one exciter per unit.
	Hydrogen		This code is limited to the complex of piping, valves, heat exchangers, oil separators and often skid-mounted devices that are associated with turbine generator hydrogen. Caution: It is important to have a clear definition of the turbine generator system boundaries to distinguish between turbine generator hydrogen fires and miscellaneous hydrogen fires being included in a separate code.
	Oil involved		Similar to hydrogen, this code is limited to the complex of oil storage tanks, pumps, heat exchangers, valves, and control devices belong to the turbine generator oil system.
Valve			This code covers large valves that include hydraulic fluid powered mechanisms. (e.g., main steam isolation valves, turbine stop valves, etc.).
Other component			This code shall be used if the predefined codes are not applicable. The coding shall be complemented by a descriptive text that describes the type of component.
Unknown			A component can be linked to the fire, but the type is unknown.

Table 5 Component specific fire occurrence frequencies in the most recent version (September 2011) of the OECD FIRE Database [1]

Type of Component	Number of Fire Events
Component (other than cable) ignited by hot work	41
Transient material	26

Type of Component	Number of Fire Events
Pump (electrically driven or turbine driven)	22
Electrical cabinet, low voltage (non-HEAF, < 1 kV)	22
Fixed heater	20
High voltage transformer (voltage > 50 kV), oil involved, catastrophic	19
Other component	19
Portable equipment (e.g., heaters, low power electric equipment, etc.)	17
Electrical cabinet, high or medium voltage (non-HEAF, ≥ 1 kV)	15
Medium and low voltage transformer (voltage level < 50 kV), dry	17
Cable run (self-ignited), power cables	13
Fan	13
High voltage transformer (voltage > 50 kV), non-catastrophic	13
Valve	12
Filter	11
Turbine generator, oil involved	11
Diesel generator	10
Breaker	9
Electrical cabinet, high or medium voltage (HEAF, > 1 kV)	9
Electric motor (not in pump)	8
Miscellaneous hydrogen containing equipment (e.g. piping)	7
Turbine generator, exciter	7
Cable run (self-ignited), I&C cables	6
Rectifier, inverter, or battery charger	6
Bus duct	5
Dryer	5
Equipment for illumination	5
Oil separator or oil stripper	5
Reactor coolant pump (RCP, for PWR)	3
Turbine generator, hydrogen	3
Medium and low voltage transformer (voltage level < 50 kV), oil filled	3
Electrical cabinet, low voltage (HEAF, < 1 kV)	2
Hydrogen containing vessel	2
Iso-phase duct	2
Junction box	2
Unknown	1

Type of Component	Number of Fire Events
Compressor	1
RPS motor generator sets	0
Battery	0
Boiler	0
Cable (fires caused by welding and cutting)	0
Main control board	0
Off-gas/hydrogen re-combiner	0
Main feedwater pump	0
RPS motor generator sets	0
Total number of events:	***392***

Table 6 provides the number of fire occurrences for the most frequently affected components. The various sub-categories of components are combined into one category each.

Table 6 Total amount of fire occurrences for main categories of components

Type of Component	Number of Fire Events
High voltage transformer	32
Electrical cabinet, low voltage	24
Electrical cabinet, high or medium voltage	24
Turbine generator	21
Medium and low voltage transformer	20

Several observations can be made from Table 5 and Table 6:

Transformers (high, medium and low voltage transformers) are the most frequent fire source with in total 52 events of 392 events in the FIRE Database representing approx. 13 % of the events collected. Catastrophic failures of high voltage transformers, accompanied by oil spills, are somewhat more frequent than those of the mostly dry medium and low voltage transformers.

Fires at electrical cabinets with 48 occurrences also provide a contribution of approx. 12 % of all events collected.

Fires associated with portable equipment, other components, transient material and components (other than cable) ignited by hot work together make up about ¼ of all events in the Database. The highest individual count among them can be observed for "component (other than cable) ignited by hot work". These fires are mainly caused by cutting and welding during maintenance and repair work; typically they are quickly extinguished by the staff involved in the hot work without developing to a significant fire. "Other component" includes various fire sources such as trailers outside the plant buildings, test loads, forest fires and other wild fires in close proximity to the plant. Most of these events are reported by those countries reporting all events, even very minor ones. More than one half of the 21 turbine fires involve oil fires, in most cases due to leaking oil pipes and seals.

Collection of Component Numbers and Estimate of Relative Component Fire Frequencies

The numbers of the components provided in the list as presented in Table 4 are to be collected in the frame of a currently ongoing activity by the participating countries for all their plants. From these numbers average generic numbers of components can be generated at least for some components. Combined with the overall occurrence frequencies as in Table 4 or with country specific occurrence frequencies, it should be possible to estimate generic relative component specific fire frequencies that may support Fire PSA, e.g., by using them as a priori information in Bayesian updates of plant specific frequencies.

Up to now, such numbers are available for selected components only from four Finnish plants (2 BWRs and 2 PWRs) and 34 nuclear plants in France with 900 MW_e PWR. For the Finnish plants the numbers were collected in an observation period of 17 years, for the French plants in an observation period of 12 years. As an example, Table 7 shows for the selected components the total numbers of components, the corresponding total years of operation of the components (reactor years), the numbers of fire events occurred, and the estimated relative component specific fire frequencies. In this table, the term "Finland" is used for four plants from Finland, the term "900 MW_e" for 34 French 900 MW_e plants. Once the collection of the numbers of components from all countries and for all plants is completed estimated relative component specific fire frequencies can be obtained by combining them with the numbers of occurrences for the individual components.

Table 7 Example of a sample of number of component types, corresponding numbers of events and estimated relative frequencies

Component Type	Data Source	Total Number of Components	Number of Component Years	Number of Events	Fire Frequency per Reactor Year
High voltage transformer	Finland	24	408	0*	7.3 E-03/a
	900 MW_e	272	3264	0*	1.2 E-03/a
Diesel generator	Finland	16	272	3	4.4 E-02/a
	900 MW_e	102	1224	2	4.8 E-03/a
Turbine generator	Finland	6	102	2	2.9 E-02/a
	900 MW_e	34	408	1	2.4 E-03/a
Medium and low voltage transformer, dry	900 MW_e	1462	17544	3	7.3 E-03/a
Pump	900 MW_e	6698	80367	4	9.3 E-03/a
Fan	900 MW_e	7072	84864	2	4.9 E-03/a
Heater	900 MW_e	16082	192984	2	4.8 E-03/a

* counted as 0.5 events

IMPROVEMENTS WITH RESPECT TO FIRE SEVERITY

Enhancements in Assessing Fire Damage Quantitatively in Fire PSA

Derivation and quantification of plant specific event trees as a tool for determining conditional probabilities of fire induced component damages is a major item in Fire PSA for the international community. For improved support of Fire PSA the OECD FIRE Database has been enhanced with respect to its capability for providing generic conditional probabilities for branch points in fire event trees. In the Database versions containing data up to the end of 2010 the effects of heat and hot gases could only be represented by one single code, for example by "single component fire" or alternatively, e.g., by the code "adjacent rooms affected". It was not possible to capture complex situations of fire spreading by the available coding. As an example, a cable fire damaging cables connected to components in adjacent rooms or in other fire compartments may impair the functionality of components in these locations. This needs to be coded by separate codes: single component fire, fire confined to one room, adjacent room affected (or other fire compartment affected). In addition, it should be possible to distinguish between direct influence of the fire by hot gases or pressure build-up and indirect influence through consequential functional effects on components. To this end, the following codes have been implemented in the OECD FIRE Database, with multiple choices coding being possible:

- Single component fire: A single component is damaged/deteriorated or destroyed by the fire:
- Multiple component fire: More than one component is damaged/deteriorated or destroyed by the fire.
- Multiple components affected: Multiple components are affected by consequential functional failures of components.
- Total loss of the room where the fire occurred: All equipment becomes unavailable by fire (or its effects) in the room where the fire occurred.
- Fire confined to one room: Fire and its effects are limited to the room where the fire occurred.
- Adjacent room affected: Components in rooms adjacent to the room where the fire originated are affected by fire or its effects. However, the fire has not propagated to other fire compartments.
- More than one fire compartment affected: The fire or its effects propagated from the original fire compartment to at least one other compartment.
- Structural influence or collapse: Structures or structural elements (including typical ones belonging to fire barriers) or buildings are damaged or do collapse.

Regarding the term "affected", distinction is made by a flag between direct influence of fire by hot gases or pressure build-up, and influence through consequential functional effects on components.

By the end of 2010 the database contained 392 fire event records. Yet this population is inhomogeneous, as the criteria for reporting fire events vary among the participating NEA member countries. For statistical analysis and use in a Fire PSA a suitably homogeneous event population is needed. Therefore, incipient (pilot) fire events that are only reported from some countries are eliminated for the purpose of the analysis of conditional branch point probabilities. This leaves approx. 300 fire events from all plant operational states.

These events were the basis for the evaluation of the more severe events, defined as follows in the FIRE Database: The term "more severe events" includes: adjacent rooms or other compartments directly affected by fire and/or fire effects, adjacent rooms and/or other fire compartments affected by consequential functional effects on components, total loss of the room where the fire originated.

Fifteen events have been identified in which adjacent rooms and/or more than one fire compartment were affected; thereof:

- Four events (representing a conditional probability of 1.3 E-02) with direct fire effects due to heat or combustion products on systems, structures and components in adjacent rooms,
- One event (representing a conditional probability of 3.0 E-03) with consequential functional effects (e.g. spurious actuation of components) on components in adjacent rooms,
- One event (representing a conditional probability of 3.0 E-03) with direct fire effects in other compartments,
- Three events (representing a conditional probability of 1.0 E-02) with consequential functional effects on components in adjacent rooms and in other compartments,
- One event (representing a conditional probability of 3.0 E-03) with direct fire effects and consequential functional effects on components in adjacent rooms and in other compartments, and
- Five events (representing a conditional probability of 2.0 E-01) with consequential functional effects on components in other fire compartments.

Five of the ten events with consequential functional effects on components in other rooms or compartments were caused by cable or bus duct fires, and three of the events with direct effects on adjacent rooms and/or other compartments were caused by high voltage transformer fires.

Total loss of the room where the fire originated occurred in four events (representing a conditional probability of 1.3 E-02). More information on the origin of fires with potentially severe consequences is provided in Table 8.

Table 8 Origin of those fire events in the OECD FIRE Database with more severe consequences

Adjacent Rooms or Other Compartments Directly Affected by Fire Note: If the location is "outside", "adjacent rooms affected" means that adverse effects of the fire have spread beyond the immediate vicinity of the origin of fire		
Building	***Room***	***Component***
Outside, near turbine building	Transformer room/bunker	HV transformer, oil involved
Outside	Transformer room/bunker	HV transformer
Outside	Switchyard	HV transformer, oil involved
Turbine building	Process room	Turbine generator, oil involved
Turbine building	Process room	Turbine generator, exciter
Waste disposal building	Room for ventilations	Fan
In 3 of the 6 cases the fire started at HV transformers, in 2 cases at turbine generators.		

Adjacent Rooms and/or Other Fire Compartments Affected by Consequential Functional Effects on Components		
Building	***Room***	***Component***
Turbine building	Room for electrical control equipment	Cable run, power cables
Turbine building	Cable room	Cable run, power cables
Electrical building	Cable penetration	Cable run, power cables
Electrical building	Switchgear room	Cable run, power cables
Reactor building	Switchgear room	Bus duct
Electrical building	Battery room	Battery charger
Turbine building	Switchgear room	Breaker
Auxiliary building	Main control room	Medium or low voltage transformer, dry
Diesel generator building	Switchgear room	Electrical cabinet, high or medium voltage
Outside, near turbine building (same as # 1 in "Adjacent rooms or other compartments directly affected by fire")	Transformer room/bunker	HV Transformer, oil involved
In 5 of the 10 cases the fire started at cables or cable ducts, in other cases the fire started elsewhere in the room where the fire started and damaged cables in that room.		
Total Loss of the Room Where the Fire Originated		
Building	***Room***	***Component***
Switchyard	Switchyard	HV Transformer
Outside, near turbine building (same as # 1 in "Adjacent rooms or other compartments directly affected by fire"	Transformer room/bunker	HV Transformer, oil involved
Auxiliary building	Other type of room	Other component (test load)
Diesel generator building	Diesel generator room	Diesel generator
In two of the four cases, the fire started at HV transformers.		

In all, Table 8 shows that cable runs and ducts, high voltage transformers and turbine generators are the dominant sources for severe consequences of fires.

The conditional probabilities, as listed above, and their uncertainty distributions, can be used as pure information (to support the decision process of the analyst) or can be applied as conservative assessment directly within the screening process or as priori information in a detailed analysis. The chosen utilization of the data depends on scope and quality of the available plant specific data.

In this context, it has to be mentioned that, e.g., Fire PRA developed in the U.S. do not assess fire damage based on fire events as described above. In order to assess the specific configuration at a particular nuclear power plant, fire models are employed to determine damage to cables and equipment. Fire models do not only assess initial damage, but can be also used to assess propagation of fires and consequential damage.

CONCLUSIONS AND OUTLOOK

The evaluations of the data collected in the OECD FIRE Database meanwhile allow providing support for major parts of Fire PSA.

It is principally possible to estimate generic building and compartment specific fire occurrence frequencies for all plant operational states, full power as well as low power and shutdown states for different reactor types. However, due to different reporting criteria and thresholds in the member countries, the data to be considered for real case applications may vary. Data from those countries reporting all fire events may be directly used, while the data pools from other countries may need some further investigation.

For BWR type reactors these analyses demonstrated that the dominant contribution results from process rooms, and, in particular from process rooms in the turbine building.

The collection of average compartment numbers for nuclear power plants with PWR is currently ongoing, after completion, results analogous to those for BWR plants will be provided.

The list of components considered in the data collection has been updated and improved in several ways. One major motivation was the desire to make it as far as possible comparable to the component list for U.S. Fire PSA in NUREG/CR-6850 [2] to facilitate comparisons and to generate generic component specific fire occurrence frequencies for PSA use.

The recent Database indicates that transformers (high, medium and low voltage ones) represent the most frequent fire source with a contribution of approximately 12 % (46 of 392 events). Fires at electrical cabinets with 45 occurrences also provide a contribution of nearly 12 % of all events collected.

The numbers of the components provided in the list as presented in Table 4 are collected from the participating countries for all their plants. From these numbers average generic numbers of components can be generated. Combined with the overall occurrence frequencies as in Table 5 or with country specific occurrence frequencies, generic relative component specific fire frequencies can be estimated that may support Fire PSA, e.g., by using them as a priori information in Bayesian updates of plant specific frequencies. Such frequencies are currently available for selected components from four Finnish plants and 34 nuclear plants in France with 900 MW_e PWR.

Last not least, for improved support to Fire PSA, the OECD FIRE Database has been enhanced with respect to its capability for providing generic conditional probabilities for branch points in the fire event trees for application in Fire PSA internationally. In order to assess the specific configuration at a particular nuclear power plant, fire models are employed to determine damage to cables and equipment, and can assess propagation of these fires.

The National Coordinators of the OECD FIRE Project member countries have decided to focus more on Database applications in the third phase of the Project. One typical example is the generation of generic compartment specific as well as the component related fire occurrence frequencies, which may be applied for Fire PSA for those plants with insufficient data on fire frequencies.

In the discussions of the Project members various analytical issues turned out to be of interest for Database application. It was clearly pointed out that OECD FIRE Database, depending on the objectives, can be used to improve, both deterministic safety assessment and PSA in regard to fire event analysis.

After the analysis of HEAF events, the Project will focus on comparing the fire protection standards in the different member countries for finding out if, and to what extent, fire events in the Database are correlated to the fire protection standard in the plants.

Further topics to be analyzed in the OECD FIRE Database Project have been proposed. In particular, the analysis of so-called challenging fires in areas relevant to safety, such as switchgear fires, relay room fires, MCR fires, has been mentioned. In this context, boundary conditions that may affect the course of a fire event could provide essential insights with respect to mitigating the consequence of a fire event.

Another topic of broad interest is the fire suppression analysis. Questions arising on this topic are how effective suppression is in the context of mitigating the consequences of a fire event or what type of suppression is more effective from viewpoint of fire event control and consequences limitation.

Fires due to fire loads being not continuously present and transient ignition sources, often related to hot work are more or less the most important ones during low power and shutdown plant operational states. The Database may give indications on what can be done to minimize the fire risk and reduce the consequences with respect to these types of fires.

The Database should also help to identify potential root causes of fire events for gaining insights on potential measures for prevention.

Human factors can affect the course of a fire event, both in the initiating phase and in mitigating the consequences. Therefore, it seems to be useful to investigate the human interface in the context of fire management from events in the Database, where possible. It should be clarified, how far human interaction can be considered as "optimum" from the operating safety point of view and to find out, when it is likely that human interaction can contribute to increase the risk or to reduce it.

The analysis of all the above mentioned issues may serve to support nuclear plant modernization projects as well as improving the fire protection standards for future reactors. In this context, a high quality of the event records with as much information being collected as possible is essential for applying the data successfully. The value of the OECD FIRE Database will increase over the years with a continuously growing amount of event data being available to the analysts.

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RELIABILITY DATA FOR FIRE PROTECTION FEATURES IN GERMAN NUCLEAR POWER PLANTS

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ABSTRACT

The German regulations for safety reviews of nuclear power plants require deterministic safety assessment as well as probabilistic safety analyses (PSA). With respect to the assessment of plant internal fire hazards, a Level 1 PSA for all plant operational states including Fire PSA is required.

In the frame of generating fire specific event and fault trees for Fire PSA purposes, a large amount of plant specific as well as generic data are needed starting with fire frequency data. The probabilities of damage to safety related components or to components triggering an incident initiating event in case of their failure are modeled by the event tree method. Important branches of the tree are fire detection and alarm including verification of the fire, fire extinguishing by different possible means, and fire propagation to adjacent compartments.

For modeling these branches in the event trees accurately, reliability data of the active fire protection features involved are needed.

For the following fire protection features, plant specific as well as generic technical reliability data have been generated and are being updated at the time being:

- Fire detection systems including alarm boards and panels as well as fire detectors including their power supplies,
- Fire and smoke dampers in ventilation systems,
- Fire doors,
- Electrically controlled hold-open systems for fire doors,
- Stationary fire extinguishing systems and equipment including the corresponding extinguishing media supplies.

The data are collected by evaluating the operating experience in nuclear power plants via the results of periodic in-service inspections. In this context, it has to be distinguished between findings providing information only on deviations from normal operating conditions and findings representing operational failures of the intended protection function in case of demand.

As a result from these evaluations failure rates (failures per hours of operation) as well as unavailability per demand are calculated for a variety of fire protection features from six German nuclear power plants (NPP) consisting of seven units representing 133 reactor years of plant operation.

INTRODUCTION

In the frame of the obligatory Safety Reviews (SR) periodically to be carried out for German nuclear power plants (NPP) at least every ten years, probabilistic safety analyses (PSA) up to Level 2 PSA have to be performed. Level 1 PSA covers all plant operational states (POS) and all plant internal events – such as fire - as well as internal and external hazards for full power operation states.

Highly important branches of the tree are fire detection and verification of the fire, fire extinguishing by different possible means, and fire propagation to adjacent compartments.

Generating fire specific event and fault trees for Fire PSA, a variety of plant specific as well as generic data are needed starting with fire frequency data. The probabilities of damage to safety related components or to components triggering an incident initiating event in case of their failure are modeled by the event tree method. Fire detection and alarm including verification of the fire, fire extinguishing by different possible means, and fire propagation to adjacent compartments represent highly important branches of the tree.

To model these branches accurately, reliability data, in particular of the active fire protection features involved are needed to be plant specifically collected and assessed. The data from the operating experience of German NPP are collected based on an evaluation of the results of periodic in-service inspections. From the evaluation of the inspection results the failure rate and the unavailability per demand are calculated for different systems and components.

For the following fire protection features with some active parts plant specific as well as generic technical reliability data have been generated in the past [1], [2], [3], and [4]:

- Fire detection systems including alarm boards and panels as well as the corresponding fire detectors including their power supplies,
- Fire dampers in ventilation systems,
- Smoke dampers as part of the ventilation systems,
- Fire doors,
- Electrically controlled hold-open systems for fire doors,
- Stationary fire extinguishing systems and equipment including the corresponding extinguishing media supplies.

The existing database from the operating experience of six reactor units, three of them pressurized water reactors (PWR) and three boiling water reactors (BWR), is being extended at the time being. This update covers extending the time periods to be observed to in total 123 reactor operation years for the nuclear power plants already under consideration up to the end of 2011. In addition, data from a further German NPP unit (with BWR) are being collected for a time period of at least ten years.

In addition, it is intended to enlarge the database further by collecting data also on passive structural fire protection elements such as cable penetration seals from the additional NPP under consideration.

METHODOLOGY FOR ESTIMATING TECHNICAL RELIABILITY DATA OF ACTIVE FIRE PROTECTION FEATURES

The operational behavior of fire protection features may vary depending on type and manufacturer as well as on the maintenance status of the respective systems and components. Therefore, generic data for the technical reliability of such equipment can only be considered to a limited extent. Moreover, plant specific data are needed for the individual features. In particular, data from nuclear installations differ from those in non-nuclear ones because of the more systematic, frequent and detailed inspections and maintenance in nuclear installations.

As a contribution to Fire PSA, input data on the reliability of various fire protection features are required. Two types of data can be distinguished:

- Failure rate λ per hours of operation and
- Unavailability per demand *P(t).*

The two types of data are connected according to the following equation:

$$P(t) = 1 - e^{-\lambda t} \approx \lambda\, t,$$

where t is the time period since the last inspection. For fire protection features in German plants, for conservative reasons, t is typically set to the time period between two regular in-service inspections.

To calculate the reliability data from the raw data on the plant specific observations and findings from the inspections, the data are statistically processed based on the approach of Bayes according to [5] and [6]. By this approach, different uncertainty factors such as model assumptions, limited extent of observations, different boundary conditions, etc. are to a certain degree taken into account. Finally, a lognormal distribution with mean values and corresponding scattering factors *k*, their distribution ranges and the scattering factors of the estimated values, are provided for the unavailability per demand as well as for the failure rate for each type of component.

For assessing the raw data, suitable criteria are needed to receive realistic values for failure rate and unavailability of active fire protection features to be applied within Fire PSA. Considering the significance of the affected component for application in the Fire PSA event tree, a careful and consistent assessment based on expert knowledge, whether the documented findings can be estimated as failures or only as deficiencies, is necessary.

FIRE PROTECTION EQUIPMENT RELEVANT FOR ASSESSMENT

There are different structures of fire event trees with different complexity described in the available literatures [7] to [9]. However, all of them include the following three functions of fire protection:

- Fire detection and verification,
- Containing the fire in the fire compartment, and
- Fire suppression.

The availability of these functions depends on the availability of suitable and reliable fire barrier elements, active protection features as well as on human actions. Examples of human activities are fire verification, as well as fire fighting either by plant staff or by the off-site fire brigade, but also activities contributing adversely to the course of the event, for example blocking a fire door in open position during operation. However, the human factor is not considered in this work. This is done within a separate analytical approach.

The focus of this paper is on the technical reliability of active fire protection features. For the following equipment, plant specific technical reliability data have been generated in the past or are currently being expanded:

- Active fire protection features:
  - Fire detection systems including alarm boards and panels as well as the automatic fire detectors with their power supplies, and push button detectors,
  - Fire dampers in the ventilation systems,
  - Fire doors, including those with electrically controlled hold-open systems,
  - Fire extinguishing systems and equipment (mainly stationary) including their corresponding extinguishing media supplies such as water deluge systems, sprinklers (wet and dry), fire water pumps, wall and field hydrants, and gas extinguishing systems;
- Passive fire protection means:
  - Cable penetration seals.

The data generation is based on the available documentation of periodic in-service inspections and partly on the documentation of maintenance and repair activities. Additionally, responsible plant staff is interviewed with regard to maintenance procedures and documentation. Detailed knowledge of the plant specific conditions is evident for the assessment. That means a thorough plant walk-through is necessary for all plant locations where relevant fire protection features are installed. In addition, a good co-operation with the plant staff is needed for a meaningful assessment.

The following documents containing the relevant information on functional disturbances as well as on deficiencies and failures of systems and components observed during the regular in-service inspections and walk-throughs were taken into consideration for the analysis:

- Records of periodic tests, regular inspections and maintenance (incl. test and inspection procedures and reports),
- Work and maintenance orders,
- Deviation reports, and
- Repair reports, if necessary.

In Germany, the active function of all systems and components is inspected regularly via a component specific inspection scheme which can be found in the plant fire protection manual. The findings observed during these inspections, e.g. functional disturbances, deficiencies and potential failures are documented in the inspection records.

Suitable criteria are necessary to reveal realistic values for purely technical failure rates and unavailability of active fire protection features. By these criteria, the reported findings must be either estimated as “failures” or only as “deficiencies”. A failure of a component implies unavailability of the required fire protection function in case of fire, whereas a deficiency is not considered for further evaluation. The assignment of a finding as “failure” or “deficiency” is based on expert judgment, and has to consider operation conditions and relevant fire scenarios in NPP.

In the following, for the fire protection features mentioned above the designed behavior and protective function is outlined. Then typical deficiencies and failures are described which may be plant specific.

FIRE DETECTION AND VERIFICATION

In this paper fire detection is considered as the process of detection of smoke or fire by fire detectors installed in the fire compartment or adjacent compartments. Verification implies that the initial information is verified by the shift supervisor, who can initiate adequate actions [9]. Typically, after an alarm of only one detector has been recognized in the main control room, plant staff is sent to the location of that detector by the shift supervisor to verify the fire. In the case of more than one detector being triggered, this is also considered as a verification of a fire. A notification by manual fire alarm buttons (push buttons) or by telephone calls is regarded as verified.

The designed function of the automatic fire detection system starts from detection of smoke/fire by the detector and ends with the notification/alert at the main control room. To generate reliability data for the whole system by means of a specific event tree for the detection, the system has been hierarchically sub-divided into

- Automatic fire detectors of different types (ionization detectors, optical detectors, others),
- Fire detection lines,
- Subsidiary fire alarm boards and/or panels, and

- Central fire alarm panel.

To collect data for the reliability of manual notification means, the inspection reports on push buttons have also been evaluated.

Automatic Fire Detectors of Different Types

In principle, four different types of automatic fire detectors are installed in German nuclear facilities:

- Ionization detectors,
- Optical smoke detectors,
- Flame detectors, and
- Differential thermal detectors.

The inspection interval for automatic fire detectors is typically one year [10]. At certain locations, where inspection is not possible during power operation due to radiation, the interval is extended to one fuel cycle (which may exceed the period of one year to 15 – 16 months). The inspection is conducted as visual inspection and the trigger function is checked with artificial smoke or in case of heat detectors with hot gas. In the frame of the inspections, the detectors are frequently replaced by new detectors on a regular basis. That means that in case of a detector exchange the regular inspection of the replaced component does not take place but the new one is installed and immediately inspected. If the new detector fails, it will be exchanged without any special comment in the documentation.

Failure of an automatic fire detector means that automatic detection function is not given. However, there is generally more than one detector installed in a compartment. Therefore, the automatic fire detection may only be delayed. A complete loss of all detectors in one compartment due to independent failures is not considered, because this combination is rare compared to failure of the complete detection line.

The expected function is that a signal is indicated at the main control room after smoke or heat having been generated and distributed to the detectors. Examples of findings considered as failures are:

- Defect detector or fire detection line,
- Detector fails to actuate, and
- Detector gives malfunction message instead of fire alarm.

Fire Detection Lines

Every third month, 25 % of the detectors of one line are required to be inspected. However, for inaccessible locations the interval is extended to one fuel cycle.

Failure of fire detector line leads to unavailability of all connected automatic detectors or push buttons. As one fire detector line may protect two adjacent compartments, also the "indirect" fire detection by signals from adjacent rooms may fail.

Subsidiary Fire Alarm Boards and Panels

Subsidiary fire alarm boards and panels are located as subsystems in some NPP. Failure of subsidiary fire alarm boards and panels leads to unavailability of all connected fire detection lines and thus to the unavailability of the detectors belonging to those lines. As one subsidiary fire alarm board or panel may protect two adjacent compartments, also the "indirect" fire detection by signals from adjacent rooms may fail.

Central Fire Alarm Panel

The central fire alarm panel is self-monitoring and equipped with redundant energy supplies protected by batteries. Therefore, and based on the entire operational experience available, a complete failure of the central panel is not considered any more at this point [1], [3]. Failures of sub-systems are considered by assessing the data of the subsidiary fire alarm boards and panels.

Push Buttons

According to the German nuclear standard KTA 2101.1 [10], push buttons need to be inspected annually under supervision of an independent inspector. In deviation to the current standard (3 months), for the reference plant called “NPP 3”, the interval was extended to one year in an early evaluation [1]. For “NPP 5”, an additional annual internal inspection has been reported [4].

The expected function is that a signal is indicated at the main control room after the push button has been pressed once. Examples for findings considered as deficiencies are:

- False alarm by moisture,
- Damaged cases or cover glass, and
- Seizure of the button after it was pushed.

If a push button does not trigger after the first attempt or do not trigger at all, this is considered as a failure.

