SMiRT 25
16th International Seminar on FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS

Ottawa, Ontario, Canada
October 27-30, 2019

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und Reaktorsicherheit
(GRS) gGmbH

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Nuclear fire safety, experimental research, Fire PSA, fire simulations, future designs, human factor, operating experience, regulation, safety assessment, standards

Kurzfassung

Im Rahmen des vom Bundesministerium für Umwelt, Naturschutz und Nukleare Sicherheit (BMU) beauftragten Vorhabens 4717R01550 fand im Oktober 2019 das mittlerweile sechzehnte internationale Seminar “Fire Safety in Nuclear Power Plants and Installations“ als Post-Conference Seminar der 25th International Conference on Structural Mechanics in Reactor Technology (SMiRT 25) zusammen mit der nuklearen Konferenz FSEP 2019 (Fire Safety and Emergency Preparedness) in Ottawa, ONT, Canada statt.

Die vorliegenden Proceedings des Seminars enthalten alle Fachbeiträge des zweitägigen Seminars mit insgesamt 42 Teilnehmern aus insgesamt 13 Ländern aus Europa, Asien und Nordamerika. Außerdem beinhalten sie drei seitens des SMiRT Post-conference Fire Seminars bei der FSEP 2019 in der letzten Plenarsitzung präsentierten übergeordnete Beiträge und weitere Fachbeiträge, die seitens Seminarteilnehmern in Parallelsitzungen bei der FSEP 2019 vorgestellt wurden.

Abstract

In the frame of the project 4717R01550 funded by the German Ministry for the Environment, Nature Conservation and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz und Nukleare Sicherheit, BMU) the meanwhile sixteenth international Seminar on "Fire Safety in Nuclear Power Plants and Installations" has been conducted as Post-Conference Seminar of the 25th International Conference on Structural Mechanics in Reactor Technology (SMiRT 25) in conjunction with the FSEP 2019 (Fire Safety and Emergency Preparedness) nuclear conference in Ottawa, ONT, Canada.

The following Seminar Proceedings contain the entire technical contributions to the two days Seminar with in total 42 participants from 13 countries in Europe, Asia and Northern America. Moreover, the proceedings include three high-level contributions from the SMiRT Post-conference Fire Seminar presented in the joint final plenary session and further contributions moved to parallel sessions of FSEP 2019.

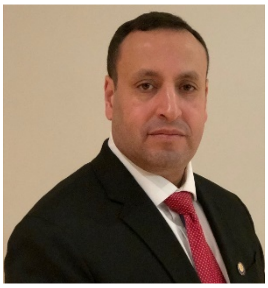
Contents

Foreword

In October 2019, the 3rd International CNS Conference on Fire Safety & Emergency Preparedness for the Nuclear Industry (FSEP 2019) and the 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations of the International Association on Structural Mechanics in Reactor Technology (IASMiRT) were held jointly in Ottawa, Ontario, Canada.

More than 150 participants from 13 countries including specialists, researchers, regulators and industry representatives, and other stakeholders in fire protection and emergency management in the nuclear industry attended both events. The experts participated in a very busy three-day program. Technical papers and presentations themes focused on methodologies for fire protection assessments, validation of fire models, experimental research and results, regulation and standards, fire safety in the design and operation of new and future nuclear power plants, deterministic fire safety analysis and probabilistic fire risk assessment. These technical meetings enabled participants to engage with recognized national and international experts from a number of areas in the nuclear sector including licensees, manufacturers, researchers, consultants, academia and regulators. Plenary sessions were devoted mainly to shared lessons learned from fires events, emergency preparedness practices, innovation, communication of the risk to public and experimental research results, and knowledge transfer.

The common events were a success as they demonstrated that fire safety and emergency management are integral to nuclear safety and security as well as to public confidence in nuclear energy. The strong international participation fostered opportunities of global partnership and collaboration to enhance the effectiveness of fire safety and emergency management.



Dr. Abderrazaq Bounagui, CSNC
– Local Organizer –

1 Introduction

The meanwhile 16th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' was held as Post-conference Seminar of the 25th International Conference on Structural Mechanics In Reactor Technology (SMiRT 25) jointly with the Canadian nuclear conference FSEP 2019 on Fire Safety and Emergency Preparedness in Ottawa, Ontario, Canada in October 2019.

In total 42 participants from Belgium, Canada, Finland, France, Germany, Japan, the Republic of Korea, the Netherlands, Sweden, Switzerland, the United Arab Emirates, the United Kingdom and the United States of America followed the 24 presentations that were presented in the different scientific sessions and participated actively in a final short expert panel discussion on future challenges with regard to fire safety of existing as well as new built nuclear facilities at the end of the seminar.

It has to be clearly pointed out that from the first Seminar 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 power plants and other nuclear facilities has significantly increased. This does in general concern the design of the plants and, in particular, of structures, systems and components (SSCs) important to safety. But this also considers the operation of such installations as well as all areas of assessment, inspection and maintenance. For more than thirty years, methodological approaches for assessing the fire risk and the corresponding analytical tools have been evolving and are continuously being enhanced.

The two-day Experts Seminar started with a session on general issues in fire protection of nuclear facilities including recent developments in regulations, standards and fire protection programs. The second Seminar session addressed the issues from the operating experience collected at nuclear installations from fires and other events related to fire safety including lessons learned from these. Another session was specifically focussing on the contribution of high energy arcing fault (HEAF) events in nuclear power stations to the overall risk taking into account that those events have the potential to induce fires and to inadmissibly impair fire protection features.

Two further sessions aimed on presenting recent results from research and modelling activities with respect to fires in nuclear facilities and on model development and applications for deterministic as well as probabilistic fire safety assessment.

The seminar topics highlighted the quite broad scope of the issues related to fire safety in nuclear installations. The presentations and discussions again indicated that fires not only in existing nuclear facilities designed according to former standards but also in modern ones are still a “hot” topic and need to be addressed not only as single events, but also in the context of event combinations with other internal and external hazards.

One major goal of this sixteenth 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 identified for existing nuclear installations in commercial operation or under final shutdown as well as new types of reactor facilities to be built and operate safely in the future.

The Canadian Nuclear Society (CNS) was the organizing body for FSEP 2019. The permanent SMiRT Fire Seminar organizers were strongly supported by the local organizing committee including Dr. Abderrazzaq Bounagui from the Canadian Nuclear Safety Commission (CNSC), Ann Turney from Canadian Nuclear Laboratories (CNL) Ltd. and Rudy Cronk from PLC Fire Safety Solutions.

The organizers want to thank all speakers, co-authors and chairpersons as well as the entire participants for their highly active and fruitful participation and valuable, high-level contributions during this 16^{h} International Seminar on ‘Fire Safety in Nuclear Power Plants and Installations’ which made this venue again a very successful one.

The next, 17th seminar of this series is intended to be held as SMiRT 26 Post-conference Seminar in Germany in late summer 2021.

Dr. Marina Röwekamp and Dr. Heinz-Peter Berg
– Scientific Chairs and Permanent Organizers –

2 Seminar Agenda

Sunday, October 27, 2019

17:00 h	***Registration and FSEP 2019 / SMiRT Fire Seminar Informal Welcome (Sneak Peak)***		

Monday, October 28, 2019

07:30 h	***Speakers and Delegates Breakfast***		
08:00 h	**Registration**		
08:30 h	***FESP 2019 Plenary Session (see FSEP 2019 Program)***		
10:15 h	***Coffee Break***		
10:30 h	***FESP 2019 Plenary Session (see FSEP 2019 Program)***		
12:00 h	***Lunch Break***		
13:00 h	***SMiRT Post-Conference Fire Seminar Introduction and Welcome by the Organizers***	R. Cronk, A. Bounagui. M. Röwekamp	PLC, Canada; CNSC, Canada, GRS, Germany
13:15 h	**Regulation, Standards and Guidelines, Protection Programs**	***Chairpersons:*** ***M. Lehto, S. Thompson***	
13:15 h	Fire and Explosion Hazards in Advanced Nuclear Reactor Technologies: Regulatory Insights from the United Kingdom	D. Lisbona, G. Williams	ONR, United Kingdom
13:40 h	Canadian Fire Protection Regulatory Requirements for Small Modular Reactors	A. Bounagui, et al.	CNSC, Canada
14:05 h	Proposed Performance-Based Code Alternative Solution Models in the Nuclear Industry	H. Shalabi, D. Esposito	Carleton University, Canada Jensen Hughes, Canada
14:30 h	**Operating Experience and Lessons Learned**	***Chairpersons:*** ***M. Roewekampe, A. Bounagui***	
14:35 h	Insights on Successful Combustible Controls Programs	P. Boulden Jr.	Appendix R Solutions, USA
15:00 h	***Coffee Break***		
15:20 h	**Operating Experience and Lessons Learned (contd.)**	***Chairpersons:*** ***M. Roewekampe, A. Bounagui***	
15:20 h	Most Important Fire Events from Finnish NPPs and Consequential Improvements and Plant Modifications	M. Lehto	STUK, Finland
15:45 h	Fires of Radioactive Materials During and After Drying Processes in German Nuclear Installations	B. Forell	GRS, Germany

16:10 h	Operating Experience with Fires in Nuclear Installations – The OECD/NEA FIRE Database	M. Beilmann, et al.	OECD/NEA, France, FIRE member countries
16:35 h	***Break***		
16:45 h	Behavior of Fire Detection and Control Systems in Case of Smoke Spreading Through Fire Barriers – Consequences and Opportunities	A. Niggemeyer	Framatome,Germany
17:10 h	Operating Experience with Fire Dampers in German Nuclear Power Plants	B. Forell	GRS, Germany
17:35 h	***Adjourn of the first Seminar day***		
17:00 h	***FSEP 2019 and SMiRT Fire Seminar Wine and Cheese Reception and Student Poster Presentation***		

Tuesday, October 29, 2019

07:30 h	***Speakers and Delegates Breakfast***		
08:30 h	***Briefing***		
08:45 h	***FESP 2019 Plenary Session (see FSEP 2019 Program)***		
10:15 h	***Coffee Break***		
10:30 h	**Operating Experience and Lessons Learned (contd.)**	***Chairpersons: L. Kuriene, M. Roewekamp***	
10:30 h	Experience Regarding Qualification of Fire Detection and Control Systems	D. Heinert, et al.	Framatome, Germany
10:55 h	Nuclear Power Stations Fire Brigade Organization, Responsibilities and Challenges in Different OECD/NEA FIRE Member Countries	K. McGrath, J.-P. Cayla, M. Roewekamp, et al.	ONR, United Kingdom, IRSN, France, GRS, Germany
11:25 h	**Special Topic: High Energy Arcing Faults (HEAF)**	***Chairpersons: K. Shirai, N. Melly***	
11:30 h	U.S. NRC Pre-Generic Issue 0-18 High Energy Arcing Faults (HEAF) Involving Aluminum	N. Melly, et al.	NRC, USA
12:00 h	***Lunch Break***		
13:00 h	**Special Topic: High Energy Arcing Faults (HEAF) contd.**	***Chairpersons: K. Shirai, N. Melly***	
13:00 h	OECD/NEA High Energy Arcing Faults (HEAF) Research – Second Phase of Testing	N. Melly, et al.	NRC, USA
13:25 h	Medium Voltage Breakers from Nuclear Power Plants to be Tested Within the OECD Nuclear Energy Agency Experimental Project HEAF 2	L. Kuriene, M. Roewekamp	ANVS, Netherlands GRS, Germany
13:50 h	**Experimental Fire Research and Modelling**	***Chairpersons: S. Suard, N. Melly***	

13:55 h	Common Cable Fire Benchmark Activity of the OECD Nuclear Energy Agency Projects PRISME 3 and FIRE	S. Bascou, et al.	IRSN, France
14:20 h	SEVEN Expert System: A Decision Support Tool for Fire Safety Analysis in the Nuclear Area	W. Plumecocq, et al.	IRSN, France
15:00 h	***Coffee Break***		
15:20 h	**Experimental Fire Research and Modelling (contd.)**	***Chairpersons: S. Suard, N. Melly***	
15:15 h	Implementation Strategies of a Semi-Empirical Cable Fire Model in the FDS Fire Simulation Code	Y. H. Jung, D. I. Kang	KAERI, Korea
15:40 h	**Fire Safety Analysis and Modelling**	***Chairpersons: W. Plumecocq, M. Roewekamp***	
15:40 h	Foundations for A Successful Fire PRA Project	P. Boulden Jr.	Appendix R Solutions, USA
16:05 h	KAERI's Research and Development Activities for the Construction and Quantification of Fire Event PSA Model	D. I. Kang, Y. H. Jung	KAERI, Korea
16:30 h	***Break***		
16:40 h	**Fire Safety Analysis and Modelling (contd.)**	***Chairpersons: W. Plumecocq, M. Roewekamp***	
16:40 h	Improving Realism in Fire PRAs for Nuclear Power Plant Applications	N. Melly, D. Stroup	NRC, USA
17:05 h	Case Study for proposed CANDU Fire Probabilistic Risk Assessment Model	H. Shalabi, et al.	Carleton University, Canada
17:30 h	**Concluding Panel Discussion**	***Chairperson: A. Bounagui***	
	Panel participants:	M. Lehto, N. Melly, M. Roewekamp, K.Shirai, S. Suard, S. Thompson	
17:45 h	***Adjourn of the second Seminar day***		
18:00 h	***Common FSEP/SMiRT FireSeminar Cocktail Reception and Banquet***		

Wednesday, October 30, 2019

07:30 h	***Speakers Breakfast***		
07:45 h	***Delegates Breakfast***		
08:45 h	***Common FESP 2019 and SMiRT Fire Seminar Plenary Session (see FSEP 2019 Program)***		
08:45 h	Fire Safety at Nuclear Sites: Challenges for the Future – An International Perspective	M. Roewekamp	GRS, Germany
09:30 h	Overview of the OECD PRISME 3 Project	S. Suard	IRSN, France
10:15 h	***Coffee Break***		

10:30 h	TBD	J. A. Milke	University of Maryland, USA
11:15 h	Proposal of an Evaluation Method for Prevention of High Energy Arcing Fault (HEAF) Induced Fires at Low and High Voltage Electrical Cabinets	K. Shirai	CRIEPI, Japan
12:00 h	***Final Conference Lunch***		
14:00 h	***Adjourn of FSEP 2019 and SMiRT Post-Conference Fire Seminar***		

3 Seminar Contributions

In the following, the Seminar contributions prepared for the 16th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' held as Post-conference Seminar of the 25th International Conference on Structural Mechanics In Reactor Technology (SMiRT 25) jointly with the FSEP 2019 conference are provided in the order of their presentation in the Seminar.

Moreover, three high-level international contributions were decided by the organizing committees to be presented and discussed to all participants of FSEP 2019 and the SMiRT Post-conference Fire Seminar during the final common Plenary Session. These contributions are provided in a specific section on the common FSEP 2029 and SMiRT Fire Seminar session, having taken place after the Seminar during the last day of the joint venue.

In addition, those presentations, which were moved to FSEP 2019, either because of fitting better in the context of the corresponding FESP 2019 sessions or due to a lack of time in the Seminar, are also provided in a specific section at the end of the corresponding chapters.

3.1 Session on “Regulation, Standards and Guidelines, Protection Programs”

The Seminar started with a session on regulatory issues, standards and guidelines applied and the corresponding protection programs ensuring fires safety in nuclear power plants, with a focus of the three presentations not only on existing facilities but also on reactor units to be built.

A first presentation was given by the United Kingdom regulatory body ONR (*O*ffice for *N*uclear *R*egulation) on their experiences with the application of the United Kingdom´s Safety Assessment Principles (SAPs) and supporting Technical Assessment Guides (TAGs) and on the expectations of ONR with respect to the protection against internal hazards and, in particular, fires, in the so-called GDA (*G*eneric *D*esign *A*ssessment) Process for in total four different types of generation IV reactors to be designed and built in the United Kingdom.

The second presentation by the Canadian Nuclear Safety Commission (CNSC) focussed on the Canadian fire protection regulatory requirements, particularly for Small Modular Reactors (SMRs) to be built proposing a graded approach addressing the particular characteristics of this type of facilities.

Moreover, a new performance system model as well as a performance-based code alternative solution model have been developed in Canada, which can be applied for assessing fire safety of existing as well as new nuclear installations.

The corresponding Seminar Contributions are provided hereafter.

Fire and Explosion Hazards in Advanced Nuclear Reactor Technologies: Regulatory Insights from the United Kingdom

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ABSTRACT

The application of ONR's Safety Assessment Principles (SAPs) to the assessment of fire and explosion hazards is discussed in this paper in the context of four Generation IV reactor technology types: sodium fast reactors (SFRs), lead fast reactors (LFRs), high temperature gas reactors (HTGRs) and molten salt reactors (MSRs).

The paper draws from experience from current operational reactor designs which is transferable to new advanced designs. It discusses, for each technology in turn, the opportunities and challenges offered by generic features aimed at preventing and/or limiting the consequences of fires, explosions and other consequential hazards.

Key themes included the consideration of defence-in-depth and hierarchy of fire protection measures, visibility of hazard consequences on an unmitigated basis, and reliance on bespoke modelling tools, methods and performance standards.

ONR has recently developed a new technical guidance for Generic Design Assessment (GDA) and undertaken an in-depth review of the Internal Hazards Technical Assessment Guide (revision 5). The revised guidance represents an extensive update in order to present ONR expectations on internal hazards more explicitly, and specifically in the context of New Build, existing facilities and small modular reactors. However, ONR regulatory expectations in the area of internal fire have not changed – simply re-presented for clarity. Whilst a review of guidance for compatibility with Generation IV technologies is currently ongoing to incorporate ONR's learning on Advanced Modular Reactors, no significant changes to ONR's regulatory expectations on internal fire are so far considered necessary.

INTRODUCTION

The Office for Nuclear Regulation (ONR) is the United Kingdom's (UK) independent regulator of nuclear safety and security. A key requirement of UK law and ONR's regulatory approach is that licensees build, operate and decommission nuclear sites ensuring that risks are As Low as Reasonably Practicable (ALARP).

ONR's nuclear safety inspectors use the ONR Safety Assessment Principles (SAPs) [1], together with supporting Technical Assessment Guides (TAGs, [2]) to guide their regulatory judgements when undertaking technical assessments of nuclear safety submissions. This includes recommendations on whether risks have been reduced to ALARP. The licensee's demonstration that risks have been reduced to ALARP may include numerical risk assessments, but also should include a comparison with Relevant Good Practice (RGP).

From 2017 and in line with the Nuclear Sector Deal [3], the UK Government Department for Business, Energy and Industrial Strategy (BEIS) has invested to ensure that regulators (ONR and the Environment Agency) develop the capability and capacity to regulate Advanced Nuclear Technologies (ANTs). ONR subsequently undertook a review of four Generation IV fission reactor types (sodium fast reactor (SFR), lead fast reactor (LFR), high temperature gas reactor (HTGR) and molten salt reactors (MSR)) to identify key safety considerations, and knowledge and research gaps in areas of regulatory interest. These reviews in turn informed ONR strategies for inspector training and international engagement. The considerations from an internal fire perspective are presented in this paper.

Also, ONR has recently reviewed its guidance for compatibility with modular reactor designs as part of the project and documented key regulatory expectations and 'lessons learnt' from Generic Design Assessment (GDA) – see [4]. A more in-depth review of ONR guidance for compatibility with Advanced Nuclear Technologies (ANTs), and in particular Generation IV designs, is currently ongoing.

SODIUM FAST REACTORS

Chemical reactivity of sodium metal with water and air has long been a key consideration in the design of sodium fast reactors. Sodium reactions with water are quasi-instantaneous, highly exothermic reactions which evolve hydrogen with spontaneous ignition. Apart from the high temperatures, the associated blast wave characteristics may challenge nuclear safety barriers / compartmentation. Sodium also exothermically reacts with air and can give rise to sodium metal fires. Sodium pool fires and jet fires are challenging to extinguish by conventional means. As the heat of reaction is dissipated, the temperatures can reach sufficiently high levels to ignite hydrocarbon-based combustible materials in the area, or thermally damage structures, systems, and components (SSCs). Sodium aerosols can be formed as sodium coolant is released at pressure, or in the event of a fire sodium oxide/hydroxide particulates. These are highly corrosive and can damage SSCs upon deposition. There are also human health (and environmental) hazards from releases of sodium oxide/hydroxide aerosols which make human intervention in the event of a leak difficult [5], [6].

A significant proportion of liquid-sodium-cooled reactors built to-date have experienced long term shutdown periods as a result of sodium fires. The prototype Japanese fast reactor, Monju underwent a major sodium fire on December 1995 and was not restarted until May 2010. The fire resulted from a sodium leak in the secondary circuit whilst the reactor power was being increased. It has been reported that the leak originated from the failure of a thermocouple under high cycle fatigue from flow induced vibration [7]. The plant was modified following the event and this included provision of sodium leakage detectors (camera monitors, smoke sensors); changes to sodium drainage system to shorten residence time; installation of a nitrogen suppression system; and the division of the secondary circuit into four smaller zones to minimise the spread of aerosols.

Sodium and water interactions and sodium fires have been reported in Russia's BN-350 and 600, the UK PFR and France's Phenix and Superphenix. A rather comprehensive compilation is available from [8], which includes how learning from the events was incorporated into the designs. Following the sodium fire in the BN-350 (1975), the evolution of this design into the BN-600 provided for the steam generators in separate locations so that accidental sodium-water reactions could not extend to the containment vessel. A total of 27 sodium leaks were reported for the BN-600 until 1999 (all detected in time) and no further leaks have been reported for this design. Extensive leak detection improvements were also reported for Superphenix (following a sodium spray fire in 1993). The UK's PFR also had operational experience in sodium leaks from secondary circuit cells and a major tube failure in Superheater 2.

In order to increase resilience against internal fire and explosion, sodium fast reactors include (with a degree of design variability) a number of the following design features:

- The effects from sodium-air reactions and sodium-water reactions as a result of leaks can be reduced by inerting with a suitable protective atmosphere (e.g. nitrogen, argon).
- Provision of further passive containment of sodium leaks (e.g. catch pots such as the "Karlsruhe tray") to reduce the amount oxygen available to oxidise leaking sodium [9].
- A double boundary usually in the form of inerted guard/ protection vessels with capacity to capture the entire primary circuit inventory (including the Intermediate Heat Exchanger).
- A double boundary between primary and secondary circuits and between the secondary circuit and the steam side fitted with leak detection. Double tubes on the steam generator side are used in the loop-type reactor design as sodium leaves the reactor vessel to the intermediate heat exchanger. The key advantage of loop-type reactors is maintainability, and this led to the loop design being favoured in the Japan sodium fast reactors. Some designs include seamless straight tubes in block heat exchanger designs (once-through flow) to minimise potential leak points. These are considered to reduce the potential for leaks from cracking in welded areas. Leaks can arise by other means as a result of restricted tube expansion in thermal cycling or vibrations causes by sodium flow.
- Leak detection systems should be able to detect leaks from water/steam into sodium (and vice versa) and sodium leaks to the inert atmosphere / air.

The delivery of safety functions under fire conditions in this reactor type is generally achieved by provision of segregated, redundant and diverse SSCs. In modern reactor designs, segregation is generally achieved by provision of reinforced concrete barriers and suitably designed penetrations that withstand the relevant internal hazard loads. Molten sodium reacts with concrete and is a known barrier degradation mechanism in the event of leaks. Steel linings on surfaces exposed to metal sodium are provided in a number of SFR designs [10].The SFR designs will need to consider concrete-sodium interactions and make adequate provisions including liners to withstand sodium leaks and fires in order to meet ONR expectations as laid out in ONR SAPs EKP.1, EKP.3 and EHA.17 (appropriate materials in case of fires).

The detrimental effects of sodium fires on SSCs, particularly in areas where safety is not delivered by the provision of safety trains segregated by lined concrete barriers, can arise not only as a result of high temperature, but also from deposition and attack by corrosive aerosols / combustion products. These can cause consequential hazards away from the seat of the fire and result in blockages of equipment such as pumps, filters, and generally instrumentation damage.

In nuclear power reactor design, the performance requirements on fire compartment barriers and other SSCs has historically been defined by the use of fire modelling tools e.g. the US National Institute of Standards and Technology (NIST)'s Consolidated Fire and Smoke Transport Model (CFAST), the U.S. Nuclear Regulatory Commission (NRC) Fire Dynamic Tools (FDT), and Computational Fluid Dynamic (CFD) codes. These aim to predict a series of fire parameters, e.g. upper gas layer temperatures, rebar temperature, temperature on the non-exposed side of barriers. The predicted parameters can be compared with performance criteria e.g. standard fire curves such as those in ISO 834-10:2014 [11] to determine whether barriers withstand the thermal load.

However, the characterisation of sodium fires is generally supported by bespoke models and tools. For example, numerical predictions for sodium leak hydrodynamics of open pool combustion and correlations have recently been carried out [9]. However, there are no systematic comparisons of sodium fire model performance using a recognised set of scenarios – it is likely that regulatory assessment will require sampling of model basis, assumptions, correlations used, experimental validation and verification status.

Sodium metal aerosols can deposit in cold spots during normal operation. Early consideration and minimisation of such locations by design is good practice and should be pursued in line with the expectations in ONR SAPs, e.g. ELO.4. Fire hazards in such locations can be prevented by the inert atmosphere; however, due consideration should be given to monitoring through-life, and in preparation to maintenance or decommissioning operations. Post operation, several areas of the plan may return to have an air atmosphere, with the associated increased risk of fire initiation. Inspection and maintenance on sodium fast reactors components may require cleanout of this residual sodium and this may involve performing a controlled reaction between residual sodium and water. During these operations, the risk for hydrogen explosions and fires should be reduced so far as is reasonably practicable by, for example, providing suitable inert atmospheres, dilution, and operations to be supported by continuous monitoring of hydrogen levels. The designs may also include the provision of recombiners to eliminate the evolved hydrogen and compartmentalisation by barriers of buildings.

Sodium Fire Challenges to Fire Compartments and the Containment Pressure Boundary

The use of sodium as a coolant allows for reactor operating temperatures which can be higher (by ~ 200 ºC) than in light water reactors (LWRs) and operating pressures at near atmospheric values. Whilst the low operating pressure reduces the challenge to the reactor pressure boundary from internal hazards associated with pressure part failure in LWR designs (pipe whip, jet impact, internal missiles), sodium fires are likely to pose one of, if not the highest, overpressure challenge to the containment boundary.

In line with the SAPs, the potential radiological consequences of a fault or accident should be evaluated assuming safety measures are absent or fail to operate (i.e. they are presented on the basis of "unmitigated consequences"). This generally excludes passive safety features such as walls or pipes, unless the fault or accident affects that feature. An understanding of the unmitigated consequences gives a measure of the importance of protective features and feeds into categorisation and classification studies. Additionally, it informs beyond design basis work and accident strategies.

It is clear that internal hazards, and fire impacts on compartment barriers specifically, have the potential to challenge the integrity of (and potentially fail) passive safety features such as the fire compartment barriers, pipework and means of isolation (even if they are passive), these may need to be shown to be resilient to the bounding sodium fire.

Consequently, dutyholders should expect that ONR will pay particular attention to the sodium inventories assumed in the postulated fire scenarios, the form of release and whether isolation is credited. ONR expects that the consequences of fires will be on the basis that the entire inventory may become involved. Any restrictions to the inventory or form of release will be subject to scrutiny. Any means of isolation will need to be demonstrably resilient to the fire and classified according to its role in protecting against the unmitigated consequences.

In line with the defence-in-depth philosophy as expressed in ONR SAPs, resilience to sodium fire hazards safety should be preferably achieved by characteristics as near as possible to the top of the list below:

- Passive safety measures that do not rely on control systems, active safety systems or human intervention e.g. fire suppression decks;
- Automatically initiated active engineered safety measures;

- Active engineered safety measures that need to be manually brought into service in response to a fault or accident;
- Administrative safety measures;
- Mitigation safety measures.

As outlined in the SAPs, the hierarchy above should not be interpreted to mean that the provision of a measure towards the top of the list precludes provision of other items where they can contribute to defence in depth.

Generally, there are built-in systems in most designs which are credited to limit the consequences of sodium fires and feature higher up on the hierarchy above. These include the use of guard piping, catch pans, fire suppression decks and cell liners. To support the adequacy of the measures credited in the safety case, ONR expects that the measures will be presented in the context of the combined hazard challenge, taking account of defence-in-depth provisions where these are reasonably practicable in combination. For sodium fast reactors the combined hazard challenge may include thermal profiles from the fire, hydrodynamic loads from the sodium inventory and environmental challenges such as the corrosivity of sodium fire products. Additional defence in depth may come from leak detection and other means of fire suppression, to extinguish any fires as early as possible.

Fire Suppression

An expectation from ONR Safety Assessment Principles (SAPs) EHA.16 is that fire detection and fire-fighting systems of a capacity and capability commensurate with the worst-case design basis scenarios should be provided. Fire detection and suppression systems should take into account chemical compatibility with sodium metal and fire products and provide for early detection of sodium releases. There is past operational experience on the difficulties of sodium fire suppression and research on the suppression of sodium fires continues at present, with recent findings on the performance of Class D extinguishing agents and nitrogen in gaseous [12] or liquid form [13]. Regardless of the extinguishing agent, a key objective should be preventing the formation of large liquid sodium pools upon release by design [9].

In line with the hierarchy of measures in the SAPs and the defence-in-depth philosophy, passive means including self-suppression decks should be provided so far as is reasonably practicable, as outlined above. Automatically initiated, active engineered safety measures can be expected to be used in conjunction (to limit the extent of any fires should they occur) or when the provision of self-suppression decks is not considered practicable. Noting the challenging environmental conditions associated with sodium fires, it is preferable to ensure that the layout of the plant, and the provision of fire detection and suppression means eliminate the need for manual firefighting of sodium fires. Where automatically initiated fire detection and suppression is not considered practicable, active engineered safety measures that need to be manually brought into service in response should be considered, and manual firefighting only relied upon as the last resort.

LEAD FAST REACTORS

The characterisation of fires involving combustible inventories in LFRs should not depart significantly from fire safety analysis carried out in light water reactors, given the low reactivity of molten lead with water and air. Notwithstanding this, it will be necessary to consider fire and/or explosion initiation following releases of molten lead, as contact with molten lead (at temperatures ~ 500 ºC) can ignite combustible inventories, result in pressurisation, release and rapid vaporisation of high flash point fluids, i.e. lubricants, which will then in turn pose a mist and/or vapour cloud explosion risk.

Standards such as BS EN 1992-1-2 [14] provide criteria for the temperature rise on the non-exposed side of barriers under fire conditions. Based on that standard, the average temperature rise over the whole of this 'back face' is limited to 140 ºC and the maximum temperature rise at any point of that surface does not exceed 180 ºC. This criterion, as specified in BS EN 1992-1-2, is intended to prevent thermal conduction through a wall igniting a combustible inventory on the other side of the wall. In light water reactors, it is almost invariably the case that unacceptably high temperature rises on the other side of barriers are only credible under fire scenario conditions.

As the design of barriers in lead-fast reactors should take into account the potentially high mechanical loadings from molten lead releases, the rate of temperature rises on the 'back face' should be low and determined by conduction through thick barrier/ penetrations. However, it is theoretically possible that releases of molten lead at primary circuit temperature of ~ 500 ºC result in unacceptably high temperatures on the other side of barriers and this should be considered in the safety case as appropriate. The released molten lead inventory, temperature and distribution in plant until solidification should therefore be considered to determine the area of plant in scope of assessment for combined/ consequential fire and explosion hazards. Also it should be considered that fires adjacent to lead inventories can result in high lead temperatures. In addition to toxic lead vapours, the rate of lead oxide generation and release increases with temperature. Lead oxide particulates are highly toxic and can also lead to damage to SSCs by deposition.

Given the low reactivity of lead-water in comparison with sodium-water, LFRs designs usually contain the steam generator within the reactor vessel. In addition to core voiding risks associated with steam entrapment and circulation in the lead coolant melt, sudden vaporisation of steam generator water released into primary lead coolant can initiate sloshing motion in the primary coolant resulting in significant pressures being exerted on the reactor vessel walls. The rapid expansion of steam following a steam generator tube rupture (SGTR) can also result in significant blast waves (commonly referred to as steam explosions) which also need to be evaluated and could act as fire initiators if impacting combustible inventories, electrical cabinets etc.

