SMiRT 23
14th International Seminar on FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS

Salford, United Kingdom
August 17-18, 2015

GRS - A - 3845

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH

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Autor / Authors:
Dr. Marina Röwekamp, GRS (Ed.)
Dr. Heinz-Peter Berg, BfS

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Kurzfassung

Im Rahmen des vom Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit (BMUB) beauftragten Vorhabens 3614R01575 wurde im August 2015 das mittlerweile vierzehnte internationale Seminar “Fire Safety in Nuclear Power Plants and Installations“ als Post-Conference Seminar der 23nd International Conference on Structural Mechanics In Reactor Technology (SMiRT 23) in Salford, Großbritannien veranstaltet.

Die vorliegenden Proceedings des Seminars enthalten alle einundzwanzig Fachbeiträge des zweitägigen Seminars mit insgesamt fünfundfünfzig Teilnehmern aus zehn Ländern aus Asien, Europa und Amerika.

Abstract

In the frame of the project 3614R01575 funded by the German Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit, BMUB) the meanwhile fourrteenth international seminar on "Fire Safety in Nuclear Power Plants and Installations" has been conducted as Post-Conference Seminar of the 23rd International Conference on Structural Mechanics In Reactor Technology (SMiRT 23) in Salford, United Kingdom in August 2015.

The following seminar proceedings contain the entire twenty-one technical contributions to the two days seminar with in total fifty-five participants from ten countries in Asia, Europe and America.

Contents

1 Foreword and Introduction

The meanwhile 14th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' was held as Post-conference Seminar of the 23rd International Conference on Structural Mechanics In Reactor Technology (SMiRT 23) in Salford, United Kingdom in August 2015. In total fifty-five participants from Argentina, Belgium, Finland, France, Germany, Japan, the Netherlands, Spain, United Kingdom and the United States of America followed the twenty-one presentations in the different scientific sessions and participated actively in a final short round table expert discussion on future challenges of fire safety for new built reactors as well as for operating plants and other nuclear facilities at the end of the seminar.

The two-day expert seminar started with a session on regulatory issues with respect to fire safety in nuclear installations and the corresponding standards and guidance. Presentations were given on the expectations of regulatory bodies on the assessment of plant internal fires in the frame of the design of future power reactors but also on the regulator's viewpoint how to apply the defence-in-depth concept to existing nuclear power plants as well as to reactors to be built. Moreover, three presentation highlighted recent enhancements in national nuclear regulations with respect to fire safety taking as far as possible also lessons learned from the Fukushima reactor accidents into account. The approaches for assessing nuclear safety provided in such advanced standards and guidelines are not limited to deterministic analyses but also cover PSA as a supplementary tool for assessing the contribution of fires to the overall risk of a nuclear facility.

The second expert session was focused on the design of nuclear power plants according to the state-of-the-art and the implications of fire hazards to the design. Design approaches and how these address fire protection were presented, from the EPR (*E*uropean *P*ressurized Water Reactor) design by EDF NNB GenCo up to the Japanese ABWR (*A*dvanced Boiling Water *R*eactor) concept by GE Hitachi, both for new reactors in the United Kingdom. An additional focus was on the role of passive fire protection means (e.g. fire barriers) and its significance for the plant design. In this context, recent developments were presented.

A non-negligible part of the seminar with two sessions was devoted to fire research activities in respect of nuclear installations. The focus of the first of these sessions was on experimental fire research for nuclear facilities. Experts from well-known research institutions, such as NIST (*N*ational *I*nstitute of *S*tandards and *T*echnology) from the United States, the French IRSN (*I*nstitut de *R*adioprotection et de *S*ûreté *N*ucléaire) and the German iBMB (*I*nstitut für *B*austoffe, *M*assivbau und *B*randschutz) of Braunschweig University of Technology provide insight from experiments being carried out in the recent past and/or still ongoing research programs. While the US presentation was on Thermal Effects of High Energy Arc Faults (HEAF), which have been observed in nuclear installations as non-negligible contributors to the fire related risk, the other two presentations underlined the significance of cable fires and their behaviour under different conditions and effects of the corresponding protection features for limiting the consequences of these fires to safety.

The second research session highlighted recent activities with regard to fire safety analysis and modelling. The progress in predicting different types of nuclear specific fire scenarios over the last decades is enormous. However, fires are such complex phenomena with high uncertainties in their behaviour over time that the modelling has not yet reached the same high level of confidence as nuclear simulations in other areas, e.g. modelling thermal hydraulics. Therefore, the progress to date in fire simulations is important for increasing reliability and traceability of the models for nuclear power plants as well as for other facilities of the nuclear fuel cycle. In this direction the presentations on ventilation effects, such as smoke propagation through vents separating rooms, or on simulations of mechanically ventilated rooms typical for nuclear stations by Large Eddy Simulations provided valuable results. In this context, sensitivity studies were also addressed making the auditory aware that the simulation results strongly depend on a variety of sensitive parameters.

Another major aspect of the seminar was the operating experience with fires and fire related events and the lessons learned from those. On an international level, several databases collecting information on fire incidents in nuclear installations are meanwhile available, in particular the OECD FIRE Database recording fire event data from nuclear power plants in member states in a manner that the needs of the analysts from licensees, regulatory bodies and TSOs (*T*echnical Safety *O*rganisations) for deterministic safety analyses but also for probabilistic risk assessment (PRA) are as far as practically possible met. Further presentations provided insights from national investigations, e.g. on aspects relevant to fire safety analyses in Spanish nuclear power stations, the human factor in prevention of fire events imposing nuclear safety, the additional aspects of fire protection for nuclear power plants under longer term safe shutdown conditions before decommissioning, an, last but not least, some lessons learned from feedback of recent fire events in operating reactors.

From the first seminars of this series starting in 1987 when the safety significance of fires in nuclear reactors had just been recognized up to today fire safety in nuclear facilities has significantly increased. This does concern the plants' design in general and of structures, systems and components (SSCs) important to safety in particular, as well as the operation of such installations, but also all areas of assessment, inspection and maintenance. Over more than three decades, methodological approaches for assessing the fire risk and the respective analytical tools have been and are still permanently being improved and extended.

However, further challenges do arise, affecting the examination of fire hazards and their consequences in nuclear installations. Nuclear fire risk assessment also requires continuous research and development on a theoretical as well as an experimental basis. The new as well as updated and enhanced methodologies and analytical approaches need verification and explicit validation for the areas of application. In this context, it has to be mentioned that the existing data have also to be permanently updated and adapted to the state-of-the art.

The seminar topics demonstrated clearly the very broad scope of the issues related to fire safety in nuclear installations. The presentations and discussions indicated that fires are still a "hot" topic and need to be addressed not only as single events, but also in correlation with other internal and external hazards.

One main goal of this fourteenth seminar on 'Fire Safety in Nuclear Power Plants and Installations' was to reflect the actual challenges and to provide insights in how to resolve fire safety issues, for existing plants as well as for reactor facilities to be built and safety operated in the future.

The seminar was hosted with great hospitality by Anastasios Alexiou und Geraint Williams from the Office of Nuclear regulation (ONR) in the "Old Fire Station" of Salford University, United Kingdom. The organizers are indebted to the invitation and support by the hosts during the two days seminar.

Moreover, the organizers want to thank all speakers, authors and chairpersons but also all the other participants for their very active and fruitful participation as well as for the valuable, high level contributions during this 14th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' which made this venue again a very successful one.

The next, 15th seminar of this series is intended to be held as SMiRT 24 Post-conference Seminar in Busan, Korea in late summer 2017.

Dr. Marina Röwekamp

- Scientific Chairperson and Organizer -

2 Seminar Agenda

Monday, August 17, 2015

09:30 h	***Introduction by Hosts***	A. Alexiou, G. Williams	ONR, United Kingdom
	Welcome	M. Röwekamp	GRS, Germany
	Regulation, Standards and Guidelines	***Chairperson: G. Williams (ONR)***	
09:45 h	Regulatory Expectations on Internal Hazards Assessment With Particular Focus on Fire	A. Alexiou	ONR, United Kingdom
10:15 h	Regulatory Point of View on Defence-in-depth Approach to Fire Protection in Nuclear Power Plants	S. Rinta-Filppula, et al.	STUK, Finland
10:45 h	***Coffee Break***		
11:15 h	Recent Update of the German Nuclear Fire Protection Standards KTA 2101, Part 1 to 3	B. Elsche, M. Röwekamp W. Neugebauer	e.on Kernkraft, Germany GRS, Germany AREVA NP, Germany
12:00 h	Recent Amendments in the KTA 2101.2 Fire Barrier Resistance Rating Method for German Nuclear Power Plants and Comparison to the Eurocode T-equivalent Method	B. Forell	GRS, Germany
12:30 h	***Lunch Break***		
	Regulation, Standards and Guidelines (continued)	***Chairperson: G. Williams (ONR)***	
14:00 h	Enhancements in PSA Regulation and Guidance on Fire Risk Analysis for Nuclear Power Plants	H.-P. Berg	BfS, Germany
	Fire Protection in the Design of Nuclear Installations	***Chairperson: H.-P. Berg (Germany***	
14:30 h	Internal Fire Protection Analysis for the United Kingdom EPR Design	A. Laïd, et al.	EDF NNB GenCo, United Kingdom
15:00 h	Overview of Internal Fire Hazards Aspects of ABWR Design for United Kingdom	K. Yoshikawa, et al.	Hitachi-GE, Japan
15:30 h	***Coffee Break***		

16:00 h	Passive Fire Protection: Role and Evolutions	T. Cerosky, et al.	Nuvia, France
16:30 h	Recent Enhancements of the OECD FIRE Database	M. Roewekamp, M. Lehto	GRS, Germany STUK, Finland
17:00 h	***Adjourn of the first day***		
19:00 h	***Hosted Dinner***	All participants enscribed and spouses	

Tuesday, August 18, 2015

	Experimental Fire Research	***Chairperson:*** ***M. Röwekamp (Germany)***	
09:00 h	Characterizing the Thermal Effects of High Energy Arc Faults	A. Putorti, et al.	NIST, USA
09:30 h	Assessment of the Burning Behaviour of Protected and Unprotected Cables and Cable Trays in Nuclear Installations Using Small- and Large-scale Experiments	M. Siemon, et al.	iBMB, Germany
10:00 h	Fully Predictive Simulations of Real-scale Cable Tray Fire Based on Small-scale Laboratory Experiments	F. Bonte, et al.	BEL V, Belgium
10:30 h	***Coffee Break***		
	Fire Safety Analysis and Modelling	***Chairperson:*** ***M. Röwekamp (Germany)***	
11:00 h	Experimental and Numerical Study of Smoke Propagation Through a Vent Separating Two Mechanically Ventilated Rooms	L. Audouin, et al.	IRSN, France
11:30 h	Large Eddy Simulation of a Mechanically Ventilated Compartment Fire for Nuclear Applications	J. Wen, et al.	University of Warwick, United Kingdom
12:00 h	Global Sensitivity Analysis Using Emulators, With an Example Analysis of Large Fire Plumes Based on FDS Simulations	A. Kelsey	HSL, United Kingdom
12:30 h	***Lunch Break***		
	Fire Safety Analysis and Modelling (continued)	***Chairperson:*** ***M. Röwekamp (Germany)***	
13:30 h	Sensitivity Analysis of FDS 6 Results for Nuclear Power Plants	E. Puente, at al.	University of Cantabria, Spain

14:00 h	Focus on the Studies in Support of Fire Safety Analysis: IRSN Modelling Approach for Nuclear Fuel Facilities	T. Vinot, et al.	IRSN, France
	Operating Experience and Lessons Learned	***Chairperson: A. Alexiou (United Kingdom)***	
14:30 h	Fire Analysis: Relevant Aspects from Spanish Nuclear Power Plants Experience	T. Villar Sánchez, P. Fernandez Ramos	Empresarios Agrupados, Spain
15:00 h	***Coffee Break***		
15.20 h	The Human Dimension: Improving Fire Safety in Nuclear Power Plants by Improving Awareness of Fire Hazards and Influencing Behaviours	G. Williams, et al.	ONR, United Kingdom
15:50 h	Defence-In-Depth Strategy of Fire Protection and Its Relevance after Final Shutdown	M. Beesen	TÜV SÜD, Germany
16:20 h	Feedback from Recent Operating Experience in Nuclear Power Plants Regarding Fire Safety	B. Forell	GRS, Germany
16:40 h	**Panel Discussion**	***Chairperson: H.-P. Berg (Germany)***	
	Panel participants:	A. Alexiou H.-P. Berg L. Kuriene M. Röwekamp G. Williams	
17:10 h	***Seminar Adjourn***		

3 Seminar Contributions

In the following, the seminar contributions prepared for the 14th International Seminar on ‘Fire Safety in Nuclear Power Plants and Installations’ held as Post-conference Seminar of the 23rd International Conference on Structural Mechanics In Reactor Technology (SMiRT 23) are provided in the order of their presentation in the seminar.

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Regulatory Expectations on Internal Hazards Assessment With Particular Focus on Fire

Dr Anastasios (Tasos) Alexiou

14th SMIRT Post-Conference Seminar on
Fire Safety in Nuclear Power Plants and Installations
Salford, United Kingdom; August 17-18, 2015

Contents

- Definition of Internal Hazards.
- ONR's Regulatory Approach.
- Key Safety Assessment Principles.
- Internal Hazards Safety Case.
- Internal Hazards Analysis.
- Fire Severity Quantification Approach.
- Safety Led Design for Internal Hazards.
- Claims, Arguments and Evidence Principle.
- Links To Other Topic Areas.
- Intelligent Customer Capability.
- Key Messages.

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ONR's Regulatory Approach

- ONR uses a goal setting regulatory approach to ensure that the risks to nuclear safety from internal hazards are ALARP.
- ONR has developed its own Safety Assessment Principles, (SAPs) which are used to judge the adequacy of licensees' nuclear safety cases.
- The SAPs were revised in 2014 to take account of:
  - Lessons Learned from Fukushima;
  - IAEA SSR 2/1 and SSR 2/2;
  - WENRA Safety Levels;
  - ONR's Recent Experience.
- The SAPs EHA series specifically address External and Internal Hazards (EHA.1 to EHA.19), however **the SAPs are applied on a holistic basis for hazards.**

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Definition of Internal Hazards

Definition of Internal Hazards from the Safety Assessment Principles for Nuclear Facilities

"Internal hazards are those hazards to the facility or its structures, systems and components that originate within the site boundary and over which the dutyholder has control in some form. The term is usually limited to apply to hazards external to the process, in the case of nuclear chemical plant, or external to the primary circuit in the case of power reactors. Internal hazards include internal flooding, fire, toxic gas release, dropped loads or impact and explosion/missiles. Again, this list is not exhaustive " (Para. 229).

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Key Safety Assessment Principles

▪ Hazard Identification:	EHA.1, 14,
▪ Hazards Analysis:	EHA.3, 4, 5, 6, 7, 13, 14, 16, 18, 19
▪ Engineering Key Principles:	EKP.1, 2, 3, 4, 5
▪ Design for Reliability:	EDR.1, 2, 3, 4
▪ Layout:	ELO.4
▪ Safety Systems:	ESS.1, 18
▪ Civil Engineering - Design:	ECE.6
▪ Equipment Qualification:	EQU.1
▪ Safety Classification & Standards	ECS.1, 2
▪ Human Factors:	EHF.3
▪ Fault Analysis:	FA.2, 3, 5, 6
▪ Validity of Data and Models:	AV.2, 3, 4, 5, 6
▪ Assessment of Safety Cases:	SC.1, 4

Note: The above list is not exhaustive

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SAPs – Hazard Identification

EHA 1 & 14

- An effective process should be applied to identify and characterise all external and internal hazards that could affect the safety of the facility.
- Sources that could give rise to fire, explosion, missiles, toxic gas release, collapsing or falling loads, pipe failure effects, or internal and external flooding should be identified, quantified and analysed within the safety case.
- The identification process should include reasonably foreseeable combinations of independently occurring hazards, causally-related hazards and consequential events resulting from a common initiating event (see Principle FA.2).

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SAPs – Hazard Analysis

EHA 5 & 6

- Analysis of design basis events should assume the event occurs simultaneously with the facility's most adverse permitted operating state (see paragraph 631 c) and d)).
- The analysis should include hazard analysis to:
  - (a) identify the potential impact of the hazard on the facility's structures, systems and components, and in particular its safety systems;
  - (b) determine the need for segregation, diversity and redundancy of plant and equipment and the location of barriers to limit this impact; and
  - (c) determine the safety functions (e.g. the withstand capability) to be provided by such barriers.
- The effects of internal and external hazards that could affect the safety of the facility should be analysed. The analysis should take into account hazard combinations, simultaneous effects, common cause failures, defence in depth and consequential effects.

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SAPs – Key Engineering Principles

EKP.1, 3, 4 & 5

- **Inherent Safety** – The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
- **Defence in Depth** - Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression .
- **Safety Functions** - The safety function(s) to be delivered within the facility should be identified by a structured analysis.
- **Safety Measures** - Safety measures should be identified to deliver the required safety function(s).

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SAPs - Safety Classification and Standards

ECS.1 & 2

- **Safety categorisation** - The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be identified and then categorised based on their significance with regard to safety.
- **Safety classification of structures, systems and components** - Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance to safety.

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SAPs – Design for Reliability

EDR.1, 2 3, 4

- **Failure to Safety** – Due account should be taken of the need for structures, systems and components to be designed to be inherently safe, or to fail in a safe manner, and potential failure modes should be identified, using a formal analysis where appropriate.
- **Redundancy, diversity and segregation** - Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components
- **Common cause failure** - Common cause failure (CCF) should be addressed explicitly where a structure, system or component employs redundant or diverse components, measurements or actions to provide high reliability.
- **Single failure criterion** - During any normally permissible state of plant availability, no single random failure, assumed to occur anywhere within the systems provided to secure a safety function, should prevent the performance of that safety function.

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SAPs – Layout & Safety Systems

ELO.4

- **Minimisation of the effects of incidents** - The design and layout of the site, its facilities (including enclosed plant), support facilities and services should be such that the effects of faults and accidents are minimised.

ESS.1 & 18

- **Provision of safety systems** - All nuclear facilities should be provided with safety systems that reduce the frequency or limit the consequences of fault sequences, and that achieve and maintain a defined stable, safe state.
- **Failure independence** - No design basis event should disable a safety system.

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SAPs – Assurance of Validity of Data and Models

AV.2, 4 & 6

- **Calculation methods** - Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.

- **Computer models** - Computer models and datasets used in support of the safety analysis should be developed, maintained and applied in accordance with quality management procedures.

- **Sensitivity studies** - Studies should be carried out to determine the sensitivity of the analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation.

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Internal Hazards Safety Case

- The safety case should consider all initiating events that could affect the performance of nuclear SSCs.
- Consideration should be given to coincident and consequential hazards.
- Be largely deterministic.
- Carried out on a conservative basis.
- Unmitigated consequences should be evaluated.
- Incorporate Design Basis Analysis (DBA).
- Include all fault sequences within the DBA, in line with SAP FA.6.

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Internal Hazards Safety Case (cont'd)

- Be subject to sensitivity analysis using PSA methods to identify plant and system vulnerabilities, if appropriate.
- Common cause failures affecting multiple trains of defence due to a single event, e.g. electrical fire, should be considered.
- The safety measures are required to be substantiated and should take account of:
  - Performance requirements;
  - Reliability data;
  - Use of appropriate design codes;
  - Relevant good practice.
- Demonstrate that the risk is ALARP.
- Hazards should be captured within the Fault Schedule.

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Internal Hazards Analysis

- Detailed Knowledge of the Plant and Site Layout:
  - Location of plant equipment;
  - Location of items important to safety;
  - Redundancy, diversity and reliability requirements of the items important to safety.
- Hazard Location and Severity Quantification:
  - Hazard identification;
  - Hazard severity;
  - Consequences of the hazard;
  - Hazard/fault combinations;
  - Verification and validation of analysis techniques.

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Internal Hazards Analysis (cont'd)

- Defence in Depth Provisions:

The safety case should demonstrate how the "defence in depth" philosophy has been applied to identify the appropriate control measures to:

  - Prevent the hazard.
  - Protect from the hazard:
    - Limit the severity of the hazard should it occur.
  - Mitigate the hazard:
    - Limit the consequence of the hazard should it occur and be severe.

- Safety Category and Safety Classification of Hazard Related SSCs
  - Safety function;
  - Performance requirements;
  - Standards.

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Fire Severity Quantification Approach

- Standard fire curves (BS 476, ISO 834).
- Time-equivalence approach.
- Parametric fire curves.
- Empirical models.
- Zone models.
- CFD models.

Questions to be Answered Before Use:

- Does it represent real fire scenario?
- Can the model answer how robust the nuclear structure is?
- How localised effects, overpressure, and adequacy of separation by the fire zoning approach will be calculated?
- What is the model limitation and validation?
- Does it reflect modern standards?
- Can you undertake sensitivity studies?
- What is the user dependency?

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Claims, Arguments and Evidence Principle

Safety cases are often structured on "Claims, Arguments and Evidence" principle. An example is given below:

Claim

- Nuclear significant hazard barriers in place to prevent failure of more than one redundant train due to the effects of an internal hazard.

Arguments

- Barrier construction capable to withstand the worst case unmitigated internal hazard scenario (including localised effects for fire).
- Limited penetrations.
- Doors and dampers designed to withstand internal hazards including the application of single failure criterion, as appropriate.
- Plant Layout.
- Redundancy.
- Administrative procedures, as appropriate.

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Claims, Arguments and Evidence Principle (cont'd)

Evidence

- Civil and structural analysis to support any arguments presented in relation to hazard loading (e.g. BS 476, BS EN 1992-1-2 and etc).
- Redundancy in damper provision to protect against single failure of an active component.
- Doors and dampers adequacy demonstrated through appropriate accredited test facility.
- Necessary to provide evidence against all potential hazards that could affect the barrier.

Note: For information only and not exhaustive or the only means by which to produce a safety case in this area.

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Claims, Arguments and Evidence Principle (cont'd)

Nuclear Significant Hazard Barriers (NSHB)

- Designed to protect against a number of internal hazards – not just fire:
  - Fire.
  - Explosion.
  - Missile.
  - Flood.
  - Pressure Part Failure:
    - Steam Release Environmental Effects;
    - Pipe Whip;
    - Missile;
    - Jet impact.
  - Dropped Load/Impact.
- The NSHB can also be in the form of enclosures.
- It is important to stress that barrier performance requires substantiation against the extent of the hazard in each case.

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Safety Led Design for Internal Hazards

Good Design Practice

- Inherently Safe Design (*avoid hazard rather than controlling*).
- Optimise Design Layout (*minimise effects of incidents*).
- Design for Reliability (*redundancy, diversity, segregation*).
- Failure Independent (*safety systems should not be disabled by an internal hazard*).
- Design for Fault Tolerance.
- Incorporation of Defence in Depth in the Design (*prevent internal hazards, limit severity, limit consequences*).
- Human Interactions Fully Understood and Supported.
- Ageing and Maintenance Requirements Consideration.
- Etc. ...

Good Justification

- Identification of Internal Hazards (*including simultaneous effects, common cause failure, combined consequential events*).
- Analysis (*largely deterministic, unmitigated consequences evaluated, single failure criterion, DBA, fault schedule, PSA*).
- Identification of Safety Functions and Measures.
- Safety Categorisation and Classification of Internal Hazards related SSCs.
- Substantiation (*performance requirements, reliability data, use of appropriate design codes, relevant good practice*).
- Demonstrate Risk is ALARP
- Safety Case Justification should be Accurate, Objective and Demonstrably Complete.

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Links To Other Topic Areas

The internal hazards safety case should be developed based on up to date design information and input from other technical disciplines including:

- Engineering Disciplines such as Civil, Structural Integrity, Mechanical, Electrical, Control and Instrumentation.
- Conventional Fire Safety.
- Fault Studies.
- PSA.
- Human Factors.
- Reactor Chemistry.
- Radwaste and Decommissioning.
- Security.

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Intelligent Customer Capability

Demonstration of Intelligent Customer (IC) Capability is a key requirement.

It is defined by IAEA as:

"As an intelligent customer, in the context of nuclear safety, the management of the facility should know what is required, should fully understand the need for a contractor's services, should specify requirements, should supervise the work and should technically review the output before, during and after implementation.
The concept of intelligent customer relates to the attributes of an organisation rather than the capabilities of individual post holders."

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SQEP Personnel

Utilisation of SQEP Internal Hazards personnel is a key requirement:

- There is a requirement to demonstrate an adequate level of knowledge of nuclear safety structures, systems and components and how internal hazards can have a detrimental effect on the performance of those SSCs.
- The use of SQEP resource in the area of internal hazards ensures that the level of regulatory risk associated is minimised.

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Key Messages

- ONR's regulatory expectations on internal hazards are based on SAPs which guide regulatory decision making in the nuclear permissioning process.
- SAPs are in line with International Guidance (IAEA and WENRA).
- SAPs should be considered in totality.
- SAPs 2014 provide further clarity and in essence remain the same in this area.

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REGULATORY POINT OF VIEW ON DEFENSE IN DEPTH APPROACH TO FIRE PROTECTION IN NUCLEAR POWER PLANT

Samu Rinta-Filppula[1], Matti Lehto[1], Pekka Välikangas[1]

[1] Radiation and Nuclear Safety Authority STUK, Helsinki, Finland

ABSTRACT

The defense-in-depth (DiD) principle is a relatively new approach to fire protection design, even though DiD has been used in nuclear power plant (NPP) safety evaluation and design for decades (IAEA 75-INSAG-3, Rev. 1/INSAG-12). It is the main design criterion in fire protection in the latest edition of Finnish Radiation and Nuclear Safety Authority (STUK) issued guide YVL B.8 for the fire protection in nuclear facilities. The DiD approach to fire protection consists of four levels of defense: preventing the ignition of fires, detecting and extinguishing of ignited fires, preventing fire growth and spreading, confining the fire so that safety functions can be performed irrespective of the effects of the fire. The design of fire protection should take all these levels into account so that fire protection is well balanced and not dependent on a single fire protection factor or level of DiD.

Despite being central to the design of fire protection, corresponding evaluations of DiD are done according to more or less unambiguous methods. The main goal of this study is to start the development of such, as much as possible, unambiguous systematic and logical method. First issue then is to build a picture of how fire safety features are executed on different levels of DiD and what is the corresponding safety importance to NPP. The Loviisa NPP was studied as an example case due to a long history of fire safety improvements since commissioning in 1977. The improvements are sorted qualitatively by their means of fire safety impact and level of DiD approach to fire protection and general plant DiD. The correspondence between the two DiD principles is an interesting issue which is discussed in this paper. Finally, Fire PRA is used to determine the safety importance of the improvements.

The method proposed for the evaluation of DiD approach to fire protection is a combined ignition root cause analysis – event tree of fire scenario – consequential failure modes and effects analysis (FMEA) where the three analyses are performed successively for a given type of fire event. The challenge is to support experts focusing in certain technical domain and exchanging relevant information between these analyses. Ignition root cause analysis is performed to find the factors leading to a fire event. Fire propagation is then modeled in the event tree, where fire protection features are taken into account. FMEA is then performed based on the fire scenario extracted from the event tree. The last stage also includes analysis on fire spreading. Measuring the effect of compartmentalization and fire spreading on safety is challenging. Accuracy of the analysis tools and initial values used in the example case is discussed.

When the fire protection concept of new design is evaluated viewpoints are effects of failures and impairments in DiD on safety of the facility. For operating units viewpoints are the effects of improvements. The described method is focusing on the framework for evaluation of the DiD approach to fire protection, wherein the actual analysis tools are only discussed briefly using an example case.

INTRODUCTION

Public opinion is hindering the prospects for commissioning new nuclear power plants (NPPs) to reduce the production of greenhouse gases via traditional fossil fuel fired power plants. One of the main concerns for people with negative perceptions about nuclear power is nuclear safety. The recent Fukushima disaster is one of only three major accidents in the history of NPPs that have led to core degradation, but it's fresh in the minds of people.

The International Atomic Energy Agency (IAEA) promotes nuclear and radiation safety. It produces guidelines and demands to guarantee the safety of NPPs. All the member states, including Finland, adhere to these guidelines. The Finnish Radiation and Nuclear Safety Authority (STUK) is responsible for the oversight of NPPs in Finland. In addition to the IAEA guidelines STUK has published its own YVL-guides which are stricter in many respects.

Thorough design of plant processes and layout creates the basis for plant safety. It is guided by many safety principles given by IAEA. While nuclear safety is the main concern in the design and operation of NPPs, there will always remain a risk for an accident. Fire events are a type of internal hazards that can compromise safety of a plant and lead to accidents. It is estimated that fire risk can comprise up to half [**1**] of the total core damage risk. Fire safety, too, is based on layout design. The separation of different safety systems and their redundancies reduces the risk considerably.

Defense-in-depth (DiD) is a safety principle for NPPs [2], [3]. It means having several separate and independent barriers between hazard and severe accident. Even with one barrier failing, the next one should be able to prevent the situation from getting worse and eventually leading to the release of radioactive materials into the environment. A barrier in this case can be considered a physical literal barrier or a safety system that for example helps to cool the reactor core. DiD approach can also be applied to fire protection, as defined later in this paper. STUK has added the term in the latest edition of its YVL-guide B.8 "*Fire Protection at a Nuclear Facility*" [4]. This guide also demands certain assessments of fire safety. As a new term, a universally validated method with acceptance criteria for assessing whether DiD for fire protection is adequate does not yet exist. The goal of this paper along with a broader master thesis [5] is to open the discussion and possible research in the field of this assessment.

This paper along with master thesis looks retroactively into the Loviisa NPP and the fire safety improvements done there as an example case. In the first part, the improvements are filtered and classified according to their means of influence, whether they are structural or active improvements and which level of defense in depth they fall into. In the second part we research associated numerical values to improvements. Risk increase factors (RIF) derived from Fire PRA (*P*robabilistic *R*isk *A*ssessment) are used for this part. Along with the classification it ca be determined which level of DiD for fire protection has been improved the most since the commissioning of the plant. The third part of the research took a closer look into the assessment of DiD. In this part, STUK started to develop a method for the assessment of possible acceptance criteria.

DEFENSE-IN-DEPTH IN REGULATION

Defense-in-depth is an essential safety principle in the design of NPPs. Assuring the effectiveness of DiD is done by multiple independent and redundant levels of defense and applying reliable materials and systems. In the Finnish regulation DiD is defined in STUK YVL-guide B.1 [6]. Safety functions shall be assured through five successive levels. The first two levels are designed to prevent accidents, whereas the remaining ones are designed to protect the plant, operators and the environment from the adverse effects of an accident. The levels of DiD are:

1. Prevention
2. Control of anticipated operational occurrences
3. Control of accidents
4. Containment of release in a severe accident
5. Mitigation of consequences

The final objective for the DiD principle is to prevent releases of radioactive substances in excess of the limits set in Government Decree 717/2013 [7].

The defense-in-depth approach to fire protection can be considered as one of the many means for realizing defense-in-depth for the entire plant. As a term it is relatively new. It can be found in an official IAEA publication at least in 1992 [8]. (U.S. Nuclear Regulatory Commission (NRC) has mentioned the DiD approach to fire protection dating back to the 1980ies.) STUK included it in the new edition of YVL-guide B.8 "*Fire Protection at a Nuclear Facility*" in 2013 [4] as the main design principle. The four levels of DiD approach to fire protection are:

a. Prevention of ignition of a fire,
b. Rapid detection and extinguishing of ignited fires,
c. Prevention of fire growth and spreading of a fire,
d. Containment of a fire such that the facility's safety functions can be reliably performed irrespective of the effects of the fire [4].

The issues addressed in the levels of DiD are not new per se. The point of defense-in-depth is to bind together the basic principles for the design of fire protection. The new idea is the successive barrier point of view. It should be assured that even if one level of defense fails, others will ensure the safety of the plant.

Prevention of ignition is the first level of the DiD approach with respect to fire protection. Construction materials allowable for Class P1 buildings in accordance with the National Building Code of Finland (Parts E1 & E2) [9] are used to minimize the danger of ignition. Validated fire resistant materials must be used wherever possible. Electrical equipment and machines containing moving parts must be reliable. Some systems contain fluids (oils, hydrogen etc.) with low flashpoints. In spaces with such substances there should be leak detection systems and extra fire protection for the systems in question.

The second level of the DiD approach to fire protection demands that a fire shall be detected and extinguished rapidly. Detection is done by an automatic system that encompasses the whole plant. Fixed extinguishing systems should also encompass the whole plant, excluding justifiable exceptions. These have to be able to function automatically when needed. In addition, components with high fire loads can be protected with separate extinguishing systems.

The third level of the DiD approach for fire protection addresses prevention of fire growth and spreading. Primary means are separation of buildings and fire compartmentation. Compartmentation is done by storey (vertical compartmentation) and by use. Safety divisions shall be separated by structures having a fire resistance rating of at least EI-M 120 (120 min fire resistance). This includes doors and hatches which also have to be locked during normal plant operation. Other compartment boundaries must be established between spaces with different operational usages and for separating high fire loads. The separating structural elements shall fulfill the fire resistance class requirements of the National Building Code of Finland [9]. Other needs in order to prevent fire spreading are ventilation control and smoke extraction.

The fourth level of DiD approach to fire protection ensures the functionality of the plant despite the effects of a fire. Main control room and emergency control rooms shall be separated by locations safe from fire risks. They shall represent their own fire compartments and they shall have isolated air conditioning systems. By our interpretation the fourth level of DiD approach to fire protection also handles functions that are implemented to improve plant safety, such as redundant systems and back-up power supply systems.

There is an analogy between nuclear safety DiD principle and DiD approach with respect to fire protection. The design of NPPs is a complex process with all the safety guides and demands on top of the complexity of the nuclear process itself. Thus the links between different safety features may not seem clear to an outsider. Instinctively one might think that the fire protection part precedes the nuclear safety DiD concept so that when fire protection is taken care of there will be no initiating event or, if it fails, the problem simply moves to general plant safety. However, this is not the case. The relations between the different levels of these two principles are intertwined. Basically it boils down to the following four questions: Can a fire in the compartment under investigation cause an initiating event? Are there components of safety systems that can be damaged by fire? Are there components of safety systems in the adjacent compartments? Can the fire event jeopardize performing of safety functions in any other way? This analysis yields a connection between all of the first three levels of DiD approach to fire protection and all of the first three levels of the nuclear safety DiD concept. In addition, the fourth level of DiD of fire protection has a connection to levels 2-4 of the nuclear safety DiD concept.

Requirements for the Assessment of Fire Safety

One of the main drivers for this research was the fact that DiD is the basis of fire protection design, but there is no defined method for assessing it. STUK Guide YVL B.8 [4] states of assessment of fire safety:

"To verify the adequate implementation of the defense in depth approach to fire protection, the following fire hazard analyses shall be conducted:

a. fire hazard analyses of the nuclear facility by deterministic, generally approved and experimentally verified methods such as

- *structural (FHA) and functional (FFHA, FHFA) fire hazard analyses*
- *fire simulation analyses to evaluate fire development and the ambient effects of fire, temperature increase in particular,*
- *analyses of heating, load-bearing capacity and integrity of load-bearing and separating structures*
- *analyses or calculations of temperature increase in the room or object of study, such as component temperature increase*

b. in addition to the above, for a nuclear power plant, a probabilistic fire risk assessment, a fire PRA (STUK Guide YVL A.7 [10])."

All of these are separate analyses, which do not really answer the question of the adequacy of defense-in-depth. They are, however, great tools for assessing fire safety in general.

Key attribute of successful DiD is balance. All the aspects must be taken into consideration and fire protection cannot rely on any single feature. According to STUK Guide YVL B.8 [4], when evaluating the implementation of the DiD approach to fire protection, one has to assume failures or impairments to plant fire protection, and remaining fire protection measures will have to compensate the deficiencies (e.g. failures to active fire protection measures and impairments to fire compartmentation, such as open fire doors). Normally, all equipment in the fire compartment shall be assumed to fail due to a fire. This means that within a safety division several systems may fail, but adequate safety functions must be possible to perform in any case. The assumption of losing the entire compartment in case of fire gives conservative result of the analyses, but is inaccurate in assessing specific fire protection characteristics inside the compartment.

RESEARCH METHODS AND GOALS

As stated earlier, this paper digs into the DiD approach to fire protection in three phases. The study is limited to threats originating during power operation. First STUK did a qualitative classification of fire safety improvements completed at the Loviisa NPP. Then a quantitative study of the safety importance of the improvements and corresponding levels of the two principles of DiD was made. Finally, the third part of the study tackled the assessment methodology for the DiD approach to fire protection, which is the most important part with long-term implications. The case study of the Loviisa NPP differs from future uses of this assessment technique: At first improvements were studied retrospectively, whereas in the future the targets shall be specific factors contributing to characteristics of a fire protection concept, when the concept is to be approved for new plants.

Classification of Improvements

In the first part of the research, the fire safety improvements connected to the turbine hall of the Loviisa NPP were studied and assessed in the cadre of defense-in-depth. Improvements were picked out from the internal STUK report ‘Safety Improvements of Loviisa NPP’. In this part, the improvements were classified by their attributes. Means of fire safety impact were determined at first. Based on that, improvements were sorted to structural and active fire protection and further on different levels of DiD regarding both fire protection and plant safety. The goal of this classification was to get a common understanding of the improvement landscape and to see if some aspects of fire protection are being over- or underrepresented. This also provided some information on the relationship between the two principles of DiD.

Probabilistic Study

The second part of the study assigned credit to the improvements’ impact on fire safety. The source material for this part was the Fire PRA of the Loviisa NPP. The Fire PRA was categorized by fire induced initiating events (IE), which were defined separately for each fire compartment (or room) and fire spreading events and given as input for the fire scenarios in the Fire PRA model. Therefore, the Fire PRA model did not show the failure combinations leading to different IEs. These IEs lead to core damage according to minimal cut sets (MCS) containing also the unavailability of safety systems needed to mitigate the IE in question. Along with the fire scenario frequency this gives the core damage frequency (CDF) caused by the fire according to each MCS.

Rooms of the plant are analyzed in the Fire PRA based on the IEs identified which are possibly induced by fires, the fire scenarios defined, and the possible fire induced failures of components and cables of systems that are needed to mitigate the IE identified. In case a fire in a room can induce different IEs, their combinations need to be handled as well.

Based on this knowledge, the safety improvements are sought from the Fire PRA data and their value to fire safety can be determined. The results from the first part are utilized in grouping the quantifiable credit the improvements get on different levels of DiD. The importance of different improvements is derived from Fire PRA results as risk increase factors (RIF). The second research phase tells us if one improvement or aspect of DiD is superior in impact compared to others. This is important, as it is required from DiD to be balanced.

Assessment of the Defense-in-depth Approach for Fire Protection

In the final part of the research, the discussion about assessing DiD approach to fire protection has been opened A three-step method is proposed. It consists of ignition root cause analysis, fire event tree analysis and consequential failure modes and effects analysis (FMEA). The three analyses are performed successively for a given fire scenario with the goal of identifying the significance of all the fire protection features. The three-step way of thinking is providing the framework where the different aspects of DiD can be quantitatively compared to each other. The actual analysis tools utilized in this method are nothing new, but are utilized mostly as stand-alone analyses of certain specific issues. This method has been first applied to the assessment of an oil pool fire in the turbine hall of the Loviisa NPP in the accompanying master thesis [5], which will be published in the near future, at first in Finnish.

The ultimate goal of the research started is to have an assessment method, which is universal and has clearly defined acceptance criteria for the DiD approach with respect to fire protection. For now, until a universal method is developed, DiD is evaluated too subjectively. It has been recognized that, in the frame of our study we probably would not meet this goal, but even to nudge the research in the right direction and open the scientific debate would be sufficient.

CASE STUDY: LOVIISA NUCLEAR POWER PLANT

The Loviisa NPP constitutes of two modified VVER-440/230 reactors commissioned in 1977 and 1980. The reactors have been uprated and currently produce at 510 MW_e each. Loviisa NPP is unique in design because the Soviet VVER-440/230 type reactors did not fulfill the Finnish safety regulations as they were. Additional safety features and Western technology were applied to the original Soviet design configuration.

The plant still contained safety shortcomings at the time of commissioning, which led to needs for safety improvements. The shortcomings were largely due to weaknesses in layout design, but the low grade of redundancy and diversity was also a problem. With the modifications, the plant became more complex. There are two safety trains for most of the safety systems, although some active components are doubled. Spatial separation of systems and compartmentation were partially incomplete and cable routings of the two redundancies could not be fully separated throughout the plant. These factors obviously increase the fire risk, for example the feedwater systems were vulnerable and could be lost in a fire event in the turbine hall. The turbine hall was chosen as the primary subject for the research by STUK.

Turbine Hall

The turbine hall contains four turbine generators, two for each reactor unit. In the original layout, the turbine hall and the feedwater tank compartment, which is located on an upper floor compared to the turbines, formed a common fire compartment. One of the main fire safety improvements of the entire plant has been the construction of a fire wall (called upper part of B-line wall) to separate the feedwater compartment from the turbine hall. The original lower part of the B-line wall also protects the control building from turbine hall fire events. Still, the remainder of the turbine hall contains safety related equipment and a fire in the turbine hall poses a threat to plant safety. Three connections between turbine hall, feedwater tank compartment and control building has been studied in the frame of fire safety improvements.

The main fire loads in the turbine hall consist of oil in the turbine lubrication and control systems, hydrogen used for cooling the generator and cable insulations. Hydrogen also carries a risk of explosion. Main ignition sources are hot surfaces, electric equipment and rotating machines. The turbine hall poses a problem to fire protection design as the complex layout includes several floors below the main floor where the turbines are located. In case of an oil leak, the oil could flow down to all floors and spread along the floors.

QUALITATIVE ASSESSMENT OF SAFETY IMPROVEMENTS

The first phase of the research was the classification of already completed fire safety improvements in the Loviisa NPP. The pertinent improvements were filtered according to the following sections: *"Fire safety and fire protection*", "*Feed water and steam systems*", "*Floods and pipe ruptures*" and "*Other Issues*". Every improvement being in, or in the vicinity of the turbine hall and the main control room was picked from the STUK's internal report 'Fire safety and fire protection' along with improvements that affect the whole plant. Fire safety improvements arising from "*Feed water and steam systems*" are not very obvious, but the logic is that building or improving redundant or back-up systems for systems located inside the turbine hall or the feedwater tank compartment are also considered for fire safety improvements. Improvements to flooding protection are taken into account if they are clearly designed against flooding caused by fire extinguishing systems. "*Other Issues*" contains few improvements to the main control room conditions, which are also within the scope. Improvements to the auxiliary systems are omitted. 46 fire safety improvements were identified to be analyzed.

Results

Filtered improvements were classified according to means of fire safety impact, to structural and active improvements and on the levels of nuclear safety DiD and fire protection DiD. The amount of improvements assigned to each means of impact is presented in Table 1. The means of impact are listed in different DiD levels, however it shall be noted that means can in fact appear on multiple levels. Such means include but are not limited to local fire protection and increasing distance. The means of impact covered all of the improvements in this study, but there can be others, which are not associated with these improvements. Some improvements are challenging to be classified because they are bigger entities impacting many factors. These are thus assigned relevant means of impact in Table 1. The most frequent means of fire safety impact are improving fire extinguishing (15 times) and fire insulation (7). The first four entries to "*Preventing fire growth and spreading*" could also be grouped together as physical separation, which would then be the most common means.

Based on previous classification and the more detailed descriptions, the improvements are divided to structural and active ones by their fire protection function. Two of the improvements belong in both categories, while some are difficult to assign to either class. There are 22 structural and 26 active fire safety improvements.

Table 1 Relevant occurrences of different means of fire safety influence among the improvements

Means of Fire Safety Influence	Number of Occurrences
Prevention of ignition	
Reducing fire load	3
Reducing ignition sources	2
Detection and extinguishing	
Improving fire detection	6
Improving extinguishing	15
Preventing fire growth and spreading	
Increasing distance	2
Fire insulation	7
Fire compartmentalization	6
Local fire protection	6
Operative fire fighting	3
Removing smoke	1
Collection of leaks	1
Limiting effects	
New redundancy	6
Improving control room conditions	2
Flood protection	3
Improving reliability	1

In terms of the DiD approach to fire protection, the results are presented in Table 2. 11 of the improvements can be assigned to the first level, which is preventing ignition. The second level, detecting and extinguishing the fire, gets 19 entries. 25 improvements target preventing fire growth and spreading, i.e. level c of DiD. The last level, containment of a fire so that safety functions can be performed, is enhanced by 10 improvements. Many improvements are again situated on two or more levels of DiD. Particularly the combination of levels a and c is prominent (8 occurrences). These are mainly improvements to structural fire protection. All the improvements to operational fire fighting are assigned at least to both levels b and c.

Assigning fire safety improvements on levels of nuclear safety DiD was more complicated, because the effect cannot be completely deduced from the description of the improvement. Instead, the effect depends on the fire compartment in question. A compartment is looked at and it is examined whether there are possible fire induced initiating events originating from that compartment. If so, all the improvements to fire safety features located in or having an effect on the compartment are deemed to affect level 1 of nuclear safety DiD, which is prevention of deviations from normal plant operation. It is then examined whether there are components of safety features designed to limit the escalation of deviations from normal operation to accidents that can be compromised by a fire. Improvements in compartments containing these are assigned to level 2 of

DiD. The same procedure is taken for level 3 concerning safety features designed to control accidents. Level 4, containment of radioactive releases, and level 5, mitigation of consequences, are not really applicable to any fire safety improvements to the plant, but could be considered in the case of improvements to operational fire fighting, where high flames originating from the plant may transport radioactive releases further from the plant. Operational fire fighting would of course be started before a release due to severe accident and occurrence of fire plume transporting the release may be eliminated by extinguishing the fire before significant release. These improvements are now assigned to level 4.

Table 2 Occurrences of improvements on different levels of DiD approach to fire protection

Fire Protection DiD Level	Number of Occurrences
a	11
b	19
c	25
d	10

The definitions in the paragraph above cause that all but one improvement affect more than one level of nuclear safety DiD. The amounts of hits for each level are listed in Table 3. The improvements inside the turbine hall affect on levels 1 and 2 based on the previous definitions. There are no components of safety systems designed to control accidents in the turbine hall. The improvements done to the control building and cable tunnels, however, affect on levels 1 through 3. The same holds for general fire protection measures, like improving fire water system, which affect throughout plant. Level 5 of DiD isn't affected by the improvements in the scope of this study.

Table 3 Occurrences of improvements on different levels of nuclear safety DiD

Nuclear Safety DiD Level	Number of Occurrences
1	35
2	43
3	27
4	6
5	0

An illustration how the different levels of both DiD principles interact is presented in Figure 1. Levels b and c of the DiD approach with respect to fire protection along with levels 1 and 2 of nuclear safety DiD form the highest cluster in the diagram. This is not surprising considering that many of the improvements studied were located inside the turbine hall, where levels 1 and 2 of nuclear safety DiD were affected. From the perspective of fire protection, levels b and c may be the easiest to be targeted retrospectively. Level a is mainly inherent in the early material choices and layout, and level d is in part concise, in part very expensive to improve. Given the explanations of how the improvements are slotted to different levels of DiD principles, everything stated in this paragraph may actually be trivial, but we wanted to show the illustration nonetheless. The distribution of cross-matches would probably change dramatically when dealing with another compartment inside the plant, and also when targeting fire protection features in the design phase as opposed to the improvements of our example case.

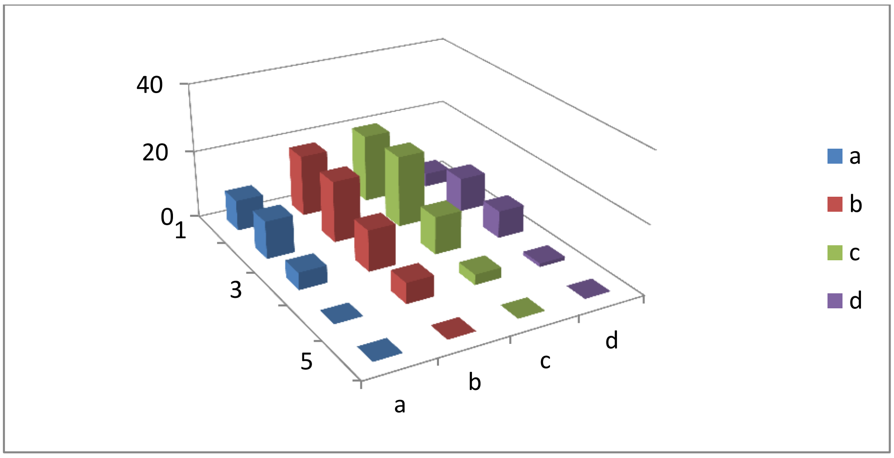

Figure 1 Illustration of how often any given two levels of the different principles of defense in depth are assigned to the same safety improvement

QUANTITATIVE ASSESSMENT OF SAFETY IMPROVEMENTS

This part of the study focuses on quantifying the importance of fire safety factors. This was done to enable comparison between the importance of different levels of DiD. The data sets studied are the improvements classified in the previous paragraph. The Fire PRA model of the Loviisa NPP serves as the source material. The model was run three times: First for fire induced core damage frequency (CDF) according to the basic Fire PRA, second for fire induced CDF in case that the additional emergency feedwater system is disabled, and third for CDF originating only from fires in the turbine hall and the feedwater tank compartment. The fire induced CDF is the overall frequency for minimal cut sets (MCS) of basic events that lead to core damage combined to initiating events caused by fire scenarios with given frequencies. The three data sets are used to determine the importance of different fire safety improvements. The measure of importance used is the risk increase factor (RIF), which determines the increase factor for CDF in case of the basic event being failed. RIFs are calculated by the PRA code and part of the available source material mentioned above. The analyses are based on the latest Fire PRA model and the plant is considered to have been in current state when the historical improvements were made. The comparison between different versions of PRA is not sensible because, in addition to the improvement in question, there have possibly been many other changes to plant conditions, and even if not, the unavailability for basic events may have changed due to advances in the reliability data. Another underlying factor that has changed over time is the ignition frequency, which has been adjusted over the years. The fact that the first Fire PRA version for the Loviisa NPP was completed in 1997, about 20 years after the commissioning of the first unit, whereas the first safety improvements date back to right after commissioning also makes the above mentioned comparison impossible.

In addition to using RIFs calculated by the PRA code, the importance of some improvements is estimated through expert analysis. This is meant to give an idea of the magnitude of RIFs for improvements that cannot be assessed just by a single basic event in the PRA model.

Results

The impacts of four stand-alone improvements could be assessed by the available PRA data, as shown in Table 4.

Table 4 Risk increase factors (RIFs) for safety improvements (values in this table are directly from the Fire PRA model)

Safety Improvement	Level of DiD for Fire Protection				Level of DiD for Nuclear Safety					RIF
					1	2	3	4	5	
Back-up feedwater supply system				x		x	x			1.02
Fire protection of safety system electric components and cables	x		x		x	x				1.64
Additional emergency feedwater system				x		x	x			42.6
New hydraulic pipelines and local fire extinguishing systems for turbine bypass valve actuators	x	x	x		x	x				3.78
Establishing professional plant fire brigade		x	x		x	x	x	x		1.83[1)]
New fire truck for the fire brigade		x	x				x	x		
Back-up fire water pump station		x	x		x	x	x	x		
New fire and rescue station		x	x		x	x	x	x		

1) RIF from the third calculation that only takes into account fires in the turbine hall and the feedwater tank compartment; other RIFs are compared to the overall fire risk

RIFs for the improvements range from very small (1.02) for the back-up feedwater supply system (based on the make-up system) to very large (42.6) for the additional emergency feedwater system (AEFW). The AEFW backs up the feedwater systems that are located inside the turbine hall and are thus susceptible to failure in case of fire in the turbine hall. The AEFW is one of the main safety improvements done at the Loviisa NPP. The RIF for the back-up feedwater supply system is very small, in part because AEFW, which was commissioned later, performs the same safety functions with higher availability. If there was no AEFW, the RIF of the back-up feedwater supply system would increase. Fire protection of safety system cables and electric components has a RIF of 1.64, while new hydraulic pipelines along with extinguishing systems for turbine bypass valve actuators have a RIF of 3.78. The new pipelines with protective casings practically eliminate oil spray fires and resulting oil pool fires can be extinguished with the sprinkler systems. In addition to the four improvements discussed above, the improvements to operative fire fighting were grouped together to get a RIF value for them. The impact of individual improvements to operational fire fighting could not be determined, but as a whole the CDF of fires originating from turbine hall or feedwater tank compartment would be 1.83 times higher without plant fire brigade actions.

The recent Fire PRA model does not go into detail regarding the impact of many safety improvements completed in the past. We wanted to expand the assessment to get an order of magnitude for the improvements in general and to be able to compare it to the few actual numbers derived straight from the PRA data. In Table 5, RIFs of some improvements are acquired through expert analysis. These RIFs are acquired through comparing different versions of PRA models, comparing the improvements to similar features in the model and other calculations. The verbal assessments are rough estimates.

All in all, the part of the analysis where we were supposed to compare impacts of different levels of DiD could not be completed due to lack of appropriate data. However, it can be stated that few major improvements dominate the overall impact of the improvements in this study. The most important safety improvements are the AEFW system, the B-line fire wall and sprinkler systems in the turbine hall. The main impacts of all of these improvements are on different levels of the DiD approach with respect to fire protection.

Table 5 Risk increase factors (RIFs) for safety improvements (values in this table are estimates acquired through expert analysis)

Safety Improvement	**Level of DiD for Fire Protection**				**Level of DiD for Nuclear Safety**					**RIF**
					1	**2**	**3**	**4**	**5**	
Additional cooling system of emergency feedwater pump compartments	x					x	x			small
Replacement of emergency feedwater pumps				x		x	x			small
Sprinkler systems to turbine hall and feedwater tank compartment		x	x		x	x	x			large
New fire wall to separate feedwater tank compartments from turbine hall (B-line)			x		x	x	x			large
Safety systems cables rerouting and fire protection in the turbine hall	x		x		x	x				moderate
Generator nitrogen supply system renewal	x		x		x	x				very small
Modernization of sprinklers in the turbine hall		x			x	x				small
Some fire doors walled up in the turbine hall			x		x	x				moderate

METHOD FOR ASSESSING DEFENCE-IN-DEPTH APPROACH TO FIRE PROTECTION

The goal was to create a method that can be used to assess the DiD approach for fire protection as a whole, while still taking into account even the smallest fire protection features. Main point of interest was the balance of fire protection features on different levels of DiD. Therefore, a three-step method is proposed for analyzing a set of fire scenarios in a fire compartment:

- Ignition root cause analysis,
- Event tree analysis of the fire scenario,
- Consequential failure modes and effects analysis (FMEA).

The analysis is not far away from Fire PRA analysis, but it aims for a higher level of accuracy with respect to the effects of fire protection measures and the consequences of a fire. The advantages of this method include the possibility to have all fire scenarios originating from the same compartment assessed in same analysis. It is also a way of bringing expertise from various fields together and interconnecting it in the same analysis. Main challenges are the transmission of information from one step to another and the fact that the analysis may expand to the point where it is difficult to maintain understanding of the big picture and control all the variables affecting the fire scenario. This method is mainly a framework, wherein the actual analysis tools may still need developing. In the following paragraphs the current ideas for the in-depth analysis are described.

Ignition Root Cause Analysis

Ignition root cause analysis takes into account all the factors impacting the ignition of a fire. For some type of fires this can be very straightforward, while for others, such as the case of oil fire in the turbine hall, it is complex. The proposed way to do this analysis is a fault tree or an event tree, depending on the fire scenario to be investigated. Frequencies are assigned to events that can cause an ignition and probabilities to failures of fire protection measures aiming to prevent ignitions. The result of this analysis should be a frequency of a fire of a certain initial burning rate. In more complex cases, the result should be a distribution of frequencies for a set of fires of different initial burning rates. The latter is e.g. used in the case of oil pool fire. This analysis encompasses level a of DiD.

Event Tree Analysis of the Fire Scenario

Event tree analysis starts with ignition and includes all the fire protection measures inside the fire compartment. It describes the development and growth of the fire depending on the initial burning rate and different fire protection measures taken. The result should be a set of temperature fields inside the compartment each with assigned conditional probabilities depending on the path the fire took along the event tree. To be able to find the temperature fields, a CFD fire simulation or rather a set of simulations of fire in the compartment must be performed. Probabilistic Fire Simulator (PFS) developed by Hostikka [11] is proposed as one of the tools to be used. The results of the simulations are then scaled for different initial burning rates, taking into account the fire protection measures actuated. The event tree of the fire scenario encompasses level b of DiD completely and fire growth part of level c.

Consequential Failure Mode and Effects Analysis (FMEA)

The final part of the assessment method tackles the question what exactly happens to the plant as a consequence of a compartment fire. Based on the temperature field in the compartment determined in the previous part, it is examined whether components of safety systems are affected by the fire either in the compartment or in adjacent compartments, if the temperature penetrates through structures. Possible impacts to structures are assessed as well. Finally, an analysis of fire spreading is performed. Spreading can happen as a result of structures failing due to fire or via open fire doors or other orifices. The effect of compartmentation is difficult to quantify. To keep the scope of the analysis feasible, event tree analysis is not applied to fire spreading events, but rather the compartment is directly assumed to be lost. After completing these tasks, the analysis continues as PRA to assess the conditional core damage probabilities and CDFs for different fire scenarios with all the failures found within the FMEA.

CONCLUSIONS

It has been concluded that it is possible to classify fire safety improvements according to different criteria. Similarly, fire protection features of a new NPP could be assessed. This information is useful and should be considered when assessing the implementation of the DiD approach to fire protection. In the example case of the turbine hall of the Loviisa NPP, the distribution of improvements on different levels of the DiD approach for fire protection was a little imbalanced, which was expected, because some fire protection features are easier to be improved than others, and some aspects of fire safety have already adequately been taken into account. The comparison of matches between the two principles of DiD may possibly be utilized to identify compartments which need to be improved from the viewpoint of nuclear safety.

Probabilistic analysis of the historical safety improvements utilizing the current Fire PRA model did not yield the desired results. Risk increase factors could be directly defined for only a handful of improvements based on the actual Fire PRA model. Due to this, the comparison of the importance of the different levels of DiD was dropped completely from this paper. Through the data and expert analysis, the most important improvements are seen to be the additional emergency feedwater system, the B-line fire wall and the sprinkler systems in the turbine hall. The problem with analyzing historical improvements is that the current Fire PRA model does not include different fire protection configurations through the past years.

The lack of data was in part due to the insignificance of the improvements or the difficulty to quantify them, but more about the nature of PRA model. To an outside observer it shows the initiating events and fire scenarios that lead to core damage, but not the failures leading to initiating events. Some features of the plant are excluded from PRA due to conservative assumptions. Fire PRA also considers a fire compartment completely lost in many cases even when that does not constitute more than a minor risk to the overall nuclear safety of the plant. Thus, no credit has been given to some local fire protection measures designed to prevent spreading of fire inside a compartment. The shortcoming of PRA for precise modeling only highlights the problem at hand: how to assess the DiD approach with respect to fire protection.

In the Finnish regulation, examples of analytical tools for assessing defense–in-depth are given. However, these are singular methods that do not aim to quantify the balance of DiD levels. Our goal was to create a method for assessing the entirety of a given fire scenario in a compartment. A three-step method with ignition root cause analysis, event tree of a fire scenario, and consequential failure modes and effects analysis is proposed. Data is transferred and shaped throughout the analysis as a continuum, so that factors impacting fire safety can be measured on the same scale. The method resembles PRA in that probabilities are assigned to all the basic events along the way. Where the analysis differs from PRA, is that there's also a variable for the severity of the fire alongside the probability. In the beginning it is the burning rate, which goes through changes depending on fire propagation and is then transformed into a temperature field inside the compartment by means of fire simulations. The challenges for this method include complexity, which can make analysis arduous. Using different experts for each step eases this issue, but creates another. Now every step has to be defined extra carefully at the interface and the quality of information transmitted has to be very dependable.

Our method has not been finalized yet to use, so the second goal of defining acceptance criteria would have been difficult to achieve. However, two types of criteria can be identified: fire safety and balanced DiD. The balance of DiD can be analyzed separately or just by assuming all kinds of failures and impairments to fire protection features and then assessing fire safety. This can be done conveniently within the assessment method as both ignition root cause analysis and event tree analysis are in a form, where a fire protection feature either succeeds or fails and it is possible to assume failure in every case. Even in the FMEA analysis which continues toward core damage as PRA and spreading analysis these failures and impairments can be assumed.

So far this method seems promising, however needs to be finalized and enhanced. The first complete version with an example case of the Loviisa NPP turbine hall will be presented in the accompanying master thesis during this fall including a first proposal of acceptance criteria.

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ONGOING ENHANCEMENTS IN THE GERMAN NUCLEAR REGULATORY FRAMEWOK WITH RESPECT TO FIRE SAFETY

Björn Elsche[1], Marina Röwekamp[2], Wilfried Neugebauer[3], Rainer Gersinska[4]

[1] e.on Kernkraft,
Hannover, Germany
[2] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH,
Köln, Germany
[3] AREVA NP,
Erlangen, Germany
[4] Bundesamt für Strahlenschutz (BfS), KTA-Geschäftsstelle,
Salzgitter, Germany

ABSTRACT

In the recent past, the regulatory framework for nuclear power plants (NPP) in Germany has been updated and enhanced comprising on the one hand comprehensive high level regulatory documents such as the 'Safety Requirements for Nuclear Power Plants' and, on the other hand, revised state-of-the-art nuclear safety standards and rules being incorporated in a corresponding legal structure.

A major enhancement concerns the nuclear fire and explosion protection standards being already available as so-called green print for final comments which are expected to be officially published end of 2015. The update became necessary after approx. ten years for better addressing some lessons learnt form the operating experience, the consideration of post-Fukushima insights, such as more systematically addressing event combinations with fires and taking into account deviations from non-nuclear standards for escape and rescue routes. Moreover, fire protections remains an important issue for nuclear power plants in Germany during the longer term post-commercial safe shutdown period before decommissioning during which the spent fuel elements remain either in the containment or in the spent fuel pool for further years requiring suitable fire protection means being in place.

INTRODUCTION

In principle, the regulatory framework for nuclear power plants (NPP) in Germany is based on deterministic requirements supplemented by probabilistic ones for safety assessment. The regulation comprises high level comprehensive claims such as the most recent "Safety Requirements for Nuclear Power Plants" [1] as well as lower level detailed technical nuclear safety standards and rules incorporated in a corresponding pyramid type legal structure as shown in Figure 1 (from [2]).

For German nuclear power plants, a variety of such technical nuclear safety standards provided by the Nuclear Safety Standards Commission (German: Kerntechnischer Ausschuss, KTA) do exist, covering also fire and explosion protection. These standards promulgated in 2000 [3] to [6] have been recently revised for updating and enhancing them according to the state-of-the-art covering lessons learned from the post-Fukushima stress tests and investigations but also the specific German situation with several reactor units having stopped commercial operation being in a longer duration post-commercial safe shutdown plant operational state before decommissioning.

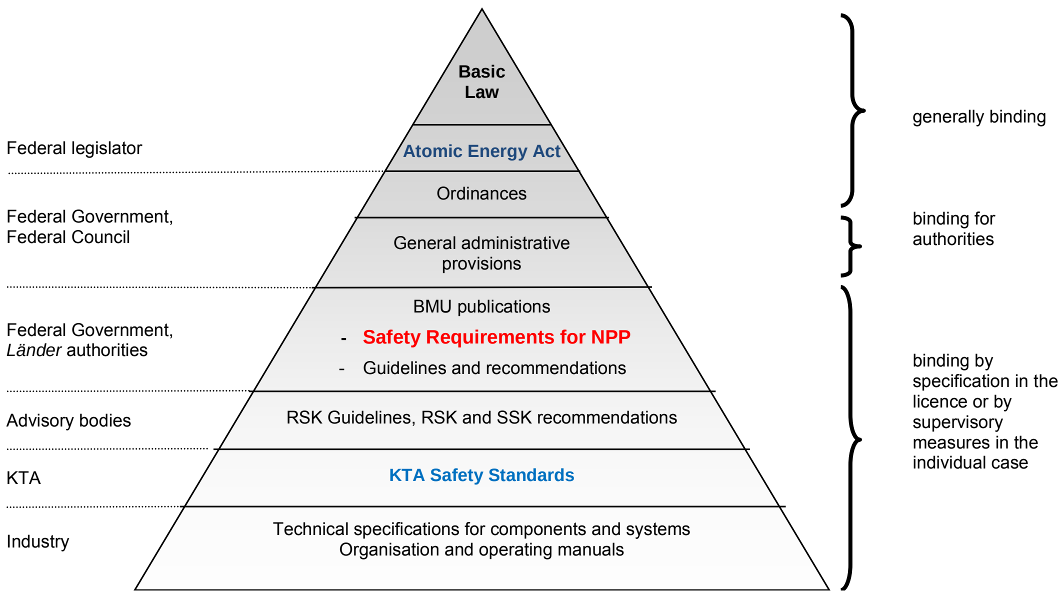


Figure 1 Nuclear regulatory framework in Germany (from [2])

All the three KTA fire protection standards as well as the explosion protection standard have already been published end of 2014 as s-called “green print” draft for final comments. Meanwhile, these final comments have been considered. It is expected that the standards will be finally published until the end of 2015. On the level of technically detailed KTA standards in total four standards are available as green print drafts with respect to fire and explosion safety:

- KTA 2101.1: Fire Protection in Nuclear Power Plants, Part 1: Basic Requirements [7],
- KTA 2101.2: Fire Protection in Nuclear Power Plants, Part 2: Fire Protection of Structural Plant Components [4],
- KTA 2101.3: (Fire Protection in Nuclear Power Plants, Part 2: Fire Protection of Mechanical and Electrical Components [9], and
- KTA 2103: Explosion Protection in Nuclear Power Plants with Light Water Reactor [10].

According to the most recent German regulation, the “Safety Requirements to Nuclear Power Plants” [1] addressing also an adequate consideration of internal hazards including fires, the fire protection standards have to be applied to land based light water reactors for all plant operating phases from the construction phase, all plant operational states of commercial operation covering power operation as well as low power and shutdown states up to the last operational phase before decommissioning, the – at least in Germany partially some years long period – post-commercial safe shutdown phase. The standard may be as far as possible also applied during decommissioning.

The following nuclear protection goals and radiological safety goals have to be met:

- control of reactivity,
- fuel cooling,
- confinement of radioactive materials.
- limiting radiation exposures.

Details with regard to the actual enhancements in the KTA fire protection standards are provided in the following sections.

NUCLEAR SAFETY STANDARD KTA 2101: FIRE PROTECTION IN NUCLEAR POWER PLANTS

The German nuclear fire safety standards of the German Safety Standards Commission (KTA), KTA 2101 "Fire Protection in Nuclear Power Plants", Parts 1-3 have been recently updated and enhanced

All the three parts of KTA 2101 are interrelated (cf. Figure 2). Therefore, one intention of the update is to harmonize the structure to provide the fundamental requirements in Part 1 and the technical details for design and operation of structures, systems and components with respect to fire safety in Part 2 and 3 accordingly, avoiding duplications.

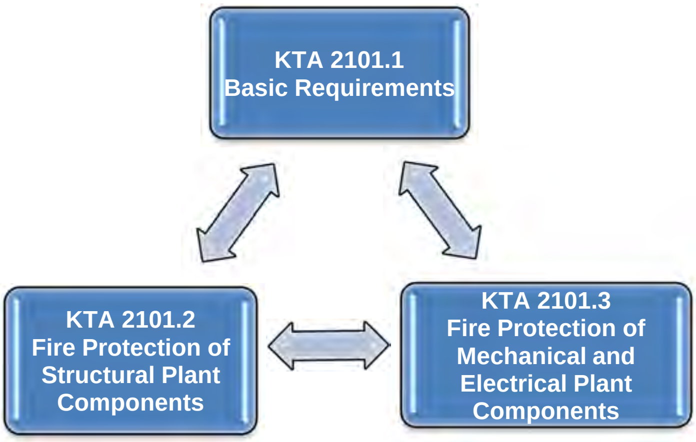


Figure 2 Interrelation of German nuclear safety standards KTA 2101 "Fire Protection in Nuclear Power Plants", Parts 1 to 3

Major goals of the update of KTA 2101, Parts 1 to 3 were the following:

- Updating requirements to the actual state-of-the-art:
  - corresponding to the most recent, also non-nuclear standards and norms,
  - providing specific compliance with requirements regarding the fire brigade,
  - considering low power and shutdown plant operational states better and more systematically,
  - addressing the fire hazard analysis explicitly and in a systematic way,
  - considering nuclear specific deviations from non-nuclear standards and norms with regard to escape and recue routes,
  - covering event combinations of fires and other anticipated events more systematically as a lesson learned from the Fukushima Dai-ichi reactor accidents in 2011.
- Compliance with the (new) "Safety Requirements for NPP" [1] in respect of the following aspects:
  - Better consideration of the defence-in-depth concept, including specific compliance with requirements for the safety demonstration, in particular requiring a fire protection concept (sometimes in the international framework also called fire protection programme) and a systematic and comprehensively documented deterministic fire hazard analysis (FHA) being kept up to date,
  - a more systematic approach and outline of the standards covering nuclear specific requirements and deviations from non-nuclear standards and norms,

- a systematic and comprehensive consideration of event combinations of fires with other anticipated event that have to be assumed, either occurring as consequence of the initial event or if their occurrence at the same time has to be accounted for due to their occurrence frequency and the extent of damage. In this context, the following event combinations have to be considered:
  - Combinations of causally related events:
    - Fire and consequential event,
    - Anticipated event and consequential fire,
  - Fire and independently occurring anticipated event.

KTA 2101.1: FIRE PROTECTION IN NUCLEAR POWER PLANTS – BASIC PRINCIPLES

For consistency with international requirements, in particular from IAEA (*I*nternational *A*tomic *E*nergy *A*gency) and WENRA (*W*estern *E*uropean *N*uclear *R*egulators *A*ssociation), several existing requirements have been formulated more precisely or new requirements have been added.

According to the fact that fire and explosion are often dealt with together in many international standards and guidelines (e.g. IAEA, United Kingdom, etc.) it has been clearly stated in KTA 2101 that requirements on explosion protection are provided in KTA 2103 "Explosion Protection in Nuclear Power Plants with Light Water Reactors" [10].

The Section on applications of the standard provides precise guidance on the goals of the standard KTA 2101.1.

The following requirements are given:

(1) This standard is applicable to nuclear power plants with light water reactors.

(2) The standard is valid for all plant operational phases

a) for protecting those plant components needed in order to maintain their required functions for meeting the nuclear protection goals and radiologic safety goals according to the Safety Requirements for Nuclear Power Plants [1], par. 2.3 and 2.5:

aa) Control of reactivity,

ab) Fuel element cooling,

ac) Confinement of radioactive materials, and

ad) Limiting radiation exposures

as well as

b) for protecting humans working there

in case of building-internal and building-external fires on site.

Note:
In this context the term 'plant component ' is used as follows according to [1]: Any structural, mechanical, process-based, electrical or other technical part of a plant. Synonyms are: equipment, system (see also structures, systems and components).

One of the most important changes in the standard affects the protection against combinations of fires and other anticipated events. The systematically structures requirements are the following:

"3.3 Combinations of Fires with other Anticipated Events

3.3.1. Basic Principles

(1) Combinations of fires with other anticipated events have to be assumed, if the events to be combined are causally related or if their occurrence at the same time has to be accounted for due to their occurrence frequency and the extent of damage.

(2) Combinations of fires with other anticipated events have to be solely considered with respect to meeting the goals mentioned in Section 1, Paragraph (2) item a).For the combinations to be considered fire protection measures have to be implemented unless effective and reliable precautionary measures have already been taken.

Note:
This requirement substantiates the extent of damage mentioned in 3.3.1 (1).

(3)The following combinations have to be distinguished:

a) Combinations of causally related events:
   - aa) Fire and consequential event,
   - ab) Anticipated event and consequential fire,

b) Combinations of independently occurring events

3.3.2 Combinations of Causally Related Events

3.3.2.1Fire and Consequential Event

(1) The following combinations of fires and consequential events have to be considered:

a) Fires and consequential component failure:
   - aa) Failure (including high energy faults) of electrical components and equipment,
   - ab) Failure of mechanical components (e.g. fast rotating parts, pre-tensioned springs),
   - ac) Failure (including high energy faults) of pressure retaining pipework and vessels, whose own intrinsic failure cannot be excluded.
     - aca) For pressure retaining vessels and systems, structures and components (SSC), for which their own intrinsic failure can be excluded because of their quality characteristics or for which their failure modes are limited, either measures for preventing a fire in the area of pressure retaining vessels or components have to be implemented or protection measures against fire impact have to be taken. Otherwise it has to be demonstrated that in case of fire the quality characteristics that preclude a failure or limit a failure mode will not be not be inadmissibly impaired.

Note:
Such pressure retaining vessels and components are e.g. the reactor containment, the steam generators, the pressurizer, the main coolant pumps and the accumulators in nuclear power plants with pressurized water reactors and the reactor pressure vessel and the reactor scram vessel in nuclear power plants with boiling water reactors respectively. The corresponding SSCs are e.g. the containment, safety related support structures and structural elements as well as the spent fuel pool. Such quality characteristics maybe e.g. the voltage exploitation. A limited failure mode is given in case of e.g. a design ensuring basis safety according to the "Safety Requirements for Nuclear Power Plants".

b) Fires and consequential internal explosion including explosions of radiolysis gases in systems and components.

3.3.2.2 Anticipated Event and Consequential Fire

The following event combinations of an anticipated event and a consequential fire have to be considered:

a) Component failure and consequential fire
   - aa) High energy faults (including arcing) of electrical components and equipment (e.g. switchgears, breakers, transformers, high voltage cables),
   - ab) High energy faults of mechanical components (e.g. fast rotating parts, pre-tensioned springs),
   - ac) High energy faults of pressure retaining pipe work and vessels whose own intrinsic failure cannot be excluded,

b) Plant internal explosion and consequential fire
   An explosion as consequential event to a fire inadmissibly impairing the required

safety functions has to be excluded. Safety functions are presumed not to be inadmissibly impaired if the provisions provided in KTA 2103 are considered.

c) Earthquake and consequential fire

ca) In safety related buildings which have to be designed against earthquakes according to the requirements of the nuclear seismic standard KTA 2201.1, it has to be ensured that the required safety functions are not inadmissibly impaired by a fire consequential to the earthquake. This requirement is met if either those plant components releasing combustibles in case of loss of their integrity or those enabling an ignition are designed against earthquake by suitable materials and construction. If a fire cannot be excluded it has to be ensured by structural fire protection means that those safety functions required after an earthquake are not inadmissibly impaired If this is not possible according to needs from systems technology or use, an equal protection has to be ensured by suitable technical fire protection means (e.g., fire detection and alarm system) or a combination of such measures. The aforementioned structural and technical fire protection means have to be designed accordingly, applying suitable building construction and other materials and construction designed against earthquake. Due to the short duration of strong earthquakes in Germany it can be assumed that a consequential fire will be effective only after the earthquake.

cb) If the plant is designed against an earthquake with a maximum intensity of I = VI (EMS-98), the required function of the structural fire protection means as well as those of the technical fire protection features are presumed without any specific design provisions.

d) Lightning and consequential fire:
Any fire consequential to lightning inadmissibly impairing the required safety functions has to be excluded. Safety functions are presumed not to be inadmissibly impaired if the provisions provided in KTA 2206 are considered.

3.3.3 Combinations of Independent Events

(1) In principle, no measures have to be taken for combinations of an anticipated fire and an independently occurring anticipated event.

Note:
In this context, it has been assumed that:

a) the occurrence frequency of combinations of independently occurring events is less than $1x10^{-5}$ per year,

b) such event combinations are excluded by suitable precautionary measures, or that

c) an event occurring independently from the fire does not inadmissibly impair the fire protection means.