SEALING OF THE FIRE COMPARTMENT

The function of sealing the fire compartment prevents the fire from spreading to adjacent compartments. In addition, it reduces oxygen supply to the fire limiting the fire development in the primary fire compartment. In German NPP, the required fire resistance rating of structural fire barrier elements, such as walls and ceilings, fire doors, hatches, fire dampers, etc., is basically 90 min or at least 30 min. The required fire resistance rating is validated through a fire test according to the ISO standard fire curve [11], [12]. In short, the principal test criteria for the elements are (1) to keep the separating function, (2) not to be heated up on the cold side more than 180 K, (3) not to allow flames going through, and (4) not to allow a piece of cotton wool (on the cold side) being ignited, if it is exposed to hot gases going through [12].

It is considered within this analysis and backed by operating experience [13] that passive fire barrier elements separating fire compartments from each other (e.g. walls, ceilings) are considerably more reliable than active elements such as fire dampers or fire doors. Therefore, the assessment is mainly focusing on active fire barrier elements.

Fire Doors, Including those with Electrically Controlled Hold-open Systems

Fire doors are inspected annually in all German nuclear power plants under supervision of an independent inspector. Only “NPP 6” has semi-annually intervals for their inspections, with the independent inspector taking part annually. According to KTA 2101.1 [10] an annually interval, bi-annually supervised by an independent inspector, is possible. The inspection is conducted visually including checking the self-closing function of the door by the door closer. This is typically done from a 30° to 45° open position.

For assessing the reliability of fire doors, mainly the active function has to be considered, i.e. closing of the door by the mechanical door closer. If self-closing of a door did not work completely, this was assumed as a failure. Typical other findings are missing sealing bars between door and frame, problems with door fittings (lose handles, missing closing cylinders, etc.) or indentations of the door leaf. These findings are considered as deficiencies, because the protective function in case of fire is not or only insignificantly deteriorated.

According to KTA 2101.1 [10] electrically controlled hold-open systems need to be plant internally inspected once per month and once per year under supervision of an independent inspector. Before the standard KTA 2101.1 was applied, the internal inspection intervals varied from once per month ("NPP 2") to four times per year ("NPP 1") or even up to once a year ("NPP 3", "NPP 4" up to 1987, then every third month), depending on the plant.

For fire doors equipped with electrically controlled hold-open systems the standard door position is open. For these doors, it is checked if the door is released in presence of smoke and if double-winged doors are controlled to close in the right order by a door coordinator.

Fire Dampers in the Ventilation Systems

As required in KTA 2101.1 [10], fire dampers need to be inspected once per year under supervision of an independent inspector. Another interval of six months is determined by the technical approval of certain types of fire dampers. However, the second inspection within one year may take place without supervision.

For the evaluated plants and time periods, intervals significantly differ. For data collected in former projects [1] inspection intervals between three months and three years where found ("NPP 3", "NPP 4"), depending on the extent of the inspection. For newer time periods, an interval of one year became typical ("NPP 1", "NPP 2", and "NPP 4"), but there is also the 6/12 months interval without/with inspector ("NPP 6"). In case of inaccessibility during plant operation the inspection intervals for certain fire dampers are throughout one fuel cycle.

The actuation of fire dampers largely differs. Simple dampers are only actuated by a fusible link that reacts (melts) to hot fire gases. At the time being most dampers are additionally actuated pneumatically or electrically by signals from local smoke detectors, local control points or the main control room.

The inspections comprise

- Visual inspection of the inner and outer parts of the dampers,
- Checking of at least one actuation mechanism including demonstration that the damper blade closes and is locked in closed position, and
- Checking that the intended signals of the blade position are indicated at the local control point and/or the main control room.

All the different ways of actuating the dampers may not be checked in each inspection or checking them may not belong to the damper inspection itself but to the inspection of the detection systems.

For the evaluation, a failure of the required damper function is considered, if the actuation did not work or if the damper blade did not close completely. If only the indication of the blade position failed or the blade closed completely but failed to lock, this is only considered as a deficiency but not as functional failure.

Cable Penetration Seals

In the future, cable penetration seals as passive fire barrier (structural protection) elements are also intended to be investigated. One reason for this is the large number of cable penetration seals installed in nuclear power plants making them a relevant path for fire propagation. Furthermore, in contrary to fire doors and dampers, the fire might spread along the cables penetrating the fire barrier ("fuse effect"). Up to now, the reliability of such passive elements has not yet been considered in Fire PSA, but may contribute significantly to the reliabilities applied within the event tree analysis.

Preliminary evaluations have shown that there are some records of defect seals, at least in one of nuclear power plants under consideration. Due to backfitting measures including routing of cables the quality of reclosed seals may be deteriorated.

Pipe seals are not considered in the investigation because most piping material is non-combustible and the quality is not affected by backfitting measures.

FIRE SUPPRESSION

The evaluation of the technical reliability of fire suppression means mainly focuses on the fire water supply and on fixed fire suppression systems (by water or gases as suppression media). The reliability of portable fire fighting equipment, which also plays a significant role for NPP fires [13], has not yet been analyzed in detail.

A typical fire water system is supplied by two redundant pressure holding pumps and two redundant water supply (feed) pumps. Depending on the location of the fire the water is distributed at the fire location by field hydrants, wall hydrants, spraywater deluge systems, and/or sprinkler systems.

Fire Water Pumps

According to the German nuclear fire protection standard KTA 2101.1 [10], annual inspection intervals of fire water pumps including their energy supply and control equipment are required under supervision of an independent inspector. In addition, plant internal monthly inspections are required for the function of the pumps and weekly inspections for the energy supply and control equipment [10].

The plant internal inspection intervals for the pumps differed between the nuclear power plants being considered from a monthly interval ("NPP 2" to "NPP 4"(further in the past)) over three months ("NPP 4" (more recently) and "NPP 6) to a semi-annual interval ("NPP 1" and "NPP 5").

For determining reliability data, it was distinguished between the function of the water pumps and the function of the remote actuation of the pumps. A significant flow reduction resulting at least in a significant delay of suppression was considered as a failure, whereas a slight flow reduction was only considered as deficiency.

Field and Wall Hydrants

According to KTA 2101.1 [10], yearly inspection intervals of hydrants are required with every second one supervised by an independent inspector. Slightly different to these requirements, at "NPP 1" for wall hydrants also the yearly inspection is performed with an independent inspector being present. In addition to the requirements by KTA 2101.1 [10], some wall hydrants of "NPP 2 are inspected annually under supervision of an independent inspector. Field hydrants are plant internally inspected monthly and yearly under supervision of an independent inspector. At "NPP 3" semi-annual inspections are conducted for both types of hydrants.

Typical findings reported in the inspection documentations are "stiffly running" valves, a finding that is difficult to assess. In the frame of the first investigations [1] such findings were conservatively considered as failure. However, after discussions with plant personnel as well as independent experts and inspectors participating in some inspections, in later analyses [3] these findings were only regarded as deficiencies. Therefore, meanwhile hydrants are considered as highly reliable.

Water Deluge Systems

For water deluge systems, different inspections of differing complexity are required. In order to generate technical reliability data, the focus of the evaluation was on the actuation and function of the main water valve, for which internal inspections are required semi-annually and annually under supervision of an independent inspector [10]. In addition to these legal requirements, plant internal inspections were carried out every three months in "NPP 4" and take place every six weeks nowadays. In "NPP 6" internal inspections are carried out on a three months basis.

In German nuclear power plants, an automatic actuation of the main water valves by the automatic fire detection system is typically not performed. Valves may be actuated locally remote controlled from the main (unit) control room. If only the remote controlled actuation fails, the local actuation may still work after a given time delay. Therefore, reliability data distinguish between failure of the remote controlled actuation and total failure.

Typical deficiencies are

- A time delay in the opening of or a failure to close the main valve,
- Broken manometers or flow detectors, and
- Leakages.

Failures of the remote controlled actuation observed were

- Motor valve opened after manual assistance,
- Remote controlled actuation did not work, and
- Valve motor was reported to be changed after inspection.

Total failures of the deluge systems were

- Main valve could not be opened at all, and
- Remote control as well as local actuation impossible.

Gas Extinguishing Systems

For gas extinguishing systems, inspections are required semi-annually with an independent inspector taking part in every second inspection [10]. In addition to these legal requirements, plant internal inspections were performed at "NPP 4" every third month and are nowadays performed every six weeks. At "NPP 6", plant internal inspections are performed every third month.

Typical deficiencies are

- Dampers of the ventilation system did not close after actuation, and
- Missing signals.

One finding at a CO_2 gas extinguishing system was considered as a failure, referring to a malfunction of a lifting magnet for weight release [3]. On the other hand, a delayed initiation of the same CO_2 gas extinguishing system was only considered as a deficiency, since the extinguishing ability of the system was not impaired.

GENERIC RESULTS OF TECHNICAL RELIABILITY DATA FOR ACTIVE FIRE PROTECTION FEATURES

To accumulate robust reliability data to be applied in a Fire PSA it is necessary to have available plant specific as well as generic reliability data. Generic data are particularly needed if the plant specific operating experience of the plant to be analyzed does not provide a statistically meaningful and suitable database. The collected reliability data of active fire protection features in the nuclear power plants being investigated have essentially the same test intervals and are in effect comparable. Therefore, the generation of generic failure rate data is possible as described in [1].

An overview of the generic failure rates for active fire protection features installed in six German NPP units is given in Table 1.

Table 1 Generic failure rates for active fire protection features from five German NPP units (data taken from [14])

Active Fire Protection Feature	NPP Units	Scattering Factor k	Failure Rate [1/h] (expected value)
Fire detection systems and components:			
- Subsidiary fire alarm boards and panels	1, 3, 4, 5	8.9	8.8 E-06
- Fire detection lines/groups	1 – 5	6.8	8.4 E-08
- Automatic fire detectors	1 – 5	5.7	8.5 E-08
- Ionization detectors	1, 2, 5	8.7	9.5 E-08
- Optical smoke detectors	1, 2, 5	23.5	2.6 E-07
- Other detectors	1, 2, 5	8.9	3.7 E-07
- Manual fire alarm buttons (push buttons)	1 – 5	38.8	6.2 E-07
Fire dampers:			
- Total failure	1 – 5	11.8	1.2 E-06
- Failure of the remote controlled actuation only	1, 2, 5	46,0	8.2 E-06
Smoke removal dampers	1 – 4	5.5	3.7 E-06
Fire doors:			
- With electrically controlled hold-open devices	1 – 5	7.1	3.2 E-06
- Without electrically controlled hold-open devices	2 – 5	11.9	2.6 E-06
CO_2 extinguishing systems	2 – 3	12.8	1.9 E-05
Water deluge systems:			
- Total failure	1 – 5	9.6	1.8 E-06
- Failure of the remote controlled actuation only	1 – 5	5.7	6.2 E-06
Wet sprinkler systems	3	28.4	6.3 E-07

Active Fire Protection Feature	NPP Units	Scattering Factor k	Failure Rate [1/h] (expected value)
Fire water pumps:			
- Total failure	1 – 5	6.2	9.9 E-06
- Failure of the remote controlled actuation only	1, 4, 5	8.1	9.6 E-06
Hydrants:			
- Wall hydrants	1 – 5	7.7	4.7 E-07
- Field hydrants	1 – 5	7,7	1.3 E-06

The lognormal fitted data distribution of Table 1 is displayed graphically in Figure 1. It shows a uniform behavior for a majority of the active fire protection features investigated with failure rates in the range of E-07 /h to E-06 /h. The failure rates (failures per hours of operation) estimated for fire detection equipment such as detection lines and automatic fire detectors are better than those determined for other features. This might be due to the fact that the first are components without any mechanical elements. Components with mechanical elements such as fire dampers and doors or valves show more frequent findings impairing the required fire protection function resulting in more than one order of magnitude higher failure rates.

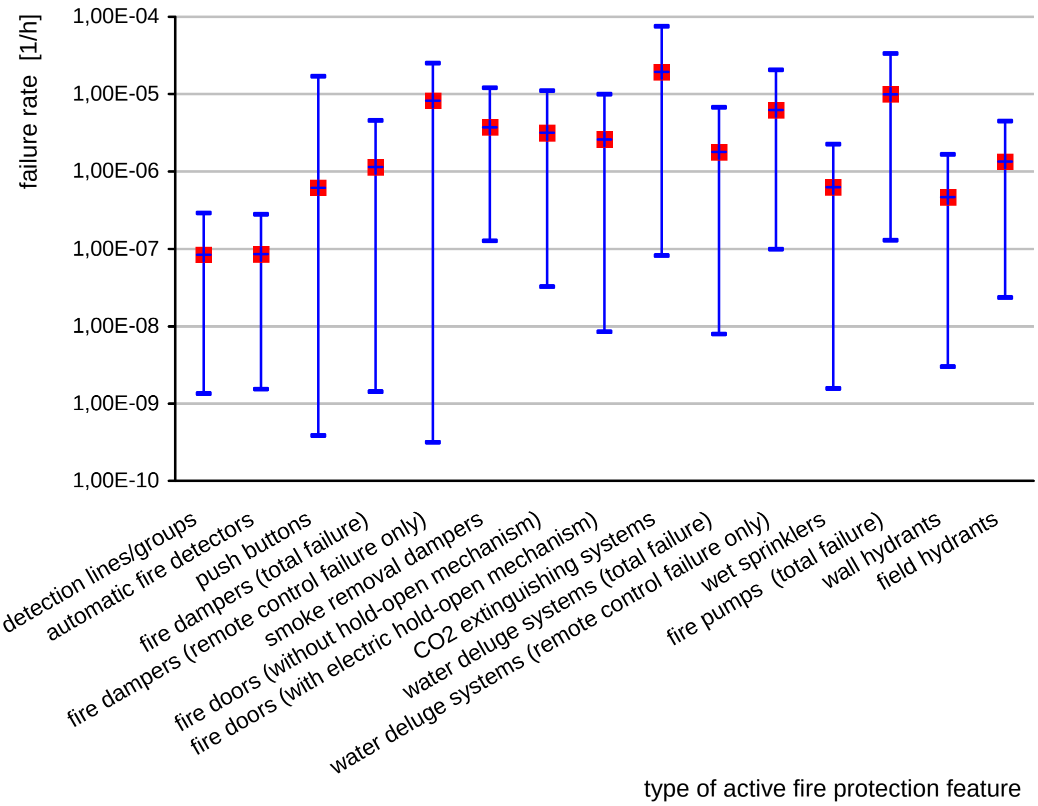

Figure 1 Lognormal data distribution of generic failure rates λ [1/h], with the 5 % and 95 % percentiles for active fire protection features as far as installed in the five German nuclear power plant units investigated in [1] to [3]

In this context, it has to be clearly stated that the highest values in the range of E-05 /h estimated for gas extinguishing systems can only be seen as a first rough estimate not applicable as verified input data for Fire PSA due to the still much too small database from only few operating hours in merely two reference plants. This is also valid for fire pumps, where the statistical database is not yet satisfactory. However, the relatively poor value of about E-05 /h is not so important since the fire water system has two redundant pumps.

The failure rates estimated are independent of time with regard to the inspection periods for in-service inspections. Therefore, the values determined for nearly all the active features mentioned above are applicable to any other German plant with an equivalent design of the fire protection systems and their active components.

CONCLUSIONS AND OUTLOOK

The generation of an extensive number of plant specific as well as of generic data from the operating experience with active fire protection features installed in German nuclear power plants representing an operating experience of 133 plant operating years has been ongoing stepwise since the late 1980's. Technical reliability data such as unavailability per demand and failure rate per operational hours have been determined and are being updated.

Throughout the assessment of the reliability data it has been repeatedly confirmed that a detailed analysis of the records of the inspection findings and further documentation correlated to the regular inspections is needed to determine the difference between a failure of the required function and a deficiency of the component behavior only. In order to assess this difference in a meaningful manner, detailed knowledge of the plant specific conditions is evident. This implies thorough plant walk-throughs for all plant locations where relevant fire protection equipment to be investigated is installed, as well as a direct and open communication with the plant personnel.

The data volume has been expanded in stages and is now going to cover six different German nuclear power plant sites, consisting of seven plant units. The assessment of the reliability data will comprise the experience of more than double the amount of plant operating years as well as the experience from an additional German plant with at least ten years of operating experience.

Furthermore, it is intended to enlarge the database by collecting data on passive fire protection means such as cable penetration seals from the additional nuclear power plant under consideration.

ACKNOWLEDGEMENT

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UPDATING OF THE FIRE PRA OF THE OLKILUOTO NUCLEAR POWER PLANT UNITS 1 AND 2

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ABSTRACT

The probabilistic fire risk assessment (fire PRA) of the Olkiluoto nuclear power plant (NPP) units 1 and 2 has been updated in the beginning of the year 2011. In the updated version, the ignition frequency of each component group is evaluated using a Bayesian approach and NUREG/CR-6850 data as a prior and the ignition frequency is divided on component basis.

The resulting total fire frequency in the PRA model of the units is 10 % higher than the historical fire frequency of the NPP unit. However, the fire frequency in certain rooms changed by orders of magnitude due to the transition to the usage of component based ignition frequencies. Especially, the fire frequency of rooms housing components with exceptionally high ignition frequency, such as main feed water pumps, large amount of electric equipment like relay rooms or rooms with hydrogen systems, increased in the updated evaluation.

In the evaluation, the safety related components are mapped to their locations. Furthermore, the cable routing database is used in order to locate the power and control cables of the safety related components. Also the locations of cables transmitting measuring data to the reactor protection system are evaluated. The resulting fire scenarios are grouped into tens of initial events of similar consequences using conservative assumptions.

The update of the fire PRA increased the core damage frequency (CDF) by 16 %. After the update, the CDF originating from fire-related initiating events constitutes of 20 % of the total CDF. The fire PRA demonstrates that in a high-redundant NPP unit, such as the Olkiluoto 1 and 2 NPP units, having four redundant divisions, each of 50 % capacity, a fire affecting two redundancies increases the conditional core damage probability by a factor of ten, compared to a situation when only one redundancy is affected by the fire, demonstrating the significance of active fire suppression systems in rooms housing two redundancies.

INTRODUCTION

Olkiluoto NPP units 1 and 2 (OL1 and OL2) are boiling water reactors (BWR) built by ASEA-ATOM in the late 70's and early 80's. Both units' safety systems are divided into four subsystems (divisions), each of which has a capacity of 50 %. Thus, two of the divisions must operate in order to successfully perform a given safety function. The components of the divisions are divided by fire compartmentalization. The exception is cables, of which maximum of two divisions may be present in the same fire compartment. In this case, the cables belonging to different divisions are divided by distance. Such fire compartments are always equipped with active fire suppression systems, usually sprinkler systems.

The probabilistic fire risk assessment (fire PRA) of OL1 and OL2 was finished in 1991. It has been updated several times, the latest in the beginning of the year 2011. In the updated version, one major modification is the evaluation of the room-based fire frequency using component based ignition frequencies. In the implementation of the fire PRA, the NUREG/CR-6850 report [1] has been used as reference.

THE IMPLEMENTATION OF THE FIRE PRA

The Scope of the Fire PRA and Partitioning of the Plant into Physical Analysis Units

The scope of the fire PRA is internal fires, which includes fires in the reactor building, turbine building, auxiliary buildings, control building, waste building. Further, also the fire water building and the water processing building are included in the scope, since they contain equipment, which is credited in the PRA model. External fires are treated in the context of external hazards. The included buildings contain about 1800 rooms. Some rooms are divided into areas having their own room codes, but such rooms are treated as one single room in the analysis. Rooms are part of a fire compartment, which can withhold a fire at least 60 minutes. In one compartment there are components and cables belonging to maximum of two divisions. The plant units are divided so that two divisions are located in the southern and western part of the unit and the two other divisions are located in the northern and eastern parts. In Olkiluoto, there is an on-site fire brigade, which is manned around the clock. All rooms in the plant unit are equipped with fire detectors. In the analysis, the basic assumption is that a fire in a room fails all components and cables in the room, but that the fire brigade will be able to withhold the fire inside the room and thus prevent spreading to other rooms.

Estimation of Room-Based Fire Frequencies

A very important part of a fire PRA is the estimation of fire frequencies. In the original implementation of the OL1 and OL2 fire PRA, the room-based frequencies was based on a few parameters evaluated with engineering judgment. In the update, the room-based fire frequencies are estimated with a method based on component based ignition frequencies.

Old Method

A crude but popular method to assess room-based fire frequency estimate is based on NUREG/CR-0654 [2], often referred to as Berry's method. In the implementation that was used in Olkiluoto, a number of parameters related to human presence, mechanical equipment and electrical equipment was used. Also judgment about the possibility to extinguish an incipient fire (pilot fire) with portable fire extinguishers and the amount of fire load was used. The decision tree used to estimate the fire frequencies is shown in Figure 1. Since the room based fire frequencies were normalized so that the sum of the room based fire frequencies equaled to the total fire frequency of the plant unit, the room based fire frequency for a typical room or compartment would be of the right range. However, the parameter values were based only on engineering judgment without knowledge of ignition frequency of specific equipment.

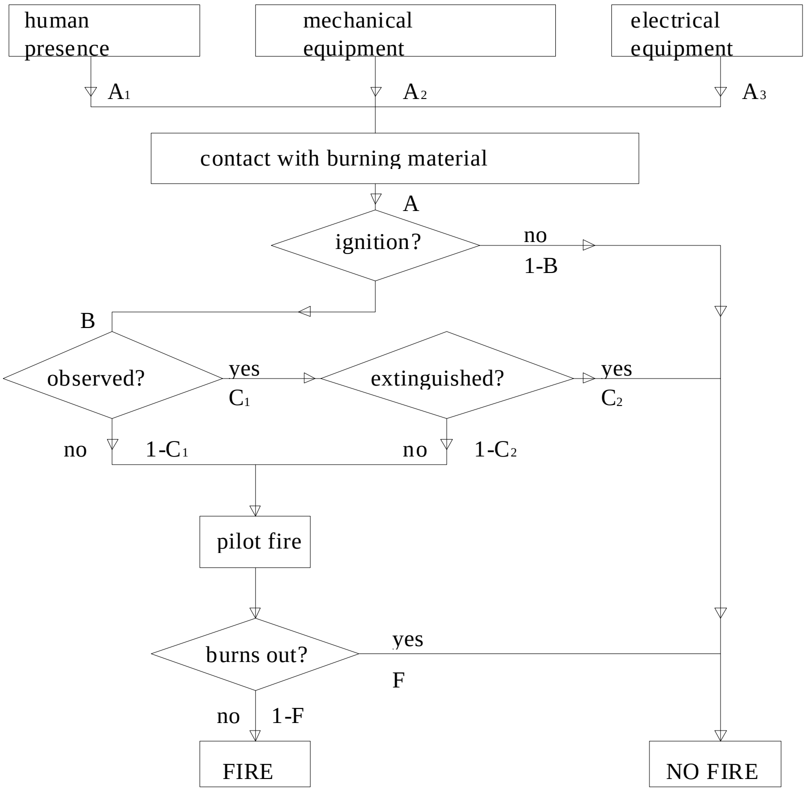

Figure 1 Fire frequency estimation with the old model

Method Based on NUREG/CR-6850

The room-based fire frequencies are evaluated based on dividing ignition sources into ignition bins. Most of these bins consist of countable components, such as pumps, motors, electrical cabinets, etc. Also self-ignition of cables, hydrogen piping, etc., is included. Transient fires, such as trash bin fires, fires induced from hot-work and other human activity are included as well. NUREG/CR-6850 [1] reports fire frequency estimates on a plant unit basis. Inherently to this method it is assumed that all units have the same amount of components, such as valves, pumps, etc.

The component fire source listing, including their locations is retrieved from the component database and the amount of cables present in the rooms is estimated using the cable pulling database. A walkdown was made during the update of the fire PRA. The walkdown was performed by a fire PRA modeler and a fire systems expert. In the walkdown, primarily the locations of the component ignition sources were confirmed and influence factors for transient factors, like occupancy of the room, amount of potentially ignition inducing maintenance activities performed in the room and presence of flammable gases and liquids, were evaluated.

The information on the number of component ignition sources, the influencing factors of transient ignition sources and the total length of cables in the room is used in calculating the room-based ignition frequency estimates. The component based fire frequencies are calculated using a Bayesian approach using the ignition frequencies presented in [1] as prior and historical fire events of the OL1 and OL2 plant units are used as evidence. The calculation of ignition frequency estimates of some component groups is shown in Table 1. Further, the calculation of room-specific fire frequency estimates of some rooms is shown in Table 2.

Table 1 Calculation of component ignition frequency estimates: some examples

Ignition source	Prior Distribution			Plant Specific Data		Posterior Distribution		
	α	β	Mean	N	T	α	β	Mean
Air compressors	5.5	2486.0	2.21E-03	0	54.7	5.5	2540.7	2.16E-03
Cable fires caused by cutting and welding	7.0	1674.0	4.18E-03	0	54.7	7.0	1728.7	4.05E-03
Cable run	12.0	2486.0	4.83E-03	1	54.7	13.0	2540.7	5.12E-03
Electric cabinets	109.5	2486.0	4.40E-02	1	54.7	110.5	2540.7	4.35E-02
Junction boxes	3.5	2486.0	1.41E-03	0	54.7	3.5	2540.7	1.38E-03
Pumps	52.5	2486.0	2.11E-02	0	54.7	52.5	2540.7	2.07E-02
Transient fires caused by cutting and welding	33.4	1674.0	2.00E-02	0	54.7	33.4	1728.7	1.93E-02
Transient fires	29.9	1674.0	1.79E-02	0	54.7	29.9	1728.7	1.73E-02
Transformer - catastrophic fires	10.5	1674.0	6.27E-03	0	54.7	10.5	1728.7	6.07E-03
Transformer - non-catastrophic fires	22.0	1674.0	1.31E-02	0	54.7	22.0	1728.7	1.27E-02
Transformer yard – others	3.5	1674.0	2.09E-03	0	54.7	3.5	1728.7	2.02E-03
Generator busbar	3.5	1674.0	2.09E-03	0	54.7	3.5	1728.7	2.02E-03

Table 2 Calculation of room-based fire frequency estimates: some examples

Room	Ignition Source	Weighting Factor		Ignition Frequency	
		Unit	Room	Unit	Room
Relay room	Junction boxes	2579	49	1.38E-03	2.62E-05
	Electric cabinets	852	144	4.35E-02	7.35E-03
	Transient fires	5126	5	1.73E-02	1.69E-05
	Transient fires caused by cutting and welding	1571	1	1.93E-02	1.23E-05
	Total				*7.41E-03*
Transformer yard	Generator busbar	9	1	2.02E-03	2.25E-04
	Cable run	2632216	260	5.12E-03	5.05E-07
	Junction boxes	2579	2	1.38E-03	1.07E-06
	Transformer - non-catastrophic fires	5	1	1.27E-02	2.55E-03
	Transformer - catastrophic fires	5	1	6.07E-03	1.21E-03
	Transformer yard - others	5	1	2.02E-03	4.05E-04
	Transient fires	5126	3	1.73E-02	1.01E-05
	Cable fires caused by cutting and welding	7146900	260	4.05E-03	1.47E-07
	Transient fires caused by cutting and welding	1571	1	1.93E-02	1.23E-05
	Total				*4.41E-03*
Pump room	Air compressors	156	1	7.28E-03	4.67E-05
	Cable run	2632216	95	5.12E-03	1.85E-07
	Junction boxes	2579	4	1.38E-03	2.14E-06
	Transient fires	5126	3	1.73E-02	1.01E-05
	Cable fires caused by cutting and welding	7146900	95	4.05E-03	5.38E-08
	Transient fires caused by cutting and welding	1571	1	1.93E-02	1.23E-05
	Total				*7.15E-05*
Pump room	Cable run	2632216	2062	5.12E-03	4.01E-06
	Junction boxes	2579	5	1.38E-03	2.67E-06
	Pumps	193	10	2.07E-02	1.07E-03
	Electric cabinets	852	1	4.35E-02	5.10E-05
	Transient fires	5126	3	1.73E-02	1.01E-05
	Cable fires caused by	7146900	2091	4.05E-03	1.18E-06

Room	Ignition Source	Weighting Factor		Ignition Frequency	
		Unit	Room	Unit	Room
	cutting and welding				
	Transient fires caused by cutting and welding	1571	1	1.93E-02	1.23E-05
	Total				*1.15E-03*

Selection of Safety Significant components and Cables

The PRA model is used in order to find components that perform mitigating functions after an initiating event and that may be affected by a fire. Initiating event frequencies are evaluated in the PRA model using historical evidence, and thus component failures leading to an initiating event are not included in the PRA model. Such components, which may be affected by a fire, are identified using the final safety analysis reports (FSAR), piping and instrumentation diagrams and electrical diagrams. Further, measurement points of the reactor protection system (RPS) are analyzed as well. The identified safety-significant components are listed together with the cabinets providing power to the component and the cabinets, where their control is performed. This information is used in order to identify the cables, which provide power and control signals to the safety significant components. The power sources of the busbars to which the power providing cabinets are connected are identified together with the power cables.

The cable pulling database is a very valuable tool in order to identify the locations of the safety-important cables. The database includes the information of the location of the end-points of segments of cable raceways. With this information and by using cable routing layouts, the locations of the cables can be identified with good accuracy.

From the list of safety important components, power sources and cables, a list of safety important rooms is compiled. Since initiating events frequency estimates in the PRA model are based on historical events, there is no need to consider fires that only lead to an initiating event. In the fire PRA, fire scenarios that may lead to an initiating event together with the failing of a component performing a mitigating function or fires that may affect a component in such a way that the plant has to be shutdown according to the technical specifications, are analyzed.

Consequences of Fires

Rooms, where the ignition probability has historically been judged to be high, or have high fire load, are equipped with active fire suppression systems. The majority of cable rooms are equipped with sprinklers, especially cable rooms, where cables belonging to two separate divisions are present. There exist, however, normal corridors where cables are routed and which are not equipped with active fire suppression systems.