Blast waves associated with rapid expansion of water/ steam mixtures upon contact with molten lead are generally well characterised. Hazard ranges have been reported from plant damage, which is generally based on incident occurrences in the lead refining industry (cf. [15], [16], [17], and [18]). In the context of LFRs, Dinh [19] reported the phenomenology of blast wave generation and suggested numerical approximations to estimate the pressure wave generated upon contact of water with molten lead. The authors built on the experimental tests in reported by Sibamoto [20], who studied the consequences of water jet injection into molten lead and observed significant blast waves when the liquid-liquid contact temperature was higher than the homogeneous nucleation temperature. Energy conversion ratios and explosion efficiencies from molten lead and water interactions reported in the open literature are generally low (in the order of 4 % to 15 %) [17], [21]. It could therefore be considered that, providing conservative loadings are used to design the reactor vessel and SG structures and any barriers exposed to lead-water reactions, design resilience against this hazard (and any consequential or independent fire hazards) should be reasonably achievable.

HIGH TEMPERATURE GAS REACTORS

Internal fire and explosion hazards in the context of gas-cooled reactors are generally well understood in the UK given experience from the existing advanced gas-cooled reactors (AGRs) fleet. Fire and explosion hazards inside containment arise primarily from air or water ingress (e.g. by contamination of the high temperature coolant gas or loss of pressure boundary (cf. [22] and [23]). Air ingress poses a risk of fire and explosion from oxidation of combustible moderators (e.g. graphite) and lubricants. Severe accident sequences generally take into account the potential for flammable gas generation and suspension of combustible dusts which may be ignited and give rise to significant overpressures. It is expected that designs will consider a postulated double-ended guillotine break of gas circulation primary pipework [22] and include defence-in-depth measures such as oxygen concentration detection.

It has been highlighted in [24] that there may be circumstances in which a 'chimney effect' can be established if two primary circuit breaks in appropriate locations were to occur. This consideration is of particular relevance to the scope of internal hazards assessment, which should check the credibility of a single guillotine break of pipework resulting in consequential breaks (due to pipe whip, jet or missile impacts).

Water ingress into the reactor vessel and primary circuit is most relevant in plant fitted with a steam cycle. A SGTR and water ingress into the core results in hydrogen generation posing an explosion hazard which should be accounted for in the design. This should include robust SG and containment designs, provision of hydrogen detection systems, moisture barriers, etc. The likelihood of water ingress can be reduced by adoption of a direct Brayton cycle and helium gas turbines. This was generally not adopted in earlier HTGR or AGR designs (as SGs were considered the mature technology of choice) and the technology would need to be proven in future designs.

Oil-based lubricants may be subject to in-service degradation in the high temperature and radiation environments inside containment and, as an alternative, the use of 'dry' (magnetic and gas) bearings have been reported in the open literature; as deployed in Dragon [24]. It is expected that justification, supported by experimental testing of lubricants will be produced to justify the choices made for HTGRs.

Generally, barriers may require fire testing as current standard fire curves (i.e. BS 476 or EN 1991-1-2) are applicable to reinforced concrete designs of NSSS buildings, which may not be applicable to modular designs of HTGRs. It is also considered that, given the proposed temperature of operation, damage to nuclear significant barriers and SSCs may arise as a result of thermal degradation or fires initiated by hot gas releases. Consequently, characterisation of hot gas releases is expected, in addition to the fire, explosion and steam release hazards traditionally presented in LWR safety cases. This should include adequate engineering design provision (by suitable venting and/or segregation of diverse and redundant SSCs in heat resistant compartments). The case should be supported by venting and heat transfer calculations that substantiate the nuclear safety significant barriers and suitably qualified or protected SSCs.

Significant inventories of inert gases are needed for HTGRs which could pose an asphyxiation risk if released in an accident or during a fire. Key considerations relate to the layout (e.g. location of gas tanks, pipework layout and isolation arrangements in relation to HVAC(*H*eating, *V*entilation and *A*ir *C*onditioning) intakes and control rooms). The layout should therefore consider the potential impact on operators performing nuclear safety-related tasks or firefighting to reduce the risk of asphyxiation to "As Low As Reasonably Practicable" (ALARP).

As highlighted previously, ONR expectations from the Safety Assessment Principles (SAPs) are that non-combustible or fire-retardant and heat-resistant materials should be used throughout the facility (EHA.17), safety measures should be identified to deliver the required safety functions (EKP.5), and that these measures should be prioritised according to the hierarchy in SAPs para. 155. It should therefore be expected that ONR will sample the level of compartmentation provided, including coverage and design of reinforced concrete nuclear safety barriers or equivalent against the combined high temperature and gas loading.

MOLTEN SALT REACTORS

Molten salt reactors include reactors in which the molten salt can be considered as acting as both the coolant and fuel (the fuel is dissolved in the salt) and those which use a solid fuel (either in suspension in the molten salt coolant or in rods) but are cooled by the molten salt. The second type use moderators such as graphite, in which case the potential for oxidation and fires associated with air ingress faults are in scope and should be considered in the safety case. An example of such graphite-moderated molten salt design is the Molten Salt Reactor Experiment (MSRE), built and operated at the Oak Ridge National Laboratory (United States) in the 1960s and the majority of MSRs use a graphite moderator.

The high operating temperature of MSRs can give rise to consequential fire and explosion hazards as a result of primary circuit leaks and accidental heating of high flashpoint fluids. Polymerisation of lubricants leaked into the primary circuit leading to blockages and accumulation of fission products in filters has also been reported – the decay heat can result in consequential fires and activity releases. Elimination, or when not possible, minimisation of hydrocarbon-based and combustible lubricants should be expected, in line with ONR SAPs EHA.13.

The chemistry of certain MSRs based on fluoride salts can be controlled by hydrogen and metal fluoride additions. Adequate control prevents fluoride attack to iron and nickel contents in materials. The specific approaches and chemicals are, of course, dependent upon the molten salt chemistry and design choices. The Aircraft Reactor Experiment (ARE) at Oak Ridge National Laboratory, for example, used a mixture of mixture of hydrogen and anhydrous hydrogen fluoride at 600 ºC [25]. Whilst the use of hydrogen for chemistry control, turbo-generator cooling, etc. is part of established light water designs and power generation, the compact nature of small modular reactors, their potential use in fuel / molten salt handling plant could result in a more direct threat to the reactor and fuel handling system upon release and ignition during normal power operations. Accidental releases of hydrogen, with the associated fire and explosion hazards, should be considered in the context of reactor, fuel and molten salt purification plants, and attention should be given to inventory and layout (to minimise the size of any releases) and the consequences of fire and explosions if a release was to occur.

The inventories, layout and storage arrangements of molten salts in relation to combustible materials and fluids (including cabling, lubricants, generator fuels) are also a key consideration in any designs where molten salts are proposed not only as reactor coolant, but as a heat transfer fluid or for energy storage. In these cases, large storage tanks holding molten salts at temperature are often required. For example, the potential for fire and explosion as a result of releases of the thermal salt and impingement on diesel or other combustible inventories should be recognised and prevented preferably by layout and suitably designed passive features such as bunds that are proven capable of containing the hot fluid.

CONCLUSIONS

ONR has recently developed a new technical guidance for Generic Design Assessment (GDA) and undertaken an in-depth review of the Internal Hazards Technical Assessment Guide (revision 5). Additionally, ONR has recently reviewed fire considerations associated to Generation IV reactor types including sodium and lead-cooled fast reactors, high temperature gas reactors and molten salt reactors.

Whilst the philosophy and design characteristics of Generation IV reactors are different from existing power reactors sites in the UK, and current experience from the Generic Design Assessment in the UK, it is generally considered that regulatory expectations as documented in the SAPs and internal hazards TAG are applicable and proportionate to the assessment of internal fire hazards. This paper has presented a series of considerations /areas of attention by ONR internal hazards inspectors when assessing internal fire in Generation IV designs:

- The expectation that the consequences of fires are evaluated on an unmitigated basis.
- Resilience to fire hazards should be preferably achieved by good design layout and passive safety measures that do not rely on active systems or human intervention. It is generally expected that measures will be implemented according to the hierarchy of measures in the SAPs, and in line with the defence-in-depth philosophy. The hierarchy should not be interpreted to mean that the provision of a measure towards the top of the list precludes provision of other items where they can contribute to defence in depth. ONR expects that all reasonably practicable measures will be implemented.
- Fire detection and fire-fighting systems of capacity, capability and compatibility commensurate with the worst-case design basis scenarios should be provided. The indirect effects of fire e.g. corrosivity and toxicity of smoke products can be more onerous in these technologies, so due attention should be given to the minimisation of manual firefighting.
- Past Operational Experience on non-light water reactor designs offers useful insights for designers and safety case specialists alike on fire initiation mechanisms, consequence assessment and plant responses. These emphasise the value of early layout considerations, material choices and selection of detection and suppression options.

Whilst a review of ONR guidance for compatibility with Generation IV technologies is ongoing, it is not envisaged that it would result in the need for substantial changes to ONR's regulatory expectations on internal fire.

REFERENCES

[1] Office for Nuclear Regulation (ONR): Safety Assessment Principles for Nuclear Facilities. 2014 Edition Revision 0, November 2014, www.onr.org.uk/saps/.

[2] Office for Nuclear Regulation (ONR): Technical Assessment Guides (TAGs): Internal Hazards, NS-TAST-GD-014, Revision 4, September 2016, http://www.onr.org.uk/operational/tech_asst_guides/index.htm.

[3] HM Government, Department for Business, Energy and Industrial Strategy: Industrial Strategy, Nuclear Sector Deal, June 2018, https://assets.publishing.service.gov.uk/government/uploads/system/uploads/attachment_data/file/720405/Final_Version_BEIS_Nuclear_SD.PDF.

[4] Office for Nuclear Regulation (ONR): New Nuclear Power Plants: Generic Design Assessment Technical Guidance, DFA-GD-007, Revision 0, May 2019, http://www.onr.org.uk/new-reactors/reports/onr-gda-007.pdf.

[5] Garcia, M., et al.: Theoretical assessment of particle generation from sodium pool fires, Nuclear Engineering and Design, Vol. 310, pp. 470-483, 2016, https://doi.org/10.1016/j.nucengdes.2016.10.024.

[6] Baskaran, R., et al.: Sodium Aerosol Studies for Fast Reactor Safety, Energy Procedia, Vol. 7, pp. 660-665, 2011, https://doi.org/10.1016/j.egypro.2011.06.089.

[7] Yanagisawa, T.: Monju, modified, Nuclear Engineering International, 2011, http://www.neimagazine.com/features/featuremonju-modified/.

[8] International Atomic Energy Agency (IAEA): Status of fast reactor research and technology development, IAEA-TECDOC-1691, Vienna, Austria, 2012, https://www.iaea.org/NuclearPower/Downloadable/FR/tecdoc-1691-status-of-fr-research-published.pdf.

[9] Diwakar, S. V., et al.: Numerical prediction of fire extinguishment characteristics of sodium leak collection tray in a fast breeder reactor", Nuclear Engineering and Design, Vol. 241, Issue 12, pp. 5189-5202, 2011, https://doi.org/10.1016/j.nucengdes.2011.09.015.

[10] US NRC, "Program on Knowledge Management for Liquid-Metal-Cooled Reactors", NUREG/KM-0007, 2014, https://www.nrc.gov/docs/ML1412/ML14128A346.pdf.

[11] International Organization for Standardization (ISO): BS ISO 834-10:2014: Fire resistance tests - Elements of building construction – Part 10: Specific requirements to determine the contribution of applied fire protection materials to structural steel elements, ISBN 978-0-580-78004-2, 2014, https://shop.bsigroup.com/ProductDetail/?pid=000000000030258985.

[12] Saito, N., et al.: Ignition and extinguishment of sodium fires in air diluted by nitrogen, National Research Institute of Fire and Disaster, Proceedings, 5th AOSFST, 2001.

[13] Deukkwang, A., et al.: Suppression of sodium fires with liquid nitrogen, Fire Safety Journal, Vol. 58, pp. 204-207, 2013, https://doi.org/10.1016/j.firesaf.2013.02.001.

[14] European Committee for Standardization (CEN): Eurocde 2: Design of concrete structures, Part 1-2: General rules - Structural fire design, BS EN 1992-1-2 European December 2004, https://www.phd.eng.br/wp-content/uploads/2015/12/en.1992.1.2.2004.pdf.

[15] Lipsett, S. G.: Explosions from molten materials and water, Fire Technology, 2(2), pp. 118-126, 1966.

[16] Reid, R. C.: Rapid Phase Transitions from Liquid to Vapor, Advances in Chemical Engineering 12, pp 105-208, 1983.

[17] Fletcher, D. F., and T. G. Theofanous: Recent progress in the understanding of steam explosions, Journal of Loss Prevention in the Process Industries, 7 (6), pp. 457-462, November 1994.

[18] Schubach, S. A., and D. F. Bagster: Some aspects of modelling steam explosions. Journal of loss prevention in the process industries, 9(3), pp.193-197, 1996.

[19] Dinh (2007), T.-N.: Multiphase Flow Phenomena of Steam Generator Tube Rupture in a Lead-Cooled Reactor System: A Scoping Analysis, paper 7497, in: Proceedings of ICAPP 2007, Nice, France, May 13-18, 2007.

[20] Sibamoto, Y., Y. Kukita, and H. Nakamura H.: Simultaneous Measurement of Fluid Temperature and Phase during Water Jet Injection into High Temperature Melt, paper 181, in: Proceedings of 11th International. Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-11), Avignon, France, October 2-6, 2005.

[21] Moriyama, K., and H. S. Park: Parameter dependence of steam explosion loads and proposal of a simple evaluation method, Nuclear Engineering Technology 47, pp 907-914, 2015.

[22] Xu, W., et al.: The influence of nuclear graphite oxidation on air ingress accident of HTR-PM, Annals of Nuclear Energy, 110, pp. 1242-1248, December 2017.

[23] Ding, H., et al.: Development of emergency planning zone for high temperature gas-cooled reactor", Annals of Nuclear Energy, 111, pp. 347-353, January 2018.

[24] Beck, J. M., and L. F. Pincock: High Temperature Gas-Cooled Reactors Lessons Learned Applicable to the Next Generation Nuclear Plant, INL/EXT-10-19329, Revision 1, April 2011,
https://doi.org/10.2172/1023461.

[25] Anderson, M., et al.: Heat Transfer Salts for Nuclear Reactor Systems - chemistry control, corrosion mitigation and modelling, CFP-10-100, DOE/NEUP--10-905, 2015,
www.iaea.org/inis/collection/NCLCollectionStore/_Public/46/130/46130016.pdf.

Canadian Fire Protection Regulatory Requirements for Small Modular Reactors

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ABSTRACT

Small modular reactors (SMRs) and advanced nuclear reactors have been the subject of significant discussion and study during the past decade. There has been growing international interest in the concept of SMRs as a possible way of introducing nuclear generating capacity in smaller and more affordable increments.

Designs of these SMRs claim that they are inherently safe and employ passive safety systems as compared to conventional reactors. Nonetheless, SMR designs may present unique fire protection challenges. This paper highlights that the National Building Code of Canada (NBCC) and the National Fire Code of Canada (NFCC) are not sufficient by themselves to achieve CNSC's regulatory objectives for fire protection. In regulating SMRs, the CNSC applies the same criteria used to regulate traditional reactor facilities by employing a risk-informed approach. In addition, this paper provides information that supports the need for SMRs to comply with CSA N293 "Fire Protection for Nuclear Power Plants". With a graded approach, the application of requirements is to be commensurate with the risks and particular characteristics of the facility or activity.

In general, new nuclear facilities are required to be designed, constructed and operated in accordance with modern codes and standards. However, performance-based approaches can be used to achieve the intended objectives of prescriptive requirements through good engineering practices.

INTRODUCTION

In recent years, novel reactor technologies have emerged to supply power to smaller electrical grids or to remote, off-grid areas. These novel technologies are commonly called small modular reactors (SMRs). SMRs are viewed by many experts as the potential way of the future in nuclear technology.

SMRs and advanced nuclear reactors have been the subject of significant discussion and study during the past decade. There has been a growing international interest in the concept of SMRs as a possible way of introducing nuclear generating capacity in smaller and more affordable increments. The designs of SMRs claim that they are inherently safer and employ passive safety systems as compared with conventional reactors. However, SMR designs may present unique fire protection challenges that can be a significant contributor to the overall nuclear plant safety. These challenges are not effectively captured in the current Canadian Building and Fire Codes.

This paper highlights that the National Building Code of Canada (NBCC) [1] and the National Fire Code of Canada (NFCC) [2] are not sufficient by themselves to achieve CNSC's regulatory objectives for fire protection. All reactor facilities, including SMRs, are classified as Class IA nuclear facilities under the Class I Nuclear Facilities Regulations [3] as mandated by the Nuclear Safety and Control Act [4] and the General Nuclear

Safety and Control Regulations [5]. In regulating SMRs, the CNSC applies the same criteria used to regulate traditional reactor facilities using a risk-informed approach.

This paper provides information that supports the need for SMRs to comply with CSA N293 “Fire Protection for Nuclear Power Plants” [6] for SMRs in addition to the NBCC and the NFCC. In general, new nuclear power plants are required to be designed, constructed and operated in accordance with modern codes and standards. Where the application of the prescriptive requirements of codes and standards is limited, the performance-based approaches can be used to achieve the intended objectives of prescriptive requirements through good engineering practices.

OVERVIEW OF SMR REACTORS

SMRs are viewed by many experts as the potential way of the future in nuclear technology. The term SMR generally refers to a nuclear reactor facility that is usually smaller than a traditional nuclear power plant and that may employ multiple novel technological approaches, such as:

- passive/inherent safety features; and
- extensive use of factory-built modules.

Common terminologies used internationally to describe such designs include advanced reactor technologies and advanced modular reactors. SMRs can vary significantly in size, design features and cooling types. Examples of different SMR technologies include:

- integral pressurized water reactors;
- molten salt reactors;
- high-temperature gas reactors;
- liquid metal cooled reactors; and
- solid state or heat pipe reactors.

According to the IAEA [7], there are about 50 SMR designs and concepts globally. These SMRs are often defined as advanced reactors that produce electricity of up to 300 MW_{el} per module. Most of them are in various developmental stages and some are claimed as being near-term deployable. There are currently four SMRs in advanced stages of construction in Argentina, China and Russia, and several existing and newcomer nuclear energy countries are conducting SMR research and development.

In Canada, there are about five different SMR designs that are undergoing vendor design reviews at different stage (cf. Table 1).

Table 1 SMRs that are undergoing vendor design reviews in Canada

	Vendor	Name of Design and Cooling Type
1	Terrestrial Energy’s Inc.	IMSR –integral molten salt reactor
2	Ultra Safe Nuclear Corporation	MMR-5 and MMR-10 – high temperature gas
3	Advanced Reactor Concept Ltd.	ACR-100 – liquid sodium
4	Moltex Energy Canada Inc.	Moltex Energy Stable Salt Reactor – molten salt
5	SMR LLC (Holtec International Company).	SMR-160 – pressurized light water

Through CNSC's webpage the following links provide additional information related to SMRs from a regulatory perspective:

- New reactor facilities [8];
- Pre-licensing vendor design reviews [9];
- DIS-16-04, *Small Modular Reactors: Regulatory Strategy, Approaches and Challenges* [10];
- REGDOC-1.1.5, *Licence Application Guide: Small Modular Reactor Facilities* [11]; and
- *Stakeholder Workshop Report: Application of the Graded Approach in Regulating Small Modular Reactors* [12].

POTENTIAL FIRE HAZARDS IN SMALL MODULAR REACTORS

SMR designs differ from the current operational power plant (NPP) designs. SMRs can vary significantly in size, design features and cooling types. Examples of different SMR technologies include the following: integral pressurized water reactors; molten salt reactors; high-temperature gas reactors; liquid metal cooled reactors and solid state or heat pipe reactors.

SMRs and current NPPs in Canada may share some similarities in fire hazards. However, there may be some unique differences. For example, liquid-metal-cooled SMR designs presents unique fire hazards associated with metal fires involving liquid sodium coolants. The unique characteristics of metal fires such as high temperatures presents fire protection challenges. The advanced reactor concepts that operate with high temperatures will affect the auto-ignition temperature of exposed combustibles. In addition, structures, systems and components may be exposed to high temperatures for prolonged periods which may affect their structural integrity.

It is important to note that on one hand the total fire load in SMR may be less than the fire load in current NPPs. However, considering the relatively smaller footprint of SMRs, the fire load densities may be higher than current conventional reactors. This means that the fire dynamics considering smaller floor areas may result in higher heat impact on fire separations (barriers). Furthermore, liquid-metal-cooled SMR designs, that uses liquid sodium as primary coolant, will have a very high fire load as compared to current NPPs in Canada. In general, quantities and locations of the fire hazard vary among NPPs and it will vary as well among the various type of SMRs.

In general, the potential fire hazards that may be found in SMRs include (but are not limited to) the following examples:

- fire hazards associated with reactor coolants (e.g., sodium coolants);
- oil fire hazards associated with various pumps (e.g., for motors, pumps diesel fuel fire hazard at diesel-driven generators);
- fire hazard associated with electrical cable insulation, pipe insulation;
- fire hazard of ordinary combustibles;
- low-level radioactive waste material (e.g., paper, plastic, rubber shoes and gloves, etc.);
- fire hazard associated with filtering materials including charcoal and high-efficiency particulate air (HEPA) filters;
- fire hazard associated with flammable off gases;
- fire hazard of protective coatings;
- fire hazard of turbine lube oil and hydrogen seal oil;

- hydrogen cooling gas fire hazard in turbine generator buildings;
- hydrogen generated in battery room as a result of overcharging a battery; and
- fire hazard associated with electrical switchgear, motor control centers (MCCs), electrical cabinets, load centers, inverter, circuit boards, and transformers.

The impact of the fire hazards on nuclear safety must be considered. Nuclear safety objectives include maintaining safe-shutdown capability and preventing or limiting radioactive releases to the environment. To meet the objectives of fire protection, the fire hazards and their consequences are required to be carefully evaluated to ensure adequate fire protection measures are in place to mitigate the fire risk and achieve the nuclear safety objectives.

OVERVIEW OF NATIONAL BUILDING AND FIRE CODE OF CANADA

The NBCC is an objective-based national model code. The provisions of the NBCC are considered the minimum acceptable measures for meeting the objectives of safety, health, structural protection, and fire protection of buildings.

Generally, when a new building code is adopted, it is not applied retroactively: existing buildings that comply with the code in effect at the time of their construction are generally not required to be upgraded so that they comply with the new code except when the building is being altered or modified. The NBCC is concerned with the health, safety, accessibility and the protection of buildings from fire or structural damage. It applies to the construction of new buildings and to the demolition or relocation of existing ones. It also applies when a building's use changes or when it is significantly renovated or altered.

Similar to the NBCC, the NFCC is also an objective-based national model code. Its provisions are considered the minimum acceptable measures required to adequately achieve specific objectives with respect to the safety, health, and fire protection of buildings and facilities.

The NFCC applies to the operation or use of buildings and facilities and regulates activities that create fire hazards. Unlike NBCC, the NFCC contain retroactive requirements that apply to all buildings, regardless of when they were built.

Specifically, the NFCC includes provisions for:

1. The on-going maintenance and use of the fire safety and fire protection features incorporated in buildings;
2. The conduct of activities that might cause fire hazards in and around buildings;
3. Limitations on hazardous contents in and around buildings;
4. The requirement for fire safety plans; and
5. Fire safety at construction and demolition sites.

The NFCC aims to reduce the likelihood of fires, particularly those that may present a hazard to the community, and to limit the potential damage caused by fires as well as by the handling and storage of hazardous materials.

LIMITATION OF THE NBCC AND NFCC FOR SMRS

The NBCC and NFCC are *model building codes* that are adopted or enacted by provincial or other federal agencies.

- NBCC and NFCC do not adequately address the unique risks associated with nuclear facilities. In this respect, the NBCC states "… code provisions do not necessarily address all the characteristics of buildings that might be considered to have bearing on the Codes objectives".

- NBCC and NFCC provide only the minimum acceptable measures to achieve fire and life safety in conventional buildings. Quoting from the NBCC, "… because the NBCC is a model code, its provisions can be considered as the minimum acceptable measures required to adequately achieve the above listed objectives, as recommended by the Canadian Commission on Building and Fire Code".
- Clause 3.1.1.2 of the NFCC states dangerous goods classified as radioactive materials shall be stored in conformance with CNSC SOR/2000-209, "Nuclear Safety and Control Act (S.C. 1997, c.9)." [4].
- NBCC allows for combustible construction, which may not be adequate for storage and handling of radioactive materials.
- NFCC does not require the facility to prepare a Fire Protection Program (FPP). The FPP is a set of planned, coordinated, controlled and integrated activities which are required by regulations, codes and standards listed in the operating licence and good engineering practice to support achievement of the fire protection objectives for a facility.
- NFCC does not require the facility to carry out and prepare a documented Fire Hazard Assessment (FHA) and Fire Safe Shutdown Analysis (FSSA). The FHA is a set of analyses and assessments for evaluating potential fire hazards as well as the appropriate fire protection systems and features used to mitigate the effects of a fire. On the other hand, the FSSA is an analysis to demonstrate that at least one means of achieving nuclear safety objectives and performance criteria is available in the event of a fire.

In consideration of the above points, additional requirements are recommended for SMRs to ensure that the requirements of the NSCA and the supporting regulations for the protection of the health and safety of persons and the environment are met. Currently, the CNSC requires SMRs to develop and implement a fire protection program in accordance with CSA N293.

CANADIAN FIRE PROTECTION REGULATORY MODEL FOR NEW REACTORS

The CNSC's regulatory model in fire protection is based upon the implementation of the defence-in-depth concept (REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants [13]) to ensure the protection of the health and safety of persons and the environment.

From a fire protection perspective, defence in depth is achieved through a combination of design (e.g., physical barriers, spatial separation, fire protection detection and suppression systems), management of fire protection (e.g., operational procedures), quality assurance and emergency arrangements. The defence-in-depth applies to fire protection at all levels of the facility and its associated activities, from establishing high-level facility objectives to defining the detailed procedures and equipment required to meet those objectives.

To achieve a high level of confidence that the fire protection goals will be met, an appropriate level of defence-in-depth should be maintained throughout the lifetime of the facility, through the fulfilment of the five elements of the defence-in-depth principles (cf. Figure 1).

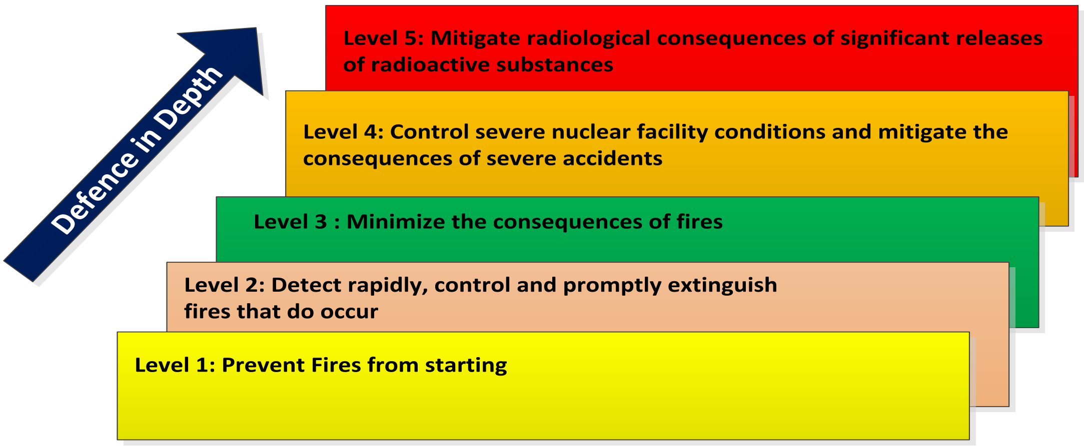

Figure 1 The five levels of defence in depth with respect to fire protection

The CNSC REGDOC 2.5.2 [13] requires that suitable incorporation of operational procedures, redundant SSCs, physical barriers, spatial separation, fire protection systems, and design for fail-safe operation achieves the following general objectives:

- prevent the initiation of fires;
- limit the propagation and effects of fires that do occur by quickly detecting and suppressing fires to limit damage and confining the spread of fires and fire by-products that have not been extinguished;
- prevent loss of redundancy in safety and safety support systems;
- provide assurance of safe shutdown;
- ensure that monitoring of safety-critical parameters remains available, and
- prevent exposure, uncontrolled release, or unacceptable dispersion of hazardous substances, nuclear material, or radioactive material, due to fires;
- prevent the detrimental effects of event mitigation efforts, both inside and outside of containment; and
- ensure structural sufficiency and stability in the event of fire.

APPLICATION OF CSA N293 TO SMRS

To achieve the fire protection objectives, the REGDOC 2.5.2 requires new NPPs to comply with CSA N293.

The CSA N293 provides the minimum fire protection requirements for the design, construction, commissioning, operation, and decommissioning of nuclear power plants, including structures, systems, and components that directly support the plant and the protected area in Canada.

The requirements of the CSA N293 standard along with its referenced standards applies to all licensed plants in Canada where referenced as part of the license conditions of the licensee. The standard is also used as the baseline for review of fire protection requirements for all new reactors including SMRs.

CSA N293 is a consensus-based standard that addresses fire protection requirements intended to reduce the risk of fires at Nuclear Power Plants. CSA N293 establishes the fire protection requirements for the design, construction, commissioning, operation, and decommissioning of nuclear power plants to address the fire protection goals and objectives. CSA N293 requirements include:

- Defence-in-depth requirements. This requires that design provision multiple levels of defence in order to prevent accidents and ensure appropriate protection and mitigation in the event of an accident;
- Design requirements for the prevention and control of fires, the mitigation of fire hazards, and the protection of plant occupants, equipment, and structures. These requirements also specify some of the means to achieve the nuclear safety and fire protection goals (e.g., redundant systems and separation between redundant fire safety shutdown systems);
- Design requirements for the design, installation, and performance of fire alarm systems, fire detection, fire suppression systems and also to specify the requirements for fire resistance rating of building structures, building materials and egress;
- Operational requirements (e.g., control of ignition sources, inspection testing and maintenance (ITM) of fire protection features, control of flammable, combustible materials);
- Fire Protection Program requirements;
- Requirements fire protection assessments (FPA). The FPA is a set of analyses and/or assessments which demonstrate that the objectives of nuclear safety, radioactive release prevention, life safety, and economic loss prevention are achieved. The FPA include the following: code compliance review (CCR); fire hazard assessment; and fire safe shutdown analysis; and
- Fire response and decommissioning.

As noted, the CSA N293 is the applicable standard for SMRs. Using a graded approach, the application of the requirements in the standard is to be commensurate with the risks and particular characteristics of the facility or activity. The CSA N293 allows for the intent of the requirements to be met through alternative means and permits the use of performance-based approaches.

Currently, the CSA 293 committee is investigating potential challenges and limitations in the application of the CSA to SMRs. The results of this analysis may lead to some enhancement to CSA N293 in the future revision.

CONCLUSIONS

The National Building and Fire Codes of Canada by themselves are not sufficient for SMRs. The CNSC's regulatory model in fire protection is based upon the implementation of the defence-in-depth concept. CNSC requires that suitable incorporation of operational procedures, redundant SSCs, physical barriers, spatial separation, fire protection systems, and design for fail-safe operation to ensure that the requirements of the NSCA and the supporting regulations for the protection of the health and safety of persons and the environment are met. Therefore, SMRs are required to comply with CSA N293 to achieve fire protection goals and objectives.

With a graded approach, the application of the fire protection requirements is commensurate with the risks and particular characteristics of the facility or activity. In addition, the CSA N293 allows for the intent of the requirement to be met through alternative means. The performance-based approaches can be used to achieve the intended objectives of prescriptive requirements through good engineering practices.

REFERENCES

[1] National Research Council Canada: National Building Code of Canada (NBCC) 2015, Toronto, ONT, Canada, 2015.