(2) Measures have to be taken for combinations of an anticipated fire and an independently occurring anticipated event listed in the following:

a) Plant internal flooding,

b) Plant internal or external electro-magnetic interference (EMI), (except lightning),

c) Earthquake (including consequential effects),

d) Flooding, or

e) Other site related external hazards.

(3) Those fire protection means needed in case of a combination of an anticipated event listed in (2) and an independently occurring fire for ensuring the fire protection goals according to Par. 1 (2) have to be made available again or be replaced by suitable other measures within one week after the occurrence of the event combination.

Note:
For a grace period of one week the occurrence frequency of the combination of an anticipated fire and one of the anticipated events listed in (2) is less than $1x10^{-5}$/a.

(4) For the combinations of an anticipated fire with one of the anticipated events listed in (2) it is assumed that the measures mentioned in (3) can be taken within one week."

Another important enhancement of the standard is that it is meanwhile required explicitly in the standard that both a fire protection concept and particularly a (deterministic) fire hazard analysis have to be provided and kept up to date.

KTA 2101.2: FIRE PROTECTION IN NUCLEAR POWER PLANTS – FIRE PROTECTION OF STRUCTURAL PLANT COMPONENTS

This part of the German nuclear fire protection standard KTA 2101.2 supplements the safety standard KTA 2101 Part 1 (KTA 2101.1) and Part 3 (KTA 2101.3). All three parts of the standard are closely related and complement each other.

The recent update of KTA 2101.2 (2014-11) [8] follows the demands of modifications prepared by the KTA (General Assembly), in particular for:

– Adapting references and definitions to up-to-date standards and codes,
– Updating and enhancing the requirements in order to make them consistent to up-to-date national as well as international standards and codes,
– Updating in particular the technical requirements for structural components to the state-of-the-art,
– Harmonizing this part of the standard KTA 2101 with the other two parts, KTA 2101.1 and KTA 2101.3.

In the KTA 2101.2 [8], specific requirements for structural components are provided. This covers in detail the followings topics:

- Design of structural fire protection means,
- Location and accessibility of nuclear power plant buildings,
- Fire compartments and fire sub-compartments
- Structural elements enclosing fire Compartments and fire Sub-compartments (fire barriers),
- Escape and rescue routes,
- Ventilation systems, heat and smoke removal systems and components.

In an informative appendix (Appendix A) a simplified validation approach for determining the fire resistance rating of structural elements is provided.

Design of Structural Fire Protection Means

Building structures and other structural elements can be designed using different methods. All these methods are in principle equivalent; however the scope of the application differs:

- Analytical validation: This kind of validation is applied in general.
- Experimental validation: Experimental validation is applied the design of special components.
- Analogy consideration: Performed on the basis of the design of structure-related fire protection measures for other load cases
- Plausibility consideration: Based on referential results of experimental or analytical confirmation that were performed for comparable structure -related fire protection measures

Location and Accessibility of Nuclear Power Plant Buildings

The buildings of nuclear power plants shall be arranged taking operational conditions and, the following additional demands into consideration:

- separation of buildings by distance,
- rapid and safe escape and rescue of persons in case of an emergency, and
- accessibility for fire suppression.

Further details are provided for

- Access roads,
- Access ways,
- Fire brigade engagement areas,
- Free movement areas.

Fire Compartments and Fire Sub-compartments

In the following, specific nuclear requirements for compartments are provided. In particular, those plant areas are characterized, where fire compartments or sub-compartments are needed for ensuring nuclear safety.

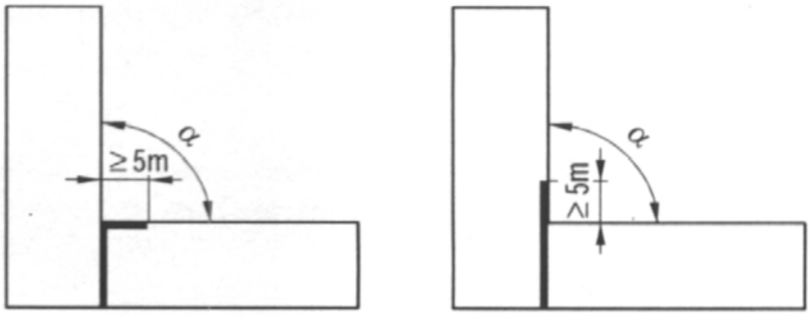
Fire sub-compartments are e.g.:

- rooms for electronic data processing equipment and their under-floor cable sections,
- rooms for switchgear/breakers and their under-floor cable sections,
- rooms for electronic equipment and their under-floor cable sections,
- rooms for emergency power supplies including and their fuel storages, redundant trains of the emergency power supply facilities,
- rooms containing redundant safety related systems or components,
- cable rooms and tunnels,
- rooms for storage of new fuel elements,
- rooms for external oil and lubricant supplies including storages for the lubrication media,
- rooms containing activated charcoal, etc.

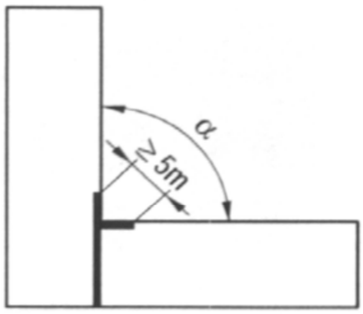
The emergency control room need to be an own fire compartment.

In addition, specific information is given for corners between different buildings:

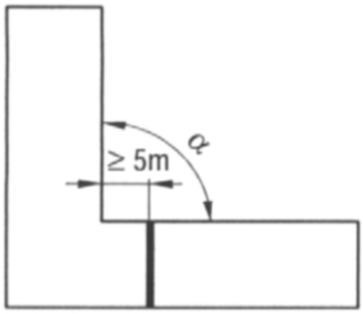
Those corners resulting from buildings installed directly adjacent to each other without any distance with structural elements located under an angle less than or equal to 120 degrees have to be separated by a fire wall and shall be designed as shown in Figure 3 below.



The fire wall at the inside corner shall be extended in either of the two directions to a length ≥ 5 m.



The fire wall shall be extended in both directions.



The fire wall shall be arranged at a distance of ≥ 5 m from the inside corner.

Figure 3 Corner between buildings

Structural Elements Enclosing Fire Compartments and Fire Sub-compartments

All elements enclosing Fire compartments and Fire Sub-compartments must be designed to be sufficiently fire resistant. For this fire resistance incombustible materials have to be used. This means that the walls and ceilings have a fire resistance time of regular 90 minutes (in appendix A is described a simplified and conservative method for calculating the fire resistance method). The requirements apply also for all openings in the walls and ceilings.

Escape and Rescue) Routes

Due to requirements for security reasons the escape and rescue rotes may be blocked to the outside. For these conditions specific fire protection requirements are needed.

The walls and ceilings of necessary stairways shall be designed to be fire resistant (fire resistance rating of 90 min (F90)). Closing elements of openings from the necessary stairways to adjacent rooms have basically to be designed to be fire resistant and smoke tight.

The vestibules of the airlocks inside the containment have to be also designed by suitable non-combustible materials with an adequate fire resistance rating.

Ventilation Systems, Equipment for Heat and Smoke Removal

In the case that fire dampers are installed in the ventilation systems which have to be leak tight, these fire dampers shall inadmissibly impair the leak tightness of the systems.

Pipes, adapter fittings, ducts and channels of ventilation conduits in a specified fire resistance class shall basically be designed of non-combustible materials. Exceptions are permissible, provided, the purpose is to remove corrosive gasses [11]).

Appendix A: Simplified Validation Procedure for Determining the Required Fire Resistance Rating of Structure-related Fire Protection Measures

The informative Appendix A provides a state-of-the-art approach for a validated determination of the fire resistance rating of fire walls and other structural elements needed to ensure nuclear safety. This approach has based on a study “Examining the possibilities for establishing standards regarding the verification of fire protection measures within the frame of KTA 2101.2 (Final Report)” by Hosser et al. [12]. The simplified validation procedure described may be used in determining the fire resistance rating of structure-related fire protection measures. Some of the parameters in this approach were no longer state-of-the-art, making an update providing precision necessary. Moreover, the applicability of the procedure was further limited because of uncertainties needing further research.

KTA 2101.3: FIRE PROTECTION IN NUCLEAR POWER PLANTS – FIRE PROTECTION OF MECHNICAL AND ELECTRICAL PLANT COMPONENTS

This part of the German nuclear fire protection standard KTA 2101.3 supplements the safety standards KTA2101.1 Basic Requirements and KTA2101.2 Fire Protection of Structural Plant Components by additional requirements that apply specifically to the fire protection of mechanical and electrical plant components in nuclear power plants. Thus these 3 standards are closely connected and complement each other.

The recent update of KTA 2101.3 (2014-11) [9] follows the demand of modifications prepared by the KTA (General Assembly), in particular for:

- adapting references and definitions to up-to-date standards and codes,
- updating and enhancing the requirements in order to make them consistent to up-to-date national as well as international standards and codes,
- updating in particular the technical requirements for extinguishing systems according to the state-of-the-art,
- reviewing the requirements for heat and smoke removal systems, in particular with respect to access and escape routes,
- including precise technical requirements for storage of pressurized gas cylinders, and
- harmonizing this part of the standard KTA 2101 with the other two parts, KTA 2101.1 and KTA 2101.2.

Moreover, the following general adaptations have been considered:

- Moving more general requirements from KTA 2101.3 to KTA 2101.1 or KTA 2101.2 or more technical requirements from KTA 2101.1 to KTA 2101.3 as far as suitable and avoiding duplications,
- Enhancing and harmonizing the wording of performance based requirements between the three parts of KTA 2101 and considering other fire protection standards and codes,
- Deleting definitions as well as requirements from this standard, if already well-established and regulated in the non-nuclear fire related building codes and standards.

The structure of the contents in the part 3 of the standard KTA 2101 has been slightly adapted accordingly resulting in the following main paragraphs:

Fundamentals

1. Scope
2. Definitions
3. Fire Protection Measures for Mechanical Components and Systems
4. Fire Protection Measures for Electrical Facilities and Arrangements
5. Facilities for Fire Detection and Fire Alarm
6. Facilities for Fire Suppression
7. Ventilation Systems, Facilities for Smoke and Heat removal

Examples for Adaptation:

Fire Detection and Alarm Systems

The former version of KTA 2101.3 included the requirement that "*…as far as the fire detection and alarm systems must be designed against earthquakes, safety standard KTA 2201.4 shall be applied. It is permissible to alternatively assume that the fire detection and alarm facility stays available after an earthquake, provided, it is proven that the support structure of the fire alarm board retains it stability during earthquakes and it is ensured that any failed components in the fire alarm control centre and the corresponding local control centres can, if required, be replaced (e.g., by exchanging the modules) or repaired at short notice."*

In the actual version of KTA 2101.3, a recommendation and a requirement for design are being distinguished. It is recommended that the fire detection and alarm systems should be designed against earthquakes according to KTA 2201.4, if they are located in building areas which need to be seismically designed according to their safety relevance and if the seismic intensity I exceeds I = VI (EMS-98). The requirement is that the fire detection and alarm systems shall be designed against event combinations of fires and other anticipated events if their function after an event combination has to be ensured according to KTA2101.1, par. 3.3.

Seismically designed fire detection and alarm systems are available and represent state-of-the-art systems.

Fire Extinguishing Systems:

The former table "*7.1-1 Suitability of stationary fire extinguishing systems in fire extinguishing areas typical for nuclear power plants*" has been deleted as a result from the availability of various new fire extinguishing systems with gas, water, foam or combinations of extinguishing agents as extinguishing media being available. The suitability of these systems has to be demonstrated for the special application and the design may follow the generally valid codes and standards.

CONCLUSIONS

Nuclear power plants in Germany are mainly designed and licensed according to the existing non-nuclear as well as nuclear fire safety standards, in general on the basis of deterministic requirements.

The recently issued state-of-the-art high level "Safety Requirements to Nuclear Power Plants" also underline the need for an adequate fire protection design and the demonstration of the reliability of the selected equipment by deterministic and probabilistic safety assessments.

Although only nine nuclear power plant units are still in operation in Germany, fire protection remains a significant issue. The main reasons are the significance of fires even in shutdown states as well as during decommissioning and the long duration of post-commercial safe shutdown plant operational phase in several plant units phase in several plant units with the spent fuel elements remaining either in the containment or in the spent fuel pool for years requiring appropriate and reliable fire protection means being in place.

The recently updated and enhanced KTA fire protection standards KTA 2101, Part 1 to 3 consider the specific German situation as well as lessons learned from various investigations of the post-Fukushima activities. They represent state-of-the-art requirements on a detailed level providing guidance to the user how to implement high level claims in the practical application. Moreover, together with other German nuclear standards, e.g. regarding internal explosions or seismic hazards, they provide a consistent approach for a safe design of nuclear power plants for the entire plant operational lifetime.

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RECENT AMENDMENTS OF THE KTA 2101.2 FIRE BARRIER RESISTANCE RATING METHOD FOR GERMAN NPP AND COMPARISON TO THE EUROCODE T-EQUIVALENT METHOD

Burkhard Forell

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Cologne, Germany

ABSTRACT

The German nuclear standard KTA2101 on "Fire Protection in Nuclear Power Plants", Part 2: "Fire Protection of Structural Plant Components" includes a simplified method for the fire resistance rating of fire barrier elements based on the t-equivalent approach. The method covers the specific features of compartments in nuclear power plant buildings in terms of the boundary conditions which have to be expected in the event of fire. The method has proven to be relatively simple and straightforward to apply.

The paper gives an overview of amendments with respect to the rating method made within the regular review of the KTA 2101.2. A comparison to the method of the non-nuclear Eurocode 1 is also provided. The Eurocode method is closely connected to the German standard DIN 18230 on structural fire protection in industrial buildings. Special emphasis of the comparison is given to the ventilation factor, which has a large impact on the required fire resistance.

INTRODUCTION

A simplified t-equivalent rating method for the structural fire design of fire barriers and other structural elements in nuclear powers plants (NPP) has been developed by Hosser and Blume in the 1990ies [1], [2]. This method became part of the German nuclear fire safety standard KTA 2101, Part 2 on "Fire Protection of Structural Plant Components" [3] in 2000. Within the regular review of the fire safety standard series KTA 2101 the rating method has also been reviewed. The method was demonstrated and recommendations for amendments were given already in a paper presented on the preceding seminar [4].

In the following, the amendments of the simplified rating method already considered in the final draft of the amended update of KTA 2101.2 [5] and expected to be put into effect by end-2015 are provided in more detail. A comparison with the t-equivalent rating method of the Eurocode EN 1991-1-2, Informative Annex F "Equivalent time of fire exposure" [6] has also been carried out. Significant parts of the EN 1991-1-2 (EC 1) method are based on the German standard DIN 18230, Part 1 on "Structural fire protection in industrial buildings" [7], which provides a better documentation of the reference scenarios. All methods correlate the fire load density and other parameters with the equivalent time to the fire exposure according to the ISO 834 standard fire curve [8].

FIRE LOAD DENSITY

The fire load density q [MJ/m^2] is the heat of combustion of the fire loads i divided by the floor area of the compartment A_f. In the KTA method, an effective fire load density q_R [kWh/m^2] is used considering two non-dimensional factors, namely the combustion efficiency

χ_j ($0 < \chi_j < 1$) and the factor Ψ_i ($0 < \Psi_j < 1$), which considers the probability that a protected fire load may not ignite.

For the combustion efficiency, Table A2-1 of the existing KTA 2101.2 was replaced by general values based on the physical condition of the fuel. According to this, for solid fuels χ_s = 0.8, for liquid fuels χ_{lq} = 0.9, and for gaseous fuels χ_{gas} = 1.0 shall be used. Other values may be used if they are underpinned by fire experiments.

By the factor Ψ_i it can be considered that some protected combustibles may not ignite in case of a given initial fire depending on the quality of the t enclosures of the combustibles. KTA 2101.2 refers to the approach in the German standard DIN 18230, which is a little bit more complex than that provided in EN 1991-1-2, Section E.2.3.

Annex F of the EN 1991-1-2 uses a design fire load density according to Annex E of the same standard. The design value is calculated from the characteristic value of the fire load density under consideration of a combustion factor *m* and three additional factors of the safety concept. The combustion factor *m* represents the combustion behaviour of the fire load. For predominantly cellulosic fire loads the factor may become *m* = 0.8. Nothing else is regulated with respect to the combustion factor. In contrary to the German nuclear KTA standard and the non-nuclear DIN 18230, the safety concept is already considered for the calculation of the design fire load density by three factors, namely for the compartment floor area ($1.10 < \delta_{q1} < 2.13$), the type of the occupancy ($0.78 < \delta_{q2} < 1.66$), and the function of active fire fighting means ($0.61 < \delta_{ni} < 1.50$). Since in EC 1 a linear correlation is assumed between the fire load density and the equivalent time of the standard fire exposure, these three factors may also be set 1 and considered latterly.

Within the German DIN 18230 the fire load density is calculated in a similar way than within KTA 2101.2. However, instead of the combustion efficiency χ_i the combustion factor $m_{18230,i}$ has to be given for each combustible. The factor should represent the dynamic combustion behaviour of a fire load under real storage conditions. The combustion behaviour of wood cribs is set $m_{18230,woodcrib}$ = 1 by definition. Common values are listed in Part 3 of the standard [9]. The factor varies considerably (e.g. for 0.05 for large rolls of paper on wooden pallets and 2.0 for a bulky storage of recovery paper) in comparison to the variation of the combustion efficiency.

EQIVALENT TIME TO THE STANDARD FIRE EXPOSURE

The EC 1 as well as the DIN 18230 assume a linear behaviour between fire load density and fire duration, whereas the KTA standard considers a diminishing increase of the equivalent time. The methods include additional correction factors for the compartment height, the thermo-physical properties of the compartment structures, and the compartment ventilation openings. For this comparison, typical NPP compartments are assumed constructed of normal concrete (λ = 1.28 W/mK, ρ = 2200 kg/m^3, c_P = 879 J/kgK) and a compartment height of 2.5 m. The ratio of vertical openings A_V related to the floor area A_f shall be 0.022. Horizontal openings are not assumed to be present.

For these parameters, the correlation of the basic value of the equivalent fire duration, $t_{e,0}$ with the fire load density is given in KTA 2101.2 (Figure 1). For comparison the curve of the uniformly distributed fire load must be taken. The other curves (non-uniformly distributed and point source fire source) come to bear in those areas of the fire compartment where an accumulation of fire load is present.

Within the EC 1 standard the equivalent time of standard fire exposure is defined by

$$t_{e,d} = q_{f,d} * k_b * w_f * k_c \tag{1}$$

where:

$q_{f,d}$ design fire load in [MJ/m^2],

k_b conversion factor to account for the thermal properties of the enclosure in [min*m^2/MJ],

w_f ventilation factor in [-],

k_c correction factor to consider the material of the load bearing structures, here: reinforced concrete, k_c = 1.0

With the values listed above the conversion factor to account for the thermal properties of the enclosure becomes k_b = 0.055 Min*m^2/MJ. With the given vertical and horizontal opening ratios of α_v = 0.022 and α_h = 0 and with the reference compartment height of H = 2.5 m the ventilation factor becomes w_f = 3.2. Finally with the correction factor 3.6 MJ/kWh Equation (1) becomes

$$t_{e,d} = 0.64\ min/(kWh/m^2) \tag{2}$$

Within the German standard DIN 18230 the result of equation (2) can be reproduced. The comparison of EC 1 with the KTA correlation in Figure 1 shows that below a fire load density of 200 kWh/m^2 KTA yields higher fire durations, whereas for above 200 kWh/m^2 EC 1 is more conservative. However, for non-uniformly distributed fire loads the local effects are considered in the design curves. A procedure to consider these local effects on partial areas is outlined in DIN 18230, but not applied in EC 1.

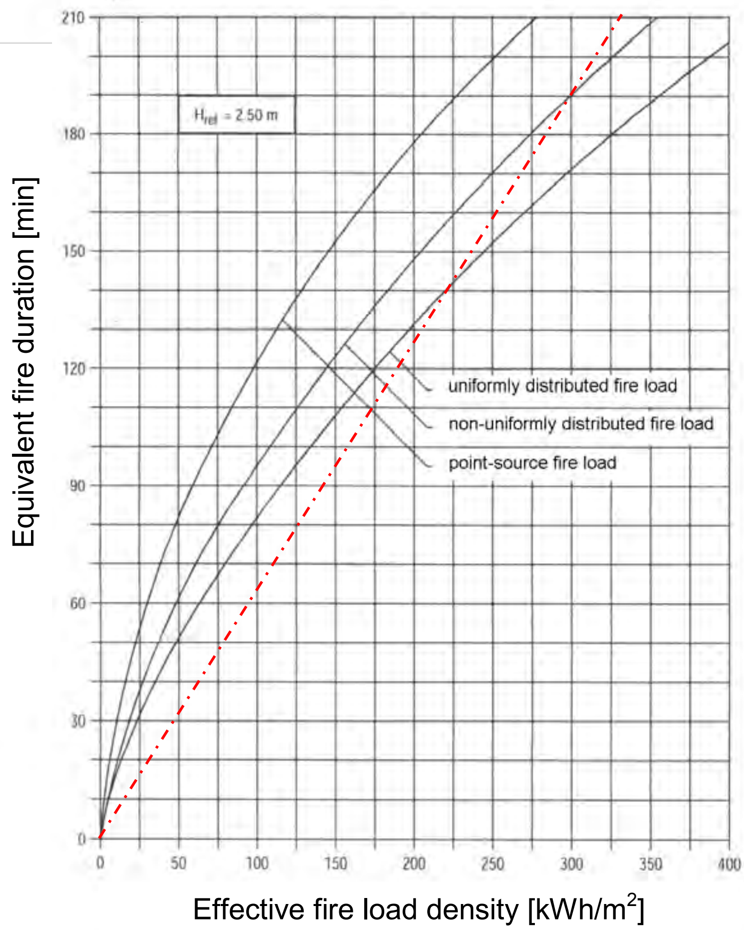

Figure 1 Basic equivalent fire duration $t_{e,0}$ as a function the effective fire load density according to KTA 2101.2 [3] in comparison to EC 1, Annex F (dotted line)

INFLUENCE OF COMPARTMENT SIZE

The compartment floor area A_f is no direct parameter in the three methods. Only in EC 1 the area affects the parameter δ_{q1} of the safety concept. Although a direct effect of A_f is not given, it should be noticed that the reference compartment floor area within the KTA is between A_f = 150 m^2 and A_f = 450 m^2 [1], whereas in DIN 18230 a reference compartment of A_f = 2400 m^2 was used [7]. EC 1 does provide no information regarding a reference compartment.

The correction factor for the actual compartment height H in comparison to the reference compartment height H_{ref} is calculated by

$$f_H = (H_{ref} \ / \ H)^{0.3} \tag{3}$$

in all methodologies; however the reference height is H_{ref} = 2.5 m in KTA 2101.2 and 6.0 m in the other standards.

VENTILATION EFFECTS

General

Ventilation has got a crucial influence on the equivalent time of fire exposure. The larger the air-inflow to the compartment, the more heat can be produced. For combustion of organic materials, the heat release per kilogram of air is $\Delta H_{air} \approx 3.0$ MJ/kg.

For low air inflows, small heat release rates in the fire compartments are given that come along with large fire durations to consume all fuel (possibly unrealistic) or under-ventilated fires (not conservative). The ventilation factor will be small for low air inflows.

For large air inflows the compartments will be over-ventilated and cooled by the excess air that leaves the compartment. Therefore the ventilation factor will be small for large air inflows.

The most severe ventilation (i.e. the maximum of the ventilation factor) has to be expected for stoichiometric conditions. However, defining the air inflow under stoichiometric conditions requires knowledge of the fuel mass loss rate, which is typically afflicted with large uncertainties. Therefore, the behaviour of the curve of the ventilation factor depends on the chosen design fire in relation to the reference compartment.

The location of the opening ratio where stoichiometric conditions occur depends on the average *h*eat *r*elease *r*ate *p*er *u*nit *c*ompartment *a*rea (HRRPUCA), which is supported by the fuels within the compartment. The air consumption per second and square-meter of floor area is $\dot{m}_{air} / A_f$ = HRRPUCA / ΔH_{air}. If the air supply is by vertical ventilation areas, by recasting the Kawagoe equation the corresponding opening ratio can be calculated by

$$A_v \ / \ A_f = \text{HRRPUCA}/\Delta H_{air} \ / \ (0.52 \ * \ h^{0.5}) \tag{4}$$

Ventilation of the KTA Reference Compartment

The ventilation of a NPP compartment may consist of natural air flow through vertical openings and/or mechanical ventilation. To consider the contribution by mechanical ventilation by the frequently used opening ratio $\alpha_v = A_v / A_f$ for the vertical ventilation area, an effective opening area was introduced by

$$A_{V,eff} = A_V \ + \ \dot{V}_{zu} \ / \ F \quad [m^2] \tag{5}$$

where:

A_V: overall surface area of vertical vents in the enclosing walls [m^2],

$\dot{V}_{zu}$: volumetric air supply rate in the case of forced ventilation [m^3/h],

A: surface area of the compartment [m^2],

F: equivalence factor between mechanical and natural airflow,
F_{old} = 6000 m^3/h in KTA 2101.1 (as of 2000) [3],

F_{new} = 2200 m^3/h in KTA 2101.2 (recent draft) [5].

The equivalence factor F from this equation used to be set to 6000 m^3/h in the former version of the Appendix to KTA 2101.2 [3], i.e. a forced ventilation of $\dot{V}_{zu}$ = 6000 m^3/h matches the inflow through an A_V = 1 m^2 vertical vent. Analytical considerations and zone model simulations with CFAST lead to the conclusion [4] that the factor F should become 2200 m^3/h in the recent version of KTA 2101.2 [5]. For the fire simulations in the reference compartment, besides a variable natural opening A_V a mechanical ventilation of $\dot{V}_{zu}$ = 3000 m^3/h has been assumed for a compartment area of A_f = 150 m^2 [1]. Therefore, the contribution by mechanical ventilation is $A_{V,mech,old} / A_f$ = 0.3 % according to the old calculation and $A_{V,mech,new} / A_f$ = 0.9 % according the recent one. The x-axis in the design diagram (see Figure 2) is shifted by 0.6 %.

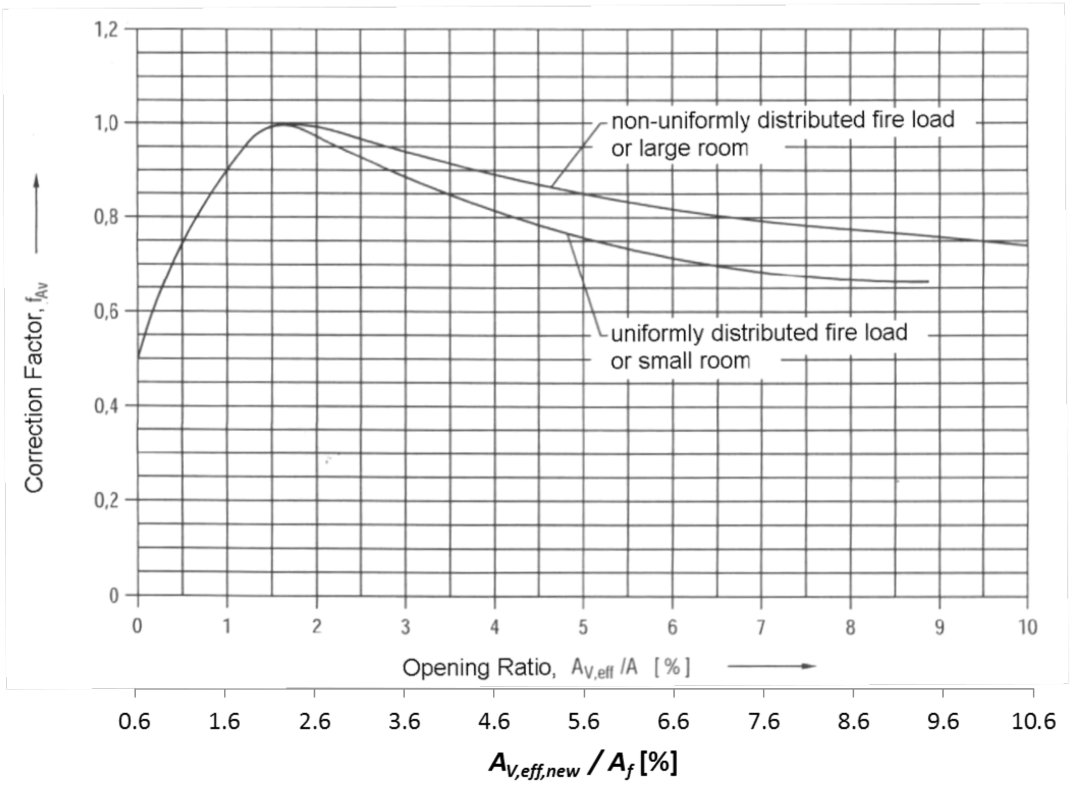

Figure 2 Ventilation factor f_{Av} as a function of the effective opening ratio, $A_{V,eff,old} / A_f$ [3] with added axis for the shifted new $A_{V,eff,new} / A_f$

As mentioned before, the quantitative behaviour of the curve of the ventilation factor in Figure 2 depends on the chosen design fire in relation to the reference compartment. The HRRPUCA in the steady state of the fire is the most important parameter of the chosen design fire. It was assumed in the original research study [1] that the value for the HRRPUCA depended on the vertical opening area. However, the majority of fire simulations within the parameter studies were over-ventilated. Maximum values for the ventilation factors were achieved under nearly stoichiometric conditions at maximum HRRPUCA of 55 kW/m^2. This resulted in a maximum value of the ventilation factor f_{Av} at $A_{V,eff,new} / A_f$ = 2.2 % (with the old factor F: $A_{V,eff,old} / A_f$ = 1.6 %). For higher $A_{V,eff,new} / A,$ the ventilation factor decreased because the fires became more and more over-ventilated in the fire simulations (Figure 2).

Revised Curve of the Ventilation Factor f_{Av}

Within the review of the KTA 2101.2 it has been demonstrated that the maximum HRRPUCA does not represent a conservative assumption [4]. In comparison to KTA 2101.2, a value of 75 kW/m^2 has been considered for design fires in the standard DIN 18230 for even larger industrial facilities [7]. A re-calculation of ventilation factors f_{Av} based on CFAST [10] fire simulations under the assumption of increased HRRPUCA leads to increased ventilation factors for larger opening ratios. It was also demonstrated that the basic values for the required fire duration (Figure 1) are in general conservative and that the ventilation factor also depends on the fire load density, particularly for larger opening ratios [11]. Therefore it was decided to keep the ventilation factor of f_{Av} = 1 for effective opening ratios $A_{V,eff,new}$ / A_f > 2.2 % and to cut the range of application of the rating method at $A_{V,eff,new}$ / A_f = 3.0 %. This range will cover most NPP compartments in bunkered buildings. For other possible fire compartments larger ventilation factors are expected and compartment-specific fire simulations must be undertaken.

Ventilation Factor in Eurocode 1 and DIN 18230

Much efforts were undertaken to calculate the ventilation factor w_0 for DIN 18230 depending on the vertical and horizontal opening ratios $\alpha_v = A_v / A_f$ and $\alpha_h = A_h / A_f$, which resulted in a set of equations that are summarized in the design diagram (Figure 3). According to the diagram, for industrial buildings, it was derived that maximum fire severity occurs up to α_v = 0.025. It can be shown that at this point a stoichiometric fire occurs in the reference compartment. For lower α_v under-ventilated fires would occur. For this reason, the line was fixed horizontally within the DIN 18230. For α_v > 0.025 cooling effects of over-ventilated fires do occur. With additional horizontal ventilation (α_h > 0) the factor w_0 also decreases.

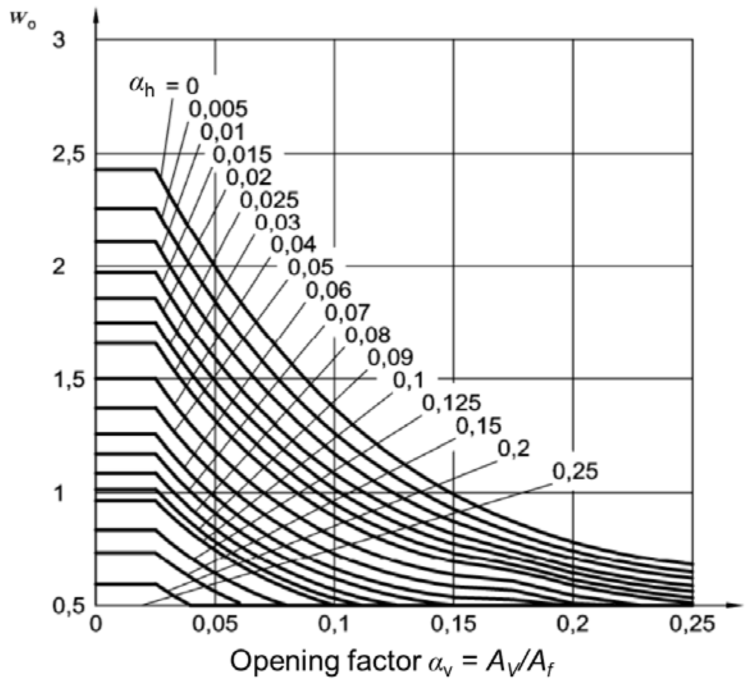

Figure 3 Ventilation factor w_0 of the DIN 18230 as a function of the vertical and horizontal opening ratios $\alpha_v = A_v / A_f$ and $\alpha_h = A_h / A_f$ [7]

The EC 1 standard principally uses a set of equations very similar to that of DIN 18230. However, the convention for a horizontal curve for α_v < 0.025 was not applied; the use of this equation is formally restricted to $\alpha_v \geq 0.025$. Nevertheless, within EC 1 a second equation can be used for the ventilation factor for small fire compartments (A_f < 100 m^2) without roof openings, which even yields higher values than the restricted equation.

Comparison of the Ventilation Factors Applied in KTA, Eurocode 1, and DIN 18230

For the ease of comparison the typical situation in an NPP compartment is assumed with zero horizontal ventilation (α_h = 0, upper line in Figure 3). The ventilation factors of EC 1 and DIN 18230 are normalized for the basic scenario of the KTA compartment. Thus the equivalent fire duration can be directly calculated by multiplying the basic values of the equivalent fire duration (Figure 1) with the normalized ventilation factors (Figure 4).

Figure 4 illustrates that the influence from different ventilation factors on the calculated fire duration is more significant than that from the basic values of the equivalent fire duration. For the recent version of the KTA 2101.2 curve the required fire resistance increases with increasing ventilation, since the decreasing part of the curve (Figure 2) has been cut off, and specific analysis is required if $\alpha_v \geq 0.03$ occurs. The equations in EC 1 for small compartments and in DIN 18230 yield correlations, where the required fire resistance decreases with increasing ventilation (in case of DIN 18230 a decrease beginning at $\alpha_v \geq 0.025$). The curve expected from the general considerations made above is not given.

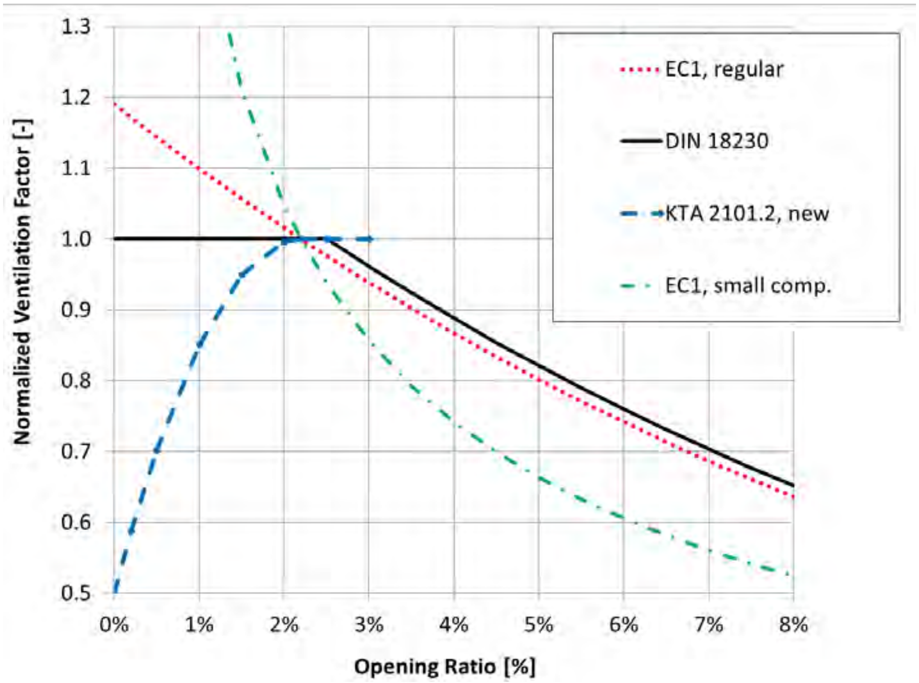

Figure 4 Normalized ventilation factors of different standards

Comparison to Experimental Data

An evaluation of average compartment temperatures depending on a factor which is calculated as relevant compartment surface area A_T (area of the enclosing walls without ventilation area plus ceiling area) divided by the ventilation factor $A_v * h^{0.5}$ of door openings was published by Thomas and Heselden and reproduced by Drysdale [12] (Figure 5). The factor A_T / ($A_v * h^{0.5}$) is basically reciprocal to the opening factor. Therefore the ventilation controlled fires are found on the right side and the fuel controlled fires on the left. The studied enclosures had floor areas of A_f = 2 m^2 (circles and rhombi), 4 m^2 (triangles), and 16 m^2 (squares).

Solid points are means of 8 to 12 experiments. The stoichiometric fires with maximum temperatures occurred at about $A_T / (A_v * h^{0.5}) \approx 12\ m^{-0.5}$.

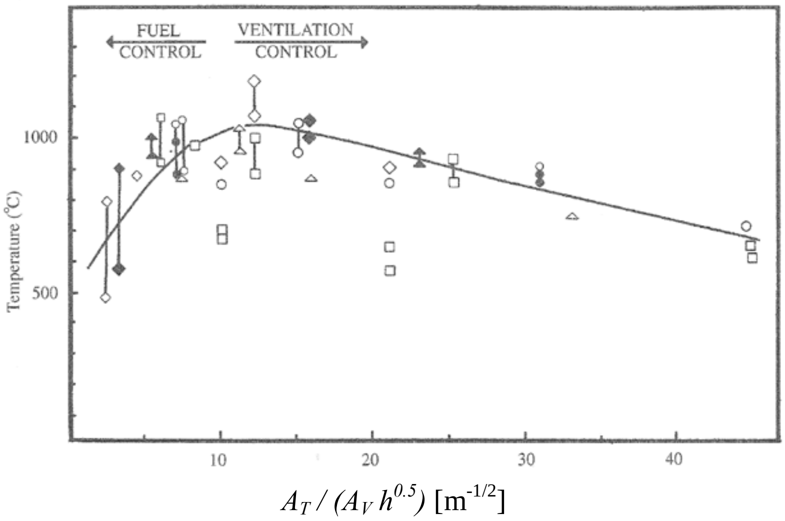

Figure 5 Average compartment temperatures during the steady state burning for wood crib fires in model enclosures as a function of a factor $A_T / (A_v * h^{0.5})$ [12]

The given data was re-evaluated and the opening ratio A_V / A_f was calculated together with a regression curve by a third order polynomial (Figure 6). Maximum temperatures in these model enclosures occurred at opening ratios of about A_V / A_f = 40 % - much larger than the assumed maximum for the ventilation factor in the non-nuclear standards (e.g. 2.5 % in the German standard DIN 18230). According to equation (4) an opening ratio of A_V / A_f = 40 % supports a maximum HRRPUCA of 880 kW/m^2.

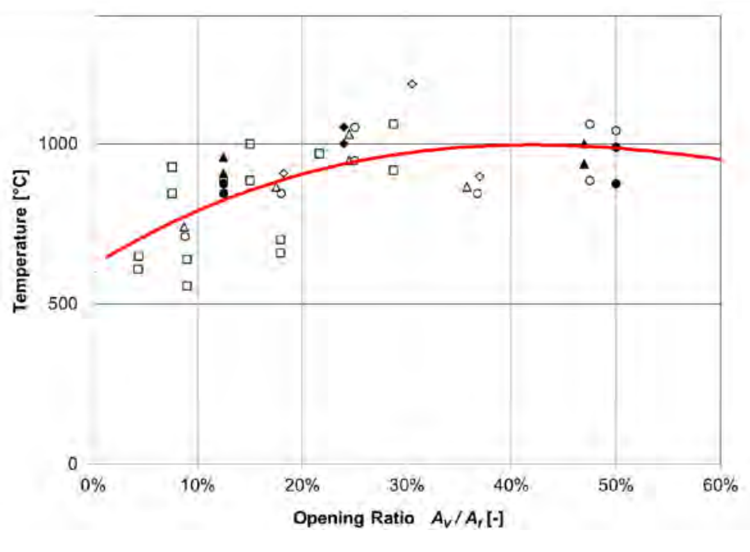

Figure 6 Average compartment temperatures during the steady state burning for wood crib fires in model enclosures as a function the opening ratio A_V / A_f

Nuclear power plant compartments are about one to two orders of magnitude larger in floor area than the studied model enclosures and industrial buildings are even larger two to four orders of magnitude. Therefore, the opening ratios where maximum fire severities are reached in real-scale compartments will shift to smaller ratios compared to Figure 6, because larger compartments will support a smaller average HRRPUCA than small compartments. For industrial buildings, the ratio of maximum fire severity will occur at smaller opening ratios than in compartments of a nuclear power plant.

The rating method as provided within the Appendix of the recent version of KTA 2101.2 [3] can only be applied up to an (effective) opening ratio of $A_v / A_f = 0.03$, which allows for a HRRPUCA of 60 kW/m^2. For comparison, in case of a cable room in a nuclear power plant with an assumed HRRPUCA of 100 kW/m^2, given a door height of $h = 2$ m, the opening ratio which allows stoichiometric conditions is $A_v / A_f = 0.045$. If such a large effective opening ratio would be given in a nuclear power plant compartment, the rating method cannot be applied and special considerations, e.g. by specific fire simulations, are needed.

Natural and Forced Ventilation under Conditions of Typical NPP Compartments

Taking into account the real conditions of compartment fires in NPP, there are additional influences from natural and forced ventilation to be considered:

- The oxygen supply by natural ventilation through doors that do not open to atmosphere but to adjacent compartments is much smaller than assumed by the Kawagoe equation. This is first because the pressure difference over a door to open atmosphere is larger than over a door to an adjacent compartment. Moreover, the atmosphere circulates between the two compartments and therefore the oxygen content is depleted.
- Forced ventilation of compartments is typically carried out by a combined push and pull system. Due to the compartment gas temperature, the smoke gas expands and the mass flows according to design will be reduced. The extent of this effect depends on the actual pressure-flow-characteristics of the HVAC system.

As natural and forced ventilation are principally overestimated with regard to real fire conditions in NPP compartments, the design with the nominal values is conservative as long as the ventilation factor f_{Av} is monotonically increasing with the opening ratio $A_{V,eff} / A$. This is achieved by the recent draft version of the German nuclear standard KTA 2101.2 [5].

CONCLUSIONS

A mayor uncertainty for structural design methods in fire safety engineering results from the source term for the heat release rate. For ventilation controlled fires, at least the maximum HRR in a compartment is known, which allows a conservative design. For fuel controlled fires, the uncertainty with respect to the resulting ventilation factor is larger. Therefore, opening ratios leading to large uncertainties are no longer covered by the design method provided in the Appendix of the German nuclear standard KTA 2101.2 and need special consideration. However, the bunkered buildings of NPP will in most cases provide compartments with boundary conditions that allow a simple and conservative design of structural elements according to the amended KTA method.

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ENHANCEMENTS IN PSA REGULATION AND GUIDANCE ON FIRE RISK ANALYSIS FOR NUCLEAR POWER PLANTS

Heinz-Peter Berg[1]

[1] Bundesamt für Strahlenschutz (BfS),
Willy-Brandt-Str. 5, 38226 Salzgitter, Germany

ABSTRACT

Fire PSA (*p*robabilistic safety *a*nalysis) is required as part of the probabilistic safety assessment within Periodic Safety Reviews (PSR) in Germany. Fire PSAs have been conducted for all nuclear power plants in Germany within the second PSR. Thus, Fire PSA has become an additional tool to supplement deterministic assessment of the fire protection for supporting decision making, but has been focused on full power plant operational states.

However, according to the German PSA Guideline and its corresponding technical documents on PSA methods and data issued 2005, the scope was extended to low power and shutdown states, however not assessing internal and external hazards for these states.

The most recent activities with regard to PSA as supplementary tool for safety assessment focus on improvements with respect to low power and shutdown (LPSD), considering fuel damage states, and in addition covering internal hazards, in particular fire. For each phase during LPSD those compartments or plant areas have to be identified where fire events may inadmissibly impair items important to safety.

The extension of Fire PSA to LPSD states has to be particularly applied to those plants, for which a third PSR is required by the German regulation. In addition, the safety significance of those plant modifications important to safety should be evaluated, for which a significant effect on the PSA results can be expected. This also covers the safety demonstration that the fire safety means needed to meet the nuclear protection goals are adequate.

INTRODUCTION

Operating nuclear power plants in Germany have been designed and constructed in different plant generations resulting in differences in the design and layout of fire protection features. Thus, it was necessary to assess the currently realized fire safety status of the nuclear power plant and its suitability.

In the past, the safety concept of nuclear power plants as well as licensing decisions by the competent regulatory authorities in charge and their experts in Germany were mainly based on deterministic principles such as prevention or control of abnormal plant operational conditions and incidents by technical means to ensure high reliability. In the meantime, probabilistic safety analyses (PSA) have come into effect as analytical tools. They are mainly applied in the frame of Periodic Safety Reviews and shall supplement deterministic safety demonstrations in order to verify the balance of the plant design related to safety.

However, in general the regulatory framework for nuclear power plants in Germany is still a deterministic one comprising comprehensive and partly very detailed regulatory documents, guidelines and recommendations of the regulatory body and advisory bodies, but also nuclear safety standards and codes incorporated in a corresponding pyramid type legal structure as shown in Figure 1.

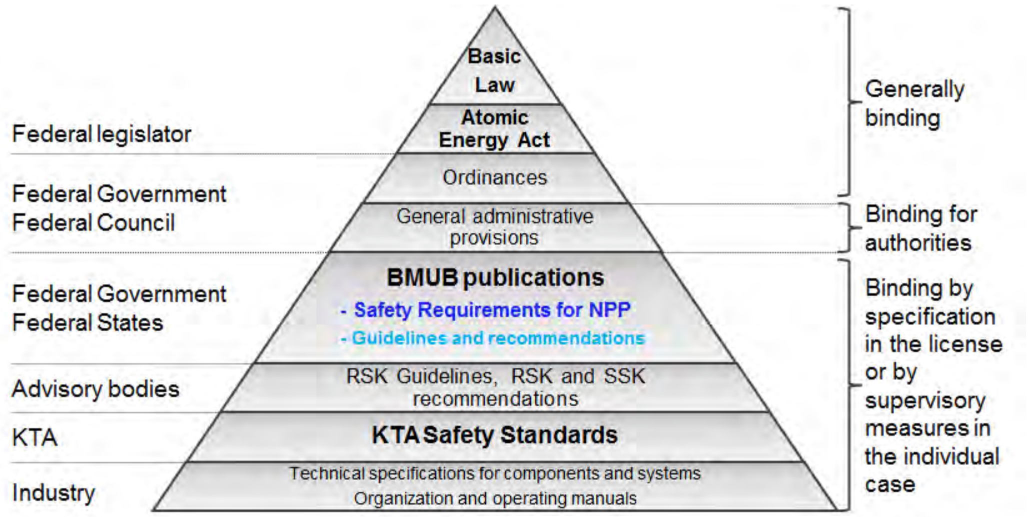

Figure 1 Nuclear regulatory framework in Germany

The German nuclear regulatory framework has been significantly enhanced in the recent past promulgating state-of-the-art "*Safety Requirements for Nuclear Power Plants*" issued by the Federal Ministry of the Environment, Nature Conservation and Reactor Safety [1]. It is systematically structured and follows the safety approach of defence-in-depth in accordance with the safety principles of the IAEA [2]. Moreover, it addresses the requirements laid down in the reference levels issued by the Western European Nuclear Regulators´ Association [3].

The recent enhancements also address the adequate consideration of internal as well as external hazards, such as fires, and with particular consideration of event combinations of fires and other anticipated events. Hazards from malevolent actions are out of scope of the German Safety Requirements [1].

In Annex 3 of [1], the basic requirement is a complete and systematic consideration of all hazards to be analysed. Moreover, event combinations of different hazards and/or hazards with other anticipated events need to be addressed systematically and considered as far as they cannot be excluded according to probability reasons. Annex 3, Subsection 3.1 "*General requirements*", provides more detailed guidance on the safe installation of protection means against internal hazards such as fires:

"3.1 (1)

Plant specifically identified and evaluated internal hazards as well as their potential combinations or their combinations with external hazards including very rare human induced ones shall be fully considered.

3.1 (2) For each hazard or combination of hazards according to Subsection 3.1 (1), the safety related impacts on the plant under consideration shall be determined considering the consequential impacts to be expected. In particular, the effects listed in the following shall be considered:

– *Plant internal flooding,*
– *Plant internal fires and explosions,*
– *…*

3.1 (3) Features for the protection against internal hazards shall preferably be installed close to the potential source of an internal hazard unless any other location is more advantageous with regard to safety."

Moreover, the Safety Requirements address safety demonstrations by deterministic as well as probabilistic safety assessment in more detail. In this context, PSA shall now also supplement deterministic safety demonstrations to assess the safety significance

- of modifications with respect to measures, equipment or the operating mode of the plant, as well as
- of findings that have become known from safety-relevant events or phenomena that have occurred and which can be applied to the nuclear power plants in Germany that are referred to in the scope of application of [1],

for which a significant effect on the PSA results is to be expected.

This new requirement has, of course, to be also applied to modifications with respect to fire protection.

GUIDANCE FOR FIRE PSA

The structure of the guidance documents providing guidance also for Fire PSA to be carried out for nuclear power plants in Germany is shown in Figure 2. The fundamental boundary conditions for performing PSA including requirements with respect to their documentation are provided in [4]. This PSA guide contains reference listings of initiating events for nuclear power plants with PWR (pressurized water reactor) and BWR (boiling water reactor) 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 corresponding technical documents on PSA methods [5] and PSA data [6]. However, the guidance on Fire PSA was limited to the full power operational state.

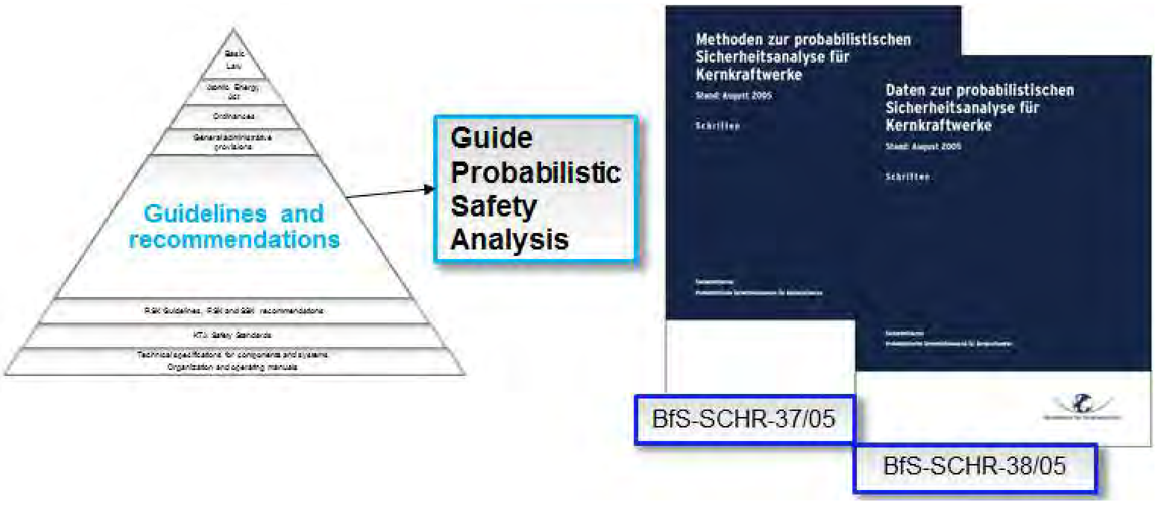

Figure 2 Guidance documents addressing Fire PSA for nuclear power plants in Germany

For fire risk assessment, different screening approaches are being applied in order to identify critical fire compartments. The models proposed have been successfully applied in several fire risk studies for German nuclear power plants.

For the detailed quantitative fire risk analysis, a standard event tree has been developed with nodes for fire initiation, ventilation of the room, fire detection and suppression, both for the incipient fire phase as well as for fully developed fires, and a node for fire propagation.

This standard event tree has to be adapted to each critical fire compartment, revealing the frequency and nature of fire induced initiating events, the list of equipment damaged, binned corresponding to different damage states, and damage frequencies.

If a complete plant specific, at least Level 1 PSA is available, the fire induced hazard states 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 included and adopted into the fault trees. The plant hazard state frequencies are estimated for each transient as the sum of the individual event core damage frequencies. The total plant hazard state frequency is obtained by summarizing the contributions of all transients. Moreover, the fire induced core damage frequency has to be calculated for the full power operational state.

The "*Safety Requirements to Nuclear Power Plants"* in [1] further demand that state-of-the-art methods, models and data have to be applied in the frame of PSA. As far as possible plant specific data have to be applied. If no suitable plant specific data are available from the operating experience, generic data may be used if justified [6].

In the meantime, guidance for performing fire PSA for low power and shutdown (LPSD) plant operational states (POS) – including the post-commercial safe shutdown state – has been developed and provided in detail in [7] and is outlined below.

For each phase during low power and shutdown those compartments or plant areas have to be identified, where fire events may inadmissibly impair items important to safety. In particular, the following changes and enhancements with respect to the input data have to be considered:

– The total compartment inventory of SSC including cables has to be associated to the POS in accordance with their safety functions to one of the following three classes (1 – no safety relevance, 2 – basic event in the PSA plant model, 3 – failure may contribute to an initiating event).
– Depending on the POS, changes of the status with regard to fire protection in given compartments are possible (e.g., BWR containment no longer filled with inert gas). These changes as well as well as those with respect to fire barriers, amount and distribution of fire loads, effects of maintenance and repair work, changes in the number of humans present and the duration of their presence in given fire compartments have to be considered for Fire PSA.
– In case of a fire induced loss of the entire compartment inventory, the list of potential initiating events should be adapted and, if necessary, further completed depending on the respective POS.
– For those compartments and/or plant areas identified to be significant within the screening, detailed analyses have to be carried out covering the following three analytical steps:
    - Fire occurrence frequency estimation:
      For that purpose, the methods in place for full power operational states can be applied, taking into account the changes in fire safety according to the conditions during LPSD.
    - Fire damage frequency calculation:
      The methods for full power operation (fire specific event tree analysis) can be applied.
    - Fuel damage frequency estimation:
      For this calculation, the event tree analysis of the respective initiating event from

LPSD PSA can be used. It has to be checked, if those human actions needed for the control of initiating events still can be performed in case of fire.

The guidance for LPSD plant operational states will be part of an additional supplementary document to [5] and [6] which is intended to be issued by the end of 2015 [7].

In the context of modelling plant specific fire event and fault trees, reliability data with regard to fire protections means are required. In [6], technical reliability data for various active fire protection features have been provided resulting from plant specific analyses of the operating experience from different nuclear power plants.

In order to update the already existing reliability data for fire protection features, in particular, failure rates per hours of plant operation, as well as to extend the database covering additional plant units the operating experience from inspections of the following components and systems installed in six nuclear power plant units has been further analysed:

- Fire detection systems with the corresponding main fire alarm panels, subsidiary fire alarm boards, detection drawers, detection lines/groups as well as automatic and manual fire detectors,
- Fire dampers and smoke extraction dampers in ventilation ducts with different actuation mechanisms (thermally by fusible link or remote controlled by typically electromechanical or pneumatic actuation),
- Fire doors between rooms, partly equipped with electrically hold-open devices, and
- Stationary fire extinguishing systems and equipment including the corresponding extinguishing media supplies, fire water pumps, hydrants, etc.

The investigations have been carried out by analysing the documentation of periodic in-service inspections as well as additional information and reports which resulted from the inspection findings. In case of more complex systems such as the fire detection systems, fault trees are presented for estimating the system's reliability in addition to the components' reliability data.

These data are then used in a second step to provide generic data, based only on the operational experience from nuclear power plants in Germany.

The updated and extended generic database covers 111 plant operational years of in total six German nuclear power plants units of different type (PWR as well as BWR) and age. This new set of reliability data for fire protections means will be part of the supplementary document [7]. More details regarding the derivation of the new data set are provided in [8] and [9].

These data may also be applied as a-priori information for estimating the reliability of components of a similar design and an equivalent inspection and maintenance practice in the frame of Fire PSA for nuclear power plants in other countries.

As already explained in the first section of this paper, PSA shall also supplement deterministic safety demonstrations to assess the safety significance of modifications of measures, equipment or the operating mode of the plant, as well as of findings that have become known from events relevant to nuclear safety or phenomena that have occurred and which can be applied to the nuclear power plants in Germany that are referred to in the scope of application of [1], for which a significant effect on the results of PSA can be expected.

Such modifications of measures, equipment or the operating mode of the plant must neither lead to an increase in the average core damage frequency (CDF) nor in the average frequency of large and early releases (LRF/LERF) if compared to the unchanged conditions of the plant. This is valid for all plant operational states, power operation as well as the entire low power and shutdown states, considering all plant internal events as well as all internal and external hazards including the very rare human induced external ones.

Hence, the new German Safety Requirements [1] contain an implicit definition of quantitative safety criteria: Mean CDF and LERF of full scope Level 1 and Level 2 PSA, respectively, must not increase due to any planned plant modification. However, no absolute value is given, by which the current risk of the plant can be assessed to be acceptable.

The CDF values have been calculated in the frame of the comprehensive (Periodic) Safety Reviews. The results of the latest safety review for the respective NPP provide the basis for the comparison in case of modifications.

This requirement in [1] results in the extended use of PSA to regulatory issues beyond PSR, such as a regulatory oversight on modifications applied by the licensee. A typical application in the past was the assessment of modifications with respect to in-service inspection intervals.

EXAMPLE OF THE CONSIDERATION OF A PROPOSED MODIFICATION

A recent example for the application of PSA according to the new requirements in [**1**] is provided in more detail in [**10**] and [**11**]. The licensee has recently requested the regulatory body for approving technical plant modification concerning the spent fuel pool cooling. Major differences of the intended plant modifications compared to the original situation are the number of emergency power supply systems available and the systems used for cooling the spent fuel.

A licensee plans technical modifications with regard to the spent fuel pool cooling. The Level 1 PSA for internal events has provided the result that the risk (here: annual frequency of fuel damage states (FDF)) is about factor 2 lower after the modifications. For approval by the regulatory authority to start licensing of the intended modification, an improvement has to be demonstrated for internal and external hazards as well.

The two alternatives of spent fuel pool cooling in the plant under investigation are outlined in Figure 3.

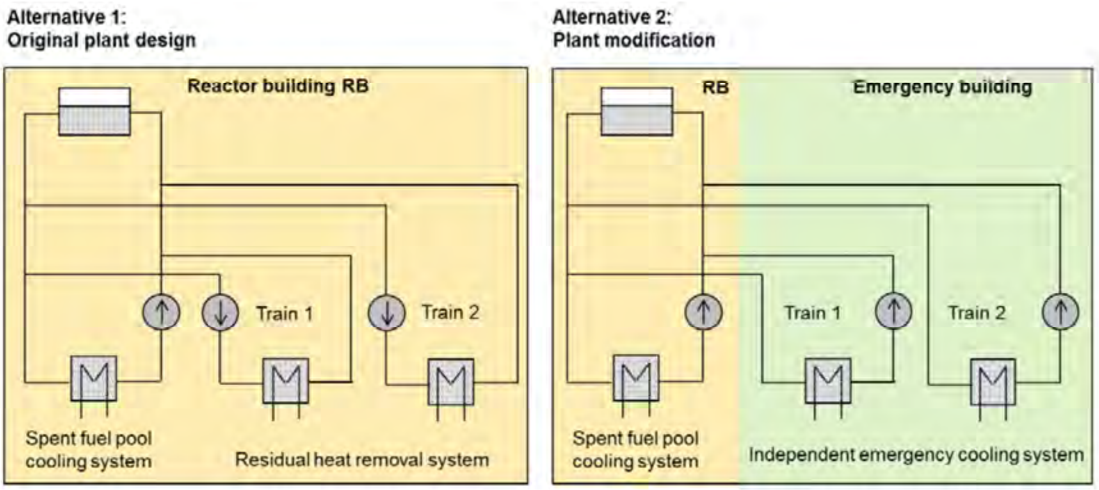

Figure 3 Alternatives of spent fuel pool cooling, from [10]

As outlined in more detail in [10] and [11] by the analysts from GRS having carried out Level 1 Fire PSA for the affected plant and also performed a probabilistic assessment of the two alternatives for spent fuel pool (SFP) cooling with respect to internal and external hazards on behalf of the Federal Office for Radiation Protection (BfS), the cooling of the SFP is normally done by the spent fuel pool cooling (SFPC) system. In case of the original alternative 1 (without plant modification), the residual heat removal (RHR) system takes over with two redundant trains, if the SFPC system is not available, either unintentionally due to a failure or due to an intended outage, e.g. for maintenance reasons. In this context, it is important that the main parts of the SFPC system and the RHR system are located inside the reactor building. In case of the alternative 2 (intended plant modification in place), two redundant trains of the independent emergency cooling (IEC) system will be used, if the SFPC system is not available. In contrary to the SFPC system and the RHR system, the IEC system is located inside the bunkered independent emergency systems (IES) building. This building is protected against fires as well as against human induced external hazards (aircraft crash, explosion pressure (blast) waves). The IES has independent ultimate heat sinks. The purpose of the intended plant modification is to allow the licensee starting already deconstruction and decommissioning activities in given areas of the reactor building, while there is still fuel on site.

For the comparison of the two alternatives, the unchanged as well as the modified Spent fuel pool cooling, the following initiating events have been considered: loss of offsite power (LOOP), failure of the residual heat removal (RHR) from the spent fuel pool, loss of water from the spent fuel pool, and flooding induced unavailability of the required system functions of the independent emergency systems. Fires may cause the initiating events LOOP and RHR failure from the spent fuel pool.

The quantitative analyses gave the result that the risk of fuel damage is much lower in case of the second alternative of spent fuel pool cooling.

The example clearly demonstrates the benefits of the enhancements in the German regulations and the need for explicit guidance as provided in [7] even if the majority of nuclear power plants will stop commercial operation within the next seven years since this tool will be applicable to the probably longer duration post-commercial safe shutdown phase, during which fires still pose a non-negligible risk.

CONCLUDING REMARKS

Nuclear power plants in Germany are mainly designed and licensed on the basis of deterministic fire safety assessment and according to the existing fire safety standards.

Meanwhile, Fire PSA is required in Germany as part of the PSA within Periodic Safety Reviews. Therefore, Fire PSA has been conducted for all nuclear power plants in Germany within the second PSR and is recognised as an additional valuable tool to supplement deterministic assessment of the fire protection for supporting decision making.

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 used in case of deterministic fire hazard analysis and probabilistic fire risk analysis and assessment.

It has to be stated that according to the German PSA Guide [4] and its corresponding technical documents on PSA methods [5] and data [6], Fire PSA in the past focused on full power plant operational states. In the frame of the second PSR the scope was extended to low power and shutdown states for internal events but not fully to internal hazards. A Fire PSA for LPSD has to be performed for those plants, for which a third safety review is required to be conducted due to the German regulations [12]. Guidance for LPSD plant operational states will be part of an additional supplementary document to [5] and [6] which is intended to be issued by the end of 2015 [7].

The recently issued state-of-the-art "*Safety Requirements to Nuclear Power Plants*" [1] also underline the need for an adequate fire safety in the design and operation of nuclear power plants as well as the demonstration of the reliability of fire related equipment by deterministic and probabilistic safety assessments.

In this context, PSA shall, on the one hand, supplement deterministic safety demonstrations with regard to the balance of the safety related plant design and, on the other hand, supplement deterministic safety demonstrations to assess the safety significance of

– modifications with respect to measures, equipment or the operating mode of the plant, as well as

– findings that have become known from events relevant for nuclear safety or phenomena that have occurred and which can be applied to the nuclear power plants in Germany that are referred to in the scope of application of [1],

for which a significant effect on the results of the PSA is to be expected. This procedure has, of course, also to be applied to modifications with respect to fire protection.

Moreover, there are considerations or already practical actions to remove equipment not needed anymore for nuclear power plants under post-commercial safe shutdown, although the formal decommissioning process has not yet started. In this context, different aspects related to fire safety have always to be taken into account.

Fire protection remains an important topic for nuclear power plants in Germany, even though eight out of seventeen plant units have been finally shutdown in 2011 and, at the time being, only eight units are still being operated commercially. One reason is that the spent fuel elements will remain either in the containment or in the spent fuel pool for further years; this still requires appropriate fire protection means being in place.

In addition, the new "*Safety Requirements to Nuclear Power Plants"* [1] resulting also in an extension of Fire PSA have already been successfully applied for assessing intended plant modifications by the licensees for an example case demonstrating the value of probabilistic risk assessment for fire safety.

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INTERNAL FIRE PROTECTION ANALYSIS FOR THE UNITED KINGDOM EPR DESIGN

Abdallah Laïd[1], Mickaël Cesbron[2]

[1] EDF Energy Plc., Nuclear New Build Generation Company Ltd. (NNB GenCo), Barnwood, Gloucester, United Kingdom

[2] EDF-SA, Service Etudes et Project Thermiques et Nucléaires (SEPTEN), Lyon, France

ABSTRACT

In the deterministic design basis analysis of the United Kingdom (UK) EPR based nuclear power plants all postulated initiating events are grouped into two different types, internal faults and internal/external hazards. "Internal Fires" is one of the internal hazards analysed at the design stage of the UK EPR. In effect, the main safety objective for fire protection is to ensure that all the required safety functions are performed in the event of an internal fire. To achieve this safety objective, provisions for protection against fire risks are taken to: (i) limit the spread of a fire, protect the safety functions of the facility; (ii) limit the propagation of smoke and dispersion of toxic, radioactive, inflammable, corrosive or explosive materials, and (iii) ensure the achievement of a safe shutdown state, personnel evacuation and all other necessary emergency actions. This paper presents the UK EPR approach on how the above provisions are applied. Such provisions involve implementing means of fire prevention, surveillance, firefighting and limiting fire consequences, appropriate to the risks inherent to the facility. Overall, the design of the UK EPR fire protection systems is based on three types of measures: prevention, containment and control.

Keywords:

UK EPR, Deterministic, Design Basis Analysis, Internal Hazards, Internal Fires, Prevention, Containment, Control.

GENERAL INTRODUCTION TO THE ANALYSIS OF INTERNAL HAZARDS

Internal Hazards (IHs) are events external to the Nuclear Steam Supply System (NSSS) that originate inside or outside buildings but within the site boundary and over which the operator has some form of control. IHs have the potential to cause adverse conditions or damage inside safety classified buildings.

The defence-in-depth (DiD) approach requires that all IHs liable to affect reactor safety should be taken into consideration at the design stage. IHs that could affect the plant must be identified, and provisions made to ensure that the risk from those hazards is commensurate with the overall frequency and release targets.

The overall UK EPR IHs design approach is as follows [1]:

- IHs identification including the consideration of credible combined or consequential hazards and the setting up of safety requirements;
- IHs impact quantification (e.g., specific loads and environmental conditions), and where necessary design basis protection of structures, systems and components (SSCs) against the impact;

- Design verification against IHs to confirm that the safety requirements have been fulfilled. This is systematically performed on a case-by-case basis (specific to each hazard, which has different characteristics) with the use of deterministic studies, such as building and equipment response, functional impact including consideration of consequential internal faults (for instance, identification of internal faults induced by an initiating internal fire hazard), etc. This process is supported by a probabilistic safety analysis (PSA) of hazards. This design verification phase can lead to design feedback including potential design developments or changes.
- Production of hazard protection schedules (HPSs) which consider and complement the design verification analyses.

The aim of the above approach is to demonstrate that classified safety systems can achieve their design objectives by minimising the occurrence of IHs and mitigating their potential consequences. This will confirm that the facility respects the UK EPR safety design objectives including the demonstration that the risks to workers and the general public are as low as reasonably practicable (ALARP).

IHs are postulated to occur during normal operating conditions or, in some cases, during post-accidental conditions. Credible combinations of hazards may be considered, with the exception of combinations which cannot reasonably be anticipated.

Evidence has to be provided that, in the event of an IH, the safety functions required to bring and maintain the reactor at a safe shutdown state and to limit radiological releases can be carried out satisfactorily.

Regarding IHs causing damage to SSCs, the design basis and design verification approach is to protect every safety categorised function required to mitigate PCC-2, PCC-3 and PCC-4 internal faults[1]. This protection is achieved by designing the SSCs to withstand the loads associated with the IHs events, or by providing physical separation between redundant elements so that the safety function can be performed despite the occurrence of the IH.

As a complement to the design verification hazard analyses, HPSs are being specifically produced for the UK EPR design. More details on the production of HPSs are presented in a later section of this paper.

Protection Principles Common to all Internal Hazards

In order to implement the IHs design approach described in the section above, the design and installation protection principles shall be such that, generally and so far as is reasonably practicable [2]:

- IHs do not prevent the fulfilment of Category A and B Safety Functions[2] claimed for the PCC analysis, even if the functions are not required after such an event.
- IHs do not trigger any PCC-3 / PCC-4 event.
- IHs do not compromise the separation of divisions.
- The frequencies of IHs that might trigger a PCC event are commensurate with the PCC's overall frequency and release targets.

In addition, an IH shall not undermine, so far as is reasonably practicable:

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1 The PCCs are the initiating events (internal faults) used as the design basis for the SSCs of the UK EPR reference plant. The PCCs are grouped into Categories 1 to 4, depending upon their anticipated frequency of occurrence (f), as follows: PCC-1: Normal operating transients; PCC-2: Design basis transients ($f \geq 10^{-2}$ pry); PCC-3: Design basis incidents ($10^{-2} > f \geq 10^{-4}$ pry); PCC-4: Design basis accidents ($10^{-4} > f \geq 10^{-6}$ pry).

2 Category A – any function that plays a principal role in ensuring nuclear safety, Category B – any function that makes a significant contribution to nuclear safety, and Category C – any other safety function.

- A Category A or B Safety Function claimed for PCC analysis, where this could lead to the loss of the function.
- the stability or integrity of:
  - The primary circuit pressure boundary (except in the case of a Loss of Coolant Accident (LOCA)).
  - Reactor internals, including the fuel assemblies.
  - Main steam and feedwater water pressure boundary.
  - The spent fuel pool and its internals, including any stored fuel assemblies.
  - Safety classified buildings, and
  - High integrity components.

The next sections provide an overview of the proposed UK EPR safety analysis for the "*Internal Fires"* hazard.

PROTECTION AGAINST INTERNAL FIRES

Safety Objective

A specific safety objective for fire protection is defined to implement and comply with the IHs design approach described in the first section. This objective is to ensure that the necessary safety functions are performed in the event of a fire inside the installation where the fire has the same characteristics as the reference fire[3], and implies that [3]:

- The fire effects shall be limited so that sufficient SSCs must remain available to permit a safe shutdown state to be reached and maintained, and limit radiological releases to below acceptable limits.
- The non-redundant systems and equipment, which perform the required nuclear safety functions, must be protected against the effects of a fire in order to ensure continuous operation.
- A fire must not compromise the habitability of the main control room (MCR). In the event that the MCR cannot be accessed, the accessibility and the habitability of the remote shutdown station (RSS) must be assured.

Fire is normally assumed to occur, during plant operation, in any room which contains combustible materials and ignition sources.

Consequential Fires

Fires could also occur as a consequence of other internal faults, internal hazards or external hazards. In such cases the fire protection requirements include the following [3].

- <u>Protection requirements for internal fires due to internal faults</u>:
  PCC or DEC-A or DEC-B that could lead to "internal fires" are: "LOCAs" and "Severe Accidents". This is due to the fact that during these events there is a potential release of Hydrogen in the containment. The necessary measures for designing the containment as well as the equipment necessary to eliminate the potential ignition of Hydrogen, or to control a Hydrogen fire or explosion are described in a dedicated "internal explosions" section of the IHs protection chapter of the safety report.

[3] The reference fire is considered as the fire which may break out in any fire area of the plant and which has the most onerous consequences (duration, severity). For a given room, it is a fire of all available loads in that room.

- Protection requirements for internal fires due to other internal hazards:
  The IH “Release of hazardous chemicals or noxious substances from on-site sources” could lead to “internal fires” in specific locations of the plant. Protection measures against “internal fires” induced by the release of Hydrogen are described in a dedicated “internal explosions” section of the IHs protection chapter of the safety report.
  The other IHs “internal explosions”, “dropped or impacting loads”, “internal missiles” and “direct vehicular impact from heavy transport within site” could potentially cause “internal fires”. The required protection measures against “internal fires” induced by these IHs are described in their dedicated sections of the IHs protection chapter of the safety report.
- *Protection requirements for internal fires due do external hazards:*
  Design rules for protection against “internal fires” induced by an “earthquake” have been produced. All the fire protection components must comply with the criteria of the earthquake effects analysis. They also must not impair the performance of safety functions as a result of either their operation or failure.
  The combination of an “aircraft crash” and an “internal fire” in a building is not applied when designing the fire protection systems. However, the firefighting water network design (geographical separation and structural protection) will ensure the availability of means of emergency firefighting in the event of an aircraft crash.
  All materials required for fire protection and concerning safety must be protected against conditions of “extreme cold”.

Fire Consequences

An “internal fire” must not cause the loss of non-redundant nuclear safety classified equipment. Otherwise, this equipment must be protected or the potential for a fire must be eliminated [3].

- Protection requirements for internal faults due to internal fires:
  Should an “internal fire” lead to a “PCC-2 internal fault”, then adequate system redundancies must remain available to control the event. Where reasonably practicable a fire must not lead to an additional PCC-3/PCC-4 event.
- Protection requirements for other internal hazards due to internal fires:
  The IHs “internal explosions”, “dropped or impacting loads”, “internal flooding” and “release of hazardous chemicals or noxious substances from on-site sources” could be caused by “internal fires”. The required protection measures are described in the sections of the IHs protection chapter of the safety report dedicated to those IHs.

Combined Hazards

The following cases of combined events are taken into consideration for DiD protection and robustness of the design [3].

- *PCC-2 to PCC-4 events*:
  An independent “internal fire” is only assumed to occur during the post-accident phase and after a controlled condition has been reached following a “PCC-2 to PCC-4 event”. Nevertheless, the fire protection measures are available for the full duration of the post-accident phase. However, the possibility of a fire in the MCR during the post-accident phase following a PCC-2 to PCC-4 event is discounted in the design. This is justified by the availability of sufficient fire protection measures and the presence of operating staff who would be able to rapidly extinguish the fire.
- *DEC-A or DEC-B events*:
  DEC-A or DEC-B type events are very infrequent. As a result, the combination of a

"DEC-A or DEC-B event" with an independent "internal fire" is assumed to occur only during the post-accident phase and no earlier than two weeks after the event.

- Design Basis Earthquake[4]:
  An independent "internal fire" is assumed to occur only during the post-accident phase and no earlier than two weeks after a "*design basis earthquake*". The following protection concepts are applied:
  - The detection and extinguishing systems within a fire compartment, where mechanical, electrical or instrumentation and control equipment for the performance of Category A or B Safety Functions are installed, must be subject to SC1 Seismic Requirements[5].
  - It is assumed that repair or replacement actions can be performed, if required, within a two week period after the occurrence of the event.