In the fire PRA model, the basic assumption is that all components and power cables in the room involved by the fire fail. It is assumed that the fire spreads to control and instrumentation cables with 20 % probability. However, if a cable room is equipped with an active fire suppression system such as sprinklers, it is assumed that the fire fails components and cables in one subsystem. If the suppression system is deemed satisfactory, it is assumed that it suppresses the fire so that it does not spread to another subsystem with 90 % probability. However, if the separation of the subsystems is deemed unsatisfactory, it is assumed that the fire spreads with 90 % probability. A fire simulation of a cable tunnel in the reactor building of OL1 and OL2 has been performed [3], which concluded that if the sprinkler system is activated, it prevented the fire from failing cables in the other divisions in all analyzed cases. If the sprinkler system did not activate, the fire failed or spread to the cables in the other division with 60 % probability.

In electrical rooms, it is assumed that a fire fails connections to components in one subdivision and if there is more than one subdivision in the room, which is the case for power systems supplying lower safety class components and relay rooms, the fire spreads with 10 % probability to the other subdivision, cf. Figure 2. Full scale fire experiments on electrical cabinets have been made [4], where it has been concluded, that fires in an electrical cabinet spread very slowly to adjacent cabinets if they are connected. Spreading to a cabinet separated by distance did not occur.

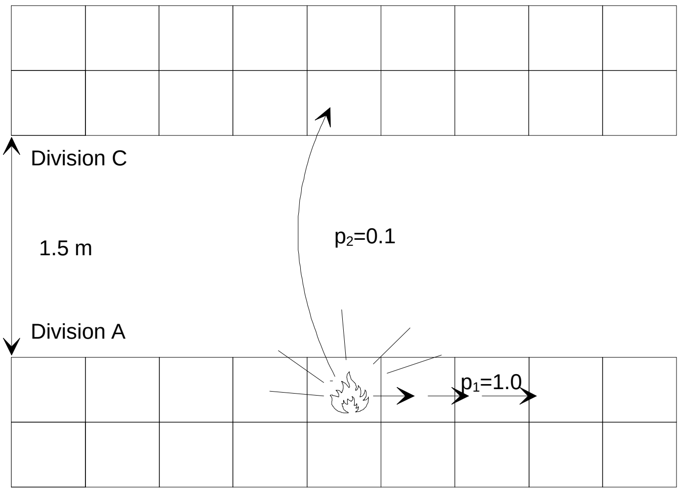

Figure 2 Modeling of spreading of a cabinet fire in a control room

The signals of the reactor protection system (RPS) are such that if power is lost, the RPS is launched. Also many solenoid operated valves (SOV) change state if power is lost. In these cases, a hot short in the cable is postulated if this results in a more unsafe state than if the power of the cable is lost.

Fire scenarios are grouped into fire initiating events based on the combination of failures of mitigating functions due to the fire. Thus, instead of modeling hundreds of fire scenarios, just a few tens of fire initiating events are modeled in the PRA model.

Fires affecting the safe shutdown of the reactor are modeled into the fire initiating events. Failure of the control rod drives, reactor scram system, emergency boron system and shutdown of the main recirculation pumps. The failure of these functions is modeled in the PRA model with a probability equal to the relative frequency of fire scenarios involving these failures in the initiating event group.

RESULTS

Changes in Fire Frequencies

The estimated total fire frequency of the unit increased by 10 % compared to the total fire frequency estimate of the unit using the old method. This discrepancy is entirely due to the fact that the new estimates include data from other plants as well. Further, the old method normalized the fire frequencies of the rooms so that the estimated total fire frequency equaled the historical fire frequency of OL1 and OL2. The room-specific fire frequencies changed for some rooms rather dramatically. This is also the case for some rooms with the highest fire frequency estimates. (cf. Figure 3)

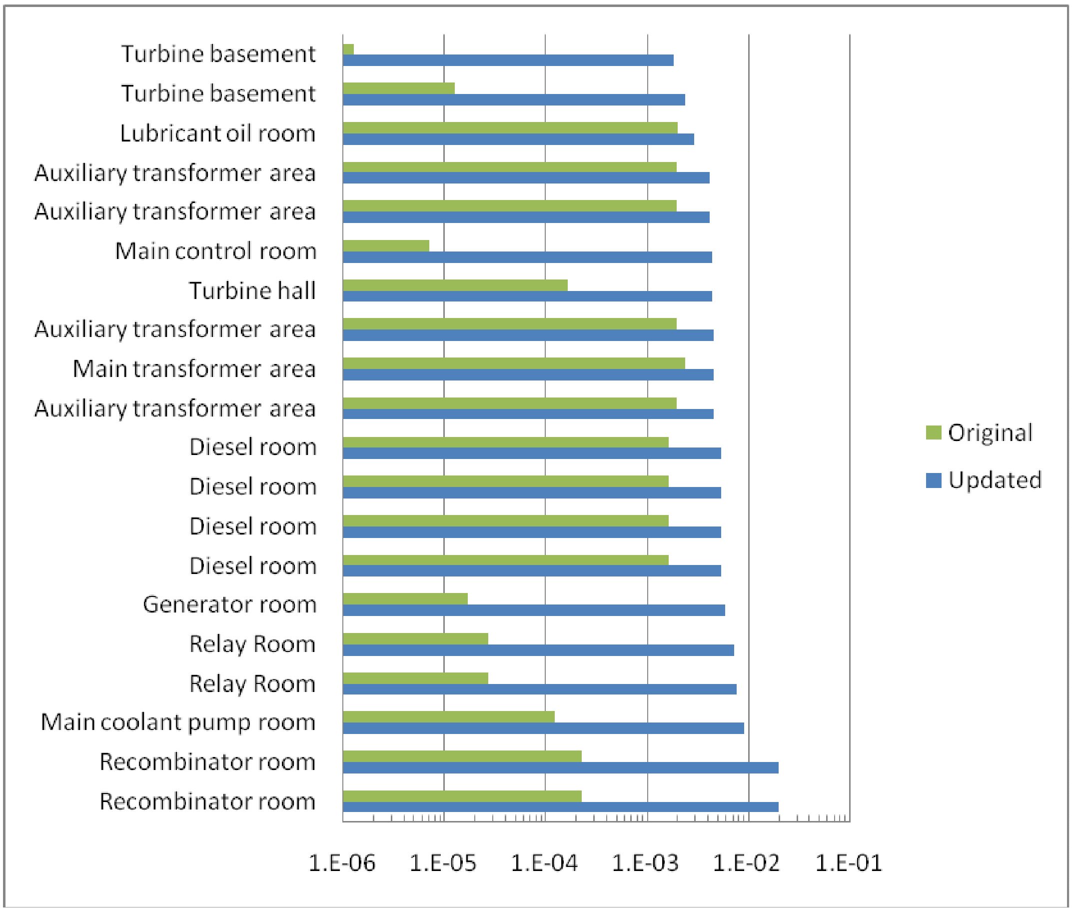

Figure 3 20 highest room-specific fire frequency estimates calculated with the updated method and comparison to the original estimates

The update of the estimated initial event frequencies increased the estimation of the core damage frequency by 18 %. The increase is, thus, significant, but not dramatic. In Figure 4 the rooms with the highest room-specific core damage frequency estimates are shown. The majority of the increase comes from the increased estimation of the core damage frequency of the relay room fires. The modeling of fires in relay rooms is very crude and conservative. The same statement is true for the main coolant pump room. Detailed modeling of these rooms would decrease the core damage frequency estimate significantly, in maximum by 8 %.

Table 3 and Table 4 show the fire initiating events modeled in the PRA model together with each fire initiating event's Birnbaum measure, which expresses the estimated core damage probability with the condition that the fire occurs in the extent that it has been assumed when modeling the fire initiating event. It is noticed, that fires just leading to a requirement to shut down the plant have a small impact on the CDF estimate. Fires leading to loss of feed water or off-site power, possibly together with failure of equipment belonging to one division, has a higher impact on the CDF estimate, still being reasonably small. If a fire leads to loss of feed water or off-site power together with loss of two divisions, the CDF estimate increases by an order of magnitude.

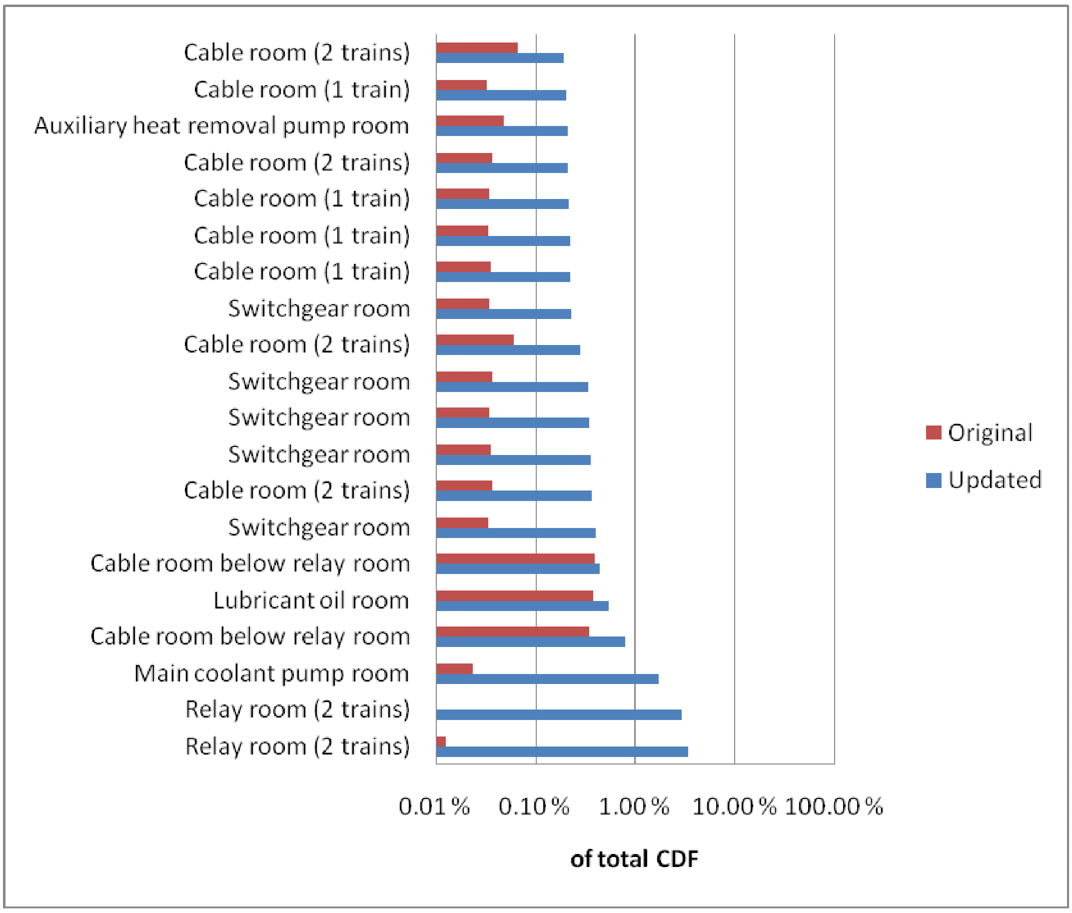

Figure 4 20 highest room-specific core damage frequency estimates resulting from using the updated fire frequency estimates and comparison the core damage frequencies using the original estimates

Table 3 Full-power fire initiating events modeled in the PRA model

Initiating Event	Description	Consequence	Birnbaum Measure
FIR-01/TP/A	Loss of containment spray, LPSI and HPSI systems, division A	Shutdown of reactor required by TechSpecs	5.70E-07
FIR-01/TP/B	Loss of containment spray, LPSI and HPSI systems, division B	Shutdown of reactor required by TechSpecs	5.74E-07
FIR-01/TP/C	Loss of containment spray, LPSI and HPSI systems, division C	Shutdown of reactor required by TechSpecs	5.71E-07
FIR-01/TP/D	Loss of containment spray, LPSI and HPSI systems, division D	Shutdown of reactor required by TechSpecs	5.72E-07
FIR-02/TP/AC	Loss of containment spray, LPSI and HPSI systems, divisions A and C and RHR system	Shutdown of reactor required by TechSpecs	5.22E-07
FIR-02/TP/BD	Loss of containment spray, LPSI and HPSI systems, divisions B and D and RHR system	Shutdown of reactor required by TechSpecs	5.74E-07
FIR-03/TF/AC	Loss of supply to divisions A and C	Loss of feed water transient	2.20E-04

Initiating Event	Description	Consequence	Birnbaum Measure
FIR-03/TF/BD	Loss of supply to divisions B and D	Loss of feed water transient	2.56E-04
FIR-04	Loss of HPSI system, one division	Not modeled	
FIR-06	Loss of LPSI and HPSI systems, one division	Not modeled	
FIR-07/TF	Loss of feed water, unrecoverable	Loss of feed water transient	4.54E-05
FIR-09/TF/AC	Loss of secondary cooling 712/721, divisions A and C and 713/723 leading to loss of RHR pumps	Loss of feed water transient	2.10E-04
FIR-10/TF/BD	Loss of secondary cooling 712/721, divisions B and D and 714	Loss of feed water transient	2.19E-04
FIR-11	Loss of one diesel generator	Not modeled	
FIR-12/TF/A	Loss of supply to division A	Loss of feed water transient	5.54E-05
FIR-12/TF/B	Loss of supply to division B	Loss of feed water transient	5.72E-05
FIR-12/TF/C	Loss of supply to division C	Loss of feed water transient	5.62E-05
FIR-12/TF/D	Loss of supply to division D	Loss of feed water transient	5.82E-05
FIR-13	Loss of two diesel generators	Not modeled	
FIR-14	Loss of one UPS	Not modeled	
FIR-15/TE	Loss of off-site power, unrecoverable	Loss of off-site power	1.11E-04
FIR-16	Loss of start-up transformer or connection to gas turbine unit	Not modeled	
FIR-17/TF/AC	Loss of 6.6 kV supply, divisions A and C	Loss of feed water transient	4.57E-05
FIR-17/TF/BD	Loss of 6.6 kV supply, divisions B and D	Loss of feed water transient	4.57E-05
FIR-18/TE/	Loss of the 400 kV grid	Loss of off-site power	2.77E-05
FIR-19	Loss of one 6.6 kV supply	Not modeled	
FIR-20	Loss of supply of two service water pumps of NI systems and two MFW pumps	Not modeled	
FIR-21/TF/A	Loss of feed water, containment spray, LPSI and HPSI systems, division A and RHR system	Loss of feed water transient	4.80E-05
FIR-21/TF/B	Loss of feed water, containment spray, LPSI and HPSI systems, division B and RHR system	Loss of feed water transient	4.93E-05

Initiating Event	Description	Consequence	Birnbaum Measure
FIR-21/TF/C	Loss of feed water, containment spray, LPSI and HPSI systems, division C and RHR system	Loss of feed water transient	4.79E-05
FIR-22/TF/AC	Loss of feed water, LPSI and HPSI systems, divisions A and C	Loss of feed water transient	6.45E-05
FIR-22/TF/BD	Loss of feed water, LPSI and HPSI systems, divisions B and D	Loss of feed water transient	6.87E-05
FIR-23/TF/AC	Loss of feed water, containment spray, LPSI and HPSI systems, divisions A and C and RHR system	Loss of feed water transient	2.15E-04
FIR-23/TF/BD	Loss of feed water, containment spray, LPSI and HPSI systems, divisions B and D and RHR system	Loss of feed water transient	2.52E-04
FIR-24/TF/AC	Loss of supply to divisions A and C and spurious opening of reactor PRV's	Loss of feed water transient	3.51E-04
FIR-24/TF/BD	Loss of supply to divisions B and D and spurious opening of reactor PRV's	Loss of feed water transient	3.84E-04
FIR-27/TF	Loss of feed water and RHR system	Loss of feed water transient	4.53E-05
FIR-31	Fire in oxygen-filled containment during startup or shutdown	Not modeled	5.70E-07

Table 4 Shut-down state fire initiating events modeled in the PRA model

Initiating event	Description	Consequence	Birnbaum Measure
FIS-03/T./AC	Loss of supply to divisions A and C	Loss of RHR systems	1.68E-05
FIS-03/T./BD	Loss of supply to divisions B and D	Loss of RHR systems	1.74E-05
FIS-09/T./AC	Loss of secondary cooling 712/721, divisions A and C and 713/723 leading to loss of RHR pumps	Loss of RHR systems	1.36E-05
FIS-10/T./BD	Loss of secondary cooling 712/721, divisions B and D and 714	Loss of RHR systems	1.55E-05
FIS-12/T./B	Loss of supply to division B	Loss of RHR systems	2.44E-09
FIS-12/T./C	Loss of supply to division C	Loss of RHR systems	4.76E-06
FIS-12/T./D	Loss of supply to division D	Loss of RHR systems	4.42E-06
FIS-32/T0/AC	Loss of feedwater, containment spray, LPSI and HPSI systems, divisions A and C and RHR system	Loss of RHR systems	2.04E-06
FIS-32/T0/BD	Loss of feedwater, containment spray, LPSI and HPSI systems, divisions B and D and RHR system	Loss of RHR systems	2.74E-06
FIS-33/T0/BD	Loss of containment spray, LPSI and HPSI systems, divisions B and D and RHR system	Loss of RHR systems	5.50E-08

Initiating event	Description	Consequence	Birnbaum Measure
FIS-34/T0/AC	Partial loss of the RHR system, divisions A and C	Loss of RHR systems	1.81E-08
FIS-34/T0/BD	Partial loss of the RHR system, divisions B and D	Loss of RHR systems	3.62E-08
FIS-35/T./AC	Loss of the RHR systems and AWF, divisions A and C	Loss of RHR systems	4.29E-06
FIS-35/T./BD	Loss of the RHR systems and AWF, divisions B and D	Loss of RHR systems	1.41E-05
FIS-36/T./	Loss of RHR system	Loss of RHR systems	1.41E-05
FIS-37/T3/B	Loss of reactor tank cooling system, division B	Loss of RHR systems	8.92E-08
FIS-37/T3/D	Loss of reactor tank cooling system, division D	Loss of RHR systems	8.62E-08

DISCUSSION

The resulting total fire frequency of the fire PRA model is 10 % higher than the historical fire frequency of the NPP units at Olkiluoto. This increase adds some conservativeness to the results. Since the number of components in US NPP units and the Olkiluoto NPP units differ, the fire frequencies should be reflecting this difference. However, no data on number of components in the source population is known, which would make it difficult to scale the fire frequencies. Furthermore, the total fire frequency in the model is already higher than the historical one, which supports the choice of being content with the results. Also the very definition of a fire event is has a great influence on the statistics.

It is clearly noticed, that the fire frequency in certain rooms changed by orders of magnitude due to the transition to the usage of component based ignition frequencies. Especially, the fire frequency of rooms housing components with exceptionally high ignition frequency, like main feed water pumps, large amount of electric equipment, like relay rooms, and rooms with hydrogen systems increased in the updated estimation. This change would be ascribed to the crude method used in the old implementation with just a few parameters governing the final room based fire frequency. The numbers of parameters were just too few and their bounds limited in order to take exceptionally high (or low) fire frequencies. However, also the choice of component groups will have an effect on the results, especially if components with low ignition frequency are joined with the same group as components with a high ignition frequency.

Further, it is noticed in Table 3, that fires isolated to one division has a small impact on reactor safety, whereas fires affecting two divisions increases the core damage frequency estimation by an order of magnitude. This can be seen when comparing the fire initiating event FIR-12/TF/.. with the fire initiating events FIR-03/TF/.. and FIR-24/TF/.. The importance of fire suppression systems in rooms, where the fire may fail two divisions is, therefore, high.

FUTURE WORK

The results of the main coolant pump room and relay rooms are very conservative due to very crude fire modeling. It is assumed, that a fire in one main coolant pump automatically spreads to the other three pumps, even though the pumps are protected by sprinkler systems, which effectively inhibit spreading of the fire to their vicinity. As for the relay rooms, the relay cabinets are grouped into rows consisting of relays belonging to one division. Divisions are separated by distance so that every second row belongs to the same division. The cabinet itself effectively delimit the fire to the cabinet, spreading very slowly to the adjacent cabinets. A fire in one cabinet will not spread to cabinets in an adjacent row. Since the fire load of each cabinet is low, the total fire power will be limited at any time. Therefore, the consequences of a fire would be very limited compared to the present assumption of failure of all safety systems in one division.

CONCLUSIONS

The probabilistic fire risk assessment (fire PRA) of the Olkiluoto 1 and 2 NPP units has been updated in the beginning of the year 2011. In the updated version, the ignition frequency of each component group is evaluated using a Bayesian approach and NUREG/CR-6850 data as a prior and the ignition frequency is divided on component basis.

The resulting total fire frequency in the PRA model of the units is 10 % higher than the historical fire frequency of the NPP unit. However, the fire frequency in certain rooms changed by orders of magnitude due to the transition to the usage of component based ignition frequencies. Especially, the fire frequency of rooms housing components with exceptionally high ignition frequency, like main feed water pumps, large amount of electric equipment, like relay rooms, rooms with hydrogen systems increased in the updated evaluation.

The update of the fire PRA increased the CDF by 16 %. After the update, the CDF originating from fire-related initiating events constitutes of 20 % of the total CDF. The fire PRA demonstrates that in a high-redundant NPP unit, such as the Olkiluoto 1 and 2 NPP units, a fire affecting safety systems in only one division has reasonably small consequences to reactor safety. A fire affecting two divisions increases the conditional core damage probability by an order of magnitude. This demonstrates the importance of adequate physical separation between redundant safety significant components and cables.

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DETERMINATION OF RELEVANT ROOMS FOR NUCLEAR POWER PLANT FIRE PSA AND THEIR CONTRIBUTION TO CORE DAMAGE FREQUENCIES FOR FIRE PROTECTION OPTIMIZATION

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ABSTRACT

Probabilistic Safety Assessment (PSA) is being used amongst other things as a decision tool for approval of operation by the regulatory authorities. The required detail and scope in performing a PSA has increased over previous years and will continue to increase in the future. The Fire PSA, as part of the Full Power Level 1 PSA can be prepared in more or less detail. The real benefits of a Fire PSA can only be gained by a very detailed analysis. On the other hand this requires much effort and work for the participating personnel. For each individual case it has to be decided how detailed the analysis has to be in order to achieve the desired results.

KEY WORDS

PSA, nuclear facility, fire, probabilistic analysis, risk analysis, data, probability, frequency

INTRODUCTION

Probabilistic Safety Analysis (PSA) In General

Probabilistic safety analysis (PSA) is used as a supplement to the safety assessment made on a deterministic basis to represent the influence of components, systems and human actions on the behavior of the plant in terms of safety. PSA can and must be performed for different types of initiating events (IE), for different operational conditions (full power as well as low power and shutdown modes) and within different areas (Level 1 to 3) [1].

Level 1 PSA identifies the sequences of events that can lead to core damage, estimates the core damage frequency (CDF) and provides insights into the strengths and weaknesses of the safety systems and procedures provided to prevent core damage. Level 2 PSA identifies the ways in which radioactive releases from the plant can occur and estimates their magnitude and frequency. Level 3 PSA estimates public health and other social risks such as contamination of land and food by radioactivity.

Probabilistic safety assessment up to Level 2 is part of the Periodic Safety Review (PSR, German: SÜ) [1] in Germany at the time being obligatory to be performed corresponding to the Atomic Energy Act [2]. A shutdown PSA is only required for Level 1 PSA. Fire PSA is part of Level 1 PSA for full power operation states.

An overview of the current state-of-the-art of science and technology in Germany is given in Table 1.

Table 1 Current state-of-the-art on probabilistic safety analysis according to the regulations in Germany

PSA Level / POS (Plant Operational State)	Level 1	Level 2	Level 3
Full Power	yes (incl. fire and external hazards PSA)	yes	no
Low Power and Shutdown	yes	no	no

The subject of this report adheres to Level 1 PSA for full power operational states. The results of Level 1 PSA are the input parameters for Level 2 PSA.

The aim of a PSA according to [1], [2], and [4] is:

- To evaluate the quantitative safety level,
- To determine the quality and quantity of potentially existing vulnerabilities of a plant, and
- To assess the balance of the safety related plant design to show that no initiating event contributes in a significantly enhanced manner to the total frequency of plant conditions beyond control.

DESCRIPTION OF LEVEL 1 FULL POWER PSA (INTERNAL EVENTS PSA)

The aim of Level 1 PSA is to

- Identify the potential event sequences that may lead to core damage,
- Estimate the frequency of all identified event sequences,
- Characterize the consequences of the accident sequences qualitatively by assigning each sequence to a core damage state.

Initiating Events

The starting point of the PSA is the identification of the set of initiating events which have the potential to lead to core damage, if additional failures of the safety systems should occur. There are two major types of accidents with the potential for core damage in LWRs: transient events and loss of coolant accidents (LOCA). Besides these internal events, the set of initiating events includes internal hazards such as fire, internal flooding, drop of heavy loads, etc.), and external hazards such as seismic events, external flooding, aircraft crash, explosion pressure wave, etc.), that can lead to initiating events.

The set of initiating events should be as complete as possible. Initiating events can be grouped if the required safety functions are identical. The set of the initiating events and their frequencies should be plant specifically determined. Reference spectra of initiating events are documented in the guidelines and have to be adapted to the examined plant.

The frequency of initiating events should be plant specifically determined; if generic values are used, the assign ability of these values has to be proven to the present case.

Event Sequence Analysis

The next step in the analysis is to determine the response of the plant to each of the groups of the identified initiating events. This requires the identification of the safety functions of operational and safety systems that need to perform as well as human actions for each initiating event. The success criteria for the safety functions shall be determined depending on the event sequence. Plant conditions which are not controlled by the safety systems shall be regarded as damage states. The essential results of the event tree analysis are the final conditions and their frequencies.

The headings of the event tree describe the various systems or functions that are required to control or mitigate the consequences of the accident. Each safety function which represents an event tree heading is performed by multiple systems or redundancies of one system.

The analysis then models the accident sequences which could occur following success or failure of the safety functions. The paths of an event tree lead to success states and to failure states, the more safety functions modeled means that more paths will result. A very simple example of the logic used in developing an event tree is shown in Figure 1.

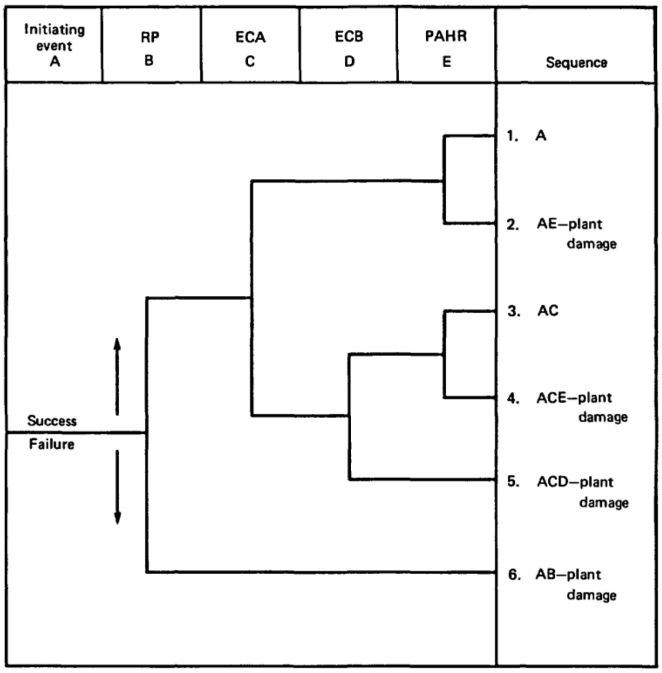

Figure 1 Simple Event Tree (LOCA), from [3]

Starting from the initiating event, the first requested safety function is being identified as branch point, where the event tree splits into two paths (branches) depending on if the required safety function is successful or not. Then the next requested safety functions are modeled in chronological sequence. The success criteria for the safety functions have to be analyzed according to the particular event tree. These criteria describe the minimum requirements to the safety systems to fulfill the safety function. They are expressed for example in terms of the number of trains of a redundant system.

The success criteria also specify the requirements for the support systems of the front line systems, e.g. electric power systems or component cooling systems. These support systems do not directly perform the required safety functions; however they could significantly contribute to the unavailability of a system when failing. Therefore, the support systems for each front-line system have also to be included in the analysis. This is done within the fault tree analysis.

Systems Analysis (Fault Tree Analysis)

The probabilities at the branching points in the event tree diagram, which stand for success or failure of the safety functions (represented in the event tree headings) have to be determined by the fault tree analysis. The top event of a fault tree is the system failure identified in the event tree analysis, that means the failure of the event tree safety function (heading). Success and failure criteria have to be identified for each event tree heading.

The fault trees extend the analysis down to the level of individual basic events which include component failures, component unavailability during periods of maintenance or testing, common cause failures and operator errors. The reliability parameters for the unique components should be gained plant specifically. The information for a collection of such parameters can be found in the operational records of the plant where all failures of components and the inspection results are documented. If the database is not sufficient generic data can be used; however, the applicability of such data has to be assessed.

The required safety functions can depend on human actions. The analysis of human actions comprises the identification, modeling and probabilistic assessment of actions by the operating personnel having an impact on event sequences. For the quantification of human error probabilities approved procedures should be used. In Figure 2 an example of a fault tree is shown. In this fault tree the component failures are expressed as basic events, the activation and the power supply of the both components are expressed as transfers to sub-fault trees. Note that the failure of the power supply is a common cause failure that affects both components and therefore can lead to the failure of the top event.