[2] National Research Council Canada: National Fire Code of Canada (NFCC) 2015, Toronto, ONT, Canada, 2015, https://nrc.canada.ca/en/certifications-evaluations-standards/codes-canada/codes-canada-publications.

[3] Canadian Nuclear Safety Commission (CNSC): Class I Nuclear Facilities Regulations, SOR/2000-204, Nuclear Safety and Control Act, Ottawa, ONT, Canada, May 2000, https://laws-lois.justice.gc.ca/eng/regulations/sor-2000-204/page-1.html.

[4] Canadian Nuclear Safety Commission (CNSC): Nuclear Safety and Control Act (NSCA), S.C. 1997, c. 9, paragraph 3(a), subparagraph 9(a)(i), paragraph 24(4)(b), last amended on January 1, 2017, https://laws-lois.justice.gc.ca/PDF/N-28.3.pdf.

[5] General Nuclear Safety and Control Regulations, paragraph 12(1)(c), (f), (d), paragraph 17(e), last amended on June 12, 2015, https://laws.justice.gc.ca/PDF/SOR-2000-202.pdf.

[6] Canadian Standard Association (CSA): Fire protection for nuclear power plants, CSA N293-12 (R 2017), Toronto, ONT, Canada, 2012.

[7] International Atomic Energy Agency (IAEA): Advances in Small Modular Reactor Technology Development, 2018 Edition, Supplement to IAEA Advanced Reactors Information System (ARIS), Austria, September 2018, https://aris.iaea.org/Publications/SMR-Book_2018.pdf.

[8] Canadian Nuclear Safety Commission (CNSC) webpage link for new reactor facilities, http://www.nuclearsafety.gc.ca/eng/reactors/power-plants/new-reactor-facilities/index.cfm.

[9] Canadian Nuclear Safety Commission (CNSC) webpage link for pre-licensing vendor design reviews, http://www.nuclearsafety.gc.ca/eng/reactors/power-plants/pre-licensing-vendor-design-review/index.cfm.

[10] Canadian Nuclear Safety Commission (CNSC) webpage link for Small Modular Reactors: Regulatory Strategy, Approaches and Challenges, DIS-16-04, http://www.nuclearsafety.gc.ca/eng/acts-and-regulations/consultation/history/dis-16-04.cfm.

[11] Canadian Nuclear Safety Commission (CNSC): REGDOC-1.1.5, Licence Application Guide webpage link for Small Modular Reactor Facilities, https://www.nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/published/html/regdoc1-1-5/index.cfm.

[12] Canadian Nuclear Safety Commission (CNSC): Stakeholder Workshop Report: Application of the Graded Approach in Regulating Small Modular Reactors, Held on November 24, 2017, nuclearsafety.gc.ca/eng/acts-and-regulations/consultation/history/workshop-report-regulating-small-modular-reactors.cfm.

[13] Canadian Nuclear Safety Commission (CNSC): REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, Ottawa, ONT, Canada, May 2014, http://www.nuclearsafety.gc.ca/pubs_catalogue/uploads/REGDOC-2-5-2-Design-of-Reactor-Facilities-Nuclear-Power-Plants-eng.pdf

Proposed Performance-Based Code Alternative Solution Models in the Nuclear Industry

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ABSTRACT

Canada is presently regulating in the third Code cycle since the introduction of an objective-based national model Code after adopting the objective-based format in 2005. The objective-based Code included adding the alternative regulatory path to permit acceptable equivalent solutions to supplement the listing of acceptable solutions that include both prescriptive rand performance-based requirements. These equivalent solutions, called alternative solutions, meet the objectives and functional statements of the Code.

The demonstrated compliance can take many forms, such as: research papers, engineering judgement, fire computational models, fire engineering calculations and engineering and/or administrative controls. The comparison should be between the performance of the compliant case to the proposed solution and not the risk. This paper proposes a new performance system model, a new proposed performance-based Code alternative solution model for existing as well as for new construction.

INTRODUCTION

Since 2005, the model National Building Code of Canada has included an alternative solution application protocol, permitting the proposal of original materials and designs to meet the objective and functional statements[1] that prompt the full intent of the Code. The building Code in each province or territory is generally based on the model Code with additions, deletions and amendments. Authorities having jurisdiction (AHJ) in Canada are presently governing in the third Code cycle since the introduction of an objective-based national model Code. The first National Building Code of Canada (NBCC) to adopt an objective-based format was issued in 2005. The Canadian Commission on Building and Fire Codes issues an updated version of the major Codes (Building, Fire, Plumbing and Electrical) nearly every 5 years. The current National Model Building Code is the 2015 edition, with a 2020 edition about to be issued late in 2020 or early in 2021 [1].

The change to an objective-based Code included adding the alternative regulatory paths to permit acceptable equivalent solutions to supplement the listing of prescriptive solutions for well-known building materials and assemblies. By defining the goals of the Code via cross-referenced objectives and functional statements, the objective-based format endeavors to give designers and Code officials' methods to evaluate a potential design for Code conformance. The term "performance-based" Codes is widely used, but it is not easy to relate how it is used in fire safety to use in other fields. Ken Richardson´s publi-

1 The functional statements are interconnected with the objectives and describe conditions to help satisfy the objectives (Preface National Building Code of Canada 2015).

cation [2] defined a widely accepted framework for performance-based Fire Safety Codes:

1) The outcome of the Code is stated in terms of "lives and property saved" and generally specifies a "level of safety";
2) The Codes and verifiable performance requirements are demonstrated and quantifiable links to the "Code objectives"; and
3) The Codes permit any solution that meets the performance requirements.

An acceptable solution is well thought out, as a set of provisions which when met will deliver the desired performance as intended by the objectives and performance requirements. The first official acceptable solutions were usually the old prescriptive Codes. Therefore, when objective-based regulations were first presented, the old Code commonly became the acceptable solution. Over time, the identification of these solutions became more obvious and fastened to the objectives and performance requirements they are satisfying [3]. It is noted that the NBCC is an objective-based Code and is considered a steppingstone toward a fully performance-based Code. However, the development of an alternative solution to satisfy the intended performance level of the Code can be based on a performance-based approach.

The starting point for any performance-based fire safety engineering calculations are design fire scenarios which include such aspects as the location of the fire, building characteristics, occupant response, fire loads, fire protection systems, etc. [4], [5]. A design fire scenario also includes a design fire (a quantitative description of fire characteristics within the design fire scenario) which is typically defined as a heat release rate (HRR) time history; but will often also include species production rates and the effective heat of combustion [4], [6].

Twenty years of experience with performance-based fire protection design (PBFPD) has been reviewed. It was concluded that shortcomings and challenges still need to be addressed in order to expand the application of the PBFPD [7]. The PBFPD needed to include political aspects, social effects, life safety, property protection, business continuity, heritage preservation, environmental protection and more [8]. A framework for PBFPD for the Built Environment was introduced in 2014 (see Figure 1) for examining the first portion of the PBFPD process established for example by the Society of Fire Protection Engineers (SFPE), one can realize that the impact of the stakeholders remains significant up to the elaboration of the design brief, that is to say even before the fire protection engineer (FPE) looks for evaluating trial designs [9].

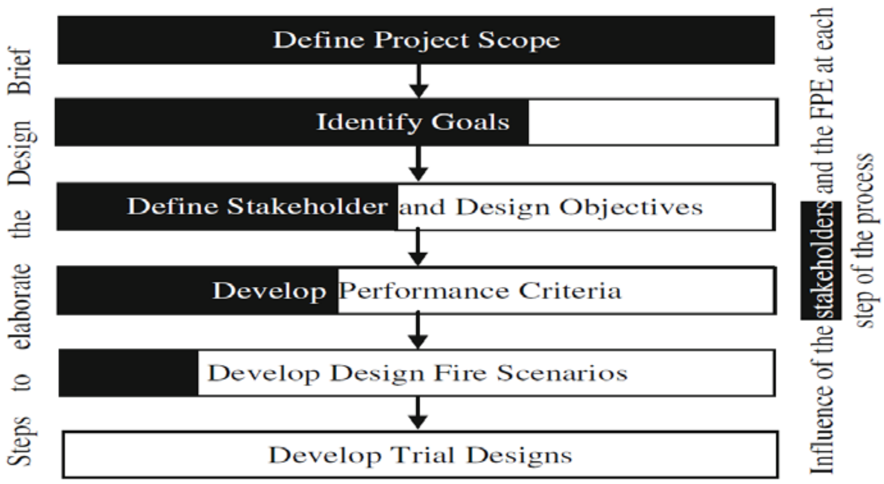

Figure 1 Framework for risk-informed performance-based fire safety analysis [10]

Performance-based codes are perceived and have many advantages; such as being less conservative and more cost-effective, yielding cost savings by designing for a particular building, having more flexibility, allowing the use of new technologies and materials. They are more suitable for large or unusual buildings and provide explicit objectives and performance requirements [11].

If a code provision cannot be met by a design or existing facility, trade-offs can be suggested, analyzed and costed, as a basis for informed decisions. This is specifically significant when dealing with historic structures which cannot be brought up to the Code without destroying their unique qualities [12]. Applying performance-based codes is also beneficial when there is a requirement for budget modification and during value engineering reviews [13].

DEVELOPMENT OF THE PROPOSED MODEL

Background

The permission to use alternative solutions is found in the NBCC, Clause 1.2.1.1.(1)(b) of Division A.

Compliance with this Code

Compliance with this Code shall be achieved by

a) complying with the applicable acceptable solutions in Division B (see Note A-1.2.1.1.(1)(a)), or

b) using alternative solutions that will achieve at least the minimum level of performance required by Division B in the areas defined by the objectives and functional statements attributed to the applicable acceptable solutions (see Note A-1.2.1.1.(1)(b)).

For the purposes of compliance with this Code as required in Clause 1.2.1.1.(1)(b), the objectives and functional statements attributed to the acceptable solutions in Division B shall be the objectives and functional statements referred to in Subsection 1.1.2. of Division B [14].

Proposed Models

In order to demonstrate compliance, the FPE must ensure that the alternative solution meets both the objectives of the Code and functional statements which are always paired as well as the intended performance level. The demonstrated compliance criteria can be met either qualitatively or quantitatively. The demonstrated compliance can take many forms, such as: research papers, engineering judgement, fire computational models, fire engineering calculations and engineering and/or administrative controls. The FPE should compare the performance of the compliant case to the proposed solution and not to the risk (see Figure 2).

Code alterative solutions for existing building can sometimes be challenging. Many old buildings are not built up to the current Code and, in some cases, it is very difficult and expensive to get these buildings up to the current Code. The proposed model in Figure 3 indicates that there are two paths for alternative solutions. The first path is done through engineering judgement or research papers and is mainly for construction materials conformity.

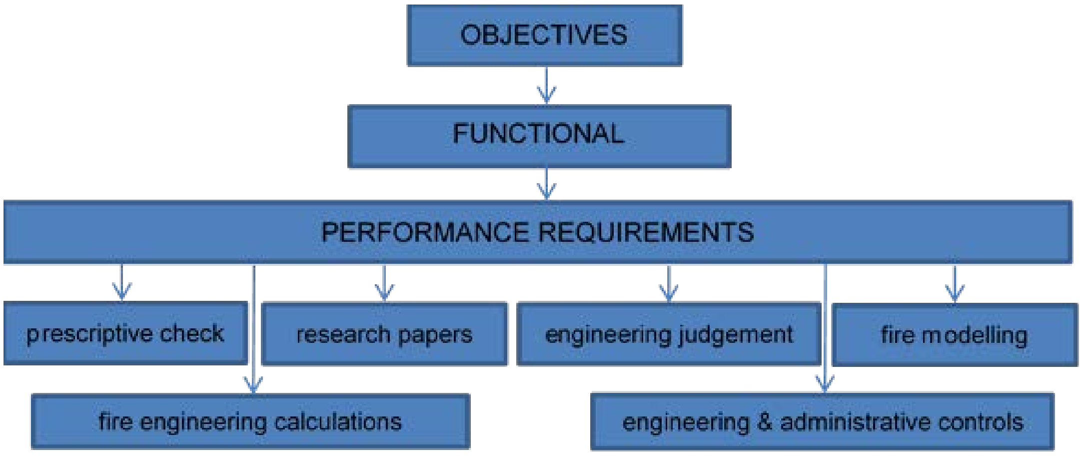

Figure 2 Performance system model

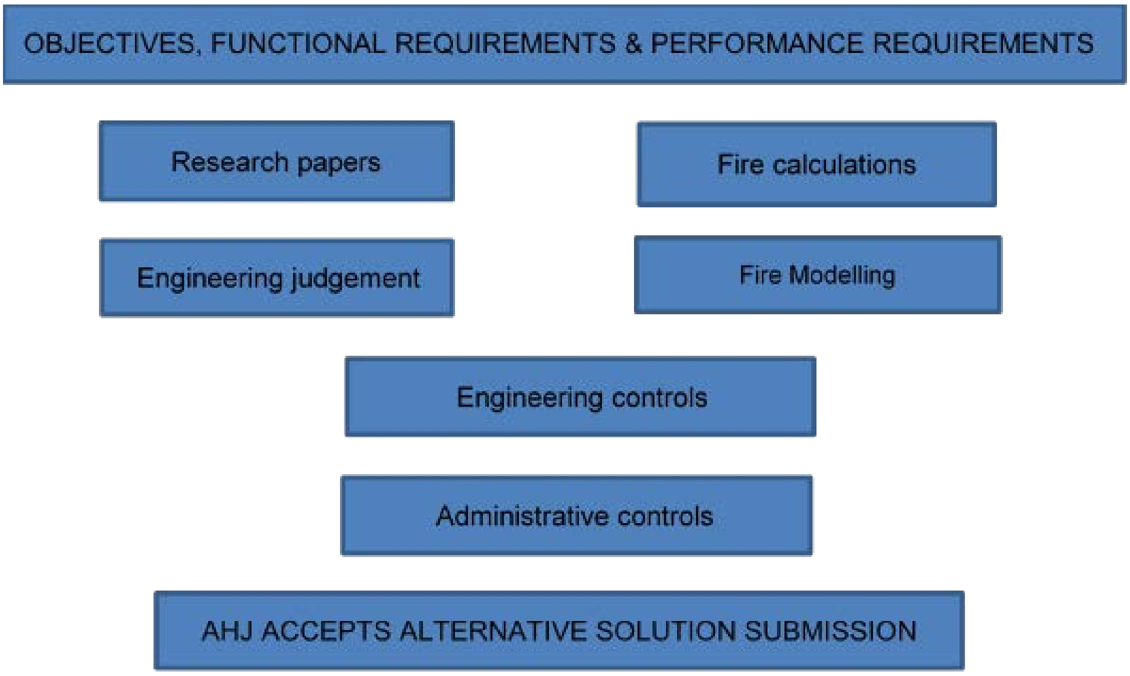

Figure 2 Proposed performance-based code alternative solution model for existing constructions

The second path is used to find an alternative solution for code non-compliance with the acceptable solutions such as: spatial separation deficiency or the need to add a sprinkler system in the building, a differing exiting configuration, etc. This is mainly done through fire modelling and calculations. Usually, the first step of this path is to develop the credible worst-case fire scenario and consequences to evaluate fire growth and development, smoke spread, detection and suppression performance, occupant response/evacuation, building members' failure, fire spread and fire department response.

If the FPE needs to reduce the consequences severity of the fire scenario developed, the FPE can suggest reducing the consequences by adding engineering controls, such as: fire alarm and detection systems, voice communication systems, additional fire barriers, specific fire suppression means, etc. The other option for the FPE to reduce the consequences of the severity is to accompany the engineering controls with administrative controls or just having only administrative controls, such as: routine maintenance, regular inspection, written operating procedures, standards and policies, work practices, etc.

This paper proposes an alternative solution for new construction in Figure 4. This proposed model has three steps; the first step is for the FPE to discuss with the AHJ and agree on the criteria of evaluation, following it by submitting an alternative solution package to the AHJ. This package should have: Building Code analysis report, performance criteria, which will establish that the objectives are met, relevant fire scenarios and developed design fires, the performance criteria which will be used to evaluate the acceptability of an alternative solution, risk and reliability analysis, fire modelling, performance-based design aspects with respect to compliance with the intent of specific Code requirements and operations and maintenance manuals.

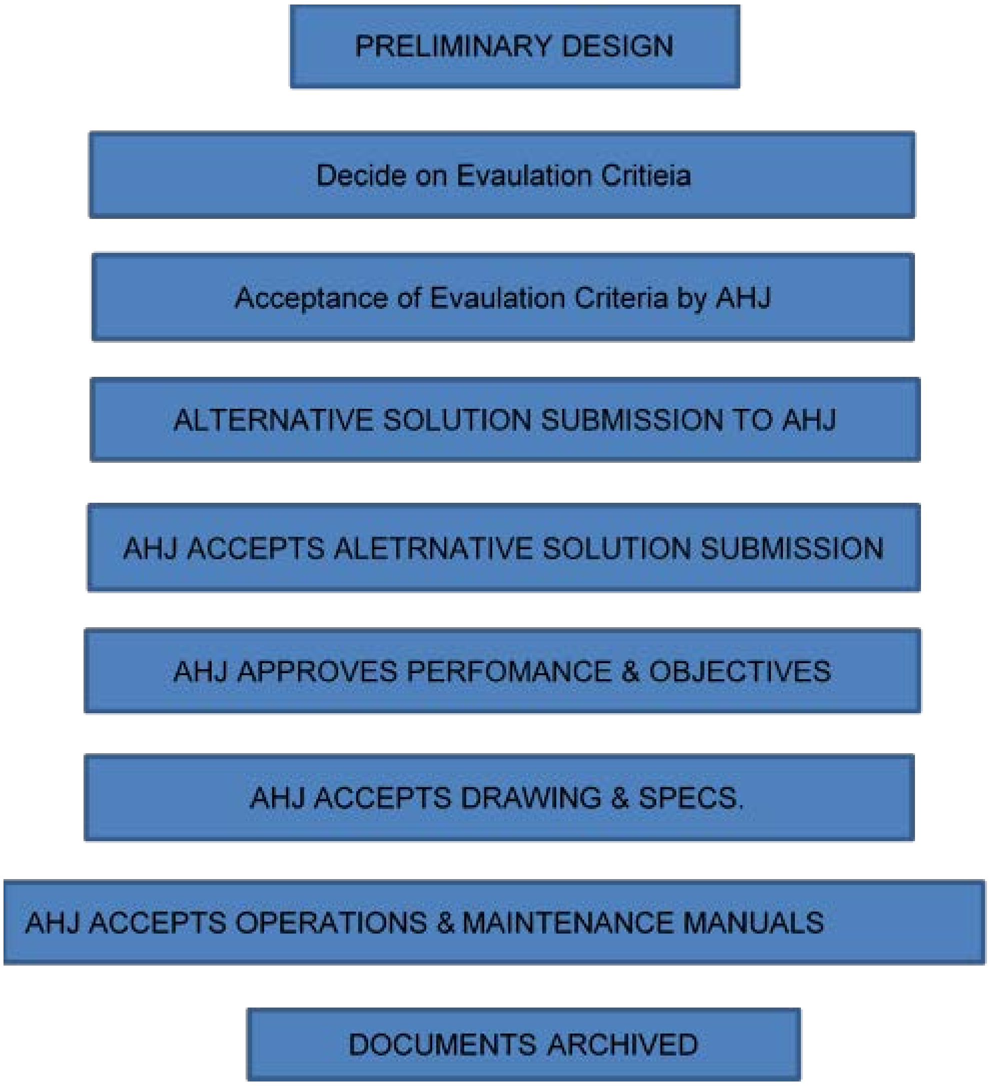

Figure 3 Proposed performance-based code alternative solution model for new constructions

The second step would be for the AHJ to review and assess the alternative solution package. The final step is to archive documents and receive permission to build with the alternative solution design.

Case Study 1

There is a high-volume space in an existing building that is not sprinklered. The existing building is required by the applicable legislation and the AHJ to comply with the requirements of the NBCC 2015. It is not desired to install a sprinkler system in the space due to potential downtime and costs related to installing a sprinkler system in this space as equipment limits access and no construction is permitted at the same time as operations are occurring in the building.

The construction requirements for the building require the building to be fully sprinklered. For the purposes of this example, the construction requirements are based on Article 3.2.2.86. of Division B of the NBCC. As per Clause 3.2.2.86.(1)(a), the building is required to be fully sprinklered.

To understand the acceptable level of performance, the objectives and functional statements (attributions) related to Clause 3.2.2.86.(1)(a) of Division B are to be evaluated.

The following functional statements and objectives are applicable to the portion of Clause 3.2.2.86.(1)(a) of Division B related to sprinklering of the building: *[F02,F04-OS1.2,OS1.3]* and *[F02,F04-OP1.2,OP1.3]*.

The functional statements and objectives are always paired and are to be read as a pairing to understand the performance level that is to be addressed such that the following pairs apply: *[F02-OS1.2], [F02-OS1.3], [F04-OS1.2], [F04-OS1.3], [F02-OP1.2], [F02-OP1.3], [F04-OP1.2] and [F04-OP1.3]*.

The referenced objectives and functional statements are stated below:

<u>Applicable Objectives</u>

<u>*OS1 Fire Safety*</u>

An objective of this Code is to limit the probability that, as a result of the design or construction of the *building*[2], a person in or adjacent to the *building* will be exposed to an unacceptable risk of injury due to fire. The risks of injury due to fire addressed in this Code are those caused by:

1) OS1.2 – fire or explosion impacting areas beyond its point of origin; and
2) OS1.3 – collapse of physical elements due to a fire or explosion.

<u>*OP1 Fire Protection of the Building*</u>

An objective of this Code is to limit the probability that, as a result of its design or construction, the *building* will be exposed to an unacceptable risk of damage due to fire. The risks of damage due to fire addressed in this Code are those caused by:

1) OP1.2 – fire or explosion impacting areas beyond its point of origin; and
2) OP1.3 – collapse of physical elements due to a fire or explosion.

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2 Terms in italics are defined in the Code as per Sentence 1.4.1.2.(1) of Division A.

Applicable Functional Statements

F02: To limit the severity and effects of fire or explosions

F04: To retard failure or collapse due to the effects of fire

An example of how the functional statements and objectives are to be read together are indicated below for the pairing of [F02-OS1.2].

An objective of this Code is to limit the probability that, as a result of the design or construction of the *building*, a person in or adjacent to the *building* will be exposed to an unacceptable risk of injury due to fire. The risks of injury due to fire addressed in this Code are those caused by fire or explosion impacting areas beyond its point of origin (OS1.2) to limit the severity and effects of fire or explosions (F02).

Intent statements for the NBCC have been released and, these intent statements assist to understand the intended performance level of the Code and should be consulted.

For attribution [F02,F04-OS1.2,OS1.3] related to sprinkler coverage:

- To limit the probability that loadbearing walls, columns and arches exposed to fire will prematurely fail or collapse, which could lead to the failure or collapse of supported floor or roof assemblies during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to harm to persons;
- To limit the probability that floor or roof assemblies exposed to fire will prematurely fail or collapse during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to harm to persons;
- To limit the probability that loadbearing walls, columns and arches exposed to fire will prematurely fail or collapse, which could lead to the failure or collapse of supported floor assemblies, which could lead to the spread of fire from a lower storey of a building to an upper storey or to the exterior during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to harm to persons;
- To limit the probability that floor or roof assemblies exposed to fire will prematurely fail or collapse, which could lead to the spread of fire from a lower storey of a building to an upper storey or to the exterior of the building during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to harm to persons;
- To limit the probability that, in the event of a fire, the absence of fire suppression systems within a storey will lead to the growth of fire, which could lead to the spread of fire within the storey during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to harm to persons.

For attribution [F02,F04-OP1.2,OP1.3] related to sprinkler coverage:

- To limit the probability that loadbearing walls, columns and arches exposed to fire will prematurely fail or collapse, which could lead to the failure or collapse of supported floor or roof assemblies during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to damage to the building;
- To limit the probability that floor or roof assemblies exposed to fire will prematurely fail or collapse during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to damage to the building;
- To limit the probability that loadbearing walls, columns and arches exposed to fire will prematurely fail or collapse, which could lead to the failure or collapse of supported floor assemblies, which could lead to the spread of fire from a lower storey of a building to an upper storey or to the exterior during the time required to achieve

occupant safety and for emergency responders to perform their duties, which could lead to damage to the building;

- To limit the probability that floor or roof assemblies exposed to fire will prematurely fail or collapse, which could lead to the spread of fire from a lower storey of a building to an upper storey or to the exterior of the building during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to damage to the building;
- To limit the probability that, in the event of a fire, the absence of fire suppression systems within a storey will lead to the growth of fire, which could lead to the spread of fire within the storey during the time required to achieve occupant safety and for emergency responders to perform their duties, which could lead to damage to the building.

With respect to sprinkler coverage, the acceptable performance level is related to life safety and protection of the building.

An approach that could be investigated to justify the performance in the existing building is to determine the credible worst-case fire scenarios and undertake fire modelling to assess the impact of not providing sprinklers versus sprinklering the high-volume space. When sprinklers operate, they have been shown to control fires which leads to increased occupant safety as well as property protection. The intent of sprinklers is to provide a level of safety to persons in the building as well as emergency responders who will respond to the building (permitting emergency responders to perform their duties without undue risk). Sprinklers will also help to mitigate damage to the building.

A deterministic analysis could be used; however, the choice of design fire is key in this assessment. In this building, there are strict administrative controls on the location and quantity of combustibles. Using a design fire based on the available combustibles, the impact of the sprinkler system can be assessed. If the design fire is such that the sprinkler system at the ceiling level of the high-volume space is not expected to active the sprinklers, this is the baseline performance of the acceptable solution that is to be compared to. If the sprinklers are expected to be actuated, the performance level of the acceptable solution is to be determined relative to the objectives and functional statements and any solution is to have a performance level that is at least as good as the performance level of the acceptable solution. Will the fire spread from its point of origin or will elements collapse affect the safety of persons and the building? This is to be assessed and compensating measures are to be provided so that the minimum performance level can be achieved.

The AHJ should be consulted early in the process (if possible) and not at the submission of the alterative solution. For example, the design fire is expected to be a key component of this analysis and acceptance of the design fire is an instrumental step for the analysis. Once the analysis is complete and documented, it is submitted to the AHJ for acceptance. It is at the discretion of the AHJ to accept the alternative solution, however, prior engagement of the AHJ to address any concerns with the approach will help to mitigate concerns of the AHJ.

Case Study 2

In an unsprinklered four-storey building, due to processing activities, openings through floors are required impacting the integrity of the floor fire separations such that all storeys are interconnected (i.e., there is a four-storey interconnected floor space). The processing activities are integral to the operations in the building.

For this example, it is challenging to determine the baseline performance level of the acceptable solutions in the Code. Sentence 3.2.8.2.(3) of Division B specifically exempts closures in openings in a fire separation, such as a floor fire separation, that would disrupt the nature of manufacturing processes, provided there are precautions taken to offset the resulting hazard. A related Note (Appendix Note) refers to guidance in NFPA 80, “Standard for Fire Doors and Other Opening Protectives” [15] and NFPA 13, “Standard for the Installation of Sprinkler Systems” and states [16] “*Combinations of methods may be required to ensure that the level of safety inherent in the requirements of the Code is maintained*” but what objectives and functional statements apply? What is the inherent level of safety?

Although Sentence 3.2.8.2.(3) of Division B is an exemption for special features that would apply when a building contains an interconnected floor space (the example building contains a four-storey interconnected floor space), the sentence contains a requirement “precautions are taken to offset the resulting hazard” such that the following objectives and functional statements have been assigned to this provisions: [F03-OS1.2] and [F03-OP1.2].

The objectives of the requirement in this provision are similar to the first example, although there is a different functional statement associated with the applicable objectives (F03: To retard the effects of fire on areas beyond its point of origin). The acceptable solution is related to mitigating the impact of fire on people and the building based on limiting fire spread.

The intent statement for attribution [F03-OS1.2] should be consulted and are as follows:

<u>Intent 1:</u>

To exempt openings in fire separations used in manufacturing operations (e.g., openings used for the flow of material from storey to storey) from the requirements of Sentence 3.2.8.1.(1) and Articles 3.2.8.3. to 3.2.8.8., which would otherwise require a vertical fire separation or certain fire protection measures, if equivalent fire protection is provided to compensate for the lack of closure.

This is to limit the probability that fire will spread through the openings to other parts of the building, which could lead to harm to persons in other parts of the building.

The intent statement for attribution [F03-OP1.2] should be consulted and are as follows:

<u>Intent 2:</u>

To exempt openings in fire separations used in manufacturing operations (e.g. openings used for the flow of material from storey to storey) from the requirements of Sentence 3.2.8.1.(1) and Articles 3.2.8.3. to 3.2.8.8., which would otherwise require a vertical fire separation or certain fire protection measures, if equivalent fire protection is provided to compensate for the lack of closure.

This is to limit the probability that fire will spread through the openings to other parts of the building, which could lead to damage to the building.

In this case, the FPE may consider solutions from NFPA 80 and provide additional features such as administrative controls on combustibles, increased detection and special suppression and could evaluate the performance via modelling. The strictest interpretation of the evaluation of the baseline performance is to consider the performance of the building if no openings are provided between floors. However, the FPE is to use judgement to determine the if the hazards are offset and the objectives and functional statements are addressed by the proposed solution.

It should be noted that attributions (objective and functional statements) are assigned only to requirements such that an alternative solution for a noncompliance with an exemption of an acceptable solution is to be based on the root requirement. For example, in building required to be of non-combustible construction, there are permissions to use combustible elements in these buildings that are indicated in Subsection 3.1.5. of Division B. If, for example, a combustible interior wall finish is desired but is more than 25 mm thick as permitted by Sentence 3.1.5.12.(2) of Division B, the applicable objective and functional statements would be based on the root requirement of Sentence 3.1.5.1.(1) of Division B ([F02-OS1.2] and [F02-OP1.2]) but the intended performance level would take into consideration the performance associated with the permission of Sentence 3.1.5.12.(2) of Division B.

CONCLUSION

This paper illustrated the requirements to adopt and/or develop the new performance system model, a new proposed performance-based Code alternative solution model for existing as well as for new constructions that can be used in nuclear industry.

REFERENCES

[1] Canadian Commission on Building and Fire Codes: National Building Code of Canada, 2015 Edition (National Research Council of Canada), Ottawa, ONT; Canada, 2015.

[2] Richardson, J. K.: Moving Towards Performance Based Codes, NFPA Journal, Vol. 88, No. 3 May/June, pp. 70–8, 1994.

[3] Bergeron, D., B. Bowen: Acceptable Solutions, Institute for Research in Construction, National Research Council of Canada, Wellington, New Zealand, April 2001.

[4] Fleischmann, C. M.: Is Prescription the Future of Performance-Based Design?, in: Fire Safety Science – Proceedings of the 10th International Symposium, International Association for Fire Safety Science, pp. 77-94, 2011.

[5] Hurley, M. J., E. R. Rosenbaum: Performance-Based Design, in: P. J. Dinenno, et al., Editors, The SFPE Handbook of Fire Protection Engineering (4th Edition), National Fire Protection Association (NFPA), Quincy, MA, USA, pp. 3-440–3-445, 2008.

[6] Department of Building and Housing: C/VM2 Verification Method: Framework for Fire Safety Design – for New Zealand Building Code Clauses C1-C6 Protection from Fire, Wellington, New Zealand, 2012.

[7] Alvarez, A. et al.: 20 years of performance-based fire protection design: challenges faced and a look ahead, Journal of Fire Protection Engineering 23, pp. 249–276, 2013.

[8] Alvarez, A, B. J. Meacham: Towards an integrated performance-based design approach for life safety across different building use groups. in: Franks, C. A., S. Grayson (Eds.) Interflam – Proceedings of the 12th International Fire Science and Engineering Conference, pp. 1259–127, Interscience Communications Ltd., London, United Kingdom, 2010.

[9] Society of Fire Protection Engineers (SFPE): SFPE Engineering Guide to performance-based fire protection, 2nd Edition, Bethesda, MD, USA, 2007.