Fire During Shutdown Conditions and Maintenance Phases

The fire protection strategy described above must also be applied to shutdown conditions and maintenance phases [3]. The maintenance periods present a potential increase in the probability of occurrence of fires. However the presence of personnel will aid the rapid detection and extinguishing of fires, thus reducing the risk. Specific administrative procedures (like hot work permits, increased monitoring, etc.) must be applied to any situation which deviates from the above general fire protection strategy.

Specific attention is also paid to the introduction of additional combustible materials and ignition sources (e.g., welding operations, paint, solvents, etc.) as well as to possible degradations in the existing fire protection provisions (e.g., loss of fire compartment integrity due to an open fire door, etc.) during such periods. A specific fire safety analysis for each shutdown period must be provided. The need to introduce combustible materials is taken into account in the plant design with the provision of dedicated storage cells which are described below.

Principles of the Fire Protection Strategy

For the UK EPR the main strategy for protection against the impacts of "internal fires" is deterministic. This strategy is complemented by a probabilistic safety assessment, but the principles of the fire protection strategy are deterministic and these are given below [4]. These principles ensure the safety objective defined in the section above is met.

- A fire is assumed to occur in any plant room, which contains combustible materials and an ignition source.
- From a hazards analysis point of view coincidental occurrence of two or more fires, from independent causes, affecting rooms in the same or different plant is not taken into consideration due to the low probability of a second fire occurring during the relatively short time until the first fire is extinguished. However, measures to protect the plant against individual fires must be assured.
- The combustion of any combustible material present in buildings must be considered, except equipment or materials protected by a fire resistant housing or cabinet.
- The auto-ignition of low and very low voltage cables is considered to be very unlikely.

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[4] A design basis earthquake is a suite of vibratory ground motions spectra chosen on the basis of the likely seismicity and geology at and around the NPP site.

[5] SC1 represents the set of seismic requirements which ensure that a safety function, needed to bring the plant to a stable state and maintain it, can be delivered in the case of an earthquake.

- Limitation of fire spreading using either the fire containment approach (fire compartments) in buildings separated into divisions or the fire influence approach (fire cells) in buildings or parts of buildings without divisional separation.
- A fire is assumed to occur during normal plant conditions (from full power to shutdown condition) or in a post-accident condition once a controlled state has been achieved or no sooner than two weeks following an earthquake due to the short time at risk and the low probability of a fire occurring in this time.
- In order to be able to set up suitable protective measures, the fire load for each room must be calculated and kept up-to-date.
- The temporary or permanent storage of fire loads during the various states of the plant as well as workshops with fixed, hot working work stations, must be identified and subject to risk analysis.
- The fire protection provisions must be optimised in order to limit the discharge of toxic or radioactive materials.
- The random failure of an active equipment item of the fire protection systems must not lead to a common mode failure of the systems needed to perform Category A or B safety functions, even if these functions are not needed following such an event. The redundancy requirement (whether functional or not) due to this principle being taken into account must be implemented within the train separation principles.
- The random failure must be applied on a deterministic basis:
  - To the active equipment of the fire protection mechanical systems,
  - To all the components of the fire protection electrical systems.
- A check on the robustness to a random failure must be applied on a deterministic basis in the event of:
  - A fire independently of the accidents, liable to impair the integrity of the fire barriers,
  - A fire leading to PCC-2 events,
  - A fire resulting from a PCC-3/PCC-4 event.
- A localised loss of integrity of the fire safety barriers may be accepted insofar as the failure of an active equipment item of the fire protection systems does not lead to a common mode on the systems required to perform Category A or B safety functions.

Applicable UK Regulations and Design Codes

An example of applicable UK regulations is given below (list not exhaustive):

- Regulatory Reform (Fire Safety) Order [4],
- Construction Design and Management Regulations [5],
- Dangerous Substances and Explosive Atmospheres Regulations [6],
- Control of Major Accident Hazard Regulations [7].

The applicable design code is the "*EPR Technical Code for Fire Protection*" (ETC-F) [8] and its associated UK Companion Document [9].

Design Basis Analysis

There are three sets of measures which provide DiD against "Internal Fires" [10]:

- **Prevention** (prevent fires from starting, prevention of spreading);

- **Containment** (fire compartments and cells, physical and geographical separation, smoke protection);
- **Control** (detection and extinguishing).

For each of these measures, it must be confirmed that an independent random failure will not undermine the fire protection safety objective.

Overall, the fire strategy for the UK EPR is mainly based on the elements of DiD depicted in Figure 1 below.

Figure 1 Elements of defence-in-depth underpinning the UK EPR fire strategy

Prevention

Prevention comprises a set of measures, aimed at preventing the fire from starting or reducing the likelihood of a fire. The requirements covering prevention are [10]:

- As a priority, the prevention measures (design and management arrangements) shall aim at limiting fire loads, to separate them or to remove them with a passive fire protection measure (e.g., enclosure or encasement), and to minimise potential ignition sources in the vicinity of combustible materials.

- The materials used shall be preferentially non-combustible (e.g.: A1 or A2s1d0 in accordance with BS-EN 13501-1 [11]).
- The use of combustible materials in fresh nuclear fuel stores is heavily restricted.

If not class A1 the material must at least be class Bs1d0 or Cs1d0 in accordance with [11] and must not produce dense or toxic smoke.

Fire Containment

Fire Compartmentation

If a fire starts, despite the preventive measures in place, measures must be taken to limit its spread and to prevent [10]:

- Impact on a system whose safety function is required to reach and maintain the safe state of the plant. Fire damage must be restricted to one redundant train in a given safety classified system.
- Spreading to other rooms and into emergency exits and disrupting any firefighting provisions.
- Environmental impact contravening applicable UK Regulations.

Limiting the spread of a fire is achieved by dividing the buildings into fire zones, which use physical or geographical separation principles. Any installed fire barrier must contain the fire so that only one of the redundant trains for a given Safety Class 1 or 2 system may be endangered by the fire. The requirements for separation are [10]:

- All nuclear safety classified buildings shall be separated from the others by partitions which are classified (at least) EI 120[6] if non-load bearing and (at least) REI 120[6] if load bearing, as required in the UK EPR Fire Protection Code ETC-F [8] and [9].
- Priority shall be given to physical separation (fire containment) rather than spatial separation. In the same way, priority shall be given to passive measures (fire rated compartments) rather than the provision of active systems (i.e., fire extinguishing).
- In case of fire, the redundant elements in a Safety Class 1 or 2 system must be protected so that failure is limited to a single train.
- Random failure is only to be considered for active equipment items such as fire dampers. Fire doors, smoke extraction ducts and floor drains are considered as passive equipment items that are not subject to the random failure requirement.

Table 1 below summarises the different types of fire compartments/cells:

Table 1 Different types of fire compartments/cells

Objective	Fire Compartment/Cells
Radioactivity containment	Type 1a/b
Nuclear safety	Type 2
Protected evacuation route	Type 3
Facilitation of the Intervention and limiting the unavailability	Type 4
Storage	Type 5

[6] Fire resistances ratings as defined in BS-EN13501-2 [12]: R (for structural resistance), E (for hot gases tightness) and I (for thermal insulation). The time criterion is given in minutes.

- The principles used take into consideration geographical separation (extinguishing – screen – distance). The containment is justified by taking into account the location of the concentrated fire loads and the combustible material properties. Fire cells must only be used in exceptional circumstances and their effectiveness must be demonstrated on both, fire propagation and radioactive or toxic waste release level.
- Where geographical separation is used, it will be justified by a fire hazard analysis.

In general, the UK EPR design considers three fire compartment types [10]:

1. The nuclear safety fire compartment (SFS) - (Type 2):
   This compartment is created to protect nuclear safety trains from a fire common mode. The partitions of these safety fire compartments shall have a fire resistance rating of at least 120 minutes. Active or passive means of fire protection shall be set up if necessary to guarantee their integrity after this time has passed. The fire resistance of the fire compartment shall be sufficient to withstand the complete combustion of the entire fire load within the compartment. The verification of the adequacy of the compartmentation is described under the sub-section 'Physical Separation' below.
   For the UK EPR, a door monitoring system will be used to detect if a door installed within the boundary of a safety fire compartment is left open and will raise adequate alarms (local and in the MCR) to alert the operator to an open or unlatched fire door. The doors, which will be equipped with such monitoring devices, contribute to the following safety functions:
   - Separation between buildings,
   - Divisional separation,
   - Segregation of safety trains.
2. The access compartment (SFA) - (Type 3):
   This compartment is intended to enable the personnel to be evacuated in full safety in the event of fire and to provide access to the firefighting teams and allow circulation of personnel for specific plant operation. It corresponds to a protected escape route. The partitions of these compartments shall have a fire resistance consistent with relevant UK regulations for adequate design of escape routes and access for the 'Fire Services', at least equal to the rating of the adjacent fire areas, without being less than 60 minutes. These compartments shall strictly minimise fire loads, and as far as possible not contain nuclear safety equipment.
3. The intervention fire compartment (SFI) - (Type 4):
   This compartment is created when the installation conditions imply the possibility of a flashover, to facilitate the intervention of firefighting crews and limit unavailability of the unit. The partitions of these fire sectors shall have a fire resistance rating suited to the consequences of the fire in the area without being less than 60 minutes.
   The size of these compartments shall be consistent with the above objective and, wherever possible, the same SFI should not be used to cover several building floors. An SFI may be:
   - included in an SFS,
   - independent of any SFS.

Examples of SFS, SFA and SFI fire compartments within a UK EPR building are shown in Figure 2 and Figure 3 below.

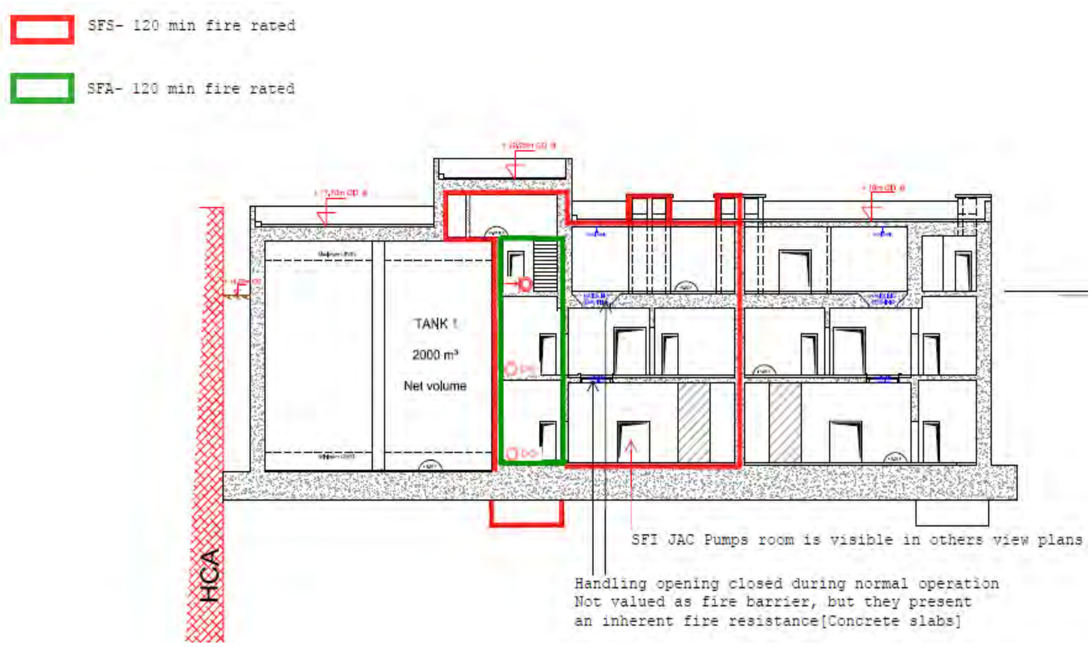

Figure 2 SFS and SFA fire compartments within the firefighting water building

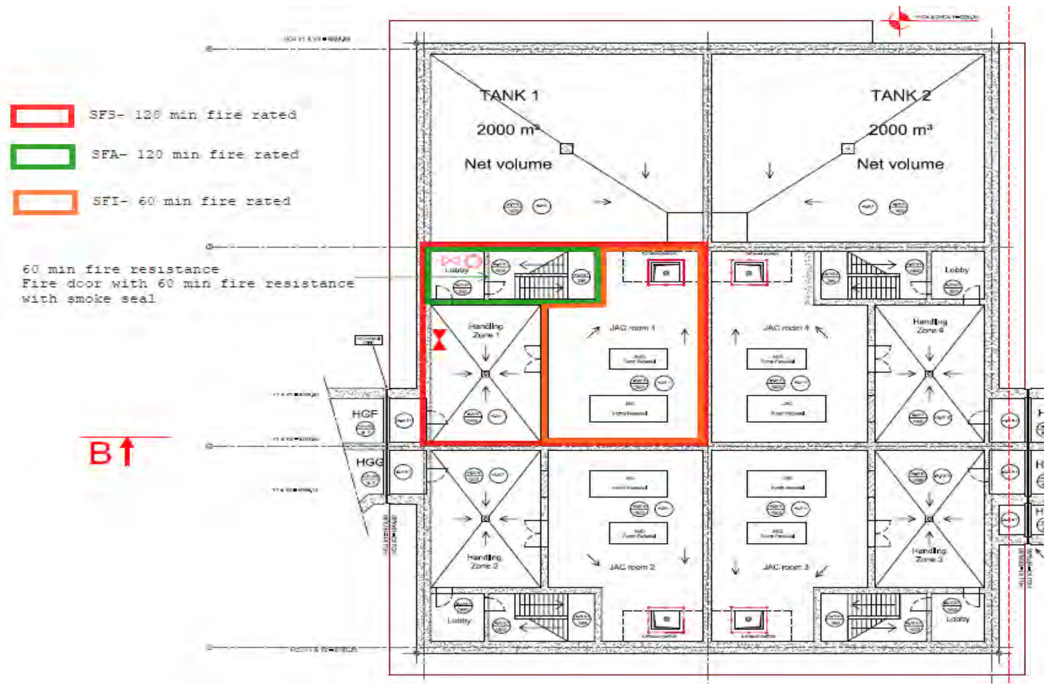

Figure 3 Example of an SFI – one of the SFIs within the firefighting water building

Fire Cells

In some buildings, in particular the Reactor Building, division into fire compartments may be limited due to construction or process factors, e.g.:

- Compact nature of the installation,
- Hydrogen concentrations (in case of a release),

- Steam releases in case of pipe break (rupture).

In this instance, some sections of the buildings may be divided into fire cells [10], where equipment is protected by spatial separation rather than physical barriers. Evidence of non-propagation of fire and avoidance of failures of safety classified equipment, must be established by assessing all possible modes of fire propagation and combustion products.

There are three fire cell types in the UK EPR design [10]:

1. The nuclear safety fire cell (ZFS) - (Type 2):
   These cells are created for nuclear safety purposes to protect safety functions needed to reach and maintain the safe state of the plant from common mode failure in case of fire. The partitions of these nuclear safety fire cells shall have a fire resistance at least equal to 120 minutes. Other means of protection shall be set up if necessary to ensure their integrity after this time has passed. As the ZFS contains at least one fire zoning equipment or openings which is not fire rated, fire risk analyses have to demonstrate the non-propagation and absence of malfunction of safety related equipment.
2. The unavailability limitation fire cell (ZFI) - (Type 4):
   These cells are created, when the installation conditions imply the possibility of a flashover and when it is not possible to create an SFI. They are also created when the installation conditions imply the possibility of a localised fire to limit the unavailability of the unit and facilitate the intervention of firefighting crews. The partitions of these fire cells shall have a fire resistance rating adapted to the fire risk analysis and of at least 60 minutes. As the ZFI contains at least one fire zoning equipment or openings which is not fire rated, fire risk analyses have to demonstrate the non-propagation.
3. The maintenance storage cell (ZS) - (Type 5):
   These cells are created during the design phase to enable the operator to store the equipment and materials required for operation, with the unit in operation, shutdown, outages and maintenance. The partitions of these fire cells shall have a fire resistance rating adapted to the fire risk analysis and of at least 60 minutes. These cells are fitted with fire prevention, detection and fighting means. If necessary, their design is based on the maximum stored fire load fixed by the operator, and takes into account partitioning. The fire load will be strictly controlled by the operator, to ensure it is not in conflict with that set out in the fire safety risk analysis.
   If possible, when the storage cell is associated with a single room, it will be reserved solely for the needs of the operator. If not possible, technical justification will be provided and sometimes dual coding will be present (e.g., ZS plus SFI due to the presence of flashover type cables). The creation of a storage (or warehousing) cell is appropriate for:
   - Rooms associated with "job" activities,
   - Rooms dedicated to temporary storage (equipment transfer, repairs and winding-down of site operations),
   - Rooms dedicated to permanent and temporary storage,
   - Rooms containing radiological sources.

Non-contained Areas

Non-contained areas (VNS):
These areas are created for rooms or groups of rooms for which no safety or nuclear safety concerns are raised after suitable analyses. They are used to justify monitoring and control of fire load parameters in these rooms. They could also be created to segregate non-controlled area from controlled areas.

Physical Separation

Separation is performed by the creation of fire compartments or by the use of fire-qualified passive protection features [10]. The fire resistance of these protection devices must be at least equal to the equivalent standard fire rating that bounds the reference fire curve. This reference fire curve is defined by the combustion of materials contained in the room and outside the enclosure. The fire protection ratings must not be less than the duration of the reference fire in the compartment or cell with the longest fire duration.

Where it is not possible to shield part of the equipment in a room from a fire, fire retardant paints can be used on the condition that all the combustible materials present are also coated.

Spatial Separation

The separation is achieved by creating fire cells or by using fire resistant barriers (e.g., screens) [10].

For spatial separations to be effective, fire risk analyses must conclude that the effects of fire will not spread to the protected equipment (temperature, heat flow, smoke, etc.).

Separation by Distance

The use of physical distance ensures that a localised fire cannot propagate towards other combustible materials [10]. This is achieved by installing the combustible materials in such a way that the space between them is free from combustible materials. Distance may also be used to prevent a fire causing failure in two redundant items of equipment. This is achieved by ensuring that there is sufficient distance between the combustible materials and at least one of the redundant items for damage to be avoided.

The distance required depends on the direct radiated heat and the hot gases generated and transported throughout the space.

As a large number of parameters (type of combustible, location in the room, severity of the fire load, etc.) are involved, it is difficult to define a general rule and as such the measures used are subject to specific assessments.

Separation Using Thermal Screens

This measure, which may be used to supplement the ‘distance’ measure, enables equipment to be shielded from direct radiated heat. A screen is installed whose fire resistance is sufficient to withstand the severity and duration of the reference fire.

Control of Fire

Fire detection and fixed firefighting systems are provided to detect and fight fires and to bring them under control. The control requirements are as follows [10]:

- The aim of fire detection is to detect fire at an early stage, locate it, trigger the alarm and initiate automatic actions when required. The fire detection system and features are classified at least Safety Class 3 when required for nuclear safety.
- The fire detection system must be operational in all cases where a fire is assumed to occur as far as technically practicable.

- In terms of firefighting, both manual and fixed extinguishing systems shall be provided. Manual systems consist of portable extinguishers and the provisions for ‘Fire Services’ firefighting. It should be noted that manual firefighting systems are not part of the demonstration of vulnerability analyses.

Vulnerability Analysis

In the first step of the analysis, where a fire is postulated in a nuclear safety fire compartment (SFS) or in a nuclear safety fire cell (ZFS), operational failure of all the equipment is assumed to occur (apart from those items of equipment which are protected by an approved fire barrier designed to resist the consequences of a fire).

The vulnerability analysis [10] shall demonstrate the suppression of any common mode or conclude in the acceptability of the risk incurred.

As a general rule, the effects of fire are limited to the investigated fire area, whether it is a compartment, a cell or a division. For cells, the analysis is also conducted between adjacent cells.

The analysis is conducted in four steps:

- **Step 1**: Search for potential common mode failures;
- **Step 2**: Perform a functional analysis;
- **Step 3**: Carry out an analysis of the fire risks;
- **Step 4**: Treat the risk(s) identified and confirmed in Step 3.

Consideration of Random Failures

Fire detection equipment is electrical equipment. Therefore, random failures must be taken into account for all the detection equipment required for safety reasons.

The pipework of the water circuits and sprinklers are considered as passive equipment items. Therefore, random failure does not need to be taken into account.

The following active equipment are to be considered for a single random failure [10]:

- Containment: fire stop devices (i.e.: fire dampers);
- Detection: main detection equipment (as the detectors and their circuits are electrical equipment);
- Extinguishing: pumps, controlled valves that change position when the systems and sprinklers are activated.

Design Verification Studies

The design verification for protection against internal fires is a deterministic demonstration that the unit has acceptable protection against this hazard. It is performed according to the methodology described below [13]:

First Phase:

Design studies are performed to define the fire compartments building by building. The justification of the fire zoning is provided for all nuclear safety significant buildings.

Second Phase:

This phase consists of checking that the fire compartments are adequately implemented and consistent with the safety objective. It includes modelling the fire compartments on the basis of the EPRESSI method [14].

Third Phase:

The objective of this phase, which is based on the implementation of the vulnerability studies described in the section above, is to either demonstrate the suppression of any common mode or conclude in the acceptability of the risk incurred.

PRODUCTION OF HAZARD PROTECTION SCHEDULES

The development of the HPSs, for the UK EPR design, is based on a methodology [15] developed to provide guidance on the methods used for design basis safety analyses with respect to internal or external hazards. The methodology relies mainly on using a qualitative approach, but employs quantitative supporting information where necessary to demonstrate that the outcome is ALARP in terms of nuclear safety risk.

An important part of the IHs safety analyses, conducted thus far, is presented within the basic design hazard protection schedules (BD-HPSs) for *[IH; Building(s)]* pairs. The main technical objective of these BD-HPSs is to contribute to the overall demonstration of the robustness of the UK EPR basic design before starting the detailed design studies. It is also an opportunity to consider any relevant operational experience from the other EPRs currently under construction throughout the world (Finland, France and China).

It is stated the first section (2nd paragraph) of this paper that "*External and internal hazards that could affect the plant must be identified, and provision made to ensure that the risk from hazards is commensurate with the overall frequency and release targets*". To achieve this objective it must be demonstrated that the three main safety functions (i.e., control of fuel reactivity; fuel heat removal; confinement of radioactive materials) can be fulfilled and the plant brought and maintained at a safe shutdown state following the occurrence of internal hazards.

The need to provide such a demonstration leads, in addition to the hazards design approach, to the production of BD-HPSs that identify the main safety functions potentially affected by the considered IHs. Thus, in order to ensure that the process is comprehensive, logically consistent and relevant to the UK safety assessment context, identification of all reasonably foreseeable hazard initiating events (HIEs) must be conducted. This is followed by the identification of the required hazard safety functions (HSFns) needed to protect against the adverse consequences of considered HIEs.

Such HSFns are categorised (in accordance with the UK EPR categorisation criteria [16] using the guidance in the BD-HPS production methodology [15], and the hazard safety features (HSFs) - SSCs that fulfil the HSFns - are classified with respect to the importance of their contribution to nuclear safety and also in accordance with [16]. This is a similar process to the production of the main fault and protection schedule for the UK EPR design. This approach is compatible with the expectation that all initiating events be given equal attention in a fault schedule, regardless of whether they relate to an internal fault (i.e.: an operational failure resulting in a core transient or direct radiological release) or an internal or external hazard.

The following procedural steps are intended to align directly with production stages and formatting defined in the BD-HPSs production methodology [15]:

- Step 1: Identification and classification of buildings requiring hazard safety analyses;
- Step 2: Identification of HIEs for which hazard analyses are required for the buildings identified in Step 1;
- Step 3: Identification and categorisation of the required HSFns and classification of their associated HSFs.

Two examples extracted from the preliminary "*Internal Fires*" BD-HPS produced for the Nuclear Auxiliary Building (HN) and the Diesel Building (HD) [17] are presented Table 2 and Table 3 below.

Table 2: Preliminary UK EPR BD-HPS for [Internal Fires; Nuclear Auxiliary (HN) Building] pair {extract from **[17]**}

Hazard Initiating Fault Applicability				Without crediting any Hazard Safety Features			Crediting all Hazard Safety Features		Hazard Safety Function		Hazard Safety Feature (HSF)				
HIE	Plant configuration (Normal operating conditions, Shutdown State, Maintenance ...)	HIE freq. (pry)	Building(s)	#	Potential consequences	Dose-band of initiated transients	Potential consequences	Dose-band of initiated transients	Function	Category (Criterion)	HSF type I/S/P/A	#	HSF title	HSF Class or Requirement	HSF code
Fire in any part of ZFS defined by HN building (including localised SFA/SFI)	- Reactor state not directly relevant. - TEG potentially venting hydrogen as effluent gas. - Building treats and contains solid, aqueous and gaseous wastes arising from the RCS primary circuit, fuel pool and elsewhere. - Building operations include electrical systems including pumps, compressors (including vacuum), ventilation, chillers and heaters. - Building envelope is designed to contain entire building inventory with containment dampers closed. - Ventilation dampers assumed to be open for normal operation. - Internal fire protection consists of SFI/SFA for post-event intervention (smoke control) and access. - No claims on fire extinguishing.	10^{-2}	HN only	1	Fire propagation to other NI buildings (HLA, HLB, HLC, HLD, HK, & HQx) through HVAC (DWN, DFL) ductwork, access doors or HGN Technical Gallery (to HQx): - Very limited connections apart from TG - Building access limited - Maximum of ONE HL division affected (RRI unaffected) One of the consequences is loss of HL electrical divisions, triggering a PCC 2.19	DB5	None – fire propagation risk minimised.	DB2	Maintain ZFS integrity	A (A-b)	I	1 2	Geographical segregation provided by walls and structures	C1/SC1	N/A
											S	1 2	Geographical segregation provided by doors, fire rating and wall penetration fillings	Class 1	N/A
									Maintain ZFS integrity (other means)	C (C-f)	A (auto)	1 2	JDT fire detection	Class 3	JDT-SF-01
											A (manual)	1 2	HN building envelope dampers	Class 3	Later
				2	Explosion induced by fire heating of SGH/TEG pipes in HN. Failure of containment envelope of HN	DB4			Prevent explosion	C (C-f)	A	1 2	JDT fire detection	Class 3	JDT-SF-01
											A	2	SKZ preventive isolation on JDT detection in HN	Class 3	Later

Table 3: Preliminary UK EPR BD-HPS for [Internal Fires; Diesel (HD) Building] pair {extract from **[17]**}

Hazard Initiating Fault Applicability				Without crediting any Hazard Safety Features			Crediting all Hazard Safety Features		Hazard Safety Function		Hazard Safety Feature (HSF)				
HIE	Plant configuration (Normal operating conditions, Shutdown State, Maintenance ...)	HIE freq. (pry)	Building(s)	#	Potential consequences	Dose-band of initiated transients	Potential consequences	Dose-band of initiated transients	Function	Category (Criterion)	HSF type I/S/P/A	#	HSF title	HSF Class or Requirement	HSF code
Fire in one HD ZFS Unit 1: 3HDA011ZFS, 3HDA012ZFS & 3HDB021ZFS Unit 2: 3HDD041ZFS, 3HDD042ZFS & 3HDC031ZFS	- Reactor state not directly relevant since fire is assumed to be credible whether diesels are running or not. - Fire could start while other division is in maintenance. - Building operations include diesel engines with auxiliary ventilation, exhaust and cooling systems, and bulk fuel storage. - Each HD is divided into three completely segregated sections with its own diesel, fuel supply and auxiliaries (2 off EDG & 1 off SBO) - Building envelope is designed to contain a fuel fire with dampers closed, including after a design basis seismic event. - Ventilation dampers assumed to be open for normal operation. - Internal and external ZFS-designated fire barriers are seismically-qualified. - Beyond ZFS level, internal fire protection consists of SFI/SFA for post-event intervention (smoke control) and access. - No claims on fire extinguishing.	Fire only: 10^{-3}	HD only	1	Total loss of all HD divisions (3 ZFS) with extensive, irreparable fire damage to that division.	DB0	Total loss of <u>one</u> HD division with extensive, irreparable fire damage to that division.	DB0	Maintain ZFS integrity	A (A-b)	I	1	Geographical segregation provided by walls and structures	C1/SC1	N/A
											S	1	Geographical segregation provided by wall penetration fillings	Class 1	N/A
									Maintain ZFS integrity (other means)	C (C-f)	A (auto)	1	JDT Fire detection	Class 3	JDT-SF-01
											A (manual)	1	HD building envelope dampers	Class 3	Later

CONCLUSIONS

This paper presents a summary of the approach adopted for the deterministic safety analysis of the “internal fires” hazard that is being performed for the UK EPR design. The main objective of the analysis is to demonstrate that sufficient SSCs remain available to permit a safe shutdown state of the plant to be reached and maintained, and limit any radiological releases to below those considered to arise from PCC events (internal faults) occurring at frequencies equivalent to the “internal fires” hazard frequency. To achieve such objective, an overall four steps internal hazards design analysis approach was developed and implemented.

- **Step 1**: IHs identification including the consideration of credible combined or consequential hazards and the setting up of safety requirements.
- **Step 2**: IHs impact quantification (e.g., specific loads and environmental conditions), and design basis protection of structures, systems and components against the impact.
- **Step 3**: Design verification against IHs to confirm that the safety requirements have been fulfilled. This is systematically performed on a case-by-case basis with the use of deterministic studies, such as building and equipment response, functional impact including consideration of consequential internal faults, etc.
- **Step 4**: Production of hazard protection schedules (which consider and complement the design verification analyses.

Regarding the “internal fires” hazard, it is expected that complete implementation of the above steps can provide a demonstration that classified safety systems can achieve their design objectives by minimising the occurrence of “internal fires” and by mitigating their potential consequences. In effect, a set of general safety principles related to “internal fires” protection have been identified and compliance of the UK EPR design with those principles is required. To achieve such compliance, three groups of defence in depth measures against the impacts of “internal fires” are adopted for the UK EPR design:

- **Prevention** (prevent fires from starting, prevention of spreading);
- **Containment** (fire compartments and cells, physical and geographical separation, smoke protection);
- **Control** (detection and extinguishing).

The initial results from the implementation of the above measures during the basic studies phase of the UK EPR design show that:

- “Internal fires” effects are limited so that sufficient structures, systems and components remain available to permit a safe shutdown state of the plant to be reached and maintained, and any potential radiological releases are kept below acceptable limits.
- Non-redundant systems and equipment, which perform the required nuclear safety functions, are protected against the effects of “internal fires” to ensure continuous and safe operation of the plant.
- In the event that the main control room becomes inhabitable due to “internal fires”, the accessibility and habitability of the remote shutdown station can be assured.
- Adequately categorised hazard safety functions and associated classified hazard safety features are identified and implemented in the UK EPR design to reduce the risks and effects of “internal fires” to as low as reasonably practicable.

These initial results confirm that the UK EPR fire protection design objective is adequately met, and therefore the risks to workers and general public are demonstrated to be as low as reasonably practicable.

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OVERVIEW OF INTERNAL FIRE HAZARDS ASPECTS OF ABWR DESIGN FOR UNITED KINGDOM

Kazuhiro Yoshikawa, Hiroki Kawai

Hitachi-GE Nuclear Energy, Ltd.,
3-1-1, Saiwai-cho Hitachi-shi, Ibaraki 317-0073, Japan

ABSTRACT

The ABWR (*A*dvanced *B*oiling *W*ater *R*eactor) is a generation III+ reactor, the most modern operational generation of nuclear power plants. The UK ABWR design is proposed for development and construction in the United Kingdom (UK), and under review by the Office for Nuclear Regulation (ONR) through Generic Design Assessment (GDA). The UK ABWR design has mainly two types of the safety system: "preventing" and "mitigating" a fault and their consequences. The prevention of internal hazards starts with design processes and procedures. These processes lead to limiting the sources of potential hazards. The mitigative safety systems are required to ensure the fundamental safety functions (FSFs): control of reactivity, Fuel cooling, long term heat removal, confinement/containment of radioactive materials, and others. Implementation of the safety philosophy is based upon redundant and diverse safety systems that deliver the FSFs. Three mechanical divisions are provided, each of which contains redundant systems, structures, and components (SSCs) capable of carrying out all the FSFs. The safety divisions are separated by robust barriers which act to contain a hazard in an affected division and prevent the spread of the hazard to a different division. The deterministic assessments and the hazard schedule argue that the rooms containing SSCs providing the FSFs are located in different fire safety divisions. The approach to maintaining the FSFs during and after internal fires is to ensure fires do not spread beyond that division to affect redundant equipment in other divisions. During the GDA process, it is demonstrated that generally barrier compartmentation (the divisional barrier walls, ceilings and floors) is sufficient to contain the postulated fires. The UK ABWR design has sufficient capability of withstanding the postulated internal fire hazard to achieve the FSFs. Further development is being undertaken with feedback in the GDA licensing review process with ONR.

INTRODUCTION

This paper discusses the internal fire hazards aspects of the UK ABWR design, which is under review in the GDA assessment process. The GDA assessment process for the UK ABWR is now in Step 3, and the Design Acceptance Confirmation (DAC) is targeted for 2017. In this paper, an introduction of the UK ABWR design and a summary of the safety design philosophy regarding internal fire hazards is described.

OVERVIEW OF THEE UK ABWR PROJECT

ABWR Design

The UK ABWR is a nuclear reactor design proposed for development and construction in the UK. The UK ABWR is being developed and offered in the UK by Hitachi-GE Nuclear Energy, Ltd. (Hitachi-GE). The design baseline of the UK ABWR is the ABWR design, which is a generation III+ reactor, the most modern operational generation of nuclear power plants. The ABWR is operational at three sites in Japan: two units at Kashiwazaki-Kariwa; one unit at Hamaoka and one unit at Shika. There are a further four units under construction, two units at Lungmen in Taiwan, and one unit at both Shimane and Ohma in Japan. At full power, a single ABWR unit produces around 1350 MW_e of electricity – enough to power more than two million homes.

The ABWR design has been developed since 1985, in collaboration with various international partners and support from power companies with experience in BWR plant operation. The main technological features employed are as follows:

- Large scale, highly efficient plant,
- Highly economical reactor core,
- Reactor coolant recirculation system driven by internal pumps,
- Advanced control rod drive mechanism,
- Overall digital control and instrumentation (I&C),
- Reinforced concrete containment vessel.

These features constitute a highly-functional, enhanced-safety nuclear reactor system, with a compact, easy-to-operate, and efficient turbine of excellent performance. For information on the ABWR, please see the website (http://www.hitachi-hgne-uk-abwr.co.uk/index.html).

The key specifications of typical ABWR design are shown below.

Plant output:	1,350 MW_e class
Reactor thermal output:	3,926 MW_t
Reactor rated pressure:	7.07 MPa
Reactor core fuel assemblies:	872
Control rods:	205 rods
Recirculation system:	internal pump system
Control rod drive:	hydraulic / electric motor drive method
Primary containment vessel (PCV):	reinforced concrete with built-in liner
ECCS*/PCV cooling system:	3 divisions
Residual heat removal system:	3 divisions

* ECCS: Emergency Core Cooling System

Figure 1 Conceptual view of ABWR

Generic Design Assessment (GDA)

Nuclear energy is highly regulated to ensure that the highest levels of safety, security and environmental protection are maintained at all times. For new nuclear builds in the UK, the first stage of this is a rigorous assessment of the reactor design by UK regulators, known as GDA.

The GDA is the process by which the UK nuclear regulators assess the potential suitability of a nuclear reactor design for development at an unspecified location in the UK, considering safety and environmental impact considerations. It is not an assessment of the principles of nuclear energy, but of the design of the UK ABWR plant itself. Passing the GDA is an important step in the process towards developing an operational station, but does not in itself give any 'permission' to develop. A Nuclear Site License is still required, as are environmental permits. Although the GDA may inform elements of these assessments, it does not replace them. Power station developers must also go through the full planning process in order to gain a Development Consent Order.

The GDA is broken down into a number of steps, each of which entails additional submissions to the regulators. A significant number of these submissions will be published via the website (http://www.hitachi-hgne-uk-abwr.co.uk/index.html) and Hitachi-GE invites comments or questions on them. A detail description of design philosophy against internal hazards including fire of UK ABWR is provided as Pre-Construction Safety Report [1] on the website.

UK ABWR SAFETY DESIGN PHILOSOPHY

Safety Functions

The design of the UK ABWR is primarily based on providing segregated, redundant and diverse safety systems to protect against faults. The safety systems for the UK ABWR can be divided into two main groups: the systems that prevent faults and abnormal conditions in the facilities, and the systems that mitigate the consequences of a fault and abnormal events.

The safety systems of the UK ABWR design are required to ensure the following fundamental safety functions are still delivered when a fault occurs:

- FSF1: Control of reactivity, particularly the ability to achieve emergency reactor shut-down,
- FSF2: Fuel cooling, i.e. all systems whose function is to prevent damage to fuel from overheating,
- FSF3: Long term heat removal, including removal of decay heat and containment venting systems,
- FSF4: Confinement/Containment of radioactive materials, and
- FSF5: Others, including remote shutdown capabilities, instrumentation and monitoring, alternative power supplies, emergency measures.

In order to achieve these fundamental safety functions throughout most of the UK-ABWR design, the implementation of this safety philosophy is based upon redundant and diverse safety systems that deliver the fundamental safety functions. Three mechanical divisions are provided, each of which contains redundant systems, structures and components (SSCs) capable of carrying out all the fundamental safety functions. The safety divisions are separated by robust barriers (designed to Nuclear Class 1) which act to contain a hazard in an affected division and prevent the spread of the hazard to a different division. The internal hazards safety case includes measures and systems designed to prevent internal hazards occurring. These preventative systems and procedures provide “defense-in-depth” against the loss of equipment that delivers a fundamental safety function.

Safety Systems

Table 1 shows redundancy and diversity of safety systems that are available to maintain the fundamental safety functions. Two diverse systems, the hydraulic system for control rods and the standby liquid control system (SLC) deliver reactivity control and reactor shutdown function. For protecting the function against consequences of internal hazards, the UK ABWR is designed to provide physical separation between those two diverse systems so as to defend at least one system from the hazard.

Fuel in the core and the spent fuel storage pool (SFP) needs to be cooled. For core cooling, there are three redundant and segregated divisions. Each of the divisions provides diversity through high pressure core injection, reactor pressure vessel (RPV) depressurization and low pressure core injection. The UK ABWR design ensures delivery of the core cooling function by installing robust barriers between the safety divisions so as to prevent hazards spreading beyond a single division.

The spent fuel cooling function is primarily delivered by the fuel pool cooling and clean-up system (FPC), which is backed up by the two redundant residual heat removal (RHR) pumps in SFP cooling mode. As noted above each of these trains is segregated by divisional barriers.

Table 1 Summary of Safety Systems of the UK ABWR design

<table>
<tr><th colspan="2">Function</th><th>System</th></tr>
<tr><td colspan="2" rowspan="2">Reactivity control and reactor shutdown</td><td>Hydraulic System for Control Rods</td></tr>
<tr><td>Standby liquid control system (SLC)</td></tr>
<tr><td rowspan="2">Cooling</td><td>Core cooling</td><td>Division I:
RCIC+(ADS+LPFL(A))
Division II:
HPCF(B)+ (ADS+LPFL(B))
Division III:
HPCF(C)+ (ADS+LPFL(C))</td></tr>
<tr><td>Spent fuel cooling</td><td>Spent fuel pool Cooling and Makeup</td></tr>
<tr><td colspan="2" rowspan="2">Containment</td><td>Primary containment vessel (PCV)</td></tr>
<tr><td>Secondary containment facility
(Reactor Building, RB)</td></tr>
<tr><td colspan="3">RCIC: reactor core isolation cooling system
HPCF: high pressure core flooder system
LPFL: low pressure flooder system
ADS: automatic depressurization system</td></tr>
</table>

The containment cooling function is delivered by the RHR system in suppression pool cooling mode as the primary means of heat removal to keep integrity of the Primary Containment Vessel (PCV) in case of an accident. A secondary means of containment cooling is provided by the PCV Spray Cooling mode of the RHR system. Containment isolation for each of the penetrations on the PCV ensures the integrity of the PCV. Additional containment function is delivered by secondary containment facility, which is supported by redundant divisions of the standby gas treatment system (SGTS). SGTS keeps the atmosphere inside the RB to be negative pressure for preventing release of radiological substances to the public from the RB.

The safety systems summarized in Table 1 are supplemented by an independent set of nuclear safety mitigation equipment that can be used for core cooling, decay heat removal and maintaining primary containment. These are located in a seismically qualified Backup Building (BB), which will be sufficiently apart from the RB as severe accident management facility. In case of beyond design based accidents, they will perform providing supplemental safety function.

Prevention and Limitation of Severity of Internal Hazards

The prevention of internal hazards starts with the design processes and procedures. These processes lead to limiting the sources of potential hazards. Codes and standards, as well as design guides are used to appropriately design equipment and structures, and develop a layout to prevent the occurrence of an internal hazard, and prevent the impacts of internal hazards whenever possible. If all the prevention measures were to fail, the general approach to ensure protection of SSCs which deliver the FSFs is to limit the impacts of internal hazards to within the redundant division of the safety system using robust barrier compartmentation.

The specific internal hazard assessments identify and assess the prevention measures included in the design. Based on the importance of the prevention measures to maintaining the Fundamental Safety Functions and reducing risk ALARP, classification of these measures are determined. As stated above, significant importance and higher classification is placed on mitigative measures and most of the preventative measures are classified as defense in depth. Some internal hazards have measures to limit severity of the hazards to reduce consequences or risk of the hazards (e.g. fire fighting).

Mitigation of Internal Hazards

If all the prevention measures were to fail, the general approach to ensuring protection of SSCs which deliver the fundamental safety functions is to limit the impacts of an internal hazard to within the redundant division of the safety system using robust barrier compartmentation. In some areas, it is necessary to consider one internal hazard causing a second internal hazard due to the proximity of these hazards. In these combined hazard cases, other prevention and mitigation measures may be necessary. Internal hazards claims are achieved by the following design features:

1. SSCs of the fundamental safety functions that require redundancy are located in separate safety division areas fully enclosed with appropriately designed barriers (floor, ceiling, wall, etc.). Divisional barriers have been designed to withstand the potential fire impact and overpressure consequences of the various internal hazards that they may be exposed to.
2. There are some areas where the functional requirements do not allow the above segregation provisions to be made. These are the primary containment vessel (PCV), the main control room (MCR), the main steam tunnel (MST) and, in some cases, redundant equipment is somewhere located in another division for reliability purposes following a hazard. These exceptions to segregation are justified on a case by case basis.

Penetrations

Penetrations of the barriers between redundant divisions are minimized. Any devices for closing penetrations such as doors, ductwork, hatches, and pipework and cable entry seals, which form part of a barrier, are appropriately designed with the same internal hazards, including fire, resistance ratings as the overall barrier.

If a barrier cannot ensure segregation of redundant SSCs, then other measures are considered, including the following for internal fire:

- Administrative controls on the amount of materials that may cause an internal fire;
- Separation by individual room walls or equipment barriers which are not claimed as internal fire divisional barriers;
- Separation of equipment by distance, without intervening materials that may cause an internal fire;
- Local passive protection that may prevent internal fire affecting the equipment required to function.

Where door access through a claimed Nuclear Safety Class barrier is required, two sets of doors separated by a lobby will be provided wherever reasonably practicable. Any remaining single doors will be fitted with an alarm system to alert operational staff where doors are left open or fail to close.

Divisional Separation Exceptions

Significant importance is placed on systems that mitigate the effects of internal hazards including fire. There are three redundant and independent divisions of equipment to maintain the fundamental safety functions. Generally for internal fire, this is provided by passive barrier compartmentation designed such that internal fire and their effects are limited to one safety division. The equipment within the different systems is grouped together within each safety division and the safety divisions are segregated from one another by robust barriers. These barriers provide the principal means of maintaining the fundamental safety functions against the effect of internal fire.

However, there are exceptions where multiple divisions of equipment or instruments are located in the same area or room. There are rooms throughout the nuclear island that contain multiple divisions because there may be specific design requirements to provide diversity; these areas are referred to as exceptions to segregation. Such exceptions to divisional segregation have been considered. These areas will be assessed to ensure that internal fire in those areas do not compromise the fundamental safety functions. In most cases of exceptions to segregation, the design of the reactor protection systems provide robustness and enhance the reliability of safety system actuation to fulfill the fundamental safety functions. The areas with exceptions to segregation include the PCV, the MST and the MCR.

Note that nuclear safety relates not only to the safe operation or shutdown of the reactor but also to the containment of contamination and radiation such that the radiological risk to workers and the public is as low as reasonably practicable (ALARP).

FIRE SAFETY DESIGN APPROACH

The following sections discuss the fire safety design approach of the UK ABWR. The key piece of legislation under the UK law is the Health and Safety at Work, etc. Act 1974 (HSWA) [2], which requires consideration of both the direct effects of fire on people (‘conventional fire safety') and the effects of ionizing radiation on people resulting from fire events ('nuclear fire safety'). For UK ABWR design, the following regulatory regime is to be considered:

Nuclear Fire Safety

1. The 'Nuclear Installations Act 1965' (http://www.legislation.gov.uk/ukpga/1965/57, as amended) is the primary regulation, under UK law, for the protection of people (both employees and the public) from harmful effects of radiation.
2. The 'Ionizing Radiation Regulations 1999' (IRRs, http://www.legislation.gov.uk/uksi/1999/3232/contents/made) are applied to nuclear licensed sites (and others containing radiological material) to regulate radiation protection. The IRRs have limited impact on the fire hazard, although some of the Basic Safety Levels (BSLs) used in deterministic analysis of hazards are legal limits in the IRRs.
3. The 'Radiation (Emergency Preparedness and Public Information) Regulations 2001' (REPPIR, http://www.hse.gov.uk/radiation/ionising/reppir.htm) require a framework for the protection of the public through emergency preparedness for radiation accidents with the potential to affect members of the public and provision of information to the public.
4. Guidance on the requirements for UK nuclear safety cases is set out in the ONR Safety Assessment Principles and Technical Assessment Guides (TAGs).
5. The Safety Assessment Principles (SAPs), which were developed for ONR use but have been made publically available, provide ONR inspectors with a framework for making consistent regulatory judgments on nuclear safety cases. Practically, the SAPs also provide guidance for design decisions of applications for and holders of Nuclear Site License.
6. The Technical Assessment Guides (TAGs) provide additional guidance to inspectors, with the relevant TAG being internal hazards, NS-TAST-GD-014, updated April [3]. This TAG refers directly to the more detailed International Atomic Energy Agency (IAEA) guidance within NS-G-1.7 [4] and NS-G-2.1 [5] within the UK context.
7. The SAPs reflect, and have been benchmarked against, the IAEA guidance. However, due to the UK's goal-setting legal framework for health and safety, the IAEA requirements are not implemented in a prescriptive manner.
8. The TAG and IAEA guidance documents have been used by Hitachi-GE to assess the UK ABWR design to ensure it meets the expectation of the UK regulators.

Conventional Fire Safety

1. The 'Regulatory Reform (Fire Safety) Order' (RRO), cf. http://www.legislation.gov.uk/uksi/2005/1541/contents/made, is the major piece of the UK fire legislation and covers general fire precautions and other fire safety duties. The RRO requires fire precautions to be put in place 'where necessary' and to the extent that it is reasonable and practicable and covers the design and occupation of buildings.
2. 'The Building Regulations 2010', http://www.legislation.gov.uk/uksi/2010/2214/contents/made embodies the description given in the 'Building Act 1984', http://www.legislation.gov.uk/ukpga/1984/55, and generally apply to the design and construction of buildings, rather than their occupation. However an exemption in the Building Regulations is included for "*Any building (other than a building containing a dwelling or a building used for office or canteen accommodation) erected on a site in respect of which a license under the Nuclear Installations Act 1965 is for the time being in force.*" Therefore the Building Regulations will not apply to the GDA for the UK ABWR.

3. The ‘Dangerous Substances and Explosive Atmospheres Regulations 2002’ (DSEAR, cf. http://www.legislation.gov.uk/uksi/2002/2776/contents/made) regulate dangerous substances used or present at work that could, if not properly controlled, cause harm to people as a result of a fire or explosion. Hitachi-GE is assessing the UK ABWR design for compliance with these regulations. An example is control of hydrogen both generated inside the reactor and used for generator cooling.
4. ‘BS 9999:2008: Code of practice for fire safety design, management and construction of buildings’ http://shop.bsigroup.com/en/ProductDetail/?pid=000000000030158436) is an accepted way of meeting the requirements of the RRO, and is therefore used as the basis for design in UK ABWR. As a code of practice, it contains elements of good fire safety management and design practice covering life safety and to enhance property protection and business continuity.
5. As with the majority of British Standards the contents are recommendations and not requirements. As such the UK ABWR design will follow the recommendations in BS 9999 in so far as in reasonably practicable and where departures from the recommendations are identified, compliance with the regulatory functional requirements (contained within the RRO) will be demonstrated through a structured approach. Some changes to the standard ABWR design have been applied for compliance with BS 9999, e.g. addition of fire fighting lobbies.
6. ‘BS 7974-1:2001: Application of fire safety engineering principles to the design of buildings - Code of practice’, cf. http://shop.bsigroup.com/ProductDetail/?pid=000000000030028692 along with eight published documents, provides guidance on the use of fire engineering techniques to provide a structured approach to meeting the functional requirements of the RRO. These documents provide guidance to Hitachi-GE for assessing fire hazards and determining the robustness of barriers against a fire hazard.

FIRE SAFETY DESIGN BASIS

Internal fire hazards can occur within the site boundary and inside the buildings of the UK ABWR. Sources of combustible inventory include fuel oil, lubrication oil, cables and transient combustibles. Fires may also arise from other internal and external hazards. The internal fire claim of the UK ABWR design is defined as follows:

“*The consequences of any internal fire hazard are limited to one safety division in principle.*”

There is suitable and sufficient fire compartmentation in place to ensure that the consequences of any fire event will be contained within the safety division of origin. Thus, a fire will not affect equipment within more than one division in principle. The above internal fire claims will be achieved in the UK ABWR design by:

a. Limitation of the fire loading,
b. Limiting the potential ignition sources so as to prevent fires from starting,
c. Limiting the severity of fires that do start,
d. Mitigating the consequences of severe fires (principally through fire compartmentation).

The number of fire sources is limited as far as possible by minimizing combustible quantities and storing potentially combustible materials in areas with limited or protected ignition sources. The sections below describe the design aspects that prevent fires and mitigate the consequences.

Fire Prevention

The UK ABWR design includes a number of features that prevent fires as follows:

1. Non-combustible materials or fire retardant materials are used for interior materials where possible.
2. Pipework containing flammable liquids is welded and sealed, and double walled where necessary.
3. Equipment containing flammable liquids is pressure tested.
4. Any leakage is detected by liquid level monitoring and captured.
5. The area around equipment containing large quantities of flammable liquids includes bunds that prevent spread of the spill. These bunds are designed to contain additional liquid from any fire fighting water/foam.
6. Low combustibility fluids are used if practicable.
7. Electrical systems use breakers and fuses to prevent overloading.
8. Dry-type transformers are used where the transformer is located within a building.
9. Electrical equipment in the areas around systems containing flammable liquids and gases, and the hydrogen supply systems have been appropriately rated per the Hazardous Area Classification guidelines.
10. Cables specified for the UK-ABWR meet standards for flame spread and smoke generation under fire conditions.
11. Cables are installed in steel cable trays, conduits or other non-combustible cable supports and are appropriately spaced.
12. Insulating lagging provided for pipework with hot surfaces.
13. Pipework for systems containing hydrogen is designed to prevent leakage and failure.
14. Pipework systems that may contain hydrogen due to the radiolysis of water has been designed in accordance with ASME Section III and/or B31.1 standards, in addition to meeting the Japanese design guide JANTI-NCG-01 [6].

Fire Protection

Although the safety case for fire hazards primarily relies solely on fire compartmentation, it is good practice to provide fire detection and alarm systems and some means of fighting fires. Operators will be trained and equipped to manually fight fires where it is safe to do so. The UK ABWR design provides the following fire protection systems for fire protection means.

Fire Detection and Alarm Systems

Fire detection and alarm systems serve to detect a fire and provide warning to occupants in the vicinity of a fire and to the MCR. The UK ABWR design will include detection and notification of a fire in areas containing SSCs of Fundamental Safety Functions, allowing operators to take actions to mitigate the effects of the fire.

Fire Fighting Systems

Fixed fire suppression systems are provided in the UK ABWR design for defense in depth as part of the nuclear safety case. Such systems are considered good practice to prevent rapid fire growth or to limit fire spread. The fixed fire suppression systems may operate either automatically or manually. Automatic or manual operation is determined by the fire hazards and/or operational requirements of the systems.

Fire Fighter Intervention

In addition to designing buildings with fixed fire suppression, consideration is given to allow fire fighters to safely enter and attack a fire where necessary. The UK ABWR design is engineered to provide the following features to allow fire fighter intervention:

a. Provision of vehicular access for fire appliances to the perimeter of the building or site,
b. Provision of quick and easy entry to the interior of the building for fire brigade members and their equipment,
c. Provision of access to sufficient supplies of fire fighting materials,
d. Means for enabling fire fighters access to all areas of a building, including provision of fire fighting lifts if appropriate,
e. Means for ensuring protected areas for fire fighters to carry out their operations,
f. Provisions for fire and rescue service communications,
g. Provision of facilities to release, or extract smoke and/or heat,
h. Provision for removing fire fighting extinguishing materials.

Mitigation of Fires

The fire hazard analysis (FHA) conservatively assumes that a fire results in the loss of all equipment within the safety division of origin. The approach to maintaining the fundamental safety functions from internal fires is to then ensure that fires do not spread beyond the division of origin to affect redundant equipment in other divisions. This is done by providing suitable barrier compartmentation between the different safety divisions. For fires, this is known as the "*Fire Containment Approach*" to fire safety and is recommended in IAEA Guide NSG- 1.7 [4]. This approach relies on passive fire compartmentation and is therefore highly reliable.

Penetrations (Door Monitoring System)

The fire doors are designed to be closed during all operational conditions. The door monitoring system monitors the fire door position and alerts operators when the doors are left open. The door monitoring system has been classified based on the importance of the system to maintaining the fundamental safety functions.

Where reasonably practicable, double fire doors with an intervening lobby are provided in the divisional barriers. This provides additional defense-in-depth against the fire hazard.

GDA DISCUSSION FOR UK ABWR DESIGN

Location of EDGs

Various discussions are being undertaken in the UK ABWR GDA internal hazards assessment. The emergency diesel generators (EDGs) location is one typical example. The conventional ABWR design in Japan and the U.S. houses EDGs in the RB with sufficient fire protection measures to limit the overall risk. However, the ONR suggested that the UK ABWR design that included EDGs inside the RB would not be acceptable when considering relevant good practices in the UK and full compliance with the SAP, which require the minimization of the potential fire hazards source. Based on ONR's clear expectation that the EDGs should be located outside of the UK ABWR RB, Hitachi-GE has undertaken preliminary engineering studies for a range of options for re-locating the EDGs outside the RB and presented the relative safety and security advantages and disadvantages to the ONR. Based on this study, Hitachi-GE identified the option which it considered best meeting the UK expectations.

The new UK design concept specifies three separate additional buildings for the UK ABWR to house the three safety class 1 EDGs. These three EDGs provide emergency power to the three respective mechanical safety divisions, and support the equipment that fulfills the fundamental safety functions of reactivity control, core cooling and containment. Housing the EDGs outside the RB eliminates the potential impact of internal hazards from the EDGs on other safety equipment within the RB.

The three new EDG buildings have three safety function,s:

a. To house and support the safety class 1 EDGs
b. To maintain environmental conditions for operation of the EDGs, and
c. To protect the EDGs against internal and external hazards.

Because of the importance of the EDG functions, the three new Emergency Diesel Generator Buildings (EDGBs) will be seismically qualified structures, to the same specification as the RB and other safety class 1 structures. The separation between buildings will also ensure all three will not be impacted by a single aircraft impact.

In order to supply power from the EDGs to the three emergency power buses, three additional trenches are part of the new design concept. These trenches will route electrical cables and other associated piping from the EDGBs to the RB. As with the EDGBs themselves, these trenches will be seismically qualified and separated from each other, and protected from impacts of other external hazards. There will be no access from one trench to another and no penetrations between the trenches. In addition, sufficiently qualified barriers between the EDGBs and the trench and between the RB and the trench will prevent the spread of hazards, such as fires, from the EDGBs to the RB. These barriers will necessarily have penetrations for cables and pipework, and these penetrations will be appropriately designed to maintain the required barrier function. The trenches will be routed around the RB between the EDGBs and the RB such that the power cables and other piping will not cross divisional boundaries within the RB.

The EDGBs will be designed to meet the guidance of BS 9999 in terms of means of escape and fire fighter access. Fire detection will be installed throughout the building and fire suppression with be installed in the basement, EDG rooms, and day tank rooms. The basement, EDG rooms and day tank rooms will also include bounding to prevent spillage of fuel outside those respective rooms.

The EDGs require a number of support systems in order to fulfill their safety functions such as air starting system, HVAC, cooling water, and lubrication, in addition to the fuel supply system. As with other versions of the ABWR design, these support systems will be divisionalized and only support a single EDG. The main components of these support systems will be housed within the EDGBs, and will be similarly protected from internal and external hazards. The HVAC systems will provide air inlet and exhaust during normal standby and EDG operation modes. Air inlet and exhaust will be local to each EDGB.

Fuel will be supplied to the EDGs during operation directly from the day tanks, which are located above the EDGs so fuel will flow by gravity. Upon indication of low levels of fuel in the day tank, a fuel transfer system will provide additional fuel to the day tanks from the bulk storage tanks in the basement of the EDGBs. Each EDGB will house a 7-day supply of fuel in bulk storage tanks to ensure all EDGs can maintain operation for that period of time. The fuel transfer system will be safety class 1 and powered from the operating EDG.

The EDGBs will include two firefighting shafts with lobbies and access directly outside, and provide sufficient escape routes within. There is an additional large external access door that allows for EDG replacement. There are additional external building penetrations for HVAC air inlet (side of the EDGB), HVAC exhaust (roof), and EDG exhaust (roof).

CONCLUSIONS

The UK ABWR design, which is proposed for development and construction in the UK, is based on the ABWR design that is the most modern operational generation of nuclear power plants. The UK ABWR design is under review by ONR through the GDA process.

Local and general fire hazard assessments are being undertaken in the development of claims and arguments for rooms within the buildings including the buildings including the RB together with an assessment of credible ignition sources and the safety equipment at risk from fire. The deterministic approach used for the fire hazard analysis is conservative and the key conservatisms relating to the technique have been detailed. The design philosophy of the UK ABWR has to reduce or remove fire hazards where possible and this technique has been largely successful. The main source of combustible material across the buildings is cabling which is prevalent in major electrical and penetration rooms.

The overriding fire safety claim is that there is suitable and sufficient fire compartmentation in place to ensure that the consequences of any fire event will be contained within the safety division of origin. Where rooms carry a significant combustible density (and potential threat to divisional barriers) the conservatisms, mitigation measures and defense-in-depth arguments used to substantiate this overriding fire safety claim will be discussed in detail during the remaining GDA assessment process.

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PASSIVE FIRE PROTECTION ROLE AND EVOLUTIONS

Tristan Cerosky[1] and Johan Perdrix[2]

[1] Export Director, NUVIA, France
[2] Export Project Manager, NUVIA Protection, France

ABSTRACT

Major incidents associated with nuclear power plants often invoke a re-examination of key safety barriers. Fire hazard, in particular, is a key concern for safe operation of nuclear power plants given its propensity to damage safety systems which could ultimately lead to radioactive release into the atmosphere. In the recent past, events such as the Fukushima disaster have led to an industry-wide push to improve nuclear safety arrangements. As part of these measures, upgrading of fire safety systems has received significant attention. In addition to the inherent intricacies associated with such a complex undertaking, factors such as frequent changes in the national and European fire regulations also require due attention while formulating a fire protection strategy.

This paper will highlight some salient aspects underpinning an effective fire protection strategy. This will involve:

A) A comprehensive introduction to the different aspects of fire safety (namely prevention, containment and mitigation) supported by a review of the development of the RCC-I from 1993 to 1997 editions and the ETC-F (AFCEN codes used by EDF in France).

B) Development of the fire risk analysis methodology and the different functions of passive fire protection within this method involving confinement and protection of safety-related equipment.

C) A review of the benefits of an effective passive fire protection strategy, alongside other arrangements (such as active fire protection) to a nuclear operator in term of safety and cost savings.

It is expected that the paper will provide nuclear operators useful guidelines for strengthening existing fire protection systems.

INTRODUCTION

Fire protection is an essential asset of the safety design of nuclear reactors. There are three main objectives for any fire protection system:

- Protection of personnel;
- Protection of reactor safety systems;
- Maintaining plant availability.

Various codes and standards are used around the world to design systems which meet the protection requirements. This paper examines the development and application of the standards for the use of passive fire protection systems in the new generation of nuclear power plants with particular reference to the French nuclear fleet and the EPR.

FIRE PROTECTION STANDARDS

The French nuclear fleet (see Figure 1) is composed of an evolving set of pressurised water reactors (PWRs).

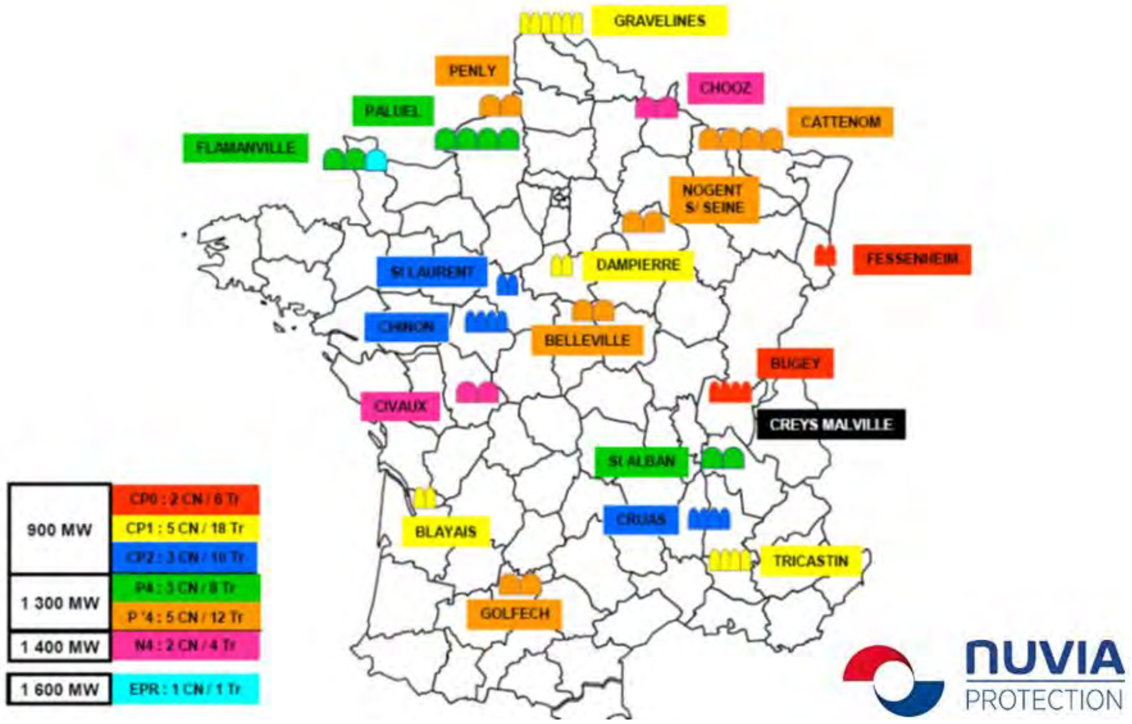


Figure 1 French Nuclear Fleet

The first reactors were subject to ENSIN/89031D in 1977 and then revised regularly up to Rev.97 for their fire protection requirements. By the N4 generation this had been superseded by RCC-I standards developed by Electricité de France (EDF). RCC-I [1] defines the global approach to fire protection in terms of overall systems and levels of fire control measures. Nevertheless, this regulation is now being superseded by ETC-F for the latest generation of European reactors, in particular the EPR [2].

Table 1 Development of French fire regulations with reactor type

Reactor Type	Fire Standard
CP0 Fessenheim and Bugey, PCY, P4, P’4	Fire Rules ENSIN/89031D (Rev.97)
CP0 Bugey	Fire Rules ENSIN/89031D (Rev.97)
N4	RCC-I (Rev.97)
EPR	ETC-F (Rev 2006) Section G

RCC-I DESIGN, CONSTRUCTION AND INSTALLATION RULES FOR FIRE PROTECTION

RCC-I [1] describes the fundamental approach to fire protection.

Physical Protection Systems

The physical protection systems are classified into three levels:

- Level 1: fire prevention – design of plants and operations to minimise the risk of fire by:
  - minimising the use of combustible materials, for example, by implementation of fire safety analysis,
  - using as much as possible equipment such as fire doors and fire retardant cables.
- Level 2: fire containment – use of physical separation, fire retardant and passive fire protection to minimise the impact of a fire on the plant. In particular, safety critical systems must be designed so that redundancy components cannot be affected simultaneously by the outbreak of a fire. This involves fire segregation of building layouts and use of fire barriers. The heat release in each defined area must be less than 400 MJ/m^2.
- Level 3: fire control – gain control over the fire as soon as possible. This will involve the use of fire detection, extinguishing and smoke control systems.

Vulnerability Analysis

Vulnerability analysis is used to assess the possibility of common mode failures (CMF).
For example by defining the redundancy items involved in safety related systems, the reliability of support systems, the probability of electrical failures and the use of mitigation systems.

Operator Procedures

Procedures need to be in place to define the actions required in case of outbreak or suspected outbreak of a fire in the plant. This includes the procedures for the periodic inspection and testing of all the fire prevention systems.

ETC-F EUROPEAN TECHNICAL CODE – FIRE PROTECTION

The Evolutionary Pressurised Water Reactor (EPR) project was started in 1989 through a joint venture between Framatome and Siemens (Nuclear Power International). The French and German utilities also participated in the evolution of the reactor as well as the involvement of the safety authorities (ASN in France and BMU in Germany). One objective was to harmonize and further develop the outstanding safety standards in France and Germany. An organization was set up to develop common codes from the French design and construction rules (RCC) and the German KTA safety standards and DIN standards related to the EPR design. This led to the establishment of the EPR Technical Codes (ETC). As part of this development it was agreed to establish a new fire protection code, the ETC-F [2].

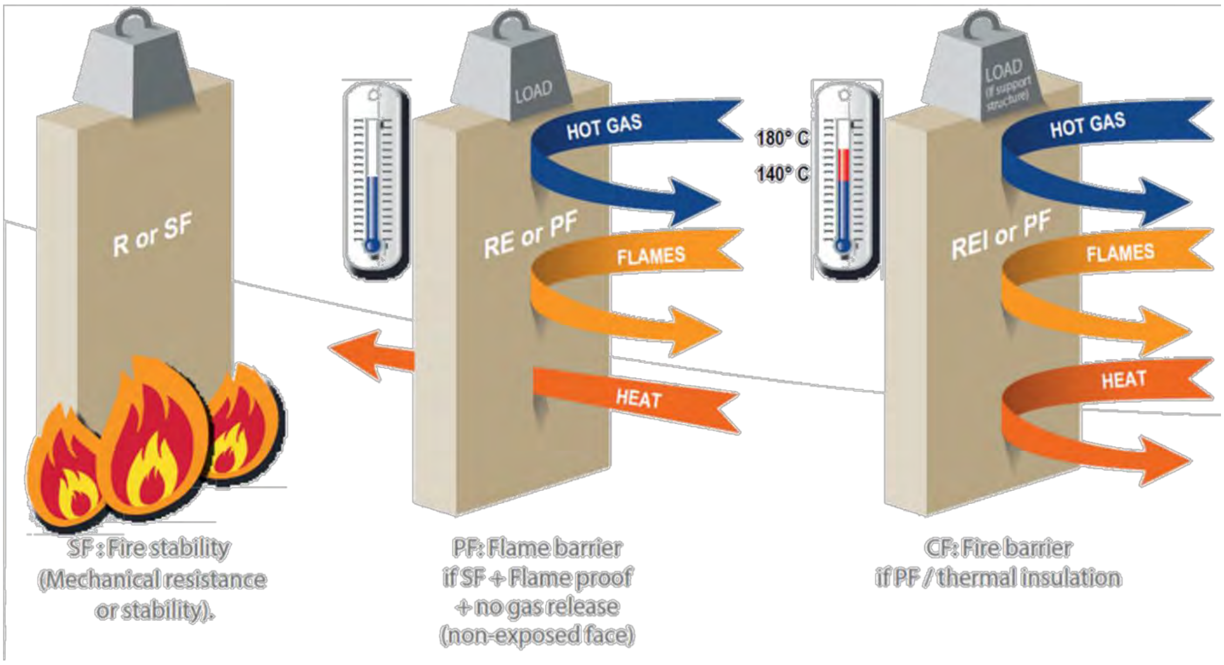
ETC-F follows the principles of RCC-I for the definitions of defence levels and analysis.

THE USE OF PASSIVE FIRE PROTECTION TO MEET ETC-F REQUIREMENTS

We will now focus on the evolution of passive fire protection induced by the new rules set by ETC-F.

Fire Requirements

Fire specific requirements have changed in two directions with implementation of ETC-F.
The first change was the application of the decree “Arrêté feu du 22 Mars 2004” [3] related to fire resistance. Fire resistance is the capacity of building materials to withstand fire for a certain period of time. It refers to European standards.



Criteria (function ensured)	Former classification (1999 order)		New classification (2004 order)
Mechanical resistance	Fire stability	SF	R
Flame and hot gas leaktightness	Flame shield	PF	(R) + E —> E
Thermal insulation	Fire barrier	CF	(R) + E + I —> EI

Figure 2 French regulation versus European regulation

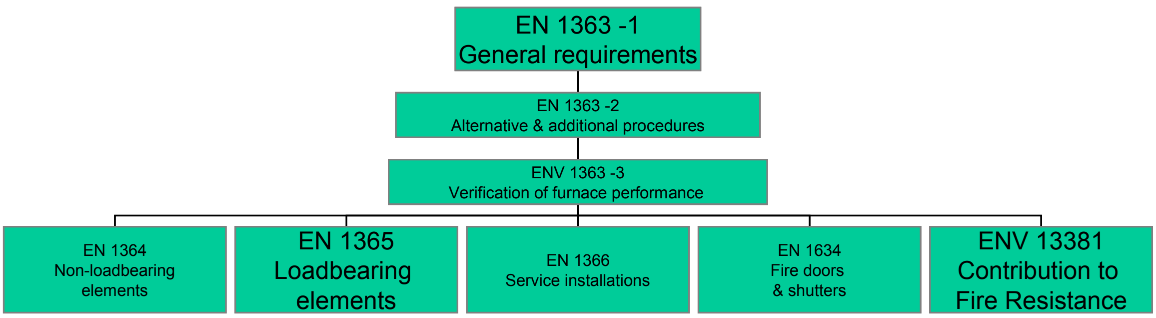


Figure 3 European Fire Standards based on EN 1363-1 [4]

The second change is the time requirement, which was mainly 1.5 hours fire protection, and now becomes mainly 2 hours fire protection.

Typical requirements are therefore:

- P120: functional protection of electrical cables and electro-mechanical equipment for 120 minutes (2 hours),
- (R)EI120 or (R) EI60: mechanical resistance, tightness to hot gases and fumes, and insulation for 120 or 60 minutes.

Earthquake Resistance

ETC-F [2] stipulates (4.3.2.1) that fire protection equipment is classified according to earthquake class 1 (SC1). The requirement is given in Appendix B. Protection of raceways and HVAC ducts must maintain their integrity, which means their dropping must not compromise the operation of F1 classified equipment.

Qualification test of products is made at SOPEMEA, the official French laboratory.

Figure 4 Earthquake tests

Severe Accidents

ETC-F [2] stipulates (4.3.3) that a fire can occur in the reactor building during the post-accident phase. Two categories of events have been defined in the reactor design for keeping safety in this phase:

- RRC-A (risk reduction category A): prevention of fusion of the core
- RRC-B (risk reduction category B): prevention of release of big quantities of contaminated particles in case of core fusion

The second category (RCC-B) is the most severe for the materials, as it involves exposure to more radiations and thermodynamic shocks. That is why it has been chosen for the testing of paintings and coatings. The testing goal is to check that the whole surfaces of paintings and coatings inside the reactor building don't release more than 50 kg of debris, in order not to block up the sumps of the borated water aspersion circuit. Therefore every material tested does not get any qualification, but an assessment of the release per square meter, which must be consolidated with all other paintings and coatings inside reactor building.

However it is clear that the goal is to get as few release of material as possible in order to be accepted by EDF.

The testing sequence for RCC-B is as follows:

- Exposure to 60 years ageing Gamma radiations: 250 kGy
- Exposure to 1st accident Beta radiations: 1 MGy
- 1st thermodynamic shock before accident: 5,5 bar and 156°C, back to 1 bar in 24 h
- 2nd thermodynamic shock after accident: 5 bar and 156°C, back to 2 bar in 12 h
- Exposure to 2nd accident Beta radiations: 22 MGy
- Post-accident weathering: 15 days à 110°C, 2 bar and relative humidity 80%
- Aggressive water spraying

Vulnerability Analysis and EPRESSI

Vulnerability analysis is used to assess the possibility of common mode failures (CMF). For example, defining the redundancy items involved in safety related systems, the reliability of support systems, the probability of electrical failures and the use of mitigation systems.

In the ETC-F, the method used for the vulnerability analysis is conventional. However the major innovation concerns the fire analysis. Indeed the conventional method of justification of resistance to fire was once based on the couple of fire curves ISO834/DSN144, where ISO834 is the curve of the performance of fire protection and DSN144 defines the duration of the fire versus fire load density (in J/m²).

An example in order to understand Figure 5: if the density of fire load is 1700 MJ/m² inside a room, a fire partition with performance EI120 is considered as sufficient.

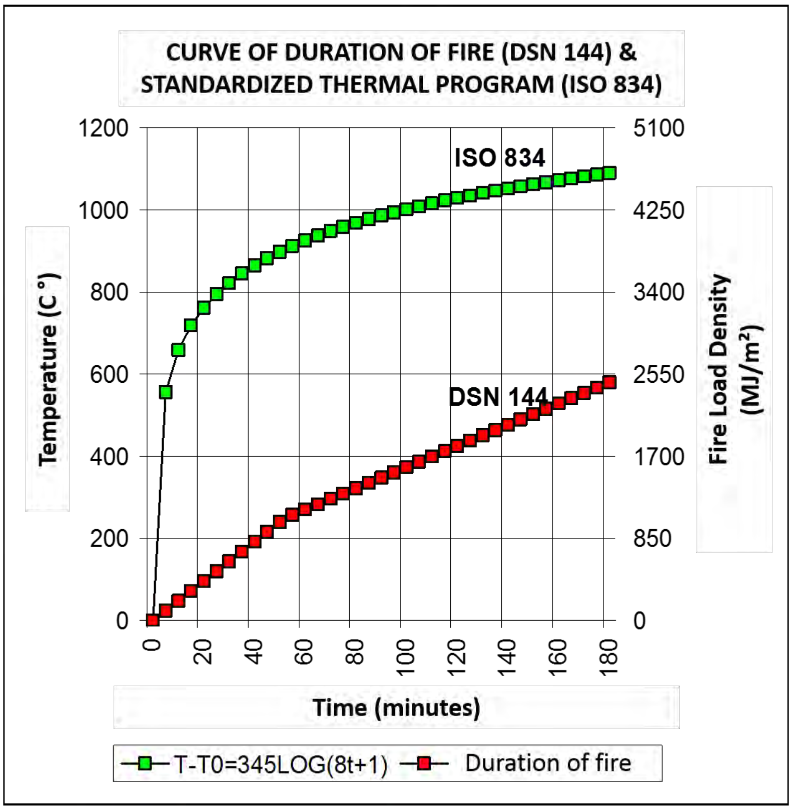


Figure 5 ISO834/DSN144 curve

However the weak point of this method is that it is not representative for actual fires (i.e.: cables fires), which have a temperature curve within the ISO834 curve, but can last longer than the regulation testing time (i.e., actual fire 6 hours compared to testing EI120 which ensures 2 hours fire protection).