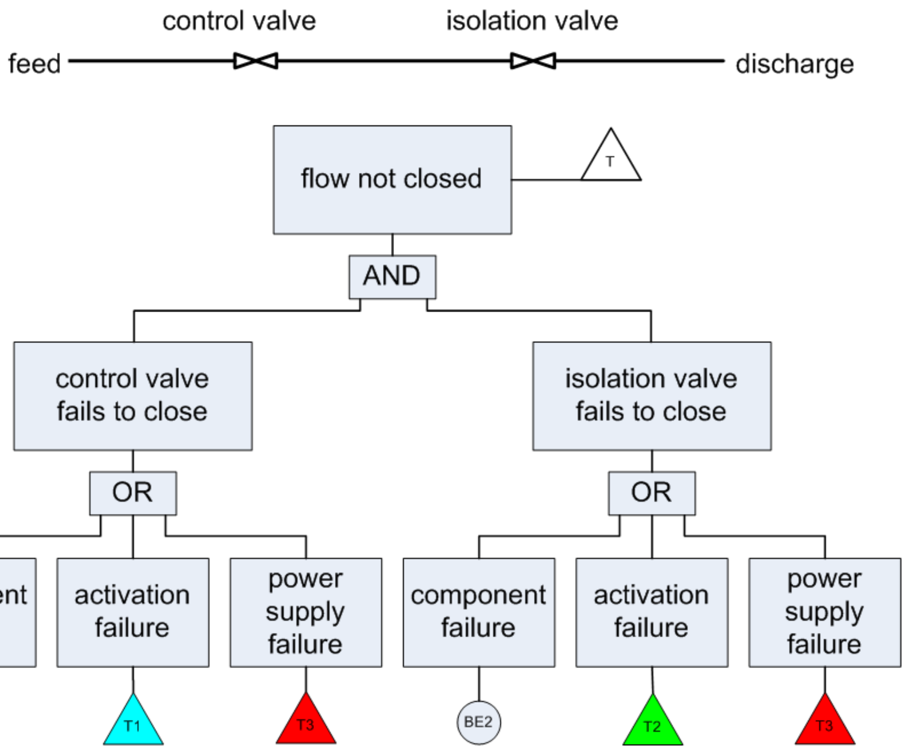

Figure 2 Fault tree, from [3]

Core Damage States (Consequences)

There are multiple final states (consequences) of the event tree diagram. These states have to be appropriately divided into success states, which can be controlled by the safety functions and core damage states. The core damage states may be characterized according to

- the general physical plant state,
- the possible availability of the safety systems which could prevent or mitigate releases,
- the initiating event.

Because the frequencies of initiating events and the reliability data of components and human actions are described by probability distributions, the uncertainties of the expectation values of the frequencies of core damage states have to be quantified in order to have a measure for the result uncertainties.

The results of the Level 1 Full Power PSA are the frequencies of event sequences which lead to core damage states.

PROCEDURE OF A FIRE PSA

The analysis of internal and external hazards comprises the identification of these hazards leading to internal initiating events such as transients or loss of coolant accidents. For a Fire PSA unique rooms have to be identified, for which a fire occurring in them can lead to an initiating event, e.g., a transient such as loss of main feed water due to the protective shutdown of a main feedwater pump because the level measurement of the main feed water tank being affected by the fire.

It is important to mention in this context that if only an initiating event is caused by a fire, the frequency of such an event is already covered by the consideration of internal initiating events in the Level 1 PSA for full power operational states. Only if there are additional fire induced failures of safety related components and/or equipment (including cables), which are required to control these internal initiating events, an additional significant contribute to the core damage frequency can be assumed.

On the other hand, failures in parts of the safety system due to a fire without causing an initiating event may cause a reactor trip in accordance with the operating manual. Because not all required safety equipment is available (e.g. one train of the residual heat removal system is out of order due to a fire) for the shutdown process, such scenarios should be regarded in the Fire PSA.

The failure of a safety related system can also occur due to the fire induced failure of a support system as described above in the fault tree analysis section, thus such systems are mandatory to be included in the analysis. Procedures on how to perform a Fire PSA are documented in several publications on an international level such as [3], [4], [5], as well as [1] and [6] on a national German basis. It depends on the particular requirements in which detail a Fire PSA has to be performed.

The following tasks describe a simplified sequence to perform a Fire PSA.

1. Plant Partitioning
2. Determination of fire frequencies
3. Review of initiating events from the internal events PSA
4. Inventory of components and cables of the fire compartments
5. Screening of compartments
6. Derivation of Initiating events to the fire compartments

7. Detailed fire modeling (fire specific event tree model)
8. Risk model (plant response model)

Note that the process of performing a fire PSA is an iterative procedure. That means that findings of a task may influence previous tasks.

Plant Partitioning

Step 1: Selection of the buildings relevant to the Fire PSA

The first step of a Fire PSA is to define those buildings being relevant to the Fire PSA analysis. Insignificant buildings will be screened out. All buildings that include components and cables identified in a subsequent task as relevant for Fire PSA have to be regarded. Buildings such as warehouses or office buildings can be disregarded, because there is no benefit of further partitioning such buildings.

The result of this step is the list of buildings that have to be considered for further analysis.

Step 2: Sub-division of the plant into fire compartments (Compartmentation)

The remaining buildings are sub-divided into fire compartments. These compartments will later be investigated with regard to their relevance for Fire PSA. A fire compartment is a room or group of rooms, which individually might not necessarily be separated by each other by qualified fire barriers.

The fire compartment itself is generally surrounded by fire barriers; therefore it is assumed that a fire will be substantially confined inside such fire compartments. If a fire compartment consists of several rooms, it might cover a complete building. A typical example of such a fire compartment is the reactor containment of NPP with BWR (boiling water reactor).

The fire compartment boundaries are typically formed by fire barriers, in more rare cases of fire sub-compartments, separation by space (distance) in large rooms with minimal fire loads or concrete walls even if these are not classified as fire barriers, may be sufficient. If there are openings in such barriers, e.g. (potentially or always) open doorways in the walls or various elevations are separated not by closed, qualified ceilings/floors but by gratings, it has to be decided for each individual case if this could be a partitioning feature or not. Therefore, no fire loads shall be present at the boundaries between fire compartments. For example unsealed cable penetrations are no qualified fire barrier elements and therefore should not be regarded as boundary of a fire compartment since unprotected cables represent a likely path for direct fire spreading. The boundaries of a fire compartment may nevertheless be open because of fire barrier elements (doors, hatches, fire dampers of the ventilation systems, etc., or unsealed penetrations) potentially being open. For these barrier elements their fire resistance rating has to be demonstrated so that it can be assumed that a fire will be confined within the fire compartment. In addition, the frequencies of these barriers being open have to be estimated within the Fire PSA.

The preferred criterion for defining fire compartments should be the typical use of fire areas inside a building, if such fire areas exist. The fire areas then can be partitioned into fire compartments. Ideally a fire compartment corresponds to an enclosed room or, as a fire sub-compartment, to an area separated by spatial separation. At the very least the compartments should be substantially fire resistant to contain the adverse effects of fires.

For the single fire compartment analysis the partial or total failure of the safety related components and cables is assumed; that means each fire compartment relevant to the Fire PSA will be individually analyzed and an individual core damage frequency (CDF) calculated. The sum of the individual compartment specific core damage frequencies of the entire fire compartments represents the overall contribution to the single fire compartment analysis. The collection of the before defined fire compartments should cover all areas of the previously identified buildings relevant to Fire PSA-

The spread of fire to adjacent compartments is principally possible due to the failure of any fire barrier element such as a fire door. However, the fire barriers, including all their active elements representing the boundaries between different fire compartments should ensure that the spread of fire is highly unlikely. The failure of these features is considered in the multi-compartment fire analysis. Within this analysis the spread of fire from one fire compartment to any other has to be estimated. The spreading of fires between compartments is limited to two fire compartments being affected in the most cases, however there may be exceptions where more than two compartments have to be regarded.

The result of this step is a list of the entity of fire compartments to be considered for further analysis.

Determination of Fire Frequencies

The occurrence frequency of fires should be determined on the basis of the operational experience. If possible plant specific data should be used, e.g. by examining the database of the plant fire brigade. If the plant specific database is not sufficient, generic data may be used, e.g., by using the international OECD FIRE Database [7] or other sources.

If the fire frequencies are determined for the entire buildings being considered in the Fire PSA, these frequencies have to be split to all fire compartments of the appropriate building. This can be done by weighting the fire compartments in matters of

- Frequency of presence of personal,
- Amount of mechanical equipment being present,
- Amount of electrical equipment being present,
- Probability of ignition (depending on flash point or ignition point),
- Distribution of fire loads (homogeneously, inhomogeneously, point source.

An approach on how to weight the fire compartments and how to gain the unique fire frequencies from the overall fire frequency of the building is described in [8].

Note that all non-relevant fire compartments of the buildings under investigation have to be evaluated as well. Extensive walkdowns are mandatory to gain the information needed for this task.

The results of this step are the fire frequencies for each fire compartment to be further considered in the analysis.

Review of Initiating Events

All initiating events (IE) from the internal events PSA have to be reviewed in order to determine whether they can be induced by a fire. Not only the thermal impact on a component itself, but also the fire induced failure of the power supply to the equipment and actuation signals for equipment operation from the main control room and the control cabinets may lead to an initiating event.

Some initiating events from the internal events PSA may be excluded due to low probability; however a fire may cause more severe faults than considered in the internal events PSA previously. Therefore new event sequences may need to be created or equivalent event sequences have to be adapted in a conservative manner.

The support systems for the systems that are involved in the normal operation of the reactor have to be included in the review of potential initiating events, because a fire induced failure of such systems may lead to an automatic or manual reactor trip.

The result of this step is a list of initiating events (IE) that need to be considered for the Fire PSA.

Inventory of Components and Cables in the Fire Compartments

All components (including cables) relevant for Fire PSA have to be identified. Components or equipment will be included in the Fire PSA if:

- Equipment which, if damaged by fire, will lead to a reactor trip (automatic scram or manual trip as specified in the procedures or other plant policies) or other initiating events,
- Equipment is required to respond to each of the initiating events identified (safety related equipment),
- Equipment whose spurious operation will adversely affect the response of systems or functions required to respond to a fire.

All cables and their equipment being required to support the operation of the selected equipment as well as cables whose failures could adversely affect credited systems and functions have also to be identified and mapped to the fire compartments. The following cables and circuits should be analyzed:

- Electric power supply circuits,
- Control power supply circuits,
- Instrumentation and control (I&C) circuits.

To decide if fire induced failures may lead to an initiating event requires extensive knowledge of the plant and analysis of system diagrams, arrangement plans, cable routing drawings of the operating facilities and limitation and protective equipment. If necessary, on-site inspections should be conducted in this context. The availability of a cable database is necessary to fulfill this task. Otherwise conservative assumptions have to be made, for example an overall covering scenario for a whole building.

It has to be decided, which initiating event has to be assigned to each fire compartment.

In the risk model all relevant components are considered within the accident sequences, so the equipment selection must occur in close coordination with the development of this risk model. All equipment identified including also all cables is then mapped to the fire compartments defined in step 1.

The result of this step is a list of all fire compartments (as defined in step 1) with the potentially affected components present (including cable routing).

Screening of Fire Compartments

It is intended to identify those fire compartments where the fire risk is expected to be relatively low or negligible compared to others. Fire compartments can be screened out, if:

- They do not contain any equipment relevant to PSA or their associated cables,

– A fire in such compartments does not lead to an automatic or manual reactor trip,
– The fire load inside the fire compartment is low (< 25 kWh/m²).

Note that all fire compartments are reconsidered as a part of the multi- compartment fire analysis, because they may cause potentially risk significant damage to equipment located in adjacent fire compartments.

In a second screening step, the further elimination of fire compartments remaining after the first screening step is possible for reducing the number of fire compartments that have to be analyzed in detail. Screening is performed on the basis of a simple, conservative estimate of core damage frequency for all remaining fire compartments. The core damage frequency is calculated with the internal events PSA model, where, as a pessimistic assumption, all assigned safety related components for a compartment are set to fail and no fire protection equipment is considered. All fire compartments that have a low contribution to the calculated overall core damage frequency can be screened out.

The result of this step is the list of fire compartments not screened out to be considered for further analysis.

Derivation of Initiating Events to the Fire Compartments

During the screening process all equipment that can lead to an initiating event is assigned to the fire compartments. Based on the knowledge of all potentially affected components in a fire compartment, the initiating events for each fire compartment can be derived.

The result of this step is the list of the fire compartments not screened out and their dedicated initiating events to be further considered in the analysis.

Detailed Fire Modeling

To calculate the fire induced frequencies of the initiating events, a fire specific event tree is used. There are two different types of fire scenarios, single compartment fires and multi-compartment fires. The single compartment scenarios cover fire scenarios where the ignition source and all the affected components are present in the same fire compartment while the multi-compartment scenarios cover those fire scenarios where it is postulated that a fire may spread from one compartment to another and damage components in multiple fire compartments.

Single compartment fires:

For each fire compartment not screened out the fire detection and suppression features and systems have to be identified. Event trees are used to calculate the frequency of the fire induced initiating events. These event trees have to be developed for the Fire PSA and adapted for each investigated fire compartment. An example of such a fire specific event tree is shown in Figure 3.

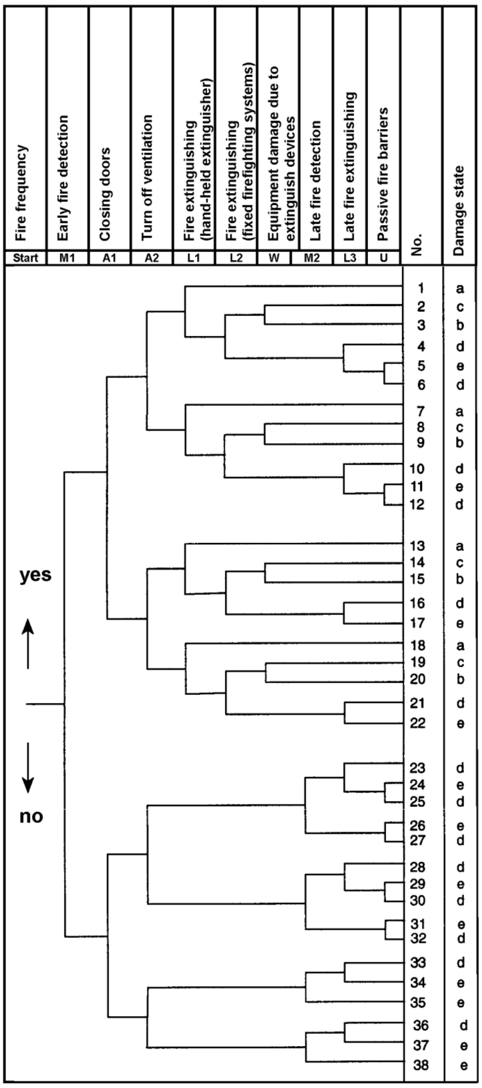

Figure 3: Example of a fire-specific event tree, from [6]

The following information should be considered for each fire compartment for the single compartment analysis:

- Fire detection ((by personnel, automatic detection through fire detectors in the compartment),
- Fire suppression due to lack of oxygen (closing of doors and ventilation),
- Fire extinguishing (by portable fire extinguishers, by stationary fire extinguishing systems).

For multi-compartment fires additional features have to be considered:

- Passive fire barriers (fire walls (including floors and ceilings) and their elements, such as fire doors or hatches, fire dampers, penetration seals, etc.) between different fire compartments,
- Fire detection (indirect detection in adjacent fire compartments),
- Fire extinguishing (in adjacent compartments).

An example of an event tree for multi-compartment fires is shown in Figure 4.

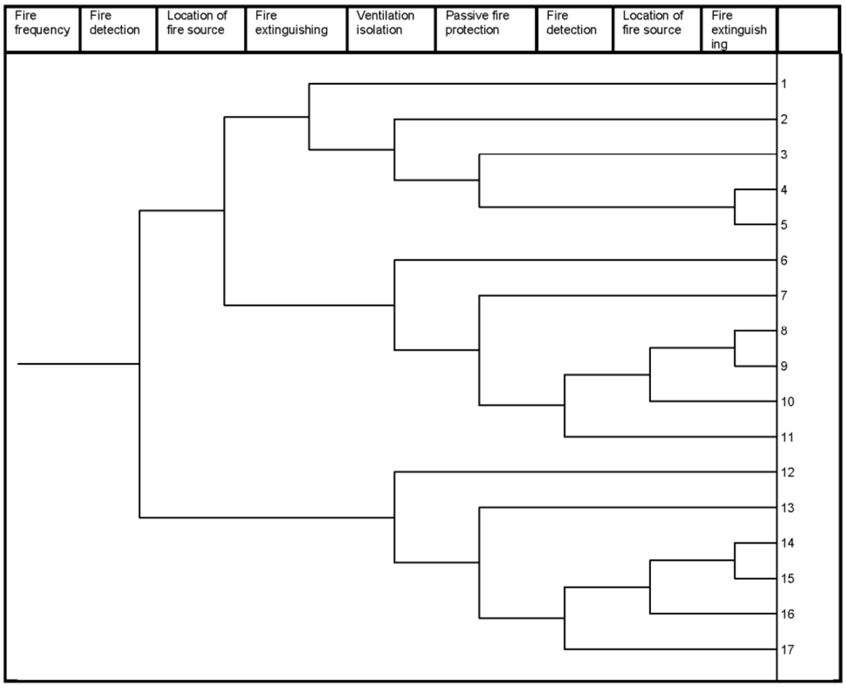

Figure 4: Example of an event tree for fires in adjacent compartments

This example of an event tree differs from the one shown in Figure 3. The reason is that there are different possibilities to develop and adapt the fire specific event tree. Although it does not matter how the event tree is developed, all important features have to be regarded.

The possibilities for fire detection, suppression, extinguishing and spreading have to be determined for each fire compartment according to the respective conditions, e.g., characteristics and number of fire detectors or extinguishing features present in a single fire compartment. The probabilities of the branching points of the headings of the fire specific event tree are calculated like the headings of the event trees of the internal events PSA. For every event tree heading there is a fault tree concerning all necessary information (e.g. fire detectors, fire extinguishing features, fire barriers). This information has to be gained for each individual fire compartment, either by using plant documentation and/or by walkdowns.

The probabilities of failure for the fire protection components and for manual actions have to be gained analogical to the proceedings for internal events PSA, which means preferred plant-specific data should be used.

The final states of the sequences of the event tree characterize the extent of damage. The following consequences can result from the sequences of the fire specific event tree:

- State a: Negligible damage (e.g. only one component; covered by the internal events PSA),
- State b: Partial damage,
- State c: Partial damage due to the extinguishing device,
- State d: Total loss of all equipment in the compartment,
- State e: Total loss of all equipment and spread of fire to adjacent compartment.

It may be useful to group the final states of the event tree, because it can be difficult to identify or define partial damage in a fire compartment.

The results of this step are the fire induced frequencies of the initiating events for each individual fire compartment and the entire multi fire compartment scenarios.

Risk Model (Plant Response Model)

To quantify the fire induced core damage frequencies the plant response model has to be created. This model is based on the internal events PSA model, as described above, which is then modified according to the particular fire compartment. The internal events PSA contain the information on those systems and components whose failure in response to an initiating event may lead to an undesired consequence. The following equipment has to be regarded:

– Safety related frontline and support systems,
– Non-safety related systems, whose failure can lead to an initiating event (e.g., main feed water, offsite power).

Only those initiating events that can occur from a fire need to be considered in the fire induced risk model. Unique additional equipment (like spurious actuation) or operator actions being not addressed in the internal events PSA need to be included to the model. For each fire compartment all the information is available through the tasks realized before to calculate the core damage frequency:

- The assigned initiating event for each compartment (including the event tree from the internal events PSA),
- The fire induced frequency of this initiating event for the given compartment,
- All relevant, respectively damaged equipment (according to partial or total damage) inside the compartment.

All relevant equipment affected by a fire in a compartment is set to “unavailable” in the analysis, thus these components are not considered to fulfill their safety function in order to control the initiating event. All redundant equipment which is not affected by the fire will keep its stochastic probability of failure as modeled in the internal events PSA.

For multi-compartment scenarios the initiating event with the higher requirements to the safety functions has to be chosen if for the adjacent compartments different initiating events were assigned. For all adjacent fire compartments all required information is available:

- The initiating event covering as worst case all other initiating events for the compartments adjacent to the initial fire compartment,
- The fire induced frequency of this initiating event for the compartment for which the fire is estimated,
- All relevant, respectively damaged equipment (according to partial or total damage) inside the both compartments.

The result of this step is the fire induced core damage frequency for each fire compartment not screened out before and for adjacent fire compartments not screened out.

With these results the initial fire compartments as well as the adjacent one with the highest contributions to the core damage frequency (CDF) are identified. To draw conclusions of these results implies to analyze if enhancements to the fire protection equipment can decrease the risk contributions of individual single fire compartments and adjacent fire compartments. The benefit is that the effects of potential enhancements to the risk contribution can be calculated in advance. If the contemplated enhancements have a significant influence on the risk contribution, they should be implemented to decrease the risk contributions of such fire compartments and therefore reduce the risk to the whole plant.

CONCLUSION

The preparation of a Fire PSA within a Level 1 full power PSA is current state-of-the-art of science and technology. The detailed approach outlined before can possibly differ depending on the level of detail of the available database of the plant, in particular the information on cable routing. If there is a lack of information in the database, conservative assumptions have to be applied.

A simple, conservative analysis can only be sufficient to evaluate the adherence to regulatory requirements, but the real benefit of a Fire PSA is to determine the risk contributions of every single room of the entire plant in order to identify potential enhancements to the fire protection features.

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Recent Advances in High Energy Arc Faults (HEAF) at Nuclear Power Plants

12th SMiRT Pre-Conference Seminar on
Fire Safety in Nuclear Power Plants and Installations
TUV SUD, Munchen, Germany
14 September 2011

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Definition of HEAF

- High-energy arcing faults (HEAF):
  - energetic electrical equipment faults
  - rapid release of electrical energy
  - rapid release of heat and vaporized metal
  - high pressure increases due to high current arcs
  - created between energized electrical components or between an energized electrical conductor and neutral or ground

HEAF Characteristics

- Characterized by an Arc Blast (as opposed to an Arc Flash) involving a sustained electrical arc followed by fire
- Complex phenomena
- Not well understood

- Components may include:

  switchgear, load centers, bus bars/ducts, motor control centers, disconnect switches, transformers, cables, etc.

Canadian Nuclear Safety Commission

Operating Experience

- Approximately 12% of fire events in the OECD Fire Event Database are identified as HEAF related
- Approximately 30 % of HEAF events result in the loss of one or more safety trains
- Approximately 55 % of HEAF events result in a change of plant operating state
- Current HEAF rule based model (NUREG-6850) is:
  - based upon one well documented event
  - under predicting damage to safety trains
- Rare but potentially consequence significant events

Canadian Nuclear Safety Commission

International Project

- International Project initiated in 2009 by Organization for Economic Co-operation and Development Nuclear Energy Agency Committee on the Safety of Nuclear Installations (OECD/NEA/CSNI)
- Mandate:
  Investigate HEAF events to better understand fire risks at nuclear power plants by an international group, pooling knowledge and research means.

Project Objectives

- Collect, disseminate and analyse HEAF event data
- Contribute to the international knowledge base on HEAFs
- Characterize and define in technical terms HEAF phenomena
- Investigate and document HEAF phenomena (electrical, chemical and physical elements)
- Perform research and testing
- Develop HEAF models

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Project Activities

- Data collection and dissemination of HEAF events and operational experience
- 46 events detailed in the OECD Fire Event Database
- Draft test plan developed
- Currently collecting equipment donations from participating countries
- Literature reviews performed

Research and Testing

- Proposal to perform testing of electrical equipment donated by member countries
- Tests to be performed at a USA facility
- Test and documentation costs funded by the USNRC (USA)

Future Plans

- Conduct tests on collected equipment
- Analyse test data
- Develop simple predictive models:
  - to identify potential HEAF scenarios
  - for damage zones around equipment
- Develop accepted input data and boundary conditions for more complex modeling (i.e. CFD)

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HIGH ENERGY ARCING FAULTS (HEAF) – UPDATE OF THE GERMAN OPERATING EXPERIENCE

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ABSTRACT

The operating experience of nuclear installations worldwide has provided a reasonable number of high energy arcing fault (HEAF) events characterized by a rapid release of energy resulting in explosive failures of the affected components with the potential of consequential fires. These events typically occur at high voltage electrical components such as switchgears and circuit breakers, or at high voltage cables.

Such electric arcs have led in some events internationally observed to partly significant consequences to the environment of these components exceeding typical fire effects. In-depth investigations have indicated failures due to the rapid pressure increase of those fire barriers and fire protection features not designed against such impacts.

Due to the high safety significance and importance to nuclear authorities OECD/NEA/CSNI has initiated an international activity on "HEAF" in 2009 for analyzing these phenomena in nuclear power plants in more detail for a better understanding of the fire risk due to this type of incidents accomplished by an international experts group to pool international knowledge and research means.

One input into this OECD project is an in-depth analysis of the German operating experience with HEAF in nuclear power plants based on a questionnaire for collection of the necessary data and information on these events.

After having analyzed the first events from the German database on reportable events at nuclear power plants the investigations have meanwhile been completed providing on the one hand insights on some typical HEAF phenomena and, on the other hand, the need for specific experiments to be carried out at equipment where HEAF typically arise.

INTRODUCTION

The operating experience of nuclear installations worldwide has provided a reasonable number of high energy arcing fault (HEAF) events resulting in explosive failures of the affected components with the potential of (partly very rapid) fires. As defined on an international basis within a specific task group "HEAF" of OECD Nuclear Energy Agency (NEA), high energy arcing faults (HEAF) are energetic or explosive electrical equipment faults characterized by a rapid release of energy in the form of heat, light, vaporized metal and pressure increase due to high current arcs between energized electrical conductors or between energized electrical components and neutral or ground. These events typically take place in high voltage electrical components such as switchgears and circuit breakers, or they occur at high voltage cables. HEAF events may also result in projectiles being ejected from the electrical component or cabinet of origin and result in fire.

In a first step, the national German database on reportable events [1] as well as the international databases for reporting incidents from nuclear installations, IRS (Incident Reporting System) and INES (international nuclear event scale), have been searched for HEAF events. The systematic query gave indications (see also [2] and [3]) that a reasonable number of reportable events with explosions and rapid due to high energy arcing faults (HEAF) have under some circumstances resulted in significant consequences to the environment of impacted components with the potential of endangering nuclear safety. In-depth investigations of these events have also identified failures of fire barriers and of a variety of fire protection features (such as fire doors, dampers, penetration seals, and the barriers themselves) due to pressure build-up and/or pressure waves.

As a result of these indications from the operating experience worldwide and first research results, an international activity has been started by OECD Nuclear Energy Agency (NEA) CSNI (Committee on the Safety of Nuclear installations) Working Group IAGE in 2009 for preparing a state-of-the-art report on HEAF of electrical components and equipment based on the operating experience of the partners in this project. More details on this activity are provided in [4].

After having analyzed the first events from the German database on reportable events at nuclear power plants [1] in 2009 (see also [2] and [3]), the investigations with respect to the German nuclear power plant operating experience have meanwhile been completed providing on the one hand insights in some typical HEAF phenomena and, on the other hand, indications on the need for experimental research to be carried out at equipment where HEAF typically arise. In the following, the German operating experience is summarized.

INSIGHTS FROM THE OPERATING EXPERIENCE WITH RESPECT TO HEAF EVENTS AT NUCLEAR POWER PLANTS IN GERMANY

As a result of analyzing the international event databases IRS and INES as mentioned before, a questionnaire has been developed covering a list of questions mainly to be answered by the licensees of nuclear power plants. Major goal of this query is to gain insights on the basic HEAF phenomena and to make possible the evaluation of such events and the identification of preventive measures in the future.

This questionnaire has been developed under the lead of experts from Gesellschaft für Anlagen- und Reaktorsicherheit mbH (GRS) and from Germanischer Lloyd Bautechnik GmbH (GL) with the aim to collect all the information and data needed for a meaningful analysis of the operating experience at nuclear stations and as a prerequisite for assessing the significance of HEAF events in probabilistic risk analysis. The corresponding analysis of the licensees' response has been done based on this query.

The insights of these investigations will also be generically processed and the feedback from the national German operating experience will be forwarded to the licensing and supervisory authorities, to the German licensees, and to the member of the OECD/NEA activity on HEAF to be used in the state-of-the-art report probably to be published in 2012.

Update of the Feedback from the German Operating Experience on HEAF at Nuclear Power Plants

The results of searching the German database on reportable events at nuclear installations [1] for HEAF events provided – based on the most recent definition of HEAF provided by the international experts in the frame of the OECD/NEA task on HEAF the results presented in Table 1, containing – in particular – the current plant state in case of the event, the component where the HEAF started, the voltage level of the HEAF component (if only the impacted component was damaged) and if existing fire barriers had been deteriorated or damaged.

From this table it can be concluded that different components were impacted, in particular switchgears and circuit breakers, as expected. In some cases it was not possible to identify the voltage level in case of the HEAF occurrence. In the majority of the events, the damage was limited to the component where the HEAF itself occurred; a fire barrier was deteriorated only in case of the HEAF events listed, and only 11 events were correlated to a fire.