[10] Alvarez, A., et al.: A Framework for Risk-Informed Performance-Based Fire Protection Design for the Built Environment, Fire Technology, DOI:10.1177/1042391513494923, Vol. 50, pp. 161-181, 2014.

[11] Ramsay, G. C.: Fire Safety Engineering Codes Role In Performance-Based Codes, Scientific Services Laboratory, Australian Government Analytical Laboratories, Port Melbourne, Victoria 3207, Australia1988.

[12] International Code Council (ICC): ICC (2000) International Building Code, Chapter 34, Falls Church, VA, USA, 2000.

[13] Meacham, B., et al.: Performance System Model – A Framework for Describing the Totality of Building Performance, in: Proceedings of the 4th International Conference on Performance-Based Codes and Fire Safety Design Methods, Society of Fire Protection Engineers (SFPE), Melbourne, Australia, 2002.

[14] Canadian Commission on Building and Fire Codes, and National Research Council of Canada: National Building Code of Canada (NBCC) 2015., Toronto, ONT, Canada, 2015, https://nrc.canada.ca/en/certifications-evaluations-standards/codes-canada/codes-canada-publications.

[15] National Fire Protection Association (NFPA): Standard for Fire Doors and Other Opening Protectives. NFPA 80, 2019 Edition, Quincy, MA, USA, 2019, https://www.nfpa.org/codes-and-standards/all-codes-and-standards/list-of-codes-and-standards/detail?code=80.

[16] National Fire Protection Association (NFPA): Standard for the Installation of Sprinkler Systems, NFPA 13, 2019 Edition, Quincy, MA, USA, 2019, https://www.nfpa.org/codes-and-standards/all-codes-and-standards/list-of-codes-and-standards/detail?code=13.

3.2 Session on “Operating Experience and Lessons Learned”

The second Seminar session addressed operating experience from nuclear power plants and the respective lessons learned. In total, eight contributions were provided.

The first presentation by Appendix R Solutions from the United States was devoted to the development and application of a specific program for the control of combustibles in nuclear installations, which allows to significantly reduce the total amount e.g. of plastic sheeting material in the plant.

The second presentation provided insights and the corresponding corrective actions from fire events having more recently occurred in operating Finnish nuclear power plants with a specific focus on roof fires, a fire in the transformer yard and high energy arcing faults. The resulting plant improvements covered modifications of different structures, replacements of components by a different component type and enhanced procedures.

In light of some countries being in nuclear phase-out, fire at a nuclear power plant site occurring during decommissioning and nuclear waste treatment become more and more relevant. The operating experience so far has indicated that incipient smouldering fires during and after the drying process of various radioactive materials may occur. Since their prevention is difficult, early detection and suppression are important.

A comprehensive overview on the international fire incidents database of the OECD Nuclear Energy Agency (NEA) FIRE representative for several hundreds of fire events during the whole lifetime (from construction up to decommissioning) nuclear power plants of different countries from Europe, Asia and Northern America was provided by the OECD/NEA Secretariat in charge for this database project. In the contribution several applications of the database as well as the challenges were discussed.

In a presentation from Framatome, Germany the behaviour of fire detection and control systems in case of smoke spreading through fire barriers was discussed with respect to its safety significance. In this context, it is relevant that such systems may close fire dampers, stop the ventilation flows and interrupt the cooling of the affected rooms by air flows through ventilation ducts.

Moreover, another presentation by an expert from GRS, Germany presented more recent insights from the evaluation of the operating experience with fire dampers in nuclear power plants in Germany focussing on fire tests according to European standards and potential effects of fire damper failures on the required functions of items important to safety .

A third contribution devoted also to fire detection and control was provided by Framatome, Germany highlighting some experience, mainly regarding the electrical qualification of such systems.

The last presentation of this session provided a first rough overview on organization and responsibilities of the fire brigades in nuclear power stations of different countries as well as the corresponding challenges in case some of the fire events. The investigations are ongoing, based on a survey, and will result in a more detailed report in the near future.

The Seminar contributions of this session are provided hereafter.

Insights on Successful Combustible Controls Programs

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ABSTRACT

As plant facilities develop stronger fire protection programs, one element that is often included is a combustible controls program. In practice a combustible controls program provides a framework in which to evaluate the amount of combustible material is allowed within a designated area. Under deterministic fire protection regulation, nuclear power plants are limited on the amount of combustible material that may be allowed within certain size rooms and the classification of the rooms based on the use and components present. A key element that should be considered when managing materials under a combustible control program should include an evaluation of whether the material is required or not required. As an example, U.S. nuclear plants have used a significant amount of plastic sheeting material. Over the past decade, U.S. nuclear power plants (NPPs) have implemented increased combustible controls and have undergone a shift. They are now not only using qualified materials where possible but are implementing an approach similar to the ALARA program. The goal is to reduce the quantities of sheeting materials to as low as reasonably achievable. This change has significantly reduced the total amount of plastic sheeting material in the plants. It is important to note that even if the material is allowed under the current combustible control limits for the area, the material may contribute to risk beyond an acceptable threshold.

This paper provides insights on combustible controls programs including a case study of sample sheet materials typical of what might be found in a Japanese nuclear power plant. For each of the sample materials the material was informally tested using a butane flame source to observe the material combustibility. The results of each test were documented by video, along with related observations. A review of the available information for each material was performed. The conclusions and observations of this study are based on physical informal burn tests of each material.

INTRODUCTION

Most nuclear plants, both in the United States and around the world use sheeting materials such as plastic sheeting for a variety of purposes, including cleanliness controls, foreign material exclusion, contamination control, and even combustible controls.

Where these materials are used for combustible controls, the basic premise for the use of the sheet-type covering material is to improve safety consistent with the defense-in-depth philosophy. This is achieved by covering combustible materials and equipment so there is a reduced chance of the combustible materials or combustible equipment contributing to an area fire. The covering material could, to some extent, isolate the covered combustible material and equipment. In order for this approach to be successful, the covering materials must be resistant to the influence of fire, sparks, and hot work that could otherwise ignite the covered materials. The covering material should be clearly non-combustible. This is not an uncommon approach and can be seen in many applications such as during welding or cutting, etc.

Because most of these materials appeared to be plastic-type products, ARS selected samples for informal testing to gain insights for combustible control programs.

DEFINITIONS AND ABBREVIATIONS

There are several terms used when discussing combustibility. Often these terms are used incorrectly or in the wrong context. In addition, there may be several similar definitions for the same term. The following is provided for clarification and is considered the best definition relative to the subject of this report.

Combustible: A combustible material is any material that, in the form in which it is used and under the conditions anticipated, will ignite and burn or will add appreciable heat to an ambient fire [1].

- Non-combustible: In the form used and under the conditions anticipated, this material will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat [1].
- Limited Combustible: This refers to a building construction material not complying with the definition of non-combustible that, in the form in which it is used, has a potential heat value not exceeding 3500 Btu/lb (8141 kJ/kg) when tested in accordance with NFPA 259, and it includes either
  (1) materials having a structural base of non-combustible material, with a surfacing not exceeding a thickness of 1⁄8 in. (3.2 mm) that has a flame spread index not greater than 50, or
  (2) materials, in the form and thickness used having neither a flame spread index greater than 25 nor evidence of continued progressive combustion, and of such composition that surfaces that would be exposed by cutting through the material on any plane would have neither a flame spread index greater than 25 nor evidence of continued progressive combustion when tested in accordance with ASTM E84, Standard Test Method for Surface Burning Characteristics of Building Materials, or UL 723, Tests for Surface Burning Characteristics of Building Materials [1].
- Fire Resistant: (1) A structure meeting the requirements for Type I or Type II construction; (2) a building of Type I or Type II (222) construction in which the structural members, including walls, partitions, columns, floors, and roofs are of non-combustible or limited-combustible materials. [1]; or (3) a material that is inherently resistant to catching fire (self-extinguishing) and does not melt or drip when exposed directly to extreme heat [3].
- Fire Retardant: (1) A coating that reduces the flame spread index of Douglas fir, or of any other tested combustible surface to which it is applied, when tested in accordance with a test for assessing surface burning characteristics (Reference 0), or (2) a material that has been chemically treated to self-extinguish [3]).
- Flame Retardant: A material constructed or treated so that it will not support flame [1].
- Flame Resistance: The property of a material whereby combustion is prevented, terminated, or inhibited following the application of a flaming or non-melting source of ignition, with or without subsequent removal of the ignition source [4].
- Incombustible: (1) Not a recognized term by [1]. (2) It can mean not combustible**;** incapable of being burned [2].
- Self-extinguish: (1) Not a recognized term by [1]. (2) Not a recognized term by [2]. (3) Self-extinguishing is the flame spread resistance, which depends on the product group (clothing, cloth, plastics, etc.). Each group is different, and its exact definition

is given by the appropriate standards. In general, the product or material is self-extinguishing if the flame is extinguished after the flame source is removed as directed by the respective standard (derived from [5] with ARS clarification edits).

CASE STUDY

ARS obtained samples of sheeting materials used in a nuclear power plant for informal testing using a butane flame source to observe the material combustibility. For this case study, eight samples of sheeting materials that were in use for various purposes were obtained for examination. The materials are sheet-type and are typically provided on rolls for large volume use. The attachment to this paper contains amongst others a photograph of a typical roll of a sheet material. In each case, the materials were identified as “non-combustible” or “incombustible”.

Material Observations

Visual observations prior to testing were:

- Sample A - medium-weight sheet-type material, white;
- Sample B - light-weight sheet-type material, white;
- Sample C - heavy-weight sheet-type material, gray pattern;
- Sample D - heavy-weight plastic sheet-type material, off-white;
- Sample E - light-weight plastic sheet-type material, pink;
- Sample F - light-weight plastic sheet-type material, light green;
- Sample G - light-weight plastic sheet-type material, blue;
- Sample H - light-weight sheet-type material, white.

The review of vendor information provided the following characterizations of the materials:

- Sample A is a fire proofing cloth or refractory cloth, also identified as a welding spark cloth; and resistant to molten metal.
- Sample B is identified by the manufacturer as a “flameproof sheet”, and it is 100 % polyester (PVC coating on polyester yarn).
- Sample C is a carbide fiber with raw material of acryl fiber. This material is qualified to JIS 1323 Type A # 03A2527 [5].
- Sample D no test data or qualification information was available.
- Sample E is identified as flame retardant to UL 94, VTM-0.
- Sample F is identified as flame retardant to UL 94, VTM-0.
- Sample G is s identified as flame retardant to UL 94, VTM-0.
- Sample H is a PVC flame proofing on a polyester base.

Methodology

Eight samples of the materials used at a nuclear power plant were provided for review and testing. Each material was informally tested by ARS using a butane flame source to observe the material combustibility. Results of each test were documented by video, along with related observations.

A review of the available information for each material was performed. The conclusions and observations of this report are based on physical informal burn tests of each material. These burn tests are much less severe than actual fire exposure because the flame source used during the tests was focused on a limited surface area due to a small flame source in comparison to an actual fire.

Observations

Sample A (cf. Figure 1) appeared to be non-combustible and may be suitable for use as a covering material for plant combustibles and equipment. The material should be verified as non-combustible via manufacturer's documentation and certification prior to continued use.

Sample B (cf. Figure 2) was identified by the supplier as a "flameproof sheet," but it is clearly combustible and is not suitable as a covering material for plant combustibles. This material contributes to the area combustible loading. The material did not continue to burn when the flame was removed.

Figure 1 Sample A, potentially non-combustible

Figure 2 Sample B, consumed by fire

Sample C (cf. Figure 3) is non-combustible, and it is the most suitable material tested for use to cover plant combustibles. Sample C satisfied JIS A 1323 Type A requirements [6]. The material reflected essentially no damage post flame exposure.

During testing, Sample D (cf. Figure 4) was characterized as non-combustible. Upon further review of the test data (video footage), this material should likely be characterized as a combustible material given the flame length when exposed to the butane flame and the amount of smoke produced. This material is not suitable as a covering material for plant combustibles. In addition, the large amount of smoke produced could result in clean-up complications, such as in contaminated areas or areas with sensitive electrical equipment. The material did not continue to burn when the flame was removed. This material contributes to the area combustible loading.

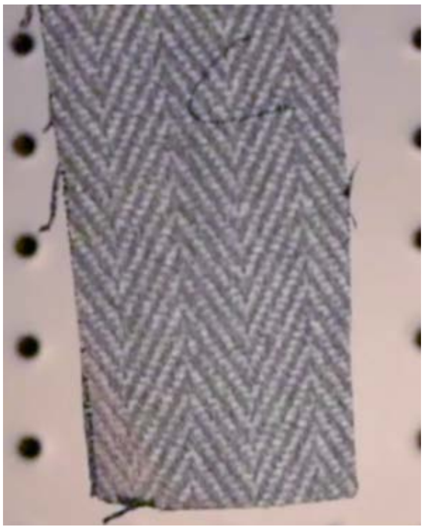

Figure 3 Sample C, no damage **Figure 4** Sample D, smoke production

Sample E (cf. Figure 5) is a combustible material that produces drops of flaming material. This material could result in the spread of fire similar to a flammable liquid. This material continued to burn once the flame was removed. The material was reported to be qualified to UL 94 [7] VTM-0. The classification VTM-0 (or V-0) requires the flaming to stop within 10 seconds on a vertical sample, and drips of particles are allowed as long as they are not flaming. The sample tested did not meet this requirement. This requirement does not satisfy the definition of non-combustible. This material contributes to the area combustible loading. The material is not suitable as a covering material for plant combustibles.

Sample F (cf. Figure 6is a combustible material as it was consumed by the flame. The material did not continue to burn when the flame was removed. The material was reported to be qualified to UL 94 [7] VTM-0. The classification VTM-0 (or V-0) requires the flaming to stop within 10 seconds on a vertical sample, and drips of particles are allowed as long as they are not flaming. The sample tested appeared to meet this requirement. This requirement does not satisfy the definition of non-combustible. This material contributes to the area combustible loading. This material is not suitable as a covering material for plant combustibles. The material potentially could exhibit pooling properties as Sample E when subjected to higher temperatures.

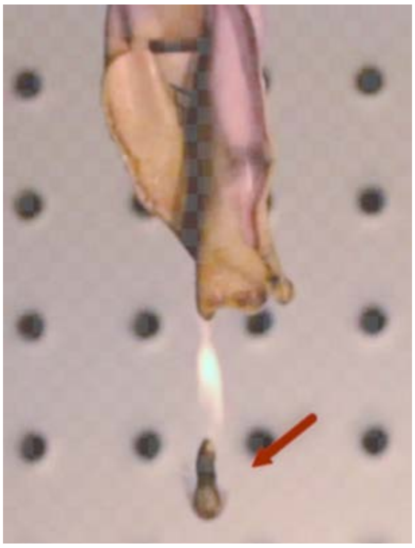

Figure 5 Sample E, flaming drop

Figure 6 Sample F, combustible

Sample G (cf. Figure 7) is a combustible material that produces drops of flaming material. This material could result in the spread of fire similar to a flammable liquid. This material continued to burn once the flame was removed. The material was reported to be qualified to UL 94 [7] VTM-0. The classification VTM-0 (or V-0) requires the flaming to stop within 10 s on a vertical sample, and drips of particles are allowed as long as they are not flaming. The sample tested did not meet this requirement. This requirement does not satisfy the definition of non-combustible. This material contributes to the area combustible loading. The material is not suitable as a covering material for plant combustibles.

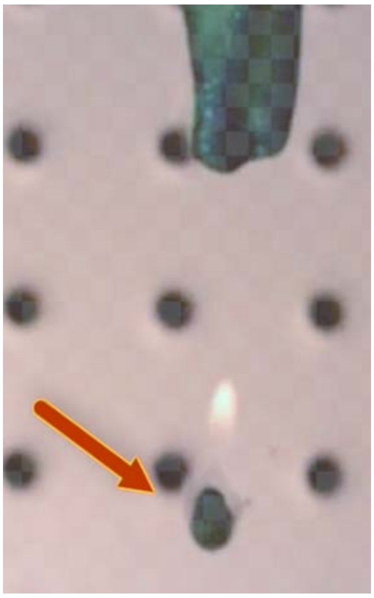

Figure 7 Sample G, no damage

Figure 8 Sample H, smoke production

CONCLUSIONS

In the past, nuclear plants have used a significant amount of plastic sheeting material.

The informal burn tests referenced in this study demonstrate that there is a wide range of material properties for plastic sheeting materials. Some of the materials such as samples A and C which were identified by the manufacturers as “fire proofing cloth”, “refractory cloth” or “flameproof sheet,” were shown to be non-combustible in our simple bench test. These materials are expected to perform such that they will isolate the covered materials from sparks, hot work, flame, and the initial stages of a fire to prevent direct ignition. Based on our tests, other materials tested could be expected to significantly contribute to combustible materials. Even some materials advertised as flame retardant or flameproof were shown to have significant combustibility.

Although Samples A and C are examples of the types of materials that may be suitable for use even as combustible controls to isolate the covered materials from sparks, hot work, flame, and the initial stages of a fire to prevent direct ignition, clearly some of the sample materials tested would not be suitable for combustible controls because the materials will easily degrade, and then the protected materials will be accessible to an exposure fire. In some cases, the sheeting materials tested would be considered combustible and should certainly be minimized in power plant applications.

Over the past decade, U.S. NPPs have implemented increased combustible controls and have undergone a shift. They are now not only using qualified materials where possible, but also are implementing an approach similar to the ALARA (*a*s *l*ow *a*s *r*easonably *a*chievable) approach. This is important because even materials that are resistant to flame spread can contribute to the total combustible loading in and area.

Based on the U.S. experience then, the goal then for the combustible control program should be to reduce the quantities of sheeting materials to as low as reasonably achievable. This change has significantly reduced the total amount of plastic sheeting material in the U.S. plants and is a good model for plants around the globe seeking to reduce the risk of fires in nuclear applications.

ACKNOWLEDGEMENTS

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REFERENCES

[1] National Fire Protection Association (NFPA): NFPA Glossary of Terms, 2019 Edition as of February 8, 2019, Quincy, MA, USA, 2019, https://www.nfpa.org/-/media/Files/Codes-and-standards/Glossary-of-terms/glossary_of_terms_2019.ashx?la=en.

[2] Merriam Webster Dictionary, latest access on March 7, 2019, https://www.merriam-webster.com/dictionary/incombustible.

[3] Gaskets Inc.: Flame Resistant vs. Fire Retardant, Rio, Wi, USA latest access on March 7, 2019, https://www.gasketsinc.com/2017/04/15/flame-resistant-vs-fire-retardant/.

[4] National Fire Protection Association (NFPA): NFPA 2113, Standard on Selection, Care, Use, and Maintenance of Flame-Resistant Garments for Protection of In-

dustrial Personnel Against Short-Duration Thermal Exposures from Fire, Selection, 2015 Edition, Quincy, MA, SA, 2019.

[5] What is the Definition of Self-Extinguishing?, KOPOS KOLÍN a.s., Kolin, Czech Republic, latest access on March 7, 2019, https://www.kopos.com/en/faq/1085-what-definition-self-extinguishing.

[6] Japanese Standards Association: JIS A 1323:1995, Flame Retardant Testing Method For Spark Droplets Of Welding And Gas Cutting On Fabric Sheets In Construction Works, 1995.

[7] Underwriters Laboratories (UL LLC): UL 94 Standard for Safety of Flammability of Plastic Materials for Parts in Devices and Appliances Testing, Fifth Edition, SBN 0-7629-0082-2, Northbrook, IL, USA, October 29,1996, http://www.lyddæmpning.dk/documents/pictu5509.pdf.

[8] Hiraoka & Co. Ltd., Pune, Japan, latest access on March 7, 2019, https://www.tarpo-hiraoka.com/en/.

ATTACHMENT: Examples of Equipment Covering Materials

Fire Events in Finland – Operating Experience and Lessons Learnt

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ABSTRACT

Finnish nuclear power plants (NPPs) started commercial operation in 1977 (Loviisa 1), 1979 (Olkiluoto 1), 1981 (Loviisa 2) and 1982 (Olkiluoto 2). Many fire events have occurred in each of them, although most of the events have been insignificant considering their impact on plant operation and nuclear safety. The paper highlights major fire events through the Finnish NPPs' operation history as well as preventive measures and modifications completed due to the events.

The preventive measures and modifications completed due to the fire events mentioned above include, for example: modifications in different structures, component type replacement, implementation of the electric arc relay protection system in switchgears, and development of procedures.

INTRODUCTION

Finnish nuclear power plants (NPPs) started commercial operation in 1977 (Loviisa 1), 1979 (Olkiluoto 1), 1981 (Loviisa 2) and 1982 (Olkiluoto 2). Many fire events have occurred in each of them, although most of the events have been insignificant considering their impact on plant operation and nuclear safety.

This paper highlights major fire events through the Finnish NPPs' operation history. The fire events to be outlined include the following: a roof fire of the diesel generator building, a high energy arcing fault (HEAF) event with an ensuing fire in a 6 kV switchgear cabinet room, a roof fire of the auxiliary building, and a current transformer explosion with an ensuing fire at the 400 kV switchyard. The first two fire events mentioned above are documented in the OECD FIRE Database [1], and the last two fire events are to be documented in the next version of the OECD FIRE Database. The last fire event mentioned above is handled briefly in STUK's annual report [2].

DIESEL GENERATOR BUILDING ROOF FIRE IN AUGUST 1985

Event Description

After maintenance of the emergency diesel generators, the ten hours test runs were ongoing during plant refueling outage. During the test, mechanics present in the diesel generator building noticed smell and started to look after its origin. Nothing unusual was noticed inside the building and the inspection was extended outside. Then, dense smoke on the roof in the vicinity of the engine exhaust pipes as well as flames around the exhaust pipe of one engine were noticed. After about four minutes, one of the mechanics called the main control room in order to report about the fire event, and an emergency stop of the engine in question was triggered within five minutes after the fire was noticed on the roof. The test runs of the three other diesel generators were also stopped.

The mechanics suppressed the flames by means of portable fire extinguishers (type of extinguishers was not reported). Dismantling of the roofing felt (bituminous water proofing), thermal insulation and bitumen around the exhaust pipe penetrations was started immediately. Portable dry chemical extinguishers were used to secure suppression of the smoldering material during the dismantling. The plant fire brigade arrived at the scene of fire approximately 9 min after the call to the main control room and at that time the fire was already extinguished. Also an offsite fire brigade was alarmed and arrived about twenty minutes after the call to the main control room. The necessary dismantling was completed in three hours after the fire was noticed. Similar dismantling was completed on the neighbouring unit after the next three hours.

Only the roof was affected by the fire. The diesel generators' test runs were stopped, but no damages due to heat, smoke or fire extinguishing agents were reported.

The root cause of the fire was human due to an erroneous design of the structure. The flammable bituminous water proofing was mounted on top of the penetration collar, thus too close to the exhaust pipe, which became hot during the ten hours test run.

Corrective Actions

The following corrective actions have been taken in two similar plant units at the same site:

- The thermal insulation material and the roofing were renewed around the exhaust pipeline penetrations (eight penetrations in total).
- The new roofing felt was not installed on top of the penetration collar, because waterproof metal sheet is used in that part of the structure.

HIGH ENERGY ARCING FAULT AND FIRE IN A 6 KV SWITCHGEAR IN APRIL 1991

Event Description

Initially, the power measurement of one 6.6 kV house load switchgear indicated a faulty value during power operation of the plant, and the personnel started to examine the reason. Some damaged terminal blocks / overheated connections were detected in the secondary circuit of a current transformer. As soon as the terminal block replacement was started, wires in the switchgear cubicle started to release smoke and the plant fire brigade was alerted. An electrician informed the main control room about the situation and all house load switchgears were connected to the power supply from the 110 kV grid via the start-up transformer and shutdown of the plant was prepared. Approximately 6 min after alerting the plant fire brigade, the fire detection system gave first alarm from an optical detector. At that time, the plant fire brigade was already at the fire scene.

Due to the smoke generated inside the switchgear cabinet an electric arc occurred and caused a fire, After approximately 9 min the plant fire brigade was alerted. The overload protection device opened the circuit breaker feeding the switchgear. The plant transformer's overload protection tripped the generator relay protection, which led to a turbine trip and reactor power was reduced to about 40 % level. It took about 4 s before the generator breaker opened after the protection signal actuated, because the generator power reduction was necessarily needed (according to the design basis, the generator breaker could not be opened with high electrical power). Therefore, the generator supplied the arc for a period of approximately 4 s. The electric arc induced a pressure wave, and one person standing in the compartment doorway fell down. One emergency diesel generator started as designed and provided power to the associated safety related bus bar, which had lost normal power supply via the damaged 6.6 kV switchgear.

During the next 7 min fire and/or smoke propagated to the neighboring cubicle containing the cable terminal from the start-up transformer (the breaker itself was previously opened due to the overload protection) and caused another short circuit by arcing (duration of the arc is not reported; at that time nobody was inside the compartment, but a boom was heard): the differential protection device of the start-up transformer opened all circuit breakers still supplying the 6.6 kV house load bus bars and also the breaker on the primary side of the start-up transformer leading to total loss of power supply from the grid. The turbine condenser and many other systems were no longer available resulting in a reactor scram. After loss of power supply from the grid three emergency diesel generators started and provided power to the safety related switchgears, as designed.

The fire brigade extinguished the fire using portable fire extinguishers (CO_2 and Halon) within approximately 30 min after two attacks. The extinguishing was hampered by the uncertainty about potential voltage inside the switchgear cubicles. Furthermore, the closed structure of the cubicles hampered extinguishing and cooling of the overheated material. Offsite fire brigade arrived at the fire scene about 25 min after the initial fire alert, just before the second fire attack was started by the plant fire brigade. The offsite fire brigade was involved only in securing the fire compartments neighbouring to the fire location.

After the fire had been extinguished, efforts to regain external supply from the 110 kV to the three not affected 6.6 kV house load switchgears were started. Before regaining voltage, the failed bus bar had to be disconnected from the transformers by opening the terminals of the connecting cables at the transformers (the cable terminal was destroyed inside the ignited switchgear). This activity took approximately 6.5 h, then two of the four house load switchgears were re-connected to the 110 kV grid. Then the reactor was cooled down to the cold shut down state to replace damaged components and cables. The third house load switchgear was taken in use three days later and the damaged one was fixed up within a week (three cubicles were almost totally destroyed, and one cubicle was destroyed partially).

The initial cause of the event was a break in a current transformer’s secondary circuit causing overheating and damages of the terminal blocks. Later on a short circuit took place in a neighbouring voltage transformer circuit causing overheating and smoke production, when the terminal block replacement was started. The initial break in the current transformer’s secondary circuit was assumed to occur in conjunction with relay testing during the previous maintenance outage and no procedure was in use to identify that kind of assumed incipient failures. Human factors can be seen as root causes.

Corrective Actions

The following corrective actions have been taken in two similar plant units at the same site:

- The maintenance practices of switchgears was improved (a circuit failure was assumed as an initial reason for the event): the integrity of secondary circuits of current transformers is to be tested after relay testing.
- Another start-up transformer was installed, and disconnectors were mounted in the supply connections to reduce vulnerability of the power supply via the start-up transformers. The original plant design contained one start-up transformer per unit.
- Fixed CO_2 extinguishing systems were installed at the 6 kV switchgears. Later on, these extinguishing systems were removed, the switchgears are now equipped with electric arc relay protection systems.

- Separation walls between the switchgear cubicles were improved to prevent smoke spreading.
- Separation of the terminal blocks was enhanced considering current transformers and voltage transformers.
- The generator breaker was renewed to enable fast opening at full power (the previous one was not able to be opened at full power causing a delay to cut off power supply immediately).

AUXILIARY BUILDING ROOF FIRE IN JANUARY 2018

Event Description

Renovation work on the auxiliary building roof had started in May 2017 and was ongoing during plant power operation when a roof fire occurred due to hot work. Previously the same day, a hole had been made to the roofing in order to install a fan for removing moisture from the insulation wool. As part of the installation, bitumen roofing felt was heated with a gas flame. After finishing the hot work some other work was continued and a fire watch was carried out for one hour according to the approved procedure. After an additional half hour all the workers left the area. More than three hours later the fire detection system gave an alarm inside the auxiliary building. Plant fire brigade checked the compartment and noticed smell but no fire. Some time later a fire on the roof was noticed by surveillance cameras.

The firefighters moved to the roof and put out the fire with a temporary fire hose provided there due to the planned hot work. The roof was partially disassembled to find hot spots. The fire extinguishing started about 10 min after the fire detection and took about 20 min. Thereafter, a fire watch was carried out the whole night.

The combustible material consisted of bitumen, wooden structures in the roof and the insulation wool. The wooden structures were remnants of the plant construction phase when large components were hauled through the lift holes.

Smoke and fire water entered the compartments below the fire area, but items important to safety were not affected. Some filters on the exhaust air radioactivity sensors had to be replaced due to reduction of sample flow.

The root cause of the event is inadequate procedure. The event showed that the one-hour fire watch was insufficient and the wooden structures as well as the negative pressure ventilation in the rooms below were not taken into account while assessing fire hazards. The reason for the fire seems to be ignition of wooden parts close to the hot work.

Corrective Actions

The following corrective actions have been taken:

- Improvement of procedures for similar works in the future and other works involving hot work. Improvements concern e.g. extended fire watch, identification of fire loads near hot work sites, reducing hot works when possible and work risk assessment forms.
- The affected opening in the roof was closed permanently with concrete structure.
- Mapping of similar openings at the site. Corrective actions are pending.

CURRENT TRANSFORMER EXPLOSION AND FIRE AT THE SWITCHYARD IN JULY 2018

Event Description

A current transformer suddenly exploded, induced missiles and an ensuing fire at the 400 kV switchyard, which is located at the site about one kilometer away from the NPP units. The event was noticed immediately by an electrician working in the neighbourhood. The same person gave a call to the national emergency centre. In addition, one person of the NPP had noticed smoke outside and the plant security patrol left to check the situation. After approximately 7 min of the initial failure the patrol arrived at the switchyard and confirmed that there was a fire. At the same time, the national emergency centre declared an alarm of an explosive fire event at the power plant (many off-site fire brigades launched due to the alarm). One minute later, the plant operator provided the information that the burning device had been de-energized by the grid operator (the grid operator had a view on the fire scene via a camera). The plant fire brigade arrived at the fire scene approximately 10 min after the fire had been confirmed and several off-site fire brigades arrived about 10 min later.

The current transformer contained approximately 600 l of oil which spread around. On the ground level wooden lids covering a small cable corridor were ignited as well as the cables below the lids. The plant fire brigade used portable CO_2 and dry chemical extinguishers for fighting the fire. The grid operator informed the cables are partially energized and they can not be de-energized, therefore water was not used for fire fighting. Later on, cable damages inside the corridor caused a loss of 400 kV and changeover to 110 kV for Unit 2, which induced stopping of the reactor coolant pumps resulting in a reactor scram of Unit 2 about 20 min after the initial failure. Unit 2 was restarted after two days.

The upper part of the damaged current transformer collapsed approximately 100 min after the initial failure. Some oil still spilled down and ignited several times on hot surfaces. Cooling of the fallen down components was performed by portable CO_2 extinguishers. The current transformer was extinguished after about 3 h, and several attacks were needed. Thereafter, extinguishing of the wooden lids of the cable corridor was completed with fire water within 30 min. Totally, more than 50 portable extinguishers were used during the event.

Efficient extinguishing of cables with water was not possible due to energized cables. Therefore, the cables in the corridor were destroyed by the fire. These cables are used for instrumentation, control and protection of the 400 kV switchyard, which is operated by the national grid company.

During recovery actions made by the grid operator, Unit 1 experienced a loss of 400 kV, continued on house-load operation and stayed at power operation mode until the 400 kV re-connection was available on the next day. Unit 2 was restarted and reconnected to the 400 kV grid after about two days.

The reason for the failure causing the explosion of the current transformer remained unclear (incipient failure causing heat up and pressure increase is assumed).

Corrective Actions

The following corrective actions have been taken:

- Temporary cables were installed for being able to operate the unaffected fields of the switchyard and reconnect the reactor units to the 400 kV grid. Thereafter, replacement of damaged components continued. A new 400 kV switchyard close to the original one was constructed and commissioned in 2019.