In Figure 6 an example for the fire curve is given Therefore, DGSNR (*D*irection *G*énérale de la *S*ûreté *N*ucléaire et de la *R*adioprotection – former central structure of ASN before reform from 2006) sent a letter to EdF requesting the replacement of the DSN144 curve by another means of justification, with the corresponding assessment and planning.

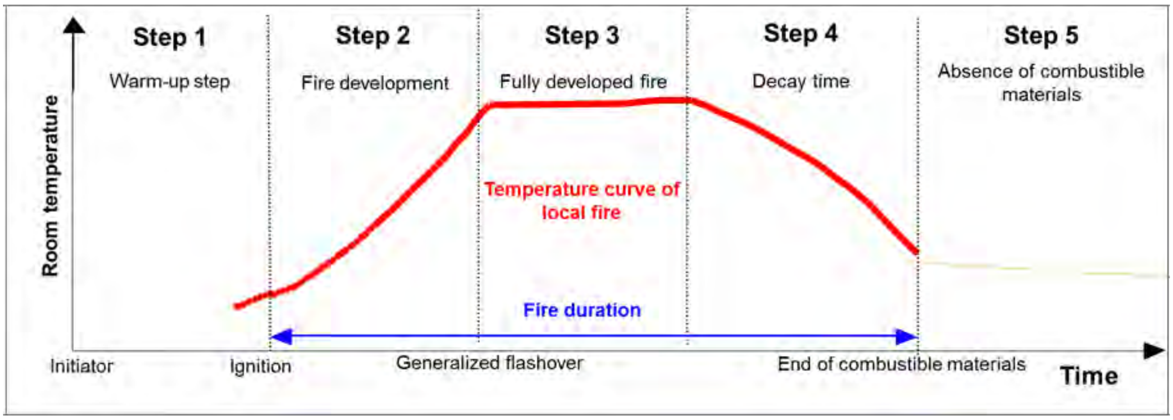


Figure 6 Typical fire curve

EdF's answer has been implemented within the frame of EPR project, and is called EPRESSI (Assessment of the actual performance of sectoring elements in the event of fire) [5].

The first step consists in defining the performance curve of protective materials after the regulation testing time has been reached. This mission has been given to French laboratory CTICM. It was a nonsense to follow the ISO834 curve after 2 hours, because it would lead to over performing materials (with impact on the costs), while actual fire curves tend to decrease after the initial increase. Therefore the testing consists in following the ISO curve during the regulation testing time (i.e. 2 hours for EI120) and then to turn off the furnace. With this input and the thermal characteristics of the protective materials, a calculation model has been set up, in order to define the performance curve for each protection.

The second step consists in defining the fire curves of each room. The main inputs are the characteristics of the room (geometry, openings, etc.), the criteria of risk of flashover (PFG) or localised fire, and the corresponding fire scenario. All inputs are put either in simplified calculation model for rooms within the frame, or through MAGIC calculation software for other cases (especially in fire cells without physical separation).

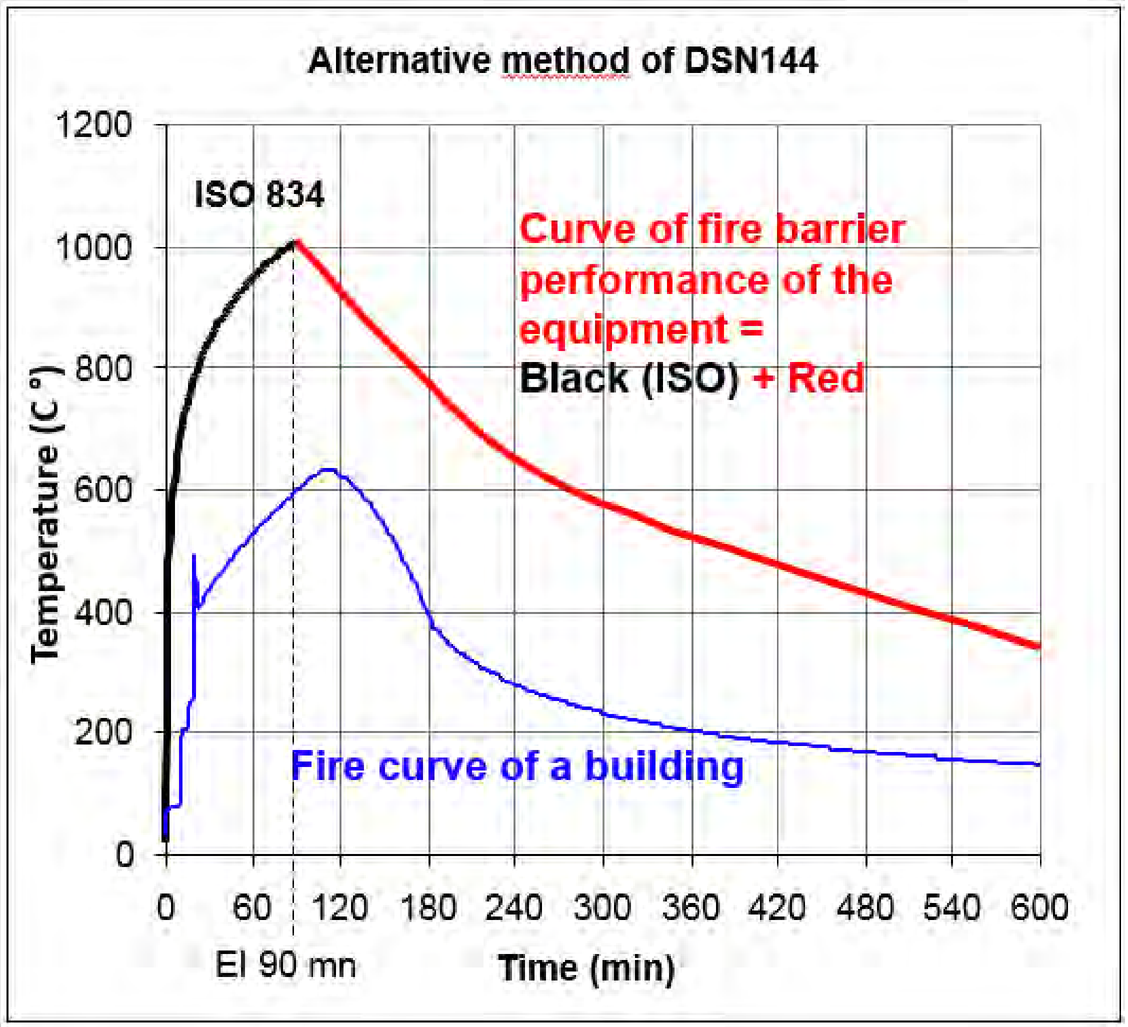


Figure 7 Performance / actual room fire curves

Fire Prevention and Fire Load Suppression (FLS)

It was shown above that one input which leads to a fire scenario is that the room meets either the localised fire risk (PFL) or flashover risk (PFG) criteria, which are (non-exhaustive extract from ETC-F):

PFG if the room with areas below 30 m² meets at minimum one of the following criteria:

1. The presence of more than 3 superimposed horizontal bearings of more than 3 m long
2. The presence of more than 3 superimposed horizontal bearings of more than 3 m long, the highest of which is less than 50 cm from the ceiling
3. The presence of fuel with rapid combustion kinetics (unless special justification is given)

PFL, if the room is not PFG but meets at minimum one of the following criteria:

1. Raceway and local installation: the presence of 2 x 200 mm horizontal secondary raceways, separated by a distance below 1 m and routed in parallel over a distance above 1m or the presence of a 200 mm vertical secondary raceway over a height of 2 m;
2. Electrical cabinets and switch boxes: electrical cabinets and switch boxes which have openings or forced convection are considered to be the potential causes of localised fires.

This requirement introduces the concept of Fire Load Suppression (FLS), the aim of which is to erase the cable trays from fire analysis point of view. In this case the goal is not to keep the function of the cables inside the wrapping, but to ensure that they don't contribute to the fire, and therefore deletes PFG or PFL criteria.

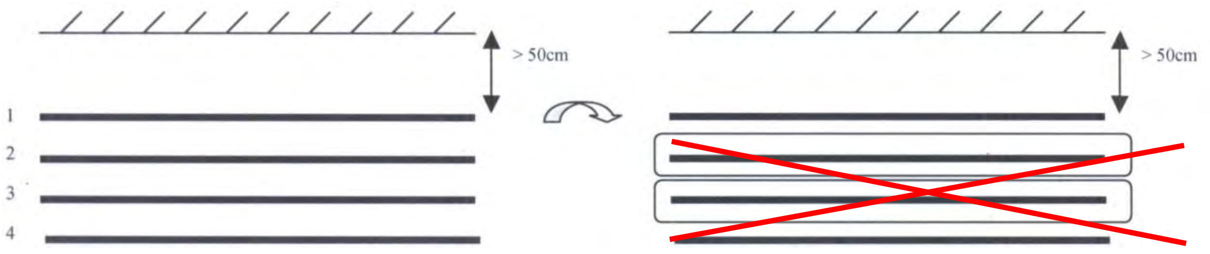


Figure 8 Example of the use of FLS on horizontal cable trays

Three different fire scenario have been selected as representative of the fire sources:

- Pool fire of 3 litres of oil,
- Cable fire,
- Electrical cabinet fire.

The corresponding curves have been modelled in the frame of EPRESSI [5], see also Figure 9:

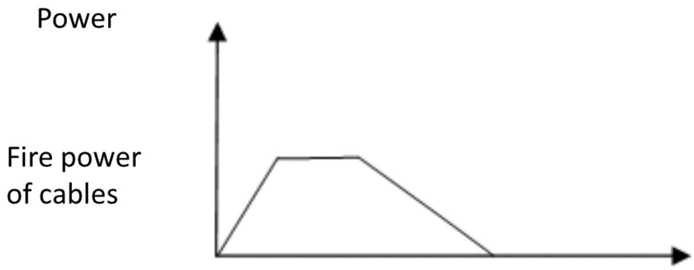


Figure 9 Cable fire curve

Figure 10 3 litres oil fire curve

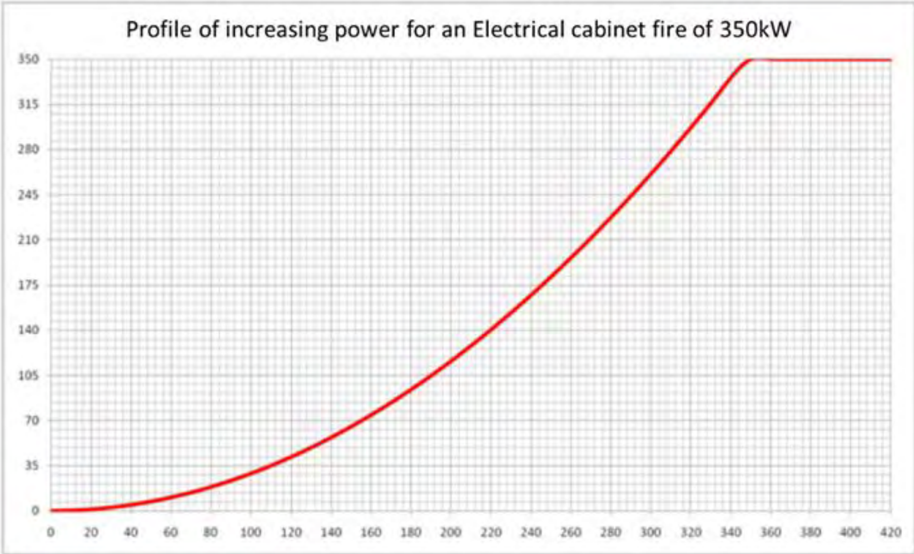


Figure 11 Electrical cabinet fire curve

The main criterion of performance of the protection is to avoid cable pyrolysis gas to be generated and spread into the air during fire. This last criterion leads to a maximum temperature of 210 °C inside the protection during the exposure to fire.

The next step is to define the testing conditions of passive fire protection for the FLS function in order to ensure that they cover all combinations of PFG and PFL criteria from one side, and fire scenario from the other side. Oil fire has been excluded of testing campaign of FLS system, because it has been demonstrated that it is not sufficient to ignite a cable tray situated above it. Other combinations are represented by:

- Protection by FLS of horizontal cable trays inside the fire plume of a cable fire;
- Protection by FLS of horizontal cable trays outside the fire plume of a cable fire;
- Protection by FLS of vertical cable trays inside the fire plume of a cable fire;
- Protection by FLS of vertical cable trays outside the fire plume of a cable fire.

Sometimes it is not possible to protect the cable trays because of harsh access. In this case the FLS protection is put directly around the fire source. That leads to a last testing configuration:

- Protection by FLS as a casing around the fire source (electrical cabinet)

The result is the development of 3 FLS passive fire protections: standard FLS for protection of cables outside the fire plume, reinforced FLS for protection of cables inside the fire plume, and FLS casing around electrical cabinet.

Figure 12 Standard FLS (vertical cable tray outside fire plume)

Figure 13 Reinforced FLS (horizontal cable tray inside fire plume)

Figure 14 Casing around electrical cabinet as potential fire source

Fire Containment

The general concept of fire compartments is implemented in the design of all buildings associated with the reactor systems. These compartments must be completely sealed with fire resistant sealing products for any opening or penetrations. In addition, the concept of fire cells can be used under special circumstances. A fire cell uses separation and protective structures to delay the spread of fire.

ETC-F [**2**] requires each building to be subdivided into fire compartments with the following characteristics: as

- Redundant safety systems must be separated.
- Areas with potential fire hazards must be isolated.
- Protected escape routes for personnel must be provided.

The boundaries between the fire compartments need to meet the requirements as given in the following Table 2.

Table 2 Certification required by ETC-F for fire containment elements

	SFC	SFE	SFS	SFA*	SR**
Structure	R120	R60	R120	R60	R60
Loadbearing walls	REI120	REI120	REI120	REI60	REI60
Non-loadbearing walls	EI120	EI120	EI120	EI60	EI60
Doors	EI_1120S_mC5	EI_160S_mC5	EI_1120S_mC5	EI_160S_mC5	EI_160S_mC5
Penetrations	EI120	EI60	EI120	EI60	EI60
Fire dampers	EI120S	EI60S	EI120S	EI60S	EI60S
Joints	EI120	EI60	EI120	EI60	EI60
Functional enclosures - cases	P120	P120	P120	N/A	N/A

* Minimum classification. The partitions of an SFA take the classification of the adjacent compartments, if this classification is higher.

** Minimum classification. The criteria may be more severe depending on the characteristics of the reference fire of the room or compartment.

Materials and products supplied to seal the compartments must meet these requirements as well as other structural and operational design considerations. Passive fire protection products can be manufactured in a wide range of forms and can be customised to meet specific fire resistant requirements. In addition the installation methods have to be designed to ensure that the fire properties of the final system meet the overall requirements.

a)

b)

c)

Figure 15 a) to c): Protection EI120 of electrical penetrations using silicone based products

a) 

b) 

c) 

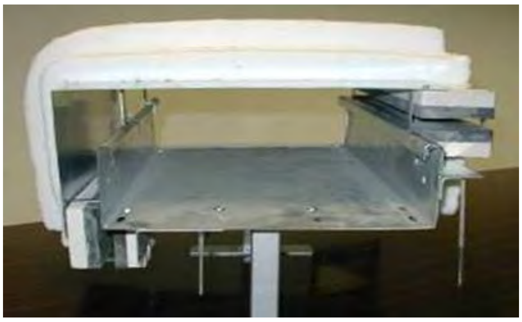
d) 

Figure 16 a) to d): Functional protection P120 of cable trays (vented system for power cables)

PASSIVE FIRE PROTECTION VERSUS ACTIVE FIRE PASSIVE PROTECTION

ETC-F [2] states that active fire protection is subject to random failure. The main consequences are that a redundancy requirement has to be taken into account for these equipment. Also a maintenance program must be put in place in order to ensure the safety level.
In contrary, passive fire protection is not subject to random failure. If installation has been made properly and no damage has been done by an external source, passive fire protection can keep its function during the lifetime of the power plant.

Figure 17 Cable tray wrapping installed outside after 25 years lifetime

CONCLUSIONS

ETC-F gives a major role to fire protection, firstly by involving it into fire prevention (Fire Load Suppression), and secondly by giving priority to physical separation of fire zones into fire containment.
It also applies the new rules of the French safety authorities (ASN), which require more precision in the justification of fire resistance of actual configurations compared to theory (EPRESSI method). The consequence is that manufacturers and installers of passive fire protection cannot be considered as only sub-suppliers, but as partners which must be involved very soon in the development of the nuclear power plants fire protection. These manufacturers and installers must also get a great knowledge of the rules of fire protection, and have to develop skills in fire analysis in order to offer and justify their best solutions for each particular case.

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Recent Enhancements of the OECD FIRE Database

Marina Röwekamp (GRS)
Matti Lehto (STUK)
Wolfgang and Andreas Werner (SAC)

14th International Post-Conference Seminar on
“FIRE SAFETY IN NUCLEAR POWER PLANTS
AND INSTALLATIONS”

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Introduction - 1

- **OECD FIRE** (***F***ire ***I***ncidents ***R***ecords ***E***xchange) Database is 1 of 5 databases collecting events from NPP operation
- In the late 1990s it became evident that IRS and INES are not suitable for specific analysis and use in fire risk assessment
- Purposes of the OECD FIRE Database Project:
  - Provide a platform for multiple countries to collaborate and exchange fire data and thereby to enhance the knowledge of fire phenomena
  - Improve the quality of risk assessment requiring fire related data and knowledge

IRS: Incident reporting system
INES: International Nuclear and Radiological Event Scale

Introduction - 2

- OECD FIRE Database collects fire incidents experience from nuclear power plants by international exchange
  - in an appropriate format
  - in a quality assured and consistent database
- Well-defined Operating Procedures (OP) ensure that the information from the Database is anonymous
- Access to the Database is possible only to individuals in organizations from member countries providing data signing a confidentiality agreement
- Coding Guidelines (CG) and Quality Assurance (QA) Manual are continuously enhanced (by users´ feedback) and extended for ensuring data validity and data quality and complete the Project documentation

Introduction - 3

- FIRE Database, version 2014:01 covers **448 fire event records** up to the end of 2014
- Database represents nearly 8000 years of NPP experience from all plant operational states (POS)
- Project is meanwhile in Phase 4 (2014 – 2015) with **12 member states** (CAN, CZE, ESP, FIN, FRA, GER, JAP, KOR, NED, SWD, SWI, USA)
- FIRE Database use:
  - Identify all types of events and scenarios for inclusion in PSA models ensuring that all mechanisms are accounted for
  - Support Fire PSA by real data from NPP operating experience, in particular to evaluate fire occurrence frequencies
  - Compare national fire event data from member states with the accumulated international data

OECD FIRE Database Search Possibilities - 1

Entry Menu Screenshot

OECD FIRE Database Search Possibilities - 2

- *'Search fire events'* function: either whole Database or already existing subsets generated by earlier queries as basis for new queries
- Different types of fields permit a selection of:
  - One attribute from a pull-down menu
  - Multiple attributes connected by logical *'OR'* within coded fields *'Operation mode'*, *'Country'* and *'Reactor type'*
  - Multiple attributes connected by logical *'AND'* or *'OR'* or *'EXCLUDE'*
  - Coding permits text string searches in all comments fields
- All fields are connected by logical *'AND'*

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OECD FIRE Database Search Possibilities - 3

- Result of any search can be stored as subset
- Different analysis modes can be chosen regarding *'Evaluate'* function:
  - Single selection fields and mutually exclusive multiple selection fields (reactor type, country, operation mode)
  - Pairs of single selection fields (cross tables)
  - Fields in which multiple attributes can be selected

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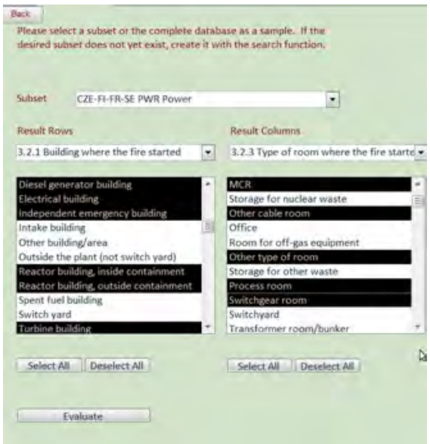
OECD FIRE Database Search Possibilities - 4

Result of exemplary query – evaluation of all ignition mechanisms

Electrical	212
Hot component	80
Hot work	61
Mechanical	40
Other	14
Self ignition	23
Unknown	18

OECD FIRE Database Search Possibilities - 5

Screenshot of the input format for pairs of single selection fields



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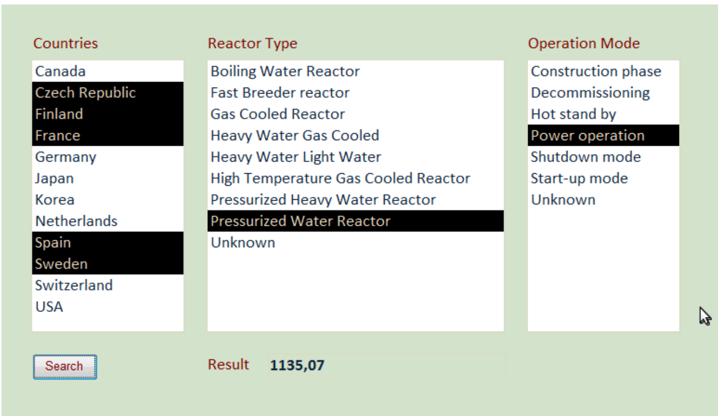
OECD FIRE Database Search Possibilities - 6

Screenshot of search by *'observation times'* function



Actual Applications of the OECD FIRE Database: Compartment Specific Fire Frequencies - 1

Example: Selected PWR buildings, power operation, countries reporting all events

Building \ Compartment-ment type	Process rooms	Rooms for electrical control equipment (incl. MCR)	Switchgear rooms
Turbine Building	2.5 E-04/ry	1.1 E-04/ry	no event
Auxiliary Building	1.2 E-04/ry	2.8 E-04/ry	2.1 E-04/ry
Reactor Building	8.6 E-05/ry	no event	no event
Electrical Building	5.3 E-05/ry	3.2 E-04/ry	1.0 E-03/ry

	Room for electrical cont	Process room	Switchgear room
Auxiliary building	1	4	2
Electrical building	7	1	5
Reactor building, inside con		1	
Reactor building, outside cc		1	
Turbine building	1	11	

Remark: Experience covers 985 ry.

Actual Applications of the OECD FIRE Database: Compartment Specific Fire Frequencies - 2

Example: Selected PWR buildings, low power & shutdown, countries reporting all events

Building \ Compartment-ment type	Process rooms	Rooms for electrical control equipment (incl. MCR)	Switchgear rooms
Turbine Building	2.3 E-03/ry	no event	no event
Auxiliary Building	6.0 E-04/ry	2.8 E-03/ry	no event
Reactor Building	8.6 E-04/ry	no event	no event
Electrical Building	no event	no event	8.3 E-03/ry

	Room for electrical cont	Process room	Switchgear room
Auxiliary building	1	2	
Electrical building			4
Reactor building, inside con		1	
Reactor building, outside cc		1	
Turbine building		10	

Remark: Experience covers 100 ry.

Actual Applications of the OECD FIRE Database: Component Specific Fire Frequencies - 1

Fire frequencies for exemplary components, countries having provided component data

Component type	Average no. of components per NPP unit	No. of fires		Component fire frequency [1/ry]	
		FP	LPSD	FP	LP/SD
Turbine generator	1	10	1	5 E-03	3 E-03
Diesel generator	4	3	4	5 E-04	3 E-03
HV transformer	4	8	3	1 E-04	2 E-03
MV/LV transformer	40	4	6	5 E-05	4 E-04
Electrical cabinet (HV/MV/LV)	1444	21	3	6 E-05	7 E-05
Electrically driven pump	324	5	5	9 E-06	5 E-05

Remark:
Experience covers 2142 ry, 1806 ry for FP (power operation), and 336 ry for LPSD (low power and shutdown states)

Actual Applications of the OECD FIRE Database: High Energy Arcing Faults (HEAF)

- Operating has shown a non-negligible number of reportable events with non-chemical explosions and rapid fires resulting from high energy arcing faults (HEAF) in high voltage equipment such as circuit breakers and switchgears
- OECD FIRE Topical Report for investigating HEAF fire events that resulted in failures of fire barrier elements and/or fire protection features due to pressure build-up in electric cabinets, transformers and/or compartments with the following conclusions:
  - Significant contribution of HEAF fire events **> 10 %** (**51** of 448 records) stored in the OECD FIRE Database
  - Conditional probability of HEAF with fire:
    FP: ~ 6 E-03/ry **LPSD: ~ 1 E-02/ry**
  - HEAF at transformers (highest contribution) increased in recent years (aging of insulations)

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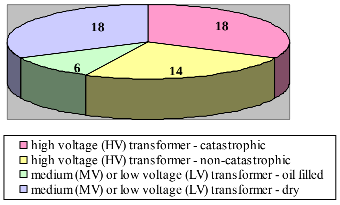
Actual Applications of the OECD FIRE Database: Analysis of Transformer Fires

- Transformer fires contribute significantly to the number of fires in the FIRE Database with > 12 % (56 of 448 records)
- Some of the transformer fires impaired plant safety
- Majority of fires affected HV transformers (oil filled, catastrophic failure)
  - Feedback from operating experience ➡ plant specific improvements
  - Generic improvements: accident management guidelines and emergency response
- Most of the severe transformer fires occur outside buildings



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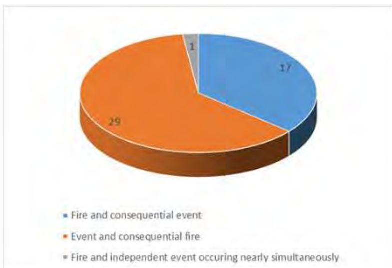
Actual Applications of the OECD FIRE Database: Combinations of Fires and Other Events

- Operating experience and lessons learned from the Fukushima reactor accidents have indicated:
  - Combinations of fires and other anticipated events do occur during the entire lifetime of nuclear installations
  - Such event combinations may impair required function of SSC important to safety
- Contribution of event combinations of fires and other events observed in the OECD FIRE Database: ~ 10 % (47 of 448 event records)
- Events occurred during all POS; even domino effects observed



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Database Challenges

Data Consolidation

- Fire Events Database (FEDB) by U.S. NRC and EPRI issued in 2013 capturing fire event history up through 2009 for ~ 2000 fire events at varying severity levels
- FEDB is intended to be merged in OECD FIRE ➔ Challenges:
  - Aligning information into a usable format
  - Large portion of supplementary information in FEDB remains proprietary plant information
- Advantage: significant increase of number of events in FIRE for PSA use

Challenges for Reducing Uncertainties

- Lack of knowledge on components or geometry of average compartment
- Clarification of overall amount of combustibles inside systems or component

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Conclusions

- OECD FIRE is a valuable tool for facilitating use of NPP fire experience and applicable as a source of generic event data for deterministic fire safety analysis as well as probabilistic fire risk assessment (Fire PRA)
- Database structure has been recently improved and extended facilitating statistical analysis
- Different applications make use of enhanced possibilities
- Various queries based e.g. on reactor type, POS, pre-selection of countries to be considered (e.g. regarding reporting criteria and thresholds), as well as trend analyses are possible
- Compartment specific frequencies can be determined, while estimation of component related frequencies remains a challenge

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Outlook

- OECD FIRE Phase 5 is intended to start in 2016 (3 years duration)
- Still limited statistical Database capabilities have to be and will be further extended
- Application already started:
  generic fire event trees for PRA purposes
- Potential issues for further Topical Reports
  - Apparent causes and direct reasons of fire events
  - In-depth analysis of recent transformer fires
  - Control room abandonment / reduced habitability

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THANK YOU
FOR YOUR ATTENTION!

For questions please contact:

Chair - Marina Röwekamp: Marina.Roewekamp@grs.de or

Vice Chair - Matti Lehto: Matti.Lehto@stuk.fi or

Operating Agent - SAC: werner.sac@gmail.com or

NEA Secretariat - Neil Blundell: Neil.Blundell@oecd.org

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CHARACTERIZING THE THERMAL EFFECTS OF HIGH ENERGY ARC FAULTS

Anthony Putorti[1], Nicholas B. Melly[2], Scott Bareham[1], Joseph Praydis Jr.[1]

[1] National Institute of Standards and Technology (NIST), Gaithersburg, MD, USA
[2] U.S. Nuclear Regulatory Commission (NRC), Washington, DC, USA*

ABSTRACT

International and domestic operating experience involving High Energy Arc Faults (HEAF) in Nuclear Power Plant (NPP) electrical power systems have demonstrated the potential to cause extensive damage to electrical components and distribution systems along with damage to adjacent equipment and cables. An international study by the Committee on the Safety of Nuclear Installations (CSNI) "OECD Fire Project – Topical Report No. 1: Analysis of High Energy Arcing Fault (HEAF) Fire Events" published June 25, 2013 [1], illustrates that HEAF events have the potential to be major risk contributors with significant safety consequences and substantial economic loss. In an effort to better understand and characterize the threats posed by HEAF related phenomena, an international project has been chartered; the Joint Analysis of Arc Faults (Joan of ARC) OECD International Testing Program for High Energy Arc Faults. One of the major challenges of this research is how to properly measure and characterize the risk and influence of these events. Methods are being developed to characterize relevant parameters such as; temperature, heat flux, and heat release rate of fires resulting from HEAF events. Full scale experiments are being performed at low (≤ 1000 V) and medium (≤ 35 kV) voltages in electrical components. This paper introduces the methods being developed to measure thermal effects and discusses preliminary results of full scale HEAF experiments.

INTRODUCTION

Switchgear, load centers, and bus bars/ducts (440 V and above) are subject to a unique failure mode and, as a result, unique fire characteristics. In particular, these types of high-energy electrical devices are subject to high-energy arcing fault (HEAF). This fault mode leads to the rapid release of electrical energy in the form of heat, vaporized copper/aluminum, and mechanical force. Faults of this type are also commonly referred to as high energy, energetic, or explosive electrical equipment faults or fires.

The energetic fault scenario typically consists of two distinct phases, each with its own damage characteristics. The first phase is characterized by a short, rapid release of electrical energy which may result in catastrophic failure of the electrical enclosure, ejection of hot projectiles (from damaged electrical components or housing) and/or fire(s) involving the electrical device itself, as well as any external exposed combustibles, such as overhead exposed cable trays or nearby panels, that may be ignited during the energetic phase. The second phase, i.e., the ensuing fire(s) typically includes ignition of combustible material within the HEAF zone of influence (ZOI). The resulting fire may be due to the ejection of hot particles or piloted ignition of combustibles. HEAF events are of concern due to their potential to im-

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* This paper was prepared (in part) by employees of the United States Nuclear Regulatory Commission. It presents information related to NRC upcoming testing programs. NRC has neither approved nor disapproved its technical content. This paper does not establish an NRC technical position.

pact adjacent items important to safety and current limitations in characterizing the ZOI as defined in NUREG/CR-6850 [2].

Due to the potential safety significance of HEAF events, the OECD (*O*rganization for *E*conomic *C*o-operation and *D*evelopment) Nuclear Energy Agency (NEA) Integrity and Ageing Working Group (IAGE) initiated a task on High Energy Arcing Events (HEAF) in 2009 to provide an in-depth investigation on HEAF events in NEA member states [3]. The objective of this working group is to determine damage mechanisms, extent of areas affected, methods of protecting systems, structures and components (SSC) and possible calculation methods for modeling of HEAF events as applicable to fire protection in nuclear power plants (NPP). As part of this effort a testing program has been initiated to investigate the HEAF fire phenomena to inform future deterministic and probabilistic methods.

This paper presents methods for measuring the heat release rates of ensuing fires and measuring the heat fluxes above and around the electrical enclosures during the HEAF experimental program. Limited data are also presented.

BACKGROUND

In order to characterize the effects of the HEAF and ensuing fire on the surrounding equipment, various phenomena were chosen for characterization in the OECD program. These include enclosure pressure, enclosure surface temperature, heat release rate, and heat flux to target equipment. Electrical test parameters such as arc voltage, arc current, and arc duration were also measured during the experiments.

Experiments were performed at KEMA-Powertest, located outside of Philadelphia, Pennsylvania, USA. The test facility includes a five-sided test cell approximately 8 m (26 ft) high, 7 m (24 ft) deep, and 9 m (29 ft) wide. The sixth side of the test cell includes a roll-up door that was fully open during the experiments. Bus bar connections for supplying low and medium voltage test current are located on opposite side walls of the test cell. KEMA-Powertest provided measurements of electrical enclosure pressure, temperatures of slug calorimeters, electrical test parameters, videography, and high speed videography during the experiments. NIST provided measurements of heat release rate, heat flux, electrical enclosure surface temperature, thermal imaging, and multiple location videography during the experiments.

The NPP equipment for the experiments was provided by OECD/NEA HEAF Project partners. Fourteen experiments have been performed to date using six electrical enclosures. Nominal test voltages ranged from 480 VAC to 7200 VAC, and nominal test currents ranged from 24 kA to 50 kA. All of the experiments conducted thus far have been performed with three phase power supplied in a delta configuration. The arcs were initiated in the enclosures by shorting across all three bus bar phases with a 2.6 mm diameter (10 AWG) tinned copper stranded wire prior to energizing the enclosures.

EXPERIMENTAL METHODS

Heat Release Rate

In order to measure the heat release rate (HRR) of the ensuing fires caused by the HEAF events, a portable oxygen consumption heat release rate hood apparatus was deployed. The portable hood was first used in the HELEN-FIRE experiments to measure the heat release rates of fires in control cabinets as described in NUREG/CR-7197 [4]. The portable apparatus was further developed and refined for use in the HEAF experiments. In the current form, the apparatus is a portable stand-alone system resistant to the effects of electromagnetic interference (EMI).

The portable hood, installed in the HEAF test cell with an electrical enclosure, is shown in Figure 1 and Figure 2. The hood was approximately 2.44 m by 2.44 m in width, with a clear height of approximately 3.0 m above the floor. Side skirts constructed of fiberglass welding curtain hung around the hood opening to reduce the quantity of smoke that escaped from the sides of the hood.

The exhaust duct exiting the top of the hood is approximately 0.46 m in diameter, and carries air and combustion products through flow measurement, gas sampling, and exhaust fan sections. The ducting is supported by scaffolding (not shown). The distance between the hood and the flow measurement section was varied with additional duct sections (not shown) to provide adequate clearance between the electrical enclosures and the metal scaffolding. The hood exhaust fan motor was powered by a dedicated portable electrical generator located outside of the test cell. The gas analyzers and data acquisition system were located in an interior hallway outside the rear of the test cell for protection from physical hazards, electrical hazards, and combustion products.

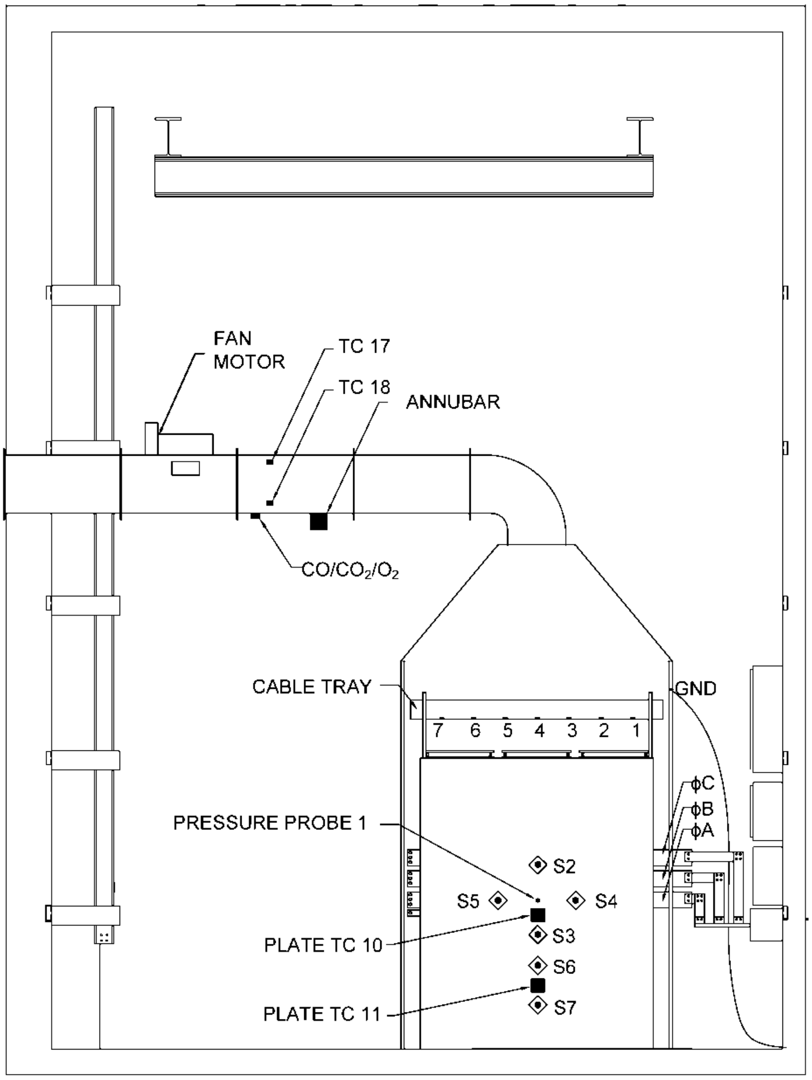

Figure 1 Elevation view of calorimetry hood, enclosure, and instrumentation. Plate thermometers facing downward under cable tray, slug calorimeters denoted by diamond symbols, stack thermocouples denoted by "TC", exhaust gas sampling probe location denoted by "$CO/CO_2/O_2$"; earth ground cable attached to hood denoted "GND", bus bars labeled with phases A, B, and C; not to scale

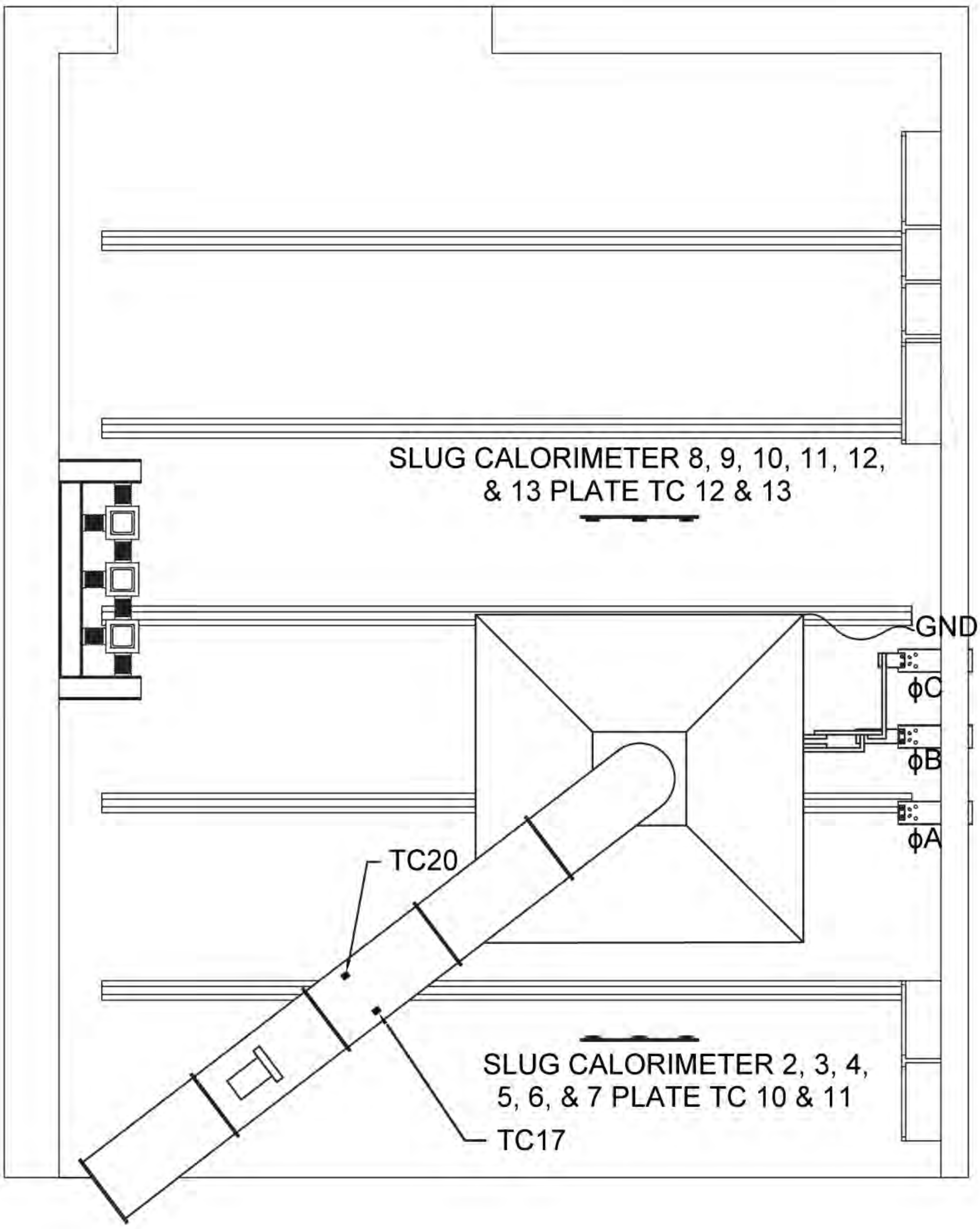

Figure 2 Plan view of calorimetry hood, cabinet, and instrumentation; not to scale

The heat release rates of the ensuing fire, $\dot{Q}(t)$ (kW), was measured by oxygen consumption calorimetry, taking into account the measured concentrations of oxygen, carbon dioxide, and carbon monoxide in the exhaust gas [5], [6].

$$\dot{Q}(t) = \left[E_{O_2}\emptyset - \left(E_{CO} - E_{O_2}\right)\frac{1-\emptyset}{2}\left(\frac{X_{CO}}{X_{O_2}}\right)\right]\frac{\dot{m}_e}{1+\emptyset(\propto -1)}\frac{M_{O_2}}{M_a}\left(1 - X_{H_2O,\infty}\right)X_{O_2,\infty} \tag{1}$$

$$\emptyset = \frac{X_{O_2,\infty}\left(1 - X_{CO_2} - X_{CO}\right) - X_{O_2}\left(1 - X_{CO_2,\infty}\right)}{\left(1 - X_{O_2} - X_{CO_2} - X_{CO}\right)X_{O_2,\infty}} \tag{2}$$

Here $\propto$ is the combustion expansion factor of 1.105, $\emptyset$ is the oxygen depletion factor, E_{O_2} is the net heat released for complete combustion of typical fuels, 13100 kJ/(kg O_2), E_{CO} is the net heat released for complete combustion of CO, 17600 kJ/(kg O_2), M_a is the molecular weight of incoming air [g/mol], M_{O_2} is the molecular weight of oxygen [g/mol], $\dot{m}_e$ is the exhaust mass flow rate in the duct [kg/s], $X_{O_2,\infty}$ is the initial oxygen volume fraction, X_{O_2} is the measured oxygen volume fraction, $X_{CO_2,\infty}$ is the initial carbon dioxide volume fraction, X_{CO_2} is the measured carbon dioxide volume fraction, X_{CO} is the measured carbon monoxide volume fraction, and $X_{H_2O,\infty}$ is the volume fraction of water vapor.

Due to the large range of possible heat release rates, the apparatus design was biased toward resolution of the relatively small heat release rates expected from the ensuing fires. The heat release rate measurement range for the hood is approximately 10 kW to 3000 kW. The velocity of gases flowing through the hood duct was measured using an Annubar®[1] averaging differential pressure element attached to a differential pressure transducer. The geometry of the duct system differed from that specified by the manufacturer, resulting in less flow straightening and flow development. Due to the difference, calibration fires were used to determine the flow coefficient for the differential pressure element.

Calibration fires were produced by a propane diffusion burner approximately 0.3 m by 0.3 m in size, providing fire heat release rates of approximately 35 kW and 50 kW. The propane burner heat release rates were calculated from the propane heat of combustion and the standard volume of propane provided to the burner as measured by a dry test flow meter corrected for temperature and pressure. For the oxygen consumption calculation of heat release from the propane burner, the value of E_{O_2} for propane is 12.78 MJ/(kg O_2) [7]. The combined standard uncertainty, composed of Type A and Type B uncertainties, in the base heat release measurements was 10 %. The expanded uncertainty in the base heat release measurements was 20 %, with a coverage factor of 2, which corresponds to a confidence interval of 95 % [8], [9].

During the experiments, the effects of wind and smoke escaping the sides of the hood increased the level of measurement uncertainty. The one open side of the test cell allowed prevailing winds to drive combustion products away from the hood. Fire resistant fabric side skirts reduced the loss of smoke, but wind conditions resulted in the loss of significant quantities of smoke in some experiments. For each HEAF fire experiment, additional uncertainty contributions due to wind and losses of combustion products were estimated using observations and video recordings.

Temperature and Heat Flux

One measure of the thermal environment during HEAF events and ensuing fires is the thermal heat flux imposed on materials surrounding the cabinets. There are various techniques available for measuring thermal heat flux, including water cooled transducers, slug calorimeters, directional flame thermometers (DFT), and plate thermometers (PT). The use of these transducers for measuring the heat fluxes in HEAF events was explored in an NRC funded project [10]. For the OECD program HEAF experiments, the choice of transducers was revisited.

The prime considerations for the experiments included a transducer that was sturdy and possessed a relatively short response time. One of the technologies frequently used in fire experiments is the water-cooled heat flux transducer (Schmidt-Boelter and Gardon Gauge types). There are two major drawbacks for their use in HEAF experiments, however. The first drawback is the presence of cooling water in the test cell, which presents logistical complications and safety hazards. The second drawback is related to the dynamic range of the

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1 Certain commercial equipment, instruments, or materials are identified in this paper in order to specify the experimental procedure adequately. Such identification is not intended to imply recommendation or endorsement by the National Institute of Standards and Technology, nor is it intended to imply that the materials or equipment identified are necessarily the best available for the purpose.

sensors. In order to capture the low heat fluxes from small ensuing fires to a reasonable level of uncertainty, a transducer with a measurement range from approximately 10 kW/m^2 to 200 kW/m^2 could be chosen, resulting in an expanded uncertainty of approximately 6 kW/m^2 (coverage factor of 2, 95 % confidence interval). A transducer of this range may be destroyed, however, by the fluxes resulting from impingement of plasma from the arcing portion of the experiment, which may be on the order of 1 MW/m^2. If a transducer with a measurement range high enough to survive the arcing is used, the heat flux measurement uncertainty would be too high for the ensuing fires.

Plate thermometers are robust sensors that can survive in hostile HEAF environments. A plate thermometer similar to that described in the literature [11], [12], and [13] was chosen for heat flux measurements in the OECD experiments due to its rugged construction, low cost, lack of cooling water, and emissivity and convective heat flux coefficients similar to power plant safety-related equipment.

The plate thermometer (PT) from the literature was modified for faster response and simpler manufacture. In order to decrease response time, the specified sheathed thermocouple was replaced by 0.51 mm diameter (24 AWG) Type-K thermocouple wires welded directly to the rear of an Inconel® 600 plate. The thickness of the mineral fiber blanket was increased to approximately 25.4 mm to decrease heat loss. A square plate of Inconel, approximately 100 mm by 100 mm in size, replaces the bent plate to reduce heat losses from the sides and simplify electrical isolation. Machine screws with ceramic washers allow for legs to be attached at the rear of the plate thermometer in order to simplify installation into cable trays and increase locational accuracy. The modified plate thermometer is shown in Figure 3 and Figure 4.

The incident heat flux on a plate thermometer can be calculated from a heat balance using the following relation, a rearrangement of Equation 18 from Ingason and Wickstrom [12]:

$$\dot{q}''_{\mathrm{inc}} = \sigma \cdot T_{\mathrm{PT}}^4 + \frac{(h_{\mathrm{PT}} + K_{\mathrm{cond}})(T_{\mathrm{PT}} - T_\infty)}{\varepsilon_{\mathrm{PT}}} + \frac{\rho_{\mathrm{ST}} \cdot C_{\mathrm{ST}} \cdot \delta \cdot \left(\frac{\Delta T_{\mathrm{PT}}}{\Delta t}\right)}{\varepsilon_{\mathrm{PT}}} \qquad (3)$$

Here $\dot{q}''_{\mathrm{inc}}$ is the incident heat flux, σ is the Stefan-Boltzmann Constant, 5.670×10^{-8} W/(m^2·K^4), T_{PT} is the temperature of the plate (K), h_{PT} is the convection heat transfer coefficient, 10 W/(m^2·K), K_{cond} is the conduction correction factor determined from NIST cone calorimeter data, 4 W/(m^2·K), T_∞ is the ambient temperature (K), $\varepsilon_{\mathrm{PT}}$ is the plate emissivity, 0.85 at 480 °C as rolled and oxidized and specified by the alloy manufacturer, ρ_{PT} is the alloy plate density, 8470 kg/m^3 from the alloy manufacturer, C_{ST} is the alloy plate heat capacity, 502 J/(kg·K) at 300 °C from the alloy manufacturer, δ is the alloy plate thickness, 0.79 mm, and Δt is the data acquisition time step of 0.2 s.

The modified PTs were heated in the cone calorimeter [14] to verify their performance and the fit of the simple thermal model in Equation (3). The plates were tested from 5 kW/m^2 to 75 kW/m^2 by heating from ambient temperature to steady state and then allowing them to cool. At a steady state flux of 75 kW/m^2 the calculated heat flux reached 63 % of the incident heat flux in approximately 0.7 s. The combined standard uncertainty in steady state heat flux measured by the plate thermometers, composed of Type A and Type B uncertainties, is 2.5 % at 75 kW/m^2. The expanded uncertainty in the steady state heat flux measurement is 5 % at 75 kW/m^2, with a coverage factor of 2 which corresponds to a confidence interval of 95 % [8].

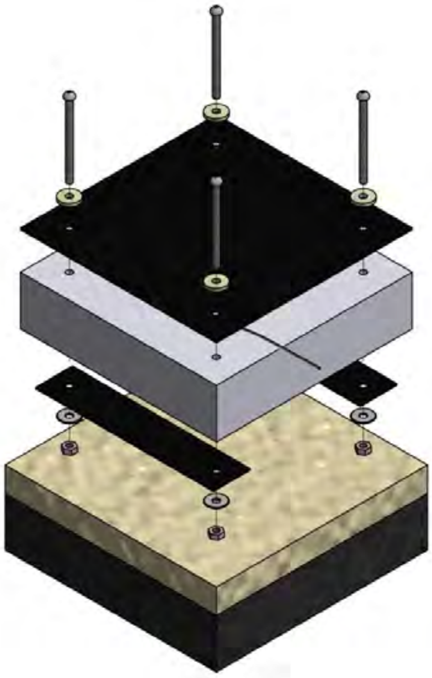


Figure 3 Exploded view of modified plate thermometer with cone calorimeter sample holder

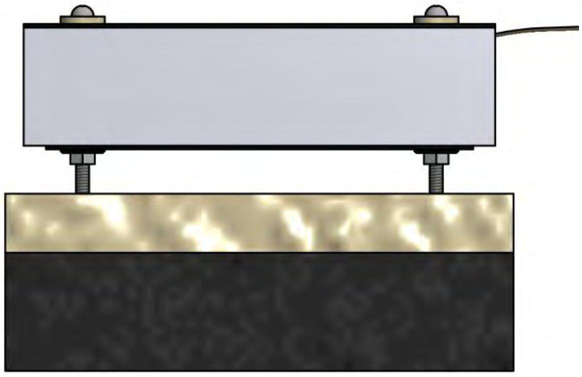


Figure 4 Elevation view of modified plate thermometer on cone calorimeter sample holder

The heating of plate TCs in the cone calorimeter was modeled in one dimension with the Fire Dynamics Simulator (FDS) [15] to verify the assumptions and property data. Agreement to within 1 % was found between the temperatures measured during exposure in the cone calorimeter and the FDS predicted temperatures. Data from heating the plate thermometer at 75 kW/m^2 in the cone calorimeter is included in the FDS validation library.

Sensor Wiring and Data Acquisition

Due to the voltages, currents, and electrical arcing that are present in and around the electrical equipment used in the HEAF experiments, electromagnetic interference (EMI) was present in the test facility. The electric and magnetic fields are capable of inducing voltages and currents in the sensor and data acquisition wiring. In order to reduce the effects of EMI, several strategies were employed in concert: shielding, isolation, signal conditioning, grounding, and electrical power conditioning. This multi-faceted approach greatly reduced or eliminated the effects of EMI on the measurement results.

A conceptual drawing of the sensors, instrumentation, and data acquisition is shown in Figure 5 and Figure 6. Figure 5 shows the wiring concept for a typical sensor, which includes plate thermometers and thermocouples. Figure 6 shows the wiring concept for the gas analyzers and differential pressure transducer for measuring hood flow rates.

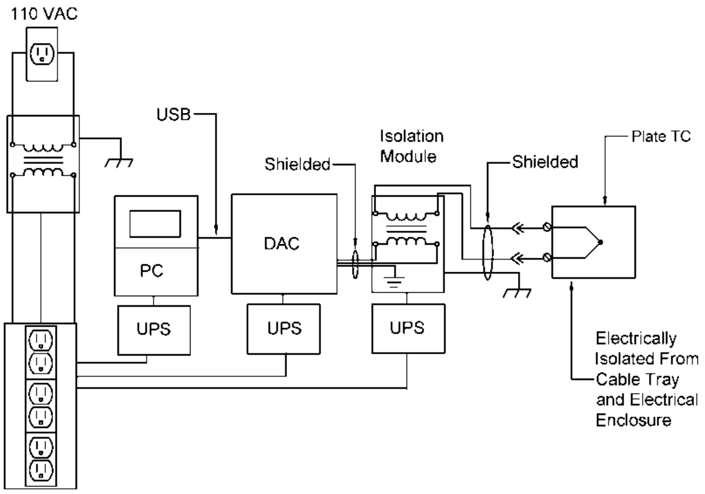


Figure 5 EMI resistant wiring concept for plate thermometer measurements

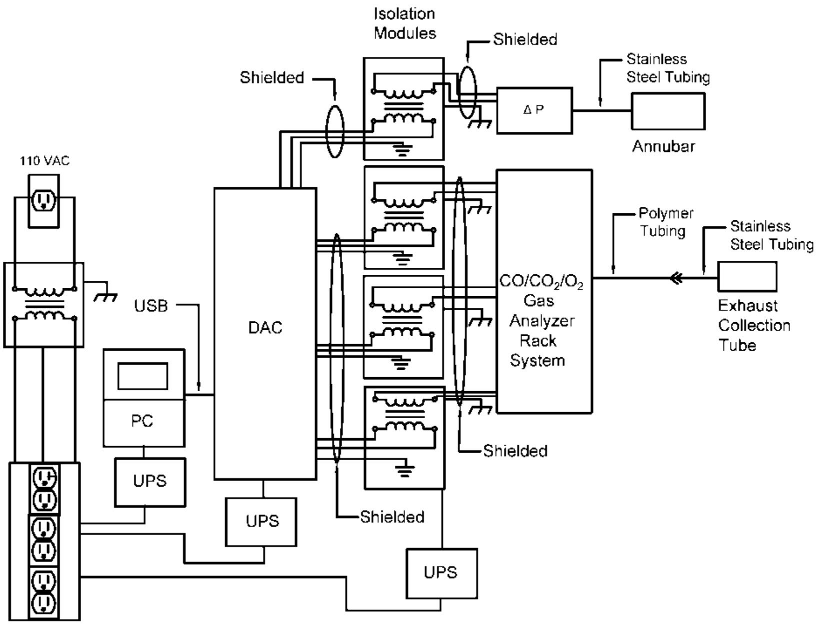


Figure 6 EMI resistant wiring concept for gas analysis and hood flow measurements

The sensor extension wiring is shielded, with the shield grounded near the sensor to earth ground. The sensor extension traveled through the test cell, via a route as far away from the current supply bus bars as practicable, through the back wall of the test cell, and to a signal conditioner and isolation transformer (isolation module). Each sensor channel had a dedicated isolation module. For thermocouple channels, the isolation modules also converted the low level mV signal produced by the thermocouple to a high level signal (0 VDC to + 5 VDC) linearized for a temperature range of – 100 °C to 1350 °C using a simulated ice junction.

Non-thermocouple sensors were served by isolation modules that converted the input signals to ± 1 VDC or ± 5 VDC output signals.

The output of each isolation module was connected to one of two data acquisition modules, housed in a separate enclosure, via a shielded cable that was grounded to earth ground. The high level signals from the isolation modules were sampled by the data acquisition system (DAC), with the results communicated to a laptop computer (PC) via a USB cable. Data were recorded by the data acquisition system at a rate of 5 Hz.

The main 115 VAC building power for the PC, data acquisition system, isolation modules, gas analyzers, and pressure transducer was supplied through signal conditioners, uninterruptible power supplies, and isolation transformers. The equipment chassis were grounded to earth. The heat release rate hood and duct support scaffolding were also grounded to earth. Grounding all of the equipment and cable shielding to the same earth ground prevented ground loops. The cable trays above the electrical enclosures were electrically isolated from the enclosure and hood and ungrounded. The enclosures were supplied with 3 phase power in a delta configuration, with the enclosure ungrounded.

RESULTS

Heat Release Rate

During the arcing phase of the HEAF experiments it was common for large quantities of rapidly generated combustion products to escape the measurement hood to the atmosphere and therefore avoid measurement. In order to measure heat release rate during the arcing phase, a larger hood would be needed to capture the combustion products, the size of which is impractical for a portable system. To measure larger fires, the exhaust mass flow rate would need to be increased, which would decrease the ability to resolve small fires, and increase measurement uncertainty. The purpose of the portable apparatus is to measure the HRR of the secondary phase of the HEAF event, i.e., the ensuing fire.

Oxygen consumption calorimetry for ordinary combustible materials such as flammable gases, flammable and combustible liquids, wood, and polymers utilizes a heat content of approximately 13.1 MJ/kg of oxygen consumed [5]. During the HEAF portion of the experiments, a significant quantity of copper and aluminum were oxidized. The heat release rate calculations do not take into account the difference in E_{O2} between oxidation of metals and the combustion of ordinary combustibles. During the HEAF portion of the experiments, the average heat release rate [MW] was estimated from the arc energy [MJ] divided by the arc duration (s) instead of oxygen consumption calorimetry.

The HEAF and ensuing fire from an experiment in a medium voltage cabinet are shown in Figure 7 and Figure 8. The nominal cabinet operating conditions were 7200 VAC and 24 kA with an arc of approximately 2554 ms in duration. The initial heat release from the cabinet due to the arc was not fully captured, but may be estimated as an average of approximately 28 MW from the arc energy expended during the arc. The heat release rate of the ensuing fire that occurred in the electrical enclosure following the HEAF event was recorded, with the primary fuel load consisting of the breaker housing. The maximum heat release rate of the ensuing fire was 165 kW. The expanded uncertainty in the heat release measurement is 25 %, with a coverage factor of 2, which corresponds to a confidence interval of 95 %.

Figure 7 Medium voltage HEAF

Figure 8 Ensuing fire

Temperature and Heat Flux

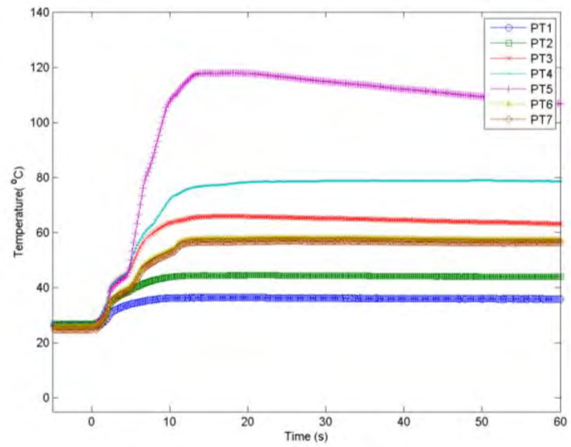
A low voltage HEAF in an enclosure with nominal operating conditions of 480 VAC and 50 kA with an arc of approximately 2115 ms in duration is shown in Figure 9 and Figure 10. The locations of the plate thermometers installed in the experiment are shown in Figure 1 and Figure 2. The temperatures reported by the thermocouples attached to the back of the nickel alloy plates of the modified plate thermometers are shown in Figure 11 and Figure 12.

Figure 9 Low voltage HEAF; front of cabinet

Figure 10 Low voltage HEAF; top of cabinet; cable tray visible in upper right of photo

During some of the experiments, plate thermometers were directly impacted by plasma ejected from the cabinet. This contact resulted in abnormal thermocouple voltages and therefore to abnormal temperature change readings from the thermocouples. The resulting abnormal voltages, temperatures, and heat fluxes could be positive or negative. The data from plate TCs directly impacted by plasma could be used outside of the time where the arc was present by using the plate TC equations. An average heat flux during the arc can be calculated by treating the plate as a well-insulated, thermally-thin solid.

The heat flux histories of the plate thermometers in the low voltage experiment were calculated from the temperature data and are shown in Figure 13 and Figure 14. In this particular case, the plasma generated by the arc event did not cause significant abnormal voltages. The peak incident heat flux measured approximately 0.9 m (3 ft) from the cabinet at PT location 10 was 17 kW/m^2 during this experiment. The peak incident heat flux measured in the cable tray located approximately 0.3 m (1 ft) above the cabinet at PT location 5 was 72 kW/m^2 during this experiment.



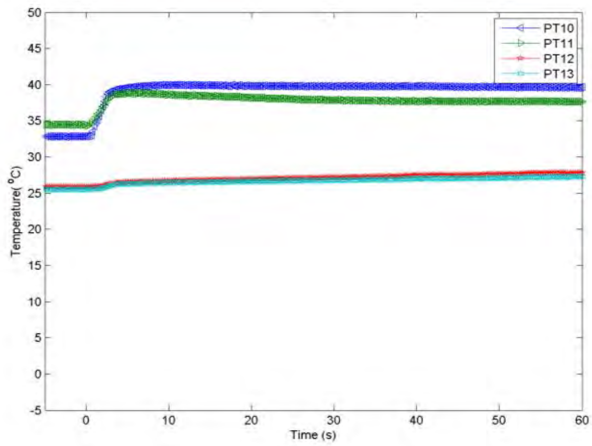
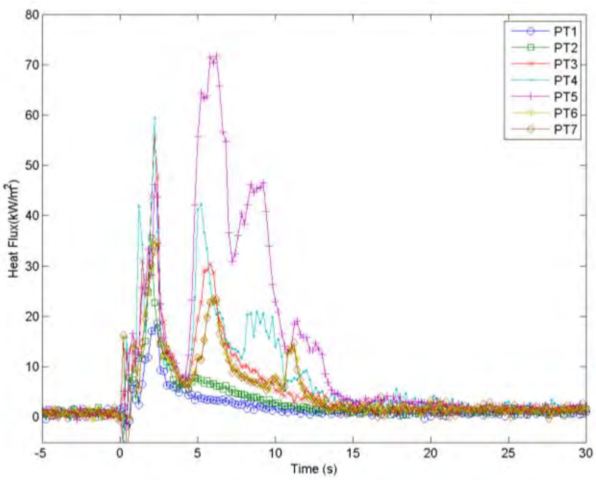


Figure 11 Cable tray plate thermometer temperatures, low voltage test. Temperature expanded uncertainty of ±3°C with coverage factor of 2

Figure 12 Vertical plate thermometer temperatures, low voltage test. Temperature expanded uncertainty of ±3°C with coverage factor of 2



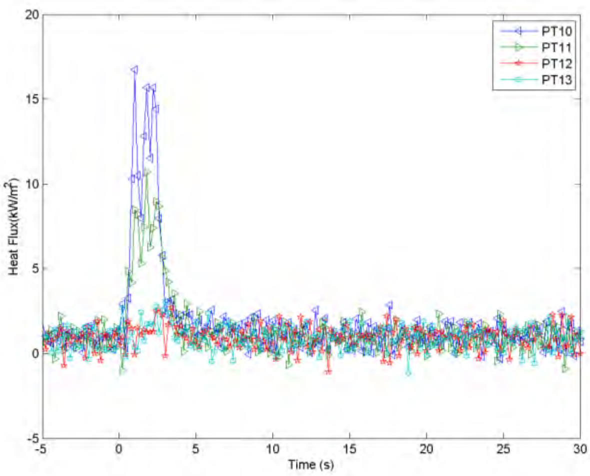


Figure 13 Cable tray PT heat flux, low voltage test; heat flux expanded uncertainty of ±4 kW/m^2 at 75 kW/m^2 with coverage factor of 2

Figure 14 Vertical PT heat flux, low voltage test; heat flux expanded uncertainty of ±4 kW/m^2 at 75 kW/m^2 with coverage factor of 2

The Fire Dynamics Simulator [15] was used to simulate the one dimensional heating of a plate thermometer (PT5 above) exposed to the heat flux history calculated from the plate thermometer measurement, Equation 3. The experimentally measured plate temperature and the corresponding FDS prediction agreed to within 2 %, which serves as verification of the method to calculate the heat flux from the measured plate temperature.

The test facility provided slug calorimeters for the measurement of the incident energy; that is, the total energy absorbed by thermally-thin targets at various locations around the enclosure. The incident energy was calculated from the temperature history according to standard methods [16]. The measurements from the slugs were also found to be adversely affected by direct impingement of plasma exiting the cabinets. The incident energy during the arc phase of the 480 V experiment was approximately 31 kJ/m^2 (0.75 cal/cm^2), measured at Slug 2.

CONCLUSIONS

The portable oxygen consumption calorimetry hood is effective for measuring the heat release rate of HEAF ensuing fires. As expected, HEAF arcing events produce too much effluent to be captured by the hood as designed. The average energy release rate during the arcing period, however, can be estimated from the electrically measured arc energy and arc time.

Plate TCs are an effective method for characterizing the thermal assault on NPP cable trays and equipment, and can serve as input boundary conditions for FDS modeling of target objects. Data during the arc event may need to be averaged over the time of the arc if plasma impingement on the plate TC causes abnormal signals.

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ASSESSMENT OF THE BURNING BEHAVIOR OF PROTECTED AND UNPROTECTED CABLES AND CABLE TRAYS IN NUCLEAR INSTALLATIONS USING SMALL- AND LARGE-SCALE EXPERIMENTS

Matthias Siemon[1], Olaf Riese[1], Jochen Zehfuß[1]

[1] Institut für Baustoffe, Massivbau und Brandschutz (iBMB),
Technische Universität Braunschweig, Braunschweig, Germany

ABSTRACT

Electric installations and cables are a main fire risk source in industrial buildings and power plants. In general, cables and cable systems are associated with flash-over phenomena due to pyrolysis of fuel gases induced by the heat of an adjacent fire, fire spread along cable trays affecting additional areas besides the fire origin, being an ignition source due to malfunction. If burning, cables can emit large amounts of smoke and toxic products affecting occupants as well as the long-term functionality of structure and installations. Paying attention to these risks has led to the development of fire retardant non-corrosive (non-halogenated) cables which are qualified to reduce the individual or all of the risks mentioned. For existing installations in industrial buildings and power plants with halogenated cables, different protection measures are available and widely applied retroactively. Important protective measures are intumescent or ablative coatings, cable casings and bindings.

For qualification of the effects of the protection measures, small-scale tests investigating a single cable specimen as well as large-scale cable tray test setups have been developed and carried out in the last 20 years at iBMB. In this paper, these test results are analysed regarding their effects on the heat release, ignition time and fire spread over cable trays. Furthermore, national and international research projects have investigated the burning behaviour of different cable types, tray installations, tray loading and spacing and ventilation conditions. As a conclusion, the main outcomes of past researches are summarized. Influence factors (e.g. pre-heating due to high power utilization, influence of cable aging) which have not been accounted for in detail are emphasized.

The modelling of unprotected cables has been internationally studied in recent years. For future applications, the question of applicability of recently developed sub-models on the fire behaviour of protected cables has to be answered. The results presented in this paper may provide the basis for the planning of further validation experiments to the fire performance of protected cables.

INTRODUCTION

During the last 20 years, extensive research efforts have been taken to investigate the burning behaviour of protected and unprotected cable trays equipped with PVC cables. One goal was the need of detailed requirements for trays and electrical installations installed in nuclear power plants to account for the risks of self-ignition, flash-over and fire spreading. In some cases, e.g. during operation, the replacement of halogenated cables with FRNC (*f*ire retardant non-*c*orrosive) cables is difficult. This led to the utilization of protective measures such as intumescent coatings, ablation coatings or protection bindings to prevent the risks of fire spreading, flash-over and burning.

The plurality in potential boundary conditions and heat exposure on cable installations has to be considered in the experimental analysis by a large amount of different test setups. Regarding the presented research campaigns, multiple test series consisting of small-scale tests like cone calorimeter [1], electric heater, radiant panel, intermediate scale tests such as the German “Brandschacht” test [2], the IEC 60332-3 [3] and large-scale tests of cable tray installations mounted in a fire compartment (see [4], [5]). The test results were used to develop a qualification method and licensing procedure for cable systems with protective coatings, based on realistic fire conditions, which is accepted by the building authorities for application in buildings in general and, in particular, in nuclear power plants. The procedure, consequences and definition of such a qualification method were published in [4], [5].

In this paper, the results are summarized and contemplated regarding the effects of the main parameters. Therefore, the results are investigated regarding the influence of the heat flux exposure, the packing density of cable trays, the material and geometry, the amount of cable trays mounted in a horizontal configuration, the effect of pre-heating and cable age on the heat release rate. A closer look at the effect of protection measures has been taken for cone calorimeter small-scale tests and large-scale tray tests, focussing on intumescent and ablation coatings. The burning behaviour of FRNC cables, as investigated in another extensive research project [6], is not discussed in this paper.

During the last 20 years, the growth in computational power has evolved in the development of CFD software capable of considering complex fuels and the resulting fires as source term of heat and smoke distribution in a compartment, e.g. FDS [7], or lumped parameter tools like COCOSYS [8] as a specialized software for nuclear power plants. The new possibilities of this calculation and design methods have been considered in guidelines of the International Atomic Energy Agency [9], [10], or nuclear fire protection standards, e.g. for German nuclear power plants [11], [12]. International research projects such as the ICFMP (*I*nternational *C*ollaborative *F*ire Modelling *P*roject, see [13], [14]), the OECD PRISME (*Pr*opagation d’un *i*ncendie pour des *s*cénarios *m*ulti-locaux *é*lémentaires) Project [15], [16], and the succeeding project OECD PRISME 2 [17] are carried out consisting of (amongst other) test campaigns specifically designed as validation basis for the aforementioned CFD software. Although cable trays and electrical installations were and are considered, some influence parameters like the effect of protection measures were not investigated to the extent needed for validation and further model development. By summarizing the results and findings of the test campaigns mentioned, conclusions are drawn and a proposal for further research goals is made in this paper.

BURNING BEHAVIOUR OF CABLES

Before ignition, burning and fire spreading, the first process is the heating of a cable and a following pyrolysis of burning gases. The heating can be caused by an adjacent fire incident, due to high power load or a damaged isolation, resulting in a short circuit. In case of a fire incident, the cables get exposed with heat fluxes from the flame, convective heat transfer (when mounted in the hot gas layer) or directly by the flame itself. This yields to a consequent heating of the cable sheath, which begins to soften (besides thermoset cable materials). In this phase, the copper conductors can move through the softened material, causing short circuits. The next phase is the thermally induced pyrolysis / degradation process of the cable and, with respect to the temperature and oxygen concentration, the ignition of the pyrolysis gases. For cables without fire retardant additives, this usually leads to a self-preserving process because the flame is heating up and igniting the surrounding cables.

Besides the cable materials and protection measures, the packing density of the cable package and the installation location (horizontal or vertical) plays an important role as an influencing factor for the burning behaviour. In combination with different protection coatings, this leads to rather complex burning processes. During the assessment of vertical cable trays, it was determined that the fire spreading over the tray is faster when compared to horizontal cable trays. For horizontal trays, a higher packing density, reached by a smaller tray with the same amount of cables, e.g. in the CHRISTIFIRE project [19], or by adding additional cable layers [18] showed that the maximum heat release rate is reduced up to 100 % and more.

Coated cables show a discontinuous flame spread because the pyrolysis gases can only escape through cracks in the coating. Coated cables usually burn at areas separated from each other. Cracks in the coating occur during the thermal expansion of the cable when the coating cannot take the tensile strains, which increase with increasing temperature. Intumescent coatings with an adequate thickness are capable of protecting themselves by closing the gaps with newly produced foam. Ablation coatings are more sensitive to cracking because they do not protect themselves. In general, the risk of cracking increases with higher heating rates and larger heat fluxes [18].

The aforementioned findings are based on extensive research mainly based on PVC cables without significant amounts of flame retardant compounds. The burning behaviour of FRNC cables is assessed and published in [6] and [19].

CURRENT PROTECTION MEASURES

Intumescent Coating

Painted intumescent coatings react to temperature exposure respectively heat fluxes with foaming and swelling, increasing the volume by magnitudes. This reaction can also be endothermic. The main protection mechanism is the foam, leading to thicknesses up to 100 x the original thickness. In case of a fire, the foam protects the cable sheath from high heat fluxes due to its low heat conductivity and isolates the whole cable against the hot environment. Due to this, the heating of the cable is slowed, the beginning of the pyrolysis is delayed and the pyrolysis rate is lowered in total. In addition, the foam prevents the pyrolysis gases from flowing into the combustion zone. As consequence, the ignition of the cables is delayed and the heat release rate is lowered.

Ablation Coating

Based on a different protection mechanism, the ablation coating systems divest energy due to an endothermic reaction, when heated. This reaction takes place until the whole ablation coating is consumed. During the reaction, even a cooling of the cables can be seen due to the strong endothermic reaction. After the consumption of the ablation material, a relatively thin material layer is left. In case of a fire, the heating of the protected cables is significantly delayed until all ablation coating material is consumed. After this point, the heating can be compared to an unprotected cable. The thickness of the remaining char (for the materials assessed) is usually too low to have a significant isolative effect.

Specific Measures at Connections and Mounting Points

When mounted on cable trays, the fixing points and mounting elements yield to gaps and insufficiently coated spots on the cable surface. To avoid fire spread at the mounting points, the construction is protected with glass fibre wires, which for themselves, are also protected with intumescent coating. These measures were taken for vertically mounted cables during the large-scale experiments [18].

In nuclear installations, water is used for decontamination measures after an incident. Some intumescent coatings get damaged when exposed to water. In this case, the coating itself has to be protected with an additional, waterproof coating which does not constrain the foam development in case of fire.