Table 1 Operating experience with respect to reportable HEAF events from German NPP (from [1])

Year of Occurrence	Reactor Type	Plant State	Component	Voltage Level	Damage Limited to Component	Fire Barrier Deteriorated	Fire and/or Explosion
2009	BWR	FP	transformer	400 kV	yes	no	-
2008	PWR	FP	circuit breaker	0.66 kV	yes	no	F
2007	BWR	FP	transformer	400 kV	yes	no	E / F
2007	PWR	FP	transformer	400 kV	yes	no	-
2006	BWR	LP/SD	auxiliary service pump	0.40 kV	yes	no	-
2006	BWR	FP	switchgear drawer	0.66 kV	yes	no	-
2006	BWR	FP	switchgear drawer	0.66 kV	yes	no	-
2005	BWR	FP	circuit breaker	6 kV	yes	no	-
2004	PWR	LP/SD	emergency power feed line	6 kV	yes	no	-
2004	BWR	FP	cable connection	10 kV	yes	no	F
2004	BWR	LP/SD	Diesel generator. exciter	unknown	yes	no	-
2003	BWR	FP	Diesel generator. exciter	unknown	yes	no	-
2003	PWR	FP	emergency power feed line	0.5 kV	yes	no	-
2002	BWR	FP	emergency power busbar	0.5 kV	**no**	no	F
2001	PWR	FP	generator transformer switch	400 kV	yes	no	-
2001	BWR	FP	emergency power distribution	0.66 kV	yes	no	-
1999	PWR	FP	ventilation exhaust	unknown	yes	no	-
1998	PWR	FP	emergency power distribution	0.66 kV	yes	no	-
1996	BWR	FP	switch drawer	0.5 kV	yes	no	F
1995	BWR	FP	switchgear drawer	unknown	yes	no	-

Year of Occurrence	**Reactor Type**	**Plant State**	**Component**	**Voltage Level**	**Damage Limited to Component**	**Fire Barrier Deteriorated**	**Fire and/or Explosion**
1993	PWR	FP	currency converter	0.38 kV	yes	no	-
1992	PWR	LP/SD	emergency power generator	unknown	yes	no	F
1991	BWR	FP	emergency power busbar	10 kV	yes	no	-
1989	PWR	FP	switchgear feed cell	10 kV	**no**	no	F
1989	PWR	LP/SD	switchgear feed area	0.38 kV	no	no	F
1988	PWR	LP/SD	switchgear	220 kV	**no**	no	E / F
1987	BWR	FP	emergency diesel generator	unknown	yes	no	-
1987	PWR	FP	auxiliary service water system	unknown	yes	no	-
1986	PWR	LP/SD	busbar	0.38 kV	**no**	no	F
1984	BWR	LP/SD	auxiliary power supply	unknown	yes	no	-
1981	PWR	FP	safety injection pump motor	unknown	yes	no	-
1979	BWR	LP/SD	switchgear	0.4 kV	yes	no	-
1979	PWR	LP/SD	control rod distribution	unknown	yes	no	F
1978	PWR	FP	switchgear	220 kV	yes	no	-
1977	PWR	LP/SD	switchgear	0.35 kV	yes	no	-
1977	BWR	LP/SD	emergency switchgear	unknown	yes	no	-

Abbreviations:

PWR: pressurized water reactor; BWR: boiling water reactor; FP: full power
LP/SD: low power / shutdown; E: explosion; F: fire

Results of the In-depth Investigations on HEAF Events at German Nuclear Power Plants

The German Questionnaire [5] covers questions directly with respect to HEAF events occurred at nuclear installations as well as questions referring to HEAF phenomena without explicit observations from events having occurred.

The questions regarding HEAF events focus on the operating experience itself including the type and size of damage, the components and plant areas affected the detection and/or identification of the HEAF and its duration, but also on the direct as well as indirect effects of the HEAF. This also includes potential consequences to nuclear safety. In case of a consequential fire the performance of fire protection means should be outlined. In addition, the licensees should provide ass far as possible information on the event causes and corrective actions taken in the affected plant.

The more general questions without observations from HEAF events occurred on site concern preventive measures taken in the plant against HEAF and the consideration of HEAF events and their potential effects in the frame of periodic safety reviews (deterministic safety status analyses as well as probabilistic risk assessment).

After the already well known more significant HEAF events presented in [2] having occurred inside a cable channel underground and at a main transformer, the German operating experienced has revealed further HEAF events, which fortunately had only limited consequences and no direct effects on the plant safety, but nevertheless the potential of impairing nuclear safety under different boundary conditions.

One event occurred at a 400 kV transformer of a Konvoi type PWR in 2007 during full power. In the area of the 400 kV electrical lead-off area, a short to ground occurred in one phase due to an electric arc. The arc induced short to ground was caused was caused by harsh weather conditions during a big storm. The short to ground was stopped by the electrical fuse for grid protection. This resulted in isolation/separation of the nuclear plant from the 400 kV external grid and an auxiliary power changeover.

The HEAF event was detected by spurious signal in the main control room. The HEAF itself was limited to the transformer area where it occurred (lead-off area) and did fortunately neither cause harm to nuclear safety nor result in a fire.

Another HEAF event, this accompanied by fire, occurred in 2008 at a PWR plant. The plant was under full power conditions. After hooking up a high pressure (HP) transfer pump from the main control room fire detectors in the corresponding switching panel of the 660 V switchgear were actuated automatically. It was not possible to switch off the pump from the main control room; therefore the corresponding busbar was switched off.

A high energetic arc occurred at the circuit breaker of the pump in the switchgear building (switchboard room) due to incorrect position of the switching contacts points. The root caused could not be identified with 100 % confidence, but it is assumed with high confidence that foreign particle impingement in the circuit breaker was the original cause.

A smoldering fire occurred as a result of the HEAF being detected in due time by the automatic detectors. The fire, which could be directly confirmed, was successfully suppressed by the on-site fire brigade by a portable fire extinguisher with CO_2 as extinguishing medium within approx.15 min.

Last not least, in 2009 another HEAF event occurred at a small oil filled transformer in a German BWR plant. A short circuit occurred at a 400 kV generator transformer located outside in the yard next to the turbine building resulting in an automatic reactor scram. Up to now, the root cause has not yet been identified. Due to the high energy release with a rapid pressure increase in the transformer vessel oil was released in the area of the flanges; however there was fortunately no ignition. In case of ignition the event potentially might have impaired the plant safety.

The HEAF was immediately recognized and identified at the main control room through faulty signals arising. The event was limited to the transformer and did neither result in fire or explosion nor to a deterioration of fire protection features.

GENERAL INSIGHTS

The operating experience with HEAF in German nuclear power plants has revealed strong indications that only few components are typically endangered to experience a HEAF with the potential of explosion and/or fire, resulting impacts on fire protection features, mainly due to the strong and rapid pressure increase, or even endanger nuclear safety.

Typically, there are specific areas and only few high voltage components affected by HEAF as outlined in Table 2 for an exemplary reference plant.

Table 2 Example for potential NPP areas with typical HEAF components

Component / Plant Area	Voltage Level		
	10 kV	***6 kV***	***0.4 kV***
Reactor Building	2		
Switchgear Building	2		4
Turbine Building	2		
Emergency Diesel Building		3	
Cooling Water Pump Station	2	2	
Transformer Switchyard	3 / 11	6	

CONCLUSIONS AND OUTLOOK

The in-depth investigations of the German operating experience with HEAF in nuclear power plants based on a query to the NPP licensees has provided several insights on the type of equipment, where such events typically occur, and the corresponding voltage levels the components affected are operated on.

HEAF as defined by the international OECD/NEA task group "HEAF" mainly occur at switchgears and circuit breakers as well as on high voltage cables. Another type of equipment showing similar behavior is high and medium voltage transformers, most of them oil filled ones, of different sizes.

Most of the equipment affected is operated on voltage levels of 0.4 kV, 6 kV and 10 kV.

All 36 HEAF events having occurred at and reported so far from German nuclear stations have been detected and identified by faulty or spurious signals of electric equipment indicating a malfunction. In case of heavy smoke arising, the events were additionally been detected by the automatic fire detection systems

Due to complex in-depth fire protection concepts being realized in all German NPP with a very high level of separating redundant safety trains the effects were always limited to only one train, if they had occurred in safety related areas. There were no relevant effects observed as a result of the explosions.

For two of in total five HEAF events with consequential fire, smoke propagation from the fire compartment or fire area to other compartments/buildings occurred

In the case of four of the events presented in this paper fire fighting was necessary for the HEAF induced fire, in three of these events the fire could be directly successfully suppressed by a portable extinguisher only. Only in one case, several attacks were needed.

Although it has been observed on an international basis that HEAF events may seriously deteriorate or even destroy firer barriers and other fire protection features either by heat and/or smoke impact or by the rapid and strong pressure rise, no such effects occurred in the German plants.

The root causes, although not always identified with 100 % confidence, were mainly technical reasons, often in combination with human failures. The more recent events have also provided strong indication that ageing of the typical HEAF related components, e.g. of the cables or the transformer windings, may play an important and increase the frequency of HEAF events.

One important result of the analysis of the German operating experience with high energy arcing faults (HEAF) is that the following prevention measures are essential:

– Quick detection and identification of the event occurrence and its location and reaction to these occurrences by the operator installing features for electric arc detection, mainly via pressure sensors in connection with overcurrent monitoring (500 ms on 100 ms);
– Timely detection/identification of slowly proceeding damages by oil monitoring and periodic inspections of insulation resistances;
– Prevention of a relevant pressure build-up by installation of pressure relief openings
– Consideration of deterioration aspects by replacement of (ageing) components such as transformers, control units, cables.

All this has already been recognized by the regulators and the NPP operators in Germany so that a variety of adequate provisions has meanwhile been taken by the German NPP licensees.

Fortunately, none of the events having occurred in German nuclear installed so far did jeopardize the plant safety. However, it is well recognized that such type of events always has the potential to result in explosions and/or fires which could impair nuclear safety or which could lead to deterioration of fire protection features essential to protect equipment of the redundant safety trains.

Therefore, the German experts see a strong need for further in-depth investigations of HEAF phenomena and to develop, based on experimental research with regard to typical HEAF components, a mechanistic model to account for the potential failure modes and consequence portions of high energy arcing faults. This should also support a better characterization of high energy arcing faults in the probabilistic risk assessment of fires.

It is therefore intended by the German experts trying to support the HEAF testing program of the United States Nuclear regulatory Commission (NRC) Office of Research to be carried out in the frame of an OECD/NEA Project by providing typical high voltage equipment to be tested. The German licensees of nuclear power plants have already been contacted and seem also be interested in the testing program and its results.

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FIRST EXPERIENCES FROM INTERNATIONAL DATABASES ON NUCLEAR POWER PLANT FIRE BRIGADE ACTIVITIES

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ABSTRACT

With regard to fire detection and suppression the availability of an adequate amount of suitable fire detectors and appropriate manual fire fighting capabilities is essential.

Special training cases for the on-site plant fire brigade are arranged at a nuclear power plant. During the training, timing of different steps of the fire brigade's operation during the attack is recorded to identify the time needed for access to the fire compartment, starting from a fire alarm.

First results of the evaluation of international databases, in particular the international database OECD FIRE, show that only a minority of fires was suppressed by automatically actuated fixed extinguishing systems. For most of the events manual fire fighting means were involved in the successful fire suppression, in some cases nuclear power plant personnel was assisted by an external fire brigade. As another result of the analysis, a large majority of the fires could be confirmed within a very short time period (less than one minute).

The event sequences at foreign nuclear power plants (NPP) can be partially used for comparison purposes in a general level to understand possible differences in plant fire brigade operation and in communication between operators and fire brigade. Anyway, comparison is only possible on a very general level, e.g., to identify the time period from departure from the fire station up to the start of the fire fighting action.

INTRODUCTION

An important aspect in the determination of fire induced core damage frequency (CDF) is the ability of the on-site plant fire brigade to respond to and extinguish fires in a timely manner before damage can occur to plant systems and components important to safety. Mostly, simplified methodological approaches have been applied for modeling fire brigade response, which have utilized either plant specific fire drill data or credited only manual fire suppression in continuously occupied plant areas such as the main control room.

First experiences from international databases on fire brigade response times and fire fighting activities are provided in the next paragraphs.

RESULTS OF AN EVALUATION OF THE OECD FIRE DATABASE

One nuclear specific international fire event database is the OECD FIRE Database [1] by the OECD Nuclear Energy Agency (NEA). For the evaluation presented in the following the version 2010:1 containing in total 373 fire events from nuclear power plants up to the end of 2009 is applied.

In the past, several specific evaluations using the FIRE Database have been performed on national level, e.g., regarding transformer fires (see [1] and [2]).

In the following, the OECD FIRE Database has been evaluated with respect to information on self-extinguishing fires and on fire events where (plant-internal and external) fire brigade activities were needed for extinguishing the fire successfully. In total, 367 fire events have been investigated; all the others had no safety significance.

Figure 1 provides the number of self-extinguishing fires and their duration. In total, 57 fires were self-extinguished. In most cases the duration of the fires is unknown. A comparable number of fires were self-extinguished within 30 min. However, 12 fires lasted longer than 30 min, two of them even more than one hour.

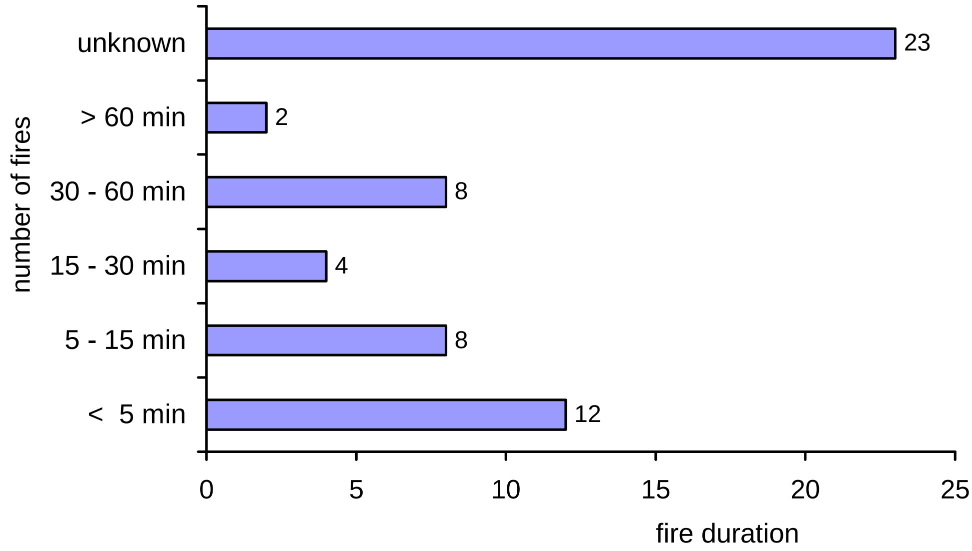

Figure 1 Number of self-extinguishing fires and their duration

Figure 2 provides information on the typical time periods of confirming those fires which had been actively suppressed by the fire brigade.

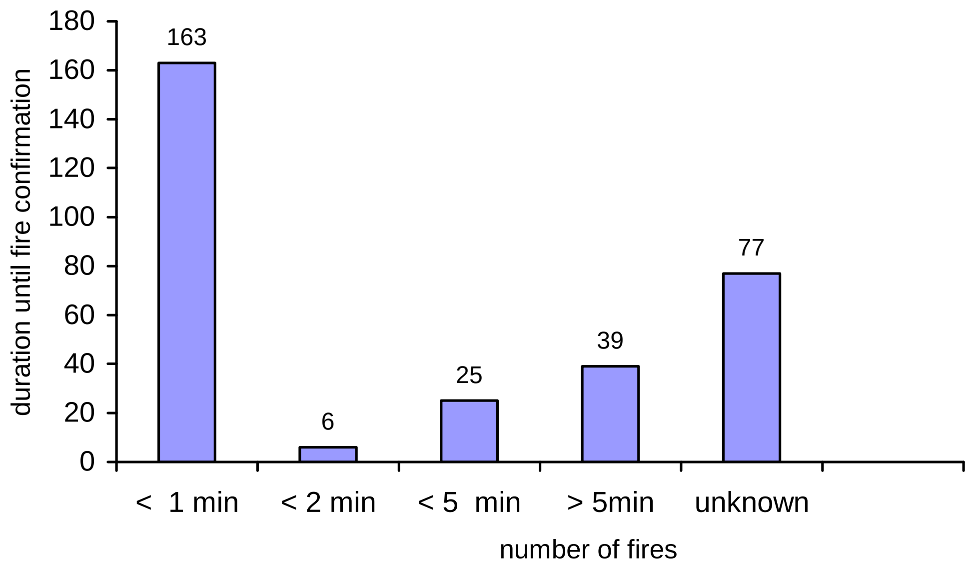

Figure 2 Time period until confirmation of fires which had to be actively extinguished

In general, confirmation time is defined as the time period between the first alarm and the verification of the occurrence of a fire and the identification of its location. However, in some countries information is only provided for the time between the first alarm and the final (second) confirmation of the fire.

From the FIRE Database the observation was made that 53 % of the fires were confirmed within one minute after the alarm, e.g. because plant personnel was present or more than one automatically actuated fire detectors sent alarm signals. On the other hand, 9 % of the fires needed more than five minutes to be confirmed. For a relative high number of event records the time to confirmation has not been provided.

Figure 3 shows the time to fire suppression of those 310 fires being actively extinguished.

The information of the coded field "duration time" has been directly used only for this evaluation, except in those cases where the information was not clear to the database applicant. Fire duration is defined as the time period from the fire alarm until the time of successful fire suppression.

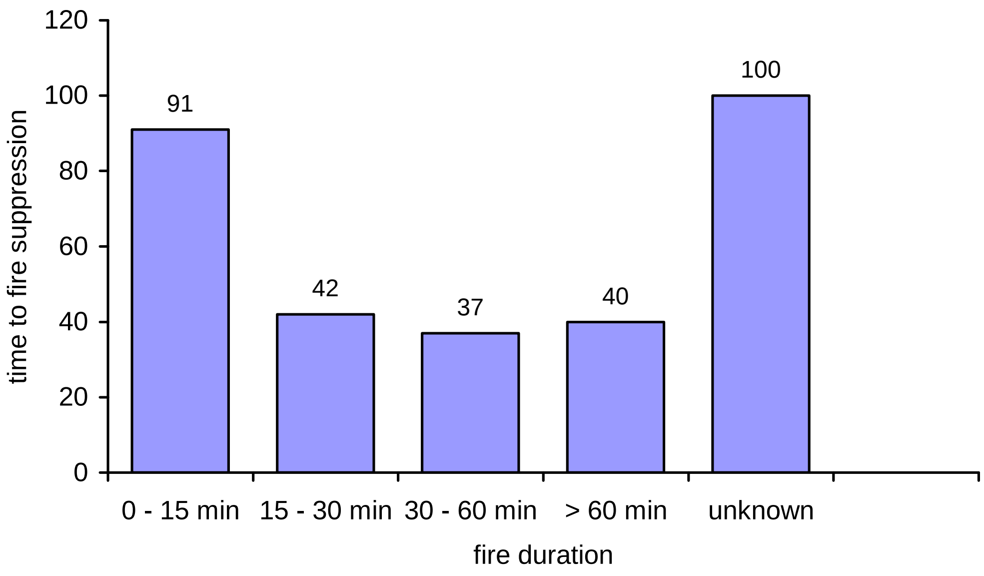

Figure 3 Time to fire suppression for fires actively extinguished

There is a steady decrease of the time to suppression for more recent fire events. Only 13 % of the fires needed suppression times exceeding one hour. Many of these were transformer fires that needed long extinguishing/cooling times because of the large amounts of hot metal. In total, 49 of these fires were events at transformers representing an amount of 13.1 % of all the 310 fires. Unfortunately, the number of reports with unknown suppression time is too large for reliable and meaningful indications from the statistics.

Figure 4 shows the time period between confirmation of the fire and the beginning of the fire extinguishing activities for the 310 fire events which were not self-extinguished.

In about 45 % of the fires, the time was lower than 15 min. In five cases the time between fire confirmation and the beginning of the fire extinguishing activities was longer than 120 min or even longer than 180 min.

Reliable information on the time period between fire being confirmed and start of fire fighting information was only available for 198 events, for 112 ones representing more than one third of the fire events with fire suppression this type of data is not available. In particular, for some events information regarding the start of fire extinguishing is missing. In these cases, the time when the fire was extinguished has been used in the statistics.

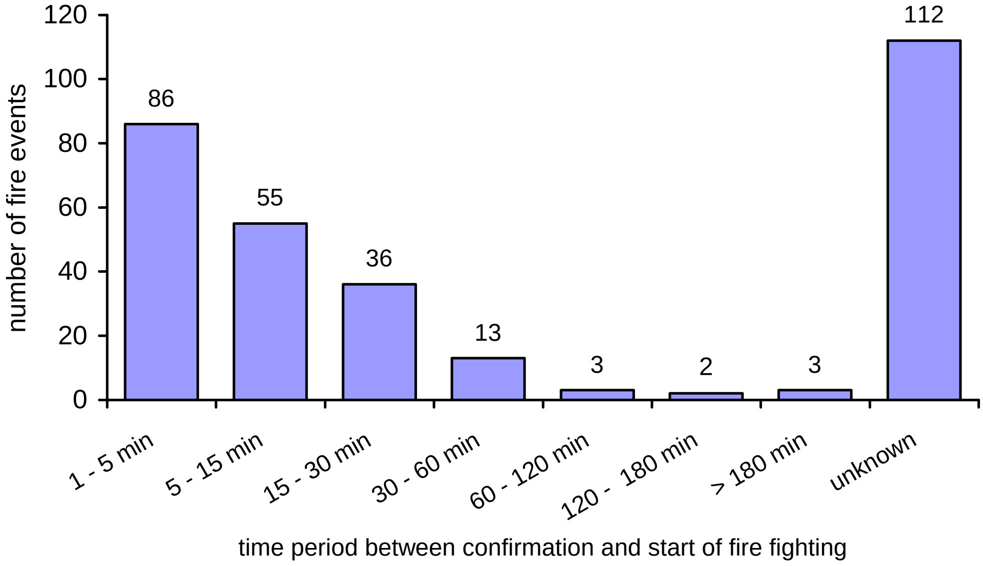

Figure 4 Time period between fire confirmation and start of fire extinguishing

Figure 5 explains who extinguished the fire. About 70 % of the fires were extinguished by the on-site plant fire brigade, by shift personnel, and other people available in the fire area, most of these by manual fire fighting alone, and a few by a combination of manual fire fighting and fixed fire extinguishing systems. In 13 % of the events not being self-extinguished plant personnel was assisted by a plant external fire brigade.

Only 3 % of the fire events (mainly transformer fires) were suppressed solely by stationary fire extinguishing systems being automatically actuated by the fire detection systems.

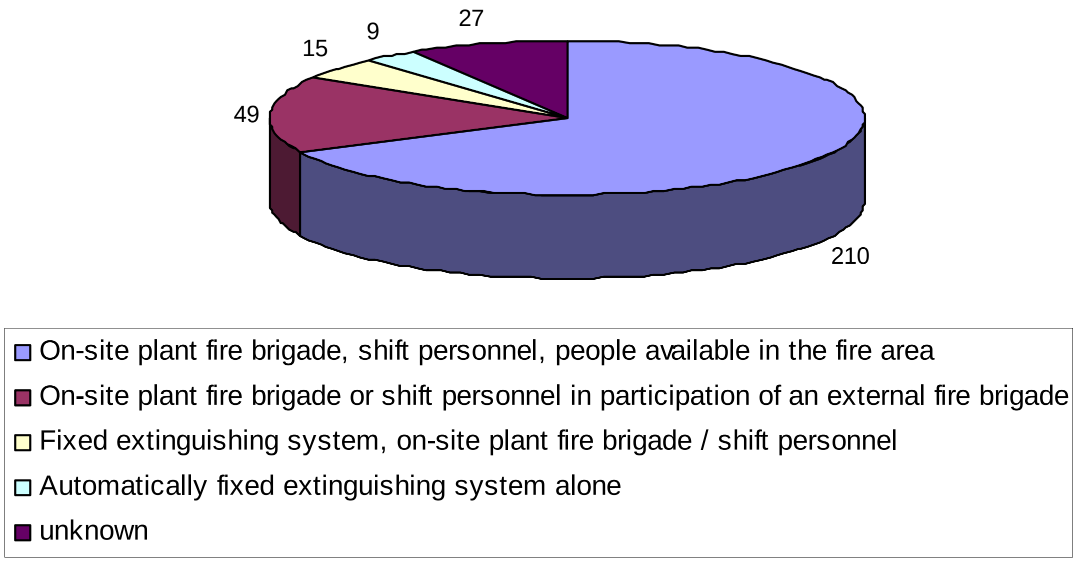

Figure 5 Who extinguished the fire

In this context, it is important to mention that it was necessary for the evaluation performed not only to search in the coded fields, but also to carefully analyze the narrative fields with the description of the event and the event sequence providing the main source of information. Otherwise, the results would be different.

Moreover, the quality of the information available in the fire event records differs. In particular for the use of the database in probabilistic fire risk assessments this fact has to be taken into account.

The quantitative information will, of course, increase with the increasing number of reported events and a careful description of the respective events to provide as much information as available.

RESULTS OF AN INVESTIGATION OF THE U.S. FIRE ADMINISTRATION

The U.S. Fire Administration has performed an investigation of the structure fire response times, not focused on nuclear or other industrial installations [3].

The definition of "response time" depends on the perspective from which one approaches the data. In the fire service, "total" response time is typically measured from the time a call is received by the emergency communications center to the arrival of the first apparatus on the scene.

Response time constituents include ignition, combustion, discovery, 911 activation, call processing and dispatch time, turnout time, drive time, setup time, "vertical" response, combat, and time until the fire is extinguished (see Figure 6).

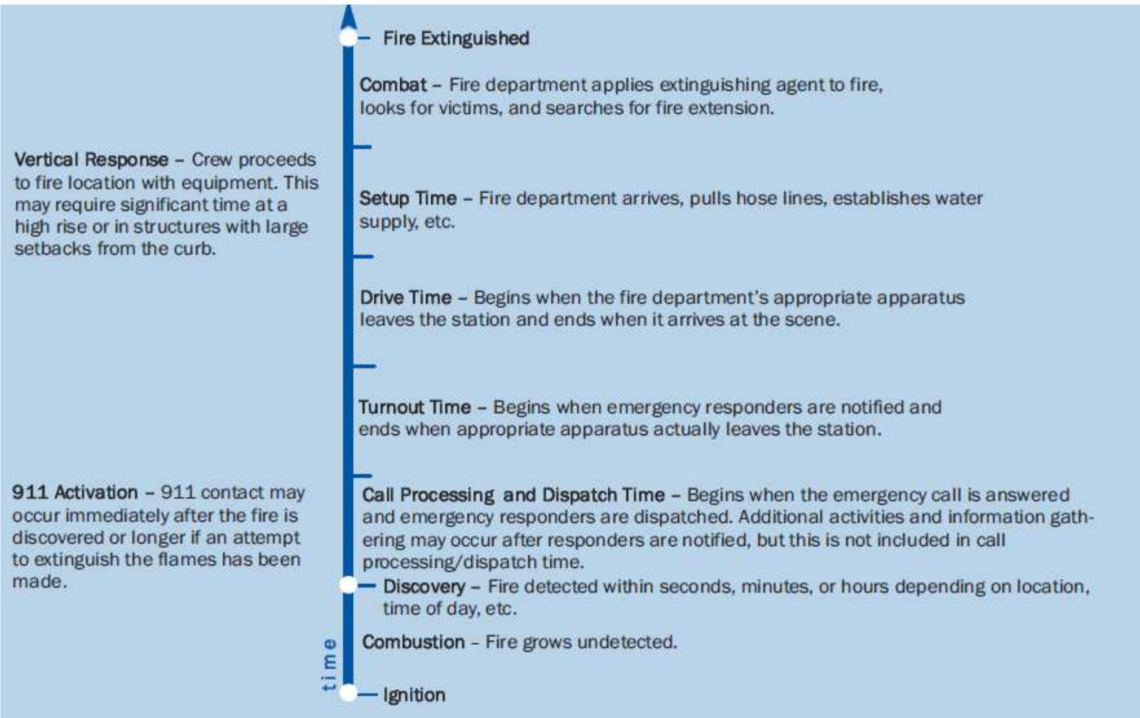

Figure 6 Components of total response time

The data for the study of the U.S. Fire Administration were queried in whole minutes. This means that response times of exactly 4 min and those up to 4 min and 59 s are all included in the 4 min category.