- The grid operator mounted metal covers over the cable corridors in those areas where similar current transformers are located (to prevent ignition of wooden lids and oil spreading to the cable corridor in case of similar explosion).
- Similar types of current transformers were replaced by October 2018. The new types contain composite insulation instead of porcelain and they contain about 200 l of oil. The new types are not expected to induce missiles in case of failures.
- The reactor coolant pumps of the Unit 2 speed control system had different settings compared to Unit 1. That was the reason for stopping of the pumps and the reactor scram. The speed control system settings were changed similar to those in Unit 1. The system was modernized during the period 2016 to 2018 and the differences between the reactor units were known thereafter.

CONCLUSIONS

Based on the operational experience and lessons learnt from the four major fire events at Finnish NPPs, the following kinds of corrective actions have been taken:

- Removing combustible material nearby emergency diesel generator exhaust pipeline penetrations on the roof;
- Survey of old lift holes on the roof containing wooden parts, then replacement of wooden structures and permanent closure of the lift hole has been piloted;
- Component type replacement (generator breaker, current transformer);
- Improvements in the power supply system (increased failure tolerance);
- Implementation of the electric arc relay protection system in most of the switch-gears (protection against electric arcing being also required in the safety design of a NPP in Finland [3];
- Enhancing procedures for maintenance practices and hot work.

REFERENCES

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Database Version 2017:02, Paris, France, May 2019 (limited to FIRE Database Project member countries only).

[2] Radiation and Nuclear Safety Authority STUK: Regulatory Oversight of Nuclear Safety In Finland, Annual Report 2018, STUK-B 235, Helsinki, Finland, May 2019, https://www.julkari.fi/handle/10024/138192.

[3] Radiation and Nuclear Safety Authority STUK: Safety Design of a Nuclear Power Plant, STUK Guide YVL B.1, Helsinki, Finland, June 2019, http://www.finlex.fi/data/normit/41774-YVL_B.1e.pdf.

Fires of Radioactive Waste During and After Drying Processes in German Nuclear Power Plants

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ABSTRACT

During the last years three smoldering fires occurred in nuclear power plants (NPPs) in Germany during and after drying processes of radioactive waste. The waste was stored in standard 200 l drums or smaller crumple drums when additional compacting was intended. The drying processes took place in facilities under elevated temperatures of up to 155 °C. One facility operated under air pressure reduced to about 50 mbar improving the water evaporation and increasing the ignition temperature. In this case, the fire started after the drying process, when the drum was parked with open lid.

Different waste materials and the inhomogeneity of waste in-between drums and within a drum make it difficult to assess the risk of ignition. Different ignition processes were reported by the licensees, which could partly be reproduced with hindsight.

For fire detection different technical means such as cameras, CO detectors, heat detectors, and simple smoke detectors were involved. This shows the variety of detection possibilities with different pros and cons.

Fire suppression using carbon dioxide as extinguishing medium failed in one case; however, suppression by water or foam was successful. Since the fires were only in a smoldering stage, the mass turnovers were small, and the radioactive releases were either not measurable or not significant.

INTRODUCTION

During normal operation of nuclear power plants (NPPs), radioactive residues are collected in sumps, steam generators, water drainage systems, hot workshops, etc. inside the controlled area. Many of these residues have a high water content, therefore they need to be dried before further processing. If the waste is prepared for disposal, certain waste acceptance criteria must be met, which also include criteria for the residual water content. As a result, during plant operation and increasingly during post-commercial safe shutdown and decommissioning, drums with radioactive waste are dried in drying facilities.

Three reportable smoldering fires having occurred in NPPs in Germany during the last years have to be mentioned in this context. The paper provides details on these the fire events, the success of different fire protection measures involved, and some potential for improvements.

FIRE EVENTS REPORTED

Event No. 1: Smoldering of Rubber Material During the Drying Process

A smoldering fire within a waste drum that was put into a plant internal drying facility was reported by a German NPP. The 200 l waste drum was filled mainly with hard rubber. The material was originally used as a coating material for a vessel and was crushed by removal, 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 (cf. Figure 1) 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 routed to the drying chamber for heat-up of the waste drum from the outside and for absorbing 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.

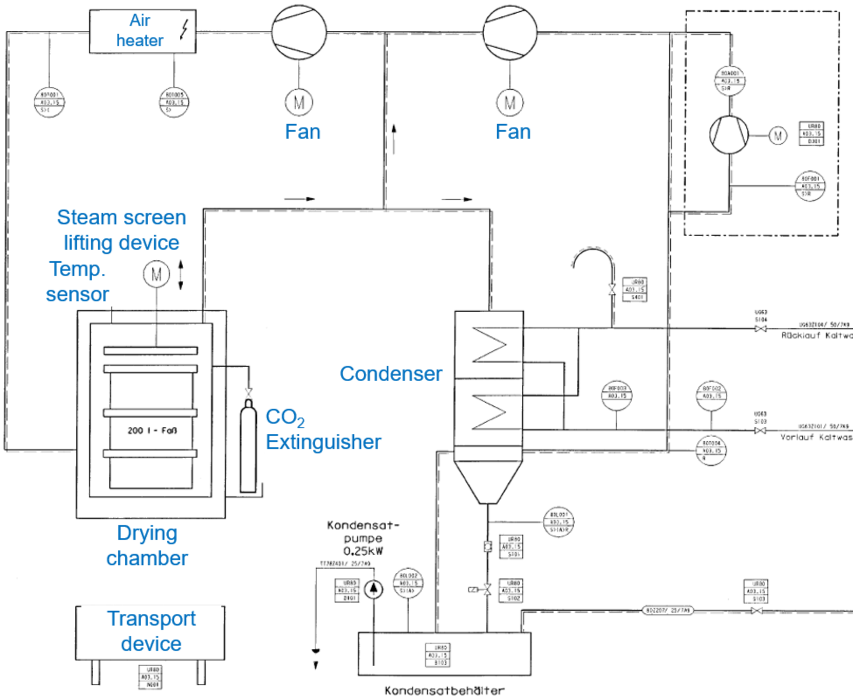

Figure 1 Schematic drawing of the drying facility

During the drying process the regular lid of the waste drum is changed by a steam screen (a metal plate with holes) that allows perfusion of evaporated water.

In the event, after the normal end of the drying process the heating was switched off and the steam screen was automatically lifted to increase the 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 again 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. The alarm was recognized after shift changeover in the morning. After investigation of the situation, plant personnel triggered the manual inerting of the drying chamber by CO_2. As the CO_2 system was not able to extinguish the smoldering fire in the waste drum, the drying facility was opened by the on-site plant fire brigade and the waste material was finally extinguished with water and foam as extinguishing agents.

The smoldering 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.

The investigations after the smoldering 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 this 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 found to be at 160 °C in experiments later carried out.

Event No. 2: Smoldering of Residues in a Waste Drum after Drying

In a technology center attached to a NPP, 16 drums of 200 l each with inner drums of 180 l were pre-dried in a reduced pressure drying facility. The drums were filled with mud from sumps of the technology center and should be squeezed after drying. Therefore, the drying process was "operational" in a sense that it did not need to meet the official schedule for processing waste for disposal. The mud was quite inhomogeneous and consisted of metal oxides (mostly Fe(II)), metals (Fe, etc.), some plastic materials, organic fluids, and other inert material. The inhomogeneity was within the drums and from drum to drum.

The drying process was performed under a pressure reduced to about 50 mbar at the vacuum pump. The reduced pressure significantly reduces the boiling temperature of water in the drums to about 35 °C. Oxygen partial pressure is low enough that ignition under this condition can practically be excluded. The drums were heated up by convection from the outside by about 130 °C furnace temperature for about 450 h.

Approximately 2 h after the proper drying process, the drums were placed for subsequent visual inspection within the building under a tent. After having taken off the drum lid and visual inspection of the contents, the drums were kept open for further processing. Two hours after having placed the drums under the tent, one smoke detector was triggered and 5 min later a second one, providing an alarm to the on-site plant fire brigade. The fire brigade found a smoldering fire in one of the drums and extinguished it successfully by mobile foam extinguishers.

In order to understand the ignition process a chemical analysis of the residues in one drum had been undertaken and showed that a redox reaction could be responsible for the ignition process. It is assumed that at first Fe(II) (parts of it such as ($Fe_3(PO_4)_2$ * 8 H_2O) was oxidized in an exothermic reaction. In the sample analyzed the Fe content was 4.6 %, 90 % of that as Fe(II)). The Fe(III) produced shows catalytic effects and allows for the autoxidation of organics. The exothermic reactions finally led to ignition. In an acidic solution of the sample one could smell H_2S, indicating the presence of sulphides. The sample showed a significant reaction after adding hydrogen peroxide.

Event No. 3: Incipient Fire after the Drying Process of Bag Filters

Two squeezing drums filled with polypropylene bag filters and filtrate from wastewater from the hot workshop and from building decontamination had been dried in a drying chamber. This method was established in the respective plant for several years. The temperature inside the chamber was about 140 °C under normal pressure and oxygen concentration. The room for the drying facility is classified as radioactive exclusion area.

The drying process was already finished for about 20 h and the chamber was cooled down to nearly room temperature, when one of the two squeezing drums was removed from the chamber (starting point 0 min). When the drum was driven out of the chamber, smoke became visible by a video camera installed in the location. Although the first squeezing drum could be driven back into the chamber, the door of the chamber could not be closed because the jamming protection signal of the door was triggered.

The CO concentration in the chamber increased and approximately 20 min after taking out the drums the CO detectors in the chamber provided a fire alarm because of a CO concentration of more than 300 ppm. By this means, the fixed CO_2 fire extinguishing system of the chamber was actuated. However, due to the open door the system was ineffective and therefore manually stopped in order to prevent CO_2 spreading all over the room.

The shift leader gave a fire alarm for the plant at about 25 min. When the members of the shift fire fighters were preparing for firefighting at the entrance of the controlled area the signal “all-clear” was given at approximately 35 min, since no flames but only smoke could be observed. However, two workers with qualification as fire fighters who were working in the controlled area entered the room of the drying chamber and confirmed ongoing smoke release. Because four members of the fire fighters from the shift personnel were still at the entrance of the controlled area, a troop of two to three firemen fitted themselves with protective equipment and self-contained breathing apparatus before entering the fire compartment. Visual inspection by the troop did not confirm that the door of the drying facility was blocked.

After staff responsible for radiation protection had confirmed that the activity inside the room was still on a normal level, the firemen took the first squeezing drum out of the chamber. By means of a visual check glowing material was observed together with smoke production. The fire was suppressed by portable CO_2 fire extinguishers. The squeezing drum was covered with the lid of the drum to keep the CO_2. The same procedure was done with the second squeezing drum, because also here glowing material was observed. The fire was reported to be successfully extinguished at approximately 130 min.

Measurements of the aerosol activity, the local dose rate, and wipe tests did not show any noticeable results.

The ignition process could not be clearly reproduced. It is considered as a combination of the temperatures elevated to 140 °C and material-specific circumstances that supported self- ignition.

ASSESSMENT OF THE EVENTS

From the aspect of nuclear safety, none of the fires was an open, fully developed one. Moreover, there was neither spreading of the fire out of the drums nor relevant releases of radioactivity out of the drums by smoke. Besides fire spreading and radioactive contamination, potential adverse effects are the production of smoldering gases which may cause ignitable atmospheres within a drying facility. Explosions or backdraft-like phenomena may not be excluded. Finally, fire propagation jeopardizing SSC (*s*tructures, *s*ystems, and *c*omponents) important to safety may occur.

In the following, the above characterized events are assessed concerning fire ignition, detection, and suppression.

Fire Ignition

Any piloted ignition can be ruled out, therefore, in principle the contributors to fire ignition in waste drums are the external heat supply from the drying facility and the exothermic reactions in the waste that lead to a self-heating and finally to ignition. The minimum temperature at which ignition is obtained under specified test conditions without any source of pilot ignition is defined as SIT [1]. A schematic illustration of the ignition process in given in Figure 2.

As stated in the definition of the SIT, the temperature is not a physical property of the material but depends on the test conditions. One important parameter is the volume of the container. The larger the container, the higher the SIT, because the heat losses are lower due the better ratio of surface area to volume. An example for this behavior is presented by Kimpel et al. [3] by the results of the SIT of brown coal. When the sample volume is displayed on a logarithmic scale there is a linear behavior to the reciprocal of the Kelvin value of the SIT (see Figure 3). The results of three different studies showed the same behavior, although there is some difference between the SIT in the different studies.

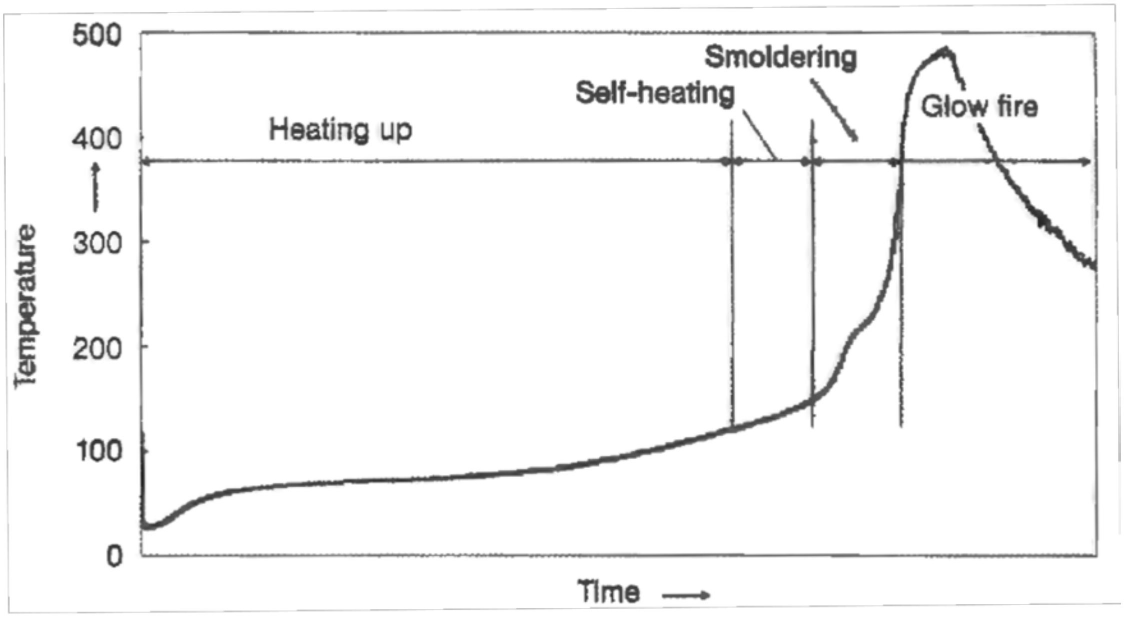

Figure 2 Scheme for the process of a fire by self-heating and smoldering of bulk material [2]

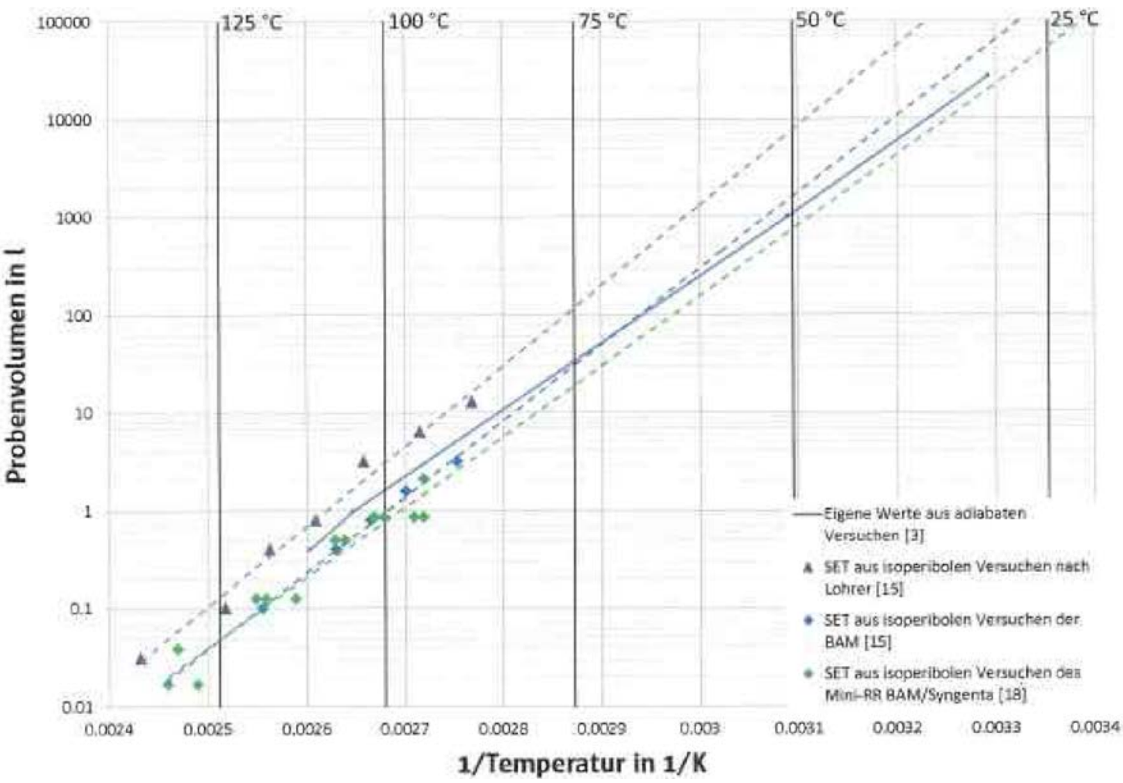

Figure 3 Relationship between SIT and volume of a test container for a sample of brown coal, from Kimpel, et al. [3]

Regarding the event no. 1, where relatively homogeneous organic material was ignited, this relationship plays a role to define a SIT from small scale test dryings. The SIT of the rubber material was additionally lowered because it was crushed when it was removed from the vessel. This increased the surface area of the material for reaction with oxygen and lowered the thermal conductivity that heat losses were reduced [2].

In the event no. 2, the composition of the waste material is quite unknown and very inhomogeneous. The influence of the Fe(III) as a catalyst was emphasized by the licensee. Such impurities within the material are generally reported [1], [2] to have a large influence on the ignition behavior.

Regarding events no. 1 and no. 2 one measure to prevent reoccurrence was to keep the drums as close as possible during the heating and cooling process to limit the oxygen content by leaving the steam screen on the drum or by closing it promptly with the lid. The relationship between oxygen concentration and SIT was exemplarily studied by Schmidt et al. [4] for different materials in small scale tests (cf. Figure 4). In these experiments the oxygen concentration was reduced by mixing with nitrogen before the mixture was let into the probe. For all materials the SIT increases with decreased oxygen concentration although the significance is different from material to material. Even at oxygen concentrations of about 2 % self-ignition did occur.

Since a steam screen only limits the air supply into a drum and the pathways within the material may be the more relevant bottleneck, it is not known if and how far it reduces the oxygen concentration at the inner parts of the drum. Therefore, a direct effect on the ignition probability might not occur, but at least the maximum heat release of an open fire can be limited by limited oxygen supply.

Moreover, the redox reactions that reportedly played an important role in the ignition process of event no. 2 would not be stopped by a reduced oxygen content.

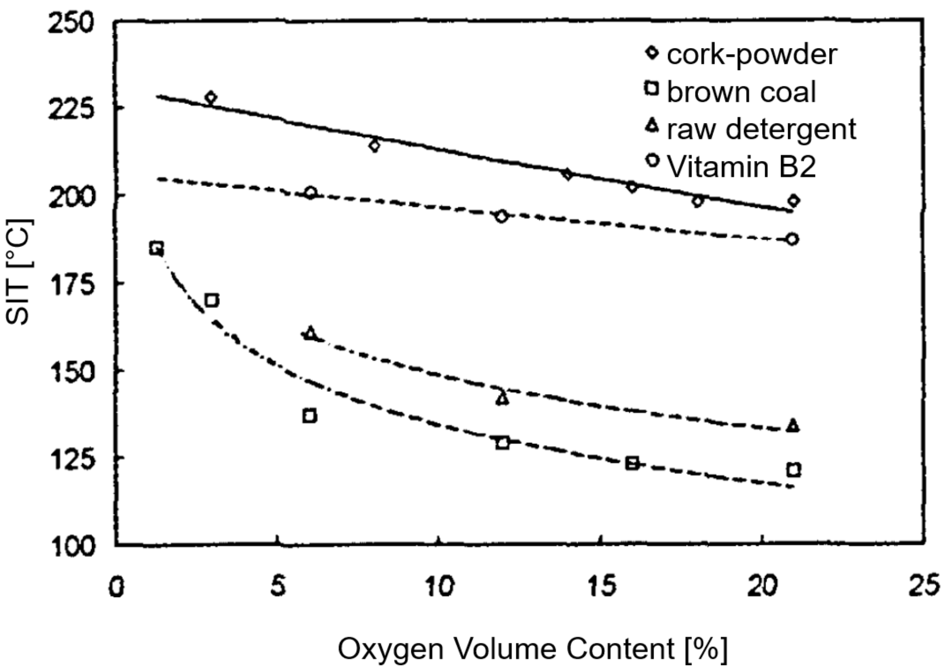

Figure 4 Exemplary dependence of the SIT from the oxygen concentration inside a sample (100 cm^3 sample volume), from [4]

Fire Detection

The three reported events were initially detected by a temperature sensor in the drying facility (event no. 1), by a standard optical smoke detector installed in a tent above the drums (event no. 2), and by a video camera followed by alarms from CO sensors in the drying chamber (event no. 3). This shows that such fires can be detected in principle by different means. Since no comparative data from different detectors for one event are available, it cannot be assessed which detector type would have detected a fire at the earliest or with the highest reliability with respect to false alarms.

Regarding the slow ignition process that is commonly assumed to start with self-heating and the early production of CO gas from decomposition and smoldering, temperature increase, and CO production are principally suited as the earliest indicators. As CO sensors have become much cheaper over the last decades, CO surveillance became state of the art for fire detection of bulky storages like silos or certain types of filters [5].

Unfortunately, in a waste-drying process, CO surveillance is conflicted with several difficulties. First, for organic materials there is a certain level of CO released from desorption processes which is no indicator for ignition. Therefore, some experience regarding the material and the drying process is needed before drawing any conclusions from CO levels. Secondly, metal fractions are very common in the residues, which oxidize under presence of water and produce hydrogen. As the frequently used electrochemical or semiconductor-based CO sensors are highly cross-sensitive to H_2 and due to the water vapour content, the required measurement technique is all but simple.

Finally, for a drum with no forced flow through the waste material, CO has to diffuse to the surface. The velocity of this process is of the same order of magnitude as the heat conduction, i.e. relatively slow. Additionally, in the cooling phase, H_2O produced by combustion will already condensate in the drum. Under these boundary conditions, the volume balance of O_2 diffusing into the drum and reacting to CO_2, H_2O, and partly CO is lower than 1. In other words, the drum is “breathing in”, delaying CO transport to a sensor [6].

For a drum that is changing temperature by external heating and latterly cooling, temperature measurements as an indicator for ignition have to be compared to the relative values for the upper limits and gradients with regard to the behavior over time, the physical processes in a drum and the exposure temperature of the drum. Air temperature measurements are commonly analyzed with regard to any risk of ignition. An early fire detection must be able to identify local hot spots in a drum that may lead to ignition. Since applying a mesh of temperature sensors in the material is rarely possible, surveillance for temperature anomalies by an infrared camera can be used at least for single-walled drums.

Fire Suppression

Fire suppression using carbon dioxide as extinguishing agent failed in the event no. 1 and was successful in the event no. 3; suppression by water or foam was successful in the events no. 1 and no. 2.

An obvious motivation to use gaseous extinguishing agents like CO_2 is that the drying process does not need to be repeated after successful application. The cooling capacity of gases is generally very small, therefore fighting smoldering fires and cooling hot spots is difficult and needs quite long holding times for the inert gases. Heat generation by redox reactions probably would not be effectively stopped by application of gases. Regarding the efficiency to suppress a fire, water-based agents are the first choice.

Depending on the composition of a specific waste, e.g. if the fraction of unoxidized metal particles within the waste is relevant, special extinguishing agents should be available.

CONCLUSIONS

The operational experience from NPP in Germany has shown three reportable fire events during or after drying processes of radioactive waste drums, none of these having more severe consequences. Since the drying facilities and processes as well as the waste material are different and inhomogeneous, there is no single strategy for further reducing the fire risk. The inhomogeneous material permitted for different ignition processes, which probably cannot be prevented completely, can change from drum to drum. Therefore, early fire detection and effective, riskless fire suppression are needed.

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REFERENCES

[1] Babrauskas, V.: Ignition Handbook, Fire Science Publishers / Society of Fire Protection Engineers (SFPE), Issaquah, WA, USA, 2014.

[2] Krause, U. (Ed.): Fires in Silos - Hazards, Prevention, and Fire Fighting, Wiley-VCH, Weinheim, Germany, 2009.

[3] Kimpel, S., et al.: Bestimmung der Selbstentzündungstemperatur von Schüttgütern – Einfach handhabbarer Weg für die Praxis, Technische Sicherheit (2), pp. 27-33, 2012 (in German).

[4] Schmidt, M., et al.: Selbstentzündung von Stäuben und Schüttgütern bei vermindertem Sauerstoffgehalt, Chemie Ingenieur Technik (74), pp. 1735-1737, 2002 (in German).

[5] Kelleter, J.: Detection of fires in silos, in: Proceedings of AUBE 2014 Conference, Duisburg, Germany, 2014.

[6] Kelleter, J.: personal communication, September 2019.

Operating Experience with Fires in Nuclear Installations – The OECD/NEA FIRE Database

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ABSTRACT

The international OECD/NEA FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) Database is one of the nuclear power plant (NPP) operational events Database Projects currently operated under the auspices of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA). This Database for the collection of detailed information from fire events at NPPs from fourteen NEA member countries is approaching the end of its fifth phase. The FIRE Database is considered mature enough for applications, with respect to not only deterministic safety assessment but also in fire probabilistic risk assessment.

The most recent version of this Database covers more than 500 well documented fire events during all operational phases of the plant life cycle from construction up to the longer duration safe shutdown, as well as a few events from NPP units under decommissioning. The number of recorded events increases continuously within each annual update. The Database structure enables analysts to carry out search queries for different aspects of fire events and allows investigations into even more complex fire scenarios. Various analyses can be systematically performed in an automated manner, from generating different samples up to a more or less complete statistical analysis.

This paper presents a brief overview of the application possibilities of the OECD/NEA FIRE Database for supporting NPP operators as well as regulators in assessing fire safety issues.

INTRODUCTION

The international Database OECD/NEA FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) is one of the nuclear power plant (NPP) operational events Database Projects currently operated under the auspices of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA). In this Database detailed information on fire events at NPPs from actually fourteen NEA member countries – Belgium, Canada, Czech Republic, Finland, France, Germany, Japan, Korea, the Netherlands, Spain, Sweden, Switzerland, the United Kingdom, and the United States of America (USA) – is collected.

The need for the FIRE Database emerged in the late 1990s when it became evident that the only international recording of fire events by the joint IAEA (*I*nternational *A*tomic *E*nergy *A*gency)/NEA International Reporting System for Operating Experience (IRS) was not suitable for specific analysis and use in risk assessments. The purpose of the FIRE Database Project is to provide a platform for member countries to collaborate and exchange fire data and thereby to enhance the knowledge of fire phenomena and in turn improve the quality of risk assessments that require fire related data and knowledge.

The Project, which was officially launched in 2001, is close to the end of its fifth phase (2016 – 2019) and is considered mature enough for applications, not only with respect to deterministic safety assessment but also in fire probabilistic risk assessment.

The objectives of the FIRE Database Project as defined according to the Terms and Conditions are the following:

- Collect fire event experience (by international exchange) in an appropriate format in a quality-assured and consistent database;
- Collect and analyze fire events over the long term to better understand such events and their causes, and to encourage their prevention;
- Generate qualitative insights into the apparent and root causes of fire events in order to derive approaches or mechanisms for their prevention and to mitigate their consequences;
- Establish a mechanism for efficient operational feedback on fire event experience including the development of policies of prevention, such as indicators for risk-informed and performance-based inspections;
- Record characteristics of fire events in order to facilitate fire risk analysis, including quantification of fire frequencies.

DATABASE HISTORY AND STRUCTURE

Discussions about a specific task concerning fire risk assessment started within the OECD/NEA several years before the FIRE joint Project was established. In 1989, the members of the Principal Working Group on Risk Assessment (PWG5) under the NEA Committee on the Safety of Nuclear Installations (CSNI) discussed several challenging aspects of Level 1 PSA and decided to write a state-of-the-art report (SOAR) on Level 1 PSA methodology [1]. Since the topic was very broad, it was divided into five sub-tasks. The sub-task “external events” focused (amongst others) on fire risk analysis and it was concluded that much work was still needed for improving fire simulation codes. In order to follow up on this recommendation and to get a better overview of the available methods and their state of maturity, the PWG5 decided in 1996 to develop a SOAR on fire risk analysis, fire simulation, fire spreading and impact [2]. As basis of this endeavor, an international workshop on fire risk assessment was held in 1999 in Helsinki, Finland [3]. In addition, a questionnaire about the application of fire risk PSA and fire simulation codes including the assessment of the impact of fire and smoke spreading was issued. Another part of the questionnaire considered a review and assessment of available data. The group identified the lack of fire analysis data as a significant challenge for the fire risk assessment and recommended the establishment of an international database which would cover various quality assured data including data from real fire events as well as the initiation and reliability of fire suppression actions.

In 2000, the OECD/NEA Working Group on Risk Assessment (WGRISK), the successor of PWG5, submitted a proposal for establishing a project for fire risk analysis data collection on a cost-sharing basis to the CSNI, which endorsed the proposal. The OECD FIRE Project was born. This activity was originally initiated by Sweden and Finland having already developed the Nordic Fire Database (NFDB) during 1999 – 2000 covering experience from over 230 reactors years. It was proposed to take the NFDB as basis for an international fire data exchange program. In 2002, the kick-off meeting of the Project was held where, next to technical aspects, administrative issues were discussed. A few months later, in January 2003, the first meeting of the FIRE National Coordinators (NCs) was held and the first phase of the Project (2003 – 2005) started with nine member countries: Czech Republic, France, Finland, Germany, Japan, Spain, Sweden, Switzerland, and the USA. During the second Project phase from 2006 to 2009 Canada, the Netherlands and Korea joined the Project. Subsequent phases of the Project followed from 2010 to 2013, 2014 to 2015. The Project is currently in the fifth Project phase (2016 to 2019). During this phase, Belgium and the United Kingdom have joined the FIRE Project.

The FIRE Database Project is one of the actually three Database Projects of the OECD/NEA. For organisation and structure of the Project, the members agreed, as general rule, that the data provided by the individual countries remain the property of their original owner, but the Project members would have the right to access the data and to conduct analytical activities. Each member country nominates a NC as single point of contact for the respective country approving the data coming from this country and being the only person with unrestricted access to the Database.

In order to recognize the support from the licensees providing the data and to increase the value of the Database for the member country, an anonymized version of the FIRE Database is also produced and made available for users in member countries. In this encrypted version, the NPP from which the data originated, is encoded but it is still possible to use the data, e.g. for general statistics. Since high quality data is essential for usability and comparability in the Database, specific guidelines and boundary conditions exist for the exact coding of the data. The compliance with the corresponding Coding Guidelines (CG), which are continuously improved by the members, is the responsibility of the NC, who is supported in this aspect also by the Operating Agent (OA) of the FIRE Project. Other responsibilities of the OA are analytical activities as well as the maintenance and constant improvement of the usability of the Database.

Currently two parallel systems are used in the FIRE Database Project. A web-interface is provided through the OECD/NEA for entering of the data by the member countries. From this web-interface the OA creates annual versions as a MS ACCESS® database with advanced search functions and tools for statistical analysis.