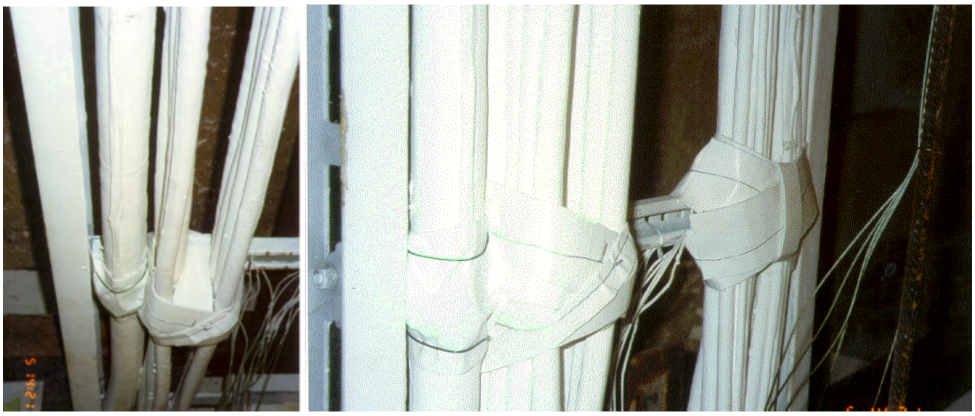

Figure 1 Protection of mounting points with glass fibre wires and intumescent coating

EXPERIMENTAL CAMPAIGN AND METHODOLOGY

Depending on the test setup and the constraints in parameter variation and boundary conditions, each test setup is only valid for specific situations and fire characteristics possible in a real installation situation. A typical example is the constant heat flux exposure in the cone calorimeter experiments which does not occur during a real fire incident. To get valid results for real fire incidents with an acceptable effort, the different small- and large-scale experiments are listed as follows:

- Cone calorimeter considering ISO 5660 [1],
- Electric heater,
- Large-scale tray experiments in a fire room with exhaust hood and connected gas analysis duct (O_2, CO, CO_2).

The electric heating furnace is shown in Figure 2. As a fourth test setup, a radiant panel apparatus was also used. Due to its low radiative heat flux exposure in standard setup, the fire spread was locally limited [18]. All following explanations and definitions are related to [18].

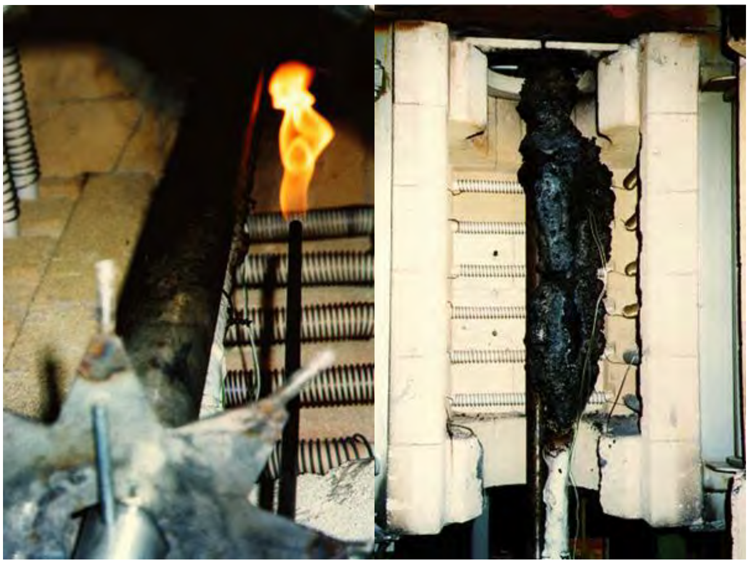

Figure 2 Electric heater with cable sample (intumescent coating) with ignition flame (left) and after the experiment (right) [20]

In Table 1, the settings, parameters and boundary conditions are listed for the test setups described above. The underlined text marks parameter variations or bandwidths of assessed parameters.

Table 1 Test matrix for the assessment of the burning behaviour of cables and cable trays

Tests / Parameter	Small-scale Tests		Large-scale Tests
	Cone Calorimeter	**Electric Heater**	**Vertical and Horizontal Tray Tests in a Fire Compartment**
Type/age of cable	6 new, 2 used	2 new	4 new. 1 used
Orientation of the cable	Horizontal	no variation	horizontal, vertical
Packing density	single cable, 3 cables, full (10 x 10 cm²)	single cable	one cable layer, two cable layer, three cable layer, control cables, power cables,
Number of cable trays and position to neighbouring trays	parameter varies considering practical issues	parameter varies considering practical issues	horizontal: 1, 2, 3 trays, vertical: single tray, 3 trays side by side, 3 trays in a row
Protective measures	intumescent coating, ablation coating	intumescent coating, ablation coating, combination of both	intumescent coating, ablation coating, tray bindings
Thermal exposure	radiative 10 - 60 kW/m²	pre-heated, max 400 °C	pre-heated, max. 450 °C
Ignition source	electric arc	Diffusion flame (methane)	diffusion flame (propane) using a sand burner with 50 - 180 kW, suspended cables, max. 300 kW
Ventilation	no variation	no variation	opening with 2.6 m²
Position in the fire room	parameter varies considering practical issues	parameter varies considering practical issues	horizontal: lower or middle position

Every test setup is based on a specific risk situation. In the large-scale room fire tests, a situation occurring in real fire incidents is simulated. A possible ignition source (cleaning materials, trash bins, wooden furniture or liquid pools is replaced by a sand burner which can be adjusted in power, location and (limited) in geometry, simulating the possible real fire source.

A fully developed fire adjacent to the trays is simulated by pre-heating the fire room (large-scale tests) or exposure with an appropriate heat flux (small-scale tests). The results of these tests can be directly transferred to practical applications if pre-heating or the heat flux exposure is linked to a real incident or ignition source.

CONE CALORIMETER AND TEST SETUP

In cone calorimeter tests, described in ISO 5660 [1], the sample is exposed with a defined heat flux. This leads to a heating of the sample surface and to the beginning of the degradation process and pyrolysis. Air and pyrolysis gases mix above the sample where it is ignited by an electric arc (called piloted ignition) or it ignites by itself. After this, the sample burns independently. Whether a cable sample ignites by itself or by piloted ignition depends, aside from their material composition, on the specific heat flux exposure in the test. A low specific heat flux where the material does not ignite is called the *minimum heat flux*. The ignition time as function of the specific heat flux provides information on the burning and ignition behaviour of cable installations, together with the measured heat release rate of the sample.

In the cone calorimeter, the maximum sample area is 100 cm² at a maximum height of 5 cm, which can be tested in horizontal and vertical installation. The specific heat flux ranges from 0 – 100 kW/m² and is generated by the conical heater. The cone calorimeter test setup is shown in Figure 3.

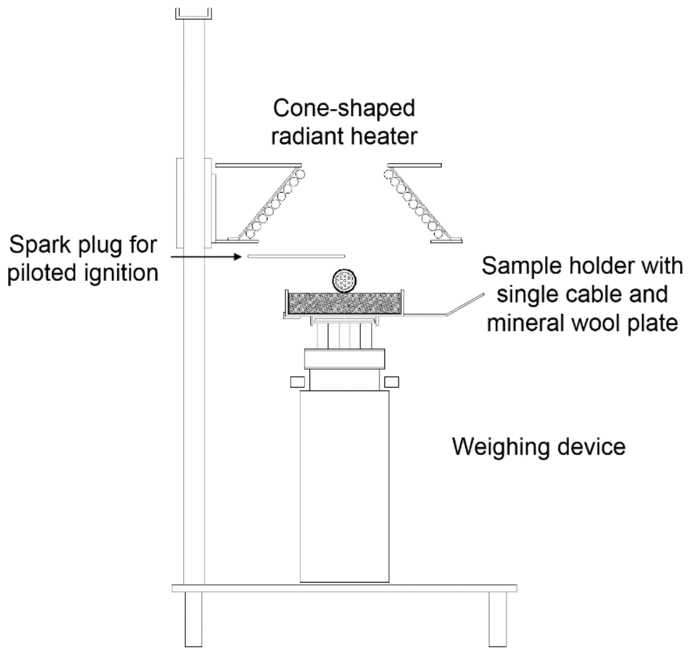

Figure 3 Cone calorimeter test with cable sample

The samples are placed in the specimen or sample holder which is mounted at a weighing device to measure the mass loss. The entire pyrolysis and burning gases are discharged through an exhaust gas system with flow measuring instrumentation and an installed gas sampling apparatus, recording the exhaust volume flow, the oxygen, carbon monoxide and carbon dioxide volume fraction. Besides the mass loss and the mass loss rate, the heat release rate can be calculated based on the oxygen consumption. The ratio of mass loss rate and heat release rate is the effective heat of combustion. Additionally, carbon monoxide, carbon dioxide and soot yields can be derived from the cone calorimeter tests.

To ensure unique boundary conditions, the specimen holder is protected with aluminium foil and a mineral wool plate to isolate the cable sample from the specimen holder. The aluminium foil prevents intrusion of melted polymer into the mineral wool plate. Before the tests, the gas sample apparatus is calibrated with a methane burner at a maximum power of 10 kW.

During calibration, the methane burner is installed instead of the specimen holder and the conical heater is not running.

ELECTRIC HEATER

The electric heater, shown in Figure 2, was developed at the iBMB. It simulates the boundary conditions of a pre-heated environment, which are defined using a specific transient heating curve. The ignition flame with a power of 1 kW is mounted at the bottom of the apparatus and is leading to a local sample exposure. In contrary to the cone calorimeter (ignition spark), the flame heat release is not negligible. The main outcome of those tests is information regarding the effect and degree of protection of the cable coating measures.

ROOM FIRE TEST SETUP

The room fire tests were conducted in a fire chamber with a base area of 3.6 x 3.6 m². Depending on the tray installation (horizontal or vertical), the room has a height of 3.6 m or 5.6 m. When needed, the room was heated with three oil burners mounted in the floor of the chamber. The ignition of the trays was done using a propane burner with a surface of 30 cm x 30 cm and a maximum power of 300 kW. The trays were mounted on a rack which itself was installed on three weighing devices. The gas exchange was assured by an opening with a width of 1 m and a height of 3.6 m, where the burning and smoke gases were discharged and delivered through the exhaust system to the gas analysis equipment which was installed over the exhaust line. The recorded quantities of the tests were the mass loss rate, the heat release rate via oxygen consumption method, the effective heat of combustion and the carbon monoxide and carbon dioxide yields. An overview of both test setups is given in Figure 4.

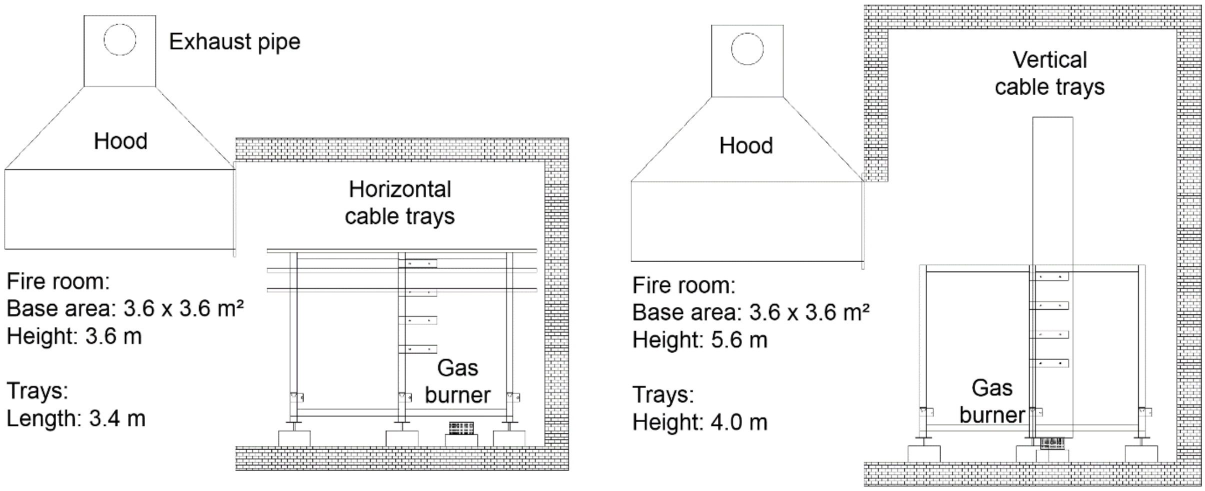

Figure 4 Test setup for the room fire tests with horizontal (left) and vertical (right) tray installations for unprotected trays

Before the actual test, the setup was calibrated using the gas burner, mounted directly under the hood. For pre-heating, applied in the tests of the coating measures, the oil burners were activated for 20 min to generate a room temperature of 175 – 350 °C (see Table 1 for details). The transient temperature curve was derived from the “smouldering curve” of [21]. The cable trays were separated from the oil burners using an insulation wall with a height of 1.4 m. Details of the test setup for protected cable trays are depicted in Figure 5.

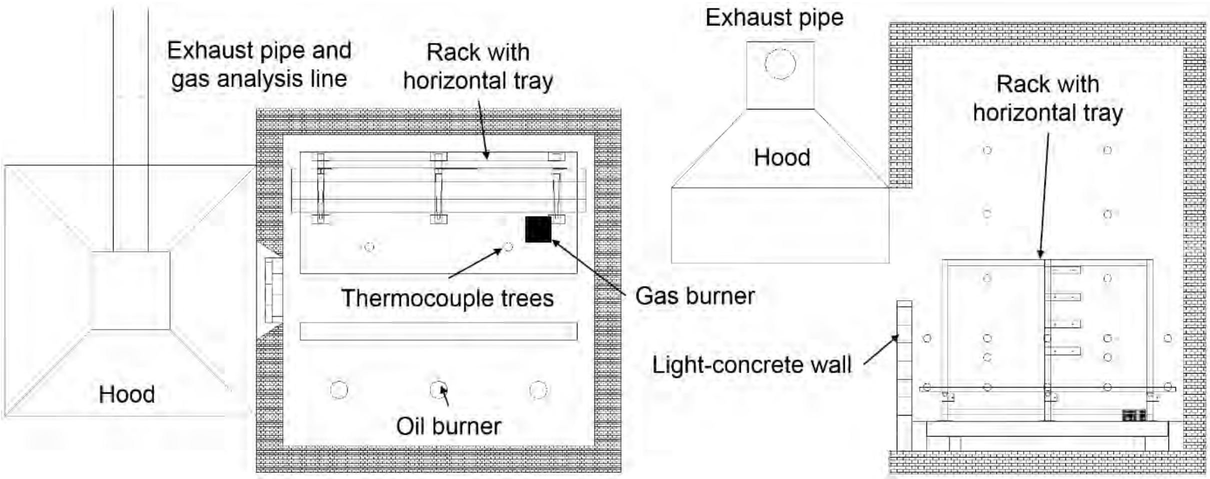

Figure 5 Test setup for the room fire tests with horizontal tray installations for protected cable trays

The thermocouple trees were installed in all tests and used to determine the hot gas layer and the general temperature development in the fire room. To prevent the heat from escaping the fire room, the opening area to the hood was decreased using a light-weight concrete wall with a height of 1.0 m. The installation height and the number of horizontal trays mounted at the rack depend on the specific tests. Figure 4 shows the location and number exemplarily for three unprotected cable trays in the hot gas layer, whereas Figure 5 shows a single cable tray was mounted about 20 cm above the gas burner.

All horizontal trays had a length of 3.4 m and were equipped with different PVC cables, with type and packing density depending on the actual test. The unprotected cable trays used one (see Figure 6a*),* 24 cables), two (Figure 6b*),* 48 cables) or three layers (Figure 6c*),* 72 cables) of the control cable JE-Y(St)-Y 40*2*0.8 mm with a diameter of 8 mm. The unprotected trays had a width of 45 cm.

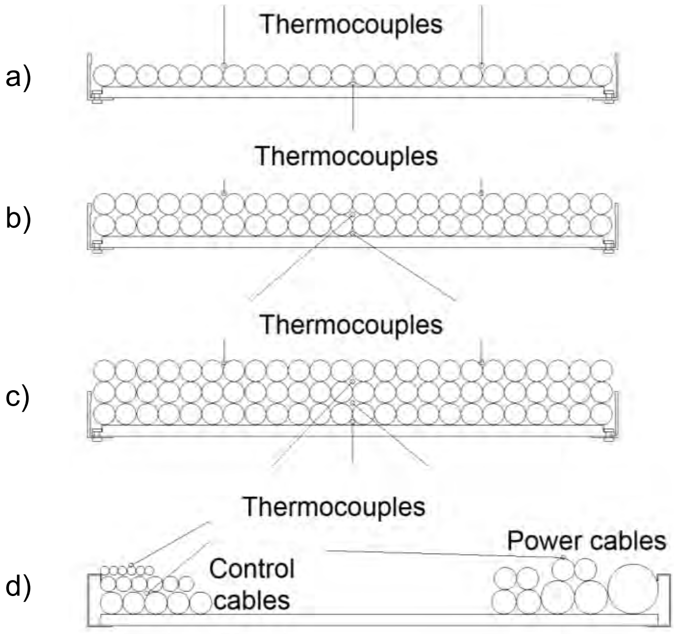

Figure 6 Cable installations on the horizontal cable trays used in the fire room experiments

The trays assessed for the analysis of the protection measures were equipped with seven different cable types, all containing PVC. They were equipped as outlined in Figure 6d). On the side facing the wall of the fire room, four different types of power cables with a diameter from 20 mm to 45 mm were located, whereas the control cables face into the fire room. This tray load was used as a standard load in most of the experiments with ablation and intumescent coatings. More details regarding the cables are published in [18]. The results of the analysis of vertical tray installations are also published in [18].

SMALL-SCALE TEST RESULTS

Cone Calorimeter

Cone calorimeter results were conducted for several cable types, cable ages, cable amounts and heat fluxes ranging from 15 to 60 kW/m². The effect of intumescent coating on the ignition time and the heat release rate is depicted in Figure 7 with dotted lines and compared to the unprotected cable (continuous line). It can be shown that the ignition is delayed by at least 450 s. The maximum heat release rate is reduced from 12 kW/m to about 3 kW/m. The results are representative for other intumescent coatings investigated with the cone calorimeter.

The other protection coating is the ablation coating, based on an endothermic reaction which consumes energy when heated. The effect of the ablation coating was investigated for one mixture and is shown in Figure 7 as dashed lines. The type of cable used in the experiments is the same as for the intumescent coating. The ablation coating leads to a delay in ignition of about 90 – 120 s. The maximum heat release rate is lowered from 12 kW/m to about 8 kW/m. All three repetition tests show a similar behaviour.

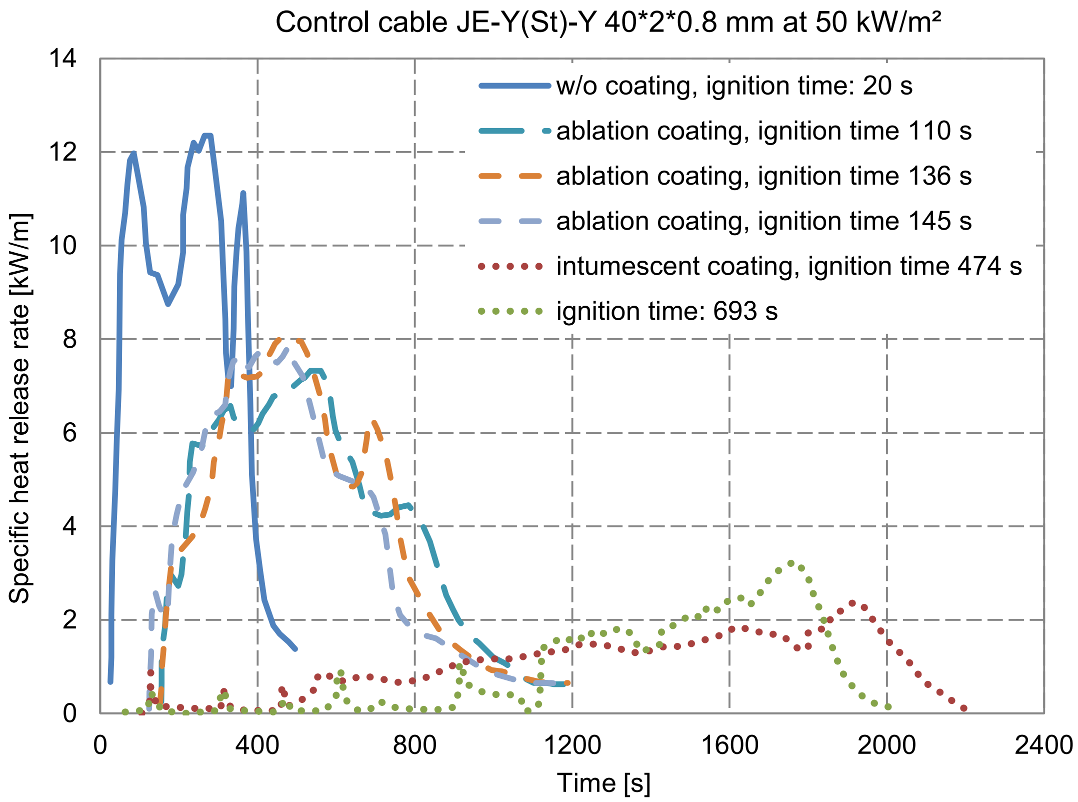

Figure 7 Specific heat release rate and ignition times measured for protected and unprotected cables type JE-Y(St)-Y 40*2*0.8 mm at 50 kW/m²

One of the cables ignited before the intumescent coating started to foam, under the same boundary conditions as the other test specimens. In this case, the intumescent coating itself started to burn after 50 – 80 s and ignited the cable surface. After foaming of the coating, the specimen behaved like the other coated samples.

To investigate this behaviour, a layer of equivalent thickness as put on the cable was laid on aluminium foil. At a heat flux of 50, resp. 75 kW/m², an ignition after 20 s, resp. 11 s at burning durations of 30 s resp. 40 s were recorded. A maximum specific heat release rate of 115 resp. 155 kW/m² was measured.

Cable Age

To check whether the age of the cable has an effect on the burning behaviour, the cable type JE-Y(ST)Y 4*2*0,8 mm, a control cable, was investigated in the cone calorimeter for heat fluxes of 20 resp. 40 kW/m² and a different cable amount (3 cables / 8 cables in the cone Calorimeter sample holder). The exact age of the *old* cables is unknown, but as they were used ones delivered from power plant operators, it can be assumed that all volatile components were outgassed at the time they were tested.

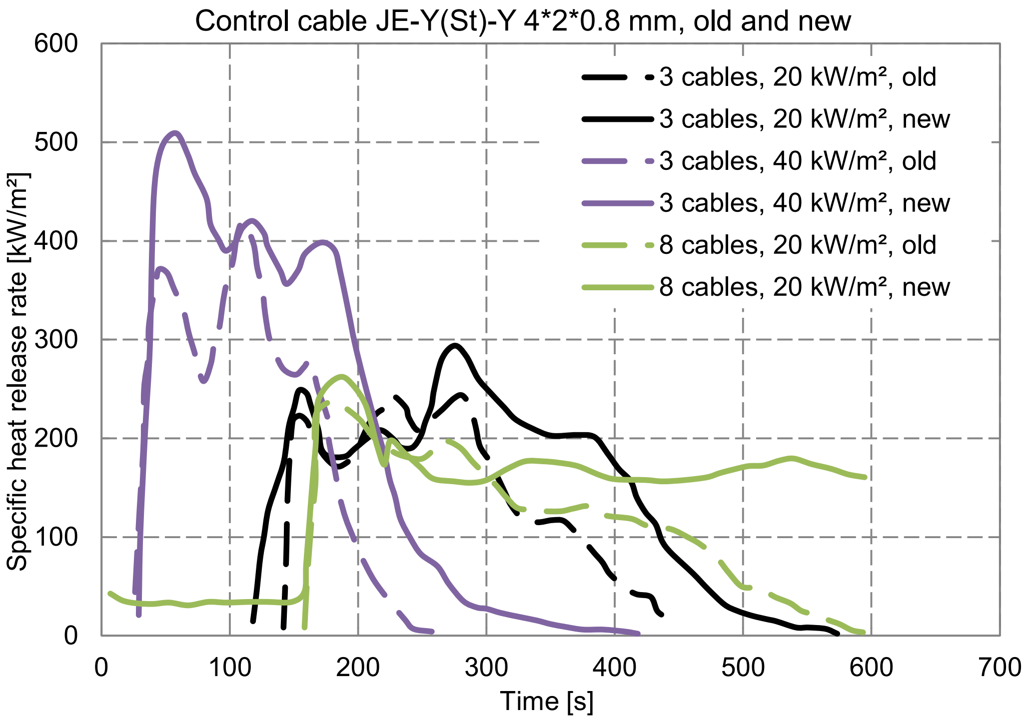

Figure 8 Specific heat release rate and measured for *new* and *old* control cables type JE-Y(St)-Y 4*2*0.8 mm at heat fluxes and different packing density

The difference between the old and the new cables can be seen in Figure 8 resulting in a generally higher maximum and average heat release rate and a longer burning duration of the new cable. One explanation might be a different PVC composition due to changes in the production. More likely is the assumption that the volatile softener in sheath and isolation is outgassing over the time. As the softener usually is one of the burning components in the materials, lower softener fraction due to the cable age results in a lower heat release rate.

Electric Heater

The heat release rate measured with the electric heater depends on the location of the ignition flame. When the cable is directly exposed by the flame (distance 1 cm away from the cable surface), it ignites after about 300 s, whether a distance of 2 cm delays the ignition to 1000 s or about 16 min. As shown in Figure 9, the flame spread of the unprotected cable starts abrupt with burning over the whole cable surface. This is plausible as the heater is controlled in three zones, leading to a homogenous temperature distribution over the height. Surface temperatures that were also recorded at three points over the cable length (see [18]) show that there is no significant difference in temperature.

The investigated cable samples had a length of 1.2 m. The furnace was pre-heated up to 400 °C (see Table 1); results are provided for 300 °C furnace temperature in Figure 9. Both protection measures are tested with a distance between methane flame and sample of about 2 cm.

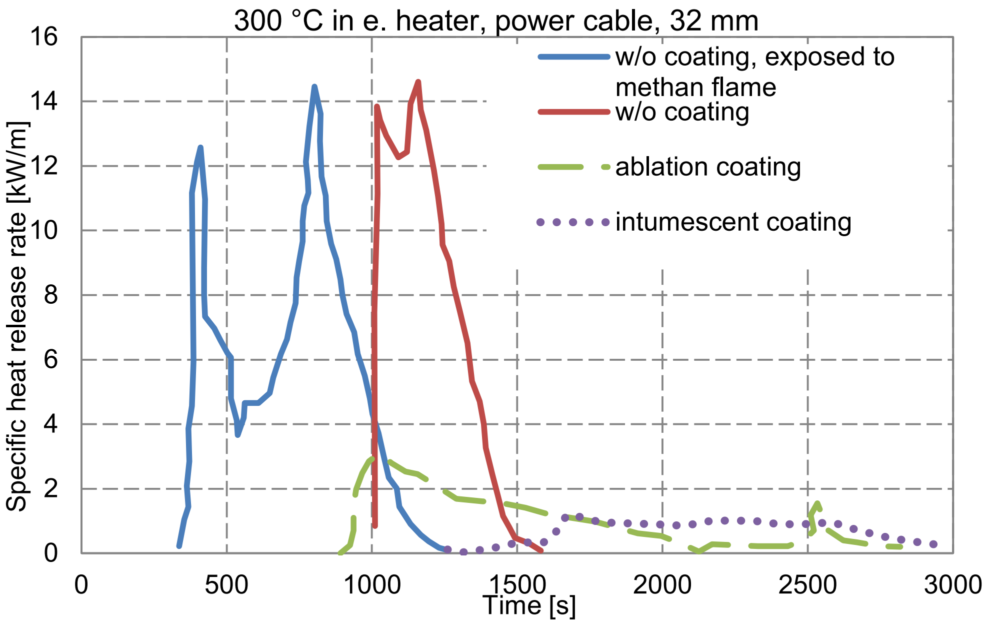

Figure 9 Specific heat release rate and measured for protected and unprotected power cable with 32 mm diameter at 300 °C in the electric heater

Figure 9 shows that both types of protection measures, the ablation coating and the intumescent coating are capable of reducing the heat release rate significantly. In contrast to the cone calorimeter results, the ignition time is not delayed in the same magnitude, and in the case of ablation coating, no delay in ignition was recorded. Like for the cone calorimeter, the results are valid in the narrow range of boundary conditions and constraints existing due to the test setup. Because of this, large-scale tests on real tray configurations were scheduled, too.

LARGE-SCALE TEST RESULTS

The test matrix of the large-scale tray tests investigating the behaviour of unprotected trays equipped with PVC cables is listed in Table 2. In all tests, the trays were mounted in the upper part of the rack in the hot gas layer (see Figure 4 as an example). The two main parameters varied in the test series were the number of cable trays and the amount of cable layers per tray like depicted in Figure 6. The second parameter increases the fuel load per tray, but on the other hand reduces the surface to volume ratio of the cable package as, e.g. for the tray with three cable layers, the second layer is protected by both upper and lower adjacent layers.

Table 2 Test matrix of the large-scale tray tests with unprotected cables and main results (yields as average values, data from [18])

	1 Cable Layer		2 Cable Layers				3 Cable Layers	
1 Tray	**Test 3**		**Test 4**					
	HRR_{max}	275 kW	HRR_{max}	230 kW				
	t_{max}	210 s	t_{max}	240 s				
	EHC	10 kJ/g	EHC	10.5 kJ/g				
	$Y_{CO2,Av}$	0.6 kg/kg	$Y_{CO2,Av}$	0.65 kg/kg				
	$Y_{CO,Av}$	0.05 kg/kg	$Y_{CO,Av}$	0.06 kg/kg				
2 Trays	**Test 5**		**Test 6**		**Test 9**		**Test 10**	
	HRR_{max}	440 kW	HRR_{max}	275 kW	HRR_{max}	430, 250 kW	HRR_{max}	380 kW
	t_{max}	660 s	t_{max}	300, 3150 s	t_{max}	360, 2400 s	t_{max}	160 s
	EHC	11 kJ/g	EHC	10 kJ/g	EHC	8 kJ/g	EHC	6.5 kJ/g
	$Y_{CO2,Av}$	0.7 kg/kg	$Y_{CO2,Av}$	0.75 kg/kg	$Y_{CO2,Av}$	0.33 kg/kg	$Y_{CO2,Av}$	(0.3 kg/kg)
	$Y_{CO,Av}$	0.09 kg/kg	$Y_{CO,Av}$	0.95 kg/kg	$Y_{CO,Av}$	0.1 kg/kg	$Y_{CO,Av}$	(0.03 kg/kg)
							Test 10a	
							HRR_{max}	320 kW
							t_{max}	2050 s
							EHC	11 kJ/g
							$Y_{CO2,Av}$	0.5 kg/kg
							$Y_{CO,Av}$	0.13 kg/kg
3 Trays	**Test 11**		**Test 7**		**Test 8**			
	HRR_{max}	880 kW	HRR_{max}	700 kW	HRR_{max}	480 kW		
	t_{max}	1020 s	t_{max}	1650 s	t_{max}	360 s		
	EHC	14 kJ/g	EHC	11.5 kJ/g	EHC	9 kJ/g		
	$Y_{CO2,Av}$	0.85 kg/kg	$Y_{CO2,Av}$	0.7 kg/kg	$Y_{CO2,Av}$	0.13 kg/kg		
	$Y_{CO,Av}$	0.16 kg/kg	$Y_{CO,Av}$	0.11 kg/kg	$Y_{CO,Av}$	0.05 kg/kg		
					Test 8a			
					HRR_{max}	800 kW		
					t_{max}	1690 s		
					EHC	9.5 kJ/g		
					$Y_{CO2,Av}$	0.7 kg/kg		
					$Y_{CO,Av}$	0.12 kg/kg		

It has to be kept in mind that this approach is different from the one taken in the CHRISTIFIRE project [19] where the packing density is increased by decreasing the tray width. The effect on the heat release rate and the fire spread is comparable, but, on the other hand, the total amount of combustible mass is kept constant.

The influence of the packing density is depicted in Figure 10, showing the results of the heat release rate of the tests number 5 (1 cable layer), 9 (2 cable layer), 10 (3 cable layer) in Figure 11a and the results of the tests number 11 (1 cable layer), 7 (2 cable layer), 8 (3 cable layer) in Figure 10b. One layer, i.e. 24 cables per tray, leads to the highest maximum heat release rate, in tests with 2 trays installed as well as in tests with 3 trays. One reason is the flame spread, which does not stop at the ends of the tray, resulting in a larger burning surface area. The fire spread velocity is also slower for trays with more than one cable layer. The effect of the number of trays is shown in Figure 10, with the combination of tests number 3 (1 tray), 5 (2 trays), 11 (3 trays) in Figure 10c, always equipped with one cable layer, and tests number 4 (1 tray), 9 (2 trays), 7 (3 trays) in Figure 10d, always equipped with two cable layers. As expected, more trays lead to a higher heat release rate in maximum.

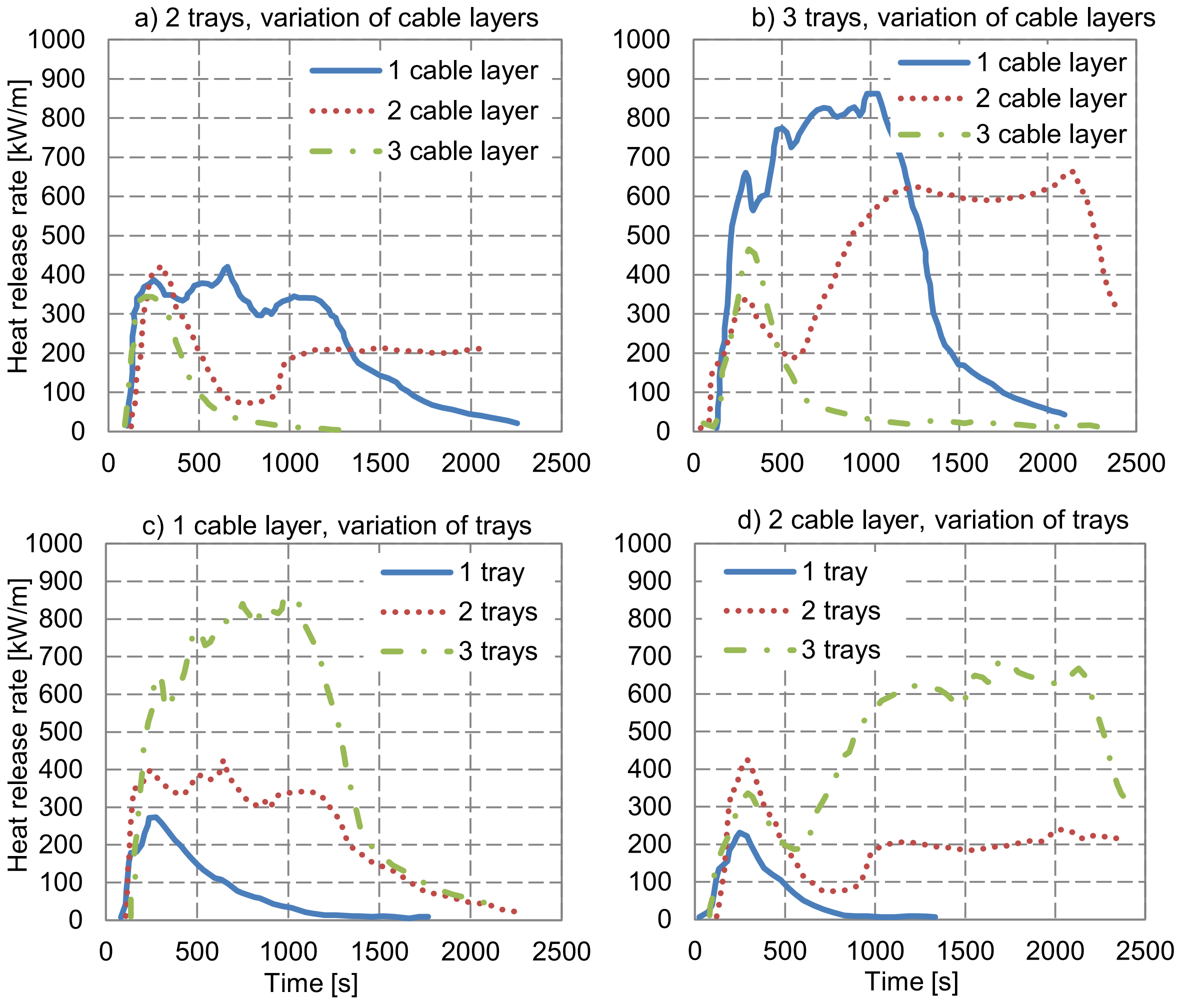

Figure 10 Heat release rate measured in large-scale room fire tests for trays located at the upper part of the rack (hot gas layer)

For two cable layers on each tray, the influence of the packing density can be seen, as the comparison of tests 5, 9 and 10 in Figure 10a shows. On the upper surface of the lowest tray, ignition starts at the same time like seen in tests with a single tray. The ignition of the upper surface of the second tray is slightly delayed (the same for test 10). If both trays are equipped with one cable layer, the fire spreads faster on the first tray than on the ones above. In a second phase, the fire spread on the first tray slows down, whereas on the trays above, the fire spreads continuously (see [18] for more details). For trays with two cable layers, the fire spread is slower and is slowed down on the second tray too, when the fire spread on the first tray starts to decrease. This behaviour can also be seen in case of three cable trays (Figure 11d). After a decline in heat release rate, (1000 s resp. 600 s for three trays), the second (and third) tray starts to burn by itself.

As a result, a local ignition of the PVC cable trays cannot be excluded in general. An increase of the packing density (in this case this means also an increase of the number of cables on the tray, see Figure 6 for tray configuration) delays the ignition time slightly and decreases the heat release rate considerably. Besides the case of self-extinction, a higher packing density leads to a longer burning duration.

The increase of the number of trays does not influence the ignition, but leads to a faster fire spread, higher heat release rates and consequently to shorter burning durations. On single trays, the fire spread is limited on the middle part of the tray.

Summarizing the results, a single cable tray is not critical, because the fire spread stops in all tests by itself. A higher packing density also has the effect of lowering the heat release rate. A condition for this is the justified installation on the tray, leaving no spaces between each cable and ensuring a positive (i.e. low) surface to volume ratio.

Protected Cable Trays

In the context of trays protected with intumescent coating, the test setup was a different one (see Figure 5), considering another fire scenario. It is supposed that a fully developed fire is located in an adjacent location in the room where the cable trays are mounted, leading to a pre-heating of the cables. Later, the fire has spread, with the flames reaching the cable trays and igniting them. At this time, the trays participate in the whole fire incident as secondary fuel load.

The results are summarized in Table 3 and Table 4. Although not taken into consideration in detail, the results for vertical tray installations are also listed in the tables. More information on the used cables and the vertical tray configuration can be seen in [18].

Table 3 Ignition and fire spread of different unprotected tray configurations

Configuration	Pre-heating Temperature	Ignition Power	Observation Concerning Fire Spreading	
			Ignition	Fire Spreading
Horizontal grid	25 °C	150 kW	yes	Locally restricted
Horizontal trays	175 °C	50 kW	yes	locally restricted
	200 °C		3 - 4 min	3 - 5 cm/min
	350 °C		< 1 min	110 - 120 cm/min
Vertical trays	25 °C	3 kW	1,5, 12 min	locally restricted
		50 kW	0,5, 8 min	16 - 30, 60 - 240 cm/min
	250 °C		45 s	130 - 160 cm/min
	300 °C		40 s	360 - 480 cm/min

Table 4 Ignition and fire spread of different tray configurations with intumescent coating

Configuration	Pre-heating Temperature	Ignition Power	Observations Concerning Fire Spreading	
			Ignition	Fire spreading
Horizontal grid	25 °C	180 kW	> 40 min	locally restricted burning, self-extinguishing
Horizontal trays	< 250 °C	50 kW	no	no
	250 - 350 °C		> 30 min	after 45 min
	> 350 °C		> 30 min	after 45 min
Vertical trays	< 250 °C		yes	locally restricted
	> 250 ° C		yes	80 cm / min *
Vertical trays, with protected mounting elements	> 350 °C		> 30 min	after 40 min

* The fire spread velocity given is related to a „jumping“ of the flames over each mounting element-

The test setup is already explained above and shortly detailed in Table 1. For the vertical trays, the protection of the mounting elements as shown in Figure 1 was necessary to avoid a "jumping" of the flames over these points with lower coating thickness. The fire spread from an unprotected to an adjacent protected horizontal tray was prevented.

INTERNATIONAL RESULTS

Besides the presented results, investigations on cable fires have been carried out in the frame of different international research projects. The ICFMP focused on the validation of fire simulation codes, with a major objective on cable fires. This was done within so-called benchmark exercises, with No. 3 and No. 5 dealing with cable trays and cable fires. Exercise No. 3 focused on the temperature development of cable trays equipped with different power and control cables and was specifically designed as a validation exercise. The experimental part of Exercise No. 5 contains of four large-scale experiments, investigating the burning behaviour of pre-heated and non-pre-heated vertical cable trays, equipped with different control and power cables (FRNC, PVC). Details can be found in [13] and [14].

Results of the international OECD PRISME Project (PRISME is the French acronym for "Fire Propagation in Elementary Multi-Room Scenarios") were published in [15] and [16]. In the PRISME project, a multi-room large-scale test, using a vertical tray installation equipped with PVC cables as fuel, was carried out in 2011. Before this, the different cables were analysed in a large-scale calorimeter in open atmosphere. The vertical tray consists of four cable bundles with eight cables each line and isolated side-walls to investigate a probable chimney effect. The results of the confined atmosphere in the multi-room fire test have shown that the burning and fire spread were weaker than observed in the open atmosphere large-scale calorimeter tests using the same cables and trays. The effect of this is depicted in Figure 11.

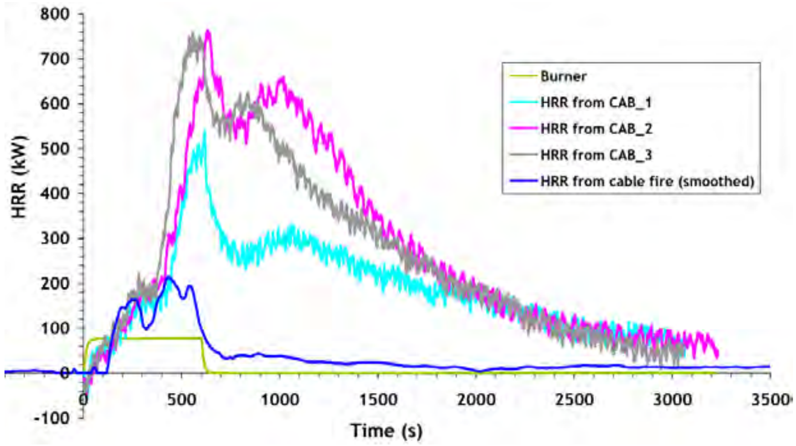

Figure 11 Heat release rate measured in large-scale multi-room fire tests and large-scale calorimeter tests at the IRSN DIVA facility [22]

Another research project with focus on the burning behaviour of cables, cable trays and tray installations was the CHRISTIFIRE project, an acronym for the term "*C*able Heat *R*elease, *I*gnition, and *S*pread in *T*ray *I*nstallations during *FIRE*". During this multi-year project, an extensive amount of small-scale tests, e.g. micro calorimetry, tube furnace measurements, cone calorimeter and radiant panel were carried out. Finally, 26 large-scale tests investigating different multiple tray configurations were realized. Overall, about 36 cables were analysed.

One goal was the systematic investigation of the influence of different heat flux expositions on the resulting specific heat release rate. The results confirm the older results from [18] and have shown that lower heat flux exposition leads to a lower heat release rate and a longer burning duration (see [19] as well).

The extensive tray test series were carried out to investigate the influence when changing the vertical distance of the trays, the number of trays, the amount of cables and the tray width. Exemplarily, the influence of doubling the tray width while keeping the amount (number) of cables constant, is shown using for the heat release rate depicted in Figure 12.

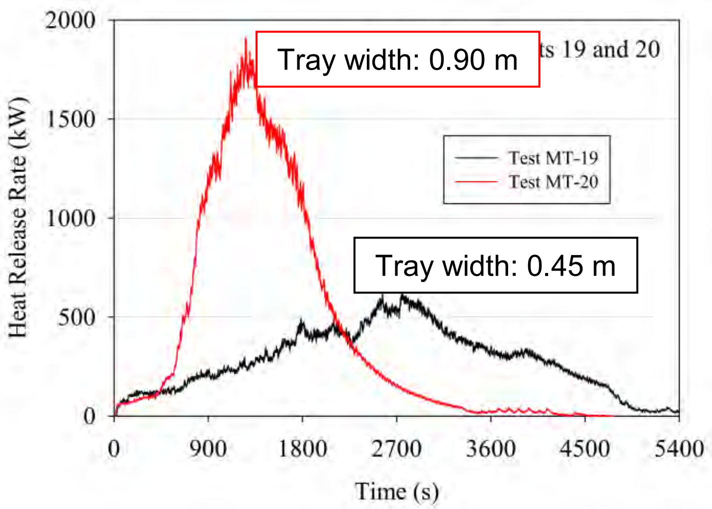

Figure 12 Heat release rate measured in tray tests varying the tray width [19]

The cables in the CHRISTIFIRE tests were loosely installed on the trays, so they cannot directly be compared to the tray experiments presented above.

INFLUENCE FACTORS ON THE BURNING BEHAVIOUR OF CABLES AND CABLE TRAYS

To summarize the results presented in the sections above, the following statements can be referred to as a conclusion of the small-scale and the tray experiments conducted at the iBMB as well as of the international projects. The influence of protection measures on the development of the heat release is omitted here and explained in detail in section "Cone Calorimeter" and "Large-scale Tray Tests".

Heat Flux Exposure

An increasing heat flux decreases the ignition time, as measured in [18] as well as in [19]. On the other hand, the maximum and the average heat release rate increase. High heat fluxes lead to a distinctive heat release curve which allows determining the degradation reactions of the sheath, filler and isolation due to the peaks in heat release rate.

Packing Density / Number of Cables

In the cone calorimeter, a single cable shows a more specific heat release curve, where a larger amount of cables in the specimen holder yield in a time-averaged behaviour. Maximum and average heat release rate decreases with higher packing density. The same is visible for cable trays. A higher packing density leads to a lower maximum and average heat release rate, and consequently to a longer burning duration. Regarding multiple horizontal trays, higher packing density results in weaker fire spread from one tray to the next tray above.

Cable Materials and Geometry

Investigations in [18] have shown no significant correlation between the diameter of the cable and the heat release rate. This is due to the fact that other parameters like the thickness of the sheath, the filler materials, insulations and other material parameters have an influence on the burning behaviour, but are uncorrelated with the diameter. As expected, the material type of the cable components has an influence on the heat release rate, as published in [19].

Amount of Cable Trays

The amount of cable trays mounted in a horizontal configuration leads to a higher maximum heat release rate at approximately constant burning duration. Significant differences can be seen when only one tray is installed. In this case, the fire spread velocity is slower and in most cases (depending on the packing density), the fire spread stops after a certain distance, leaving unburned cables at both ends of the tray.

Pre-heating of the Fire Room and Cables

The pre-heating of the fire room and consequently, the cables has a significant effect on the fire spread velocity and the heat release rate, as shown in the ICFMP Benchmark Exercise No. 5 test series [14] and the test series conducted at the iBMB on unprotected and protected cable trays as well as in the electric heater test investigating a single cable [18]. Due to the decrease of the temperature difference of the cable and its degradation temperature, emitting pyrolysis gases, a weaker burning and flame process is sufficient to ignite the adjacent parts of the cables.

Cable Age

As shown in Figure 8, older cables burn at a slightly lower but significantly decreased maximum and average heat release rate. Like mentioned before, the escape of volatile softeners from the PVC sheath and insulation material is regarded as main reason for this behaviour. Cables with other sheath and insulation materials than PVC were not investigated.

CONLCUSIONS AND OUTLOOK

Having a look at the effect of the coating measures, the intumescent coating has a large impact on the resulting heat release rate measured in the Cone Calorimeter (see Figure 7). The results of the tests investigating protected trays with intumescent coating and their unprotected counterparts are listed in Table 3 (unprotected trays) and Table 4 (protected trays). Referring to the test with a pre-heating temperature of 350 °C and a 50 kW ignition source for both tests, the unprotected tray ignited in < 1 min, whereas the protected tray needed more than 30 min to ignite. The flame spread velocity for the unprotected tray was measured with 110 – 120 cm/min while the fire spread on the protected tray was continual and locally limited. The same magnitude of the difference in ignition time was measured for vertical trays protected with intumescent coating, especially for the protected trays with additional protection measures for the mounting elements. Again listed in Table 3 and Table 4, the unprotected cable tray ignited after 40 s compared to an ignition time of > 30 min for the protected trays, although the pre-heating temperature was lower in the unprotected tests (300 °C against 350 °C). The test results for the unprotected vertical tray have shown high flame spread velocities of 360 – 480 cm/min.

Trays protected with ablation coating were also analysed under the same test and boundary conditions. Looking at pre-heating temperatures up to 400 °C, the ignition was delayed by about 40 min. After this, the fire starts to spread faster than for cables protected with intumescent coating. On the other hand, vertical trays protected with ablation coating did not need extra protection of the mounting elements like it was needed for trays with intumescent coating.

Vertical tray experiments with protected trays using intumescent coating revealed weak spots at the construction / mounting elements, where the decreased thickness of the coating layer yields to an observed “jumping” of the flames at these points. Trays protected with ablation coating did not show that results (see [18]).

The extensive small-scale and large-scale tests have also shown that, besides the protection measures, a lot of different influence factors exist which determine the burning behaviour of cable trays, which are one of the most significant fire sources in nuclear power plants. As shown for the tray configurations with high packing density, it can be stated that the risk of fire spread for trays equipped with PVC cables which fail to comply with the IEC 60332-3 test [3] is reduced. This is not valid for installations with more than one tray, as the results presented here have shown that the trays mounted above burned over the whole surface area.

The presented national test campaign focused on the development of a qualification and licensing procedure for cable systems. Today, the progress in computational power and modelling allows us to consider cable fuels as a fire source in CFD simulations. Regarding the results presented in this paper, a closer look on protection measures like intumescent and ablation coatings might be useful. Current fire simulation codes such FDS [7] are generally capable of considering both protection mechanisms of ablation and intumescent coatings in their fundamental equations. By specifically analysing these parameters in combination with other important factors like the packing density and the pre-heating of the surrounding environment, existing models can be validated and improved regarding their prediction capabilities when protection measures on cable trays and installations are investigated.

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FULLY PREDICTIVE SIMULATION OF REAL-SCALE CABLE TRAY FIRE BASED ON SMALL-SCALE LABORATORY EXPERIMENTS

Tarek Beji[1], Frederick Bonte[2], Bart Merci[1]

[1] *Department of Flow, Heat and Combustion Mechanics, Ghent University, Sint-Pietersnieuwstraat 41, 9000 Ghent, Belgium*

[2] *Bel V, Rue Walcourt 148, 1070 Brussels, Belgium*

ABSTRACT

This paper presents a computational fluid dynamics (CFD)-based modelling strategy for real-scale cable tray fires. The challenge was to perform fully predictive simulations (that could be called 'blind' simulations) using solely information from laboratory-scale experiments, in addition to the geometrical arrangement of the cables. The results of the latter experiments were used (1) to construct the fuel molecule and the chemical reaction for combustion, and (2) to estimate the overall pyrolysis and burning behaviour. More particularly, the strategy regarding the second point consists of adopting a surface-based pyrolysis model. Since the burning behaviour of each cable could not be tracked individually (due to computational constraints), 'groups' of cables were modelled with an overall cable surface area equal to the actual value. The results obtained for one large-scale test (a stack of five horizontal trays) are quite encouraging, especially for the peak Heat Release Rate (HRR) that was predicted with a relative deviation of 3 %. The time to reach the peak is however overestimated by 4.7 min (i.e. 94 %). Also, the fire duration is overestimated by 5 min (i.e. 24 %). These discrepancies are mainly attributed to differences in the HRRPUA (*h*eat *r*elease *r*ate *p*er *u*nit *a*rea) profiles between the small-scale and large-scale. The latter was calculated by estimating the burning area of cables using video fire analysis (VFA).

INTRODUCTION

The work presented in this paper has been performed in the frame of two large international collaborative research projects of the OECD (*O*rganisation for *E*conomic *Co*-operation and *D*evelopment) Nuclear Energy Agency (NEA) called PRISME and PRISME 2 [1], [2]. The general objective of these projects is to deepen the understanding of fire dynamics in nuclear power plants (NPPs).

Fire dynamics in NPPs is mainly driven by a strong interaction between the fire and a mechanical ventilation network that connects several well confined rooms. Therefore, within the PRISME and PRISME 2 framework, a large-scale concrete facility, called DIVA, has been built at the French IRSN (*I*nstitut de *R*adioprotection et de *S*ûreté *N*ucléaire) in order to perform an extensive number of experiments [1], [2]. In the fire scenarios investigated, several parameters are examined: (1) the nature of the fire source (e.g. liquid pools, cables, and electrical cabinets), (2) the geometrical structure in terms of number of rooms involved, (3) the mechanical ventilation (e.g. volume flow rates, number of fans, positioning and effect of fire dampers), and (4) of fire extinguishing systems.

The variety of fire scenarios examined has led to several experimental campaigns, each focussing on a specific aspect. For instance, in PRISME 2 [2], a campaign called CFS (*C*able *F*ire *S*pread) has been dedicated to cables, since they are the most important combustible load in nuclear facilities. Such campaign aims first at characterizing the full-scale burning of cable trays (with several types of cables) in open atmosphere conditions before investigating compartment fire dynamics.

The series of tests having performed in open atmosphere conditions is called CFSS (*C*able *F*ire *S*pread *S*upport) [3]. In this paper, one of these tests is considered; it is referred to as CFSS1. The cables in CFSS1 were provided by one of the partners in PRISME 2, namely VTT (Finland), who conducted small-scale laboratory experiments on the same type of cables [4]. The main objective of the work described here is to challenge the current computational fluid dynamics (CFD) capabilities in predicting the large scale behaviour of cable fires based solely on information from small-scale tests. Due to the complexity of the problem and despite the current advances in the modelling as well as the computational capabilities, the reliability of such approach for the time being remains questionable. That is why, in practice, design calculations for CFS in NPPs are based on a simplified modelling approach (e.g. FLASH-CAT [5]) where several input parameters are provided from the results of a series of large-scale experiments called CHRISTIFIRE tests carried out in the United States. However, such an approach remains limited by the finite number of large-scale experiments that could be carried out, and that do not reflect necessarily real conditions (e.g. interaction with a mechanical ventilation system).

Therefore, despite the increased number of experiments and the efforts in their analysis (for example through Video Fire Analysis (VFA) [6]), we argue that CFD in conjunction with more affordable small-scale tests can be, in the long-term, a viable option for the analysis of CFS in NPPs.

Predictive calculations of cable tray fires remain very scarce. In 1979, Hunter [7] developed a set of models that capture the behaviour of horizontal cable trays exposed to a fire plume. In these models, the ignition of cables is associated to a 'critical' surface temperature as well as the local oxygen level. A similar approach has been adopted in [8] and implemented in a CFD-based methodology where the localized burning is estimated from cone calorimeter measurements. The methodology presented here is essentially the same as in [8]. However, in this work an attempt is made to a have a more realistic modelling of the geometry as opposed to [8] where it has been noted that the "*tray geometry [is] perhaps unreasonably oversimplified*". The configuration of cables is in fact considered in [8] to play a role as significant as physical and chemical properties on the cable response to fires. In [9], in-depth pyrolysis modelling was adopted. However, predicted large-scale HRR results were not compared to experimental data.

In the remainder of the paper, first the large-scale experimental set-up is presented, in order to understand the challenges in the CFD modelling of the configuration of interest. Then, the numerical modelling is discussed and more specifically how small-scale data is used to model the chemical reaction as well as pyrolysis. The second aspect of the numerical simulation is devoted to modelling the geometrical arrangement of the cables for the large scale. Finally the results are discussed before drawing the main conclusions.

DESCRIPTION OF THE EXPERIMENTAL SET-UP

As mentioned above the experimental test considered in this work (named CFSS1) has been performed by the French institute IRSN in the context of a large international collaborative research project PRISME 2 [2]. The fire source, as shown in Figure 1, is composed of five horizontal trays 2.4 m long and 0.45 m wide, with 0.3 m spacing. The trays are set-up against an insulated side wall. Each tray contains 49 power PVC cables of 13 mm outer diameter provided by one of the PRISME partners, the technical research centre of Finland (VTT). Each cable is composed of: (1) a metallic material with a linear mass of 85 g/m, (2) a sheath layer with a linear mass of 85 g/m, (3) a filler layer with a linear mass of 35 g/m, and (4) an insulation layer of 30 g/m. Chemical analyses have identified mainly three elements in the composition of each cable: PVC, $CaCO_3$ and phthalates.

Figure 1 The five horizontal cable tray device used for the CFSS fire tests

The ignition source is a sand burner of 0.4 × 0.4 m^2 located 0.2 m below the first tray at the centre. The gas burner supplied with propane delivers a fire power of 80 kW. The supply is stopped when the total HRR reaches 400 kW. This time is considered as the ignition time of the cable tray. For the test considered, the ignition time is 80 s.

NUMERICAL MODELLING

Small-scale Experiments - Standard Cone Calorimetry

The Finnish research institute VTT (Finland) and Aalto University (Finland) have performed several small-scale tests on the cables that were supplied for the CFSS1 test [4], the first ones being standard cone calorimetry results at two radiative external heat fluxes of 50 kW/$m^2$and 60 kW/m^2. The results displayed in Figure 2 show a good repeatability for the two 50 kW/m^2 tests. The solid lines in Figure 2c to 2f show average results for the quasi-steady state between t = 50 s and t = 500 s.

The averaging between the three tests is performed as follows:

$$\bar{\phi} = \frac{\left\lfloor \left(\phi^{1}_{50\mathrm{kW/m^2}} + \phi^{2}_{50\mathrm{kW/m^2}} \right) / 2 \right\rfloor + \phi_{60\mathrm{kW/m^2}}}{2} \tag{1}$$

where ϕ is the quantity to be averaged.

The measured effective heat of combustion (EHC) is EHC = 18.21 MJ/kg. The measured yields of oxygen, carbon dioxide and carbon monoxide are, respectively, y_{O2} = 1.38 g/g, y_{CO2} = 1.80 g/g and y_{CO} = 0.05 g/g. Unfortunately there are no measurements of soot yield and unburned hydrocarbon yields. Soot will be incorporated however in the modelling using the value of y_C = 0.17 g/g based on Tewarson's measurements for PVC [9].

Simultaneous Thermal Analysis (STA)

Simultaneous thermal analysis (STA) tests [4] showed that there are mainly three decomposition steps that occur roughly in the following temperature intervals: (1) 200 - 300 °C, (2) 430 - 530 °C for sheath and insulation and 380 – 440 °C for filler in air, and (3) temperatures higher than 660 °C. The primary information provided here (and used later in the modelling) is that the degradation process of the cables starts to take place at 200 °C.

X-ray Fluorescence (XRF), Gas Chromatography (GC) and Mass Spectrometry (MS)

The results obtained so far from cone calorimetry do not provide information on the chemical composition of the cables. The combined analysis from XRF, GC and MS shows that the main two compounds are C_2H_3Cl (which is PVC) and $CaCO_3$ as provided in Table 1 [4]. Many other elements were detected, with percentages, however, of less than 0.1% each. Based on Table 1 and the composition of the cables provided in the second section, the overall mass fractions in the cables are as follows: Y_{C2H3Cl} = 0.35 and Y_{CaCO3} = 0.41.

Table 1 Mass proportions [%]

Compound	Sheath	Filler	Insulation
C_2H_3Cl	45.83 %	11.11 %	33.49 %
$CaCO_3$	32.46 %	67.42 %	32.46 %

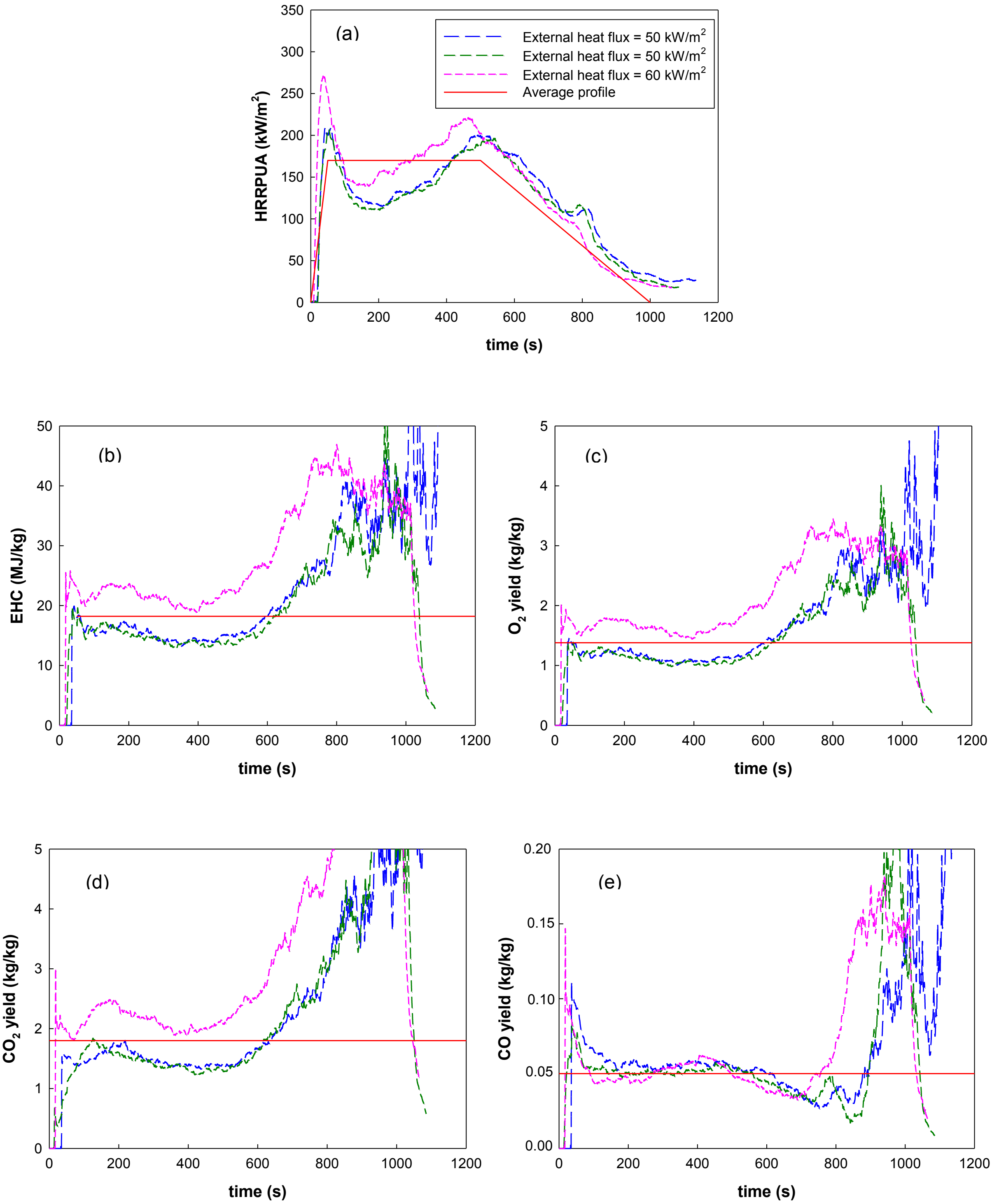

Figure 2 Standard cone calorimetry experimental data used in the simulation; (a) heat release rate per unit area (HRRPUA), (b) effective heat of combustion (EHC). (c) O_2 yield, (d) CO_2 yield, (e) CO yield

Exploitation of Small-scale Results for the Modelling

The exploitation of small-scale results for the modelling is performed at two levels. The first is related to the construction of the chemical reaction and the second is related to pyrolysis modelling.

Construction of the Chemical Reaction

In order to construct the chemical reaction, one must first consider the chemical composition of the fuel. The small scale tests described above (and more particularly XRF, GC and MS) showed that the cables are mainly composed of PVC (C_2H_3Cl) and $CaCO_3$. Unfortunately, in the cone calorimetry tests, compounds which may result from the combustion of PVC such as HCl were not measured. Also, the production of elements such as Ca was not quantified. As a result, the chemical reaction that is proposed here is mainly based on carbon (C), Hydrogen (H) and Oxygen (O) elements. Such reaction is written as follows:

$$CH_nO_m + v_{O2}\, O_2 \rightarrow (n/2)\, H_2O + v_{CO2}\, CO_2 + v_{CO}\, CO + v_s\, C \quad (2)$$

where the stoichiometric coefficients, *v*, are calculated based on the measured yields, *y*, using the following equation:

$$v_\alpha = \frac{MW_F}{MW_\alpha} y_\alpha \quad (3)$$

where *MW* denotes the molecular weight, *F* the fuel and α the species considered.

The purpose then is to find the coefficients *n* and *m*.

A carbon atom balance gives:

$$\left(\frac{y_{CO_2}}{MW_{CO_2}} + \frac{y_{CO}}{MW_{CO}} + \frac{y_s}{MW_C} \right) MW_F = 1 \quad (4)$$

where the molecular weight of the fuel is expressed as:

$$MW_F = 12 + n + 16m \quad (5)$$

An oxygen balance gives:

$$m + \left(2\frac{y_{O_2} MW_F}{MW_{O_2}} \right) = \frac{n}{2} + \left(2\frac{y_{CO_2} MW_F}{MW_{CO_2}} \right) + \left(\frac{y_{CO} MW_F}{MW_{CO}} \right) \quad (6)$$

The above system of equations gives *n = 0.8* and *m = 0.3*. The reaction is thus written as:

$$CH_{0.8}O_{0.3} + 0.785\, O_2 \rightarrow 0.4\, H_2O + 0.72\, CO_2 + 0.03\, CO + 0.25\, C$$

Pyrolysis Modelling

In order to simulate the flame spread process in a cable tray fire and the subsequent fire growth, one must consider a model for fuel response to the thermal stress exerted by flames and hot smoke. In general, there are two strategies concerning this modelling aspect: (1) an in-depth pyrolysis model or (2) a surface pyrolysis model.

The in-depth pyrolysis model computes the degradation of the fuel layer by layer. If the fuel is composed of several layers of different properties (as it is the case for cables) one must characterize the burning behaviour and thermal properties of each. For instance, the solid degradation of the sheath layer is described by an Arrhenius equation where the pre-exponential factor and the activation energy are estimated based on small-scale experiments. Furthermore, if one layer is not composed of one homogeneous material (which is often the case) the description of the burning behaviour must take this into account.

The complexities inherent to the in-depth pyrolysis modelling are bypassed in the surface pyrolysis model as follows. In this model, one assumes that when the surface temperature reaches a predefined value, the pyrolysis process is triggered and the burning behaviour occurs at rates similar to what is measured in the small-scale cone calorimetry tests. In other words, for the case at hand, when the surface temperature of the cables reaches 200°C (value measured in the STA tests), we assume that the heat release rate per unit area follows the average curve provided in Figure 2a. The computation of the surface temperature using Fourier's equation for conduction requires the thermal properties of the cables. Since the latter were not measured explicitly, the following values for PVC were considered [11]: (1) density, ρ = 1400 kg/m^3, (2) specific heat, c = 1.05 kJ/(kgK), (3) thermal conductivity, k = 0.16 W/(mK). The emissivity was taken as ε = 1. The heat of vaporization (required for the computation of the heat balance at the surface) was taken as h_v = 2300 kJ/kg [11].

Knowing that the cable diameter is 13 mm, it is difficult to track the pyrolysis of each of the 49 cables of each tray individually in a CFD model where a typical cell size would be in the order of 5 to 10 cm. It is also difficult to model the interstices between the cables that would allow the initial propane burner flame to penetrate through the trays in a vertical direction before starting a horizontal flame spread process. In the next section, a solution is proposed to overcome these difficulties.

Large-scale Modelling

In order to build the large-scale model in CFD, one must consider an important aspect regarding the pyrolysis modelling option used here. Since the pyrolysis modelling is based on the burning surface of the cables, it is imperative to reproduce the latter in the large-scale CFD model.

The total surface area of the cables per tray is:

$$A_{cbl} = n_{cbl/tray} \pi d_{cbl} L_{tray} \tag{7}$$

where $n_{cbl/tray}$ is the number of cables per tray, d_{cbl} the cable diameter, and L_{tray} the tray length.

Considering that 49 cables per tray is a "loose" arrangement, it is assumed here that the total surface area of the cables is involved in the burning. In other words the burning area to be simulated has the value provided in Equation (8).

A second constraint in building the CFD model is the presence of interstices that allow vertical flame propagation through the trays before radial spread. In this way, a realistic flame spread pattern is reproduced.

Taking into account the two elements discussed above (i.e., surface area of cables and presence of interstices), the modelled arrangement for the stack of trays displayed in Figure 3 is proposed. One can see that each tray contains four "groups of cables" of height $H_{gp,cbl}$ and width $W_{gp,cbl}$. The total surface area of the cables per tray according to this arrangement is:

$$\left(A_{cbl}\right)_{\mathrm{model}} = n_{gp,cbl/tray} \times 2\left(H_{gp,cbl} + W_{gp,cbl}\right) L_{tray} \tag{8}$$

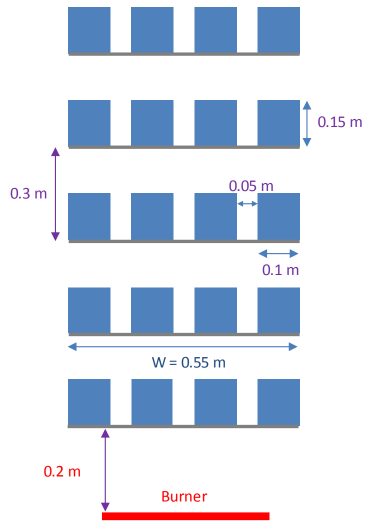

Figure 3 Side view of the large-scale model constructed in CFD

There are several possibilities in terms of values for $H_{gp,cbl}$ and $W_{gp,cbl}$ that lead to a surface area that equates the actual one (provided by Equation (8)). The values used here are $H_{gp,cbl}$ = 0.15 m and $W_{gp,cbl}$ = 0.10 m. The interstitial space is taken as 0.05 m. Several remarks regarding this modelling are provided in the following:

- All the dimensions are a multiple of 5 cm, which was predefined as the cell size for the simulation performed here. Therefore the gas phase mesh is well aligned with the solid burning items (i.e., groups of cables).
- With the proposed arrangement, the tray width is 55 cm, which is 10 cm more than the actual width. This has been taken into account in the positioning of the radiative heat flux gauges.
- Considering the dimensions of the groups of cables (and thus their volume) and the prescribed density for PVC do not yield the actual mass of the cables. This discrepancy does not (significantly) alter the outcome of the predicted burning behaviour since the pyrolysis model is based on the total surface area of the cables. The density value (as well as the other thermal parameters) nevertheless has a direct influence on the heat transfer calculation and the time to reach the critical surface temperature of 200 °C.

The modelling strategy described above has been implemented in the Fire Dynamics Simulator (FDS 6) [12], a CFD package developed by the National Institute of Standards and Technology (NIST). All the default models and constants in FDS were not changed except for the radiative fraction that was set to 0.56 as suggested by Tewarson [11] for PVC based on small-scale calorimetry tests.

RESULTS

Heat Release Rate (HRR) and Heat Release Rate per Unit Area (HRRPUA)

The results are first compared in terms of HRR profiles. Figure 4 shows rather encouraging results considering the fully predictive aspect of the simulation. The most interesting result is the excellent prediction of the peak HRR (with 3 % deviation from experimental data, see Table 2). The occurrence of the peak is however predicted with a delay of 4.7 min. This could be attributed to the modelled arrangement of the cables and the subsequent heat transfer and ignition. In fact, since the cables are modelled in 'groups', there are less interstices through which flames and hot gas penetrate, slowing down thus the heat-up and ignition process. Furthermore, there are uncertainties in the thermal properties of the 'groups' of cables, which are assumed to be made of one homogeneous material, whereas in reality they are composed of several layers (in addition to air gaps) with different properties.

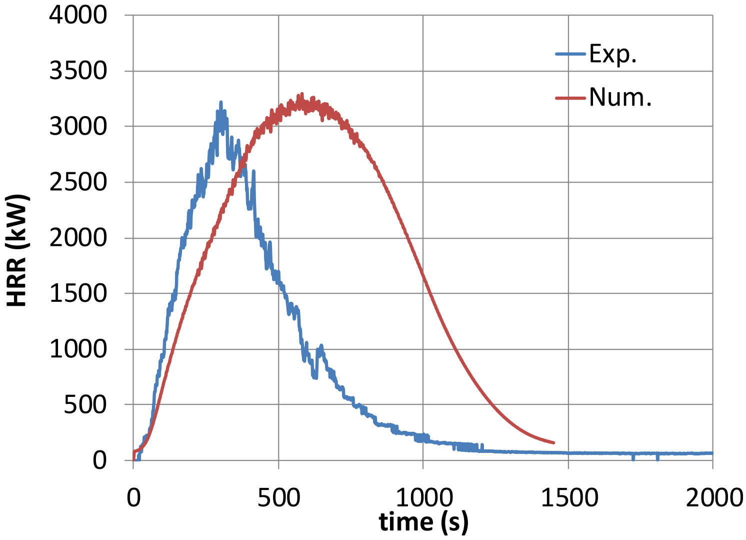

Figure 4 Comparison between the predicted and measured HRR profiles

Figure 4 shows also that the duration of the fire is substantially overestimated by 5 min (see Table 2), which resulted also in an overestimation of the total Heat Release by 96 % (see Table 2). The latter result might seem alarming because knowing the combustible mass, the heat of combustion of its main elements and their burning efficiency (from cone calorimetry) should allow a better prediction of the released energy content (i.e. total heat release). The discrepancy is nevertheless not surprising, because a surface pyrolysis model is used here. That means that the information on the total combustible mass (and its energy content) is not incorporated, but rather the information on the surface ignition temperature in conjunction with a HRRPUA profile. In our model, the source of uncertainty probably stems thus from one of these two elements (i.e. surface ignition temperature or HRRPUA profile). Arguing that the surface ignition temperature might not change from small-scale to large-scale, focus has been put on the HRRPUA profile. The idea is to estimate the latter profile for large scale and compare it to the cone calorimetry profile. An estimate of the burning area of the cables at large scale is provides thanks to the video fire analysis (VFA) technique described, applied and assessed in [6].

Table 2 Comparison between the main features of the predicted and measured HRR profiles

	Experiment	Numerical Model	Absolute Deviation	Relative Deviation
Time to ignition [s]	80	76	- 4	- 5 %
Fire duration [min]	19	24	5	26 %
HRR peak [MW]	3.2	3.3	0.1	3 %
Time to reach the HRR peak [min]	5	9.7	4.7	94 %
Total heat release [MJ]	1300	2552	1253	96 %

The temporal evolution of the cables' burning area is illustrated in Figure 5. Dividing the large-scale HRR profile by the burning area (see Figure 5) allows having an estimate of the large scale HRRPUA profile, which is displayed in Figure 6 and compared to its counterpart for the small scale. One can see that: (1) the average steady-state HRRPUA value for the large-scale is slightly lower than the small scale, and (2) the decay stage starts much earlier for the large-scale. As a consequence, strong deviations are obtained when transposing the HRRPUA profile from small-scale to large-scale. Other elements of uncertainty are worth mentioning, such as (i) the possible shrinkage in the cable area due to burning (which is not accounted for here), or (ii) the contact surface between the cables (despite the 'loose' arrangement) that reduces the burning area (also not accounted for here). It is important to note that these uncertainties confer to the calculation a rather conservative aspect (i.e. overestimation of the total heat release) except at the early stage where the predicted fire growth rate is lower than the actual one.