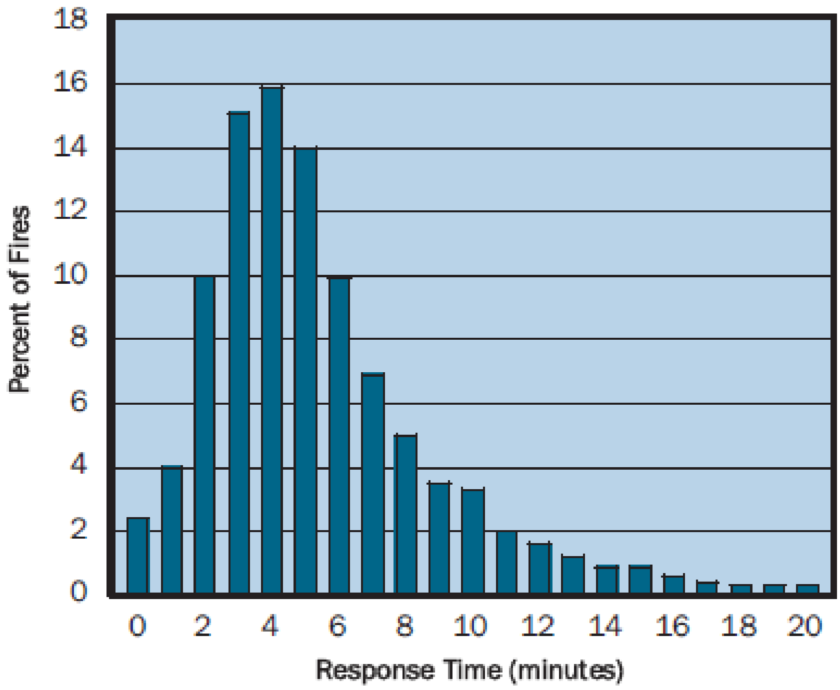

Figure 7 Fire response time according to [3]

Because the vast majority of response times are 20 min or less (98.7 %), the charts and graphs in this paper do not reflect response times of more than 20 min.

As shown in Figure 7, the highest percentage (16 %) of structure fires had a response time in the range of 4 min. The percentage of structure fires with response times of 3 and 5 min were not far below at 15 % and 14 %, respectively. Overall, 61 % of structure fires in the time period under consideration had a response time of less than 6 min.

In most of the analyses done here, response times of nearly 50 % of the fires were less than 5 min and less than 8 min for about 75 % of the events. Nationally, average response times were generally less than 8 min.

The overall 90th percentile, a value of very high confidence often cited in the industry, was less than 11 min.

USE OF KRSKO NUCLEAR POWER PLANT SPECIFIC DATA TO MODEL FIRE BRIGADE RESPONSE

At the Krsko nuclear power plant, the on-site fire brigade consists of five members, three of these being are professional fire fighters and two local operators [4]. The off-site fire brigade is comprised of nine members, at least three of them being professional fire fighters. In a fire incident, the on-site brigade members are officially notified of a fire condition by an alarm panel located in the fire brigade station and, in parallel, one in the main control room.

Upon receiving an alarm, the shift supervisor dispatches a local operator to the suspected fire location to determine fire conditions. The on-site fire brigade immediately begins to suit up and gather equipment. After assessment of the fire condition, the local operator notifies the shift supervisor of the fire location and the fire brigade response needed. The shift supervisor then notifies the on-site and off-site fire brigades to respond.

If the fire can be easily extinguished, the local operator will then extinguish the fire, otherwise the operator waits for the arrival of the fire brigade. This twofold action minimizes the time in the assessment of fire conditions and establishment of the fire location.

For off-site fire brigade personnel, to reduce delay time in response, personnel are accompanied by a security escort or by a qualified escort from their time of arrival at the plant until their exit.

Currently, for those fire PSA in which fire brigade response has been modeled, drill times were taken to be equivalent to the time to detect, respond, and extinguish the fire. However, drill times typically only record the time for the fire brigade to respond to the scene of the fire after actuation of a fire alarm. Therefore, the time to extinguishment may be underestimated. This general approach has been utilized in the fire PSA for the Krsko nuclear power plant to model the probability of manual fire suppression.

A thorough walk-through of the Krsko NPP and review of its fire brigade practices was performed to determine the probability of manual suppression in a given time frame for all critical plant areas. Information gathered was used to determine the time to reach the discrete fire phases of established burning to suppression.

Interview of plant fire department personnel, a comprehensive tour of the fire department facilities, and a review of drill records were performed to assess fire department response time, distances to fire locations, training, and equipment provisions.

Table 1 provides an example of the results of the analysis for the fire brigade manual fire fighting response to postulated fire incidents at the Krsko nuclear power plant. The table presents minimum, average, and maximum times for each fire phase.

Table 1 Krsko manual fire suppression upper cable spreading room as described in [4]

Event/Phase/Description	Minimum [min]	Maximum [min]	Average [min]
1. Detection	1.0	5.0	2.0
2. Alarm	1.0	5.5	2.0
3. Fire Brigade Building Response	3.0	9.5	4.5
4. Arrival at the Room of Origin*	7.0	17.5	10.5
5. Finding the Fire	7.5	18.5	11.0
6. Agent Application	8.0	19.5	11.5
7. Extinguishment	9.0	21.5	12.5

* A two-minute delay in arrival at the room of origin was assessed due to the potential for smoke transport into the emergency switchgear rooms and consequent multiple alarms in the control room and on-site fire brigade station.

Results of this analysis have been compared to and found to be consistent with earlier fire data [4].

Crediting fire brigade response and suppression before critical damage occurs on a consistent plant-wide basis has allowed for removal of unnecessary conservatisms in the analysis and avoidance of potential skewing of fire area importance (if manual fire suppression is credited in only limited plant areas).

APPROACH FOR FIRE DEPARTMENT RESPONSE TIME MODELING

In order to provide a framework for describing a fire department's suppression activities, the events that occur from the start of a fire until it is extinguished are considered.

The sequence of events, which is shown in Figure 8, starts with fire ignition. After some time the fire is detected (by a person or by an automatic device) and it is reported to the fire department.

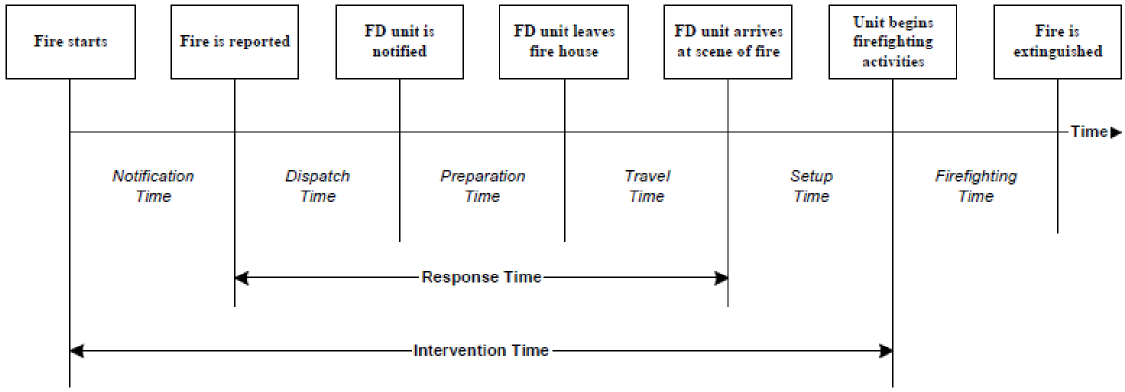

Figure 8 Typical sequence of events in a fire from fire ignition to fire being successful extinguished

One or more dispatchers at the fire department process the alarm and then fire services are notified to respond. Equipped fire fighters respond to the fire scene where they operate to extinguish the fire. The effectiveness of the fire department in minimizing loss of life and property depends in part on the elapsed time between ignition and intervention by the fire department.

In this model [5], the response of the fire department to a fire incident is analyzed in detail. All the steps taken are being evaluated in order to determine the time needed to carry out a number of activities that add to the response time. These events are the following:

- ***Notification time:*** Once a fire has started in a building it will eventually be detected either by automatic means or by the building occupants, and the fire department will be notified. The time elapsed from fire ignition to fire department notification is called the notification time and it is calculated by either the so-called occupant response model or the detection model [5].
- ***Response time:*** This is the elapsed time from the moment the fire department is notified until such time when the first fire unit arrives at the scene. As indicated in Figure 8 response time includes:
  - *Dispatch time:* the time between the receipt of an alarm and the dispatch of a unit (notifying it to respond) by the dispatcher.
  - *Preparation time:* the time required for the dispatched unit to prepare and be ready to leave the fire department.
  - *Travel time:* the time required by the first unit to travel from the fire department to the scene of the fire.
- ***Setup time:*** the time required to setup and prepare equipment, vehicles and assemble fire fighting teams just before any fire fighting and rescue operations begin. The model does not compute this time, but it is given as input based on statistical values.
- ***Fire fighting time:*** the time required by fire fighters to extinguish the fire. It depends on the status of fire on arrival, the type of building and fuel loading, the resources, equipment and men on the site.

CONCLUSIONS

With regard to fire detection and suppression the availability of an adequate amount of suitable fire detectors and appropriate manual fire fighting capabilities is essential.

Fire service emergency response to fire is based on the fact that the earlier the fire is attacked the smaller will be the consequences to people and property. This investigation considers the statistical relationship between fire service response and the effects of the fire.

Special training cases for the plant fire brigade are arranged at a nuclear power plant. During the training, timing of different steps of the fire brigade's operation during the attack is recorded to identify the time needed for the access to the fire compartment, starting from a fire alarm.

First results of the evaluation of international databases show that only a minority of fires has been suppressed solely by automatically actuated stationary extinguishing systems. For most of the events manual fire fighting means were involved in the successful fire suppression, in some cases nuclear power plant personnel was assisted by an external fire brigade. A large majority of the fires could be confirmed within a very short time period (minutes).

The evaluation of the OECD FIRE Database underlines the importance of the on-site fire brigade and other plant personnel trained in fire fighting.

The event sequences at foreign nuclear power plants can be partially used for comparison purposes with regard to the situation in the country under investigation in a general way to understand potential differences in plant fire brigade operation and in communication between operators and fire brigade. Anyway, comparison is only possible on a general level, e.g. to identify the time from departure from the fire station to the start of the fire fighting actions.

A comparison of the nuclear situation with non-nuclear fire brigade activities and fire brigade response times is even more difficult. In most of the analyses done in that area, response times were less than 5 min for nearly 50 % of the fires investigated and less than 8 min about for 75 % of the events.

In this context, it has to be taking into account that response time in these studies means arrival at the area where the fire takes place and not the beginning of the fire extinguishing activities as in the evaluation of the nuclear fire data base. Thus, time needed for external fire brigades to arrive at the fire location is nearly the same in both cases.

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PREDICTING INDUSTRIAL FIRE BRIGADE TACTICAL FIRE FLOW RATES

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ABSTRACT

A methodology for estimating the required suppression agent flow rates for typical fire hazards encountered by industrial fire brigades at nuclear power plants is proposed. The methodology addresses both fuel controlled and ventilation controlled fires over a range of ventilation conditions typical of a nuclear power plant. Prediction of fire suppression agent application rates, called the "Tactical Fire Flow Rate" (TFFR) is approached by evaluating the fire scenario and selecting one of three TFFR approaches based upon the influence the compartment boundary has on the fire dynamics within the enclosure.

For rooms with the potential to proceed to flashover, TFFR for water is based upon the floor surface area of the enclosure. For compartments where the flashover potential is precluded by fire hazard or construction, the suppression agent application rate for water is related to the maximum heat release rate of the fire hazard within the compartment. For applications where the compartment construction and arrangement does not substantially affect the fire growth rate the TFFR is based upon the exposed fuel surface area. A trial application of the methodology is performed for a CANDU nuclear power plant.

INTRODUCTION

Canadian nuclear fire protection requirements [1] require an industrial fire brigade (IFB), commonly called emergency response teams, at Canadian nuclear power plants (NPP) to be available at all times (i.e. 24 hours per day, 365 days per year), sufficiently staffed and sufficiently equipped to meet two response criteria:

- to be capable of effectively responding to and mitigating all fire hazards at the licensed facility. Assistance by offsite fire departments or other resources may be credited, and
- to be capable of protecting safety related areas of the licensed facility without offsite assistance.

The minimum staffing level of the IFB (which form part of the minimum shift compliment), onsite equipment and offsite resources credited to satisfy the above two fire response criteria are required to be documented for acceptance by the regulatory authorities. No specific methodology for determining minimum IFB staffing levels is prescribed by Canadian requirements [1] or endorsed by the Canadian regulatory authority, the Canadian Nuclear Safety Commission (CNSC) or its staff.

Determination of the minimum staffing levels to satisfy the response criteria above is typically approached via a systematic review of fire hazards and response risks at the NPP by subject matter experts combined with qualitative or quantitative assessments of suppression agent application requirements. Suppression agent application requirements on a hazard by hazard basis (usually expressed in terms of application rates such as liters per second) are evaluated for the most demanding application rate and used to establish minimum resource requirements for that application rate. The relationship between agent application rate and IFB staffing levels are beyond the scope of this paper which is focused on estimating agent application requirements only.

Although a number of methods have been employed by Canadian licensees to assess minimum staffing levels, this paper reviews pervious work in estimating minimum agent application rates on a hazard or by compartment geometry basis and proposes methodologies applicable to construction, hazards and operational arrangements at a typical CANDU NPP. A typical CANDU NPP unit is evaluated using the methodology recommended in this paper and the results are evaluated against analysis performed previously by expert elicitation for comparison and qualitative validation of the proposed methodology.

NOMENCLATURE

A:	area [m^2]
A_{fs}:	fuel surface area [m^2]
A_{floor}:	floor surface area of a compartment
A_t:	total surface area of a compartment including walls, floor and ceiling
E:	energy [MJ]
F:	flow rate in [l/s]
FLED:	fire load energy density [MJ/m^2]
K_f:	heating efficiency of an enclosure fire (conservatively estimated at 0.5 for most compartment conditions)
K_w :	cooling efficiency of available agent (conservatively 0.5 for a water delivered via manual hose streams)
m_{air}:	mass flow rate of air
Q_{max}:	maximum (peak) heat output of fire [MW]
Q_w :	absorptive capacity of water at 100 C = 2.6 MW/l/s
TFFR:	Tactical Fire Flow Rate [l/s]
V:	volume [m^3]

PROBLEM DESCRIPTION

Suppression agent application requirements, typically in terms of application rates or total agent volute, are used as part of a more complex systematic assessment of fire hazards and fire response risks at CANDU NPPs to support the determination of minimum fire response requirements (staffing and resources). A methodology for estimating the required application rate or total volume of agent on a hazard by hazard basis is therefore required. Although critical application rates, the minim rate of agent required to suppress a fire for a specific fuel determined through testing, are documented in literature there are only general correlations available for mixed fuels typically encountered in real enclosure fires. Accurately calculating suppression agent requirements can be difficult due to the limited accuracy of available correlations and large variability in the efficiency and effectiveness of the manual fire fighting process.

METHODOLOGY

To determination the required agent application rate, called the Tactical Fire Flow Rate (TFFR) in this paper, a literature review was performed to identify current methodologies and their limitations. The identified methodologies were then evaluated for applicability to NPP construction and fire hazards.

In reference [2] Grant, Brenton, and Drysdale provide a comprehensive summary of experimentally supported work in the area of determining critical water application rates for a number of fuels under varying environmental conditions. One of the main conclusions of their review is that for scenarios where the enclosure does not significantly affect the fire dynamics fuel surface area (A_{fs}) governs suppression agent application rates and that for suppression to occur by manual hose streams using water, sweep time of the agent stream is required to be less than the re-flash time of the fuel. For two dimensional flammable liquid fires of depth these two concepts were previously well established by testing [3], [4] and operational experience now forming the basis of agent application rates codified by commonly used engineering standards [5]. The relationship with exposed fuel surface area and TFFR is supported by the work of Sardqvist and Holmstedt in [6]. Where the enclosure does not substantially affect the combustion dynamics, such as in a very large room, it is recommended that suppression agent application rates for free burning fuel in the open be used.

Based on reference [2] and [6], it can be established a bonding water application rate of 0.3 $l/s/m^2$ of <u>fuel surface area</u> based upon live burn scenarios and operational experience. And for flammable liquid hazards with a typical AFFF foam proportioned at 3 % by volume (current design basis at Canadian nuclear power plants) results in an application rate of 0.0678 $l/s/m^2$ (0.1 $usgpm/ft^2$).

Pre-Flashover Enclosure Fires

Barnett in [7] developed a calculation method based upon combusting efficiencies of enclosure fires supported by combustion engineering analysis. The TFFR relationships are presented in terms of maximum heat release rate and floor energy density.

In reference [8] eleven TFFR prediction methods, including reference [7] were compared, with the conclusion that the relationship in [7] provided a good estimation method of TFFR applicable to a wide variety of room geometries and ventilation rates. Since the approach in Barnett work [7] was based upon maximum heat release rate at the point when a compartment fire changes over from a fuel surface controlled (FC) fire to a ventilation controlled fire (VC), termed the FC/VC point, the influence of ventilation rates become bounded in the method.

However, the use of fire load energy density (FLED) as a basis for determining TFFR for applications not directly evaluated by the supported tests is problematic and on the surface contradicts the current understanding that heat release rates and suppression agent application rates are depended on exposed surface of the fuel, not on a fuel density per unit floor area. It was noted that in reference [6] that statistical survey work on developing a relationship for average fuel surface areas based upon floor area is ongoing. For application to NPPs, it is therefore recommended to use the correlation with maximum heat release rate (Qmax) which is indirectly related to fuel surface area. This relationship is given in [7] as equation 2004/1 (3), by:

$$F = \frac{k_f * Q_{max}}{k_w * Q_w} \tag{1}$$

substituting the recommended k_f and k_w of 0.5 and Q_w of 2.6 MJ/l/s absorption capacity of water at 100 ^{0}C results in:

$$\frac{k_f}{k_w * Q_w} = \frac{0.5}{0.5 * 2.6} = 0.385 l/s/MW$$

rewritten as:

$$F\ [l/s] = 0.385\ Q_{max} \tag{2}$$

Post-Flashover Enclosure Fires

Investigation by [9] and [10] concluded that for post-flashover enclosure fires, compartment conditions are well mixed and consist of a uniform zone of burning gas. Testing concluded that fire suppression agent application rates are a function floor surface area and exposed fraction of the fuel surface. For the application of water by manual hand lines, the application rate was also dependent on mean droplet diameter. Application rates were not significantly affected by changes in enclosure height over the range of enclosure dimensions investigated. For mean droplet diameters typical of manual handlines and assuming that the fire fuel surface is fully shielded (bounding condition) TFRR is bounded by 0.2 l/s per m^2 of <u>compartment floor area</u>. The 0.2 l/s per m^2 is supported by the statistical survey work of Sardqvist and Holmstedt in [6] where water application rates were measured at real fires. This relationship is:

$$F\ [l/s] = 0.2\ A_{floor} \tag{3}$$

Flashover Potential Evaluation

To establish if flashover was possible in the compartment the McCaffrey, Quintiere, Harkleroad (MQH) correlation [11-15] flashover test was employed which accounted for the compartment thermal boundary properties. This resulted in the following test for concrete compartments

Flashover is possible if:

$$Q_{max} \geq 0.193\ (A_t\ m_{air})^{1/2} \tag{4}$$

TRIAL APPLICATION

To support the development of the proposed TFFR methodology a trail application was performed at a Canadian NPP. The name and location of the NPP used in this trial application is not published, however, it is noted that the construction charateristics and operational practices are typical of CANDU and PWR (pressurized water reactor) NPP types. This work involved the systematic assessment of fire hazards was performed to determine each zone's most demanding Tactical Fire Flow Rate (TFFRz for a zone) at a Canadian NPP. Agent selection and application method were based upon current fire attack strategies as documented in the NPPs pre-fire plans, standard response procedures and ERT training documentation.

To manage the effort required for this task the review focused on hazards which are postulated to challenge safety related areas, hazards may be within the safety related area or an exposure to a safety related area. Where possible the analysis relied upon bounding assumptions to ensure conservatism in the analysis. The assessment of TFFRz is summarized in the following table with the most demanding TFFRz.

As detailed in the referenced methodologies the assessment of TFFR is not intended to be a precise determination of TFFR, but a prediction useful in established minimum agent application flow rates to support a minimum resource assessments.

Brief Summary of the TFFR Assessment Methodology

Where water suppression was employed, the TFFRz was determined either by a fuel surface area demand model or a compartment post flashover demand model. Fuel surface areas and other critical parameters were determined based upon the hazard data in the facilities FHA. Where the compartment post-flashover model was employed, the information from the FHA was used to postulate a vent controlled post flashover compartment environment. The assessment as to if an individual compartment is capable of proceeding to flashover was determined using the MQH correlation [11] – [15] which permitted a conservative assessment of flashover potential and allowed the inclusion of compartment boundary material. Ventilation rates and compartment doors were assessed in the analysis based upon the design manuals and design documentation of the facility.

The conclusion of the assessments as determined by the site specific calculations for each NPP was that two attack lines (offensive or defensive) would be required for a number of postulated fire scenarios. Review of the calculated TFFRz revealed that numerous scenarios resulted in the above demand.

CONCLUSION

For hazards and construction types found in NPPs, a methodology is presented to predict TFFRs. In a trial application of the proposed methodology for a CANDU NPP plant, the predicted TFFRs are consistent with current estimates for suppression agent flow rates determined by field reviews by subject matter experts.

RECOMMENDATIONS

The application of the methodology discussed in this paper requires for post flashover compartment scenarios a TFFR based upon floor area. Although the literature review established that room height effects had limited influence on the water application density and the selected application rate bound all test scenarios, including where ceiling heights were doubled, an improved correlation based upon room geometry including height is more desirable. In reviewing the post flashover data generated and plotting against total compartment surface area (see Figure 1) it was noticed that it appears a bounding linear relationship that should be explored.

RESULTS OF TRIAL APPLICATION TO A CANDU REACTOR

Table 1 Predicted TFFR based upon data from trial use at a CANDU NPP

Room Code	Qmax [MW] - from FHA	Eq. (4) MQH Flashover Qmax [MW]	Possibility of Flashover (yes/no)	Area [m^2]	Total Area of Compartment Surfaces [m2]	Eq. (2) Compartment Demand for Water [l/s]	Eq. (3) Post Flashover [l/s]
109	41.94	9.45	yes	74	419.56	N/A	14.8
321	41.94	0.57	yes	74	419.56	N/A	14.8
428	41.94	14.58	yes	74	419.56	N/A	14.8
41	29	4.69	yes	46.75	411.50	N/A	9.35
320	27.96	1.12	yes	197	833.46	N/A	39.4
42	19.225	8.47	yes	169.99	667.02	N/A	34
45	19.225	60.56	no	700	2459.90	3.70	N/A
70	19.225	31.69	no	240.3	1111.78	3.70	N/A
199	11.65	0.46	yes	45	259.40	N/A	9
306	11.65	8.91	yes	45	259.40	N/A	9
134	3.508	5.31	no	74	264.48	0.68	N/A
346	3.508	0.81	yes	74	264.48	N/A	14.8
453	3.508	7.53	no	74	264.48	0.68	N/A
295	3.077	0.74	yes	56	228.28	N/A	11.2
404	3.077	0.74	yes	56	228.28	N/A	11.2
570	2.847	1.94	yes	306.2	1040.68	N/A	61.24

Room Code	Qmax [MW] - from FHA	Eq. (4) MQH Flashover Qmax [MW]	Possibility of Flashover (yes/no)	Area [m^2]	Total Area of Compartment Surfaces [m2]	Eq. (2) Compartment Demand for Water [l/s]	Eq. (3) Post Flashover [l/s]
208	2.8	4.05	no	439	1651.80	0.54	N/A
316	2.8	22.47	no	439	1651.80	0.54	N/A
423	2.8	13.43	no	439	1651.80	0.54	N/A
72	2.4	3.16	no	56.95	186.56	0.46	N/A
12	2.33	14.31	no	118	537.32	0.45	N/A
151	2.075	42.87	no	165	3840.24	0.40	N/A

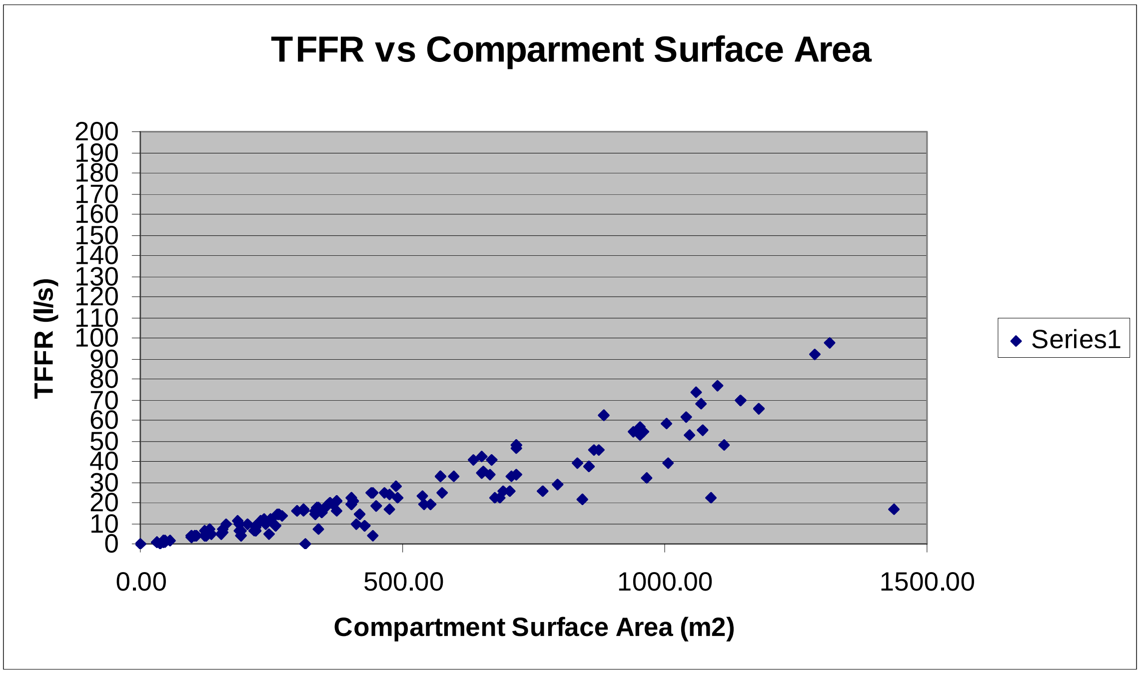

Figure 1 Relationship of predicted TFFRz with compartment surface area A_t

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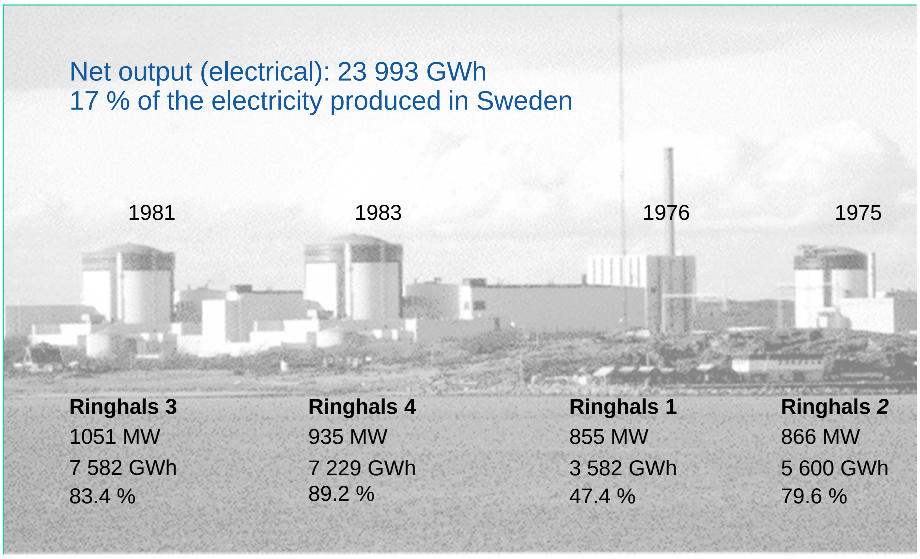
Fire in the Containment During Pressure Test Causing Great Damage at Ringhals NPP, Unit 2 2011-05-10

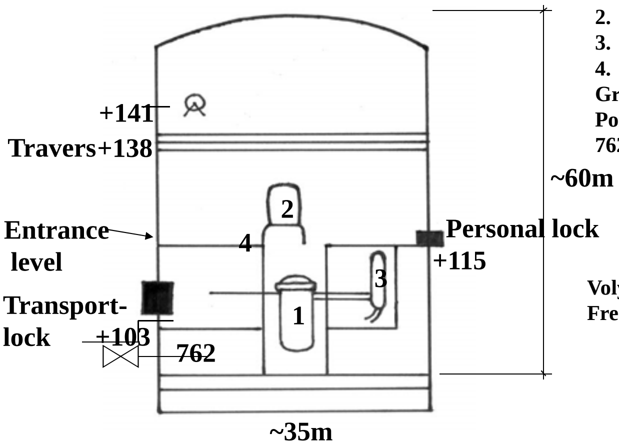
Tommy Magnusson

Planned start is 18.11.2011 +/- 30 days

Ringhals 2010



Containment



1. Reactor
2. Steam generator
3. Reactor Coolant Pump (RCP)
4. Place of fire +115 entrance level

Ground floor +103
Polar Crane +138
762 Fire main

Volyme ~$62x10^3m^3$)
Free volyme ~ $50x10^3m^3$)

Before the Fire 08:23 the same day

Melted lead (Melt point 327°C)

Kick sled

Aluminium ladder that melted (Melt point 660°C)

The dry/wet vacuum cleaner that caused the fire

After the Fire

2012-03-28 Ringhals Fire Containemnt 20110510 5

Before the Fire 08:23 the same day

2012-03-28 Ringhals Fire Containemnt 20110510 6

After the Fire

All the available fire load is completely burnt

Fire Start

- On 20110510 at 23.32 the temperature increased in the containment
- CAT (Containment Air Test/ 3.16 bar overpressure) started the same day
- Access/Entry was not possible
- The fuel was in the fuel building
- The initial plan had been changed. The CAT was performed before reloading, normal terms were not fulfilled. Some material that had to be used again was left.