FIRE DATABASE APPLICATIONS

One of the aims of data collection is to generate generic fire occurrence frequencies for different reactor types and plant operational states (POS) for PSA use. Other applications with significance for regulatory assessments include but are not limited to in-depth investigations of fires and event combinations of fires and other events or, the analysis of apparent causes as well as root causes of fire events. Additional objectives of the next, sixth Project phase (from 2020 to 2022) will be to investigate fire records having some multi-unit or multi-source effects in order to support PSA on the site level and the investigation and analysis of parameters for exploring fire suppression success by both extinguishing systems and/or manual firefighting. The following paragraphs provide insights from the ongoing activities.

Fire Occurrence Frequencies

For the NPPs in FIRE Database member countries the observation times are systematically collected and updated on a yearly basis. The information collected for each NPP unit in the member countries this context is i.e. as follows according to the recent version of the FIRE Database Coding Guideline (see [4]):

- Plant unit name and identification number;
- Reactor type;
- Whether the reactor is part of a multi-unit and/or multi-source NPP site;
- Start of commercial operation: date of the start of commercial operation of the respective reactor unit;
- End of commercial operation / start of post-commercial safe shutdown: date for the end of commercial operation of the respective reactor unit, being the same date as for the start of post-commercial safe shutdown phase;
- End of post-commercial safe shutdown / start of decommissioning: date for the end of the post-commercial safe shutdown of the respective reactor unit, being the same date as for the start of decommissioning[3];
- End of decommissioning: date for the end of decommissioning, representing also the end of observation;
- Observation start: either the date of start of commercial operation, or the date of starting reporting events from the respective reactor unit to the FIRE Database, whatever comes last;
- Observation end: for the respective reactor unit either the end date of post-commercial safe shutdown, or the end date of decommissioning, depending on whether events are reported also during decommissioning, or the end date of the actual observation period (e.g., December 31, 2018).

From this information the observation periods for each reactor unit can be subdivided into the following three different plant lifetime periods for statistical use:

  - power operation,
  - low power and shutdown (including post-commercial permanent safe shutdown), and
  - decommissioning.

Based on the observation times generated using the aforementioned information, the FIRE Database user can generate fire occurrence frequencies for different countries, different NPP sites, different reactor types, and for the above plant operational states. The fire occurrence frequencies can be used at least as generic prior information within Fire Probabilistic Safety Assessment (PSA), if the number of fire events in the plant or reactor unit under investigation is statistically not sufficient. Depending on the Fire PSA approaches in the FIRE member states, room (or fire compartment) specific fire frequencies for each building relevant for Fire PSA or component related fire frequencies are needed. Both types of fire frequencies can in principle be derived from the FIRE Database.

Room-Based Fire Frequencies

The FIRE Project members have developed the following Table 1 for categorizing different room types of fire occurrence sorted by different buildings. In these tables, the numbers of rooms of a given type per building are completed according to par. 3.2.3 and 3.2.4 of the Coding Guideline in its most recent version (see [4]) by the National Coordi-

[3] *Note*: In some countries this is the date when the official permit is given to start decommissioning.

nators (NCs) of the fourteen member countries for the different NPP units for which they provide fire event data.

The Database incudes a specific feature to calculate room-based fire occurrence frequency values either for an individual reactor type or for various types of reactors. In this context, the analyst can also decide, if only specific types of fire (e.g., for electrical fires, etc.) shall be considered. Moreover, such frequencies can be provided for the different POS as well.

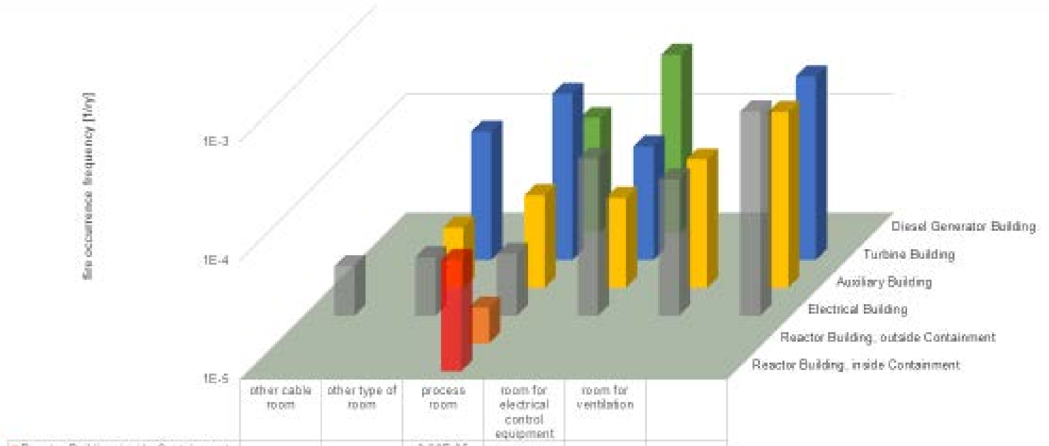
Table 1 Room types per buildings table to be filled for each NPP unit for deriving room specific fire frequencies from the FIRE Database

OECD FIRE Database – No. of Rooms (Compartments) per Building Plant ID: Date:

Room Type	Cable rooms		Room for electrical control equipment incl. MCR	Switch-gear room	Battery rooms	Room for ventilation	Room for off-gas equip-ment	Process room	Staircase / corridor	Office	Workshop	Storage rooms			Diesel gene-rator room	Trans-former room / bunker	H_2 cylinder bunker	Other room	Total
Building	Cable spreading room	Other cable room										for nuclear waste	for other waste	for com-bustibles					
Reactor Building																			
Contain-ment																			
Outside Contain-ment																			
Electrical Building																			
Auxiliary Building																			
Turbine Building																			
Diesel Generator Building																			
Intake Building																			
Spent Fuel Building																			
Independent Emergency Building																			
Workshop Building																			
Other building																			
Total																			

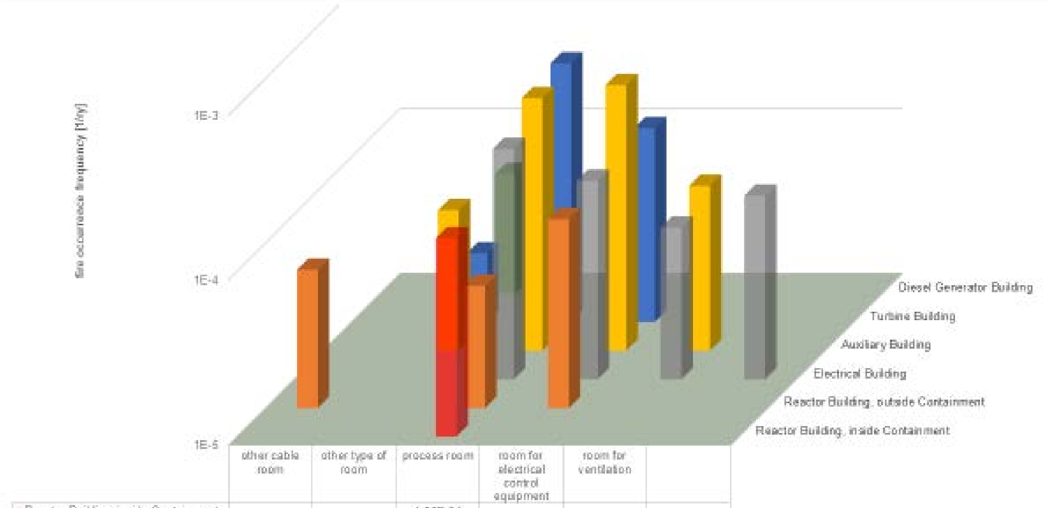
Note: Definitions of rooms (compartments)and building , see OECD FIRE Coding Guideline 2017:02, par. 3.2.3 and 3.2.4

Examples are provided in the following Figures 1 to 6 for pressurized water reactors (PWRs), boiling water reactors (BWRs), and pressurized heavy-water reactors (PHWRs) for power operation (FP) as well as for low power and shutdown (LPSD) states.



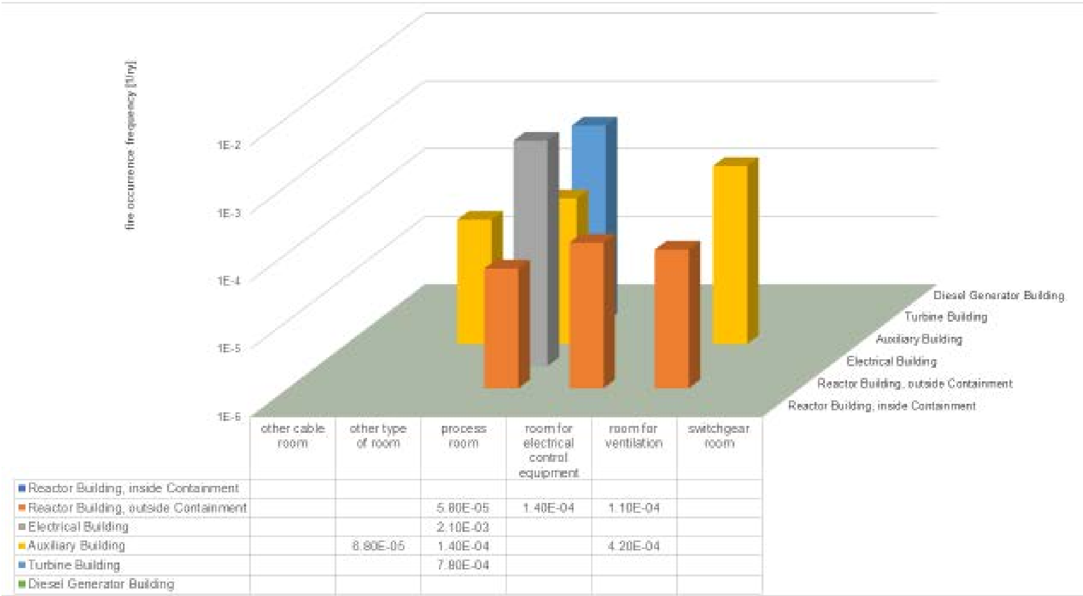
	other cable room	other type of room	process room	room for electrical control equipment	room for ventilation	
Reactor Building, inside Containment			8.60E-05			
Reactor Building, outside Containment			2.00E-05			
Electrical Building	2.60E-05	3.10E-05	3.30E-05	2.10E-04	1.40E-04	8.40E-04
Auxiliary Building		3.20E-05	6.00E-05	5.70E-05	1.20E-04	3.00E-04
Turbine Building		1.20E-04	2.50E-04	9.00E-05		3.50E-04
Diesel Generator Building			9.10E-05	3.10E-04		

Figure 1 Room type specific fire occurrence frequencies for PWR during power operation, from [4]



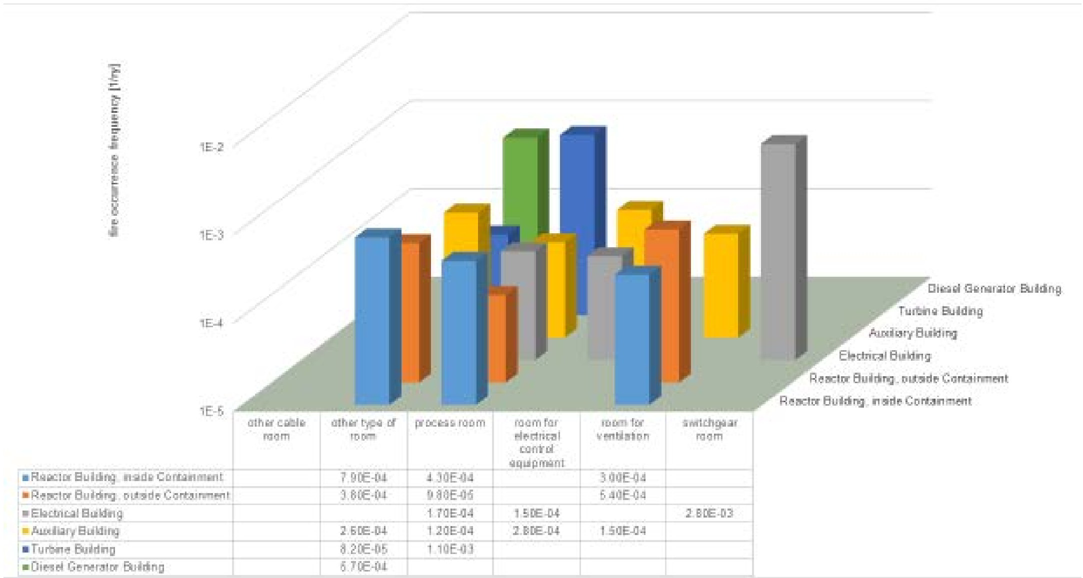
	other cable room	other type of room	process room	room for electrical control equipment	room for ventilation	
Reactor Building, inside Containment			1.60E-04			
Reactor Building, outside Containment	5.90E-05		5.50E-05	1.40E-04		
Electrical Building			2.50E-04	1.60E-04	8.30E-05	1.30E-04
Auxiliary Building		7.10E-05	3.40E-04	9.40E-04	1.00E-04	
Turbine Building		2.60E-05	3.70E-04	1.50E-04		
Diesel Generator Building		5.30E-05				

Figure 2 Room type specific fire occurrence frequencies for BWR during power operation, from [4]



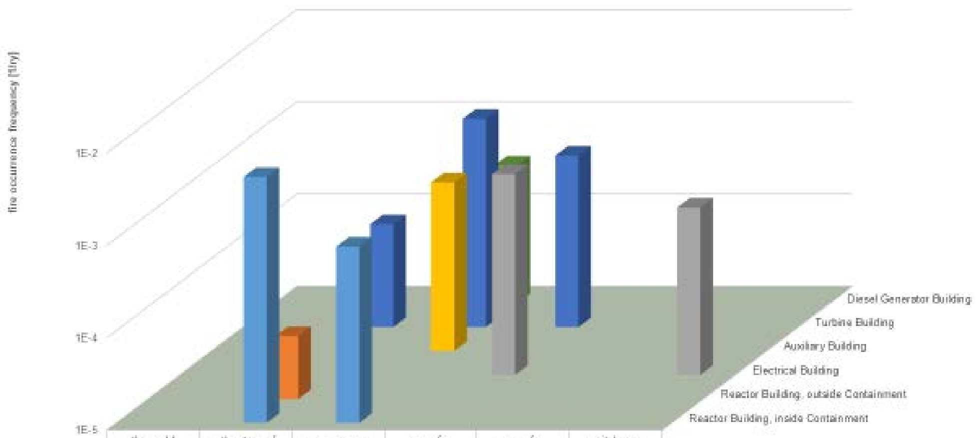
	other cable room	other type of room	process room	room for electrical control equipment	room for ventilation	switchgear room
Reactor Building, inside Containment						
Reactor Building, outside Containment			5.80E-05	1.40E-04	1.10E-04	
Electrical Building			2.10E-03			
Auxiliary Building		8.80E-05	1.40E-04		4.20E-04	
Turbine Building			7.80E-04			
Diesel Generator Building						

Figure 3 Room type specific fire occurrence frequencies for PHWR during power operation, from [4]



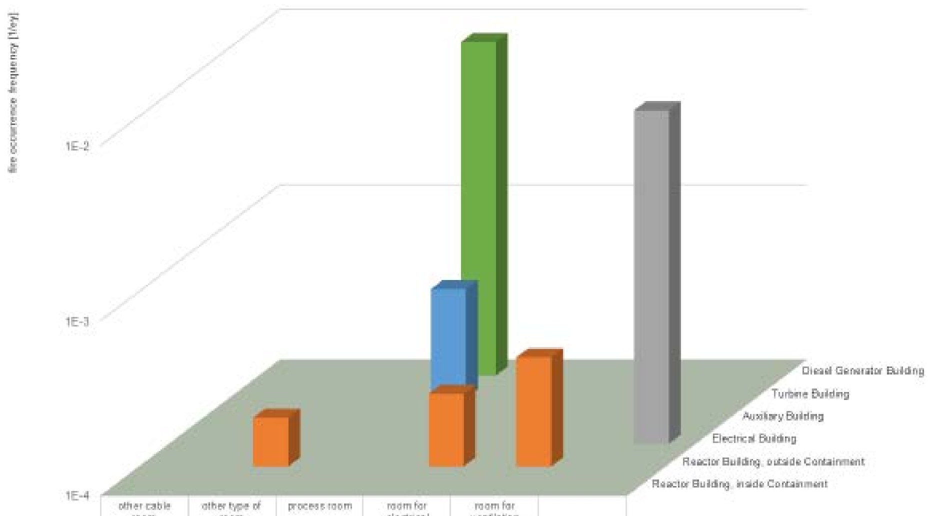
	other cable room	other type of room	process room	room for electrical control equipment	room for ventilation	switchgear room
Reactor Building, inside Containment		7.90E-04	4.30E-04		3.00E-04	
Reactor Building, outside Containment		3.80E-04	9.80E-05		5.40E-04	
Electrical Building			1.70E-04	1.50E-04		2.80E-03
Auxiliary Building		2.60E-04	1.20E-04	2.80E-04	1.50E-04	
Turbine Building		8.20E-05	1.10E-03			
Diesel Generator Building		6.70E-04				

Figure 4 Room type specific fire occurrence frequencies for PWR during low power and shutdown, from [4]



	other cable room	other type of room	process room	room for electrical control equipment	room for ventilation	switchgear room
Reactor Building, inside Containment		4.60E-03	8.10E-04			
Reactor Building, outside Containment		4.80E-05				
Electrical Building				1.50E-03		6.50E-04
Auxiliary Building			6.70E-04			
Turbine Building		1.30E-04	1.80E-03	7.20E-04		
Diesel Generator Building			3.10E-04			

Figure 5 Room type specific fire occurrence frequencies for BWR during low power and shutdown, from [4]



	other cable room	other type of room	process room	room for electrical control equipment	room for ventilation	
Reactor Building, inside Containment						
Reactor Building, outside Containment		1.90E-04		2.60E-04	4.20E-04	
Electrical Building						8.00E-03
Auxiliary Building						
Turbine Building			4.20E-04			
Diesel Generator Building			8.00E-03			

Figure 6 Room type specific fire occurrence frequencies for PHWR during low power and shutdown, from [4]

The figures indicate that during power operation fires in the electrical building are the dominant contributor with fire frequencies varying between E-05 and E-04 per reactor year (ry) for rooms in BWR and PWR type plants and in the order of E-03 /ry for PHWR type reactors. The main contributions come from electrical and process rooms. For reactors during low power and shutdown phases, the fire frequencies in the electrical building are significantly higher reaching the order of E-03 /ry for BWR,PWR and PHWR type NPPs in some rooms, with switchgear rooms and rooms for other electrical control equipment providing the highest room specific contributions to fire risk. The intended in-depth investigations on the apparent and root causes (see below) may help to identify potential for improvements and better prevention of these types of fire.

Component Based Fire Frequencies

During Phase 4 of the FIRE Database Project an effort has started for gaining not only fire frequencies specifically for different types of rooms but also for the different types of components important for Fire PSA. Since the collection of this information is time consuming and not easy for all components of fire ignition for which codes are provided in par. 3.2.5 of the Coding Guideline (see [4]), data collection is still ongoing. The template shown in Table 2 provides the list of components for which numbers are to be collected for each reactor unit reporting fire events to the Database.

Table 2 List for collecting number of components of fire ignition for different component types per reactor unit according to FIRE Coding Guideline

Component Code			**Number of Components**
Battery			
Compressor			
Diesel generator			
Electrical cabinet:			
	High or medium voltage (non-HEAF, ≥ 1 kV)		
	High or medium voltage (HEAF, ≥ 1 kV)		
	Low voltage (non-HEAF, < 1 kV)		
	Low voltage (HEAF, < 1 kV)		
Hydrogen containing vessel			
Pump:			
	Electrically driven or turbine driven		
	Reactor coolant pump (RCP, for PWR)		
	Main feedwater pump		
Transformer:			
	High voltage (voltage ≥ 50 kV)		
	Medium or low voltage (voltage level < 50 kV):		
		Dry	
		Oil filled	
Turbine generator			

The collection of data on cables is particularly challenging ; therefore to date, cable data have not been collected systematically. However, at least one country has provided cable data, which was collected by the licensees as input information for their Fire PSAs (the length of cables was derived from NPP cable databases or that was counted approximately in situ, room by room).

Based on the component numbers and number of fire events occurred at the component types listed in Table 2, component type specific fire ignition frequencies have been derived, which can be compared to e.g. fire ignition frequency bins from Fire PSA in member countries. Figure 7 shows such fire ignition frequencies for reactor units during power operation, and Figure 8 for those under low power and shutdown conditions.

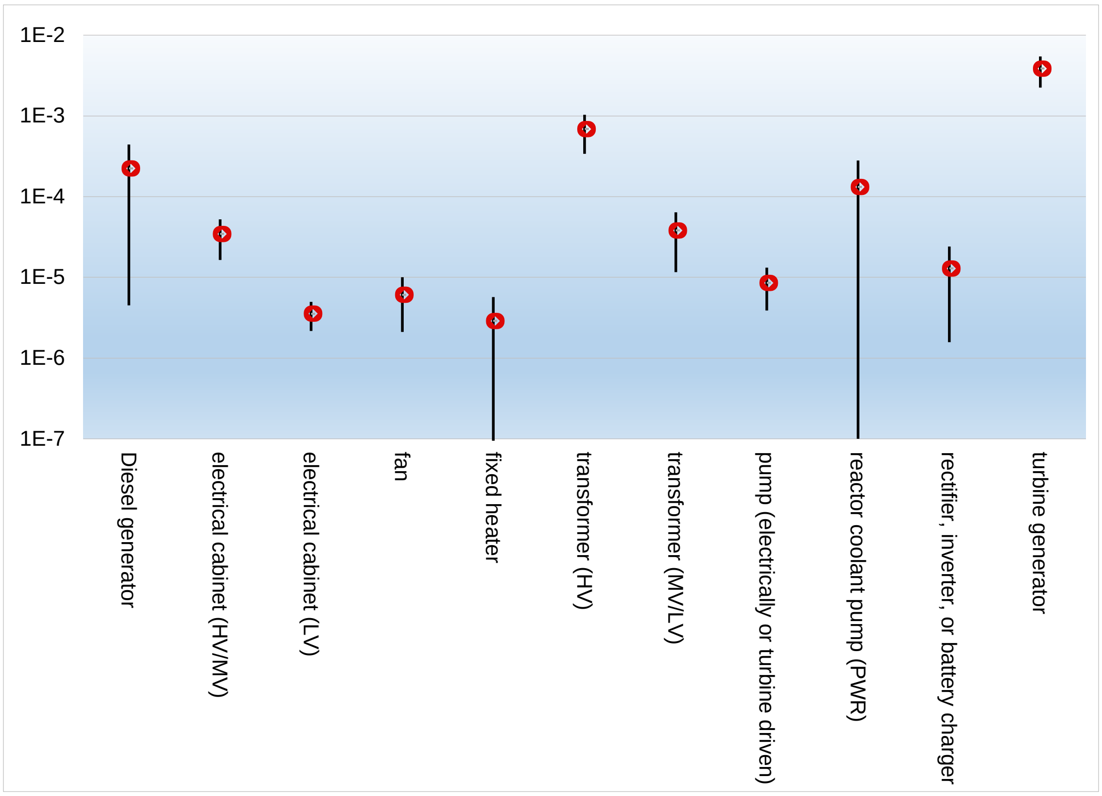


Figure 7 Component type specific fire occurrence frequencies for reactor units in power operation, from [4]

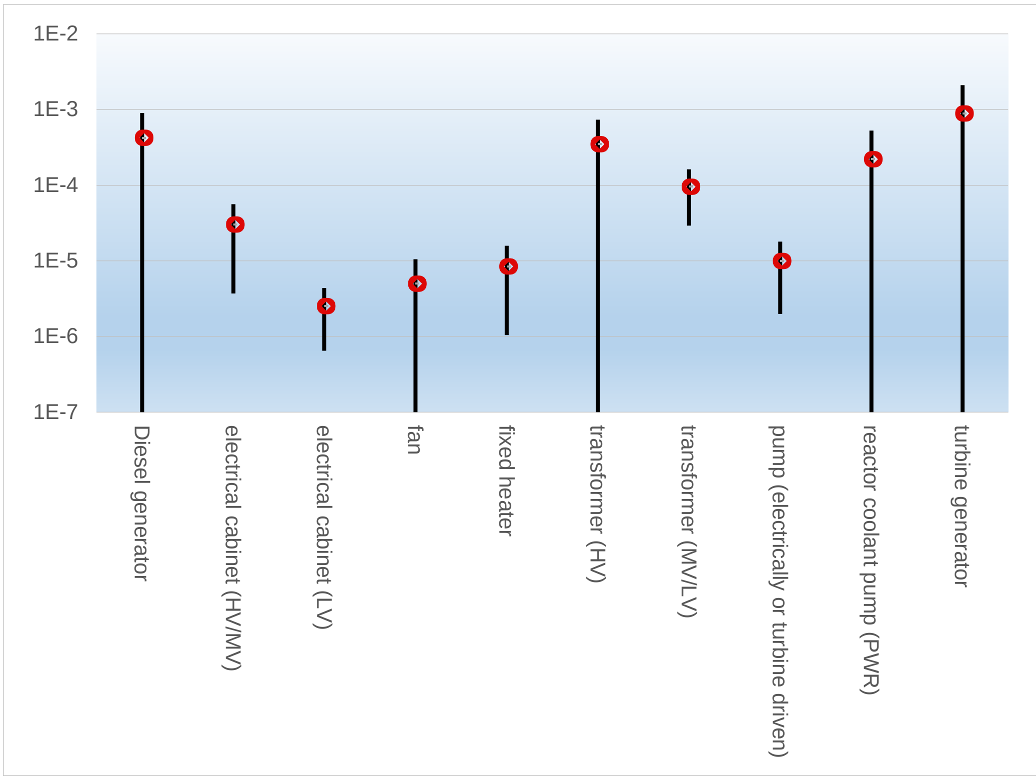


Figure 8 Component type specific fire occurrence frequencies for reactor units in low power and shutdown, from [4]

In principle, various fire frequency data can be generated. For statistically meaningful results it is important that according to the different reporting criteria and threshold in FIRE member countries, not all events can be binned together. However, different fire ignition bins – room specific or component specific ones – can at least be used as a priori information for statistical assessments, in particular if the individual NPP unit specific operating experience is insufficient for statistical analyses.

Indications Regarding Causes of Fire Events in the FIRE Database

An activity has been initiated to learn more on the apparent causes of fire in order to improve fire safety in NPPs in member states and to gain more insights on fire event root causes.

In a first step, a rough trend analysis has been performed with respect to the most common apparent causes of fire events in the FIRE Database. For a statistically meaningful analysis only events from those countries reporting all fire events (without any reporting thresholds or exclusion criteria) have been analyzed. These 290 events from PWRs, BWRs, and PHWRs represent a total of 2763 reactor years of operating experience. It is important to note that only events with known POS at the time of fire occurrence were accounted for.

The following major causes have been identified and analyzed, see Table 3 below. However, only those 169 of the 290 events for which the cause could be identified without doubt have been considered. The results of this first trend analysis provide further suggestions for detailed analyses.

Table 3 Major apparent causes of the fire events as observed from the FIRE Database [4]

Type of Cause	Number of Events							
Reactor Type	**Total**		**PWR**		**BWR**		**PHWR**	
POS	**FP**	**LPSD**	**FP**	**LPSD**	**FP**	**LPSD**	**FP**	**LPSD**
Observation Time [ry]	2274.80	488.10	1491.71	300.42	311.06	63.20	472.00	125.05
Hot short	10	4	6	3	4	0	0	1
Short to ground	5	0	5	0	0	0	0	0
HEAF	13	6	8	4	3	1	2	1
Mechanical overheating	16	8	7	4	4	3	5	1
Electrical overheating	18	11	10	7	7	4	1	0
H_2 fire	3	0	1	0	1	0	1	0
Oil/lubricant fire	14	9	4	4	10	3	0	2
Hot work	30	22	18	5	6	10	6	7

Interesting apparent cause categories are those with non-negligible contributions from human error, because understanding their causes promises a high potential for preventing future events. Such categories are:

– Electrical overheating (34 % human error): A large part of this is caused by the operation of equipment without assuring the availability of required cooling devices, operating equipment over longer time than it is designed for and connecting the wrong power sources to consumers. The occurrence rate of such events could be reduced by improving operating procedures and training.

– Mechanical overheating (25 % human error): In most cases the overheating is caused by mechanical friction between moving parts or by bearings damaged by insufficient lubrication. Improvements could be achieved by enhanced maintenance procedures.

– HEAF (26 % human error): The faults are mainly caused by equipment problems due to winding shorts, cable shorts and shorts in coils of switching equipment. Components mainly affected are high as well as medium and low voltage transformers, bus bars or other switching equipment in electrical cabinets, breakers, and cables in cable ducts as well as on cables routes. Improvement may be achieved by enhanced quality controls, maintenance procedures and training.

– Hot work (100 % human error): Mainly affected are components (other than cables) ignited by hot work, transient material and filters. Strengthening of rules for maintenance and repair work as well as for dealing with transient material could result in reduced occurrence rates.

A first rough analysis of the events regarding their root causes has been carried out. While apparent causes may also vary depending on the reactor type and design, root causes can only be subdivided into technical cause (named “equipment”), typical human erroneous actions (“human”) and inappropriate procedures or not following the procedures (“procedures”). These may be different during power operation and low power and shutdown states. The analysis provided the results listed in Table 4. In this context, it has to be noted that because of the still missing observation times for construction and decommissioning phases only the in total 500 events (6871 reactor years for power operation and 2043 for low power and shutdown phases including post-commercial permanent safety shutdown), for which observation times are available, have been accounted for.

Table 4 Root causes of fire events in the recent version 2017:02 of the FIRE Database [4] for different POS

Type of Root Cause	Number of events, where the root cause type occurred, and Corresponding Frequencies [1/ry]	
	Power Operation	**Low Power and Shutdown**
Equipment	234 / 3.4 E-02	85 / 4.2 E-02
Human	89 / 1.3 E-02	60 / 2.9 E-02
Procedures	36 / 5.2 E-03	17 / 8.3 E-03

Note: This information is based on the 500 fire events which neither took place during construction nor during decommissioning.

The analysis of event causes will be continued, focusing on events with more than one root cause. Their root causes will be analyzed with respect to their type and, in the event of multiple occurrences of root causes of one type, their sequencing will be examined in detail.

HEAF Fire Events and Event Combinations of Fires and Other Hazards

The operating experience from FIRE member countries has revealed (see also [6] and [7]) the importance of high energy arcing faults (HEAF) as an important phenomenon for causing fires, either at the component where the HEAF occurred itself or as consequential event. The most recent version of the Database [4] covers in total 62 HEAF fire events, 44 of these having occurred during power operation representing a fire frequency of 6.6 E-03 /ry and 18 having been observed during low power and shutdown states. Compared to power operation, the latter represent a factor of 1.4 higher fire frequency of 9.2 E-03 /ry. This indicates the non-negligible contribution to the overall fire risk by HEAF induced fires, even if a NPP unit has stopped commercial operation. The overall HEAF fire frequency of 7 E-03 /ry therefore underpins the need for further investigations into this fire initiator.

In total, 51 event combinations of fires and other internal and external hazards including event chains of three events (representing approx. 10 % of the entire event records) have been observed until the end of 2017 (cf. Figure 9), 34 of these were combinations involving HEAF events. This again underpins the significance of HEAF with respect to plant internal fires.