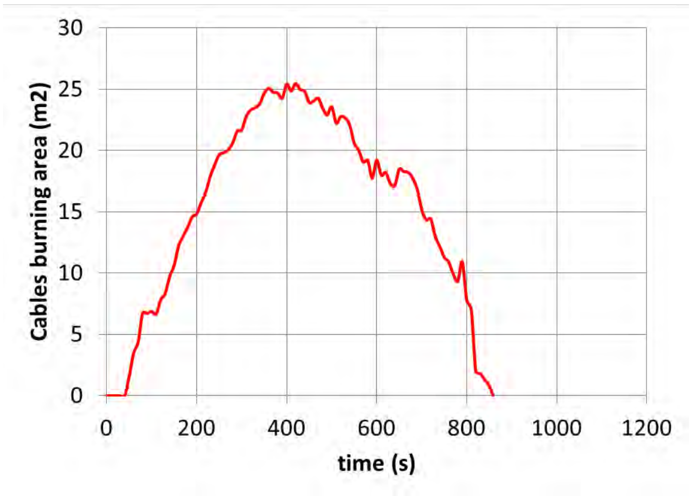

Figure 5 Estimated temporal profile of cable burning area from video fire analysis (VFA)

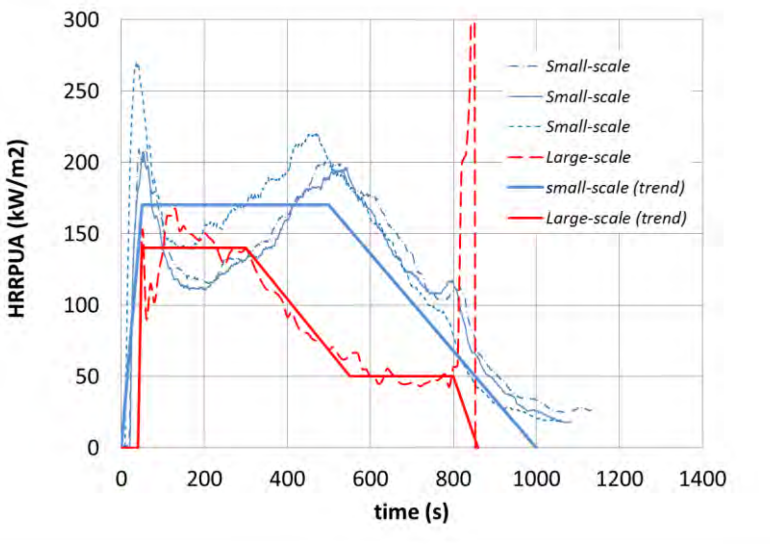

Figure 6 Comparison between the measured small-scale and large-scale HRRPUA

Species Concentrations

Table 3 shows a comparison of the species' peak concentrations between the simulation and the experimental data. The deviations displayed in Table 3 are partially attributed to limited or incomplete information on species production. The cone calorimetry tests have indeed not been calibrated for soot and unburned hydrocarbons (HC). Also, the fact that unburned hydrocarbons (HC) have not been taken into account into the main chemical reaction might have contributed to an overestimation of the other species (i.e. CO_2, CO and soot). Furthermore, it is important to bear in mind that since the exact chemical composition of the fuel and the species produced (including not only soot and HC but also HCl) are not well known, the modelled chemical reaction remains only a 'crude' approximation. Therefore, a more detailed characterization of species production at small scale could provide enhanced predictions for species' concentrations.

Table 3 Comparison between measured and predicted peak concentrations of species

Species	Experiment	Prediction
CO_2	4 %	10 %
CO	2000 ppm	4200 ppm
Soot	1800 mg/m^3	5000 mg/m^3
HC	1000 ppm*	-

* There was saturation in the measurement.

Radiative Heat Fluxes

Two radiative heat flux gauges were placed centrally in front of the stack of cable trays at 0.5 m (respectively 1 m) distance and 0.5 m (respectively 1 m) height. The radiative heat flux profiles displayed in Figure 7 show, as expected, similar features to the HRR results in Figure 4. The peak values are very well predicted but with a delay in time. Furthermore, the predicted profiles span over a longer period of time in comparison to the measured data. These results confirm that the key to 'good' predictions of the fire-induced environment is strongly linked to the prediction of the HRR profile. The influence of the model for the chemical reaction and the composition of the species produced needs however to be investigated.

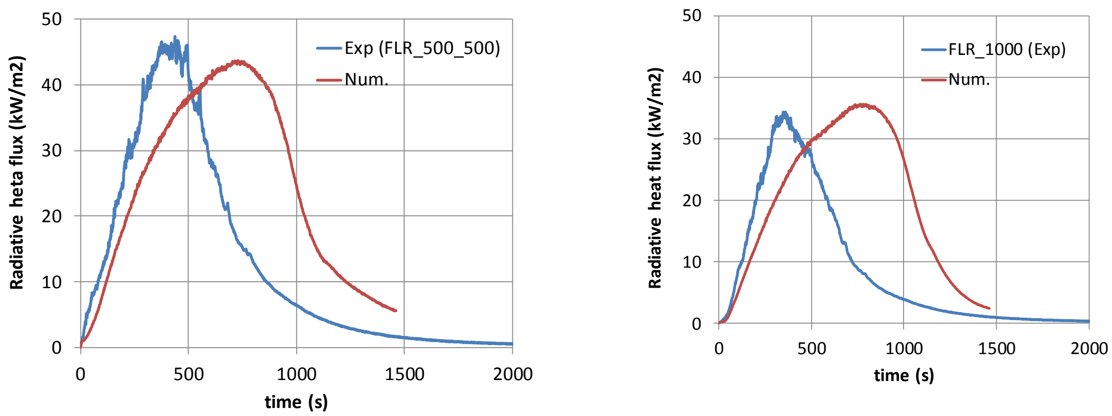

Figure 7 Comparison between measured and predicted radiative heat fluxes

CONCLUSIONS AND FUTURE WORK

The aim of the work presented here was to perform and assess a fully predictive CFD simulation of a large-scale fire involving a stack of horizontal cable trays.

The methodology is mainly based on a surface pyrolysis model that relies upon the specification of a ‘surface ignition temperature’ in conjunction with a temporal profile of HRRPUA. This information for the set of cables, specific to the large scale test considered in this work, is obtained from small-scale laboratory tests. Since the heat-up and ignition of the large number of cables in the large-scale cannot be modelled on an individual basis (i.e. cable by cable) due to limitations in the computational resources, the arrangement in the CFD model has been set in terms of ‘groups’ of cables. The dimensions of the ‘groups’ of cables have been set to yield an overall cable surface area (i.e. cumulative perimeter of the cables) equivalent to the actual configuration. Furthermore, the ‘groups’ of cables were separated by small interstices, in an attempt to mimic the actual pattern of the fire spread according to which flames and hot gas penetrate through the cable trays in a vertical direction before starting to spread radially. An additional modelling point discussed in this work is the set-up of the chemical reaction. The several valuable small-scale tests did not allow a full characterization of the species produced from the burning of the cables (made mainly of PVC and a plasticizer), inducing therefore an additional source of uncertainty.

The results obtained are quite encouraging, particularly for the peak heat release rate (HRR) which was predicted with a relative deviation of 3 %. The time to reach the peak is however overestimated by 4.7 min (i.e. 94 %). The fire duration is also overestimated by 5 min (i.e. 24 %). These discrepancies are mainly attributed to differences in the HRRPUA (heat release per unit area) profiles between small-scale and large-scale. The latter was calculated by estimating the burning area of cables using video fire analysis (VFA). There are also other elements of uncertainty worth mentioning, such as (i) the possible shrinkage in the cable area due to burning (which is not accounted for here), or (ii) the contact surface between the cables (despite the ‘loose’ arrangement) that reduces the burning area (also not accounted for here). The results in terms of radiative heat fluxes were similar to the HRR results. However, larger discrepancies are obtained for the concentration of species, which could be mainly attributed to uncertainties in the yields of species produced (such as HCl and unburned hydrocarbons, which were not quantified and therefore not considered in the single-step chemical reaction).

Future work will be performed to assess the overall level of uncertainties by performing simulations with different options for (1) the set-up of the chemical reaction, (2) the choice of the overall thermal properties of the ‘groups’ of cables, and possibly (3) alternative geometrical arrangements. The methodology will also be applied and evaluated for additional large-scale tests.

ACKNOWLEDGMENTS

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EXPERIMENTAL AND NUMERICAL STUDY OF SMOKE PROPAGATION THROUGH A VENT SEPARATING TWO MECHANICALLY VENTILATED ROOMS

Laurent Audouin, Hugues PRETREL, Samuel Vaux

Institut de Radioprotection et de Sûreté Nucléaire (IRSN),
Centre de Cadarache - Bât 346 - 13 115 Saint Paul Lez Durance, France

ABSTRACT

The paper presents an experimental and numerical study about smoke propagation through a horizontal opening between two superposed compartments, as can be encountered in nuclear installations, in case of a fire taking place in the lower room. The experimental configuration proposed in this study consists in two rooms mechanically ventilated and connected each other by a horizontal opening. The fire source is simulated by a propane burner located in the lower room. The inlet ventilation duct is located in the lower room and the exhaust ventilation duct is located in the upper room. For such experimental configuration, several flow regimes at the horizontal opening connecting the two rooms can be encountered depending on the fire power, the opening size (diameter, depth) and the ventilation set-up (location of inlet/outlet ducts, flow rate). Indeed, flow at the opening is governed by buoyant forces due to the hot gases produced by the fire, the inertia effect due to the forced ventilation and the momentum effect due to smoke flow nearby the horizontal opening (for instance, ceiling jet or thermal plume from fire). Consequently, such complex mixed (natural/forced) convective flows are still a challenge for CFD fire codes to make properly calculations of these experimental scenarios. The objective of this paper is to assess the capability of ISIS code (CFD) to simulate the behaviour of smoke propagation inside these two superposed compartments. Results of this study are presented with details (especially, thermal stratification and flow rates through the horizontal vent) and are discussed thoroughly.

INTRODUCTION

Smoke movement in nuclear installations is a major issue for safety assessments. Indeed, due to the transport of hot gases and soot particles from the initial fire room to neighbouring compartments, safety systems (i.e. EIS, Equipment Important for Safety) can be damaged as clogging of high efficiency filters (HEPA) located in the ventilation network or failure of electrical or electronic devices. Smoke can flow through horizontal or vertical openings (for example, doorways, vents or reduced orifices used for running cables) controlling the mass and heat transfer of hot gases between compartments. As the nuclear facilities are equipped with a ventilation network, these flows are governed by mixed convection (both buoyancy and inertia), are often mono or bidirectional and turbulent in nature. When the opening is directly connected to the fire enclosure, the flow may have a significant effect on the evolution of fire source by modifying the amount of air (and consequently of oxygen) entering into the enclosure and the exhaust of combustion products to adjacent rooms. Thus, predicting such flows from engineering zone codes and CFD tools remains a key issue for nuclear safety assessments. From this challenge, this study proposes to focus especially on the heat and mass transfer of hot gases through a horizontal opening separating two superposed compartments.

A brief overview of the literature [1] to [6] shows that previous experimental studies have established the theoretical basis of such vertical flows. From small-scale experiments, these authors proposed some analytical approaches to calculate the natural flows due to buoyant effect based on the Froude number as well as the inertia term when mechanical ventilation is applied. From these approaches and the corresponding database, Cooper [4], [5] has proposed correlations in order to determine flowrate for pure natural convection and mixed convection in the goal of being used in zone codes. However, there are very few experimental studies focusing on this type of smoke flow and the Cooper models, which are used in most fire zone codes, have been validated only on very limited number of experiments (mainly at small scale). Thus, their predictive performances are still discussed, especially for large scale fire tests and for confined and mechanically ventilated enclosures as pointed out by Emmons [6] and Li [7]. Consequently, it is worth to investigate further for experimental and theoretical studies concerning vertical flows through horizontal openings based on large-scale tests including the effect of forced flow. The expected outcomes are to improve the physical understanding of these types of flows with variable density, to validate/enhance the existing correlations and to assess the CFD codes to simulate such flows.

This study investigates the capability of CFD code developed in IRSN [8], [10] to capture the main behaviour of mixed convective flow (natural/forced) through a horizontal opening separating two superposed rooms during large-scale fire tests carried out in IRSN experimental installation. In this work, an engineer approach is used with the CFD code, as a first step. In order to assess the relative effect of natural convection due to thermal plume vs forced flow induced by the ventilation system, two initial regimes of forced ventilation are more especially investigated in this paper for a constant heat release rate of 97 kW: (1) high ventilation rate (i.e. about 2300 m^3/h) and (2) low ventilation rate (i.e. about 600 m^3/h).

DESCRIPTION OF THE FACILITY AND EXPERIMENTS

The DIVA Facility

The IRSN DIVA experimental facility is a large scale multi-room facility (see Figure 1 and 0) representative of nuclear installations. It includes four compartments (labelled 1 to 4) and a corridor. All the walls are 0.3 m thick and are built with reinforced concrete designed to withstand a gas pressure ranging from – 100 hPa to 520 hPa. The compartments labelled 1 to 3 are 6 m in length, 5 m in width and 4 m in height.

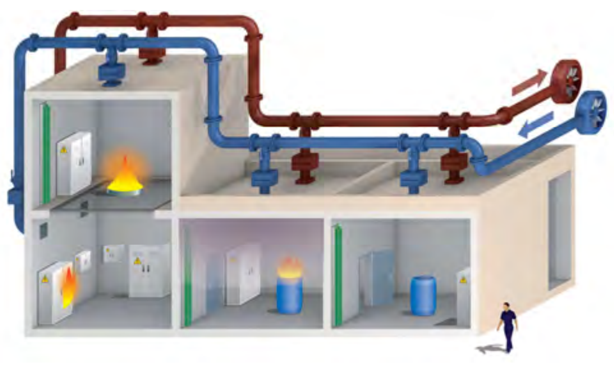


Figure 1 Schematic of the DIVA facility and its ventilation network

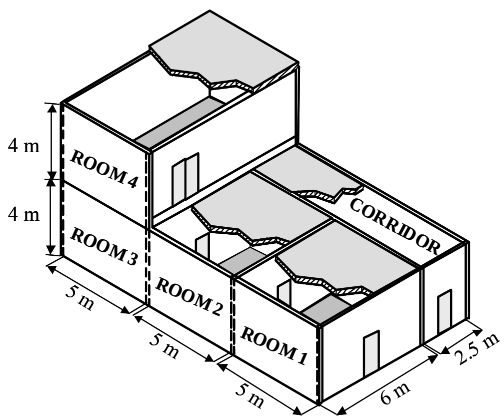


Figure 2 Main geometrical dimensions of the DIVA facility

The room 4 (length × width × height = 8.8 × 5 × 4 m^3) is designed to study the vertical hot gas propagation from a lower (room 3) to an upper room (room 4) through a horizontal opening having a surface of about 1 m². The corridor (length × width × height = 15 × 2.5 × 4 m^3) is located along the rooms 1 to 3.

All rooms of the DIVA facility can be connected with a mechanical ventilation system by means of inlet and outlet ducts, which can be set up at any height in each room depending of the fire scenarios. The lower rooms (compartments 1 to 3 and corridor) can be connected through a single doorway or different types of elements (simple openings, fire door, etc.). The DIVA installation can be 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 (Nuclear Power Plant) as well as nuclear laboratories and nuclear reprocessing plant.

For this study, the aim is to investigate the vertical smoke propagation through a horizontal opening (or hopper) for mechanically ventilated fire room scenarios. The tests are focused on the study of the flows going through the vent. Indeed, these types of flows are complex in nature (typically, mono or bi-directional) because of the competition between the buoyancy force due to the density difference on one hand and, on the other hand, the inertia force due to the relative pressure induced by the ventilation system. Moreover, the flows could be significantly more complex if fire plume is located just below through the hopper (i.e. including the direct effect of plume momentum). The fire scenario in Figure 3 for the two tests studied in this paper comprises two rooms with mechanical ventilation (one lower compartment as fire room, i.e. room 3; one upper room, i.e. room 4, connected by a horizontal opening of 1m² in area). The inlet duct of fire room and the outlet duct of upper compartment (or target room) are located in the upper part of each compartment (about 0.8 m from the ceiling).

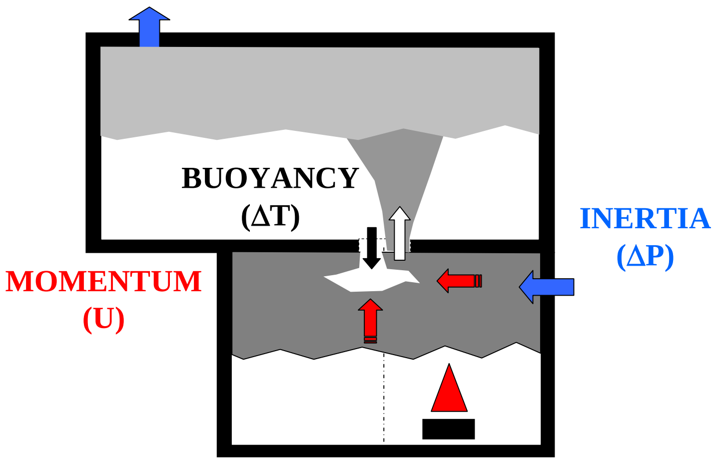


Figure 3 Experimental configuration

The walls are made in re-enforced concrete, of which some areas are covered with 30 mm rock-wool panels for safety requirements. The wall areas are the ceiling of both rooms and the upper parts (about 2 m) of the side walls in the fire room.

The horizontal vent is a rectangular section with the dimensions 1.03 m × 1.03 m = 1.06 m^2, located at the centre of the fire room (off-centre in the upper room, as shown in 0 b)). The overall depth of the orifice is 0.385 m, which corresponds to the 0.300 mm thickness of the concrete wall separating the two rooms and a layer of rock wool.

Fire Source

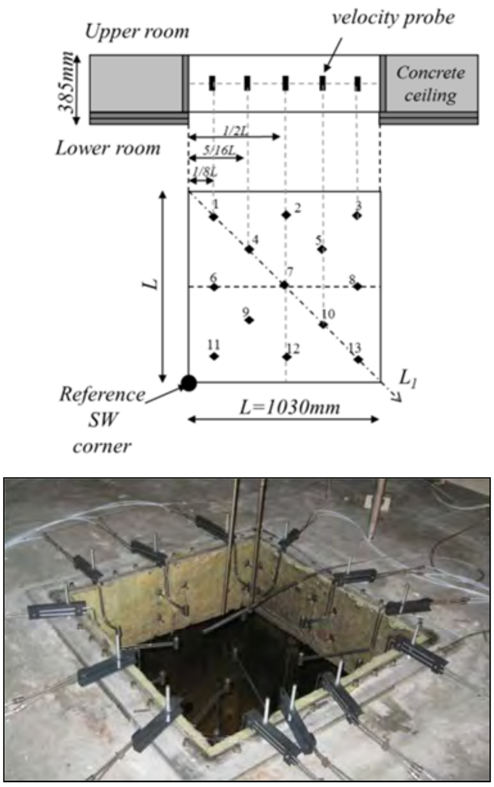
The fire source (cf. Figure 4) is a rectangular propane gas burner (with an equivalent diameter of 1.13 m) located at the Northwest corner of the lower room (off-centre location). The gas is supplied directly through a circular pipeline within a pan filled by water. Propane gas bubbles through the water before supplying the reactive zone of the flame. The ignition is piloted by an electric spark system. The level of fire HRR is adjusted by monitoring the propane flow rate.

Before ignition During the fire test

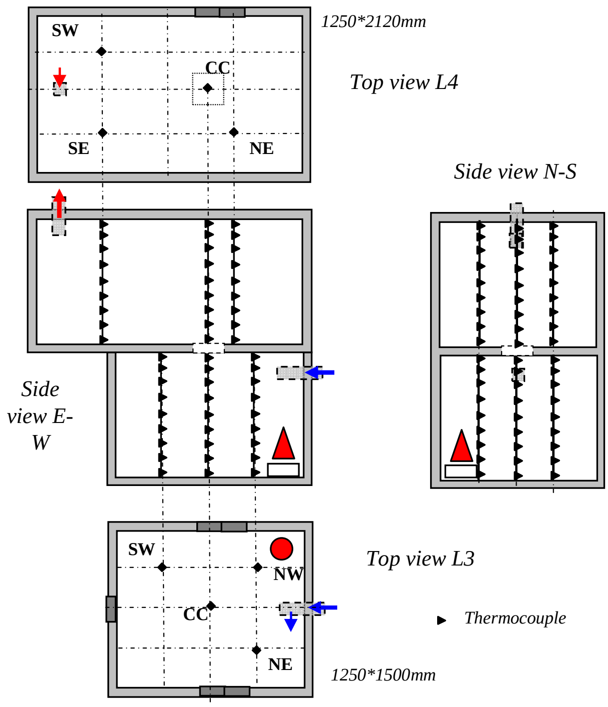
Figure 4 Propane gas burner

Measurement Techniques

The propane flow rate of the burner is measured by means of a mass flowmeter. The corresponding fire HRR is obtained from the propane flow rate measurement multiplied by and the effective combustion enthalpy for propane (ΔH_c = 46 MJ/kg).



(a)



(b)

Figure 5 Location of measurements (flow and temperature):
(a) Set of 13 bi-directional McCaffrey-type pitot tubes at horizontal opening,
(b) Trees of thermocouples in both compartments

The ventilation flow rates are measured with an average Pitot probe devices, located in ventilation ducts (inlet and outlet) and connected to membrane pressure transducers. A probe coefficient and the measurement of temperature are considered to compute the volume flowrates. The horizontal vent in Figure 5 (a) is equipped with 13 bidirectional velocity probes (numbered 1 to 13) linked to K-type thermocouples in a cross section of the vent as shown in Figure 5 (a). Each bidirectional probe connected to pressure transducers has been tested previously in a wind tunnel in order to determine its own probe coefficient.

The lower and upper compartments include four thermocouples trees (cf. Figure 5 (b)) named SW, CC, NE and SE for the upper room L4, and SW, CC, NW and NE for the fire room L3 as described in Figure 5 (b). Each tree is equipped with nine K-type thermocouples located at 0.05 m, 0.55 m, 1.05 m, 1.55 m, 2.05 m, 2.55 m, 3.05 m, 3.55 m and 3.90 m from the floor.

Experimental Determination of the Vent Flow

The flow rate at the horizontal vent is calculated from the spatial integration of the velocity fields taking into account the gas temperature. The following expression is used:

$$\dot{m}_{vent} = \int_S U\rho(T)ds = C\left\{\frac{PM}{R}\sum_S\left(\frac{U_i}{T_i}ds_i\right)\right\} \quad (1)$$

P is the absolute static pressure, M the molar mass of air and R the constant for perfect gas. U_i are average velocity of each probe during few minutes. The integration method of the 13 velocities has been proposed taking into account the “no slip” condition at the boundaries and also the fact that each velocity probes are associated to different surface element “ds_i” (see details in [9]). In addition, a coefficient C is introduced to take into account the error due to coarse mesh near the boundaries. This coefficient is determined experimentally at ambient temperature by comparing a reference flow rate (measured in the ventilation network) and the one computed from the integration of the velocity field with the relation (1). The value for the coefficient C is 0.96 ± 0.08 for these tests [9]. Then, the corresponding upward and downward flow rates are deduced from the two relations hereafter:

$$\dot{m}_{vent}^{up} = \frac{1}{2}\left(\int_S U\rho(T)ds + \int_S |U|\rho(T)ds\right) \quad \text{and} \quad \dot{m}_{vent}^{down} = -\frac{1}{2}\left(\int_S U\rho(T)ds - \int_S |U|\rho(T)ds\right) \quad (2)$$

For large velocity through the horizontal vent (i.e. large flow rate), the uncertainty on flow rate is assessed about 5 % (see Table 1, test Q12). For low velocity (i.e. small flow rate), the uncertainty on this one is as high as 66 % due to the weak velocities measured by the bi-directional probes (see Table 1, test Q17). Fortunately, the natural convection flows due to buoyancy effect increases significantly the velocities at the horizontal opening during fire tests and thus allows a rather good estimation of the smoke mass transfer through the vent.

Table 1 Initial experimental conditions for the two tests

	Test Q12	Test Q17
HRR [kW]	97	97
Tr [1/h] (*)	8.0	2.2
Before ignition (experimental values)		
Q adm (fire room) [m^3/h]	2270	630
Q ext (target room) [m^3/h]	2380	680
Q net [m^3/h]	2205	225
Q(FR -> TR) [m^3/h]	2250	365
Q(TR -> FR) [m^3/h]	45	140
Uncertainty [%]	~ 5	~ 66

* The renewal rate Tr is computed as the ratio between the volume flow rate in inlet duct before ignition and the room volume.

Experimental Procedure During Fire Tests

The test procedure is as follows: First, the ventilation network is put into place in order to achieve the expected ventilation flow rate. Then, the gas burner is turned on. A mixed convection flow starts to occur at the horizontal vent, and smoke progressively fills up the adjacent compartment. The fire HRR remains constant over all the test duration. The test is stopped (i.e. the gas burner is turned off) once a quasi-steady state is obtained. The criteria are based on the velocity and gas temperature recordings at the vent and the vertical temperature stratification within the two rooms.

NUMERICAL MODELLING

Model Overview

A detailed description of the ISIS code (CFD code) is out of the scope of this paper and is available with more details in [8], [10], and [11]. The physical modelling used in ISIS is based on the low-Mach number assumption. As a result, the thermodynamic pressure in confined environments is constant in space but varies in time. Turbulence modelling is dealt with the Favre-averaging approach. The turbulent terms describing the Reynolds stress tensor and turbulent scalar fluxes are modelled using the eddy viscosity hypothesis and the standard k-ε model with usual wall laws (i.e. logarithmic law). Turbulent combustion is based on the fast chemistry conserved scalar approach and the mean reaction rate, which is controlled by the turbulent flow mixing, is determined by the Eddy-Dissipation Concept model (EDC). A one-step irreversible chemistry is assumed. The soot production and oxidation are simulated by a semi-empirical one equation model and radiative transfers are dealt with P1 radiative model. The gas absorption coefficient of the mixture used the total emissivity approach of the Weighted Sum of Gray Gases model and the soot absorption coefficient is related to the soot volume fraction (scattering effects are neglected according to the Mie theory). The wall conduction is taken into account through the 1D Fourier's equation and the convective flux is given by standard thermal wall laws. The mass flow rate of the ventilation network at each branch of the compartment is solved using a general Bernoulli equation.

The system of equations is discretized on a staggered mesh based on the common finite volume method. The time integration scheme is performed using a fractional and semi-implicit scheme. The balance equations are then solved in a step-by-step sequential process including turbulence, mixture fraction, fuel mass fraction, enthalpy, radiative transfer and Navier–Stokes equations, written with a projection method.

Simplified Approach for the Fire Source

This work is a preliminary step to assess the capability of ISIS code to describe the mixed convective flows (natural/forced convection) through a horizontal vent. In this first step, a coarse grid (less than 300000 meshes) is used and the study is only focused on the thermal flows between the two compartments and thermal stratification inside these rooms. Thus, as an “engineer” approach, the modelling of fire, in addition to the standard k-ε model described just above, is simplified by using the Volumetric Heat Source (VHS) method. This type of simplification is appropriate if accurate predictions of the shape of the flame and the near field region of the fire are not important. The heat source is added thanks to a cuboid shape corresponding to the burner surface and the flame height obtained from Thomas [12]. The radiative heat transfer modelling is either not considered in the simplest approach or calculated by the P1 radiative modelling and Fletcher model [13] for the absorption coefficient. Radiation loss by the fire is considered by the radiative fraction coefficient χ_R (taken as 0.4 for propane). Finally, three calculations are performed per fire test as described in Table 2 hereafter.

Table 2 Modelling proposed in this study

	Modelling of Combustion	Modelling of Radiative Transfer	Modelling of Absorption Coefficient	Complexity
VHS[(1)] + χ_R	no (VHS)	no	no	+
VHS + ray P1	no (VHS)	P1	Fletcher 0	++
Standard k-ε	EDC	P1	WSGG[(2)] 0, 0	+++

(1) VHS: Volumetric Heat Source (thermal approach)
(2) WSGG: Weighting Sum of Gray Gases

Computational Grid and Boundary Conditions

Boundary conditions of the computational domain are consistent with the design of tests discussed previously (thermal properties of concrete walls and insulation layers for some walls and the ceiling, etc.). The initial and boundary conditions of the ventilated system are fixed by the experimental values obtained before the fire ignition. The total head losses of the inlet and outlet ducts are determined from the initial state (pressures and flow rates) and are kept constant during all the simulation time. The computational domain for the two compartments is presented in Figure 6, where the pool fire and the ventilation ducts have been drawn. The mesh has a grid size of 10 cm inside the compartments and 5 cm near walls (see Table 3). Thus, the horizontal area corresponding to the horizontal vent is described with about 100 meshes. Overall, the computational domain is composed of about 265 000 meshes.

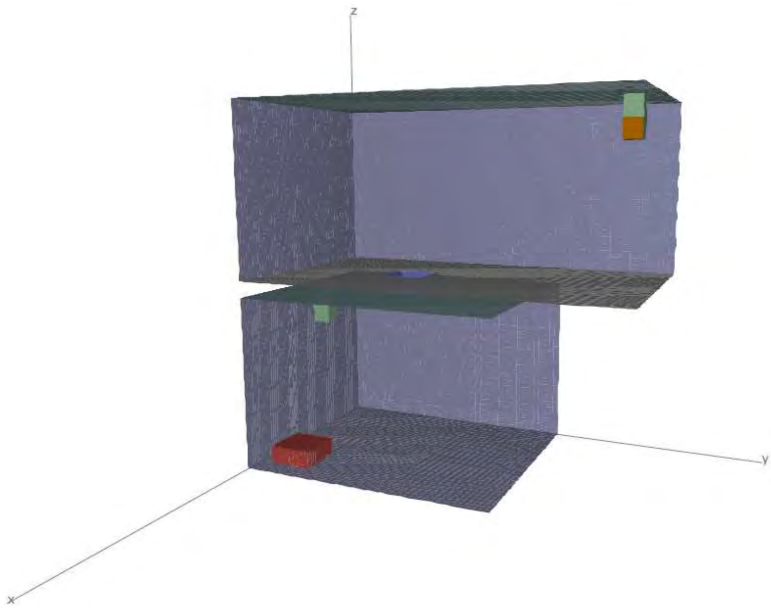


Figure 6 Scheme of computational domain

Table 3 Description of the numerical grid

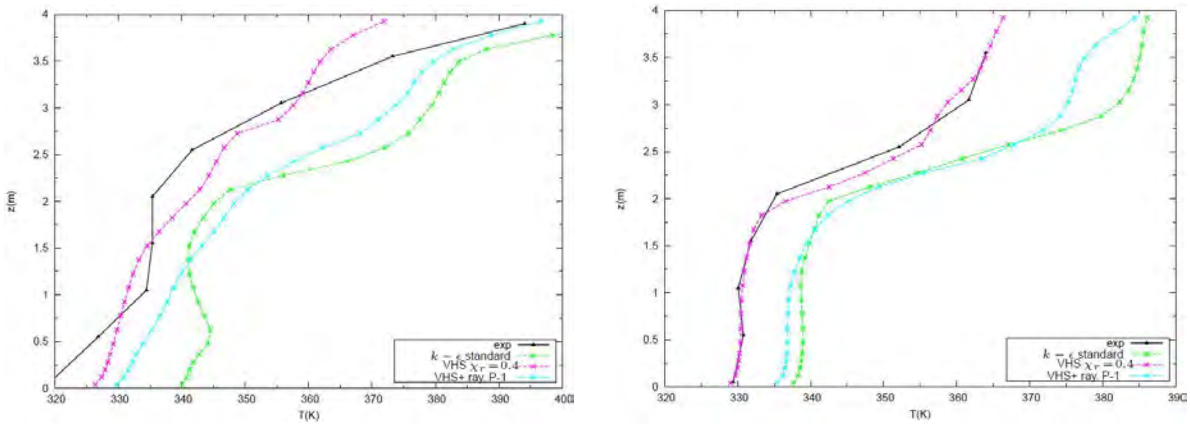
Nodes	Fire Room	Target Room
N_x	56	56
N_y	52	66
N_z	40	40

DISCUSSION

The thermal stratifications calculated for the Q12 test at 1000s are presented in Figure 7 (fire room) and Figure 8 (target room).

Overall, for all the modelling, the comparisons with experimental temperatures at two locations (NW and centre CC) in fire room are rather correct with an overestimation of temperature profiles for two models (i.e. “VHS + ray P1” and “Standard k-ε“). The maximum temperature close to the ceiling is close to about 400 K in fire room and is calculated with a pretty good agreement by them. The shapes of vertical temperature are also assessed properly.

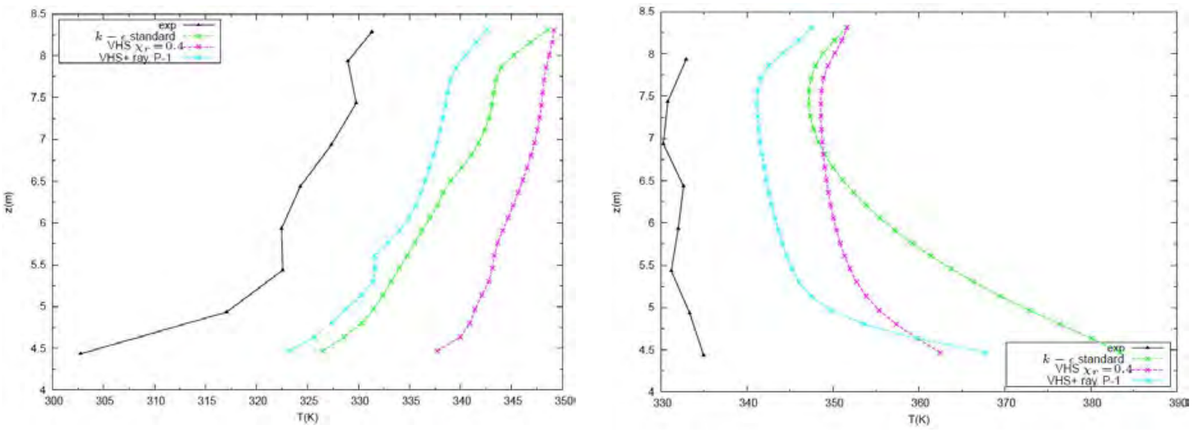
But, whatever the modelling used for these calculations (“VHS + χ_R”, “VHS + ray P1” and “Standard k-ε“), the thermal stratifications obtained at two locations (NE and centre) in target compartment (cf. Figure 8) are poorly predicted. For the Q12 test, the high forced ventilation rate (i.e. about 2300 m^3/h) from fire room to upper compartment involves an important coupling with the natural convection due to buoyancy. This complex type of flows (mixed convection through the horizontal vent and then thermal plume in target compartment) seems to be difficult to simulate with the three models and the coarse grid proposed in this work.



NW tree

Centre (CC) tree

Figure 7 Temperature (thermocouple tree) in the fire room, test Q12 at t = 1000 s



NE tree

Centre (CC) tree

Figure 8 Temperature (thermocouple tree) in the fire room, test Q12 at t = 1000 s

Concerning the upward mass flow rate at the opening (see Figure 9), a rather good prediction is obtained during the fire test with a discrepancy less than about 10 % regardless the modelling. The numerical result is in good agreement for the downward mass flow (its value remains close to zero). Both experimental and numerical outcomes show that the flow through the vent is mainly upward and mono-directional, corresponding to the first regime (see Figure 10) as discussed by Prétrel in [9]. This result can be also highlighted by the mapping of temperature presented in Figure 11. Compared to the initial flow rate of 2320 m^3/h (i.e. 0.76 g/s), the upward mass flow at t = 1000 s is nearly the same meaning that the forced flow induced by the ventilation system is dominant in relation with the natural convection due to the fire source.

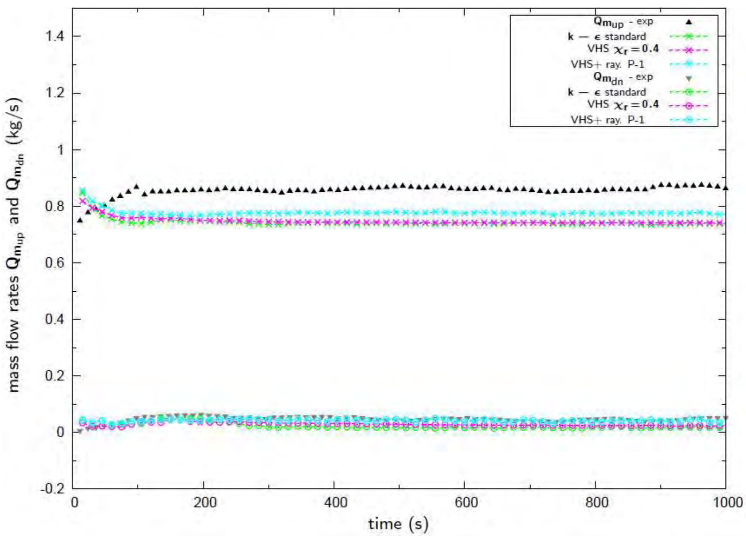


Figure 9 Mass flow rates (upward and downward), test Q12 for t = 0 to 1000 s

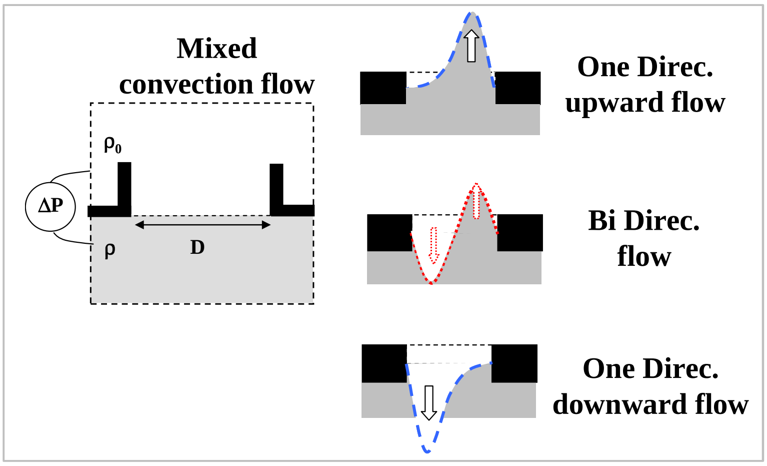


Figure 10 Smoke movements involving mixed convective flow in confined and ventilated compartments (from Prétrel [9])

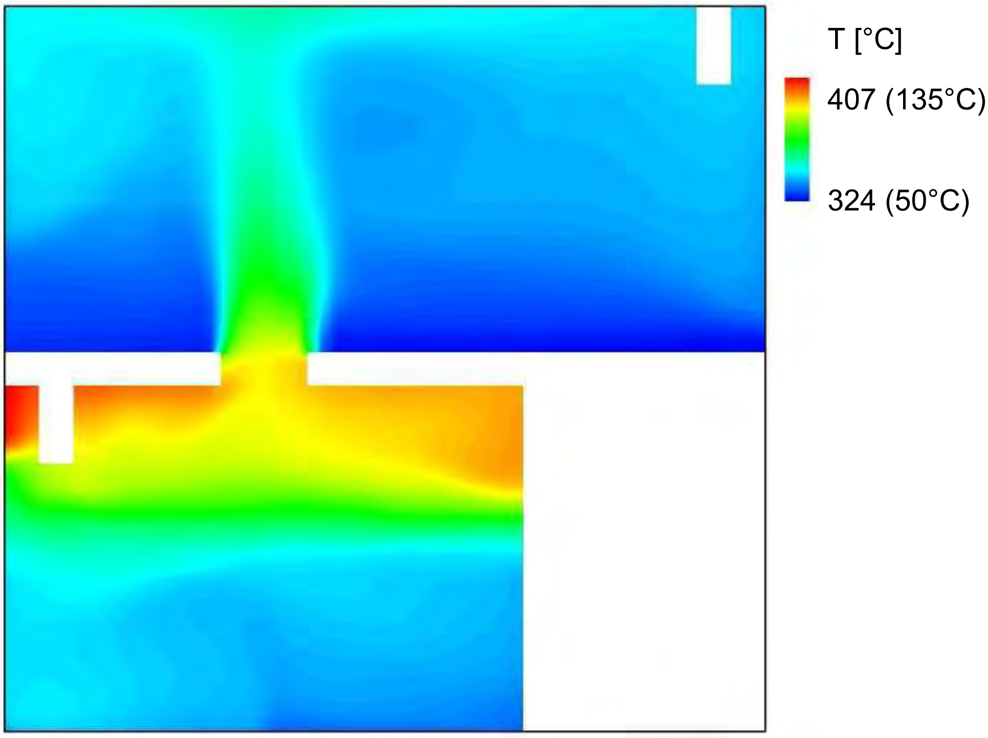
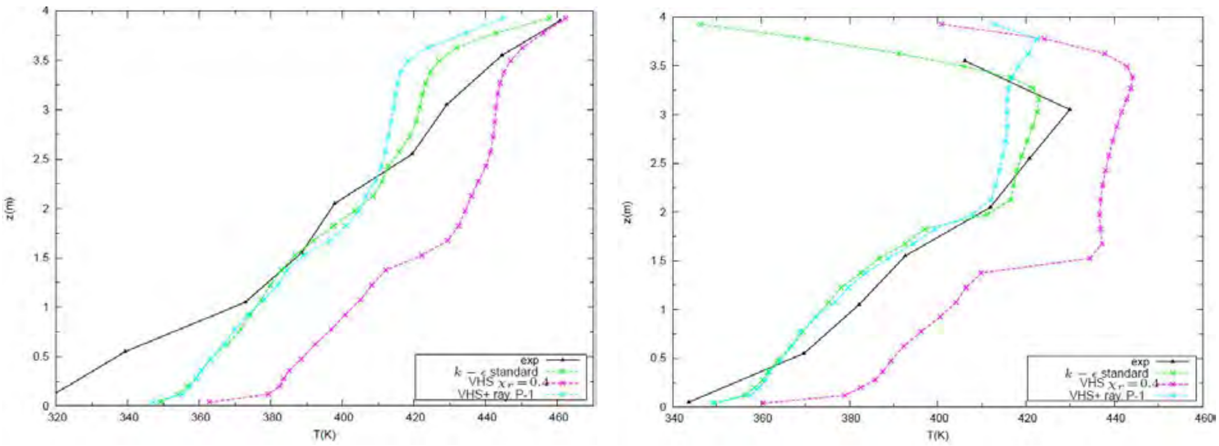


Figure 11 Temperature mapping (ISIS simulation) with "k-ε standard" modelling Test Q12 at t = 1000s

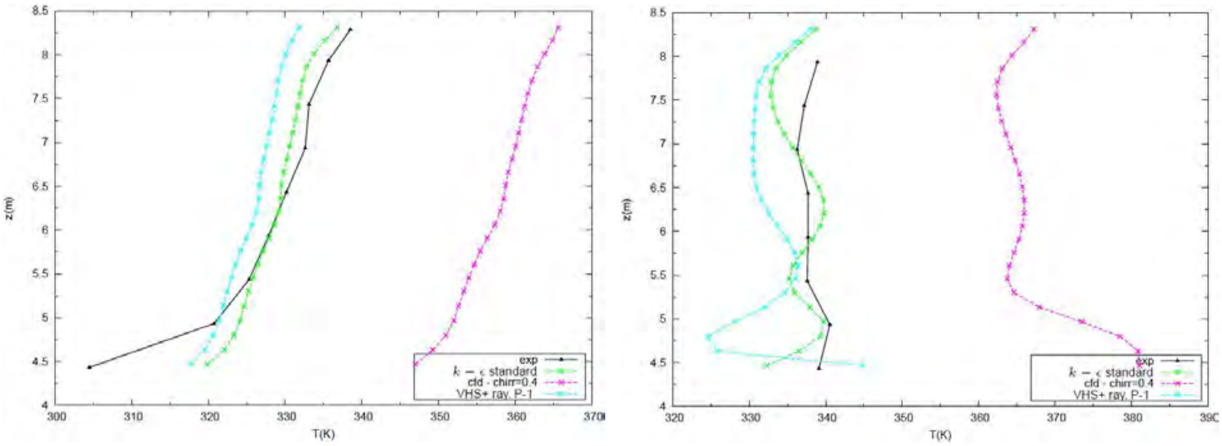
The thermal stratifications assessed by the CFD code for the Q17 test at 1000 s are presented in Figure 12 (lower room) and in Figure 13 (upper room). Compared with the previous simulation of Q12 test, the vertical temperatures (values and shapes) at the two same locations as previously are in good agreement with experimental data for the two models taking into account the radiative heat transfer, namely "VHS + ray P1" and "Standard k-ε". The modelling "VHS + χ_R" does however not calculate the temperature profiles both in fire and target compartments properly. Indeed, the temperatures inside rooms are widely overestimated whatever the height considered.



NW tree

Centre (CC) tree

Figure 12 Temperature (thermocouple tree) in the fire room, test Q17 at t = 1000 s



NE tree　　　　　　　　　　　　Centre (CC) tree

Figure 13　　Temperature (thermocouple tree) in the fire room, test Q17 at t = 1000 s

The numerical results presented in Figure 14 show that the experimental data concerning the upward and downward mass flow rates through the horizontal vent are well predicted. The downward mass flow seems to be weakly underestimated by ISIS but the experimental data of velocity are very difficult to measure for low ventilation rate as in Q17 test. This particular point was discussed previously in the paper. Consequently, the uncertainty on the downward mass flow is likely quite large and the calculations obtained by ISIS can be considered as realistic. Nevertheless, compared to the initial flow rate of 640 m^3/h (i.e. 0.21 g/s), the upward mass flow at t = 1000 s increases of about 50 % meaning that forced and natural flows are of the same order of magnitude. For low ventilation rate as in Q17 test, the thermal convection due to buoyancy effect plays an important role in the smoke and heat transfer between the lower and the upper compartments. Moreover, in average, the numerical results estimate that the upward mass flow is about 0.3 g/s and the downward mass flow is about 0.1 g/s. This outcome points out that the flow through the horizontal opening is bi-directional corresponding to the second regime (see Figure 10) described by Prétrel in [9]. It is also observed on the mapping of temperature presented in Figure 15.

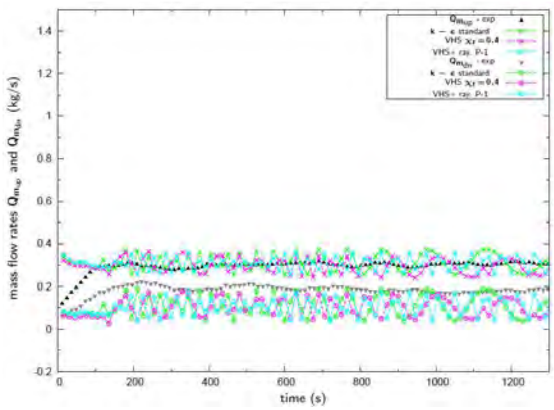


Figure 14　　Mass flow rates (upward and downward), test Q17 for t = 0 to 1000 s

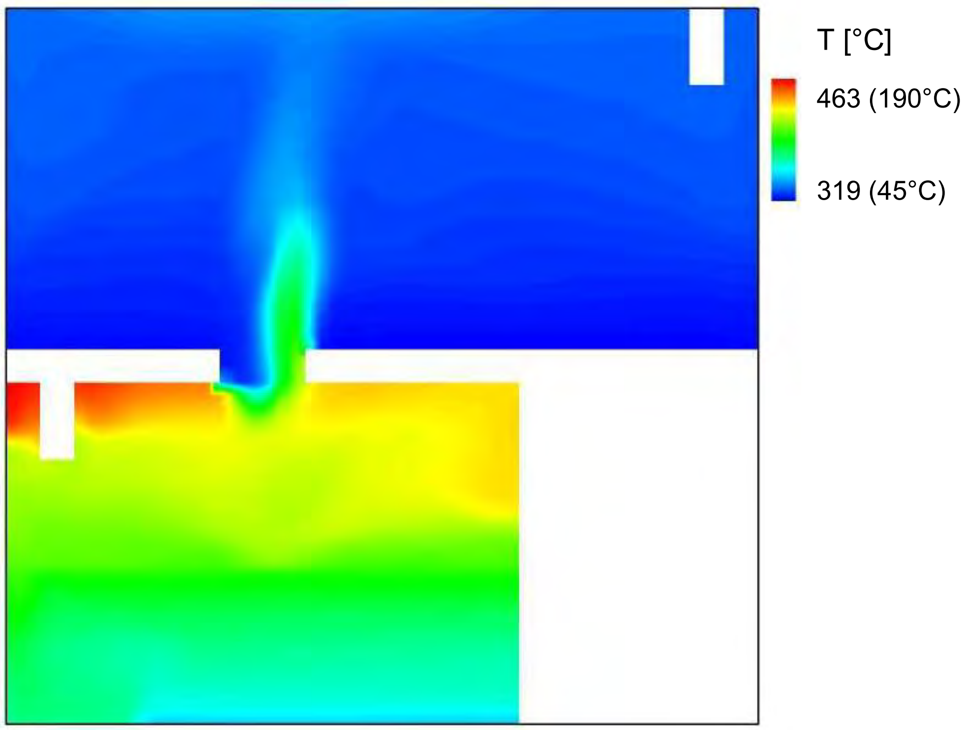


Figure 15 Temperature mapping (ISIS simulation) with "k-ε standard" modelling Test Q17 at t = 1000s

Overall, for the two simulations of tests, the modelling based on "VHS + ray P1" provides numerical results close to those of the "Standard k-ε" modelling. But the simplest approach "VHS + χ_R" with no consideration of the radiative heat transfer is not relevant to simulate this type of scenarios. It is therefore necessary to simulate these two tests by taking into account a radiative model.

CONCLUSIONS AND FUTURE WORK

The objective of this paper is to investigate the capability of the ISIS code 0, 0 to capture the main behaviour of mixed convective flow (natural/forced) through a horizontal vent separating a lower fire room and an upper target compartment. The fire source is a propane gas burner of 97 kW located in the lower room. As a first step, an engineer approach is favoured by use of three simple models (two thermal with or without radiative transfer, and one with combustion) and of a coarse grid (less than 300000 meshes). Concerning the ventilation system, two initial regimes of forced ventilation are more especially investigated: (1) high ventilation rate (of about 2300 m^3/h) and (2) low ventilation rate (of about 600 m^3/h). These scenarios of interest have been carried out previously in a large-scale experimental installation 0 and the numerical outcomes are compared with the experimental data.

For this study, the main conclusions can be drawn as:

- An engineer approach (simplified modelling, coarse grid) can give satisfactory results under the condition that the radiative heat transfer is considered in the simulation.
- This type of simplified approach seems to be not relevant for high ventilation rate condition (forced convection =>> natural convection) but seems to provide quite good results for low ventilation rate (forced convection ~ natural convection).

- For the two tests, the regimes of flow (mono and bi-directional) are properly assessed.
- Moreover, the numerical outcomes concerning the upward/downward mass flows are in good agreement with the experimental data.

Based on this first study, the next step will consists to go further in the simulation of these two tests, especially by using finer grids and a better description of turbulent flow (LES modelling). It is expected that this step provides better predictions concerning the mixed convective flow at the opening, the thermal stratification in the compartments and the plume above the horizontal vent (upper room).

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LARGE EDDY SIMULATION OF A MECHANICALLY VENTILATED COMPARTMENT FIRE FOR NUCLEAR APPLICATIONS

Bao P. Xu[1] and Jennifer X. Wen[2*]

[1] Faculty of Energy & Power Engineering, Dalian University of Technology, Dalian, China
[2] Warwick FIRE, School of Engineering, University of Warwick, Coventry, United Kingdom

ABSTRACT

This paper deals with the modelling of a mechanically ventilated compartment fire which is a commonplace in nuclear fire scenarios. An advanced Computational Fluid Dynamics (CFD) field model with a wall conjugate heat transfer treatment is proposed. It simultaneously solves the compartment fire flow and the wall heat conduction. The flow solver is based on the Large Eddy Simulation (LES) based fire simulation solver FireFOAM within the frame of open source CFD code OpenFOAM®. An extended eddy dissipation model is used to calculate the chemical reaction rate. A soot model based on the concept of smoke point height is employed to model the soot formation and oxidation. A finite volume method is adopted to model the radiative heat transfer. The ventilation flow is modelled by a simplified Bernoulli equation neglecting the detailed information on the ventilation system. The proposed model is validated against a single room fire test with forced mechanical ventilations. The predictions are in reasonably good agreement with experimental data.

Keywords:
compartment fire; large eddy simulation; conjugate heat transfer; forced ventilation

INTRODUCTION

A confined and mechanically ventilated compartment fire is a commonplace in nuclear fire scenarios, where fire compartments are connected to ventilation networks to prevent radioactive releases. The mechanically ventilated compartment fires differ from the naturally ventilated ones in that fires are confined in enclosures with forced ventilations, leading to significant thermodynamic pressure variations [1]. Although naturally ventilated compartment fires have been extensively studied in the literature [2] to [5], mechanically ventilated compartment fires are less documented due to the lack of large-scale fire tests. A fire test program PRISME [6] was designed to investigate the fire growth in full-scale confined and mechanically ventilated compartments. The test results have been widely used as benchmark data for the validation of fire models [1], [7] to [10].

The underlying physics of the mechanically ventilated compartment fires are complex. Combustible solid/liquid material is firstly pyrolysed into gaseous phase and then ignited by heat sources, resulting in a buoyant fire accompanied by the formation and oxidation of soot particles. Soot particles enhance the radiative heat transfer, and the radiative heat feedback to the surfaces of combustible material can modify the fire burning rate. The time-varying burning rate induces pressure variations which alter the ventilation flow rates. Combustion heat is extracted from the confined compartment in two ways: one is through ventilation exhaust flows; the other is via heat transfer to the walls.

* Correspondence Jennifer.Wen@warwick.ac.uk

The modelling of the above mentioned fire scenario is challenging. Some simplifications have to be made. There are generally two fire modelling approaches: integral zone models [11], [12] and CFD field models. In the zone models the fire compartment is normally divided into an upper hot zone and a lower cold zone, where a homogeneous mixture is assumed in each zone. The zone properties such as pressure, temperature, species concentrations, etc. are solved by the integrations of the mass and energy balances in each zone. The field models solve differential transport equations of mass, momentum and energy, closed by more elaborate physics-oriented models such as turbulence model, pyrolysis model, combustion model, soot model, radiation model, ventilation model and conjugate heat transfer (CHT) model etc. In this study, an advanced CFD field model with a wall CHT model is proposed and validated against PRISME Source test data [6].

NUMERCAL DESCRIPTIONS

Fire spreading in a compartment is a low Mach number flow of variable density. A fully compressible flow solver becomes inefficient due to the need to capture acoustic waves; hence a low Mach approximation is widely adopted to speed up simulations in the fire modelling, by dividing the pressure into a time dependent thermodynamic pressure for the energy equation and a spatial and time dependent hydrodynamic pressure for the momentum equation. The flow solver in this study is based on our previous studies on fire modelling [13], [14], in which the eddy dissipation concept (EDC) was extended to the framework of large eddy simulation (LES) by taking into account the distinctive roles of the sub-grid scales (SGS) and using the partially stirred reactor (PaSR) concept to relate the filtered soot formation rate to the soot chemical time scale which is assumed to be proportional to the laminar smoke point height (SPH). The turbulent mixing time scale for soot is computed as a geometric mean of the Kolmogorov and integral time scale. A finite volume based radiation model is adopted for radiative heat transfer. More details about the LES solver for fire modelling can be found in the reference [13], [14].

A major feature of mechanically ventilated compartment fires is in that fire rooms exchange mass, momentum and energy with the ambient environment through ventilation networks. The ventilation networks are usually rather complex, composed of different types of components such as ducts, bends, valves and fans etc. Computing the ventilation flow of the whole networks requires detailed information of all the components. To simplify the ventilation calculation, the flow resistance coefficient between network nodes can be calculated using test pressure data if the detail of the ventilation structure is known [9]. The ventilation flow is usually solved separately from the flow solver, and the ventilation flow is coupled to the compartment flow via boundary conditions. In this study a simplified ventilation model [1], [8] based on a general Bernoulli equation is adopted, which neglects the detailed information on the ventilation system. Because of the pressure variation, reverse flows are observed at both admission and exhaust ventilation branches. Therefore, the ventilation model can be written in a general form as follows:

$$P_{vent} - P_{room} = R_{vent}\rho Q|Q| \tag{1}$$

where P_{vent} is the pressure of a node inside the ventilation system, which keep nearly constant during operation; P_{room} is the pressure at the ventilation openings connected to the compartment; R_{vent} is the resistance coefficient; Q is the ventilation volumetric flow rate and keeps positive if ventilation flow is directed into the compartment and otherwise keeps negative; ρ is the upstream density, i.e. taken as the density at the ventilation openings if the ventilation flow is directed out of the compartment. The resistance coefficient R_{vent} is priori calculated using test data and keeps constant during the simulation.

The hot fire smoke exchanges heat transfer with wall surfaces in the forms of radiative and convective heat transfer. The most accurate method to calculate the heat transfer is the CHT approach which solves the gas phase flow in the compartment and the wall heat conduction simultaneously and the wall surface heat flux is used as a coupled boundary condition between the two phases. The modelling of the convective heat transfer in the confined compartment is not a straightforward task. The convective heat transfer can be calculated from the empirical heat transfer coefficient which is related to the properties of gas phase, the type of convection, the orientation of solid wall. However, there lacks of well-defined large-scale fire tests for determining the coefficients in the literature. Therefore, a fixed coefficient or one-dimensional heat conduction calculations [9] were normally adopted in previous studies. In this study, a three dimensional CHT model, which solves a 3D heat conduction equation (2) for the solid wall, is proposed to calculate the wall heat transfer. At the wall surface a balance equation (3) of heat flux is used as a coupled thermal boundary condition between the solid and gas phases.

$$\rho_s c_s \frac{\partial T}{\partial t} = k_s \left[\frac{\partial^2 T}{\partial x^2} + \frac{\partial^2 T}{\partial y^2} + \frac{\partial^2 T}{\partial z^2}\right] \quad (2)$$

$$k_s \frac{\partial T_s}{\partial n}|_{wall} = k_g \frac{\partial T_g}{\partial n}|_{wall} + Q_r \quad (3)$$

where ρ_s, c_s and k_s are respectively density, specific heat and heat conductivity of solid phase and assumed to constant; k_g is effective heat conductivity of gas phase corrected by SGS turbulence, counteracting the under-prediction of the temperature gradient at the under-resolved wall boundary, Q_r is radiative heat flux.

VALIDATION CASE AND PROBLEM DESCRIPTIONS

The numerical models in the second section are validated against PRISME Source test PSR-SI-D3, which is a single room test. The test room has an internal dimension of 5 m x 6 m x 4 m ventilated by an admission branch and an exhaust branch. The ventilation branches enter the room through two 0.4 m x 0.4 m rectangular ducts. The test room consists of 30cm thick concrete walls and the ceiling is covered by an insulation layer of 5cm thick rock wool. A fuel pan of a surface area 0.4 m^2 is situated in the room centre and hydrogenated tetra-propylene ($C_{12}H_{26}$) is used as fuel. The detailed information on the test facility and test conditions can be found in [1], [6].

The computational domain shown in Figure 1 is divided into two regions, a fluid region enclosed by a solid region (black). The fluid region is meshed using non-uniform grids with a maximum grid size of 5cm, clustering at the wall surfaces. Uniform mesh is created for the solid region, 10 grids are placed in the thickness direction. The simulation starts from zero velocity, an initial pressure of 98384 Pa and an initial temperature of 307 K for the two regions. No pyrolysis model is attempted for modelling the fuel burning rate in this study, in order to avoid its uncertainties affecting the predictions. The burning rate is defined by the experimental data as shown in Figure 2, and applied as an inflow boundary condition at the fuel inlet. Reverse flows were observed at both the admission and exhaust branches in the fire test, an inlet-outlet boundary is applied at the ventilation openings where inflow/outflow velocities are calculated using the ventilation model in Section 2 according to the room pressure. On the all the wall boundaries of the fluid region, a no-slip velocity boundary condition and a coupled thermal boundary condition (Equation 3) are applied.

A one-equation SGS turbulence model [15] is employed to represent the SGS stress and turbulent Viscosity, and two model coefficients are set to be $C_k = 0.05$ and $C_e = 0.4$. An etended Eddy dissipation combustion model [14] and a soot model based on smoke point concept [15] are used to model the chemical reaction rate and the formation and oxidation of soot particles, and the smoke point height of $C_{12}H_{26}$is set to 0.029 in the simulation.

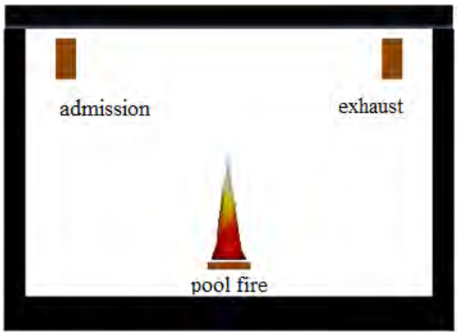

Figure 1 Computational domain.

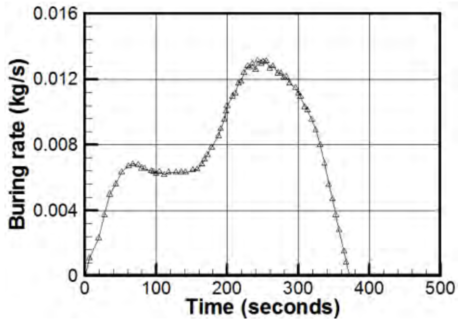

Figure 2 Fire burning rate of the test.

A finite volume method (FVM) [14] is used for radiative heat losses, in which the radiative transfer equations within multiple solid angles are solved to evaluate radiative absorption and emission. With the Gray assumption, the total absorption coefficient is decomposed into the gas absorption coefficient and the soot absorption coefficient. A total of 16 solid angles covering a hemisphere is used for the radiative transfer equations (RTE) as a compromise between computational time and accuracy.

For the heat conduction of the solid region, Equation (2) is spatially discretized using a second order Gauss linear scheme and the solution is marched in time using a first order explicit scheme. The thermal properties of solid region are assumed to be constant in the simulation, and can be found in the reference [1]. The outer boundaries are assumed to thermally adiabatic.

RESULTS AND DISCUSSION

Figure 3 shows the comparison of the room pressure variations. The predictions follow closely with the changing trend of the experimental data. Three factors contribute to the pressure variations: combustion heat release, ventilation flows and wall heat transfer. After the ignition the combustion heat release induces a rapid pressure rise due to the increasing burning rate. The predicted variation is initially higher than the experimental data, which increases the exhaust flow rate and reduces the admission flow rate resulting in an early reverse flow at the admission branch (see Figure 4). The first pressure peak, which occurs at t = 65 s corresponding to the first peak of the burning rate, is under-predicted by 700 Pa due to the over-prediction of the pressure variation which increases the energy and mass losses through the ventilation system. After the first peak, owing to the increasing heat losses via the wall heat transfer and the ventilation system, the room pressure then drops to a valley value which is well predicted by the current simulation. As the burning rate increases again after t = 160 s, the pressure variation increases and reaches to another peak because the combustion heat release outnumbers the heat losses. The extinction at t = 370 s induces the lowest valley value which is also moderately under-predicted by about 700 Pa due to the under-prediction of the wall heat transfer resulting from the inadequate grid resolution at the wall boundary layers.

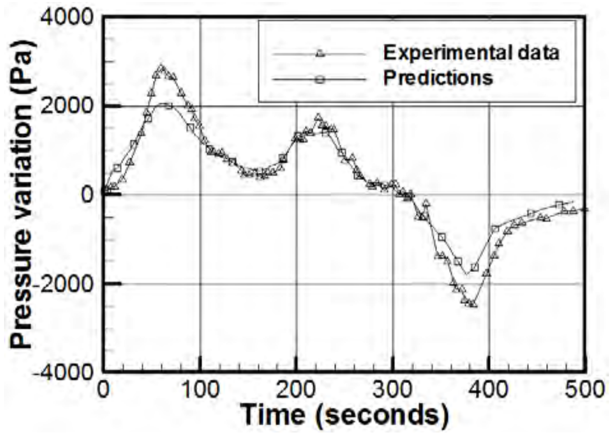

Figure 3 Comparison of the pressure variations.

Comparisons of the volumetric flow rates at the admission and exhaust branches are shown in Figure 4. The ventilation flow is closely coupled with the pressure variations and its changing pattern resembles that of the pressure variations. The predictions are generally in reasonable agreement with the experimental data. Over-predictions of the ventilation flow rates are observed prior to t = 250 s for the admission flow rate and t = 300 s for the exhaust flow rate. After the ignition, a reverse flow quickly establishes at the admission branch due to the initial pressure increase, and the reverse flow last for approximately 250 seconds. A reverse flow is also observed for the exhaust branch after the extinction, but it only lasts for 80 seconds. The predictions agree rather well with the experiment data after t = 300 s.

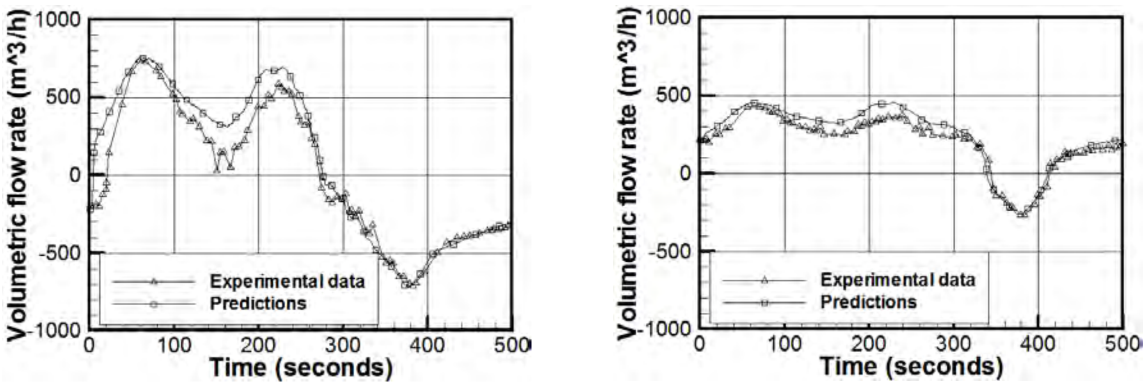

Figure 4 Comparisons of the admission flow rate (left) and exhaust flow rate (right).

Figure 5 shows the comparisons of oxygen molar fraction at three locations. Overall, the predictions agree well with the experimental data at all the locations. The fire smoke tends to spread towards the ceiling due to the buoyancy effect and then forced to move downward. Therefore, the oxygen concentration starts to drop sooner close to the ceiling, and then almost drops linearly with time to a minimum value before the extinction. After the extinction, the concentration starts to gradually recover due to the intake of fresh air from the admission branch. The moments when the oxygen molar fraction starts to drop and the final oxygen concentration are well predicted in current simulation.

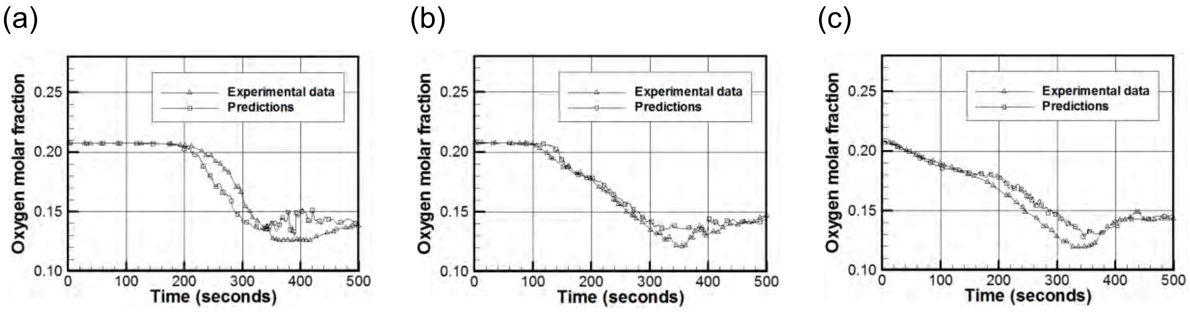

Figure 5 Comparisons of oxygen molar fraction at three locations - (a) X = - 0.8 m, Y = 0 m, Z = 0.35 m; (b) X = 1.5 m, Y = - 1.25 m, Z = 0.8 m; (c) X = 1.5 m, Y = - 1.25 m, Z = 3.3 m.

Figure 6 shows the comparisons of temperature at three locations. One peak value is observed almost at the same moment t = 300 s for each location, which is higher close to the ceiling. The peak value and its location are well predicted for all the locations. After the ignition the temperature is over-predicted, and after the peak moment the temperature is moderately under-predicted, except at the location of NE380 which is significantly over-predicted after t = 400 s due to the over-heating from the heated ceiling. The discrepancies of the temperature can be mainly attributed to the imperfectness of the ventilation model, wall CHT model and the combustion model. Among these causes, the error resulting from the CHT model due to the less resolved boundary layer is the most possible reason responsible for the discrepancies.

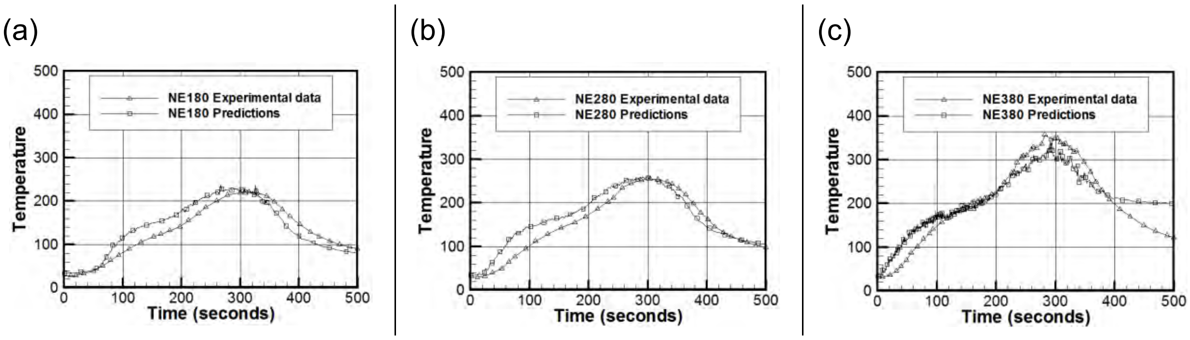

Figure 6 Comparisons of temperature at three locations - (a) X = 1.5 m, Y = - 1.25 m, Z = 1.8 m; (b) X = 1.5 m, Y = - 1.25 m, Z = 2.8 m; (c) X = 1.5 m, Y = - 1.25 m, Z = 3.8 m.

Figure 7 displays the contours of temperature in the middle plane at four different moments. After the ignition, a fire plume establishes and spreads towards the ceiling. The maximum flame temperature is observed around 1300 K. The temperature is higher inside the plume, and a hot upper layer is developed and extended towards the floor. After the extinction a temperature gradient still prevails inside the room.

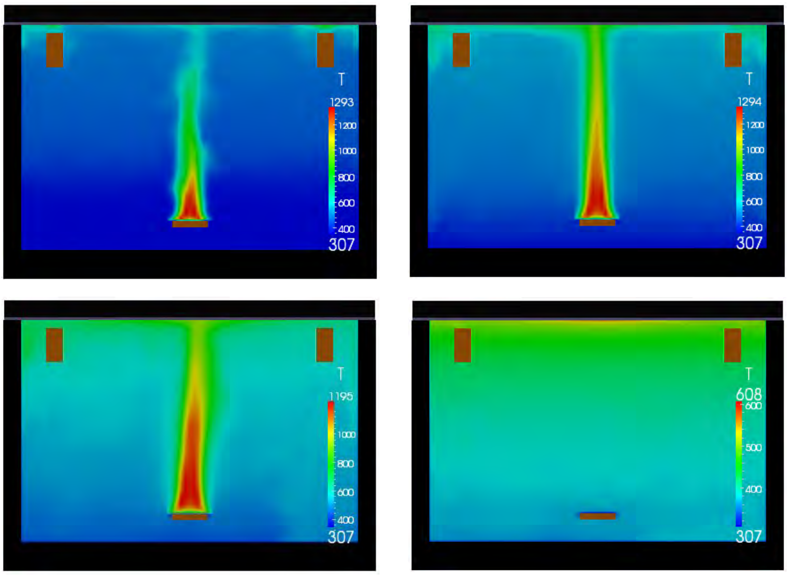

Figure 7 Contours of temperature in the middle plane at t = 100 s, 200 s, 300 s, 400 s.

Figure 8 displays the contours of oxygen mass fraction in the middle plane at four different moments. The fire plume consumes oxygen inside the room. As the fire plume induces pressure rise causing a reverse flow at the admission branch, which prevents the intake of fresh air, the depletion of oxygen results in the fire extinction.

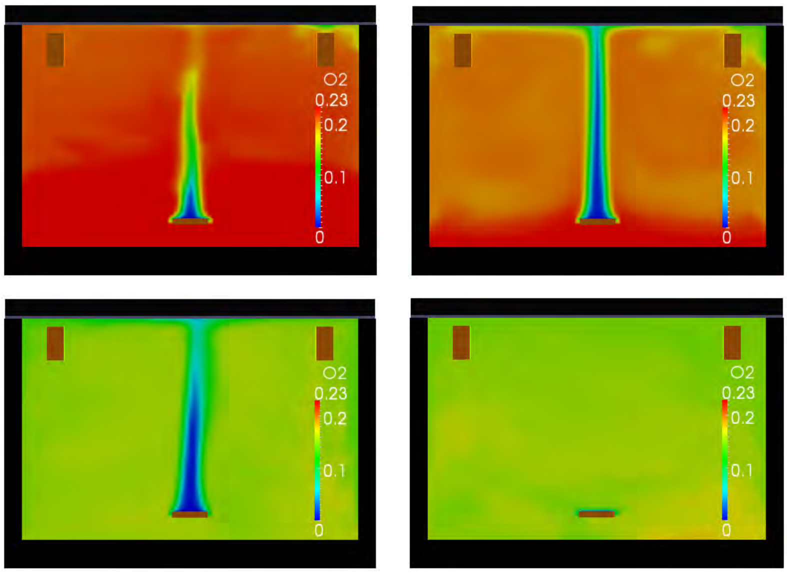

Figure 8 Contours of oxygen mass fraction in the middle plane at t = 100 s, 200 s, 300 s, 400 s.

Figure 9 displays the contours of carbon dioxide mass fraction in the middle plane at four different moments. The maximum of carbon dioxide mass fraction is found to be 0.18 inside the fire plume. After the extinction, a nearly uniform distribution of carbon dioxide mass fraction around 10 % is observed.

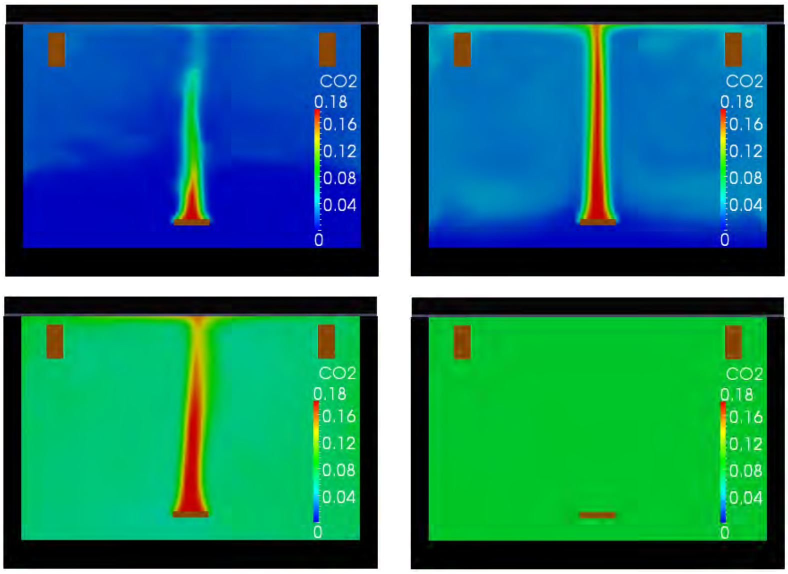

Figure 9 Contours of carbon dioxide mass fraction in the middle plane at t = 100 s, 200 s, 300 s, 400 s.

Figure 10 displays the contours of soot mass fraction in the middle plane at four different moments. The maximum of soot mass fraction is found to be 2.4 % inside the fire plume. After the extinction, a nearly uniform distribution of soot mass fraction around 0.5 % is observed.