First Reaction in MCR

- Temperature increase of 10°C in a few minutes
- The CCTV cameras were hard to use because of the soot on the lenses
- There was uncertainty what had happened
- The only fire extinguishing system inside the containment is the containment spray system. The plant operating procedure says that the containment spray system could be used in case of fire.
- After a long safety discussion depressurization was started

Sequence of Events

1	2	3	4	5	
Mond.10	Tues.11	Wedne.12	Friday13	Satur.14	Sund.15

- 20110510[1] 23:32 - Detection of heat/smoke
- App 20110511[2] 00:00-30 - The fire was burnt out
- 20110512 07:30[3] - Depressurization started after safety discussion
- 20110514 02:45[4] - First entrance / fire located
- 20110515 15:00[5] - Dryer was in operation

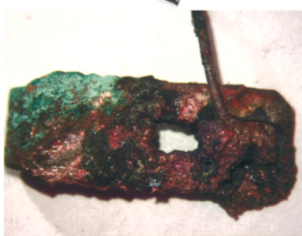
The humidity was reduced from 80 % to 40 %

Damage in Containment

- The damage by heat or hot gases is concentrated to two cables
- The soot collared the whole containment above the fire, black, with high chloride level (100µg/cm² limit 1 µg/cm²)
- Below the fire soot has penetrated some components, but the soot was not visible (4300 components)

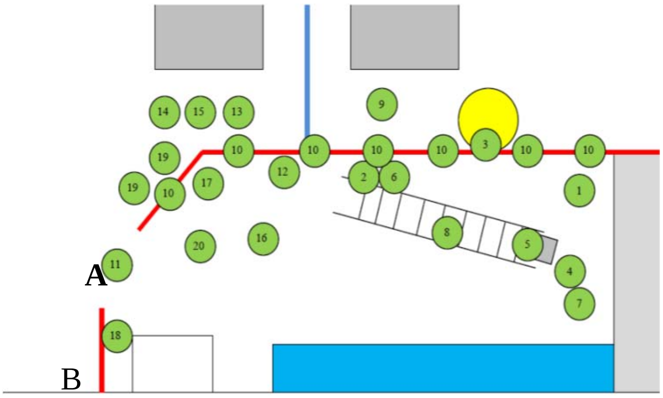
Fire Cause

- There was an arc in the vacuum cleaner
- All other reasons can be excluded
- SKL The Swedish Laboratory of Forensic Science have investigated the objects from the vacuum cleaner



Connection to one of the engines.
The surface covered with copper splash.
The bearing shows no overheating

Burned Material



1 Divinghose
3 Dry suit
5 Bucket/ tools
7 Plastic
9 Water Hose
11 Shoe border
13 Cable wall machine
15 scouring machine
17 Vacum cleaner dry/wet
19 Trashbag open top

2 Diving line
4 Wieght belt
8 Armored plastic
10 Kick sled
12 Plastic bags
14 Scotch-Brite
16 Divning lamp
18 Kabels
20 Explosive safe can

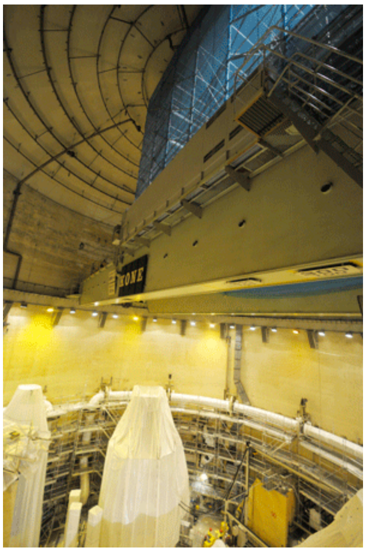
A Box with shoe covers

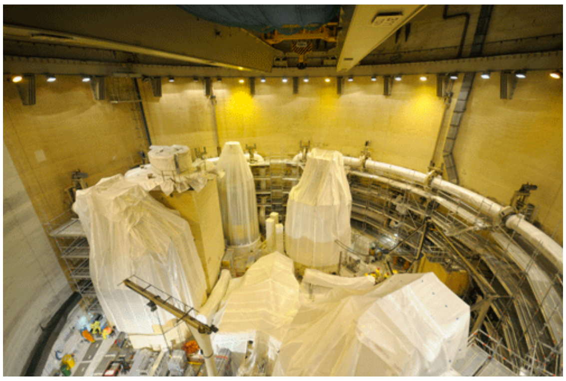
B Box with rubber gloves

In total approx. 5 kg rubber and 30 kg plastic

On the top of the pressurizer

Containment spray pipes on the top of containment





After cleaning and
covering components
and cavity

2012-03-28 Ringhals Fire Containemnt 20110510 15

Lessons Learned

- Flame spread: higher with overpressure
- Transient fire load; must be controlled
- Fire cause elimination
- Communication on what is an acceptable fire load
- Soot /Chlorides: effect on components
- Fire protection in the containment
  - RCP oil/cables and transient fire load
  - No fixed fire extinguishing systems
  - Manual fire fighting is difficult in areas with lock or overpressure

To handle (avoid, detect and extinguish) a fire in the containment has to be very well planned

2012-03-28 Ringhals Fire Containemnt 20110510 16

THE MULTI-STAGE FIRE SAFETY CONCEPT IN GERMAN NUCLEAR POWER PLANTS

Björn Elsche[1], Günter Fischer[2], Stefan Kirchner[2]

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2 TÜV SÜD Industrie Service GmbH, Germany

During the construction and the operation of a nuclear power plant many requirements have to be met. These requirements are derived from the atomic energy act, which contains different specific aspects. The requirements are defined in a graded set of rules and standards.

Figure 1 shows the so-called pyramid of regulations. According to the highest value the laws (the German constitution named Basic Law and the Atomic Energy Act [1]) build the top of the pyramid, followed - from top to the bottom - by ordinances and administrative regulations including general requirements.

Below, the more detailed requirements specified in the regulatory guidelines published by BMI/BMU (e.g. Nuclear Power Plant (NPP) Safety Criteria [2]), Guidelines for the Assessment of the Design of PWR Nuclear Power Plants against Incidents (short form: Incidents Guidelines) [3], etc.) and the RSK Guidelines for Pressurized Water Reactors (PWR) [4] can be found. On another lower level the nuclear safety standards of the German Nuclear Safety Standards Commission KTA (KTA Standards) are provided covering specific detailed requirements and criteria corresponding to the higher level requirements and their realization in nuclear power plants. The wide bottom of the pyramid is built of the generally accepted technical norms and standards for industrial facilities.

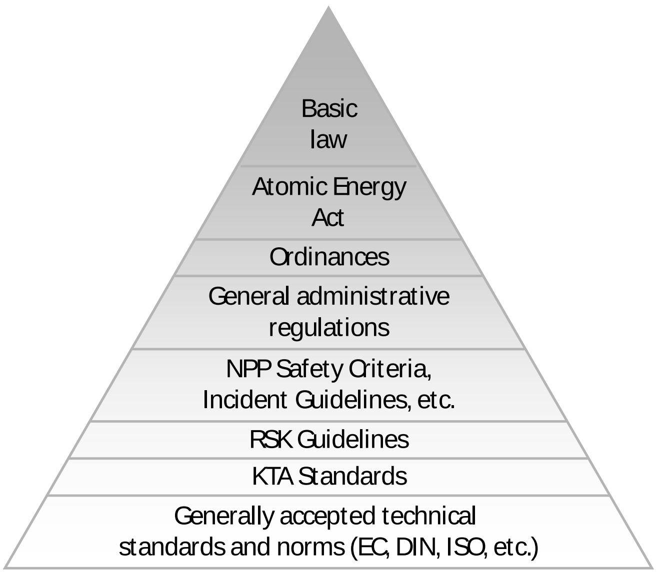

Figure 1 Pyramid of regulations for nuclear installations

Moreover, requirements which are derived from the conventional non-nuclear regulations have to be considered. The basis for these regulations of civil engineering (as outlined in Figure 2) is also the Basic Law, followed by the building codes, ordinances, administrative regulations, technical norms and standards, and private guidelines as well as specifications by the manufacturers.

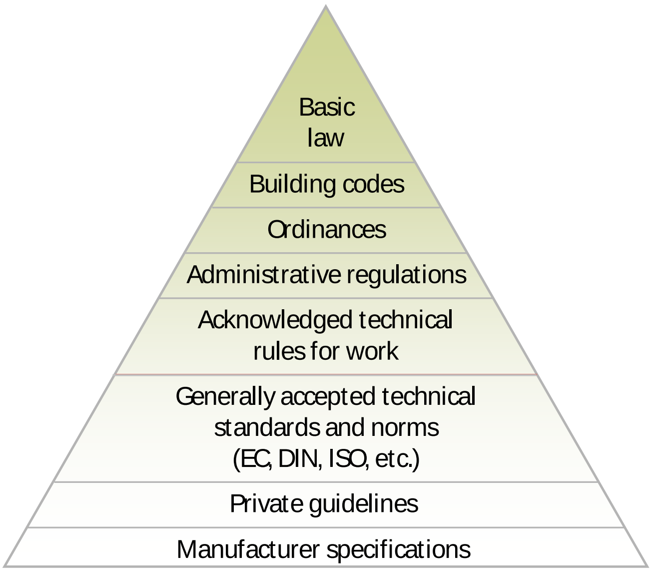

Figure 2 Pyramid of regulations – conventional requirements (civil engineering)

Below it is shown, which nuclear and conventional requirements with regard to fire safety have to be met to build and operate a nuclear power plant.

In the Basic Law the primary requirement is to ensure the life and physical integrity of persons. That also means that a fire shall not have any consequence to the life of an individual. This applies both for persons, who are close to the fire (in the NPP) and those, who are far away from the fire (outside the NPP), but could be affected by the fire.

In the nuclear regulations, fire safety is addressed in several ways. The nuclear regulations contain national standards and regulations regarding fire prevention and fire protection /mitigation aspects. In addition, § 7 of the Atomic Energy Act regulates [1], which safety precautions have to be made. By implementing these precautions, all measures with respect to fire safety have been taken into account for preventing damage resulting from the design and operation of the nuclear power plant.

In the following regulatory documents the requirements are described in detail. The German Nuclear Power Plant Safety Criteria [2] require that safety related plant components shall be in the condition to fulfill their functions in case of fires and explosions. Moreover, an early detection of fires shall be ensured and fire extinguishing features shall be available. Further detailed requirements are described in the RSK Guidelines, which consider the nuclear specific criteria. The fire safety specific nuclear standard KTA 2101 [4] describes in detail the specific requirements with regard to fire including their implementation at the nuclear power plants. KTA 2101 [5] consists in total of three parts:

- Part 1: Basic Requirements
- Part 2: Fire Protection of Structural Components
- Part 3: Fire Protection of Mechanical and Electrical Components.

Besides nuclear requirements, civil engineering requirements with regard to fire safety must be fulfilled as well. In accordance with the conventional protection goal of the federal building codes buildings have to be designed, constructed and maintained that the public security and order, particularly life, health and the natural life conditions shall not be endangered. In particular, the requirements demand with respect to fire safety that "all structures shall be planned, constructed, modified and serviced to prevent the initiation and spreading of a fire and in case of fire to ensure effective fire fighting and the rescue of people …" [6].

In the KTA Standards, detailed nuclear specific requirements are provided. The KTA fire protection standard KTA 2101 [5] demands a NPP specific fire safety concept, which will be described in detail further on in this paper.

In accordance with the standard KTA 2101, Part 1 the following nuclear protection goals have to be met:

"Protection against internal and external building fires, with respect to

a) Plant components whose safety functions must necessarily meet the protection goals on which the Safety Criteria are based, i.e.,
   - aa) Control of reactivity,
   - ab) Cooling of fuel assemblies,
   - ac) Confinement of radioactive materials and
   - ad) Limitation of radiation exposure,

b) Structural plant components which enclose such plant components and

c) Plant personnel."

The standard KTA 2101 implements the requirements with respect to fire safety, which are described in the higher level regulations. To meet these requirements several measures are taken into account.

As a general requirement fire loads and potential ignition sources have to be prevented or at least minimized as far as practically feasible. In areas where they cannot be totally prevented separation has to be considered. This supports the prevention of the occurrence of fires. It is not possible to optimally reduce all fire loads and potential ignition sources. Hence, a fire safety concept is necessary to meet both the conventional and nuclear protection goals. To accomplish this, four types of fire protection measures are provided (see also Figure 3):

- Structure related fire protection,
- Equipment related fire protection,
- Operational fire protection,
- Fire defense.

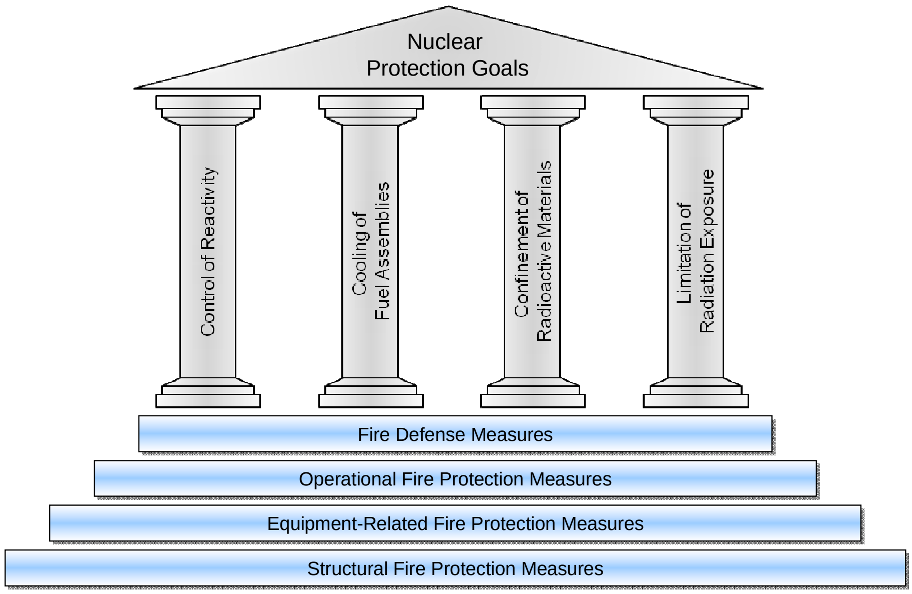

Figure 3 Fire safety measures to accomplish the nuclear protection goals

The spread of fire shall be prevented by structure related fire protection measures, such as the choice of adequate and suitable building materials (mainly non-combustible or flame-resistant ones) as well as separation by forming physically separated compartments (fire compartments, fire sub-compartments, smoke compartments). Priority is given to these primary passive measures.

In addition, active fire protection measures have to be taken into account to ensure early fire detection, the automatic extinguishing of fire, fire fighting as well as the rescue of people.

The corresponding equipment related means include fire detection systems and equipment , stationary fire extinguishing system and fixed as well as portable fire extinguishing equipment, ventilation and smoke extraction systems. Operational fire protection includes administrative measures (e.g., minimization and separation of combustible operating and working media, etc.) as well as portable equipment for fire fighting and rescue of people. In addition, fire defense measures can be performed by the on-site plant internal professional fire brigade and/or by off-site fire brigades.

Regarding fire safety, Figure **4** demonstrates the coaction of all types of fire protection measures to meet the conventional and nuclear protection goals by using the illustration of a four stranded cord.

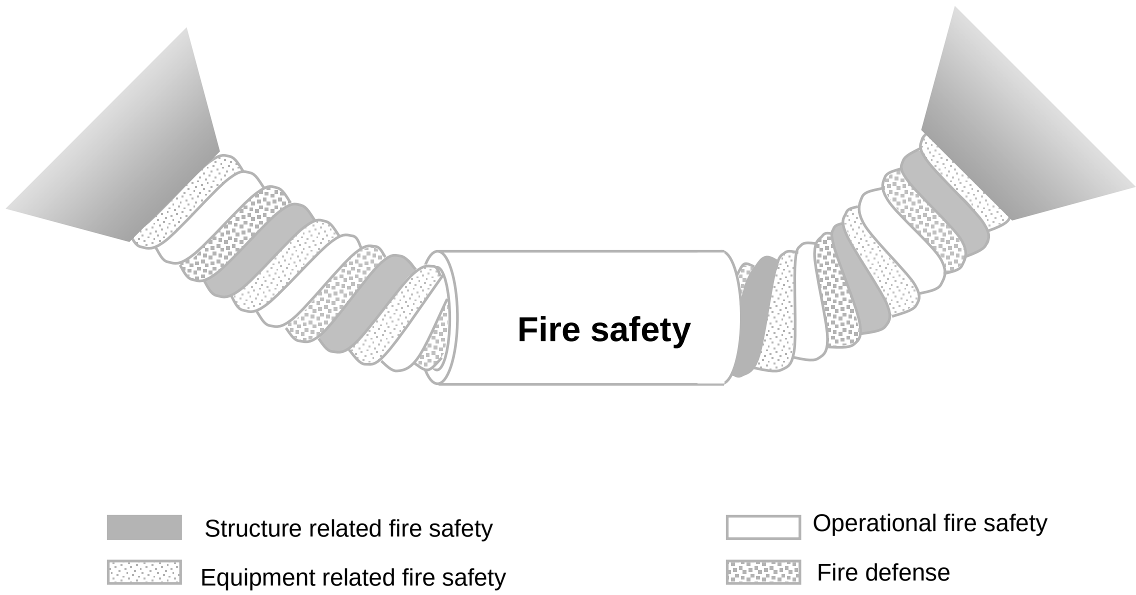

Figure 4 Interactions of the different types of fire protection measures

Not all fire protection measures and types of fire protection means are always needed in case of fire to meet the nuclear protection goals. Regarding fire safety, a fire has to occur at first before a nuclear protection goal will be violated. The minimization and separation of fire loads and ignition sources is the primary measure of the fire safety concept. Hence the fire risk is strongly minimized.

In the case of fire the separation of redundant trains, either by qualified rated fire barriers or at least by physical separation through distance supported by fire retardant protective shields. Coating, etc. ensures that a fire in an area relevant to safety can only affect components or equipment of one redundant train and therefore does not violate the nuclear protection goals. Nuclear power plants´ design considers even the unavailability of one safety train by maintenance activities and a random failure of another train in case of an event such as fire to ensure the availability of at least one redundant train.

The failure of a fire protection measure will normally be detected before a fire occurs. Therefore, signals of fixed active fire protection features (e.g. fire detection system) are displayed at the control room and periodic preventive inspections and preventive maintenance take place for passive fire protection measures on a regular basis. The undetected random failure of a single measure in case a fire occurring does not lead to the violation of the nuclear protection goals because of the remaining plurality of other fire safety measures (see Figure 5).

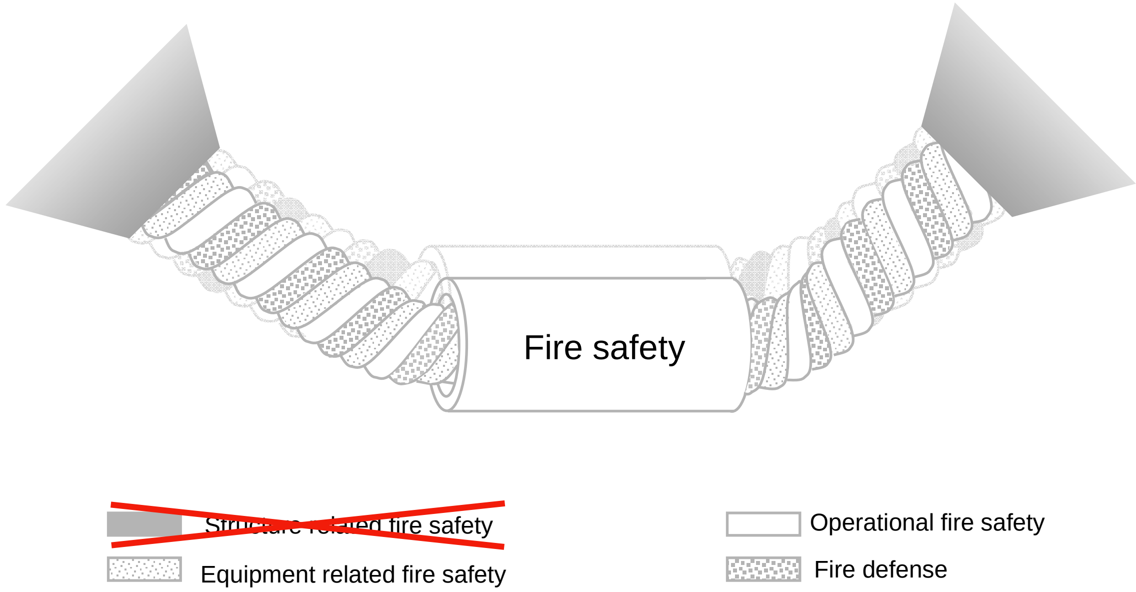

Figure 5 Interaction of the different types of fire protection measures, if one measure fails

So far, fire is considered as a singular event. Fire might also appear in combination with other internal or external events. The standard KTA 2101, Part 1 also deals with the combinations of:

- A fire and a subsequent event,
- A postulated event and consequential fire, and
- A postulated event with an unrelated fire.

If the combinations of these events are significant with regard to nuclear safety, specific measures have to be considered to meet the nuclear protection goals under these conditions. Two examples illustrate the interaction of the fire protection measures:

Penetration Seals

If cables or pipes have to cross two fire compartments, the wall ducts have to be protected to keep the fire resistance of the compartment. Therefore, penetration seals have been installed to prevent the spread of fire and smoke for a limited period of time. Usually a failure of the penetration seal can be excluded, because the building materials used and the assembly of the penetration seal by itself are proven to resist the fire and verified during the course of assembly, which is maintained by specialists. In accordance with standard KTA 2101, Part 1 [5] an authorized expert has to inspect the penetration seal after the assembly and during recurrent inspections.

In the case of fire, coupled with an additionally damaged penetration seal the spread of fire and smoke to another fire compartment can be assumed. Due to the large number of fire detection devices, smoke leakage can be detected early. According to the KTA standard 2101, Part 1 [5] the fire detection and alarm system is designed to ensure the localization of the fire source via smoke location including the identification of the location and the verification of the alarm at the fire alarm boards. Areas with a large fire load are basically protected by automatically actuated stationary fire extinguishing systems. Finally, fire-fighting measures will be performed by the available on-site fire brigade within a fast response time. Alternatively the off-site fire brigade can be alerted as well.

Therefore, the protection goals are reached by the interaction of the remaining types of fire protection measures (see also Figure 6).

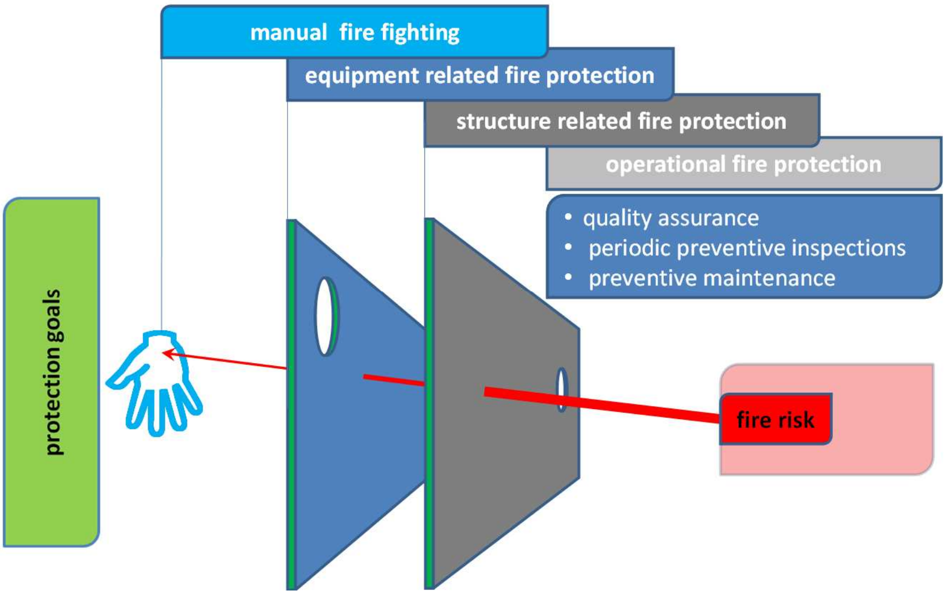

Figure 6 Effects of the fire risk on the protection goals regarding the fire safety measures

Earthquakes and Consequential Fire

As a result of an earthquake it is possible that a fire could occur at a component, which is not designed against earthquake. Every redundant train is separated with regard to fire safety. That means a fire cannot affect more than one redundant train at the same time. Fire fighting can be performed inside the compartment/s of the affected redundant train. The protection goals are not violated. Figure 7 shows a possible situation in a building with four redundant trains (typically given in the switchgear building of modern for German PWR plants) after an earthquake. The redundant train affected by the fire is separated by structural fire protection means designed against earthquake from compartments containing equipment of other redundant trains. Due to this, fire fighting is possible without any restrictions and the remaining redundant trains are left unaffected.

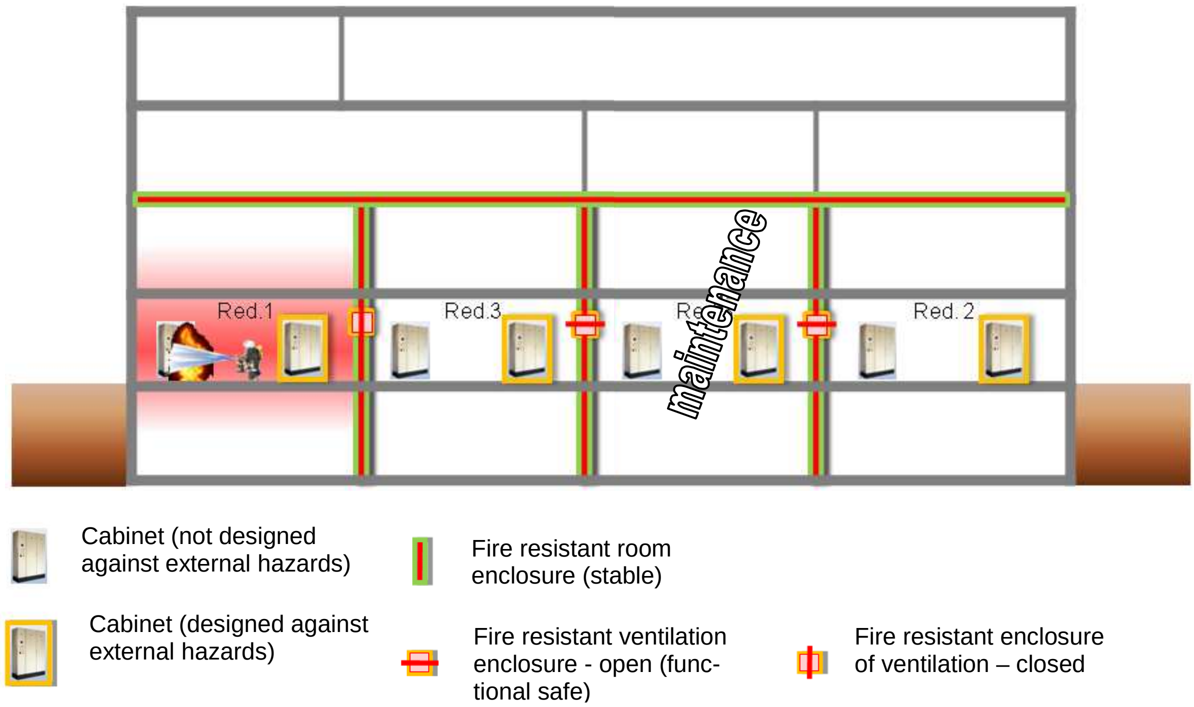

Figure 7 Situation after earthquake

The examples described can be assigned to other fire safety measures and combinations of events. Due to the safety orientated fire safety concept an effective fire safety is ensured even if one individual measure fails or is restricted.

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ANALYSIS FOR THE OPTIMIZATION OF GERMAN NUCLEAR POWER PLANTS AFTER THE INCIDENTS IN FUKUSHIMA

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ABSTRACT

On March 11th 2011 several tsunami waves caused by the Tōhoku earthquake (magnitude M_w = 9.0) struck the east coast of Japan. These incidents, amongst others, led to a long-lasting station blackout along with a total failure of the cooling water supply at the Fukushima Daiichi nuclear power plant (NPP) site [1].

As a result of these incidents several of the German local state authorities decided to carry out special safety inspections (SSI) at selected German nuclear power plants (NPPs). The beyond design basis events induced by natural external hazards, such as earthquakes, floods and extreme meteorological hazards (e.g. snow, heavy rain and wind loads) as well as the man-induced hazards (explosion pressure wave, airplane crash, and hazardous gases) have been taken into account within the investigation [2].

In consideration of the resulting consequences of these hazards, the goal of the SSI was to identify optimization potential (OP) regarding the robustness of the nuclear power plants taking into account the nuclear protection goals defined by the nuclear regulations, e.g. "Sicherheitskriterien für Kernkraftwerke" (English: Safety Criteria for Nuclear Power Plants) [3].

The failure of the cooling water supply, the power supply and other internal events, such as internal fires, shattering effects, internal flooding and loss of coolant have been considered [2].

This paper presents the optimization potential with relevance to fire safety. The results demonstrate that even small optimization measures can increase the availability of buildings, systems, and equipment including portable items such as vehicles in case of a beyond design basis incident. Therefore, the robustness of the NPPs in case of beyond design basis scenarios should be improved.