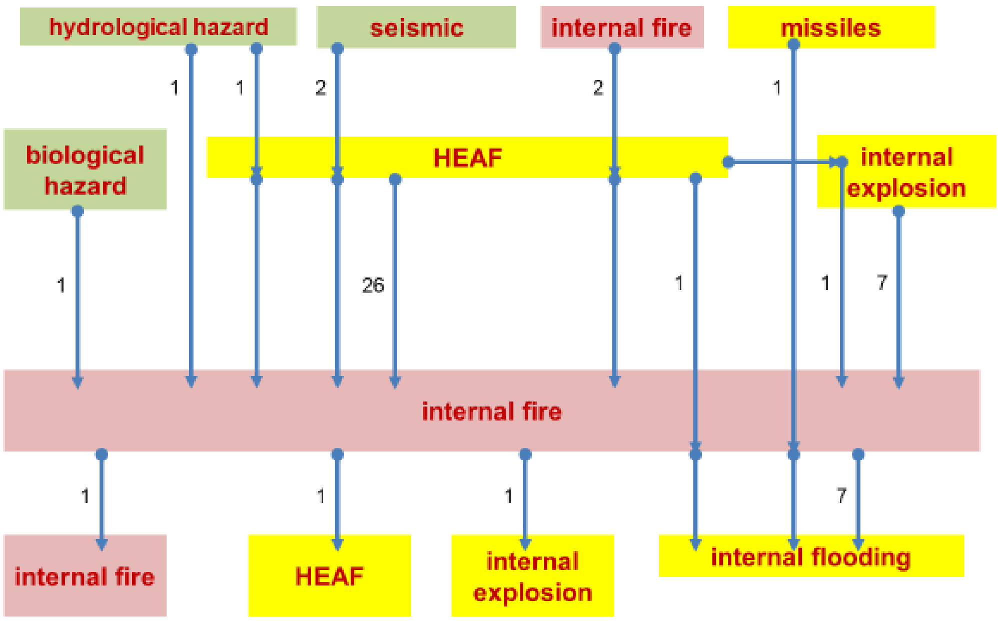


Figure 9 Number of event combinations of fire and other hazards from [4]

Moreover, the 62 HEAF induced fire events observed, thereof 28 individual events and 34 event combinations, result in an occurrence frequency of 7 E-03 /ry. For NPPs during power operation the occurrence frequency is 6.6 E-03 /ry (based on 44 events), for plants in low power and shutdown states (18 events) the frequency is even higher (9.2 E-03 /ry). For the 34 event combinations of HEAF and fire (cf. Figure 10) the occurrence frequencies are in the order of E-04 /ry to E-03 /ry. These observations have supported further HEAF related activities to be carried out in an international framework (details see [7], [8] and [9]).

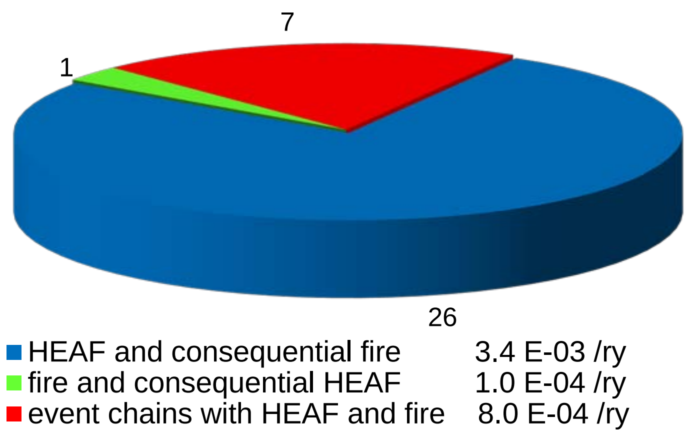


Figure 10 Event combinations of HEAF fire events, from [4]

Furthermore, one of the lessons learnt from the analyses of event combinations so far is that these should have been considered in the plant design against hazards, in particular fires, because of their risk significance. Updates of national as well as international regulations meanwhile take these insights into account (see, e.g., IAEA Special Safety Guide SSG-64 “Protection against Internal Hazards in the Design of Nuclear Power Plants” [11].

CONCLUSIONS AND OUTLOOK

At the time being, a variety of analytical activities based on the FIRE Database are ongoing, such as the above-mentioned comparison of fire ignition frequency bins from the member countries´ generic operating experience with plant specific frequency bins from Fire PSA or the investigations on fire suppression success by both extinguishing systems and/or manual firefighting.

Proposals for further extending the data collection in the sixth Project phase planned for a period of three year from 2020 to 2022 have been provided and will be discussed by the OECD/NEA FIRE members. The proposals comprise cross-cutting topics between OECD/NEA Databases such as FIRE and ICDE (*I*nternational *C*ommon Cause *F*ailure *D*ata *E*xchange) on common cause failures (CCFs) of active fire protection features.

Another topic is to collect fire event data not only from operating power reactors but also from reactor units under decommissioning as well as from research and demonstration reactors. A thus extended scope may attract new countries to join the FIRE Database Project.

ACKNOWLEDGEMENTS

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REFERENCES

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): State of the Art of Level-1 PSA Methodology, NEA/CSNI/R(92)18, Paris, France, January 1993, https://www.oecd-nea.org/nsd/docs/1992/csni-r1992-18.pdf.

[2] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics, NEA/CSNI/R(99)27, Paris, France, March 2000, https://www.oecd-nea.org/documents/1999/sin/csni-r99-27.pdf.

[3] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Proceedings from International Workshop on Fire Risk Assessment, NEA/CSNI/R(99)26, Paris, France, June 2000, https://www.oecd-nea.org/documents/1999/sin/csni-r99-26.pdf.

[4] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD

FIRE Database Version 2017:02, Paris, France, May 2019 (limited to FIRE Database Project member countries only).

[5] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Project - Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R(2013)6, Paris, France, June 2013, http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf.

[6] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): A Review of Current Calculation Methods Used to Predict Damage from High Energy Arcing Fault (HEAF) Events, NEA/CSNI/R(2015)10, Paris, France, 2015, https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-10.pdf.

[7] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Experimental Results from the International High Energy Arcing Fault (HEAF) Research Program Testing Phase 2014 to 2016, NEA/CSNI/R(2017)7, Paris, France, 2017, http://www.oecd-nea.org/documents/2016/sin/csni-r2017-7.pdf.

[8] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research: An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arc Faults (HEAFs), NUREG-2218, Washington, DC, USA, 2018, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2218/.

[9] Melly, N., K. Hamburger, and M. H. Salley: OECD/NEA High Energy Arcing Faults (HEAF) Research – Second Phase of Testing, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

[10] International Atomic Energy Agency (IAEA): Protection against Internal Hazards in the Design of Nuclear Power Plants”, IAEA Safety Standards Series No. SSG-64, Vienna, Austria, in publication, 2020.

Behavior of Fire Detection and Control Systems in Case of Smoke Spreading Through Fire Barriers – Consequences and Opportunities

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ABSTRACT

The fire detection and control system (FDCS) function is the timely detection of initial state (incipient) fires, to report fire locations and to control certain fire protection functions such as closing of fire dampers.

Closing of fire dampers and therefore the confinement of fires is one of the fundamental defence lines for both non-nuclear specific and nuclear specific safety demands. If, however, the FDCS closes fire dampers of ventilation duct branches serving areas containing heat sensitive equipment (I&C electrical devices), dysfunctions may occur after a grace period due to loosing proper cooling by the ventilation. Indeed, in case of fire, the fire is confined in the respective fire compartment, but even in this case it might affect the ventilation of areas remotely from the fire.

This opens a safety issue with regards to the separation concept of redundant parts of the safety systems. If smoke spreads to an adjacent division the FDCS may close fire dampers and will interrupt cooling by ventilation as a consequence.

Doorways in the separating barriers, independent if the fire door is closed or opened for passages, constitute per se a potential path of smoke spreading. The doorways however cannot entirely be prevented as being necessary for personnel circulation and access.

The guiding question is how consequential loss of cooling can be prevented under consideration of timely closing of fire dampers and the presence of doorways for personnel circulation and access.

INTRODUCTION

Fire is a phenomenon jeopardizing humans life safety and can cause large damages to the plant (investment and radiologically). One of the main fire protection design solutions is to confine the fire inside dedicated areas.

Therefore, it is important to ensure the fire compartments integrity and requires a timely closing of the respective fire dampers. It is noted that the fire compartments function is generally not limited to the confinement of the heat and flames it is also the confinement of the smoke which is of importance for the personnel's health and safety. This excludes the approach exclusively relying on fire damper actuation by heat due to its delayed closing. The FDCS is required in order to provide the closing function.

If the FDCS closes fire dampers of ducts serving areas containing heat sensitive equipment (e.g., I&C or electrical devices), dysfunctions of these may occur after a grace period due to loosing proper cooling by the ventilation. This is independent if there is a fire or not and in first order a concern of the properties investment (consequently damage) but becomes significant if another train of a safety system is concerned.

Upon fire detection the FDCS closes the fire dampers of the respective fire compartment. This has an effect on the ventilation and, in consequence, also on the cooling of different areas, at first on the insulated fire compartment but also to remote areas. Furthermore, the balance of the system will be disturbed causing pressure effects and undefined air-flows. For example, the supply air is interrupted while exhaust is still in operation, which generates huge pressure differences between the different areas with significant effect e.g. at doors, where it might not be possible to open doors or even to close them, and huge air flows through leakages at the concerned boundary. Those air flows, which are resulting from the pressure difference, boost smoke spreading through the building in the event of fire. Furthermore, subsequent fire dampers will be closed by the FDCS upon fire detection which will further increase the effect.

It might be concluded that doors certified as smoke control doors will entirely prevent smoke spreading to the unexposed side, even when exposed to fire. This is a misconception because the basic ideas of fire resistance and smoke control are different and therefore tested under different exposure conditions.

FIRE DOOR AND SMOKE CONTROL DOOR PERFORMANCE

The following performance characteristics are used to determine the fire resistance ratings and smoke control performances of doors (example EN 1634-1 [1]):

- Integrity (E);
- Insulation (I);
- Smoke tightness (S).

FIRE RESISTANCE PERFORMANCE

Integrity (E) is a performance characteristic used to describe the ability of a construction element that is used as a means of separation to withstand exposure to fire on a single side without allowing the propagation of fire (flames and hot gases) to the unexposed side resulting from the passage of flames or hot gases. The determination parameters are:

- Visual observation of flames (limited in size und duration);
- Cotton pat test (no ignition);
- Cracks (< 25 mm; < 6 mm through a distance of < 100 mm).

The corresponding test devices are shown in Figure 1.

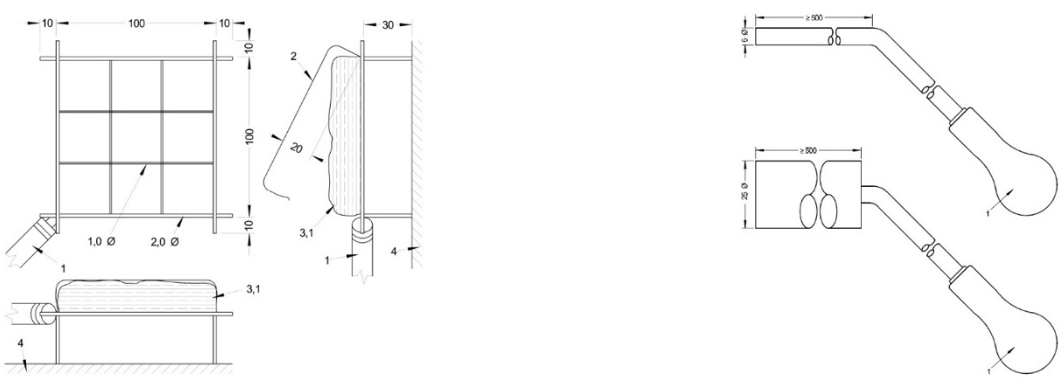

Figure 1 Cotton pad test device and gap gauge

Insulation (I) is a performance characteristic used to describe the ability of a construction element to withstand exposure to fire on a single side without allowing the propagation of fire to the unexposed side resulting from a significant transfer of heat on the exposure side. The transmission of heat through the construction element shall not cause ignition of the unexposed surface or any other material in close proximity to the heated surface. The insulation performance of construction elements is determined by the surface temperature on the unexposed side of the construction element. The determination parameters are:

- Maximum temperature limit < 180 °C;
- Maximum average temperature limit < 140 °C (at dedicated locations).

Note:

The temperature in areas close to gaps and components penetrating the door (25 mm or 100 mm depending on the class) are not respected for the determination of the insulation performance criteria.

The classification of construction products and building elements is expressed as “rating” according to the period of time where the components will continue to perform their function based on the performance characteristic tests described above.

SMOKE CONTROL PERFORMANCE

Smoke tightness (S) is a performance characteristic used to describe the ability of a construction product or building element to withstand exposure of smoke on a single side without allowing the propagation of extensive smoke volumes to the unexposed side. The propagation of smoke through the construction element shall not cause such smoke concentrations that safe egress on the unexposed side is endangered (hypothesis). The determination parameters are:

- Leak rate of < 20 m^3/h; < 30 m^3/h (single; double leaf door) at 200 °C and 50 Pa;
- Leak rate of < 3 m^3/h per meter gap at ambient temperature and 25 Pa.

For the fire resistance classification only the Integrity (E) and the insulation criteria (I) are considered. There is no requirement with respect to the transfer of smoke. Indeed, a fire door provided large contribution to the prevention of smoke spreading but it can only act as a type of resistor limiting the flow rate. This is in general not different to smoke control doors where its tightness is determined by the leakage rate.

Therefore, fire doors as well as smoke control doors allow the transfer of smoke towards the unexposed side where it, depending on the leakage rate and the room geometry, will accumulate.

Figure 2 shows exemplarily smoke leak through a fire door during a fire resistance classification test.

Figure 2 Smoke penetration through closed fire door

As visible from Figure 2, there is certain smoke release to the unexposed side which will accumulate inside the connected room (probably below the ceiling) in a real room configuration.

It is noted that the gas pressure inside the test furnace during a standard fire test is such adjusted that the upper part of the fire doors is exposed by a certain overpressure and the lower part to a sub-pressure. This corresponds to typical room fires. Fires in confined spaces (e.g. rooms without windows) might generate higher pressures through the entire door surface (overpressure even at the bottom) thus smoke spreading might probably be larger.

Fire Detection Systems

The function of a fire detection system is to detect fires. A “fire” however is a hypothetically term which stands generally for a combustion process causing damages and cannot uniquely be specified. Therefore, a fire detection system is not able to identify a “fire”, it assesses phenomena aligned to a “fire”. Such phenomena are in general:

- Heat / Temperature absolute or rise (temperature gradient);
- Opaqueness light (transmission through a test length);
- Thermal radiation (mainly infrared radiation).

The detectors are sensitive to one or more of these parameters. The task of the fire detection system is to assess the different parameters and combinations thereof and to determine if the values are in such a range typically aligned to a fire.

If the parameters are within such range the FDCS will trigger a signal that a fire has been detected and may perform further actions such as isolating the concerned fire compartment by closing all associated fire dampers.

Since the FDCS function is a timely detection of incipient fires and to report detections to a location providing permanent supervision, such as the main control room or other locations which are permanently supervised, FDCSs typically have a looped architecture connecting the different FDCS sub-units and other interfaces (see Figure 3).

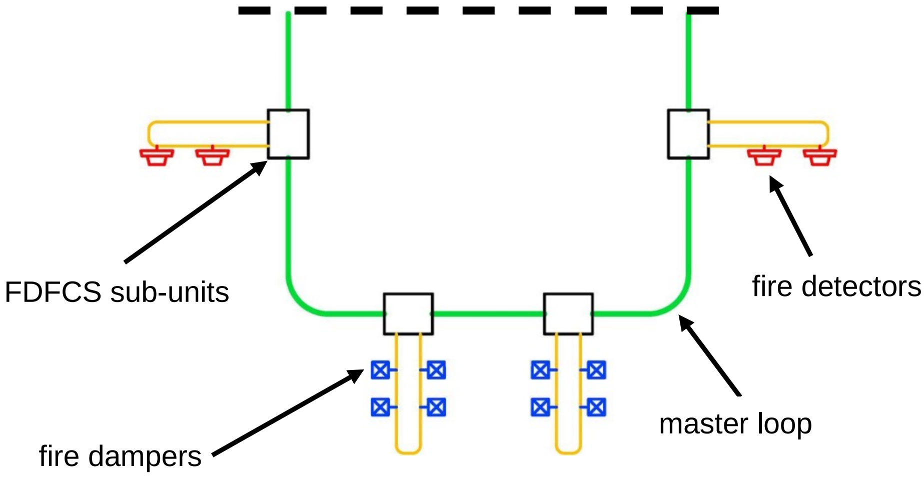

Figure 3 FDCS general architecture – common loop

This loop constitutes a possible source of a common cause failure (CCF). A possible failure sequence might be that a spurious failure in one of the FDCS sub-units propagates to other sub-units demanding the closure of fire dampers controlled by the respective units.

It is however noted that the interface between the different sub-units requires a specific data protocol. If it can be demonstrated that a spurious command does not meet the data logic or is not reasonable for the respective other sub-units, they probably do not react.

Otherwise no looped architecture can be used. This will require entirely autarkic FDCS units without any connection between each other having separate interfaces to the permanent supervised area (e.g., main control room (MCR).

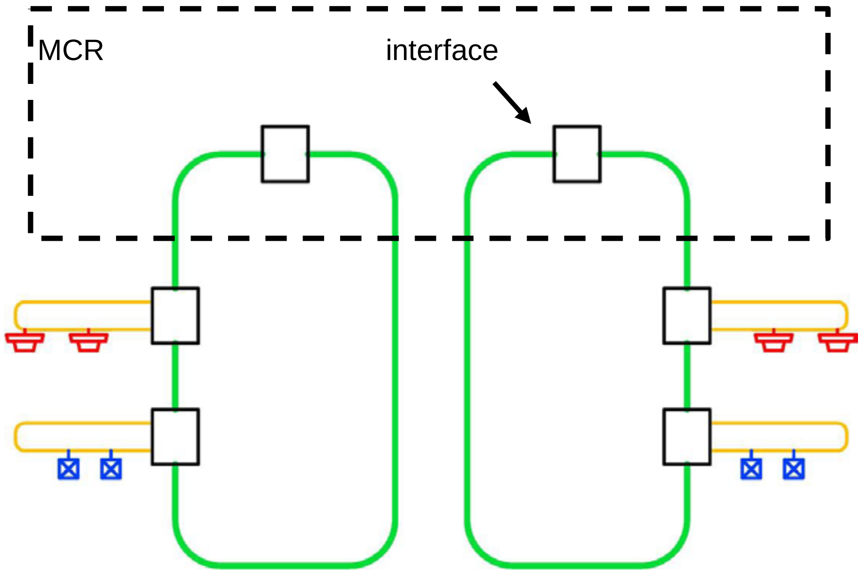

Figure 4 FDCS general architecture – no common loop

FDCS WITH RESPECT TO NUCLEAR SAFETY ASPECTS

Generally, there are two safety aspects addressed to the function of the FDCS, one is the demand for a timely closing of fire dampers and he other is the cooling of heat sensitive equipment. The closing of fire dampers will affect the ventilation system and the cooling of heat sensitive equipment.

Fire doors and smoke control doors however cannot be considered as smoke tight and smoke propagation cannot entirely be prevented. This effect is boosted by the ventilation system due to the disturbance of its balance and the resulting pressure effects. There is potential of subsequent closing of fire dampers (cascading effect) by the FDCS which in turn will enforce the effect on the ventilation system.

It is noted that smoke propagation occurs through all leakage and is not limited to doors whereby doors constitute the largest source of leakage.

In principle, a train separation concept is applied, therefore, a safety related ventilation function is not extended through multiple trains. From that point of view there is generally no major safety issue since the effect is limited to a single train. A safety issue will arise if smoke propagates through multiple trains of safety systems resulting in subsequent closure of fire dampers.

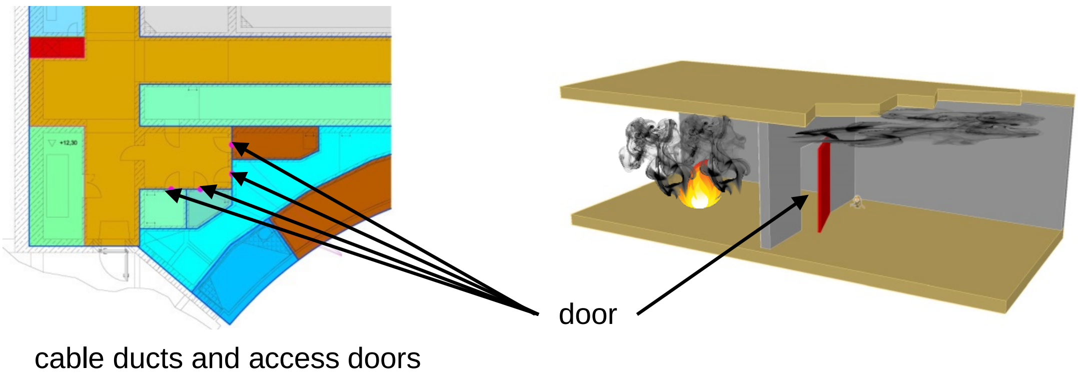

Figure 5 Doors separating different trains of safety systems and potential smoke spreading

As a general approach, penetrations (e.g., for cables or pipework) through fire barriers separating different trains of the safety systems shall be limited. This however cannot be avoided due to necessary interconnections. Moreover, for doors there are security and radiological constraints as well. These are generally a result of the interaction of radiologic confinements, access control, and the protection of the outer building shell. This prevents the plant design from having alternative access routes from the outside.

Furthermore, FDCSs generally meet non-nuclear industrial standards which gives rise for demanding a safety classification serving as control function of a safety classified system. The general rule is that safety classified systems shall not be controlled by a lower or non-safety classified system. The safety classification approach implies a certain robustness against malfunctions, but for the particular case of the FDCS it exceeds the boundaries of the classification idea.

Even if the FDCS system is safety classified it will not be more robust to prevent faulty closing of fire dampers. This is due to the subjective evaluation of the fire phenomena and the possible detection parameters. It is quite difficult to distinguish between a fire and other sources to which the detectors are sensitive, and it is even not possible to identify the origin of heat or smoke. The governing question is how to deal with all these constraints and still meet the required safety functions.

Potential options to be assessed for risk prevention as identified above are:

- Avoiding connections (potential smoke leakage paths) between redundant trains of safety systems;
- Providing sufficiently tight barriers;
- No automatic closing of those fire dampers belonging to rooms containing equipment sensitive to heat. Those dampers will then be closed by thermostatic switch or manually;
- Automatic closing by the FDCS.

In the event of fire, the fire compartment where the fire has been detected will be isolated by closing the associated fire dampers. This closing of fire dampers will more or less impact the building's ventilation.

If a ventilation duct is interrupted principally two conditions result:

- Parts of the ventilation system go out of operation and cooling is not any more provided for these areas. This is however limited to a single train due to the redundant architecture of the ventilation systems.
- The balance (equilibrium) of the ventilation system is disturbed which may significantly affect smoke propagation through leakages as flow rates, flow path and pressure differences change.

The range of the balance disturbance and the resulting pressure differences depends on the ventilation systems architecture, the size of the disconnected volume flow rate and which part of the ventilation system is no longer operating (supply only, or exhaust, or both). Particular larger pressure differences significantly affect potential smoke propagation through leakages. In consequence, further fire compartments might be isolated by the FDCS and at worst the entire ventilation system might be stopped.

From this point of view and its aligned difficulties in the assessment, it might be better stopping the ventilation system even for personnel safety purposes. However, the time period needed for restarting the ventilation needs to be considered (including local or remote-controlled reopening of fire dampers) in particular with respect to small fires or false alarms. This is because a lack of ventilation might cause damage of heat sensitive equipment already in case of negligible events. The described sequence become particular importance if smoke spreads through borders between redundant trains of safety systems.

It is noted that the performances of the construction products and building elements to be installed in fire barriers are identically for all types of fire compartments (see "Fire door and smoke control door performances"). There are no specific smoke tightness performances for borders between divisions. It can therefore be assumed that smoke spreading through safety division borders is possible and will result in the same sequence as smoke spreading inside the division.

As stated before, the consequential lack of ventilation and thus its cooling function are not limited to the fire compartment next to the leak in the division barrier. It might be also possible that further areas will be affected.

Note: A division is considered as an area containing only one redundant train of a safety system. A division can contain different safety systems, but its redundant train is contained in the respective other divisions.

From nuclear safety point of view this is not acceptable since the entire ventilation system is required to be fully in operation (no loss of more than one train of a safety system). It is therefore mandatory that all dampers of the ventilation system serving the division not containing the fire shall remain in their required (non-fire) position.

For this case, automatic damper closure by smoke detection has to be prevented by e.g. equipping all these fire dampers with a closing device not sensitive to smoke such as a fusible link. It is noted that this will concern all fire dampers inside a division due the balancing of the ventilation system.

However, as smoke constitutes per se a risk for human life safety a non-closing or delayed closing of fire dampers resulting from the use of fusible links is not acceptable. This is however different for the event that smoke spreads to another division. In this case, smoke spreading is rater limited due to the limited number of connections and tightness requirements (e.g., air and water tightness). The resulting smoke concentrations will not jeopardize human life safety but are still capable of triggering fire alarms.

Inside a division smoke propagation caused by non-automatic closure of fire dampers is not acceptable from human life safety point of view and patrimonial aspects.

It is noted that smoke propagation may also induce a safety issue due to possible damages of equipment (long-term corrosion, or instantaneous failure, particularly of I&C) outside the fire compartment. In particular for equipment sensitive to smoke (e.g., I&C) the view is not clear (see IRSN CATHODE test series presented in [2]).

By superposing the consequences aligned to smoke propagation inside a division and between divisions all fire dampers are required to be controlled by the FDCS. This is because it is postulated that a (single) fire can start anywhere (vice-versa assessment) and smoke propagation by non-automatic closure of fire dampers is not acceptable from life safety and nuclear safety point of view (long-term corrosion, or instantaneous failure in particular of I&C) and property aspects.

In order to cope with the consequences aligned to smoke propagation between divisions a subsequent closure of fire dampers in multiple divisions has to be prevented. This requires that the closing commands of the FDCS of all fire dampers inside the division not containing the fire are blocked or recirculation cooling units are provided for those areas containing equipment sensitive to heat.

CONCLUSIONS

The general fire protection approach demands a timely detection of incipient fires following the defence-in-depth concept. A timely initiation of fire countermeasures, which is in first order the fire confinement, requires a prompt closing of fire dampers. Therefore, it is mandatory that the all fire dampers are controlled (to close) by the FDCS. This is in particular essential for human life safety aspects because of the smoke toxicity.

Because an automatic closure of fire dampers might jeopardize essential ventilation functions (cooling of items important to safety) certain cases are known where an automatic damper closure has to be prevented or an alternative cooling source has to be provided. This is in particular the case if multiple divisions are concerned. The analyses of alternatives proposed such as closing by temperature gauge has shown that these proposals do not capture the timely detection and fire confinement needs or are rather costive. The scenario to be dealt with is a multiple closure of fire dampers in different divisions due to smoke spreading.

The essential safety function which has to be performed is the cooling of safety related equipment. In general, this is performed by the central ventilation system(s) the function of which is jeopardized by closing fire dampers. Therefore, it is necessary to provide an alternative cooling method such as the use of recirculation cooling units or to ensure that cooling by ventilation can be performed. At boundaries between divisions this can be provided by a specific control mode of the FDCS or by providing sufficiently tight smoke barriers such as “smoke” airlocks at doors.

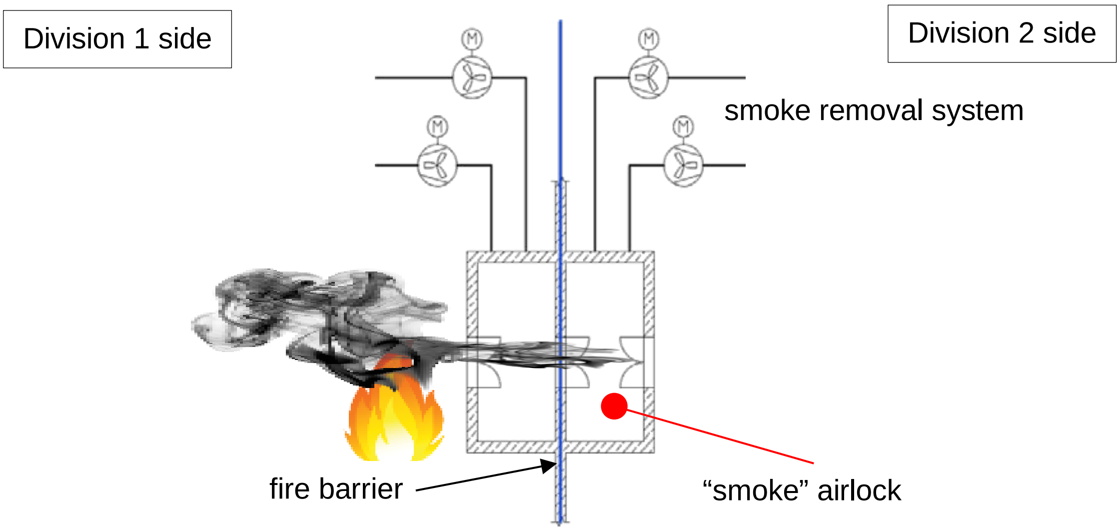

Figure 6 “Smoke” airlock

A so-called “smoke” airlock as shown in Figure 6 provides a means of extracting leaked smoke from the door. The fire in one division is separated from the adjacent division by a fire barrier (blue line). A dedicated ventilation system removes the smoke from the room. It is noted that the airlocks as shown in Figure 6 are required to be free on any fire load thus as fire inside these rooms is excluded. Two of such airlocks are required side-wise of the door due to symmetry reason (fire on the other side). Variations are shown in Figure 7.

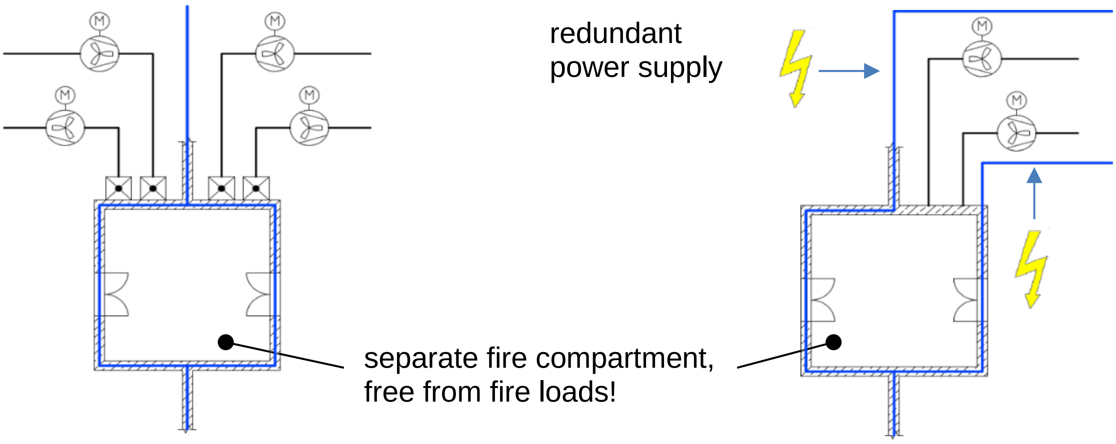

Figure 7 Variation of a “smoke” airlock

The variations in Figure 7 require a separate fire compartment which is free of any fire load (fire exclusion). As this alternative is rather costive and space consuming other protection features are investigated. It is shown that all assessments result in the blockage/suppression of faulty closing signals except for the spurious failure of the FDCS and subsequent failure propagation through the master loop. In the latter case, the item can only be captured by using a failure robust signal protocol (data logic) or by autarkic FDCS per each division. The following approach for signal blockage/suppression might be possible:

In the event of smoke spreading through division borders three functions are available:

a) The blockage/suppression signal is transmitted via the master loop towards all FDCS sub-units (see Figure 8).