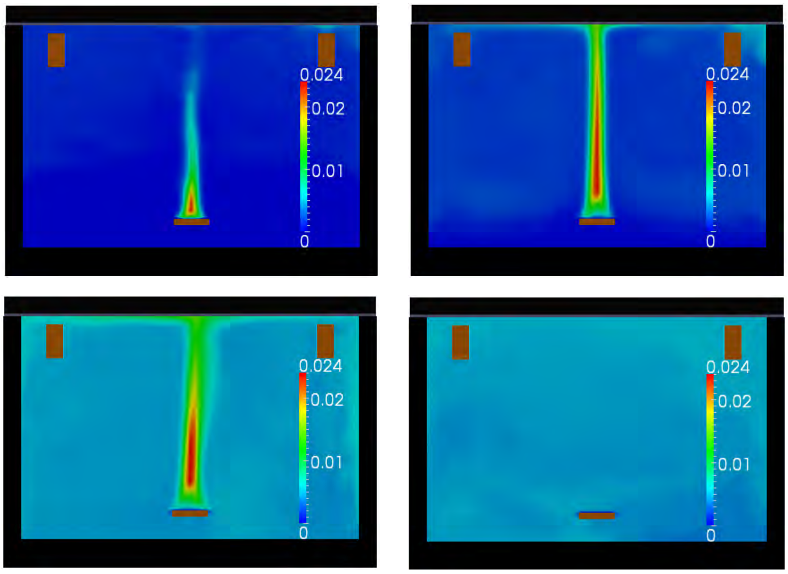

Figure 10 Contours of soot mass fraction in the middle plane at t = 100 s, 200 s, 300 s, 400 s.

CONCLUSIONS

An advanced CFD field model has been proposed for modelling of mechanically ventilated compartment fires. The field model is based on a LES flow solver coupled with a wall CHT model. The LES flow solver adopts a one equation SGS turbulence model, an extended eddy dissipation combustion model, a soot model based on the concept of smoke point height, a FVM radiation model, a ventilation model based on a general Bernoulli equation.

The proposed model is validated against a single room fire test, ventilated by an admission branch and an exhaust branch. The predictions of the pressure variation, the volumetric flow rates at the admission and exhaust branches, oxygen concentration and temperature are compared with experimental data. Overall, these predictions are in reasonably good agreement with the experimental data, closely following the experimental changing patterns. The ventilation flows and the wall heat transfer are two important features of the mechanically ventilated compartment fires, their model accuracies greatly affect the predictions and responsible for the discrepancies in the current simulation. To improve the accuracies of the proposed field model, a detailed ventilation model by solving the ventilation flow in the whole ventilation system and a CHT model based on empirical correlations on the convective coefficient needs to be implemented, if the detailed information of ventilation networks and large-scale fire tests on the empirical correlations is available.

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GLOBAL SENSITIVITY ANALYSIS USING EMULATORS, WITH AN EXAMPLE ANALYSIS OF LARGE FIRE PLUMES BASED ON FDS SIMULATIONS

Adrian Kelsey

Health and Safety Laboratory, Harpur Hill, Buxton, United Kingdom

ABSTRACT

Uncertainty in model predictions of the behaviour of fires is an important issue in fire safety analysis in nuclear power plants. A global sensitivity analysis can help identify the input parameters or sub-models that have the most significant effect on model predictions. However, to perform a global sensitivity analysis using Monte Carlo sampling might require thousands of simulations to be performed and therefore would not be practical for an analysis based on a complex fire code using computational fluid dynamics (CFD). An alternative approach is to perform a global sensitivity analysis using an emulator. Gaussian process emulators can be built using a limited number of simulations and once built a global sensitivity analysis can be performed on an emulator, rather than using simulations directly. Typically reliable emulators can be built using ten simulations for each parameter under consideration, therefore allowing a global sensitivity analysis to be performed, even for a complex computer code.

In this paper we use an example of a large scale pool fire to demonstrate an emulator based approach to global sensitivity analysis. In that work an emulator based global sensitivity analysis was used to identify the key uncertain model inputs affecting the entrainment rates and flame heights in large Liquefied Natural Gas (LNG) fire plumes. The pool fire simulations were performed using the Fire Dynamics Simulator (FDS) software. Five model inputs were varied: the fire diameter, burn rate, radiative fraction, computational grid cell size and choice of turbulence model. The ranges used for these parameters in the analysis were determined from experiment and literature. The Gaussian process emulators used in the analysis were created using 127 FDS simulations. The emulators were checked for reliability, and then used to perform a global sensitivity analysis and uncertainty analysis.

Large-scale ignited releases of LNG on water were performed by Sandia National Laboratory (SNL) in 2009. At the largest LNG release rate the flames did not cover the entire area of the LNG spill, this behaviour had not been observed in previous large-scale experiments. Also the height of the flames was also greater than expected from previous large-scale tests. One possible explanation for the observed behaviour is that in this very large-scale release the speed at which air and fuel vapour was drawn into the fire exceeded the flame speed. Therefore the flames could not propagate upwind to ignite the whole surface of the LNG pool. Fuel vapour from the unignited region, drawn into the fire, may also account for the higher flame height. A global sensitivity analysis allows the influence of uncertain parameters on the quantities of interest to be examined.

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INTRODUCTION

Developing and using models, sensitivity analyses are used to identify influential parameters. Once influential parameters have been identified care can be taken in finding or collecting data, to set the parameter value used in the model, or to ensure that a suitable representation is used in the model. A local sensitivity analysis, varying parameter values about a base-line can often identify significant parameters for this purpose.

In consequence modelling for safety analysis, for example, fire safety analysis, there is frequently interest across a range of possible operational conditions and there will often be uncertainty about the values of parameter that should be used. Rather than a local sensitivity analysis, considering the sensitivity at a single set of conditions, there is interest in the response over the full, global, range of conditions. In this situation a global sensitivity analysis can be used to identify the parameters that are important over the full range of operating conditions [1].

"Design of Experiments" [2] was developed to identify influential parameters in experiments. It has been applied to the sensitivity analysis of simulations from fire models [3] and shown to be able to produce useful information about the sensitivity of output quantities of interest using a small number of simulations using a fractional factorial design.

However, deterministic computer models, such as fire models, do not have the same behaviour as experiments. In particular repeating a simulation should result in the same result. One result is that if a parameter has no effect on an output of interest simulations that only vary in that parameter do not provide additional information. This has led to the extension of the "Design of Experiments" to the "Design and Analysis of Computer Experiments" [4].

Performing global sensitivity analyses directly, using a Monte Carlo approach is not possible unless the duration of model runs is very short. A Monte Carlo analysis may require thousands of model evaluations and the computing resource and time soon becomes prohibitive. One approach to this problem is the use of efficient designs, for example, extended-FAST (Fourier amplitude sensitivity test) [5] is a computationally efficient method to calculate main and total effect sensitivity indices. An alternative is the use of statistical emulators. In this approach a relatively small number of simulations are used to build an emulator. Once built an emulator can be used as a cheap way of making predictions at conditions for which simulations have not been performed. It can also be used as the basis for sensitivity and uncertainty analysis, calibration and optimisation. The emulator approach is more flexible in the information that can be obtained than extended-FAST or a factorial design for experiments. Building an emulator may require more simulations to be performed than a fractional factorial design. Loeppky et al [6] investigate and agree with a rule of thumb suggesting that a reliable emulator needs ten simulations per parameter. However, the resulting emulator can be used as a cheap predictor of the original model, in uncertainty and sensitivity analysis, calibration, and optimisation.

In this paper we describe performing an emulator based global sensitivity analysis using an example of large scale LNG fire plumes using FDS simulations, based on work described in Kelsey et al. [7].

GLOBAL SENSITIVITY USING AN EMULATOR

The approach to global sensitivity analysis described here is based on Bayesian Analysis of Computer Code Output (BACCO). O'Hagan [8] reviews this approach in an article providing a tutorial introduction. A software tool to perform global sensitivity analysis using the BACCO approach, GEM (Gaussian Emulation Machine), was developed by the research group of O'Hagan; the sensitivity analysis is described in Oakley and O'Hagan [9]. The GEM software uses a graphical interface to setup and perform the sensitivity analysis and is freely available for non-commercial use [10]. The interface can be used to create a design of simulations to perform, fit and check an emulator, and perform sensitivity analyses. The simulations themselves are not run from within GEM. Setting up input files, based on a design from GEM, running the simulations then extracting the results must be performed on suitable computers. GEM itself can be run on a laptop or desktop PC with no particular processor or memory requirements. The workflow of a complete sensitivity analysis involves a number of stages and frequently the use of more than one computer.

A Gaussian process based emulator fits a response surface for a specified output quantity over a number of input parameters. Simulations are performed at specified input values for each of the input parameters. Emulators can be built for different output quantities but each emulator will only predict a single quantity. Once built it is much cheaper to perform an evaluation of the emulator at new values of the input conditions than it would be to run the simulator, but the emulator is doing much less work than the simulator, only predicting a single quantity. An underlying assumption of the emulator is that the output quantity varies smoothly and is a continuous function of the input variables. At design points, where simulations have been performed, the output is known perfectly. Moving away from the design points the assumption of smoothness constrains the surface, but uncertainty about the actual value increases, see Figure 1. By contrast in simple Monte Carlo analysis the selection of each sample points is independent, taking no account of the points that have been sampled previously; this is one reason why an emulator can be more efficient than simple Monte Carlo sampling.

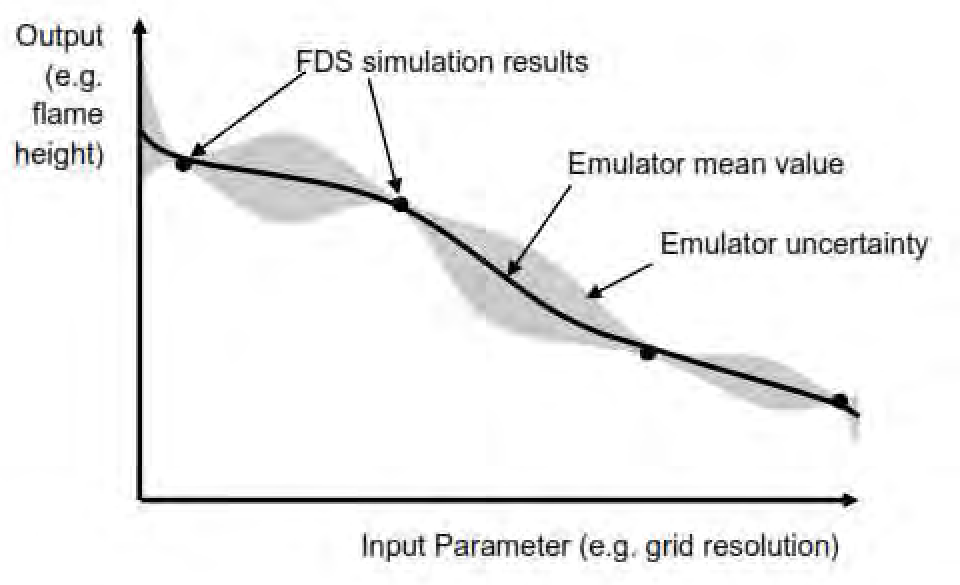


Figure 1 Illustration of Gaussian process emulator fit to training points

A design must be created to specify the input conditions at which simulations, used to build the emulators, must be performed. These designs are based on covering the range of input parameters without repeating any of the input values. This is one difference between Design of Experiment, where repeating values can give information about measurement uncertainty, and computer experiments, where repeating values can reduce the information obtained from simulations. A solution to this is to use Latin hypercube designs for the sample values; the GEM software provides two ways of generating Latin hypercubes. Maximin Latin hypercubes maximise the minimum distance between samples filling the design space evenly, reducing to equal spacing in one dimension. Even if some input variables have no effect on the output there is still good coverage of the remaining input variables, see Figure 2. In this design an optimisation is performed for a specified number of simulations and there is no simple way to increase the number of simulations in a design. Alternatively an LP-tau design can be used; this is a pseudorandom number sequence [11]. The LP-tau design is faster to generate and the design can be extended, filling in gaps in the existing design, but it does not fill the space as evenly as the maximin Latin hypercube. In models of physical systems it is often found that only a few parameters are active. This is why Latin hypercube designs, that do not repeat values, are useful, as when one dimension is removed from the design the input conditions of the simulations remain distinct.

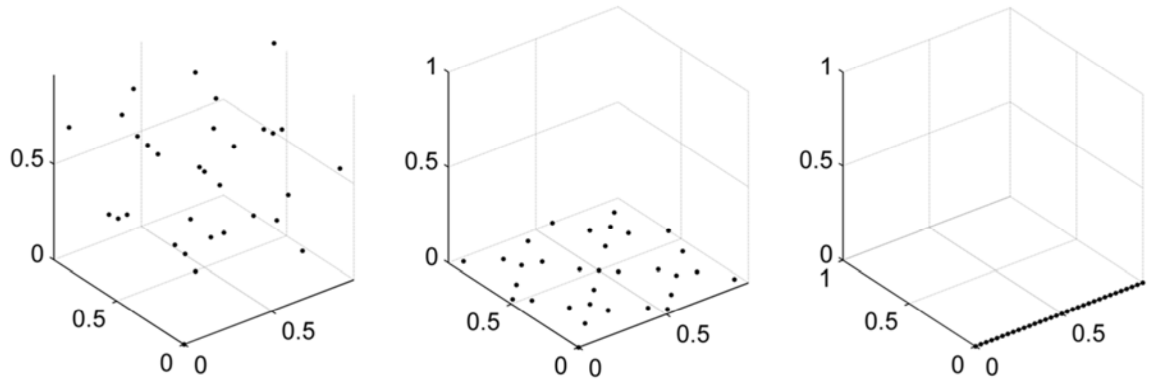


Figure 2 Maximin Latin hypercube design, reducing from three to one dimension

Once an emulator has been built the performance should be checked. The approach used in GEM is to perform a ‘leave-one-out’ cross-validation. Since the surface of a Gaussian process emulator passes through the predicted output values from all the simulations comparison of simulation and emulator cannot be used as a direct check on the fit of the emulator. Instead emulators are fitted leaving out the prediction from one simulation at a time, a ‘leave-one-out’ cross-validation, giving the same number of emulators as simulations performed. Each emulator is then used to predict the simulation that it does not include and the fit is checked.

Once an emulator has been fitted and checked the sensitivity analysis can be performed. GEM can perform two types of sensitivity analysis, variance based and mean based. The variance based analysis predicts main and total effects, the quantities predicted by an extended-FAST analysis [5]. The main effect of a parameter is the amount of the variance explained by that parameter alone; the total effect of a parameter is the sum of its main effect and all the possible interactions including that parameter. If the total effect of a parameter is equal to the main effect then no interactions involving that parameter contribute to the variance. If they differ then interactions involving that parameter do contribute to the variance. The sensitivity analysis can then be repeated including specified interactions to examine their effect. A variance based analysis can identify important parameters and interactions, but does not, by itself, explain how the parameters contribute to the variance. The other sensitivity analysis that can be performed using GEM is mean based. The mean value of parameters can be plotted across their range to examine how they change and hence how that parameter contributes to the variance.

APPLICATION OF SENSITIVITY ANALYSIS TO LARGE LNG FIRE PLUMES

An emulator based global sensitivity analysis was performed to investigate behaviour observed during the Phoenix tests performed by Sandia National Laboratory in the United States. These tests involved the largest LNG releases ever made; the LNG was released onto a large pool of water and ignited [12]. In the largest of the Phoenix tests only part of the LNG pool was covered by flames and the flame height was greater than expected. This behaviour is analysed by Betteridge et al [13] and a global sensitivity analysis is described in Kelsey et al. [7]. The outputs of interest in the sensitivity analysis were the plume flame height and the entrainment velocity into the base of the fire plume, as the velocity may have been high enough to limit flame spread and unignited vapour drawn into the fire plume may account for the greater than expected flame height being.

The fire plume simulations were performed using FDS version 6 [14], which is a freely available CFD code developed by the National Institute of Standards and Technology (NIST) in the United States. The model inputs used in the global sensitivity analysis are described in Table 1.

Table 1 Global sensitivity analysis inputs and their ranges

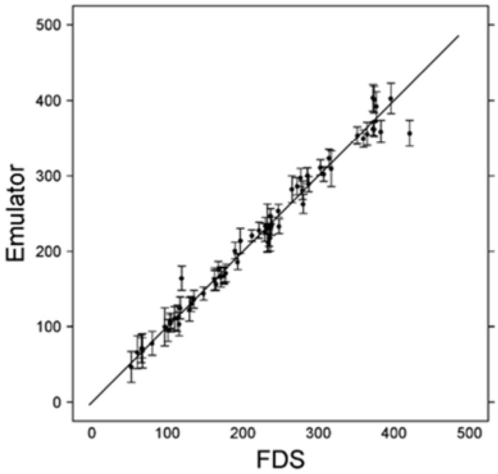
Model Input Parameter	Range
Pool fire diameter	10 – 100 m
Burn rate	0.05 – 0.5 kg m^{-2} s^{-1}
Radiative fraction	0.20 – 0.35
Grid resolution ($D^*/\Delta x$)	16 – 40
Turbulence model	Deardoff or Smagorinsky

The range of pool fire diameters considered covers both the smaller, 21 m, and larger, 83 m, LNG spill diameters in the Phoenix tests. In the former flames spread over the whole surface of the spill, in the latter the flames were limited to a region approximately 56 m in diameter. The range of burn rates examined is large, this allows for the possibility, considered in Betteridge et al. [13], that vapour from the whole of the LNG spill contributed to the burn rate in the smaller region where burning occurred. The other three parameters considered are related to the setup of the model. The radiative fraction is a pragmatic approach to the problem that the mesh is not sufficiently fine to resolve flame fronts and therefore maximum temperatures will be under-predicted. The lower end of the range corresponds to a clear flame, the higher a smoky hydrocarbon fire, the default in FDS. The range of grid resolution studied run from the top end of the range used for the Heskestad flame height in the FDS validation manual [15] to a value suggested by Chung and Devaud [16] as a good compromise between computing time and accuracy. All of these parameters are continuous; the final parameter though is not continuous, rather it is a “switch” between two different turbulence models. Smagorinsky [17] and Deardorff [18] are the default turbulence models in FDS versions 5 and 6 respectively.

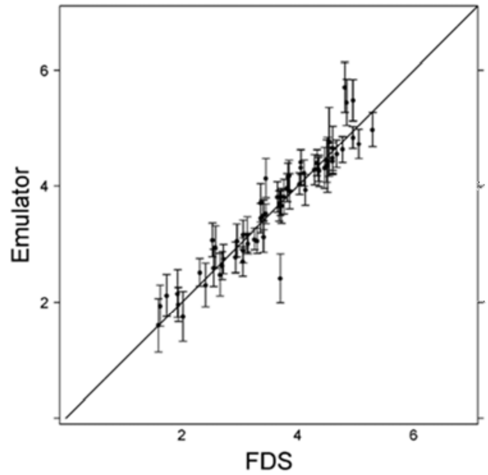
A 127 member LP-tau sequence was used for the design of simulation input conditions. Challenor [19] examined this type of design for looking at the influence of a switch in a global sensitivity analysis. The 127 member design is a Latin hypercube design with $2^m - 1$ members, where $m = 7$. The first $2^{m-1} - 1$ members and the remaining 2^{m-1} members are also each Latin hypercube designs. If one turbulence model is used for the members of the first Latin hypercube design, with 63 members, and the other turbulence model for the remaining 64 members of the 127 member design, then separate emulators can be fitted for the two turbulence models. This avoids potential problems with the smoothness and continuity assumptions underlying a Gaussian process emulator. The choice of a 127 member design allows a design containing two Latin hypercube designs and follows the rule of thumb of 10 simulations per parameter.

RESULTS

The results of a leave-one-out cross-validation are shown in Figure 3, indicating that the emulators are suitable for use in a sensitivity analysis.



a) flame height (m)



b) entrainment velocity (m s^{-1})

Figure 3 Cross-validation of emulators for the Smagorinsky turbulence model

The variance based global sensitivity analysis showed that fire diameter has the greatest effect on both flame height and entrainment velocity, accounted for over 75 % of the total variance.

The burn rate accounted for almost all the remaining variance for the flame height, with a small amount of interaction between fire diameter and burn rate. The remaining parameters contributed very little to the variance in the flame height.

Plotting mean emulator predictions of flame height, Figure 4, shows the influence of fire diameter and burn rate. The values of the grid resolution and radiative fraction are set at the middle of their ranges, as the variance based analysis showed these had little effect on the flame height. Emulator predictions for both turbulence models are plotted showing that they also have little effect on flame height, as indicated by the variance based analysis.

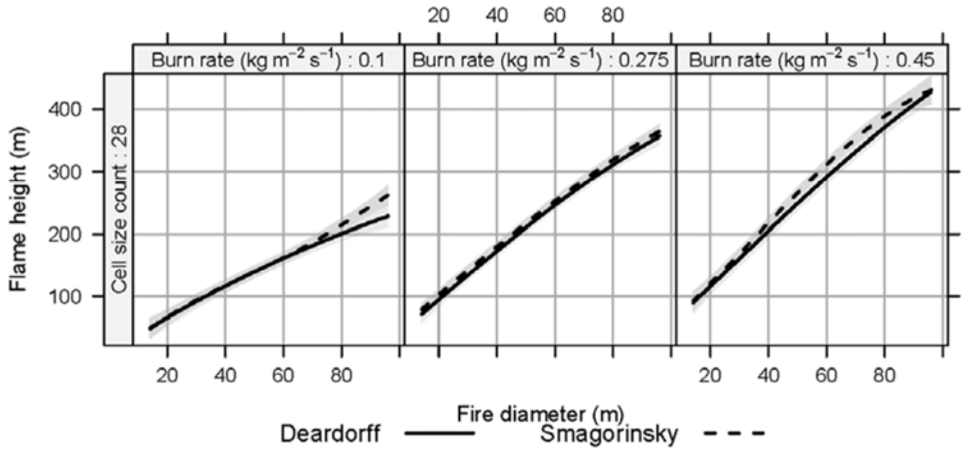


Figure 4 Emulator predictions of flame height against fire diameter, showing influence of fire diameter and burn rate

The variance in the entrainment velocity is dominated by the fire diameter but burn rate, grid resolution and the turbulence model also contribute to the variance. The effect of grid resolution is greater with the Deardorff than the Smagorinsky turbulence model.

Plotting mean emulator predictions of radial velocity, cf. Figure 5, shows the influence of fire diameter and burn rate, but also the effect of grid resolution and turbulence model. Predictions are shown for values of grid resolution close to the ends of the range used in the sensitivity analysis. These show that the predictions using both turbulence models converge at the higher grid resolution. The predictions made using the Deardorff turbulence model are more affected by grid resolution than those using the Smagorinsky turbulence model. This explains the greater contribution to the variance for the Deardorff turbulence model. Variance based sensitivity analysis shows that both flame height and entrainment velocity are most sensitive to the fire diameter and burn rate. The other parameters considered have much less influence. Though care must be taken to ensure adequate mesh resolution when predicting the entrainment velocity, the flame height shows less sensitivity to the mesh used.

A variance based sensitivity analysis by itself shows which parameters affect the flame height and entrainment velocity, but does not explain how they affect them. Plotting the mean response of the parameters shows how they respond to the sensitive parameters. The predicted flame height is sensitive to burn rate, and increases with increasing burn rate. An increased burn rate, due to vapour from non-burning regions of the LNG pool, could therefore increase flame height. The predicted entrainment velocity is sensitive to both fire diameter and burn rate. Betteridge et al. [13] note that Gottuk and White [20] report flame spread velocities of 2 m s^{-1} for liquid pool fires. The predicted entrainment velocities are higher than this value so tend to support the hypothesis that entrainment velocities could inhibit fire spread causing non-burning regions. However, further work would be required to quantify burning velocities in LNG pool fires to be sure that this was an explanation of the observed behaviour.

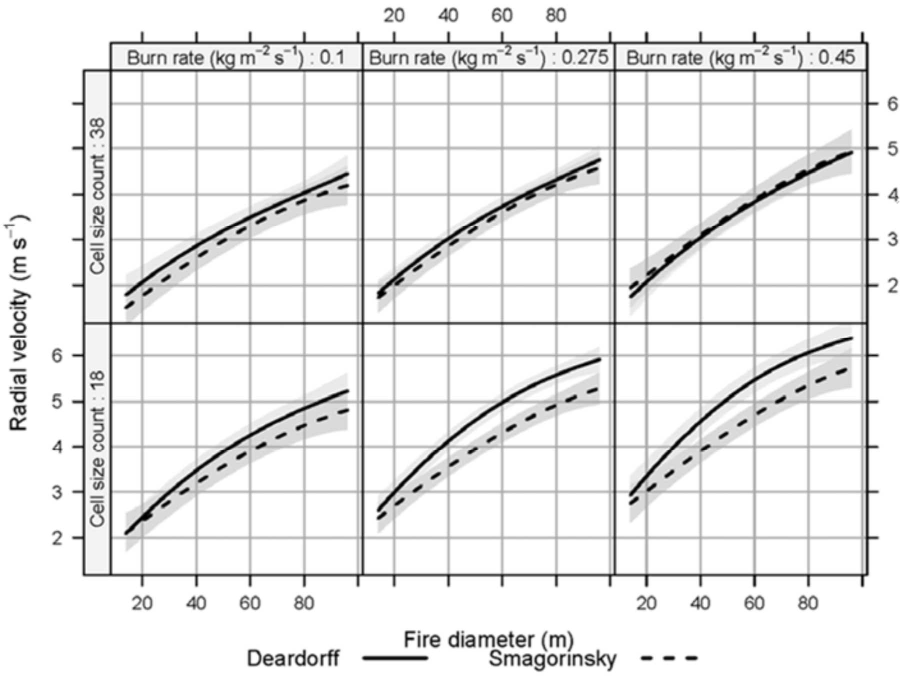


Figure 5 Emulator predictions of entrainment velocity against fire diameter, showing influence of fire diameter, burn rate, grid resolution and turbulence model

CONCLUSIONS

The combination of variance based and mean based sensitivity analysis, which is possible using emulator based sensitivity analysis, can be used to identify sensitive parameters and how they affect the quantities of interest. The analysis of fire plume behaviour shows that the fire diameter is the most influential parameter and that for large diameter fires the entrainment velocity into the plume could be sufficient to limit upwind flame spread, stopping the burning area from covering the whole of a large LNG pool.

Predictions of quantities of interest show different sensitivities to parameters. In the fire plumes studied both flame height and entrainment velocity are most sensitive to fire diameter followed by burn rate. However, the entrainment velocity also shows sensitivity to the turbulence model and grid resolution.

During the international OECD PRISME Project, mechanically ventilated multi-compartment fires were studied experimentally. These tests are complex; to capture the observed behaviour the interaction between mechanical ventilation and fire in the compartments must be modelled. To simulate these experiments with CFD a network ventilation model must be coupled with the CFD code, therefore increasing the number of physical and model parameters needed. Sensitivity analysis can be used to to identify important parameters and understand the effect that inputs have on output quantities. By identifying where effort should be concentrated the sensitivity analysis can help to inform the design of validation studies. Sensitivity analysis can also be used in the evaluation of consequence models and their uncertainties, complementing scientific assessment, verification and validation. Emulators allows global sensitivity analyses to be used with CFD models, where otherwise the number of simulations needed could not be performed, and therefore can contribute to the use of CFD in nuclear fire safety.

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SENSITIVITY ANALYSIS OF FDS 6 RESULTS FOR NUCLEAR POWER PLANTS

Daniel Alvear[1], Eduardo Puente[1], Orlando Abreu[1] and Julián Peco[2]

[1] Group GIDAI – Fire Safety-Research and Technology, University of Cantabria, Santander, Spain

[2] Consejo de Seguridad Nuclear, Madrid, Spain

ABSTRACT

The Spanish standard "*Instruction IS-30, Rev. 1*" (February 21, 2013) allows the new approaches of risk informed performance based design (PBD) The Spanish standard "Instruction IS-30, rev. 1" (February 21, 2013) for demonstrating the safe shutdown capability in case of fire in nuclear power plants. In this sense, fire computer models have become an interesting tool to study real fire scenarios. Such models use a set of input parameters that define the features of the physical domain, material, radiation, turbulence, etc.

This paper analyses the impact of the grid size and different sub-models of the fire simulation code FDS, version 6 with the objective to evaluate and define their relative weight in the final simulation results. For the grid size analysis, two different scale scenarios were selected, the bench scale test PENLIGHT and a large-scale test similar to Appendix B of NUREG - 1934 (17 m x 10 m x 4.6 m, with an ignition source of 2 MW and 16 cable trays). For the sub-model analysis, the PRS-INT4 real scale configuration of the INTEGRAL experimental campaign of the international OECD PRISME Project has been used.

The results offer relevant data for users and show the critical parameters that must be selected properly to guarantee the quality of the simulations.

INTRODUCTION

The work is based upon computational modelling of fires during the last decade. In the fire community, the use of computational fluid dynamics (CFD) has become usual and among all the codes developed, it is mandatory to highlight the LES (*L*arge *E*ddy *S*imulation) code Fire Dynamics Simulator (FDS) [1]. It has been released the sixth version of the code and it fights against important challenges since it is widely used for many fire scenarios. The range of spatial scales covered goes from millimetres to hundreds of meters.

When using the FDS code, a first question arises on the selection of a suitable mesh size: it must be little enough to resolve turbulence-related questions, but it must be also large enough to ensure a realistic time scale. For that purpose some stability of the solution (i.e. temperature, velocity) is needed to be ensured.

The selection of a mesh is the first step to get a numerical solution and has been widely treated in many documents; however in the field of modelling for nuclear power plants the main reference is the NUREG-1824 [2].

Another consideration to highlight is the dependence between input and output values when a criterion is fixed. From the results of this relation, the parameters that need a more precise evaluation to input the code are identified, and complementary, which parameters do not need such precision. In this sense, it is worth to highlight the work done by Hostikka [3]. He has created a model that combines Monte Carlo analysis and Computational Fluid Dynamics (CFD) modelling. Najm [4] used probabilistic uncertainty that propagates from input to output in a number of different CFD codes when the input has been characterized probabilistically. Recently, in the frame of the international PRISME (*Pr*opagation d'un *i*ncendie pour des *s*cénarios *m*ulti-locaux *é*lémentaires, French acronym for "Fire Propagation in Elementary Multi-Room Scenarios") Project launched by the OECD (*O*rganisation for *E*conomic *C*o-operation and *D*evelopment) Nuclear Energy Agency (NEA), Suard et al. [5] have applied a factorial design to study sensitivity of the fifth version of the FDS code.

However, NUREG-1824 [2] evaluated the results of the fourth version of the code, which had a different finite difference term to resolve the flow model and therefore, we deal with the evaluation of the sixth version of FDS for different values of the cell size, which is very important to use the code properly.

GRID SENSITIVITY

In this study, different values of the cell size have been tested using different types of heat sources. The temperature was used as the criterion to get a value that quantifies the difference between both base and test cases.

Following [2] or the User´s Guide of FDS, Section 6.3.6, we have a first approach to evaluate the mesh size. In this approach, the characteristic cell size, *Dz*, is related to the characteristic diameter of fire, *D**. In these documents, values of the ratio *D*/Dz* between *4* and *16* were tested and recommended.

$$D^* = \left(\frac{\dot{Q}}{\rho c_p T \sqrt{g}}\right)^{2/5} \tag{1}$$

where $\dot{Q}$ is the heat release rate [J/s], ρ, c_p and *T* are respectively the density [kg/m^3], specific heat [J/kg.K] and temperature of ambient air, and g [m/s^2] is the acceleration of gravity field.

$$MSE(T_{test}) = \sqrt{\frac{\sum_t (T_{test}(t) - T_{base}(t))^2}{(N-1)}} \tag{2}$$

$$\varepsilon = \frac{MSE(T_{test})}{\overline{T_{test}}} \tag{3}$$

Where $T_i(t)$ is the temperature of the case *i* at time *t*, and *N* the number of values of temperature for each case performed. ε is the squared relative error and $\overline{T_{test}}$ is the average temperature.

Penlight Scenario

The airdrop configuration of the Penlight tests [6] was used as scenario to test grid sensitivity because it is a single geometry composed by a cylindrical oven and a cable in the symmetrical axe of the cylinder. In this configuration, only radiation has influence on heating the surface of the cable. Nonetheless, we are interested in testing two configurations of the cable: on the one hand, an inert cable inputted by using both the PART and the OBST instructions of the code. On the other hand, the same scenario was tested with a reactive cable and pyrolysis effects were included.

The cylindrical oven has length of 0.8 m and a diameter of 0.51 m. Cable temperature was evaluated 2 mm deeper into the cable as stated in the validation tests performed by the developers of FDS during the PRISME Project [7]. For the grid analysis, the same temperature program as the LOWER case studied in the validation examples was used. Air temperature did not show any change of the cell sizes tested and therefore it was not further discussed. The computational domain was a cube of 0.6 m x 0.45 m x 0.45 m.

For the non-reactive cable, a plastic material is used, with a density of 1959 kg/m^3, a conductivity of 0.2 W/m.K and a specific heat, and 1500 J/kg.K, respectively.

Table 1 and Table 2 provide a summary of cases studied and the results obtained.

Table 1 Tests cases with non-reactive cable

Case	No. of cells	Size (shape)	Code
1	970	5 cm (cubic)	PART
2	150	10,9,9 cm (parallelepiped)	PART
3	7760	2.5 cm (cubic)	PART
4	970	5 cm (cubic)	OBST
5	150	10,9,9 cm (parallelepiped)	OBST

The results were compared for each criterion to a different base case. Thus, the first column of the results shows the relative squared error regarding the base case with a cell size of 0.025 m. The second column points out the differences between PART and OBST cable characterizations. The results were compared against two base cases of cell sizes 0.05 m and 0.1 m.

Table 2 Results of non-reactive cable cases

Case	ε Squared	ε OBSTvsPART
1	3.97 %	BASE
2	5.67 %	BASE
3	BASE	-
4	4.25 %	0.57 %
5	6.38 %	0.99 %

Figure 1 shows the temperatures obtained at 2 mm depth inside the cable from numerical treatment. In green colour, we can see the PART results, while black colour characterizes the OBST results.

From the results obtained, it can be observed that there are no differences between OBST and PART descriptions at the same cell size (below 1 %). However, for different grid sizes, the squared relative differences are in the order of 4 %, if the mesh size is multiplied by 2, and around 6 %, if it is multiplied by 4.

It can be observed that some properties of the curves are quite different, for example the temperature peak reached by the cable from coarse cells (0.1 m) is at least 30 ºC higher than those obtained from cell sizes of 0.05 and 0.025 m.

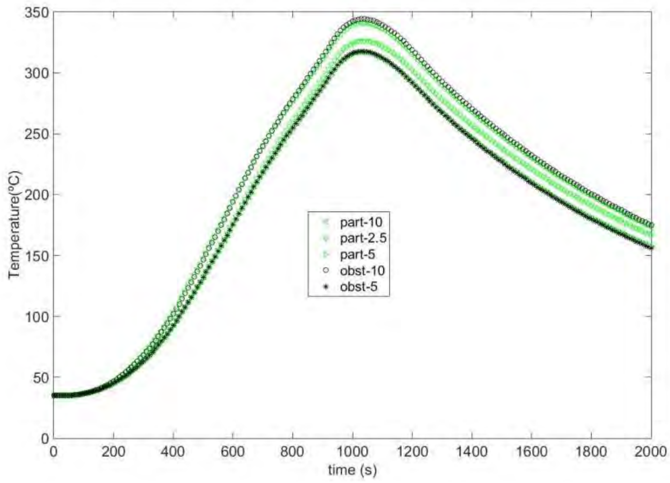


Figure 1 Cable temperature during numerical simulation

For reactive cables, a different definition of the main components of the cable were used, PVC. The definition of the PVC components was made by the parameters provided in Table 3.

Table 3 Properties of reactive cable components

Property	Value
Density	1200 kg/m^3
Specific heat	1370 J/kg.K
Conductivity	0.12 W/m.K
Emissivity	0.21
N_Reactions	1
Nu_Spec	1
Spec_Id	PVC_surrogate
Reference Temperature	257 ºC
Heat_of_Reaction	1500 J/kg

The PVC_surrogate had a heat of combustion of 16400 J/kg.

Table 4 Tests cases with reactive cables

Case	No. of cells	Size (shape)	Code
1	7760	2.5 cm (cubic)	PART
2	970	5 cm (cubic)	PART
3	150	10,9,9 cm (parallelepiped)	PART
4	150	10,9,9 cm (parallelepiped)	OBST
5	970	5 cm (cubic)	OBST
6	7760	2.5 cm (cubic)	OBST

Table 5 demonstrates that the error, when comparing characterizations (PART vs. OBST), has a different behaviour regarding to the no reactive cable. It seems that there are some differences between both if the size of the cell decreased enough (3 % at cell size of 0.025 m). It is likely because the PART characterization was created in the development of THIEF [8] model and for that model, the cable must be non-reactive. However, for squared relative error, there is the same trend as for the non-reactive case. Higher cell size corresponds to higher squared error but it shows a difference with regard to reactive cables, the mesh with a cell size of 0.05 m seems to be more accurate.

Table 5 Results of reactive cable cases

Case	ε Squared	ε OBSTvsPART
1	BASE	BASE
2	5.39 %	BASE
3	12.33 %	BASE
4	12.76 %	0.28 %
5	2.27 %	0.24 %
6	BASE	3.18 %

Figure 2 demonstrates that the shape of the temperature curve inside the cable was similar for the cases of cell size 0.05 m and 0.025 m; however, the coarse mesh (cell size 0.1 m) predicts a maximum temperature higher than the finest meshes and shows a thermal increase faster than these ones.

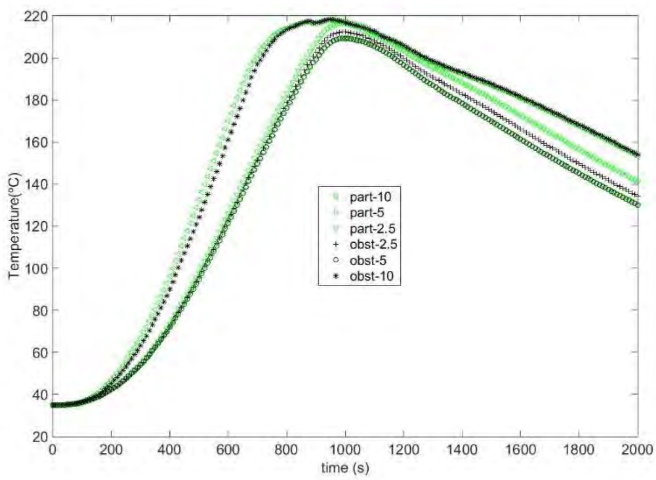
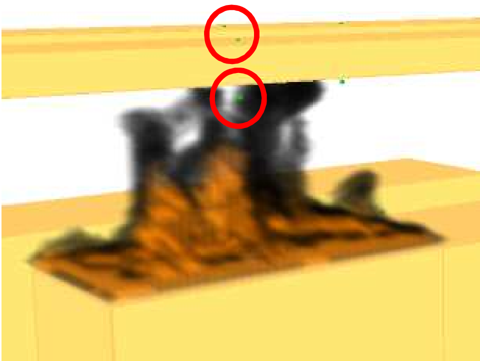


Figure 2 Cable temperature during numerical simulation

Switchgear Room Scenario

The sensitivity analysis in a scenario accounting for fire sources which Equation 1 can describe completes the study of the grid influence to the results. In that sense, a simulation of a typical nuclear power plant room was perfomed, a switchgear room [8] with electrical cabinets and cable trays. In this scenario, neither non-reactive cables nor the PART characterization were considered.

The room has a size of 17 m length x 10 m width x 4.6 m height. Inside the room, 16 cable trays, 12 electrical cabinets, an exhaust duct, 3 vents and a fire source of 2 MW are installed. The analysis includes environment temperature and cable temperatures 2 mm in depth of the cable for two locations in the room. Figure 3 shows these locations highlighted with red circles on the right side (top view), one in the vertical direction of the fire source (Figure 3 on the left side) and the other at a location that has no direct contact with the plume. Cable trays have two components: the metallic structure and the cable bundle. In fact, the flame does not affect the reactive cable directly.



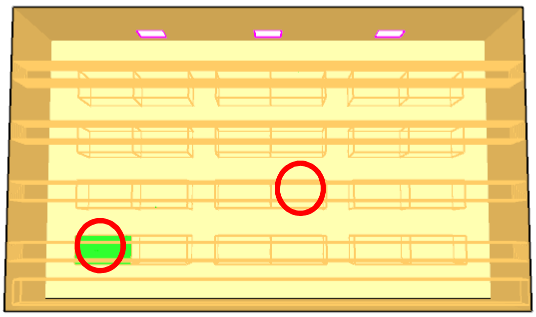


Figure 3 Fire source above a cabinet (left), top view of the switchgear room (right)

The heat is transferred to the cable by conduction through metallic structure and convectively through environmental gas.

Table 6 summarizes the characteristics of mesh sizes tested with the evaluation of the parameter *D*/Dz* that is in the range between 6 and 26. The base case to evaluate the differences corresponds to the finest mesh (cell size 0.05 m).

Table 6 Mesh characteristics and simulation times

Case	$D^*/\Delta z$	Cell size	# of cells	Time
1	6.3	0.2 m	102236	4h27’
2	8.4	0.15 m	242338	11h42’
3	12.6	0.1 m	818890	36h09’
4	25.3	0.05 m	6543110	428h47’

Table 7 provides the results obtained during numerical treatment.

Table 7 Results from a switchgear room scenario

Case	ε gas plume	ε cable plume	ε gas	ε cable
1	24.61 %	24.89 %	6.65 %	14.14 %
2	13.72 %	15.98 %	5.37 %	10.75 %
3	8.77 %	1.56 %	2.26 %	1.84 %
4	BASE	BASE	BASE	BASE

The squared relative error values from temperature in the plume are significantly higher than those ones reported away from the fire source. In all the temperatures evaluated, one can see how an increase in the cell size involves less accurate temperature values during the simulation. It seems that cable temperature prediction is worse than gas temperature except for the plume evaluation. However, when a fire source is included, the gas temperatures needs a grid sensitivity analysis, too.

PARAMETER SENSITIVITY

In order to analyse the sensitivity of parameters for a case of interest in nuclear power plants the PRISME INTEGRAL test number four (PRS-INT4), see Figure 4, has been used. For that purpose, a simple method denominated “*Elemental Effect Met*hod” [9] has been applied. For each parameter, a 5-fold vector within the range of values indicated in Table 8 was test-ed. Since the parameters belong to different sub-models of the FDS code, the family upon which they are related is also specified.

The fire source is a pool of heptane with an equivalent diameter of one meter located in the fire room (upper compartment in the centre). For the analysis of results, we deal with the maximum average temperature (three places of measurement for each compartment, a, b, c, as indicated in Figure 4). The ventilation has three nodes, two of them are air inlets located in the room 1 (upper left in Figure 4) and in the corridor (compartment down), the third one is an extraction node located in the third room (upper right). Rooms 2 and 3 have an insulation of rock wool to protect the walls against heat from the pool.

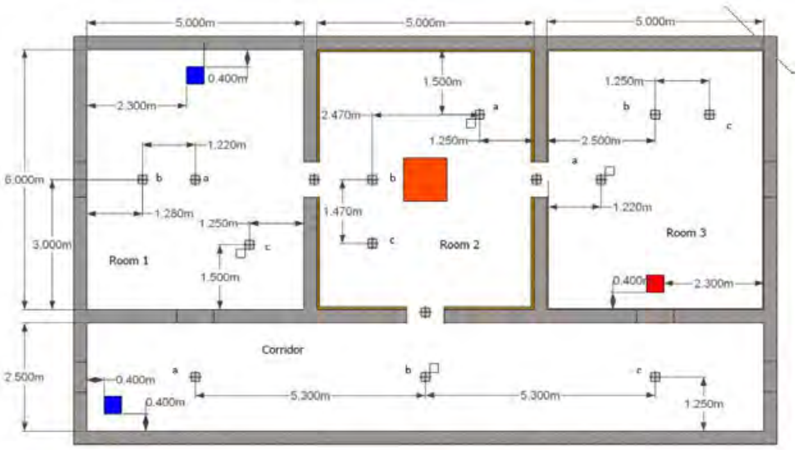


Figure 4 Top view of PRS-INT4 test

Table 8 Range of values for sensitivity analysis

Parameter	Units	X_{min}	X_{max}	Family
Soot yield	-	0.01	0.10	Fuel gas
CO yield	-	0.06	0.15	Fuel gas
Specific heat	J/kg.K	600	1300	Insulation
Conductivity	W/m.K	0.04	0.24	Insulation
Emissivity	-	0.75	0.98	Insulation
Mass loss rate	$kg/m^2.s$	0.014	0.064	Pool
Heat of vaporization	J/kg	150	500	Pool
Radiative fraction	-	0.25	0.50	Fuel gas
Volume flow	m^3/s	0.59	1.00	Ventilation

Cubic cells of a characteristic size of 0.1 m composed the mesh and 41 simulations were performed to do the sensitivity analysis. The reference value of each parameter was the mean value of the range shown in Table 8. Figure 5 shows the temperature gradient profiles in locations a (black triangle), b (red triangle) and c (blue triangle) when the maximum temperature is reached in the fire room.

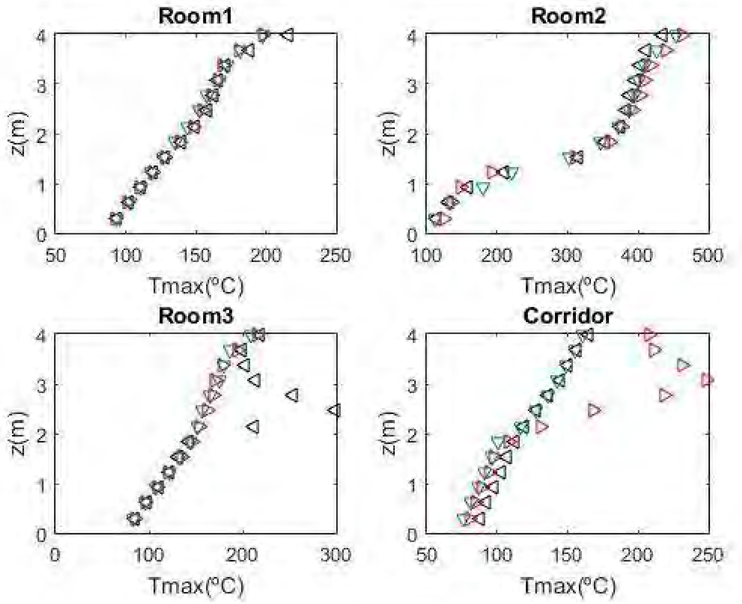


Figure 5 Profiles of gradient of temperature in the rooms

It is noticeable that maximum temperatures reached in the room 3 and in the corridor, correspond to the thermocouple trees located in front of the doors that communicate such compartments with the fire room. In the fire room, it is observed temperatures higher than 450 ºC and a gradient profile close to the ideal one if the compartment is divided into two independent vertical zones. Room 1 seems to show a little influence from the fire room, but this is because there are no trees of thermocouples in front of the door that connects room 1 with the fire room. However, temperature reached at 2 m high is about 150 ºC. Lowest temperatures are reached in the corridor except for the tree of thermocouples directly located in front of the fire room.

The gradient of temperatures in the vertical line of each door, see Figure 6, is also analysed. Due to ventilation, the highest temperatures are observed to be reached at the door that connects room 2 and room 3.

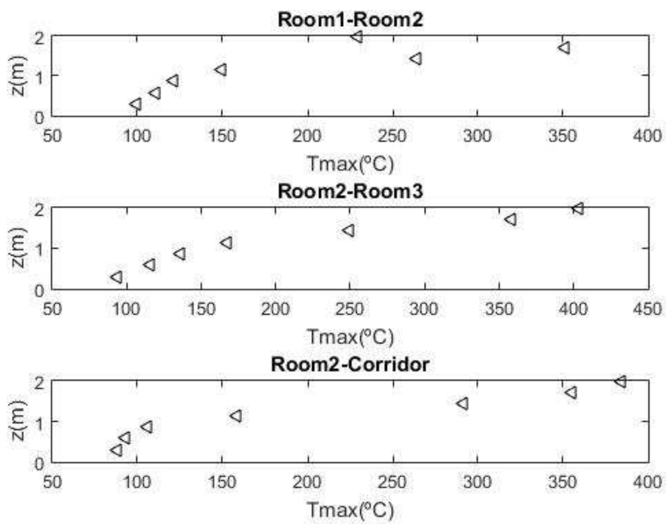


Figure 6 Profiles of gradient of temperature in the doors

Finally, the variation of the maximum average temperature for the fire compartment by evaluating the sensitivity coefficient is shown.

$$\delta(\overline{T_{max}}) = \sum_{i=1}^{5} \delta_i(\overline{T_{max}})/5 = \sum_{i=1}^{5} \frac{X_i}{\overline{T_{max}}} \frac{\Delta \overline{T_{max}}}{\Delta X_i}/5 \qquad (4)$$

where $\overline{T_{max}}$ is the maximum average temperature and X_i is the value of the parameter in Table 8 for each case.

From the results given in Table 9, one can see that those parameters related to combustion of gas showed different behaviours: the radiative fraction, as a prescribed input related directly to the energy released, has a positive influence of 0.09 but soot and CO yields have only influence in the species concentration and therefore have no influence on the temperature. Among the parameters related to insulation, only the conductivity has some influence on the energy and thus, on the temperature. The sensitivity coefficient for this parameter is negative, which is coherent to the fact that a better insulation prevents heat losses to the outdoor compartment.

Table 9 Results of sensitivity analysis

Parameter	$\delta(\overline{T_{max}})$ fireroom
Soot yield	no influence
CO yield	no influence
Specific heat	no influence
Conductivity	- 0.08
Emissivity	no influence
Mass loss rate	0.62
Heat of vaporization	- 0.02
Radiative fraction	0.09
Volume flow	- 0.58

The parameters related to liquid phase, mass loss rate and heat of vaporization, show different responses both qualitatively and quantitatively. The first one has a positive sensitivity coefficient with the highest value of the parameters analysed, the second one shows a weak and negative relation between input values and temperature results. Finally, the ventilation parameter shows a negative role, contributing to the cooling of the gas through the continuous extraction of hot gases and injecting cool air. However, due to the ventilation conditions the flames are being maintained.

CONCLUSIONS

The grid sensitivity has been tested for two fire scenarios: The first scenario has no fire sources but a temperature profile prescribed, and it has laboratory scale (dimensions lower than one meter. The second one has a strong fire source of 2 MW and dimensions of a real fire scenario in a nuclear power plant.

From the mesh sensitivity study it has been concluded that if radiation is the main factor to heat transfer, gas temperature is accurate enough with significantly large cell sizes. However, the temperatures of the other compartment components need a sensitivity analysis with smaller cell sizes. On the other hand, if there are fire sources and thus plume effects, the sensitivity of gas temperature against mesh size needs to be analysed. If that is not the case, larger errors may occur and, because of that, large real scenarios need to be carefully treated when mesh size is selected. For example, a common cell size when large areas are simulated is 0.2 m, however as results show, this one could be of little accuracy if the characteristic fire diameter is about a car fire (2 MW).

Regarding parameters sensitivity, it is important to fix the goals of simulation since they provide the criteria under which the sensitivity of the model is being analysed and for that, we will know which parameters have to be used with special care and which ones can be avoided from the analysis. The parameters that have shown to be more sensitive from the analysis of the PRISME test facility are those related to heat release rate and mass flow ventilation in case that temperature prediction is the goal. However, it is necessary to analyse each criterion in terms of sensitivity to achieve the safety goals.

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FOCUS ON THE STUDIES IN SUPPPORT OF FIRE SAFETY ANALYSIS: IRSN MODELLING APPROACH FOR NUCLEAR FACILITIES

Julien Espargillière, Raphaël Meyrand, Thierry Vinot

Institut de Radioprotection et de Sûreté Nucléaire (IRSN),
Fontenay-aux-Roses, France

ABSTRACT

For a fire safety analysis, in order to comply with nuclear safety goals, a nuclear fuel facility operator has to define the elements important for safety to be maintained, even in the case of a fire. One of the key points of this fire analysis is the assessment of possible fire scenarios in the facility. This paper presents the IRSN method applied to a case study to assess fire scenarios which have the most harmful effects on safety targets. The layout consists in a central room (fire cell) containing three glove boxes with radioactive material and three electrical cabinets. This room is linked to two connecting compartments (the fire cell and these two compartments define the containment cell) and then to two corridors. Each room is equipped with a mechanical ventilation system, and a pressure cascade is established from the corridors to the central room. A fire scenario was studied with fire ignition occurring in an electrical cabinet. This scenario has a set of safety goals (prevention of fire cell and containment device failure, propagation of the fire). This case study was conducted with the IRSN code SYLVIA based on two zones modelling. Safety goals were associated with key parameters and performance criteria to be fulfilled. Modelling assumptions were defined in order to maximize physical effects of the fire. Sensitivity studies were also conducted on key parameters such as oxygen limitation, equivalent-fuel definition. Eventually, a critical analysis of the code models was carried out.

INTRODUCTION

Over several years now, the French nuclear fire safety regulation [1] and [2] has turned from prescriptive to objective requirements. Consequently, the fire safety analysis (FSA) focuses now on the compliance of fire effects with performance criteria for fire protection measures. These performance criteria are mainly related to the vulnerability of targets (for instance, containment equipment) within nuclear fuel facilities in order to avoid accidental sequences potentially compromising facility safety functions and leading to a significant release of nuclear materials in environment. To assess this demonstration in its FSA, the licensee has to ensure that nuclear safety objectives are fulfilled in the case of accidental fires. A key step in this analysis process is the identification of the targets associated to the elements important for safety to be maintained. These targets may include structural elements, all types of nuclear systems and various components important to nuclear safety. Furthermore, the protection of employees who have to ensure safety operations for nuclear facilities must also be considered. The last step of the analysis consists in the evaluation of the effects of a fire on targets, performed with numerical simulations.

To illustrate this method, a case-study of a typical containment cell of a nuclear fuel facility is presented below. This cell contains both electrical equipment and radioactive materials. For this case-study, the fire safety analysis highlights a scenario in which fire starts in an open-door electrical cabinet facing the glove boxes.

In order to assess the fire consequences for this scenario, a series of fire simulations was carried out with the SYLVIA fire simulation code [3] developed by IRSN (*I*nstitut de *R*adioprotection et de *S*ûreté *N*ucléaire). This fire code is based on a two-zone modelling approach [3] and [4] to calculate smoke and heat transfers from fire source to other compartments or ventilation network and the impact on targets. The results from theses computations allowed the accurate identification of all the important safety equipment that could potentially be damaged by fire consequences. To properly perform these fire simulations and predict the relevant target damages, IRSN needs to establish fire properties and rupture criteria of containment equipment by means of both experimental tests and literature data:

- fire source characteristics (heat release rate (HRR), fire growth, mass loss rate, combustion products, etc.) based on open fire tests representative of fire scenarios in facilities,
- rupture criteria of containment equipment (gloves, doors, pipes, etc.) due to pressure and heat stresses,
- fire spreading criteria for electrical cabinets or glove boxes.

After this introduction, a complete description of the case-study is detailed. Then, some experimental tests performed at IRSN are presented concerning the determination of characteristics of electrical cabinet fire, glove box fire and rupture criteria of containment equipment. From these experimental outcomes and a dataset (geometry, ventilation network, etc.) describing the case-study, the criteria related to the safety issues, the fire modelling and the key assumptions needed for the calculations by means of SYLVIA fire code are detailed. A discussion is proposed about the computations results before concluding this paper.

CASE STUDY DESCRIPTION

The case study is a typical containment cell of a nuclear fuel facility. This cell contains both electrical equipment and radioactive materials. A full description of the fire cell is presented in Figure 1 and Figure 2.

The fire compartment includes a row of three electrical cabinets which powers equipment important for safety. This room also includes three glove boxes containing radioactive materials and facing the cabinet row at a distance of 1.5 m. This room is linked with the rest of the facility by two connecting zones. In a simplified manner, the rest of the facility is represented by two large corridors. The fire compartmentation only includes the fire compartment and the fire resistance rating according to the standard nominal fire curve from ISO-834 [5] being 90 min. Every room is equipped with a nuclear ventilation system. To prevent radioactive material dispersal, inlet air vents are located near the ceiling and exhaust air vents near the floor. The containment compartmentation consists in the room itself and the two connecting zones; the pressure drops between rooms have to be kept to ensure the dynamic containment.

The glove boxes are composed of four metal panels and two opposite “working” panels of LEXAN (polycarbonate polymer, 10 mm thick) with a biological protection in leaded PMMA (polymethyl methacrylate, 50 mm thick). Four glove holes in each “working” panel allow operators handling. One of these “working” panels is facing the cabinet row. The glove box ventilation is made of vertical PVC pipes routed from the ceiling.

In case of fire, a safety interlock closes the fire room dampers on ventilation inlets and exhausts after fire detection. A detection time of 2 min 30 s after the ignition is assumed. The ventilation of connecting zones and corridors is maintained.

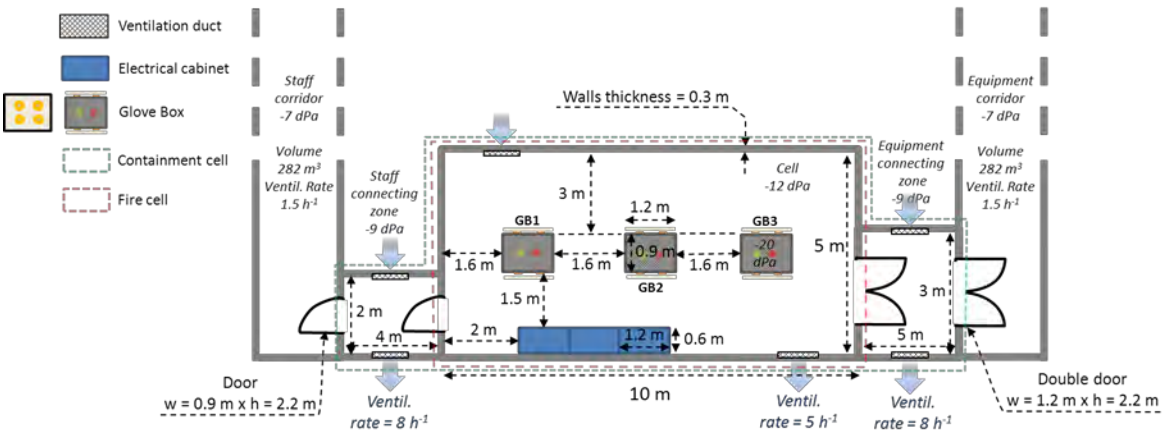

Figure 1 Top view of the cell

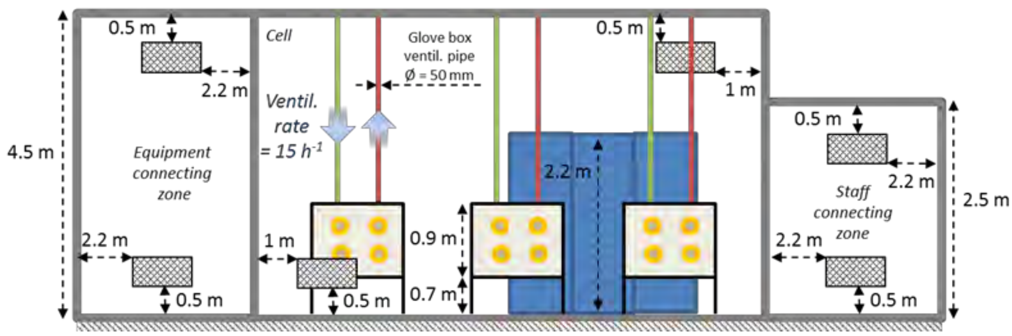

Figure 2 Front view of the cell

IRSN EXPERIMENTAL TESTS FOR DETERMINING THE CHARACTERISTICS OF FIRE AND CONTAINMENT EQUIPMENT

In order to simulate properly a fire scenario with a fire code, the first stage consists in the determination of the fuel source fire properties. Consequently, open atmosphere fire tests concerning real electrical cabinets were performed under a large-scale calorimeter [6]. Heat release rate, incident radiant heat fluxes in front of the cabinet and combustion products were measured. Two configurations were mainly investigated in open atmosphere:

- open-door cabinets allowing the fire to freely growth along wires and components inside cabinet,
- closed door cabinets with two square openings on each door (one on the upper part and the other one on the bottom part of door) limiting the oxygen consumption.

For real electrical open-door cabinets, a significant quantity of smoke appeared just after ignition and the flame spread slowly from the bottom to the top, all along the electrical components. A few minutes later, the fire was fully developed leading to a powerful fire. In this configuration, all the combustibles burnt. Just after ignition, for real closed door cabinets, smoke was observed exiting from the upper ventilation openings. Moreover, flames could also appear through these openings [7]. Sometimes, puffs of smoke could exit through the lower ventilation openings [6]. Depending on material nature, combustible load and opening sizes, the fire could quickly extinguish by lack of oxygen [6], [8] leading to a weaker amount of material pyrolysis in comparison with real open-door cabinet. More technical details and main outcomes about these fire tests are available in [6], [7] and [8].

An IRSN experimental program is in progress on the issues of resuspension radioactive materials in case of glove box fire involving powder of PuO_2. As a part of this program, fire tests involving various configurations of 1 m^3 glove box (number of combustible panels) were performed in open atmosphere under a large-scale calorimeter in order to determine the fire behaviour of a glove box (heat release rate, combustion products, etc.). The main outcomes show strong heat release rate fires (up to roughly 3.5 MW) with a fast fire grow and an important release of soot in spite of the reduction of the number of combustible panel. Moreover, a very important radiative flux of the flame in front of the glove box is measured. The analysis of the main outcomes of these tests is in progress and technical details will be published during 2016 [9].

Concerning rupture criteria, IRSN performed an experimental program to investigate the behaviour under thermal stress of some combustible equipment in nuclear facilities (such as glove boxes equipment or PE waste drums). The tests were performed under a hood using a radiative panel which gives a defined incident heat flux. For each device, a series of tests were performed: a constant heat flux, varying from 1 kW/m² to 60 kW/m², is applied during a maximum period of 30 min (1800 s). The experimental observations focus on the occurrence time of relevant thermal events: first plastic deformation, glass transition and self-ignition. For each event and device, two methods are used to determine its critical (i.e. minimal) heat flux of degradation:

- a semi-empirical method, based on 1D heat transfer laws and an extrapolation of experimental data;
- a fully-experimental method, consisting in an interpolation of the experimental data.

More technical details and main outcomes about these tests are available in [10].

Because of a lack of data on the behaviour of containment equipment under pressure stresses representative of a fire scenario in a facility, IRSN builds an experimental facility, called STARMANIA. The main objective of the STARMANIA facility is to determine the mechanical strength which is the differential pressure value corresponding to the equipment failure but also to determine the aeraulic resistance [11]. Four experimental studies were led: two on fire dampers and two on fire doors. The results obtained during these tests have permitted to determine the aeraulic behaviour and the rupture pressure of the equipment [11]. More technical details and main outcomes about these tests are available in [11].

SCENARIO DESCRIPTION

Assuming maintenance work, electrical cabinet facing the glove boxes has its doors opened and the other cabinets have their doors closed. Fire ignites in the open-door cabinet and the fire room ventilation system is closed after fire detection. The ventilation of connecting zones and corridors is maintained. The fire spreading from a cabinet to the adjacent one is modelled with the recommended approach of NUREG/CR-6850 [12], i.e. assuming a fire propagation delayed by 15 min.

Based on the safety goal of preventing radioactive release (i.e. prevention of fire and containment cell failure: glove box, pressure cascade system, door, filter, etc.), a safety analysis has to identify the related safety issues and associated safety criteria which come from both IRSN experimental tests and literature data. A major assumption in this work ensures conservative results, i.e. a material ignites or breaks immediately when the critical flux or temperature is reached.

The computations related to this scenario shall address the four following issues:

- **Issue a:** Does cabinet fire cause a loss of containment of glove boxes?

  Gloves rupture (named [G] in the results tables) is assumed if a thermal flux of 2 kW/m² [10] or a temperature of 85 °C [13] is reached. Ventilation PVC pipes rupture (named [V] in the results tables) is assumed if a thermal flux of 4 kW/m² [10] or a temperature of 125 °C [13] is reached.
- **Issue b:** Does fire spread from cabinets to glove boxes?

  Due to lack of data, leaded PMMA is assumed to behave like PMMA. Therefore, an ignition flux of 15 kW/m² [14], a material ignition temperature of 250 °C [14] for PMMA (named [P] in the results tables) or an ambient gas temperature of 500 °C, corresponding to flashover conditions (named [F] in the results tables), are assumed. The fire spread to working panel is assumed if PMMA panel ignites. If fire spreads to glove boxes, the effects of these secondary fires must be taken into account.
- **Issue c:** Does fire cause a failure of fire compartmentation?

  The calculations have to determine if a fire can last longer than 90 min (named [D] in the results tables) or if the gas temperature exceeds the standard nominal fire curve ISO-834 (named [G] in the results tables).
- **Issue d:** Does fire cause a failure of containment cell?

  The computations have to determine if a fire cause a pressure cascade inversion (named [R] in the results tables) or a double-door rupture (named [D] in the results tables). These rupture is assumed if door pressure gap exceed 18 hPa [11].

FIRE SIMULATIONS: NUMERICAL TOOLS, ROOMS AND FIRE MODELING

Numerical Too

The SYLVIA software system [3], developed by IRSN, is designed to simulate the fire growth and its consequences in an industrial facility featuring a ventilation network. SYLVIA particularly calculates the development of the fire, the transportation of hot gases and soot, the transportation of aerosols (whether radioactive or not), the clogging of filters, the environmental conditions inducing the failure of electrical equipment and the mechanical damage on fire barriers such as firebreak doors and fire dampers. Based on a two-zone modelling, each compartment is divided into two zones of variable volume, in which the thermodynamic properties are uniform (temperature, combustion products, etc.). The ventilation network is modelled using a set of elements, such as ducts, filters, valves, fans, etc. Mass and heat exchange correlations (between zones, flames and walls) supplies the mass and energy balance equations performed in each zone. This software is especially designed to perform low time consuming simulations that are required for safety assessments.

Modelling the Fire Source: How to Model Fire Spreading from One Electrical Cabinet to Adjacent Ones

For this study, the fire source is considered to be a row of three electrical cabinets. The first cabinet of the row, where fire breaks out, is assumed open-door. For other cabinets, their doors are closed. Based on realistic approach, the time evolutions of HRR for open and closed-door cabinets are taken from the IRSN fire tests which could lead to the most harmful effects on safety issues (CA02 test in [8] for closed door cabinet and PXA.3.2 in [6] for open-door cabinet). Moreover, the fire spread between cabinets is assumed all along the cabinet row. Consequently, the total heat release rate sums the different HRR for each single cabinet already in fire. The fire spread from a cabinet to the adjacent one is modelled with the recommended approach of NUREG/CR-6850 [12], i.e. assuming fire propagation delayed by

15 min (i.e. 900 s). The special cases dealing with the absence of fire spreading between two cabinets are very complex and hard to connect with real fire scenarios. So, it is assumed that the fire spreads in any case (conservative approach).

Based on these assumptions, the time-evolutions of heat release rates are presented in Figure 3. During the first 15 min, the HRR curve (blue curve) follows exactly the HRR from the experimental open door cabinet fire (green curve) showing a HRR peak at about 1.1 MW at 710 s. After 900 s, the fire propagates to the adjacent cabinet. In the same time, the HRR begins to decrease slowly from about 400 kW due to both HRR decrease of open-door cabinet and HRR increase from the closed-door cabinet. Then, 900 s yet later, the next cabinet ignites and the HRR remains constant. Of course, this simple approach can be used a priori for any number of cabinets.

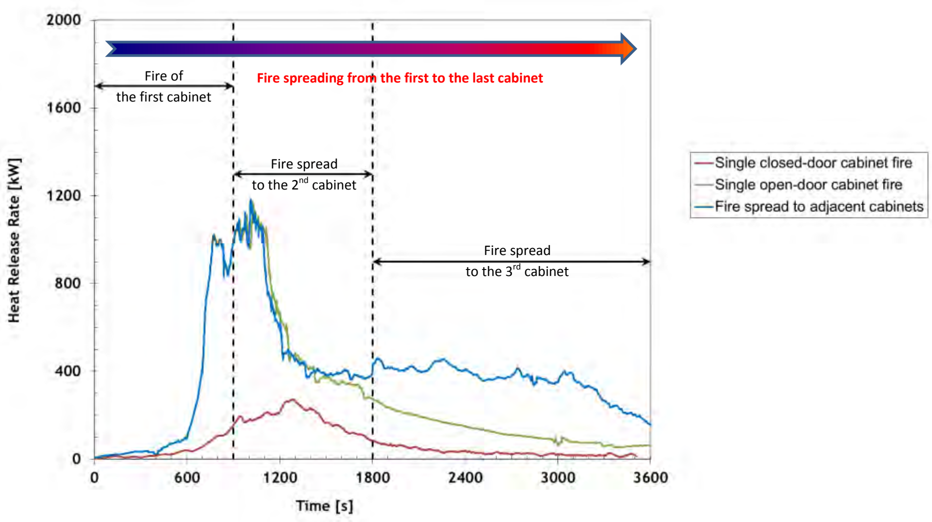

Figure 3 Modelling of heat release rate (blue curve) for three electrical cabinets fire (green curve: single open-door cabinet; red curve: single closed-door cabinet)

In addition, the fire is modelled as a pool fire in the SYLVIA code. The effect of oxygen depletion in the room, affecting the pyrolysis rate and the fire duration, is simulated using empirical models built for pool fire. The effect of confinement involves that the fire could extinguish below a given oxygen threshold. Here, a sensitivity study is performed with two oxygen limit laws:

- a Lower Oxidant Limit (LOL) model which assumes a sudden fire extinguishment below a defined oxygen threshold. Oxygen thresholds of 8 %, 10 % and 12 % in volume are considered in the computations;
- a Peatross and Beyler model [15] which assumes a linear decay of pyrolysis rate with oxygen concentration. An oxygen threshold of 11.5 % in volume is considered in the computations. This threshold is deduced from IRSN full-scale experiments of an open-door cabinet fire inside a mechanically-ventilated compartment [16].

Concerning the simplified combustion reaction defined for fire simulation, the major products of combustion are introduced following the chemical reaction hereafter:

$$\text{Fuel} + Y_{O2}\, O_2 \rightarrow Y_{CO2}\, CO_2 + Y_{CO}\, CO + Y_{H2O}\, H_2O + Y_{soot}\, C$$

In this equation, Y_{O2}, Y_{CO2}, Y_{CO}, Y_{H2O} and Y_{soot} are respectively the mass rate of oxygen consumption, the mass rate of production of carbon dioxide, carbon monoxide, water vapour and soot. Combustion products and oxygen consumption yields are determined from both experimental data [6], [7] and [8] and mass balance.

Modelling Rooms and Ventilation network

A simplified approach considers that the inlet and exhaust of the ventilation network of each room are modelled as a fixed pressure boundary condition. Leakage resistances of closed-doors and closed dampers are taken from IRSN aeraulic experiments [11]. This simple model accurately assesses the fire development and propagation of hot smoke in the rooms and takes into account the success or failure of safety actions such as ventilation management. A sensitivity study is also performed to take into account the failure of the fire detection, implying that the computations were conducted with either closed or maintained ventilation.

REVIEW OF SUPPORT STUDIES FOR FIRE SAFETY ANALYSIS

From the comprehensive modelling of fire scenarios detailed in the previous paragraphs, a set of eight fire simulations were performed with SYLVIA code and the safety conclusions deduced from the numerical outcomes were summarized in Table 1.

	O_2 limit law	Issue a	Issue b	Issue c	Issue d
Closed ventilation	**LOL 8 %**	Yes [G][V]	No	No	No
	LOL 10 %	Yes [G][V]	No	No	No
	LOL 12 %	Yes [G][V]	No	No	No
	P&B 11.5 %	Yes [G][V]	No	No	No
Maintained ventilation (detection failure)	**LOL 8 %**	Yes [G][V]	No	No	No
	LOL 10 %	Yes [G][V]	No	No	No
	LOL 12 %	Yes [G][V]	No	No	No
	P&B 11.5 %	Yes [G][V]	No	No	No

Table 1 SYLVIA computation results

Concerning glove boxes containment, all the computations conclude that the containment is lost in case of an open-door cabinet fire. In terms of safety, it implies that a fire protection is requested. However, due to the distance of 1.5 m between the open-door cabinet and the facing glove box, the absence of fire spreading to glove boxes is surprising. Indeed, fire radiation in SYLVIA code is modelled with a point source model. This limits the accuracy of the radiation model in the vicinity of fire sources and a solid flame model would be more appropriate to evaluate radiative heat fluxes for short distances. To fix this limitation, SYLVIA computed HRR and flame height were used as input data of a solid flame model to assess a relevant radiative heat flux. For a LOL oxygen limit law, this last method predicts that fire could spread to glove boxes in case of an open-door cabinet fire. These results are consistent with measured heat fluxes of open-door cabinet fires [6]. For Peatross and Beyler oxygen limit law, the solid flame model computation predicts no fire spread. A reason could be that the SYLVIA predicted heat release rate with a Peatross and Beyler model is underestimated for all of the open-door cabinet fire tests performed in a confined atmosphere [**16**]. The validity of the Peatross and Beyler model, which is based on liquid pool fires, could be unsatisfactory for such a complex solid fire source. Investigations are under progress at IRSN to find a more appropriate model for complex solid fire sources. So safety conclusions will be only deduced from LOL computations. For these cases, computations were rerun considering glove boxes ignition at the time predicted with the solid flame model. Table 2 summarizes the safety conclusions deduced from the numerical outcomes of these last computations.

	O_2 limit law	Issue a	Issue b	Issue c	Issue d
Closed ventilation	**LOL 8 %**	Yes [G][V]	Yes [P]	No	Yes [R]
	LOL 10 %	Yes [G][V]	Yes [P]	No	Yes [R]
	LOL 12 %	Yes [G][V]	Yes [P]	No	Yes [R]
	P&B 11.5 %	Yes [G][V]	No	No	No
Maintained ventilation (detection failure)	**LOL 8 %**	Yes [G][V]	Yes [P]	No	Yes [R]
	LOL 10 %	Yes [G][V]	Yes [P]	No	Yes [R]
	LOL 12 %	Yes [G][V]	Yes [P]	No	Yes [R]
	P&B 11.5 %	Yes [G][V]	No	No	No

Table 2 SYLVIA computation results taking into account "solid flame" conclusions

In case of an open-door cabinet fire, an incident heat flux greater than 4 kW/m² is quickly reached on gloves or pipes and the glove box containment is lost. Then, during the cabinet fire HRR peak, an incident heat flux of 15 kW/m² is reached on the facing glove box panel. It implies that fire spreads to the glove box panel and the cabinet fire is followed by a glove box fire. In this case, all the computations show gas temperature lower than standard nominal fire curve and fire duration does not exceed 90 min. Therefore, the fire cell is properly designed for thermal effects. However, all the computations predict a pressure cascade inversion during the glove box fire which means that containment cell is not properly designed. In terms of safety, design improvements shall be considered: action on ventilation, implementation of an automatic fire extinguishing system.

CONCLUSIONS

Based on literature data, SYLVIA software tool and experimental tests concerning both electrical cabinet fires, glove box fires and investigations about rupture of containment equipment, IRSN proposed some original methods to perform calculations in order to assess fire scenarios in the framework of fire safety analysis in the French nuclear fuel facilities. This study proposes:

- a simple and conservative approach for modelling the fire spread between cabinets based on fire tests carried out previously for open and closed-door cabinets [7], [8] and [9];
- a full set of safety criteria (in accordance with safety issues) to take into account the rupture of containment equipment due to fire effects. These criteria come from both IRSN experimental tests and literature data.