Moreover, further optimization potential was identified regarding organizational and logistical aspects. In order to prevent complications during fire fighting operation at a NPP or vehicle and equipment failure, the results of this research should be considered for a continuous optimization process of the existing fire safety measures. This includes the on-site fire brigade, external forces as well as the fire fighting measures.

INTRODUCTION

The Tōhoku earthquake occurred on March 11th at 05:46 UTC (14:46 JST), 130 km ESE off Ojika Peninsula at a focal depth of 24 km on the subduction zone between the North American plate and the Pacific plate. The earthquake had a magnitude (M_w) of 9.0 and intensity (JMA) of 6+ at Fukushima Daiichi NPP site. Besides extreme ground motion the earthquake also generated large tsunami waves which struck the east coast of Japan. The largest tsunami wave occurred in Aneyoshi, Miyako with a height of 38.9 m [1]:

The "Great East Japan Earthquake" [2] affected five of the fifty-four existing nuclear power plants in Japan. Figure 1 shows the location of the affected nuclear power plants Higashi Dori, Onagawa, Fukushima Daiichi, Fukushima Daini and Tokai which are located at the north-eastern coast of Japan. All reactors were in operation when the earthquake occurred, except the unit in Hagashi Dori and Units 4 to 6 of Fukushima Daiichi NPP. The earthquake caused an automatic reactor trip (scram) of all operating units.

As a result of the effects of the earthquake all five NPP sites got hit by several tsunami waves. The most affected NPP sites were Fukushima Daiichi (6 units, all BWR) and Fukushima Daini(4 units). The maximum estimated height of the tsunami at Fukushima Daiichi was 14.0 m. The tsunami protection measures at the NPP site were designed for a tsunami up to 5.7 m height [2].

Figure 1 Nuclear power plants affected by Tōhoku earthquake; Source: JAIF, Inc.

Except for one emergency diesel serving Unit 6, Fukushima Daiichi NPP site (Figure 2) lost most of its safety related equipment and all off-site and on-site power owing to the inadequate design for the tsunami experienced. The loss of cooling for Units 1 to 3 and the spent fuel pools (SFP) of Unit 4 as well as cooling for other safety related equipment caused accident conditions in these four units [1].

Figure 2 Fukushima Daiichi NPP site, 6 Units; Source: IAEA mission report [2]

For this reason, several German local state authorities decided to carry out special safety inspections (SSI) at selected German NPPs.

ROLE OF THE TÜV® IN THE SSI

In Germany the local state authorities of the Federal States (so-called "Länder") are responsible for the licensing and supervision of nuclear power plants and other nuclear installations. According to The German Atomic Energy Act § 20 [4], the local state authorities are permitted to assign licensed technical experts e.g. from TÜV SÜD. Figure 3 demonstrates the SSI procedure. At the end of March 2011, a concept for the SSI in NPPs was developed by some of the German local state authorities. Therefore TÜV®-Organizations were assigned to form a commission to carry out special safety inspections at NPPs in order to provide a report on findings and optimization potential (OP) considering specific assumptions with regard to the incidents in Fukushima.

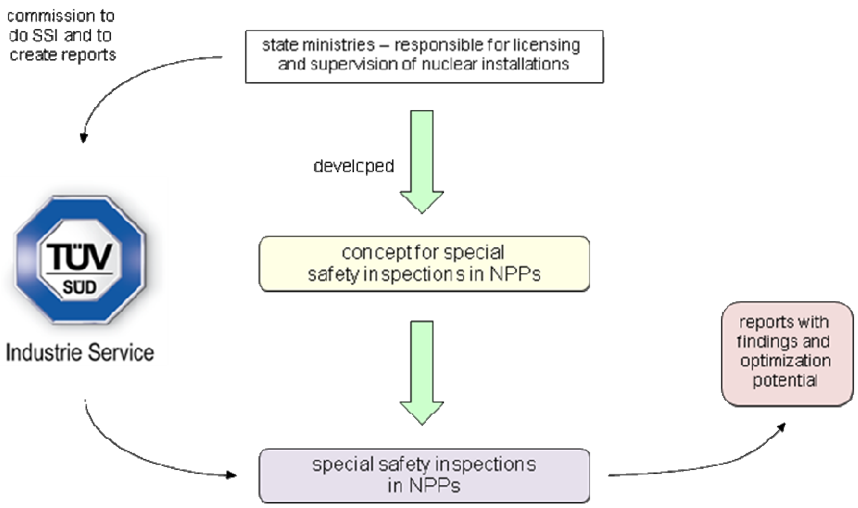

Figure 3 SSI process including the role of TÜV SÜD Industrie Service

SPECIAL SAFETY INSPECTIONS (SSI) AT SELECTED GERMAN NUCLEAR POWER PLANTS

The NPP site specific boundary conditions and the approved design of each NPP were used as a basis for the SSI. In addition, **beyond design** basis events have been considered. Because of their low probability, such **beyond design** basis events are not part of the current nuclear regulations or plant design.

Furthermore, the investigations took into account aspects such as manpower (on-site), vehicles and equipment of the on-site fire brigade [2].

With regard to the Fukushima incidents, the SSI considered the natural hazards of earthquake, flooding and extreme weather conditions e.g. snow precipitation and wind loads (all **beyond design** basis) as the three main scenarios. In addition the investigations also focused on structural protection in case of intentional aircraft crash as well as on protection of NPPs against man-made attacks [2]. The results regarding these additional aspects are not part of this paper.

Findings as well as the suitable optimization potential should be defined and mentioned in the report.

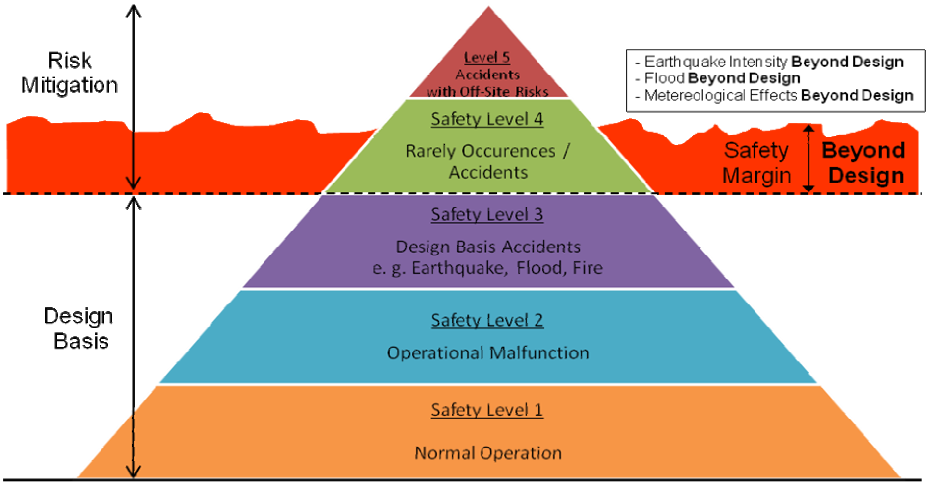

Figure 4 Defense-in-depth model of nuclear safety

Figure 4 illustrates the defense-in-depth model of nuclear safety. In order to ensure the nuclear protection goals

- control of reactivity
- cooling of fuel assemblies
- confinement of radioactive materials
- limitation of radiation exposure

the nuclear safety concept comprises in five levels.

Incidents within safety levels 1 to 3 are covered by the design of the NPP. Thereby, the nuclear protection goals are met. Due to their low probability of occurrence beyond design basis incidents belong to levels 4 to 5, which are not or not completely covered by the plant design. In that case the nuclear protection goals are violated and measures for these incidents shall only mitigate the risks after an occurred incident.

Within the SSI the assumed beyond design basis scenarios have been regarded to safety level 4.

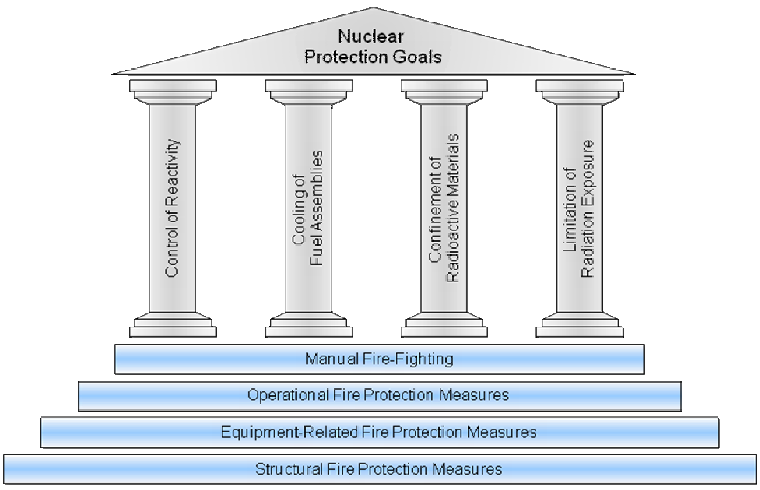

Figure 5 Nuclear protection goals regarding fire safety measures

Figure 5 shows the four steps of fire safety measures as one method to ensure the nuclear protection goals are met. In regard to the beyond design basis scenarios, the structural fire protection measures have been considered as failed (e.g. fire dampers in fire walls or blocked emergency exits) in the SSI. Equipment-related fire protection measures were also considered as not available after the assumed incidents, due to the lack of seismic withstand requirement of these systems. Operational fire protection measures, such as the use of fire hose stations, could also be assumed as being not available as well (e.g. non-designed extinguishing water supply). The fire fighting means are the final possibility of fire safety measures that should be successful at the NPP due to the existing on-site fire brigade. The greatest optimization potential can be assumed in that area.

GOALS OF THE SSI

The main goal of the SSI was to analyze the selected NPPs with regard to meet the nuclear protection goals [3] in case of the above mentioned beyond design basis events. Therefore the following consequences were considered:

- Loss of cooling water supply,
- Loss of power supply, and
- Further consequential events (e.g. internal fires and flooding).

The nuclear power plants were particularly investigated to determine if, and how, fundamental safety functions considering the specific nuclear protection goals "Control of Reactivity" (reactor pressure vessel) and "Cooling of Fuel Assemblies" (fuel element storage pool) shall be ensured. For the assumed scenarios, the practicability and the efficiency of existing emergency measures have been taken into account. In order to create a standard questionnaire for all NPPs inspected a wide variety of assumptions was considered.

SPECIFIC ASSUMPTIONS FOR THE SSI WITH REGARD TO FUKUSHIMA

The following general assumptions relevant to fire safety have been considered in the SSI:

- Consideration of the natural hazards: earthquake, flooding and extreme meteorological conditions in terms of the availability of coolant supply and power supply;
- Beyond design basis earthquake and flooding (but without specific seismic intensity or flood height);
- No loss of function of seismically qualified buildings; however an interference of accessibility was assumed;
- The simultaneous occurrence of a beyond design basis earthquake and a beyond design basis flooding was <u>not</u> assumed (no causal connection such as for Fukushima: earthquake → tsunami);
- Aspects regarding disaster management have not been investigated

Due to the failure of the cooling water supply and the failure of the power supply, as a result of the tsunami waves at the Fukushima Daiichi NPP site, the following specific assumptions (limited to fire safety aspects) have been considered for the investigation:

Earthquake:

- Access roads are blocked due to debris.
- Buildings that are seismically qualified have not collapsed but could be damaged (e. g. a fire wall is no longer able to fulfil its safety function).
- Buildings that are not seismically qualified have collapsed.
- Fire supply system is not available.

External Flooding:

- Access roads are flooded.
- Internal fire during long-term flood is assumed.

FINDINGS AND OPTIMIZATION POTENTIAL (OP)

Considering the assumed scenarios - beyond design basis earthquake and beyond design basis flooding - the SSI has demonstrated different findings and optimization potential (OP). The following findings and OP measures apply for fire safety aspects:

Beyond Design Basis Earthquake:

OP for quick access to buildings in case of fallen concrete wall panels

As shown in Figure 6, the external facades of seismically qualified buildings are often panel constructions (e. g. concrete wall panels). In the event of a beyond design basis earthquake, the displayed concrete wall panels could detach and interfere access to buildings containing systems relevant to nuclear safety. This could cause delays for the staff or fire fighters attempting to enter the buildings. The availability of emergency exits could also be affected.

Figure 6 Blocked building entrances due to fallen concrete elements

It is concluded that the design of the external facades should be improved and/or access to the building should be possible by other means.

OP for quick access of buildings in case of damaged / blocked physical protection doors

Safety related buildings are designed against earthquake loads. These buildings usually have physical protection doors because of their safety significance. The accessibility of physical protection doors are not verified in case of a beyond design earthquake.

Figure 7 illustrates that the doors of a building might be damaged (electronically or/and mechanically) or mechanically blocked due to the lack of seismic withstand requirement of these doors. Although the building would withstand an earthquake. In addition, the keys for these doors are sometimes stored in buildings with minor earthquake design. In case of the assumed destruction of these buildings, the keys might be inaccessible.

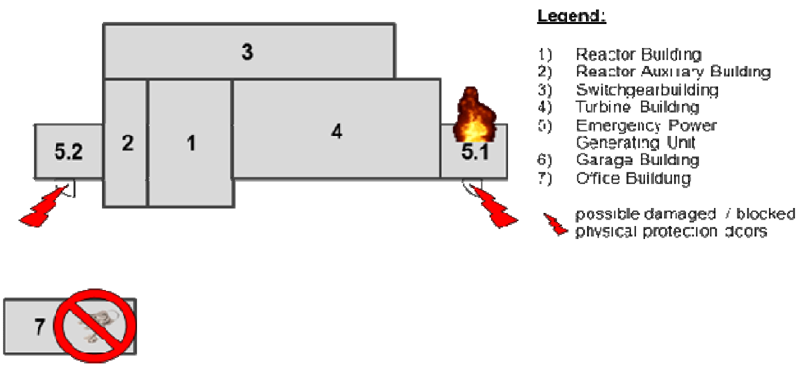

Figure 7 Potentially blocked physical protection doors

To increase the availability of physical protection doors and to ensure a quick accessibility after an earthquake the doors should be functionally tested and -if necessary-seismically qualified. The location of the key-storage should be improved (e.g. kept in seismically qualified buildings).

OP for availability of the on-site fire brigade vehicles and equipment:

On-site fire brigade buildings, such as garages or spare-part buildings are usually not seismically qualified (nuclear specific). In general these buildings are built as concrete or reinforced concrete construction (see Figure 8). The vehicles or equipment located in these buildings might not be available due to heavy damages or the destruction of the buildings after an earthquake.

Figure 8 Not seismically qualified on-site fire brigade garage buildings

The availability of fire brigade equipment/vehicles has to be verified and -if necessary-to be improved.

OP for the availability of external personnel (off-site fire brigade, NPP staff on call – needed for fire fighting and emergency organization)

In case of a beyond design basis earthquake the availability of external personnel (off-site fire brigade, NPP staff on call needed for fire fighting and emergency organization) is uncertain. Figure 9 shows that, due to the possible damage or destruction of dispatch centers, phone company buildings or radio masts, it might not be possible to alert internal and external personnel. Blocked or heavily damaged access roads could prevent site access. Within the SSI it could not be answered how the reinforcement of internal and external personnel is currently organized or ensured.

In addition to the possible destruction of off-site fire brigade buildings, external personnel (and their equipment) might not be available for operations at NPPs due to their requirement in other public operations.

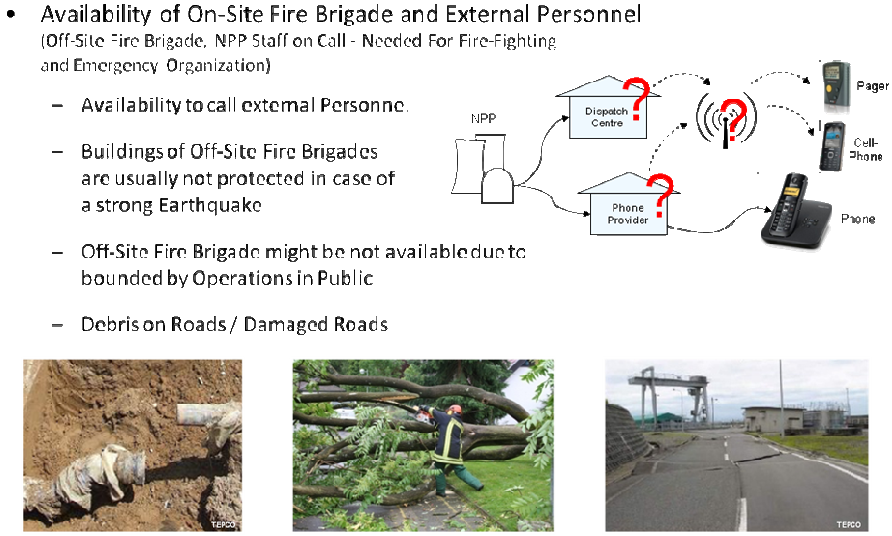

Figure 9 OP - Availability of on-site fire brigade and external forces

The organization of external resources (and associated equipment) in case of a beyond design earthquake should be considered and optimized.

Beyond Design Basis Flooding

OP for the availability of the on-site fire brigade and external forces (off-site fire brigade, NPP staff on call – needed for fire fighting and emergency organization)

In case of beyond design basis flooding the access roads at or around the NPP site might be impassible. The availability of the roads depends on the ground level of the site, the flood height and the damage due to the flood. The availability of the fire brigade vehicles (on-site and/or off-site) is limited by the wading depth, the position/height of the exhaust pipe, the position of the vehicle batteries and the position of air pressure tanks of the vehicle.

Figure 10 shows the position of the vehicle battery, the air pressure tanks and the exhaust pipe of a fire engine. The height of these limiting parts of the vehicle determines the availability of the vehicle in case of beyond design basis flooding. A flood level higher than the limiting parts must to be assumed to lead to the unavailability of the vehicle. Private vehicles of plant staff or members of the on-site fire brigade are expected to already be unavailable.

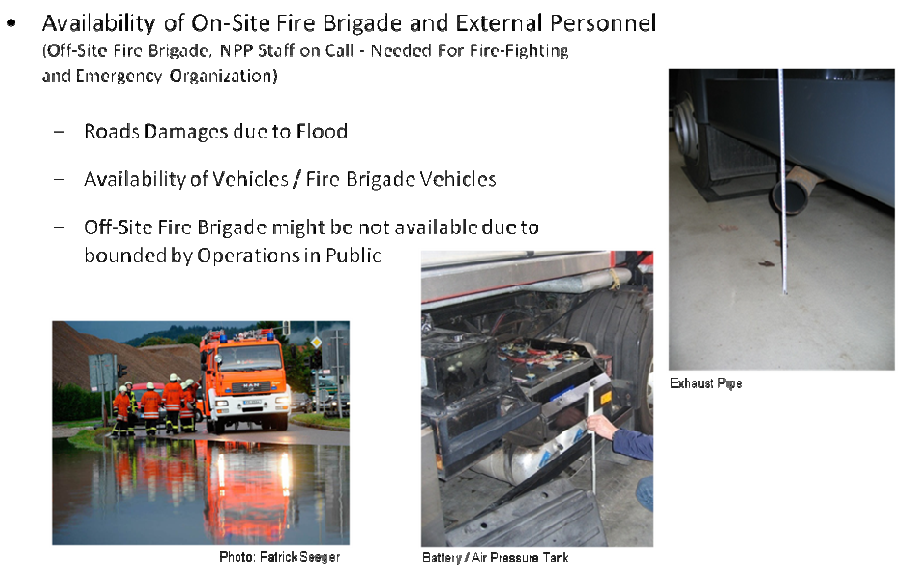

Figure 10 OP - Availability of on-site fire brigade and external personnel

The position of limiting parts of fire brigade vehicles should be optimized to ensure maximum availability of the vehicles. The transportation of on-site emergency organization staff and fire brigade team members should be considered and optimized.

OP for the availability of the fire water supply system

The fire water supply system is needed for automatic extinguishing systems, manual fire fighting and emergency measures, e.g. feed-in to the fuel element storage pool. In contrary to the results after an earthquake, no fracture of the supply pipes is expected following a beyond design basis flood.

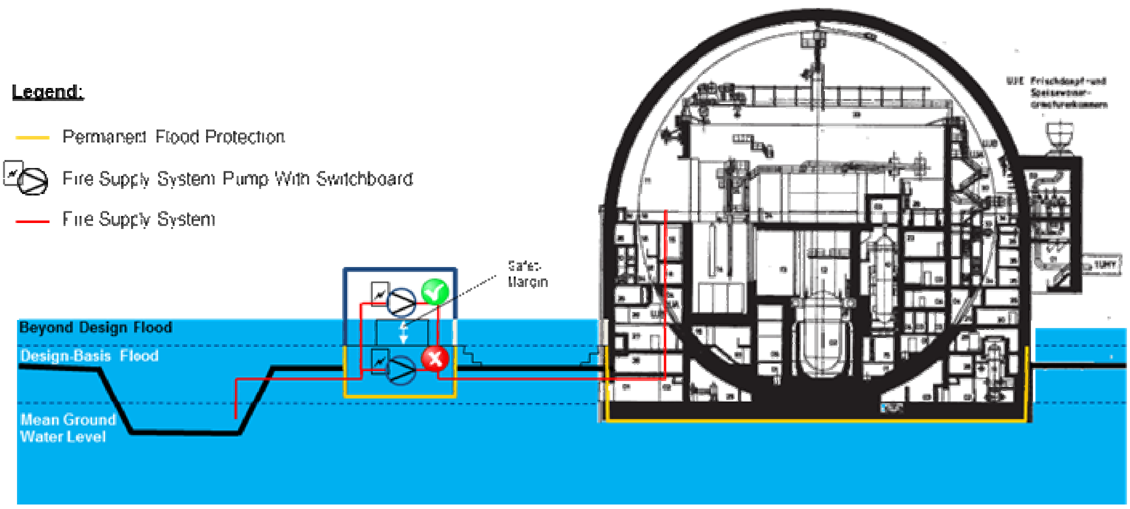

Figure 11 Availability of the fire water supply system

As shown in Figure 11, the availability of the fire water supply system depends on the position of the water pumps with the suitable switchboard. In case of a design basis flood, the water pumps will be protected by the design of the building. In case of a beyond design basis flood, the building might be flooded and the water pumps might not be functionally capable if there is insufficient (safety) margin on the position of the water pump and the switchboard.

To ensure the function of the fire supply system, the position of the fire supply system pumps should guarantee a suitable safety margin.

OP for quick access of building entrances in case of flooded and/or blocked physical protection doors

As shown in Figure 12, there are three main aspects to be considered in case of a beyond design basis flood. First, the electrical function of the door could fail. Secondly, the use of the door keys could be hindered by the fact that the door locks might be submerged. Thirdly, the doors might not be able to be opened due to the pressure of the floodwater.

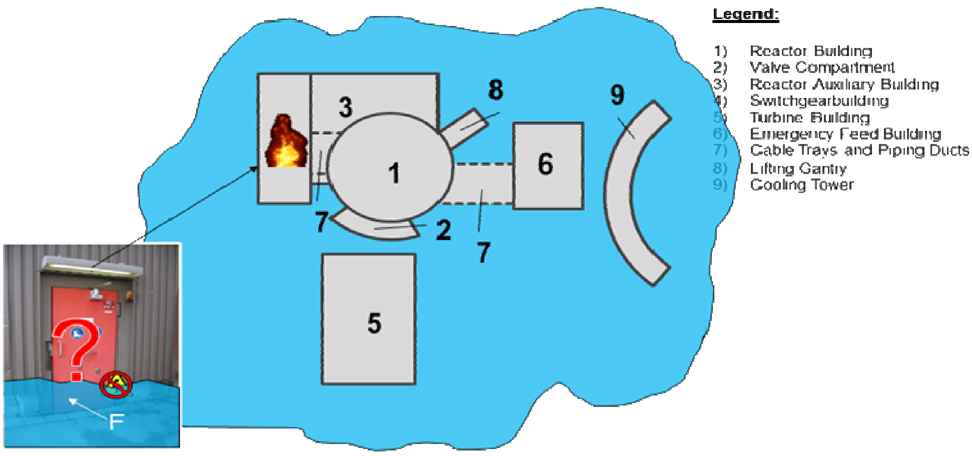

Figure 12 Flooded and/or blocked physical protection doors

The position of electrical parts and door locks should be reviewed and optimized if necessary. Further measures might be necessary to ensure the required function of doors due to pressure of the floodwater.

CONCLUSIONS

This paper provides findings and OP relevant to fire safety for compliance with the nuclear protection goals control of reactivity, cooling of fuel assemblies, confinement of radioactive materials and limitation of radiation exposure. The results presented are restricted to fire safety aspects. However, compliance with the nuclear protection goals will also be ensured through other areas, such as systems engineering and radiation protection. Disaster management was not considered in the SSI.

Results regarding the structural protection due the impact from aircraft, as well as NPP security protection, are outside the scope of this paper.

Regarding the different types of existing fire protection measures (structural, equipment-related, operational and manual fire fighting as shown in figure 5) at a nuclear power plant, the findings and optimization potential shown in this paper are only applicable to manual fire fighting. Therefore manual fire fighting has a special importance for future improvements.

The results clearly demonstrate that small improvements of buildings, vehicles or systems could lead to higher availability and efficiency of safety functions and therefore greater likelihood of satisfying the nuclear protection goals in case of a beyond design basis event. Due to the identified optimization potential, safety margins of safety related systems can be defined. By this means it is possible to further increase the robustness of a nuclear power plant against the effects of beyond design basis events.

The results of this paper may be applicable for nuclear power plants outside of Germany and may lead to similar safety inspections at nuclear plants all over the world. Fire protection measures should be constantly reviewed to ensure the highest practicable availability of buildings, systems, and equipment.

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Hinweis: Änderung durch Art. 1 G v. 31.7.2011 I 1704 (Nr. 43) textlich nachgewiesen, dokumentarisch noch nicht bearbeitet

6 Seminar Conclusions and Outlook

The 12th International Seminar on ‘Fire Safety in Nuclear Power Plants and Installations’ demonstrated significant progress in nuclear fire safety with respect to experiments, analysis, and assessment. However, there are still challenges since the knowledge on several fire related phenomena is not yet as mature as expected two decades ago and due to the continuous need of further enhancing the analytical tools.

The following conclusions have been drawn from the seminar:

With respect to research activities focusing on real case fire scenarios in nuclear installations´ fire safety there is a strong expert opinion that the actually ongoing experimental nuclear fire research programs provide valuable insights on the behavior of the nuclear facility in the case of fire. A typical example is the international OECD PRISME Project, which is being continued in a second phase with as far as possible realistic nuclear power plant specific scenarios to close still existing knowledge gaps. The intended experiments should help to solve specific questions important for the analysis, such as the consideration of under-ventilated conditions, the effects of specific conditions by forced ventilation, or the effects of fire extinguishing systems on fire sequence course.

State-of-the-art fire simulation codes are meanwhile well established as analytical tools for deterministic and probabilistic assessment of fire hazards. The most recent fire modeling activities, e.g. simulations of the OECD PRISME experiments have however demonstrated that there is a continuous need for enhancing fire models, in particular with respect to modeling pyrolysis rates for different types of fuels and scenarios or for predicting fire suppression effects. In this context, sensitivity and uncertainty analysis are recommended.

The nuclear power plant operating experience indicated the significance of an impact by fires due to high energy arcing faults (HEAF), in particular with regard to specific components and structures. This failure mechanism with explosive component faults may significantly impair the fire protection features of a nuclear facility and thus have the potential of degrading nuclear safety. The fire expert community identified the need for in-depth experimental investigations on HEAF fires and initiated an international OECD Project, where such destructive tests will be carried out.

It was again highlighted that, in particular for Fire PSA, plant specific data on fire events as well as on the reliability of different protection means and human actions in case of such events are evident. The international database on fire events in nuclear power plants OECD FIRE represents a valuable tool for collecting fire specific operating experience which in future may also be used for fire occurrence frequency and fire suppression probability calculations.

Last, but not least, the nuclear accidents as a result of earthquake and Tsunami.at the Fukushima Daiichi nuclear power station in Japan in March 2011 revealed further questions, in particular on combinations of fires with external or internal hazards. In principle, the fire protection standards and design cover these aspects and especially seismically induced fires and explosions. However, the operating experience has shown vulnerabilities in this respect. As already observed in the previous, 11th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' the most recent earthquakes having impaired the safety of nuclear installations have re-started the discussion on fires consequential to external hazards between regulators and analysts in several countries. The existing regulations and standards have to be adapted to the state-of-the art in this respect.

For beyond design basis accidents with consequential fires there is nevertheless a need for further investigations. The results from international stress tests and other national in-depth investigations resulting from the above mentioned nuclear accidents will show in the near future, where research and analytical effort is needed and what could be done to improve the safety of operating installations further. However, the more frequent fire events have to be analyzed with highest priority.

The participants from all over the world, representing all parties involved in nuclear fire safety, nuclear industry as well as regulators, research institutions as well as technical expert and support organizations (TSO), strongly emphasized the value and benefits of the information provided in this experts seminar to be shared inside the nuclear fire community. They expressed their wish of continuing this series of fire seminars on a regular basis in time intervals of at least two years.

The next, 13th seminar of this series is therefore planned to be conducted in late summer 2013 in conjunction with the 22nd 'International Conference on Structural Mechanics In Reactor Technology' (SMiRT 22), which will take place in San Francisco, CA (USA) in August 2013 (cf. http://www.smirt22.org/).

Attachment

CD of the 12th International Seminar on ‘Fire Safety in Nuclear Power Plants and Installations’ held as Pre-Conference Seminar of SMiRT21