The numerical outcomes of this study were obtained with SYLVIA fire code. However, the accuracy of these results must be taken into account before drawing any conclusion for safety analysis. Indeed, in the case-study, the impact on close targets has to be carefully assessed and may have to be rerun due to the limitation of the point source model or the relevance of the oxygen limit law.

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FIRE ANALYSIS: RELEVANT ASPECTS FROM SPANISH NUCLEAR POWER PLANTS EXPERIENCE

Pedro Fernández[1], Tómas Villar[1]

[1] Empresarios Agrupados A.I.E., Nuclear Safety Department, Madrid, Spain

ABSTRACT

Empresarios Agrupados A.I.E. leads the development and updating of fire analysis for the Spanish NPP's. Some of them decided to voluntarily adopt standard NFPA-805 [1] as an alternative to the current fire protection rules. Fire Probabilistic Risk Assessment (PRA) methodologies have been continuously evolving during recent years. This paper will briefly present experience gained in relationship with some relevant aspects of fire risk analysis.

Associated circuits need to be evaluated to determine if cable faults can prevent or cause the maloperation of redundant safety related systems. If a circuit is not properly protected by an isolation device, fire damage to a cable could propagate to other safe shutdown cables. In order to check that the coordination is adequate, existing electrical protections coordination studies have been analyzed and, for some plants, additional analyses have been performed for DC and AC for instrumentation an control (I&C) systems.

Spurious actuations are also a basic part of the analysis of the consequence of a fire, which should consider any possible actuation that can prevent or affect the performance of a system or safety function. In this context, it was furthermore necessary to take into account the possibility of a combination of several spurious actuations that can result in a specific consequence, according to Appendix G of NEI 00-01 Rev. 2 [2]. These are the so-called Multiple Spurious Operations (MSOs).

One key element in fire analysis is the availability of validated fire models used to estimate the spread of fire and the failure time of cable raceways. NFPA 805 [1] states that fire models shall only be applied within the limitations of the given model. The applicability of the validation results is determined using normalized parameters traditionally used in fire modeling applications. Normalized parameters assessed in NUREG-1934 [3] may be used to compare NPP fire scenarios with validation experiments. If some of the parameters do not fall within the range of the study, further justification must be made. Additionally, in order to compare the results and to check that the semi-empirical equations described in NUREG-1805 [4] lead to conservative results, some calculations for a few fire zones were performed using the more accurate CFD model "Fire Dynamics Simulator" (FDS).

As a result of Human Reliability Analysis, more complete and accurate information about equipment and instrumentation that can fail due to a fire in the different fire areas will be provided to the operators in order to assure them about the equipment they can rely on in case of fire, thus helping them respond better to a plant transient caused by the fire.

As part of the transition process to NFPA-805, an estimation has been performed of the increase in risk associated with non-compliance with Appendix R of 10 CFR 50 [5], for both Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), in order to determine its acceptability within the criteria established in the Regulatory Guide 1.174 [6].

CONTENT OF THE PAPER

INTRODUCTION

Fire Probabilistic Risk Assessment (PRA) methodologies have been continuously evolving during recent years.

Empresarios Agrupados, A.I.E. leads the development and updating of fire analysis for the Spanish NPP's. Some of them decided to voluntarily adopt standard NFPA-805 [1] as an alternative to the current fire protection rules.

- Almaraz I/II NPP. Analyzes performed as part of the transition process to NFPA 805 [1]. Peer review for fire PRA performed in February 2011.
- Ascó I/II NPP. Analyzes performed as part of the transition process to NFPA 805 [1]. Peer review for fire PRA performed in June 2015.
- Vandellós II NPP. Analyzes performed as part of the Periodic Safety Review.
- Trillo I NPP. Analyzes performed as part of the Periodic Safety Review.

The baseline methodology described in NUREG/CR-6850 [7] has been used to perform the fire Probabilistic Risk Assessment, taking into account the Frequently Asked Questions that have been released over these years and also the Supplements of NUREG/CR-6850 [7] (general fire PRA methodology) and NUREG-1805 [4] (fire growth calculations).

Some relevant aspects derived from this experience are highlighted with relation to associated circuits of concern, fire models, operation and increase in risk calculations.

EVALUATION OF ASSOCIATED CIRCUITS

Both NEI 00-01 [2] and Generic Letter 81-12 [8] define associated circuits of concern as those cables and equipment that have a physical separation less than that required by Section III.G.2 of Appendix R [5], and have either: a) a common power source with the shutdown equipment and the power source is not electrically protected from the circuit of concern by coordinated breakers, fuses, or similar devices, b) a connection to circuits of equipment whose spurious operation would adversely affect the shutdown capability and c) a common enclosure with the shutdown cables not electrically protected with the possibility of not preventing propagation to it.

This type of circuits needs to be evaluated to determine if cable faults can prevent or cause the maloperation of redundant safety related systems.

For the circuits related to cases a) and c), in order to check that the coordination is adequate, coordination studies on electrical protections have been analyzed and, for some plants, additional analyzes have been performed for DC and AC for instrumentation & control systems.

These analyzes help ensure that the electrical protections on the different loads of the plant are appropriate and coordination between different levels of protection works adequately with regard to AC and DC for instrumentation & control systems.

These studies show that the impact of circuits that share a power source or electrical raceways is adequately contemplated in the design, and that no further action is therefore considered necessary within fire analyses.

For the circuits in case b), the analysis of each fire area identifies all the affected components, either because they failed when performing their active function or because they may undergo spurious operation.

Fire-induced spurious operation is an event where the fire causes damage to the cables so that a given component changes to an undesired status that prevents or affects the performance of one or more safety functions. Since fires can affect several cables, the occurrence of more than one spurious operation during a fire can be contemplated.

Combinations of spurious operation in different systems that could affect one or more safety functions have been taken into account.

The concept of multiple spurious operations basically involves several spurious operations to coincide, which may occur under different conditions.

Reference document NEI-00-01 Rev. 2 (Appendix G) [2] provides a list of multiple spurious operation (MSO's) scenarios that fall into one of the above groups. These lists are based on the experience at similarly designed plants and are used as a basis for a systematic review of combinations of spurious operations that yield scenarios that differ from the ones considered.

Spanish nuclear power plants have been analyzed using the above document as a basis to identify any potential scenarios that have not been included in the analyzes. Their applicability and potential impact have been analyzed.

In some plants, the applicability of the various multiple spurious operation (MSO) scenarios has been addressed in expert panels formed by PRA, electrical, instrumentation and mechanical engineering specialists, as well as by operating personnel.

These analyses have yielded the following results:

- Several scenarios were already included in the PRA models;
- Some scenarios have been discarded because of the large number of spurious actuations required or because of specific assumptions or calculations;
- Some scenarios have required additional malfunctions that were not initially included in the models to be considered;
- Some scenarios have called for certain additional initiating events apart from those initially considered in the models;
- Some scenario resulting from the impossibility of accepting local recovery actions on motor-operated valves.

VALIDATION OF FIRE MODELS (FDT's)

One key element in fire analysis is the availability of validated fire models to estimate the spread of fire and the cable raceways failure time. This time is related to the time available for fire suppression.

The calculations performed for the fire PRA analysis were based on the methodology described in NUREG/CR-6850 [7] and on the semi-empirical equations and correlations explained in NUREG-1805 [4] (FDT's Fire Dynamics Tools).

NFPA 805 [1] states that fire models shall only be applied within the limitations of the given model. The applicability of the validation results is determined using normalized parameters traditionally used in fire modeling applications. Normalized parameters assessed in NUREG-1934 [3] "Nuclear Power Plant Fire Modeling Analysis Guidelines" may be used to compare NPP fire scenarios with validation experiments.

These parameters are the following:

- Fire Froude Number: related to fire diameter and thus to plume temperature and flame height;
- Flame length ratio: related to the "size" of the fire relative to the height of the zone;
- Ceiling Jet Distance Ratio which measures the ceiling jet position;
- Equivalence Ratio: which relates the heat release rate and the ventilation rate;
- Compartment Aspect Ratio which measures the general shape of the zone;
- Radial Distance Ratio: related to the radiative heat flux calculations.

If not all of the parameters fall within the range of the study, additional justification is necessary to be made, by means of performing sensitivity analysis. These analyzes refer to varying selected input parameters in the "conservative" direction so that they fall within the applicability range. If the conclusions (fire growth and CDF of the fire scenario) are not affected by the variations in the parameters, the sensitivity analysis results may be used to further justify the conclusions.

The most relevant parameters outside the validation range for the fire growth calculations performed for fire PRA's were shown to be the Fire Froude Number and the Radial Distance Ratio.

For the former, the results of the validation for the heat release rates of the most common fire sources such as vertical cabinets (fire limited to only one cable bundle and fire in more than one cable bundle), electrical fires in pumps, motors, ventilation subsystems, transformers and transient combustibles are shown in Table 1 below.

Table 1 Results of the validation in terms of the Fire Froude Number for the typical Heat Release Rates of fire sources

HRR [kW]	Minimum Valid Diameter [m]	Maximum Valid Diameter (m)	Diameter Used in the Model [m]	Valid
32	0.171	0.350	0.600	No
69	0.233	0.475	0.600	No
142	0.310	0.636	0.600	Yes
211	0.364	0.745	0.600	Yes
317	0.428	0.876	0.600	Yes
702	0.588	1.204	0.600	Yes

This means that the calculations for fires in motors, transformers and ventilation subsystems (for both severity factors 75 % and 98 %) and in one cable bundle vertical cabinets (for 75 % severity factor) are outside the validation range.

A sensitivity analysis for these scenarios was carried out using a fire diameter of 0.350 m and 0.475 m, respectively.

For the latter, the scenario would be within the validation range if the heat flux calculation radius falls between 1.32 m and 3.42 m, assuming a fire "size" of 0.6 m.

Likewise, another sensitivity analysis for the few scenarios outside the validation range for the radial distance ratio was performed, assuming the failure of the cable trays involved in the conservative direction.

Both sensitivity analyzes lead to a decrease in the time available for fire suppression, because the flame height increases due to the decrease in the fire diameter, and thus the time in which the flame reaches the target diminishes, on the one hand, and if the target raceways in the heat flux heat are deemed to be damaged in the conservative way, the time available for fire suppression decreases too. The typical reduction in the time available was shown to be approximately one minute or less.

This might be relevant because a reduction of for example 40 seconds in the time available would lead to a 6 % increase in the non-suppression probability and so in the CDF.

The results of the evaluation of the impact of these analyzes once validation criteria were used show that the increase in the total CDF is low, on the order of 1 – 2 %.

This means that the conclusions of the fire PRA do not change significantly, but the results of the validation should be taken into account for the future revisions of the documentation.

Nevertheless, a draft report of the Supplement 1 of NUREG-1824 [9] “Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications” released on November 2014 includes additional test data for validation of the models and suggests that the range of validation could widen, so that all calculations would meet the validation range.

COMPARISON OF FDS AND FDT’S FIRE GROWTH CALCULATIONS

The fire propagation calculations aim to determine the raceway (cable trays) failure time that could be affected by the fire. As shown before, the calculations were based on the methodology described in NUREG/CR-6850 [7] and on the semi-empirical equations and correlations explained in NUREG-1805 [4]. These calculations are based on very conservative in many aspects empirical mathematical formulations, and do not include a realistic methodology for the calculation of some key parameters of the fire growth.

To complement this, additional calculations for a few fire zones were made using the advanced fluid dynamics computer code "Fire dynamics simulator" (FDS), which can perform more accurate estimations of the consequences of a fire.

The goal of these calculations was to compare the results and to check that the semi-empirical equations described in NUREG-1805 [4] lead to conservative results. The parameters of the fire used in the simulations as well as the time evolution of HRR and damage criteria were the same as those used in manual calculations (FDT's).

In some cases, the purpose of the simulations was to compare FDS tray damage times with those obtained from the spreadsheets (FDT’s).

The results of one of those scenarios are shown in Table 2 and Figure 1 below.

Table 2 Comparison of tray damage times obtained by FDT calculations to those obtained by FDS simulations

Parameter	FDS Result	FDT Result
Damage time of tray A	9 min	4.5 min
Damage time of tray B	11 min	5.5 min
Damage time of tray C	13 min	9.5 min
Damage time of tray D	15 min	10.5 min
Damage time of tray E	16 min	8.5 min
Damage time of tray F	22 min	11.5 min
Damage time of tray G	No damage	14.5 min
Damage time of tray H	No damage	14.5 min
Damage time of tray I	No damage	14.5 min
Damage time of tray J	No damage	13.5 min
Damage time of tray K	No damage	15.5 min
Additional HRR due to secondary targets (trays)	248 kW	2747 kW

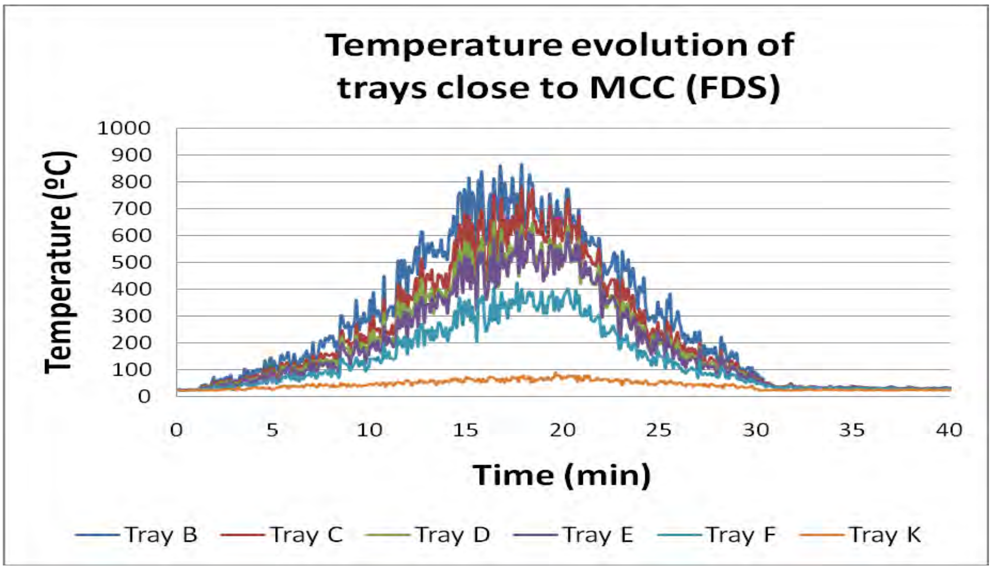

Figure 1 Temperature development of trays close to a MCC (FDS)

In some other cases, the purpose of the simulations was to check if a fire could propagate from a fire zone to the adjacent zone. The results show that the temperatures in the hot gas layer in the adjacent zone for all scenarios simulated reach approximately 200 ºC for a short period of time, below the damage criteria for qualified cables.

As a consequence of these simulations, some conclusions can be drawn:

- Greater cable tray damage times for FDS simulations;
- No additional damage to secondary targets;

- Duration of the fire less than 45 minutes;
- Self-extinction;
- No propagation to adjacent fire zones.

INFORMATION GIVEN TO THE OPERATOR AS A CONSEQUENCE OF FIRE ANALYSIS

Fire risk analysis analyzes in detail equipment and cables that can be affected by a fire in a given fire area. This is valuable information that can be used not only to develop the fire analysis, but also as a help for the operators in the control to discriminate which equipment can fail in a fire area and to identify which indications can be given credit and which indications could give false or no indication.

Following this idea at one plant, an AOP (*A*bnormal *O*perating Procedure) called “fire in some area of the NPP” was developed in collaboration with the fire PRA team. A similar approach has been adopted at the rest of the plants.

The purpose of the AOP is to describe the symptoms and actions to be taken by the operation staff if a fire in any area of the plant takes place. It also provides information to address the loss of a safety function, system or safety equipment and possible equipment outages due to fire. First of all, after confirming the presence of the fire, it is necessary to assess its magnitude and to form the fire brigade.

A key aspect to managing a fire is to identify its location. Because of that, it is compulsory to check if the fire began in the containment building, the main control room or somewhere else. Once the fire is located, it must be dealt with. For this purpose, an area action sheet was developed for each fire area. Operators were given feedback for this. These sheets tell the operator which equipment and instrumentation could be affected by the fire and the EOP’s related to them.

If the fire affects safety systems, it is necessary to check the equipment which could be damaged by the fire and therefore the equipment and instrumentation which the operator can rely on. The information can be found in the area action sheets of the AOP.

After that, operation strategies are needed in order to achieve and maintain safe shutdown checking if the fire could prevent it. If the fire affects equipment needed for plant operation, in the same way, strategies will be established.

Once the fire is extinguished, the state of the plant must be evaluated and, if necessary, actions are to be taken using the proper procedure of the plant.

INCREASE IN RISK CALCULATIONS

For those nuclear power plants which decided to voluntarily adopt standard NFPA-805 as an alternative to the current fire protection rules, it is necessary to analyze and justify compliance with chapter 4 of NFPA-805 [1]. The baseline methodology described in FAQ 08-0054 [10] has been used for this purpose.

Once fire areas that meet the deterministic criteria were identified, and CDF (core damage frequency) / LERF was calculated, an increase in risk was calculated for fire areas which do not meet the deterministic criteria, i.e. the risk resulting from variations from the deterministic requirements (VFDR). For those fire areas screened out in the fire PRA and for those areas whose CDF is negligible, it is assumed that the increase in risk will be also negligible. For the rest, a safe shutdown path for each fire area was chosen, as basis for the VFDR.

In a first step, the safe shutdown functions free of damage for each fire scenario have been determined. After that, the paths free of damage and those affected by the fire were compared with the base defined for the VFDR. If all the safe shutdown functions are guaranteed in the fire scenario, the increase in risk is zero. If not, it is assumed conservatively that the increase in risk will be the CDF of the fire scenario.

In this context, it should be noted that the increase in risk calculations are made for each fire area. In that area, some fire scenarios may meet the deterministic criteria and others will not. The results of the increase in risk calculations are used to determine its acceptability within the criteria established in the Regulatory Guide 1.174 [6] shown below. The increase in risk associated with non-compliance with Appendix R of 10 CFR 50 [5] is acceptable, as it falls into Region II for CDF and Region III for LERF. But in this approach, the design modifications proposed as a consequence of the transition to NFPA-805 were not taken into account for the CDF/LERF calculation. If these design modifications are taken into account, the new CDF/LERF was obtained and it turned out to be lower than the CDF/LERF, assuming compliance with Appendix R.

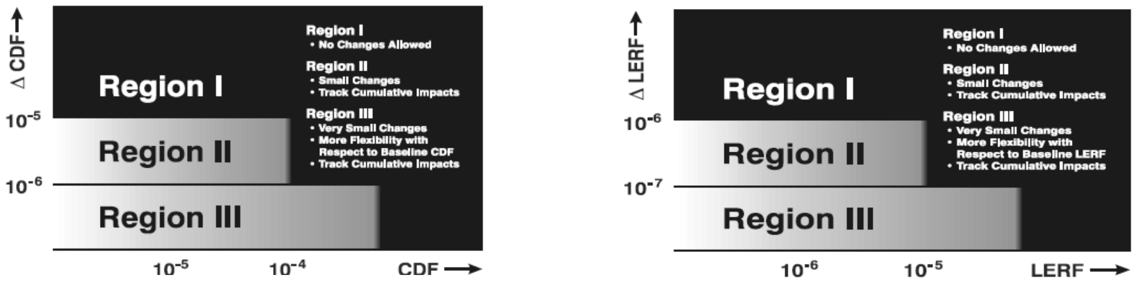

Figure 2 Acceptance guidelines for CDF and LERF

CONCLUSIONS

As a consequence of the experiences derived from the fire analyses for Spanish nuclear power plants, the following conclusions have been drawn:

- Spurious operation of equipment, including the analysis of Multiple Spurious Operations (MSOs) plays a key part in fire risk analysis.
- One key element in risk-informed/performance based fire protection is the availability of verified and validated (V&V) fire models that can reliably estimate the effects of fires.
- Fire risk analysis provides valuable information to help operators deal with transients scenarios caused by fires.
- The fire risk analysis team must keep in close contact with experts in areas such as fire protection, operation, systems engineering and electrical circuits. Feedback from these experts is needed.
- The transition to a risk-informed and performance-based licensing basis means a more efficient use of resources. (cf. NFPA-805 [1]).

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Fire - the Human Dimension

Improving fire safety in nuclear power plants by improving awareness of fire hazards and influencing behaviours

Geraint Williams & Pauline Woodward

ONR (Office for Nuclear Regulation), UK

What are we sharing with you?

- **Key messages** from the work we do with licensees – other than assessment of designs
- Some of the **reasons or evidence** why we have reached these insights…

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Key messages: (1) some "givens"

- Given:
  - Fire is undesirable on nuclear plants…
  - Fire events often have a strong human dimension
  - High performing licensees/ duty holders recognise this and try to manage fire risks proactively

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Key messages: (2) the meat in the sandwich

In order to improve fire safety management it helps if

- There are clear **lines of responsibility** for fire safety
- There are ways of **measuring improvements** in fire safety management
- Fire safety is recognised as a duty of **all staff** (and contractors) and supported by **awareness training** in aspects of fire
- Degraded plant condition is more evident if efforts are made to **improve local environment** (lighting, cleanliness, etc.)

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Key messages: (3) with what purpose

Improved performance in fire safety management should lead to

- Improved **fire safety culture** on the ground
- Improved **accountability** and performance
- More **success stories** to share both within the company and with industry

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Key messages: (4) a bit more context…

- Not all licensees/duty holders are the same
  - right that they have different points of focus / different ways of implementing improvement
  - ONR's regulatory philosophy based on:
    - persuasion and influencing / open and effective dialogue / positive and enabling approach.
    - supports and strengthens licensees' self-regulatory processes rather than provide a substitute for them
    - concentrates on prevention of major nuclear accidents whilst at the same time regulating risks to workers

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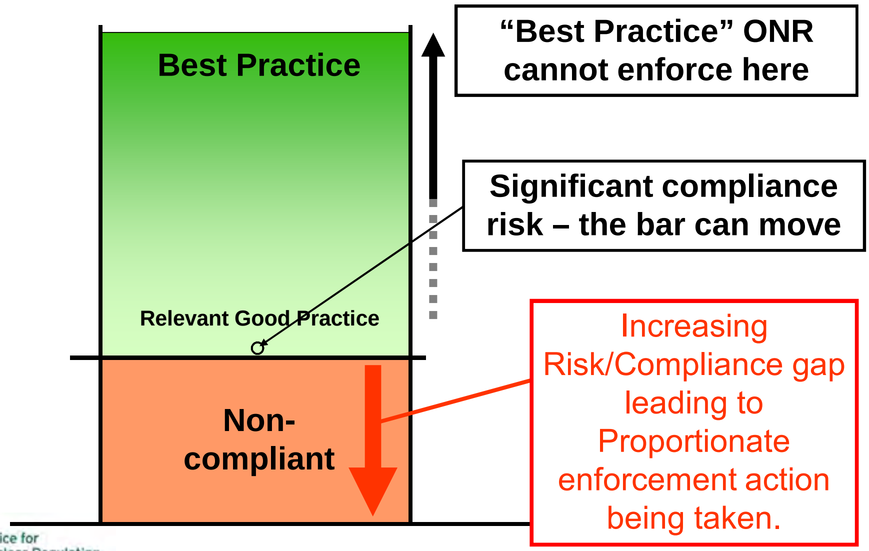
The Regulatory “Goal Posts”



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Supporting evidence for our insights..

- Following slides expand upon the following to give some background:
  - Fire events
  - Fire safety management improvements and self assessment approaches

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ONR's interest in fire

- Not only
  - Assessment of design,
- But also:
  - Compliance with the fire safety case at UK nuclear facilities,
  - "Conventional" fire with ONR's Fire Safety Inspectors
    - Personnel safety, Means of escape, Risk Assessments, etc.
    - Regulated under non-nuclear legislation – Regulatory Reform (Fire Safety) Order 2005 and Fire (Scotland) Act 2005
  - OPEX (Operational Experience)/LFE (learning from experience) includes fire events

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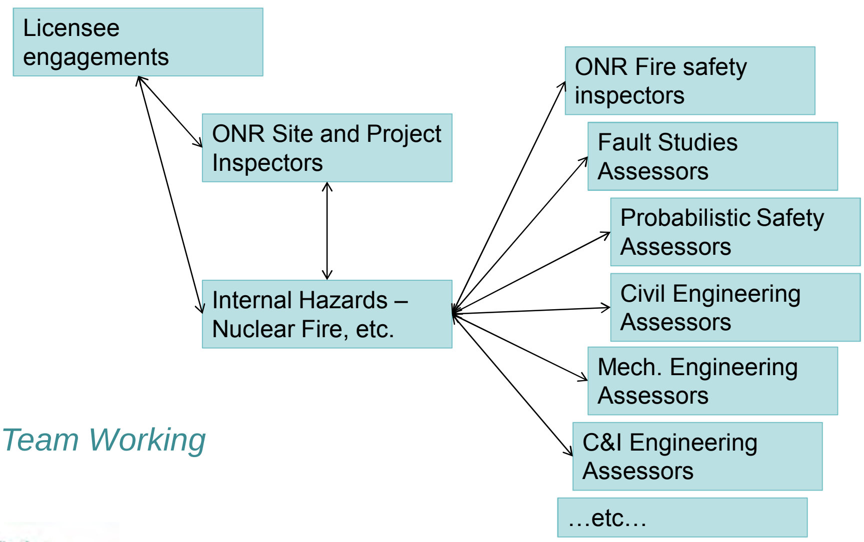
Interconnectivity within ONR



Team Working

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ONR insights from events

- Historically:
  - a few big fire events causing release
    - International: Brown’s Ferry / Vandellos
    - UK: Windscale Fire, fires in waste silos
  - A larger number of middle sized events where release was avoided/less significant
    - Turbine plant fires, RCP fires, HP oil spray fire, oil-filed transformers, etc.
  - Regular smaller fires on nuclear plant – some self limited and burn out, some extinguished at plant
- Generally
  - not many real or significant events per year – but danger is complacency

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Events (cont)

- Do they have the same cause?
  - No, but quite a lot have common themes (UK and worldwide)
    - Electrical fires, particularly on older and less well maintained systems, or infrequently-used systems
    - Control of works, hot working, etc.
      (subset: fire in decommissioning work)
    - Transient combustibles, housekeeping
    - Oil systems: leaks, failures, controls – oil under insulation
    - Fires at standby diesels or gas turbines
    - Some traffic-related fires

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Success stories…

- Some recent examples
  - Licensee improvement initiative for construction projects
    - Included: responsibilities, increased monitoring and audits, improved procedures and company standards, improved awareness of fire hazards, shared learning briefs, oversight, etc.
  - Licensee developed and delivered at all their sites an improved fire safety awareness course, delivered to all relevant staff, including contractor support staff

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Success stories (cont.)…

- (more examples)…
  - Licensee focus on improving fire safety management at all levels of the company
    - Many aspects including: development of a fleet fire safety index to measure aspects of fire safety, fire "referential" inspections – a high level audit, comparison with French practice, improvements in plant condition, etc.
    - Has made definite improvements in "look and feel" of plant and fire events have reduced
      (causal or coincidence?: beware complacency!)
- UK good practice is shared at a cross-licensee forum – allows learning from others
  - Not all licensees have same problems, but all can learn from other

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Measuring fire safety management performance?

- Why measure?
  - Good way of communicating with higher management and stakeholders, and to get buy-in for improvement plans
- How to measure? No clear approach:
  - INPO Performance Objectives, Criteria and Guidance
  - WANO Peer reviews
  - IAEA NS-G-2.1 (also Safety Series No.50-P-9, marked as obsolete)
  - Various company-developed approaches
    (e.g. EDF NGL developed a "fire referential" based upon all the references above)
  - Some potential for developing an industry wide "Capability Management Model"

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DEFENCE-IN-DEPTH STRATEGY OF FIRE PROTECTION AND ITS RELEVANCE AFTER FINAL SHUTDOWN (BY THE EXAMPLE OF GERMANY)

Michael Beesen, Benjamin Ernst, Günter Fischer

TÜV SÜD Industrie Service GmbH, Germany

ABSTRACT

Nuclear power plants (NPP) are protected against internal and external fires by a fire protection defence-in-depth concept including the following precautionary measures: operational, structural and equipment related fire protection measures as well as manual fire fighting. The fire protection measures are designed in consideration of fires to be expected (from fire loads permanently and temporarily present together with potential ignition sources) in order to prevent a violation of both the protection goals of public law and the nuclear protection goals / radiological safety objectives in case of internal and external fires.

The aspect "What is the future significance of the fire protection defence-in-depth concept?" needs to be considered with regard to the situation following the final shutdown. From our point of view as a TSO (*t*echnical *s*afety *o*rganization) both the non-nuclear protection goals (e.g. prevent occurrence of a fire; ensure escape and rescue of humans) as well as the nuclear ones have to be ensured after final shutdown of a nuclear plant. The protection goals of public law will almost completely remain after the plant has stopped commercial operation while the nuclear safety objectives will be stepwise reduced in consideration of the decommissioning status until the end of the nuclear supervision. Nevertheless, the fire protection concept must clearly specify those fire protection measures that are necessary to ensure the plants' safety. The situation on site regularly needs to be under examination to check if the fire protection concept covers all conditions to be considered and if the existing fire protection measures are sufficient or if an adaption is necessary.

INTRODUCTION

German Nuclear Phase-out

In 2010, the German Parliament agreed a lifetime extension for German NPP. As a result of the Fukushima Dai-ichi reactor accidents the German Parliament decided in 2011 that several energy suppliers had to stop commercial operation of their NPP. The German Atomic Energy Act [1] was changed by August 2011. Following this, the energy suppliers had finally to stop production of energy at eight NPP units. The change of the German Atomic Energy Act [1] also includes the decision to stop the production of nuclear energy in all NPP latest by the end of the year 2022 (German nuclear phase-out). Following 2011, the remaining operating NPP will have to stop one after the other commercial operation. Most recently, the Grafenrheinfeld NPP stopped operation in June 2015. Figure 1 shows the German NPP used for commercial production of energy, their status, and the latest date of final (safe) shutdown.

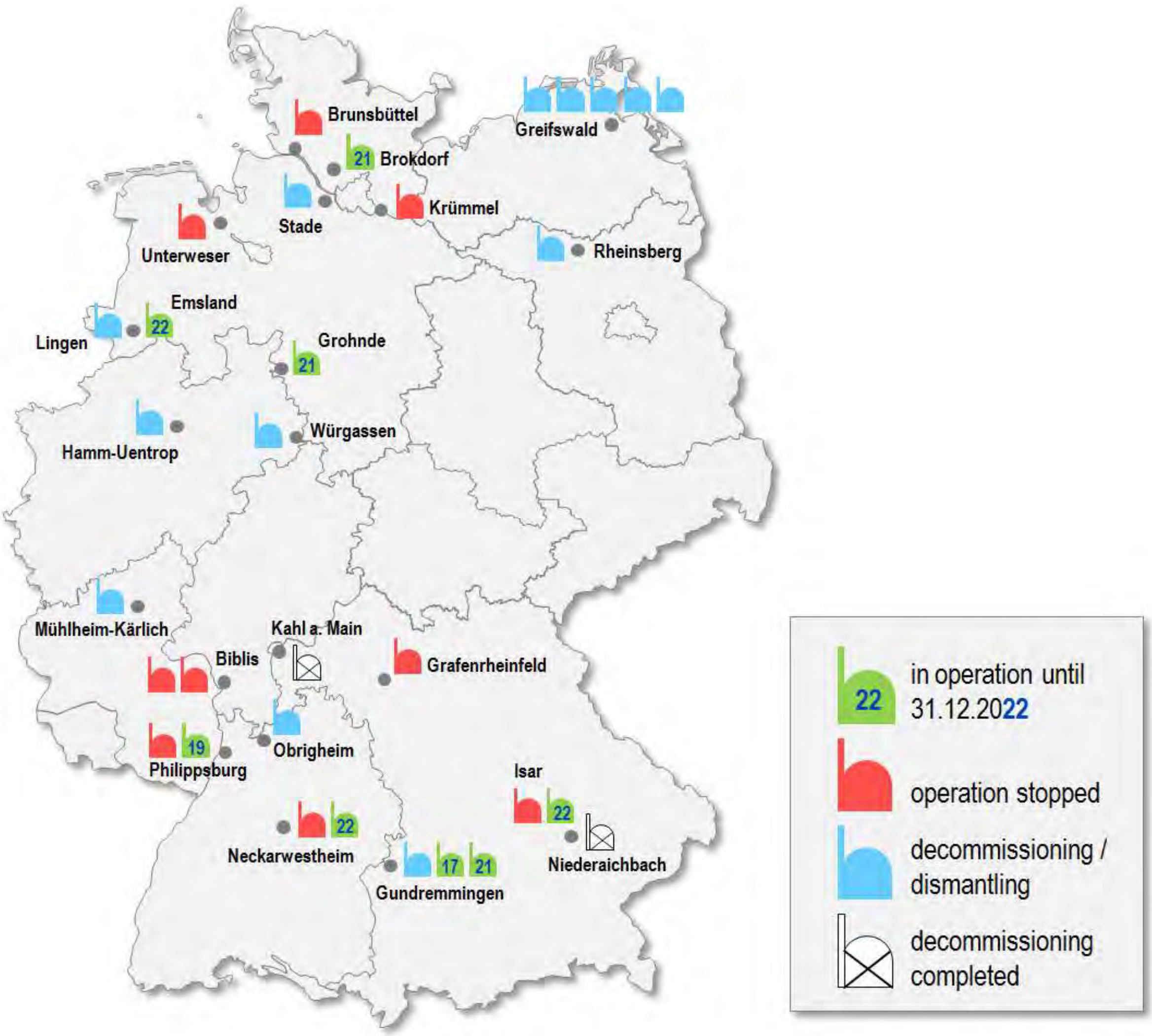

Figure 1 German NPP - status and latest date of shutdown

Concerning those NPP with shutdowns since 2011, the energy suppliers prefer immediate dismantling for decommissioning of their NPP. In addition to the support of the remaining operating German NPP and nuclear projects outside Germany, German nuclear industry will have to handle the disused NPP in the next years and decades to achieve nuclear dismantling goals (nuclear fuel no longer present on site, radioactive materials no longer present, end of nuclear supervision). The required time for phase-out operation ("post-commercial safe shutdown"), decommissioning and dismantling is estimated to be 10 to 15 years per plant unit following the proclaimed authorities' permission (license) for decommissioning.

Legal Requirements and Protection Goals

Concerning constructing, using, modifying and dismantling of industrial buildings in Germany different legal requirements have to be taken into account, particularly the non-nuclear building law, the occupational health and safety law and the nuclear safety law. Relating to public safety and order attention should be paid to the building law with its buildings codes, technical building regulations and technical rules, etc. Occupational health and safety is achieved by complying particularly with the German labour protection act, technical rules for workplaces as well as with standards and guidelines of the employers´ mutual insurance association. Nuclear safety is ensured if claims according to the state-of-the-art of science and technology are met. Therefore, attention should be paid especially to the Atomic Energy Act [1], the Radiation Protection Ordinance], the Safety Requirements for NPP [3], the RSK[1] Guidelines, the ESK[2] and SSK[3] Recommendations as well as the KTA safety standards (e. g. KTA[4] standards series 2101 "Fire Protection in Nuclear Power Plants" [4]). The relevant legal requirements are summarized briefly in Figure 2.

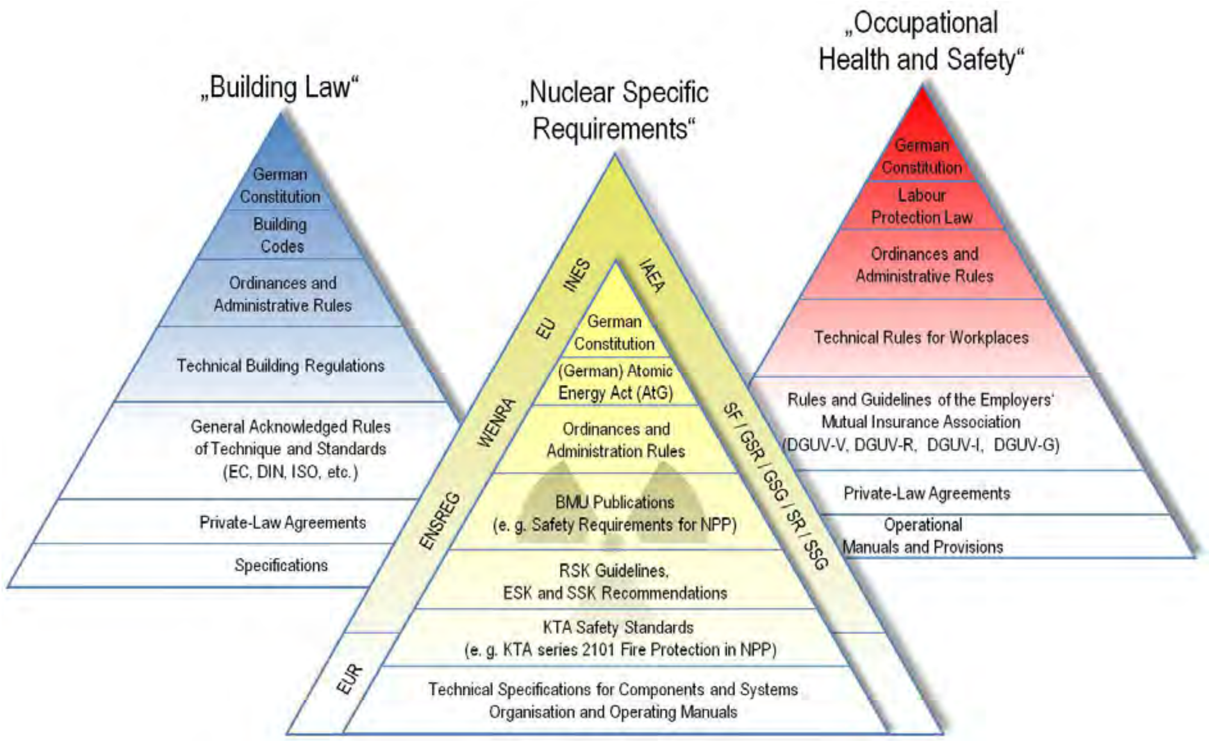

Figure 2 Legal German requirements

The major aim of the German building codes is to ensure that public safety and order, in particular life, health and the basis of life, are not endangered. Concerning fire safety this aim is put in concrete terms by the fire protection goals "*prevention of fire occurrence*", "*prevention of fire and smoke spread*", "*enabling rescue of people*" and "*enabling effective fire fighting*" (cf. Figure 3).

Figure 3 General and fire related protection goals of building law

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1 RSK: Reactor Safety Commission (in German: *R*eaktor-*S*icherheits*k*ommission)

2 ESK: Nuclear Waste Management Commission (German for: *E*ntsorgungs*k*ommission)

3 SSK: Commission on Radiological Protection (German for: *S*trahlen*s*chutz*k*ommission)

4 KTA: Nuclear Safety Standards Commission (German for: *K*ern*t*echnischer *A*usschuss)

According to the German Atomic Energy Act [1] it is necessary that precautions have been taken in light of the state-of-the-art with respect to science and technology for preventing damages resulting from the construction, operation and dismantling of NPP to ensure nuclear safety. This aim is put into concrete terms with the “Safety Requirements for Nuclear Power Plants” [3] by the nuclear protection goals “*control of reactivity*”, “*cooling of fuel assemblies*” and “*confinement of radioactive materials*” as well as by the radiological safety objectives “*limitation of radiation exposure*”. To ensure the nuclear protection goals and the radiological safety objectives in case of on-site building-internal and building-external fires, it is required to have a defence-in-depth fire protection concept including fire protection measures that suitably protect relevant structures, systems and components as well as plant personnel (cf. Figure 4).

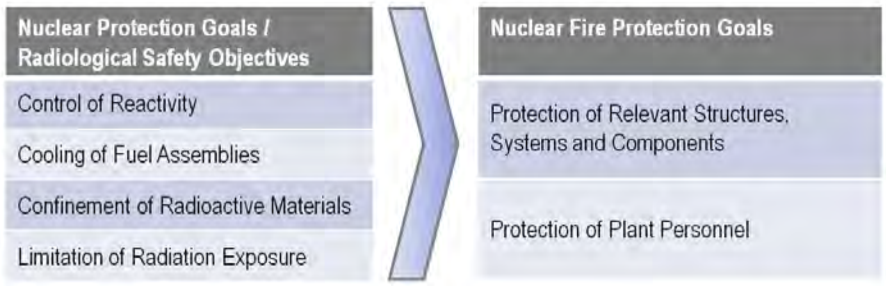

Figure 4 General and fire related nuclear protection goals

Figure 5 shows during which operating phases, e.g. normal operation, shutdown states such as outages for refuelling, inspection and maintenance, safe shutdown for starting removing components, and dismantling NPP, the building law’s protection goals and the nuclear protection goals respectively the radiological safety objectives are relevant and have to be considered.

Building Law’s Protection Goals	Power Operation	Post-Operational Phase	Phase-Out Operation	Post-Utilisation
Ensure that public safety and order - in particular life, health and the natural basis of life – are not endangered	✓	✓	✓	✓

Nuclear Protection Goals / Radiological Safety Objectives	Power Operation	Post-Operational Phase	Phase-Out Operation	Post-Utilisation
Control of Reactivity	✓	at least temporarily	posibly temporarily	- if no post operation utility after AtG applies
Cooling of Fuel Assemblies	✓	at least temporarily	possibly temporarily	
Confinement of Radioactive Materials	✓	✓	✓	
Limitation of Radiation Exposure	✓	✓	✓	

Figure 5 Building law’s protection goals and nuclear protection goals / radiological safety objectives and their significance subject to plat operational states

FIRE PROTECTION DEFENCE-IN-DEPTH STRATEGY AND ITS SIGNIFICANCE

Fire Protection Defence-in-depth Concept

The "Safety Requirements for Nuclear Power Plants" [3] require precautionary measures for fire protection in order to limit plant internal fires (on-site building-internal or building-external) and their consequences to ensure acceptable conditions with respect to nuclear safety. Therefore a fire protection defence-in-depth concept consisting of active and passive fire protection means has to be developed and implemented to ensure "*prevention of occurrence of incipient fires*", "*fast and reliable fire detection*", "*fast and reliable fire suppression*" and "*spreading of fire to be limited*". Thus the "Safety Requirements for Nuclear Power Plants" [3] requires a fire protection defence-in-depth concept meeting both the building law's fire protection goals as well as the nuclear fire protection goals respectively the radiological safety objectives.

According to the nuclear KTA fire safety standards series 2101 [4] the fire protection defence-in-depth concept consists of:

- operational fire protection measures,
- structure related fire protection measures,
- equipment related fire protection measures, and
- manual fire fighting capabilities.

To decrease the fire risk and to increase the efficiency of the entire fire protection defence-in-depth concept the following measures are implemented:

- quality assurance (QA),
- periodic preventive inspections,
- preventive maintenance.

Figure 6 shows the measures of the entire fire protection defence-in-depth concept, their interaction and their effects to the nuclear protection goals and the radiological safety objectives respectively the equipment and materials to be protected.

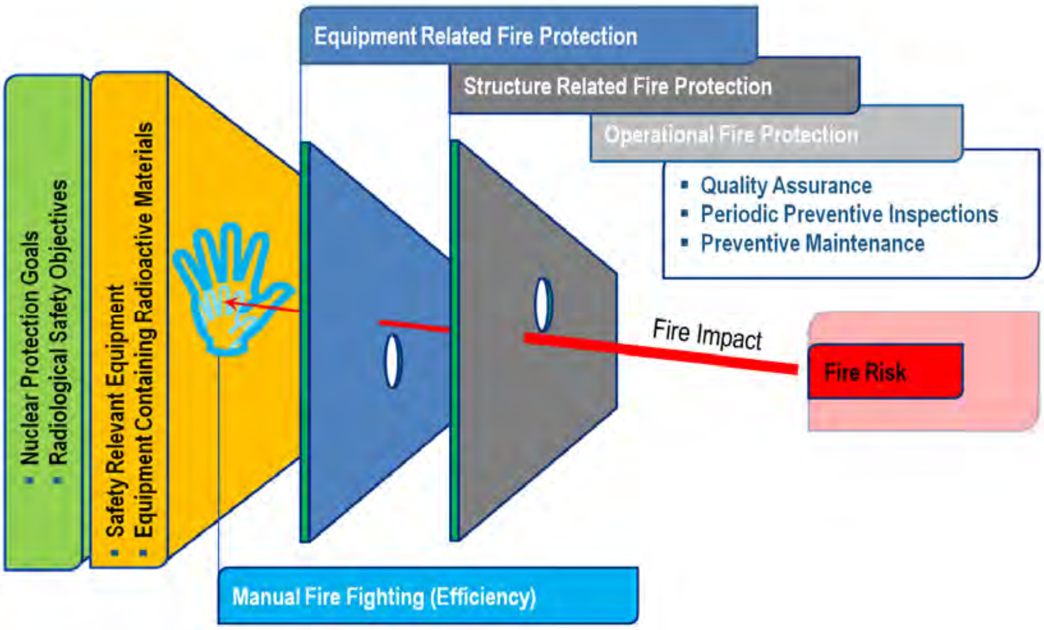

Figure 6 Fire protection defence-in-depth concept

Nuclear Defence-in-depth Concept and the Importance of Fire Protection

According to the "Safety Requirements for Nuclear Power Plants" [3] the safety of German NPP is based on an overall defence-in-depth concept with four safety levels. On level of defence no. 1 the goal is to prevent abnormal operation by use of high level and monitored quality of systems as well as approved and frequently trained personnel. Control of abnormal plant operation and prevention of accidents is the goal of level of defence no. 2. Therefore, additional measures are implemented. The third level of defence (representing the last level of design towards incidents) is the control of accidents by use of safety systems that are designed to mitigate the effects of accidents and that are sufficiently effective in case of assumed defects/failures. On level of defence no. 4 the effects of beyond design basis accidents (e.g. crash of civil aircraft) are limited.

The fire protection defence-in-depth concept aims to prevent spread of a fire occurred at level of defence no. 1 such that it will not exceed level of defence no. 3 (cf. Figure 7).

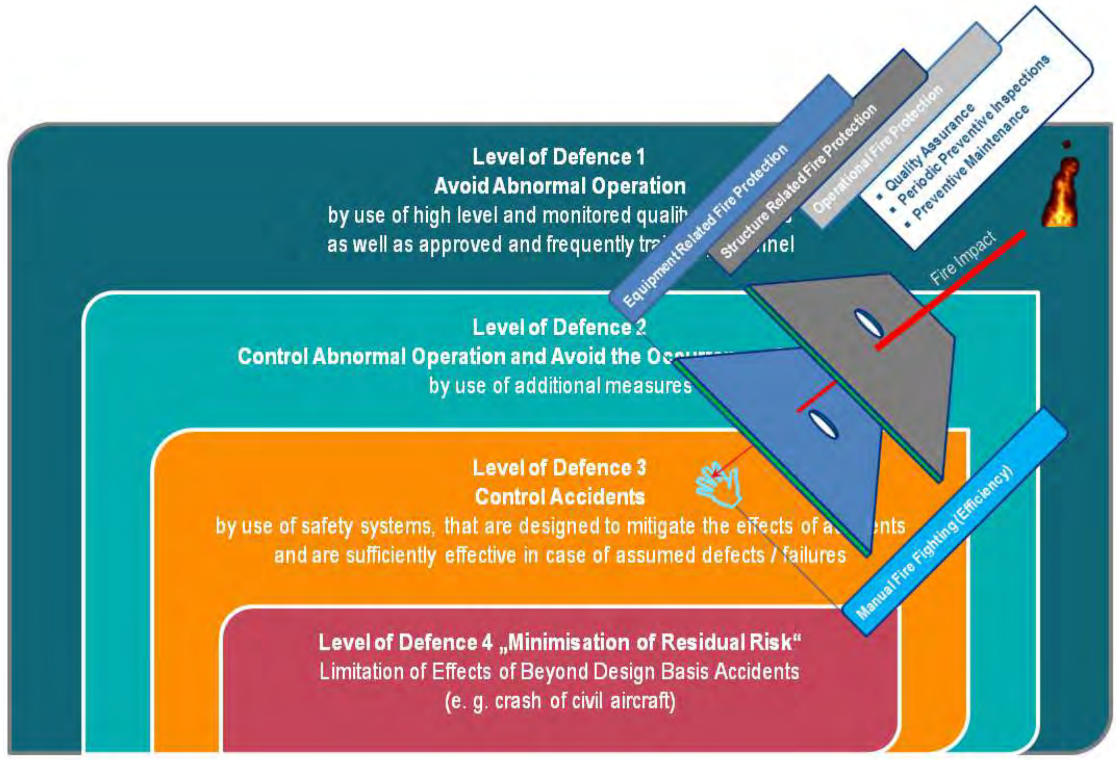

Figure 7 Nuclear defence-in-depth concept and importance of fire protection

It is important trying to prevent a shift from a low level of defence to a higher one, because the "Safety Requirements for Nuclear Power Plants" [3] demanding the following:

- On level of defence no. 1 and 2:
  Taking into account all circumstances of any individual case, the radiation exposure of the personnel shall be kept as low as achievable for all activities, even below the limits of the Radiation Protection Ordinance [2].
- On level of defence no. 3:
  The maximum design limits for the plant for protecting the population against any release-induced radiation exposure shall not exceed the relevant accident planning levels of the Radiation Protection Ordinance [2].

Furthermore, the Radiation Protection Ordinance [2] requires that anyone who owns, operates or decommissions a NPP shall minimize any radiation exposure or contamination of man and environment – even below the respective limit – by taking into consideration the state of the art and by taking into account all circumstances of individual cases.

The confinement of radioactive materials present in the nuclear power plant as well as the shielding of the radiation emanating from them shall be ensured according to the “Safety Requirements for Nuclear Power Plants” [3]. The confinement of radioactive materials shall be ensured by implementing sequential barriers and retention functions. Barriers are understood to be the fuel rod cladding, the reactor coolant pressure boundary and the containment. Retention functions are measures or equipment for the retention of radioactive materials (e.g. by filtering, water coverage, directed flow through maintaining sub-atmospheric pressure, delay lines, building seals, drain pans, vessels or other confinements).

According to the “Safety Requirements for Nuclear Power Plants” [3] the entity of barriers and retention functions shall be designed in such a way and maintained in such a condition over the entire plant service life that, in combination with the measures and equipment of the respective levels of defence, the respective safety related acceptance targets and acceptance criteria (cf. Annex 2 of [3]) as well as the radiological safety objectives according to Section 2.5 of [3] are met on the different levels of defence for all events or plant conditions and the associated mechanical, thermal, chemical and radiation-induced impacts.

The barriers and retention functions in their entirety shall also be reliably effective enough in all events resulting from internal and external hazards or very rare human induced external hazards that the radiological safety objectives according to Section 2.5 of [3] are met.

Plant Modifications and their Effects on the Fire Protection Defence-In-Depth Concept

At first, fire protection measures (precautionary measures) in order to meet the nuclear fire protection goals are required as long as the protected safety system or parts thereof are necessary to fulfil the superior nuclear protection goals / radiological safety objectives. For example, fire protection measures that protect the residual heat removal system or parts thereof are necessary to be kept functioning until all fuel assemblies have been removed.

At second, fire protection measures (precautionary measures) are needed to prevent radiation exposure in case of fire. Therefore, these fire protection measures are necessary to remain as long as the protected radioactive material remains at the NPP site and could be released as a consequence of fire. The minimization of the radiation exposure is required also in case if the exposure does not exceed the accident planning values of the Radiation Exposure Ordinance [2].

Fire protection measures are usually inspected and tested regularly in recurring intervals. Modifications of the testing manuals (that are part of the safety specifications) are allowed, if the fire protection measures are not required any more from the perspective of nuclear safety relevance. For example, periodic in-service inspections of fire protection features inside the turbine building of PWR[5] are no longer necessary after final shutdown. One prerequisite for this is that no safety system or parts thereof are operated inside the turbine building and that there is no storage of radioactive materials.

The on-site fire brigade is frequently considered as a means of compensation when assessing deviations from requirements. A reduction of the amount of fire fighters is only possible, if both, an assessment by the authority responsible for fire prevention as well as an assessment by the nuclear authority regarding the significance for nuclear safety have been carried out.

Concerning the fire load present at a NPP, it is sometimes argued that by disconnecting electrical installations during the post-commercial safe shutdown plant operational phase, the hazardous potential with respect to fire safety is significantly reduced. In our point of view, the hazardous potential is reduced sustainable, when the fire load is removed.

5 PWR: Pressurized Water Reactor

After final shutdown, a modification of prevailing conditions concerning the utilization may occur. For example, performing a decontamination of the reactor coolant system may result in additional fire loads and cable or pipework routing along access and escape routes when the number of personnel is significantly increased. With regard to fire safety this is acceptable if this situation is covered by measures of the fire protection defence-in-depth concept being already in place or if deviations from regulatory requirements are compensated by additional fire protection measures (e.g. installation of additional fire detectors; immediate alert of the on-site fire brigade in case of a fire detector being actuated instead of alert after verification by shift personnel; installation of self-rescuers / smoke hoods, etc.). Further examples for an assessment being required with regard to adequacy, suitability and efficiency of the fire protection defence-in-depth concept including the respective measures being in place are the installation of additional work stations for waste treatment or the installation of additional laboratory work stations.

CONCLUSIONS

As part of a fire protection defence-in-depth concept, fire protection measures as precautionary measures against fire hazards are required to prevent that functions of items important to safety are inadmissibly impaired. The nuclear protection goals and radiological safety objectives always have to be taken into account and therefore the fire protection goals have to be ensured.

The limitation of the effects of a plant internal fire to the valid accident planning values according to the Radiation Protection Ordinance [2] is not sufficient. According to [2] the radiation exposure always has to be minimized to a level below the permissible limits of the Radiation Protection Ordinance [2].

Plant modifications require a regular verification of the existing fire protection defence-in-depth concept with its fire protection measures regarding their suitability and efficiency.

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FEEDBACK FROM RECENT OPERATING EXPERIENCE IN NUCLEAR POWER PLANTS REGARDING FIRE SAFETY

Burkhard Forell

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Cologne, Germany

ABSTRACT

The paper provides insights from the operating experience in German nuclear power plants with reportable fire safety related events. The three reportable events outlined in more detail resulted in German Information Notices prepared by GRS and distributed on behalf of the Federal German regulatory body. The events concerned deficiencies at pipe penetration seals, which were not filled with mineral wool. Also at fire doors deficiencies of the insulation inside the door wings occurred. One event concerned a smouldering fire of rubber material in a waste drum which was placed in a plant internal drying facility for radioactive waste.

INTRODUCTION

Two reportable events have occurred recently in German Nuclear Power Plants (NPP) affecting fire protection means. These events had some generic aspects in common. In the first event, in penetration seals of pipes leading through fire barriers the filling with mineral wool was missing. Pipes with larger diameters additionally need a thermal insulation of the pipe itself, which was also missing for some pipe penetration seals. In the second event, deficiencies at fire doors constructed according to former standards were found, which also concerned the thermal insulation.

A third reportable event being important to nuclear fire safety is a smouldering fire of rubber material within the plant internal drying facility for radioactive waste drums.

These recently reported events related to fire safety have been analysed in more detail.

FINDINGS AT PENETRATION SEALS OF NON-COMBUSTIBLE PIPES

The pipe penetration seals in walls or ceilings with requirements to fire protection have to be constructed according to the non-nuclear building code with the associated guidelines for pipework and cable installations [1]. These require that starting with a certain diameter not only the ring gap between pipe and wall/ceiling has to be filled with non-combustible mineral material, but also the pipe itself has to be insulated in a way that the temperature on the cold side of the pipe and the seal does not exceed 180 K in case of a standard test fire according to the ISO 834 [2] standard fire curve for a defined time.

For different situations the detailed pipe penetration seal design is typically described in a so-called "pipe penetration seal catalogue" of the given NPP. According to the catalogue, a typical design of a pipe penetration seal of a non-combustible pipe consists of a filling of the ring gap between the pipe and the jacket tube which is fixed to the wall/ceiling. The filling is made by mineral wool with a melting point exceeding T > 1000 °C. Both ends of the jacket tube are typically sealed with a silicone sealing or with a reversible silicone expansion joint (cf. Figure 1). Empty seals consisting of the jacket tube only are filled with mineral wool and covered with a sheet metal cap on both sides (see Figure 1, bottom-middle and top-right).

Figure 1 Pipe penetration seals of non-combustible pipes sealed by reversible silicone expansion joints and one reserve pipe seal closed with a sheet metal cap (bottom-middle and top-right); image source: G+H Isolierung GmbH, www.guh-brandschutz.de

For non-combustible pipes with diameters larger than DN 160 to 200 (depending on the specific requirement), an insulation of mineral wool with a melting point T > 1000 °C additional to the one of the ring gap of the pipe itself is needed (cf. Figure 2).

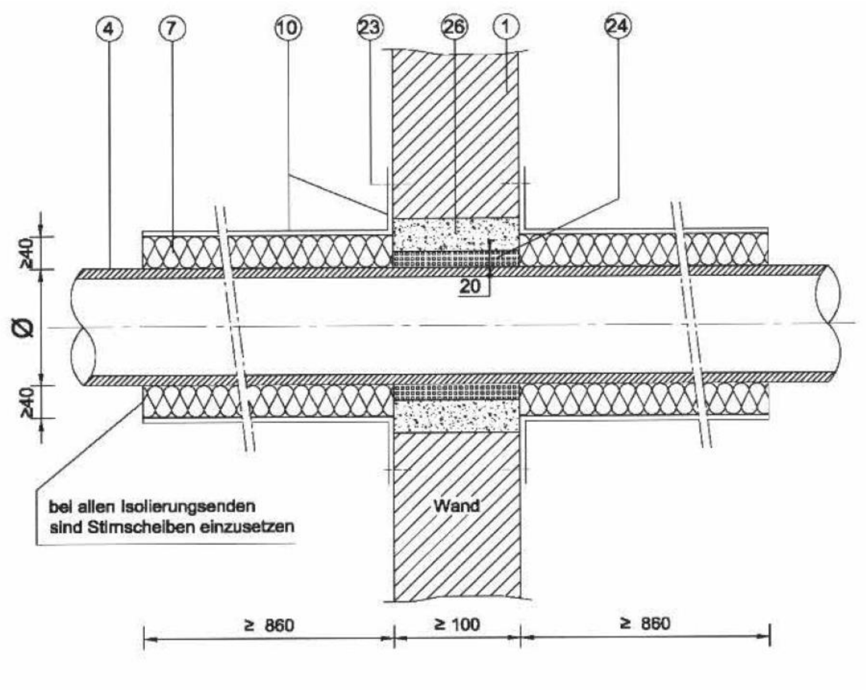

Figure 2 Schematic pipe penetration seal for non-combustible pipes with diameters over DN 160 to 200 (depending on the specific requirement) with insulation of the ring gap and additional insulation of the pipe; image source: G+H Isolierung GmbH

One German NPP has recently reported a finding that at a number of pipe penetration seals the seals were closed with the silicone expansion joints or the sheet metal cap, but the inner filling of the ring gap with mineral wool was missing. Therefore, the pipe penetration seal could only be considered as being smoke-tight but not tight against enduring heat by hot gases or flames. Moreover, the insulations of the pipe were missing at in total 29 pipe penetration seals with diameters larger than 200 mm.

The deficiencies obviously resulted already from the construction phase of the plant in the 1980ies, when the pipe penetration seals had been closed from both sides without filling in the mineral wool. Regular in-service inspections on pipes have been performed; however these inspections only consisted of a visual inspection without opening the seals.

The relevance for nuclear safety can be considered as minor. This is first because the pipe seals still ensure a certain structural fire protection at least for natural fires according to the German operation experience, which are less severe than the ISO 834 standard test fire. Furthermore, the structural fire protection means are only one layer of protection in the plant fire protection concept.

By means of a German Information Notice [3] prepared by GRS on behalf of the Federal German nuclear regulatory body BMUB the German nuclear power plant operators were informed about the event including the following suggestions for preventing recurrence:

- In addition to the regular in-service inspections on pipe seals, random tests shall be performed where a number of pipe seals are opened and the filling of the ring gap by mineral wool is checked. In case of any findings the size of the sample has to be extended.
- All pipe penetration seals shall be checked for the diameter that will require the seals to be equipped with an additional pipe insulation to meet the 180 K temperature criterion.
- Pipe penetration seals should be clearly indicated by a number and a sign showing that the seal meets the fire safety requirements. The documentation to be provided and kept up to date should include number, location, and type of the pipe penetration seal as well as any opening and closing having been performed.

FINDINGS AT FIRE DOORS MANUFACTURED ACCORDING TO FORMER CONSTRUCTION STANDARDS

Findings at fire doors manufactured according to former construction standards were reported from two German NPPs. After some old fire doors had been replaced by new ones and cut open after use, a considerable number of these fire doors showed problems with the insulation inside the door wing. In some cases, the mineral wool insulation had shrunk causing a gap of approximately 30 mm at the upper part of the door. The fire doors had been in service for more than 35 years. For comparison, Figure 3 gives an example of a door wing where the mineral wool filling is complete. Other doors that were checked in the two plants also showed missing parts of thermal insulation close to the lock casing. Concerning the shrinking it is assumed that the deficiencies result from aging effects. The missing parts of the insulation were attributed to a lack of quality insurance during the manufacturing of the fire doors in the beginning of the 1970ies. The findings described were observed at fire doors which were produced by at least three different manufactures.

Similar to the reported event on pipe penetration seals, the relevance with respect to nuclear safety of these events on fire doors can be considered as minor. This is particularly due to the fact that according to the operation experience in Germany the fire doors still provide a considerable level of structural fire protection for natural fires.

In the beginning of the 1970ies, most fire doors in Germany were manufactured according to the German standards DIN 18081, DIN 18082, and DIN 18084 consisting of construction plans of major parts of the doors, including the door wing insulation by mineral wool that was fixed by wire. These standards from 1969 were withdrawn at the end of 1976, because stronger test criteria were enacted before. Only the standard DIN 18082 [4] on 30 min rated fire doors was republished in a revised version, which also included a door wing insulation of mineral fibre boards instead of the old insulation by mineral wool.

In addition, in those times, industry was encouraged to develop own types of fire doors that met the test criteria in a fire test according to the ISO 834 [2] curve. These doors could enter the market because they had a test certificate instead of being constructed according to a construction standard.

Figure 3 Example of a fire door according to the old DIN standard being cut open after removal from the plant showing no deterioration of the insulation

Although major changes have taken place looking at the design of fire doors during the last decades, it is important to note that there is no principle change in the requirements on quality insurance for manufacturing of fire doors. However, it is commonly assumed that quality insurance has been improved by the manufactures.

By means of another German Information Notice [5] prepared by GRS the nuclear operators were informed about the event and the following suggestions against recurrence were made:

- All fire doors manufactured according to the German standards DIN 18081, DIN 18082, and DIN 18084 having been withdrawn by the end of 1976 shall be checked randomly. In case of findings the size of the sample has to be extended and, if necessary, a full check has to be performed.
- All fire doors that are constructed or have a test certificate according to standards that were published from 1977 on might also be affected by quality deficiencies or by aging processes. For these doors it has to be demonstrated by the quality documentation or by the construction plans of the doors that deficiencies are excluded with a high probability. Otherwise, the fire doors should be checked randomly such as outlined above.

SMOULDERING FIRE OF WASTE MATERIAL WITHIN THE DRYING FACILITY

A smouldering fire within a waste drum that was put into a plant internal drying facility was reported by one German NPP. However, latterly GRS was made aware of similar, but non-reportable events that did not fulfil the reporting criteria.

The 200 l waste drum was filled with mainly with hard rubber. The material was originally used as a coating of a vessel and therefore was crushed, which decreased the self-ignition temperature (SIT) of the material. The material consisted of natural rubber, hard-rubber powder and sulphur.

The drum was put into the facility for drying. The drying process is performed by air that circulates in the facility. The air is heated up by electric heaters to a temperature of about 155 °C. Then it is led to the drying chamber to heat-up of the waste drum and to take up the water vapour. Finally, the air is cooled down to collect the condensate before it is heated up again. The drying process is finished as soon as the mass flow of condensate is below a certain limit.

During the drying process the regular closure of the waste drum is opened and the drum is covered with a steam screen (a metal plate with holes) that allows evaporation of water.

After the drying process ended as normal the heating switched off and the steam screen was automatically lifted up to increase cooling of the drum. The air in the chamber of the drum cooled down to almost room temperature. After that the air temperature measured in the chamber increased slowly but continuously. Thus a temperature surveillance alarm was automatically triggered.

The surveillance alarm of the system was not indicated as fire alarm and was first ignored in the main control room. It was recognized after shift changeover in the morning. After examination of the situation, plant personnel triggered the manual CO_2-inerting of the drying chamber. As the CO_2-system was not able to extinguish the smouldering fire in the waste drum, the drying facility was opened by the plant fire brigade and the waste material was finally extinguished with water and foam as extinguishing agents.

The smouldering fire did not cause any radioactive releases that could be measured outside the drying facility. There was no contamination or harm of personnel. The fire severity did not exhibit any impact on nuclear safety. Probably, fires on radioactive waste drums may lead to contamination. A smouldering fire may cause an open fire or may produce smouldering gases leading to ignitable atmospheres within a drying facility. Explosions or backdraft-like phenomena may not be excluded.

The investigations after the smouldering fire showed that the SIT of the rubber material was significantly lowered by crushing it. The SIT was assumed to be in the order of 300 °C, but was value was valid for the material when it was applied according to the design as coating of vessels. With crushing the material, the SIT decreased and was in later laboratory experiments reproduced at 160 °C.

The SIT is a safety related parameter depending on the boundary conditions. The SIT e.g. decreases if the size of the sample increases, because the ratio of sample volume divided by the sample surface increases. The SIT also increases with decreasing oxygen concentration within the sample, however even at oxygen concentrations of about 2 % self-ignition in granules, etc. does occur [6], [7].

In an German Information Notice [8] prepared by GRS on behalf of the Federal German regulatory body, BMUB, GRS informed the German NPP operators about the event, including the following proposals for preventing recurrence:

- Bevor using the drying facilities with new materials where the ignition properties are unknown, self-ignition processes shall be ruled out. This can be done by test dryings, where scaling effects concerning the influence of the sample size on the self-ignition temperature should be considered.

- It shall be checked if the time between the end of the drying process and lifting up of the steam screen can be increased to limit the oxygen access into the waste drum. (This measure is assumed by GRS to reduce the risk of open flames in case of fire. The risk of ignition may also be reduced by the steam screen, however it is not known if and in how far the oxygen concentration within the waste drum is reduced by the steam screen.)
- Fire detection within the drying facility shall be improved (CO measurements are the first choice in fire detection; however difficulties were reported for CO measurements because of the high vapour content in the system. Temperature measurements within the waste drums were also recommended.) The indication of the signals on the main control room should be improved.
- During operation and cooling phases of the drying facility it shall be surveyed regularly by plant walk down.
- Since the installed CO_2 inert gas fire suppression system is not suitable to extinguish a smouldering fire within a drum, it should be considered if more reliable extinguishing equipment could be installed at the drying facility.

CONCLUSIONS

German nuclear operators reported several events with respect to fire safety during the last years. Three of these events showed generic aspects and therefore resulted into German Information Notices which where prepared by GRS. Concerning the findings at pipe penetration seals, the first feedback from German operators shows that there were no further systematic deficiencies at any other German NPP. Since the two other information notices are quite new, the feedback of the operators is still ongoing.

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4 Seminar Conclusions and Outlook

The 14th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' demonstrated ongoing progress in nuclear fire safety with respect to performed experiments and assessment. However, there are still challenges since the knowledge on several fire related phenomena is still not yet mature and the analytical tools applied need further enhancement.

The presentations at this seminar provided an added value to the state-of-the-art in nuclear fire protection highlighting recent developments, but also describing in an open manner still unsolved issues in this field.

The following conclusions have been drawn from the seminar:

Although progress has been achieved with respect to fire safety in nuclear installations and its assessment, fire safety is still an important issue which has to be addressed in all types of nuclear facilities from their construction throughout their operational and post-operational lifetime until decommissioning has been completed. Currently, as the seminar has indicated, the main focus is on the planning, construction and operational phase.

With respect to research activities focusing on real case fire scenarios in nuclear installations it has to be stated that the actually ongoing experimental nuclear fire research programs provide valuable insights on the behaviour of the nuclear facility in case of fire. A typical example is the international OECD PRISME Project, which is currently in its second phase with as far as possible realistic nuclear power plant specific scenarios for closing still existing knowledge gaps. The intended experiments should assist to resolve 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 the fire sequence.

A confined and mechanically ventilated compartment fire is one scenario in a nuclear facility, where fire compartments are connected to ventilation networks to prevent radioactive releases. Mechanically ventilated compartment fires strongly differ from naturally ventilated ones insofar that fires are confined in enclosures with forced ventilation, leading to significant thermodynamic pressure variations. Whereas naturally ventilated compartment fires have been extensively studied in the past, mechanically ventilated compartment fires are less documented due to the lack of large scale fire tests. In the frame of the fire PRISME test program experiments were designed for investigating the fire growth in full scale confined and mechanically ventilated compartments. The test results have been used as benchmark data for the validation of different fire models.

Thermal effects of high energy arcing faults (HEAF) have been observed in nuclear installations representing non-negligible contributors to the fire related risk as they have demonstrated the potential to cause extensive damage to electrical components and distribution systems along with damage to adjacent equipment and cables. Full scale experiments are being performed at high and medium voltages in electrical components which will provide a more in-depth understanding and could be the basis for deriving appropriate countermeasures.

In this context, new insights from the operating experience, inspection and maintenance activities should be consequently used for improving reliability and effectiveness of active and passive fire protection features.

Regarding the validation of fire models, the heat release rate for electrical cabinet fires is still an important and not completely solved issue. One reason for this situation is that the existing databases do not provide the information needed for that purpose. Therefore, further investigations to reduce as far as possible uncertainties in the fire characteristics are needed.

Moreover, complex mixed (natural/forced) convective flows are still a challenge for CFD fire simulation codes for proper calculations of these experimental scenarios.

The challenges regarding development of the fire risk analysis methodology as well as modelling the different functions of passive fire protection means within this approach involving confinement and protection of equipment important to safety have not yet been completely resolved.

Modelling of unprotected cables has been internationally studied in recent years. The corresponding results may provide the basis for the planning of further validation experiments to the fire performance of protected cables. For future applications, the question of applicability of recently developed sub-models on the fire behaviour of protected cables has to be answered.

Associated circuits need to be investigated to determine if cable faults can prevent or cause maloperation of redundant systems important to safety. If a circuit is not properly protected by an isolation device, fire damage to a cable could propagate to other safe shutdown cables. In order to check that the coordination is adequate, existing electrical protection coordination studies have been analysed and, for some plants, additional analyses have been performed for AC as well as DC for instrumentation and control (I&C) systems.

Spurious actuations are also a fundamental part of the analysis of the consequence of a fire, which should consider any possible actuation that can prevent or affect the performance of a system or a safety function. In this context, it is necessary to take into account the possibility of a combination of several spurious actuations that may result in a specific consequence.

However, fires are such complex phenomena with high uncertainties in their behaviour over time that the modelling has not yet reached the same high level of confidence as nuclear simulations in other areas, e.g. modelling thermal hydraulics. In this context, sensitivity studies were also addressed making the seminar auditory aware that the simulation results strongly depend on a variety of sensitive parameters.

Uncertainty in model predictions of the behaviour of fires is an important issue in fire safety analysis for nuclear power plants. A global sensitivity analysis can help identifying the input parameters or sub-models that have the most significant effect on model predictions. An alternative approach is to perform a global sensitivity analysis using an emulator.

Further uncertainties do still exist in the human response to fire. Efforts are ongoing to reduce these uncertainties for enabling the analyst to quantify them, in particular in the frame of probabilistic analyses.

Plant modifications require a regular verification of the existing fire protection defence-in-depth (DiD) concept with its fire protection measures regarding their suitability and efficiency. The nuclear fire protection goals have also to be ensured after final safe shutdown of a nuclear facility taking into the actual plant status. The situation on site needs regular examination for checking if the fire protection concept covers all conditions to be considered and if the existing fire protection measures are adequate or if an adaption is necessary.

The method proposed for the evaluation of the DiD approach to fire protection is a combination of an ignition root cause analysis, an event tree for the fire scenario and a consequential failure modes and effects analysis (FMEA) where these three analyses are performed successfully for a given type of fire event. The challenge is to support experts focusing on certain technical domains and exchanging relevant information between these analyses. Ignition root cause analysis is performed to find the factors leading to a fire event. Fire propagation is then modelled in the event tree, where fire protection means are taken into account. FMEA is then performed based on the fire scenario extracted from the event tree. The last stage of the analysis also includes analysing fire spreading. Measuring the effect of compartmentation and fire spreading on safety is still challenging.

An appropriate assessment of plant internal fires is not only important in the frame of designing future nuclear power plants but also on the regulator's viewpoint how to apply an enhanced DiD concept to existing nuclear power plants. In thiscontext, new insights from the operating experience, inspection and maintenance activities should be consequently used for improving reliability and effectiveness of active and passive fire protection features.

An additional focus of the seminar was on the role of passive fire protection means (e.g. fire barriers) and their significance for the plant design. Part of this activity was a review of the benefits of an effective passive fire protection strategy, alongside other arrangements (such as active fire protection) to a nuclear operator. Passive fire protection is not subject to random failure. Thus, if the passive protection means have been properly installed and no damage by an external source has occurred, passive fire protection feature can keep their function during the entire lifetime of the power plant. Therefore, this aspect should be part of the investigations to enhance fire safety for new built nuclear facilities.

For demonstrating the robustness of the fire protection in nuclear facilities, post-Fukushima issues such as consideration of potential 'cliff-edge' effects or combinations of fires and other anticipated plant internal and external events, being either correlated in cause or time have to be investigated. Some steps in this direction are the results of the European Union stress tests including a time schedule for implementing necessary measures. In this context, mobile equipment for ensuring adequate protection becomes more and more important as well as the availability of accident mitigation equipment needed to be reliably operable after an event which also have to prevent or mitigate consequential fires.

Last, but not least, the nuclear accidents as a result of earthquake and Tsunami.at the Fukushima Dai-ichi nuclear power station in Japan in March 2011 revealed further questions, in particular on combinations of fires with external or internal hazards. The most recent earthquakes having impaired the safety of nuclear installations have re-started the discussion between regulators and analysts in several countries on fires consequential to external hazards. The existing regulations and standards have to be adapted to the state-of-the art in this respect. This has already been done to some extent with respect to the requirements for design basis accidents, however further enhancements should be enforced in all countries with nuclear installtions.

For beyond design basis accidents with consequential fires, a need for further investigations remains. For that purpose, international activities regarding research and analytical effort to further improve the safety of operating installations in critical situations.

The participants from all over the world, representing the different parties involved in nuclear fire safety, from nuclear industry as well as from regulatory bodies, research institutions as well as from technical expert and support organizations (TSO), emphasized the value of and benefits from the information provided in this experts' seminar to be shared inside the nuclear fire community. They clearly expressed their wish of continuing this series of fire safety seminars on a regular basis in time intervals of approximately two years. The next, 15th seminar of this series is therefore planned to be conducted in late summer 2017 in conjunction with the 24th 'International Conference on Structural Mechanics In Reactor Technology' (SMiRT 24), which will take place in Busan, Korea in August 2017 (cf. http://www.smirt24.org/).

Attachment

CD of the 14th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' held as Post-conference Seminar of SMiRT 23

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH

Schwertnergasse 1
50667 Köln
Telefon +49 221 2068-0
Telefax +49 221 2068-888

Forschungszentrum
85748 Garching b. München
Telefon +49 89 32004-0
Telefax +49 89 32004-300

Kurfürstendamm 200
10719 Berlin
Telefon +49 30 88589-0
Telefax +49 30 88589-111

Theodor-Heuss-Straße 4
38122 Braunschweig
Telefon +49 531 8012-0
Telefax +49 531 8012-200

www.grs.de