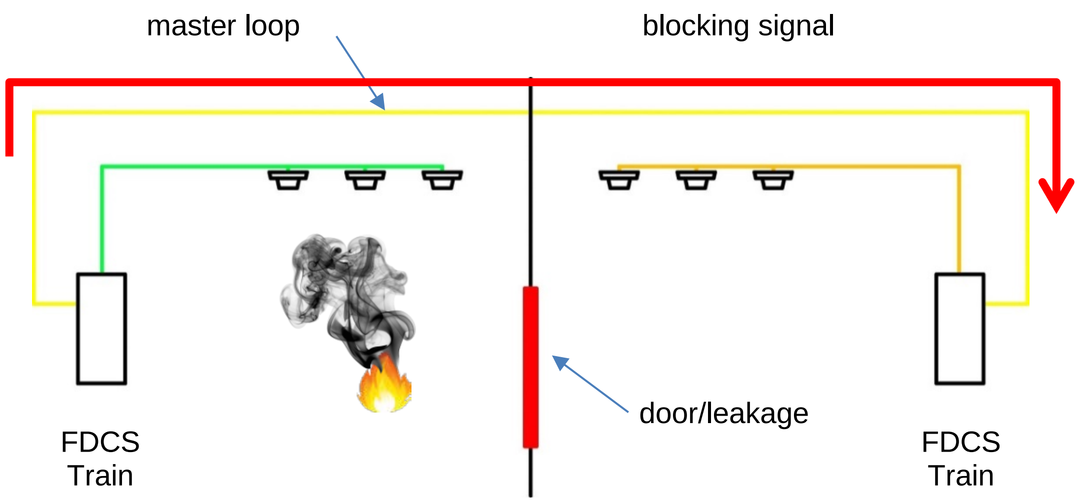

Figure 8 Blocking signal automatic by master loop

This feature will only work if a master loop exists and concerns also divisions remotely to the division containing the fire in case that no selective signal is provided. Furthermore, the FDCS which detects the fire shall not close those fire dampers inside the divisions which do not contain the fire, but which are controlled by the FDCS sub-unit.

b) The blockage/suppression signal is initiated by a dedicated fire detector which is part of the FDCS sub-unit adjacent to the respective division containing the fire.

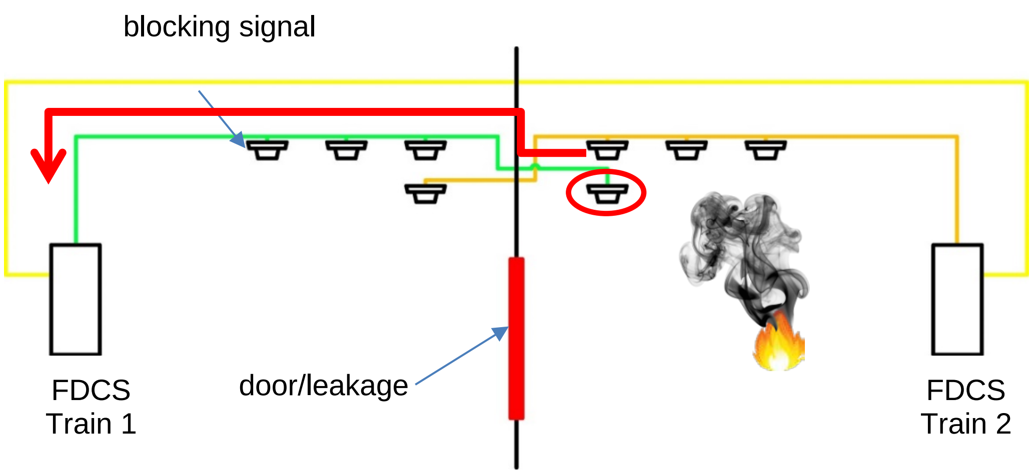

Figure 9 Blocking signal by dedicated detector

This architecture is independent from the master loop and will not affect all divisions of the plant. Only the division which is next to the one containing the fire will be affected, all other fire compartments operate ‘as usual’.

c) The blockage/suppression signal is initiated manually by a supervisor (e.g., main control room personnel).

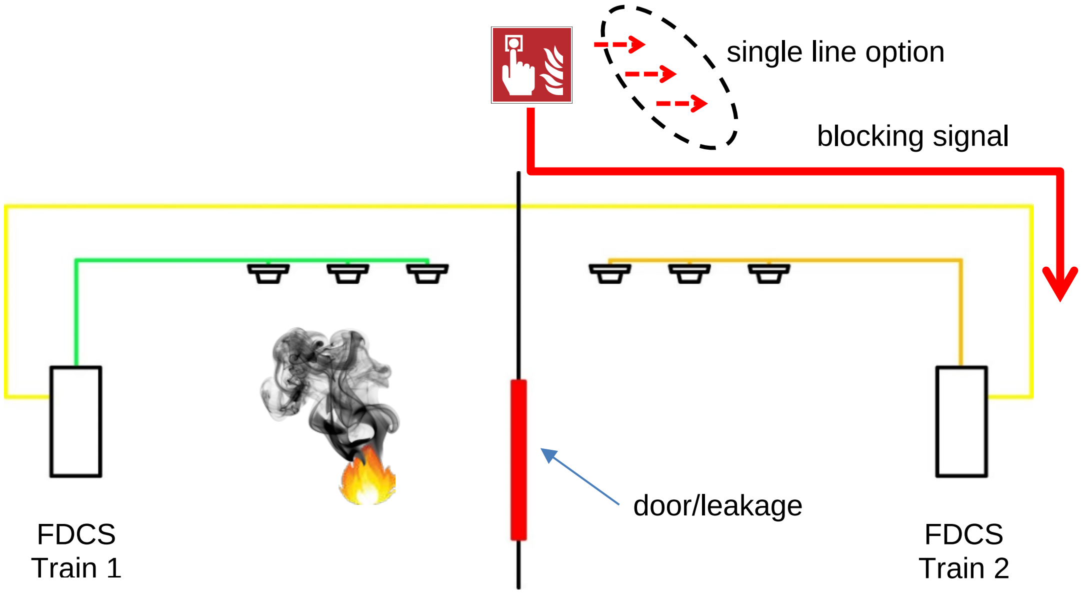

Figure 10 Blocking signal manually by master loop

The feature is not necessarily dependent from the master loop. The signal can be transmitted by the master loop or by dedicated separate lines. A selection logic shall prevent the division containing the fire from applying the blockage signal. The approach is however error-prone for delayed activation if smoke has already been leaked towards an adjacent division thus a prompt activation from the supervisor is requested.

In order to prevent CCF due to smoke spreading through different divisions two different approached were identified, one relies on capturing the smoke by a “smoke” airlock, the other one on a specific blocking signal which prevents from further fire dampers closing by the FDCS. Bothe approaches have advantages and disadvantages. However, although the “smoke” airlock solution implies that smoke spreading to the adjacent division is entirely prevented it requires as specific control by the FDCS. This is in first order the starting of the associated smoke removal fans, but also as it cannot be guaranteed that more than one door of the airlock is open simultaneously thus the approach having a blocking signal is required anyhow.

REFERENCES

[1] European Norms (EN): EN-1634-1: Fire resistance and smoke control tests for door and shutter assemblies, openable windows and elements of building hardware - Part 1: fire resistance test for door and shutter assemblies and openable windows, 2014.

[2] Piller, M., and M. Coutin: Effects of smoke and thermal stress on electrical equipment failure, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 24, 15th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 4-5, 2017, Bruges, Belgium, GRS-A-3912, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2017, https://www.grs.de/en/publication/grs-3912.

German Operational Experience with Fire Dampers in Nuclear Installations

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ABSTRACT

Fire dampers installed in German nuclear power plants (NPPs) have in the most cases been fire-tested according to the criteria of the German non-nuclear standard DIN 4102, Part 6. Since September 2012, fire dampers are in the scope of application of the harmonized European EN standards resulting in minor design changes.

All fire dampers can be in principle triggered by fusible links at a temperature threshold of 72 °C inside the ventilation duct. Dampers installed in locations important to safety can additionally be triggered remotely.

The fire damper function is divided into the subfunctions “actuation” (by different, partly redundant means) and “closing/barrier function”. Failure characteristics of dampers are illustrated by different cases and by statistical data including failure rates for the subfunctions. The scope of the regular in-service-inspections is explained.

The design of the ventilation systems and the position of fire dampers in the ductwork are also illustrated. In this context, the potential effects of damper failures on the required functions of items important to safety as well as on the conventional non-nuclear fire safety are discussed.

INTRODUCTION

A large amount of rooms in NPPs and other nuclear installations is ventilated by mechanical ventilation systems. If ventilation ducts are routed through walls, floors or ceilings with a required fire resistance rating (fire barriers), fire dampers are used to maintain the specified fire resistance rating. According to the German nuclear fire safety standard KTA 2101 [1], the need for separating fire compartments by rated fire barriers is given for

- walls and ceilings separating rooms with different redundant trains of the safety system,
- walls and ceilings protecting access and escape routes,
- walls and ceilings protecting high fire loads, and
- all ceilings (because fire compartments (and sub-compartments) shall be limited to one building level), if not otherwise required by NPP specific needs.

According to these requirements, typically an amount of approx. 1,000 fire dampers are installed in the German Konvoi type pressurized water reactor (PWR) plants.

QUALIFICATION OF FIRE DAMPERS

Fire dampers installed in German NPPs are qualified according to the conventional non-nuclear building code requirements and have certain design features in common.

Since 1974 fire dampers need an approval mark such as "*PA-X …*") of the Deutsches Institut für Bautechnik (German Institute for Civil Engineering) in Berlin. The approval mark was provided after fire tests and additional tests for leakage and endurance had been successfully performed according to guidelines of a technical experts committee. In September 1977, the original guidelines were superseded by the German standard DIN 4102-6 [2] on the "Behaviour of building materials and components in fire; Ventilation ducts; Definitions, requirements and tests", which include criteria for the qualification of fire dampers.

For the closed damper the leakage rate of the damper had to be smaller than 10 m^3/h per 1 m circumference at the smallest diameter. This value counted for a pressure difference of

- 200 Pa in regular flow direction, if the fan is not switched off after closing, or
- 40 Pa against regular flow direction and in all other cases.

During fire exposure according to the standard fire curve (e. g. ISO 834) for following criteria had to be kept:

- The temperature increase on the surfaces of the cold side of the wall, the chassis of the fire damper, and movable parts of the damper must be lower than 140 K as a mean value and must be lower than 180 K as a maximum value (for this criterion a 5 cm wide strip at the edge of the wall and the fire damper is not considered).
- The temperature increase of the gases leaking through the closed damper must be lower than 140 K.
- Visible flames on the cold side of the damper are not permitted.
- A cotton-wool ball which is held on the cold side of the damper is not permitted to be ignited.

Before the fire test, the fire damper had to be closed by the closing device 50 times.

Since 1995, instead of the approval mark a general technical approval (e.g., "*Z-41.3-xyz*") was required.

Since September 2012 fire dampers in ventilations ducts have to comply with the DIN EN 15650 [3]. The test standard is EN 1366-2 [4] and the classification is according to the harmonized European standard EN 13501-3 [5].

As one classification criterion, for the E-classification (French: Etanchéité) concerning the integrity of the damper in the wall/ceiling, the leakage rate through the damper measured starting 5 min after the beginning of the fire test has to be lower than 360 m^3/m^2*h) at 300 Pa pressure difference. In Germany the classification of dampers for smoke tightness (S) is needed, therefore the value is reduced to 200 $m^3/(m^2*h)$ [5].

The old German requirement of 10 m^3/h per 1 m circumference at 200 Pa (see above), for a typical damper with dimensions 0.8 m * 0.4 m (circumference: 2,4 m, area: 0,32 m^2) at 300 Pa represents 92 $m^3/(m^2*h)$ according to the EN standards. Although 92 $m^3/(m^2*h)$ is smaller than 200 $m^3/(m^2*h)$ for S-classified dampers, it is important to consider that the EN standards require the maximum leakage rate during the fire test and not only in a cold test such as the former German standard [2].

To comply with the new requirements, one change of the fire damper design in Germany (Figure 1) is e.g. the implementation of seals with intumescent coating material in the casing to fit with the damper blade. However, such improvements were made before the harmonized requirements were introduced. The design of fire dampers kept very much like it was according to the German qualification. Figure 1 shows a typical fire damper, in this example only equipped with an actuation by fusible link. An additional remote-controlled actuation leads to changes at the actuation and release mechanism (no. 6 to 9 of Figure 1).

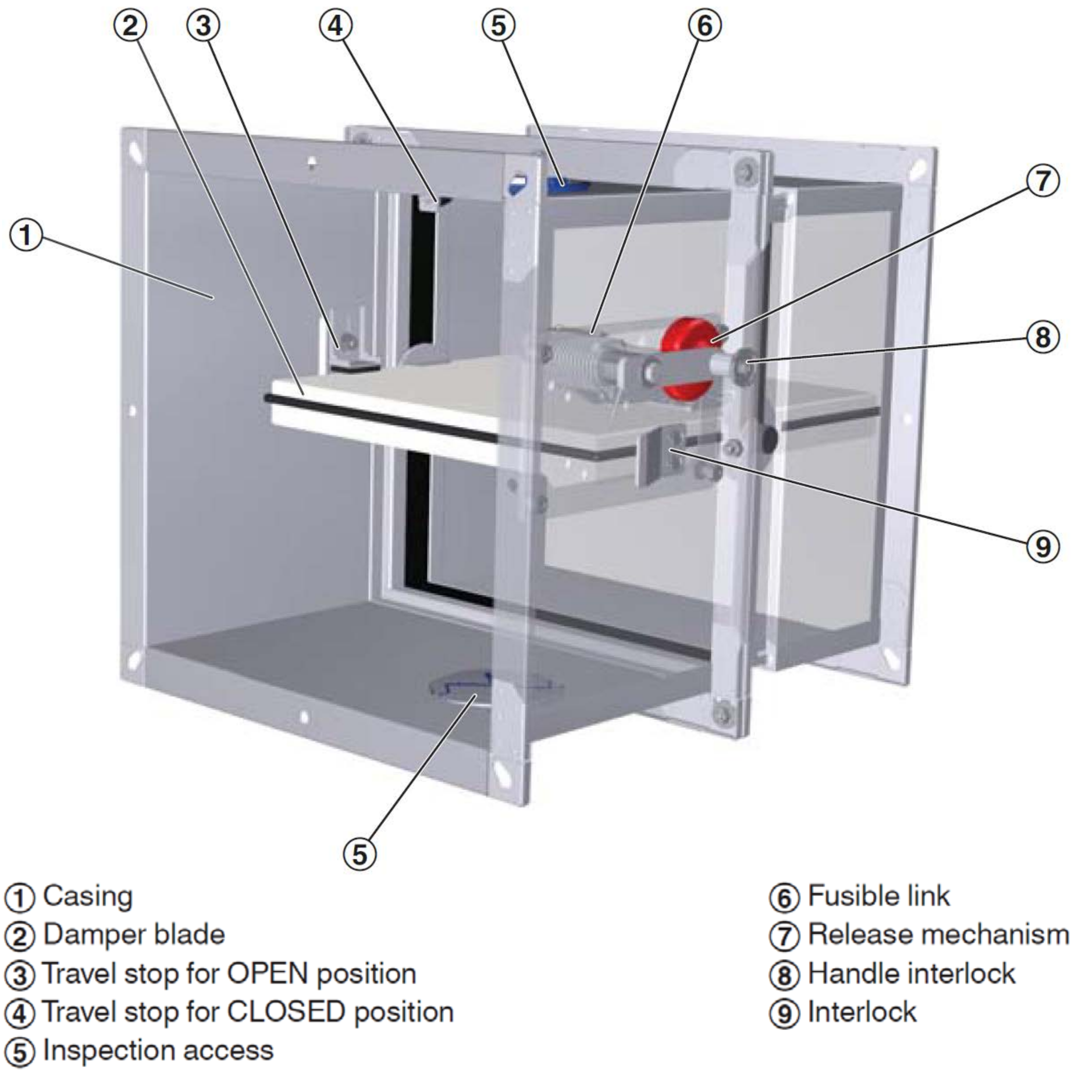

Figure 1 Schematic illustration of a frequently used fire damper with fusible link (image source: Trox GmbH, www.trox.de)

OBSERVATIONS AND FINDINGS FROM IN-SERVICE INSPECTIONS AND REPORTABLE EVENTS

Based on non-nuclear requirements fire dampers have to be tested every six months. If two consecutive tests did not show any relevant deficiency, the test interval can be extended to twelve months.

A test interval of twelve months is also required by the German KTA 2101.1 [1]. Practically, depending on the arrangements regarding the inspections at a specific NPP, e.g., if the testing of the manual actuation is separated from the testing of the automatic actuation, shorter test intervals of the (sub-)functions maybe possible and demanded.

For fire dampers installed in plant areas important to safety, the thermal actuation mechanism by fusible link has been included in the in-service inspection program due to a German Information Notice [6] as a result of findings at fire dampers in the mid-1990s. Meanwhile, in some plants these inspections are carried out periodically every ten years, where the fusible link is molten by a hot air dryer. In other plants, within the yearly inspections the fusible links are unhinged to ensure that all movable parts of the closing mechanism of a damper are moved.

If in-service inspections of fire dampers show findings (deteriorations or failures of the required function), it needs to be checked whether these are reportable events according to the German reporting criteria, which were published first at the end of 1992. The relevant reporting criterion of the current version [7] is defined as follows:

The failure of or damage to a feature of equipment-related or structural fire protection is reportable. Non-reportable are failures or minor damage to individual components of the fire protection features, by which the fire protection functions were not unduly affected.

The meaning of “minor damages” was clarified by the official annotations [8]:

- Failures of individual fire or smoke detectors being self-reported without the fire detection being impaired;
- Damage to structural fire protection features that does not affect the required fire resistance or the retention of smoke (e.g., bump in a fire door, fire damper not opening again after closing);
- Failure of the remote-controlled actuation of a single fire damper without affecting the thermal actuation of the damper.

Although the reporting criteria together with the annotations still leave some room for interpretation. The reported events allow a comprehensive overview of the condition of fire dampers in German NPPs to GRS as TSO (*T*echnical *S*afety *O*rganization).

From 1993 up to the summer of 2019 a total of 65 events with fire dampers was reported according to the German reporting criteria [7]. Most of the findings characterized in the reports were observed during periodic in-service inspections or additional inspections based on a German Information Notice [6]. The remaining cases are events which were discovered during an acceptance test, during a plant walkthrough or during maintenance at another system.

Considering the reportable events with fire dampers, three major damage mechanisms or causes have been found.

On the one hand, there were difficulties at the actuation mechanism or at the damper blade due to hardening (of grease or similar) or impurity. These include, for example, a reportable event where the mobility of the release bolt was restricted by impurity, or an event, where residuals of asbestos fibre binder led to sticking of the damper blade in the housing.

On the other hand, in some cases the closure of the damper blade was prevented due to mechanical deformations. This was caused, for example, by blocking bolts in which grooves were rubbed through multiple closing processes.

Events caused by deficiencies in the design, construction or manufacturing can be classified as a third category of findings. These include, for example, a release spring being undersized or the installation of an incorrect anchor guide tube by the manufacturer. Occasionally, operating errors occurred which, in connection with inadequate design, caused the failure of the closing mechanism. In this context, deficiencies in the thermal actuation resulted in a German Information Notice [6] that was supplemented six times.

In addition, there were events where a fire damper failed for other reasons. For example, a fire damper could not be closed because the motor to drive the damper was overheated and blocked. Further error mechanisms, specific for pneumatically controlled dampers, were a defective piston of a solenoid valve or defective exhaust air throttle valves.

STATISTICAL RELIABILITY ESTIMATIONS

Based on the evaluation of the documented results of in-service inspections, which include deteriorations that were considered below the reporting thresholds, reliability data were derived as input for Fire PSA (*P*robabilistic *S*afety *A*nalysis). Among other active fire protection features this study [9],[10] included data for fire dampers from six German power reactor units comprising more than one-hundred reactor years.

Different types of fire dampers used in the German NPPs were investigated. All fire dampers are equipped with thermal actuation, the wide majority of them by a fusible link. Dampers in plant areas important to safety can additionally be actuated remote-controlled. The following remote-controlled fire damper actuation types were analyzed:

- electro-magnetic valves which release air-pressure from a pneumatic system to close the dampers (closed-circuit principle) (type 1),
- lifting magnets which draw back a bolt when actuated to close the dampers (open-circuit principle) additionally equipped with a pneumatic support to re-open the blade (type 2),
- lifting magnets moving back a bolt when actuated to close the dampers (open-circuit principle), partly equipped with a crank lever to re-open the blade (type 3), and
- magnetic clamps (closed-circuit principle) that release the blade when deactivated (type 4).

Remote-controlled actuation and thermal actuation are diverse actuation mechanisms. To consider this for the damper reliability, the failures observed were assigned to different sub-functions and a fault tree for technical failures was generated (cf. Figure 2). In addition to the actuation, in case of fire the damper blade has to move to the closed position and remain structurally intact to fulfill its required ‘closing/barrier function’. Since not all fire dampers have the additional remote-controlled actuation mechanism, this mechanism is marked with a dashed line in Figure 2. All dampers can be manually operated by means of a test button on one side of the fire barrier penetrated by the ventilation duct. However, this actuation mechanism is not accessible in many cases and is therefore marked with a dotted line in Figure 2. A failure of a fire damper occurs in case of failure of either all actuation mechanisms of the damper or failure of the ‘closing/barrier function’.

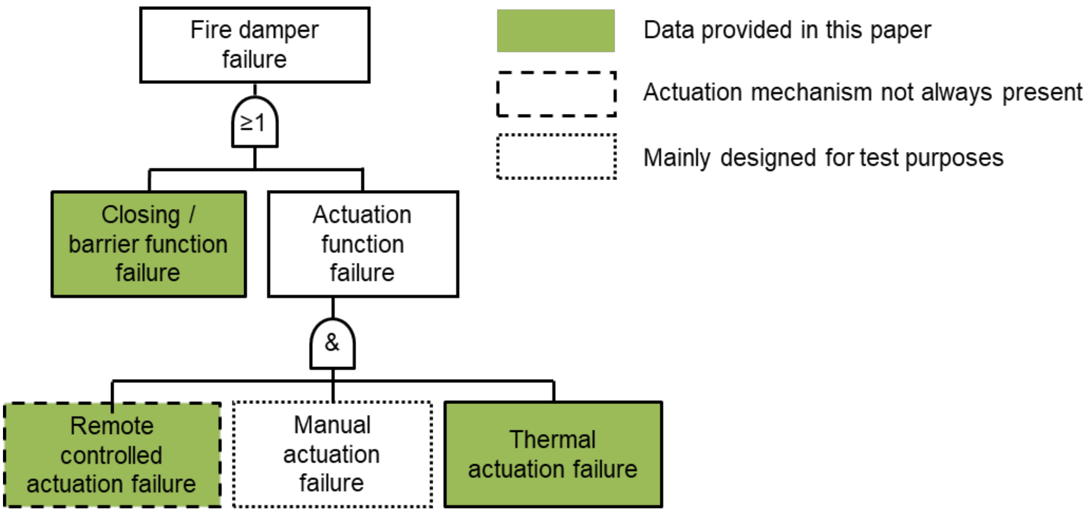

Figure 2 Fault tree for technical failures of fire dampers

The results of the evaluation are presented in Table 1.

Table 1 Results of the statistical evaluation for fire dampers, from [10]

Active Fire Protection Features	Number of components	Test interval [years]	Time observed [h]	Number of failures	Failure rate [1/h]				
					5 % quantile	50 % quantile	95 % quantile	Mean value	Standard deviation
Fire dampers [a]									
- Closing/Barrier function (all dampers)	3 799	1/2 / 1[b]	500 581 612	117	3.91E-08	2.10E-07	6.07E-07	2.51E-07	8.88E-08
- Actuation:									
- Remote controlled									
- Electro-pneumatic (type 1)	505	1[b]	72 262 028	10	2.93E-09	4.75E-07	6.12E-06	1.66E-06	4.75E-06
- Lifting magnet plus pneumatic reopening (type 2)	539	1[b]	61 423 362	148	3.69E-07	1.98E-06	5.96E-06	2.42E-06	1.81E-06
- Lifting magnet (type 3)	1 308	1[b]	185 930 706	74	3.75E-08	5.55E-07	2.66E-06	8.22E-07	5.79E-07
- Magnetic clamp (type 4)	74	1[b]	6 354 822	4	9.09E-09	7.52E-07	7.99E-06	4.82E-06	2.99E-05
- Thermal (fusible link)	3 370	1 , 10[c]	321 640 836	125	1.54E-09	2.07E-07	2.00E-06	4.83E-07	5.85E-07

[a] A share of the fire dampers and fire doors is located inside exclusion areas where the testing interval is extened to one fuel cycle (≈15 months)

[b] The most common test interval for fire dampers is 1 year, sometimes 6 months. In some plants manual and remote testing is separated, that on average every 6 months the blade is moved.

[c] In some plants, the fusible link is unhinged every year - other plants do destructive tests with a hot air dryer every 10 years

Concerning the ‘closing/barrier function’, 117 functional failures were observed. Most of these failures occurred because of dust deposit or resinified oil on the moving inner parts of the dampers which blocked the closing function. Sometimes, moving parts of the damper itself were bent over the time. A small number of failures were caused by significant damages of the damper blades, i.e. deteriorating the barrier function. A resulting mean failure rate of λ = 2.5 E-07/h was estimated.

Regarding the remote-controlled actuation, the complete signal line from the trigger (e.g., the main control room, the fire detection system or a local control panel) to the damper is covered. Failures are always assigned to the dampers, even if the failure is located at the trigger or the signal line to the damper, because these are not modelled. Typical failures are electrical like malfunctions of pneumatic valves or stuck lifting magnets. Although the actuation mechanism types 2 and 3 are based on the outdated open-circuit principle, there was no significant difference in the failure rates of all four types observed; all failure rate mean values are within one order of magnitude. Moreover, the functional unavailability of the thermal actuation via fusible link is in the same order of magnitude. A mean failure rate of λ = 4.8 E-07 /h was estimated, which is the lowest value of the entire mean failure rates for the different types of actuation. However, as the test interval for the thermal actuation could be about ten times longer than that for remote-controlled actuation, the resulting unavailability per demand may increase up to the upper boundary in comparison to all other actuation mechanisms.

EFFECTS OF DAMPER FAILURES DURING FIRES

To the knowledge of the author, there has been no reportable fire event in a German NPP where a fire damper failed. Moreover, none of the fire events internationally reported in the OECD Nuclear Energy Agency (NEA) Database FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) [11] describes an effect of a fire damper failure during an accidental fire. Therefore, the following considerations are theoretical.

Regarding the fact that in German NPPs each redundant train of the safety system has got basically an own ventilation system (e.g. in the electrical building, see Figure 3) or the ductwork is assigned to one safety train only, the number of fire dampers separating compartments with equipment of different safety trains is very small. This concept limits the potential effects of damper failures significantly.

Generally, if an inlet-air damper does not close in case of fire, the air flow will support combustion. In case the pressure increase in the fire compartment allows for a contraflow into the inlet duct, a certain spread of smoke and hot gases cannot be excluded. Other leakages like those at fire doors will also contribute to smoke carryover. If an outlet air damper does not close in case of fire, smoke is immediately drawn into the (main) duct.

A closer look should be taken to the ventilation system of one redundant train of a German electrical building (see Figure 3 below). The main air supply duct (blue, left), the main extract air duct (yellow, right) and the smoke extraction duct (grey, far right) are made of concrete. The air supply duct and the air extraction duct are connected to horizontal sheet metal ducts via wall-mounted fire dampers (red). The air supply ducts are additionally equipped with rated bypass dampers close to the floor and the smoke extraction ducts are connected to the compartments by rated smoke extraction dampers close to the ceiling which are both closed under normal operating conditions.

Figure 3 Schematic layout of the ventilation system in one redundant train of the electrical building

In the event of fire, the train where the fire occurs is supplied with external air, whereas the other trains are operated in recirculation mode. For the fire compartment, in order to fight the fire, the fire damper to the air extraction duct closes and the smoke extraction damper to the smoke extraction duct opens. Additionally, the supply air is increased by opening the bypass dampers.

Smoke diffusion into the air extraction occurs as soon as the fire damper fails to close. However, a certain overpressure is needed for smoke spreading against normal flow direction into adjacent rooms. In this case, the damper of another building elevation level can possibly close to prevent significant vertical smoke carryover.

Smoke diffusion into the main supply air duct can theoretically occur, if the smoke extraction damper fails to open, and the fire compartment would be very tight. Practically, the leakages, e.g. by fire doors, will allow for a flow direction out of the supply air duct into the fire compartment at both the fire damper level and the bypass damper level.

This design practically prevents smoke carryover by a single failure of a fire damper.

CONCLUSIONS

GRS continuously evaluates the operating experience in NPPs on behalf of the Federal German regulatory body, the Ministry for the Environment, Nature Conversation and Nuclear Safety (BMU). This also includes reportable events concerning fire dampers. The majority of fire dampers installed in German NPPs are qualified based on requirements from German non-nuclear standards. The most recent common European standards resulted in minor design changes of the dampers. In the 1990s, GRS drafted in total six German Information Notices with respect to systematic functional failures or deteriorations of fire dampers. In consequence, numerous corrective measures to prevent recurrence were taken in German plants. The review of reportable events on fire dampers demonstrates that the systematic failure mechanisms observed from the operating experience of fire dampers in German NPPs are covered by the corresponding Information Notices and the recommendations given there. The remaining events can be assessed as single failures.

The failure rates for fire dampers, which can be estimated from the detailed analysis of findings from periodic in-service inspections, are not higher compared to failure rates determined for other active fire protection features installed in NPPs. Nevertheless, the design of the ventilation system limits the possibility of fire propagation to adjacent compartments or inadmissible impacts to redundant safety trains by single failures of fire dampers.

ACKNOWLEDGEMENTS

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REFERENCES

[1] Nuclear Safety Standards Commission (KTA, German for Kerntechnischer Ausschuss): KTA 2101.1 (2015-11) Fire Protection in Nuclear Power Plants, Part 1-3, November 2015,
http://www.kta-gs.de/e/standards/2100/2101_1_engl_2015_11.pdf.
http://www.kta-gs.de/e/standards/2100/2101_2_engl_2015_11.pdf.
http://www.kta-gs.de/e/standards/2100/2101_3_engl_2015_11.pdf.

[2] Deutsches Institut für Normung (DIN) (Ed.): DIN 4102-6, Behaviour of building materials and components in fire; Ventilation ducts; Definitions, requirements and tests, 1977-09, 1977.

[3] Deutsches Institut für Normung (DIN) (Ed.): DIN EN 15650, Lüftung von Gebäuden – Brandschutzklappen; Ventilation for buildings – Fire dampers; 2010-09, 2010.

[4] Deutsches Institut für Normung (DIN) (Ed.): DIN EN 1366-2. Feuerwiderstandsprüfungen für Installationen – Teil 2: Brandschutzklappen; Fire resistance tests for service installations – Part 2: Fire dampers; German version: 2015-09, 2015.

[5] Deutsches Institut für Normung (Ed.): DIN EN 13501-3, Klassifizierung von Bauprodukten und Bauarten zu ihrem Brandverhalten – Teil 3: Klassifizierung mit den Ergebnissen aus den Feuerwiderstandsprüfungen an Bauteilen von haustechnischen Anlagen: Feuerwiderstandsfähige Leitungen und Brandschutzklappen; Fire classification of construction products and building elements – Part 3: Classification using data from fire resistance tests on products and elements used in building service installations: fire resisting ducts and fire dampers; 2010-02, February 2010.

[6] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH: Weiterleitungsnachricht (German: for Information Notice) zu meldepflichtigen Ereignissen in Kernkraftwerken der Bundesrepublik Deutschland (WLN 1994/01 and Supplements A-D) „Funktionsstörung in der Mechanik der Schmelzlotauslösung von Brandschutzklappen im KXX und KYY“ (Malfunction in the mechanics of the thermal-actuation of fire dampers in NPP XXX and YYY) (title anonymized), Köln, Germany, 1994 (in German).

[7] Verordnung über den kerntechnischen Sicherheitsbeauftragten und über die Meldung von Störfällen und sonstigen Ereignissen (Atomrechtliche Sicherheitsbeauftragten- und Meldeverordnung - AtSMV) vom 14. Oktober 1992 (BGBl. I 1992, Nr. 48, S. 1766), zuletzt durch Artikel 18 der Verordnung vom 29. November 2018 (BGBl. I 2018, Nr. 41, S. 2034) geändert, 2018 (in German).

[8] Länderausschuss für Atomkernenergie (Ed.): Erläuterungen zu den Meldekriterien gemäß Anlage 1 der AtSMV, 2015-09, 2015 (in German).

[9] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH: Ergänzung und Aktualisierung von Zuverlässigkeitskenngrößen für Brandschutzeinrichtungen in deutschen Leichtwasserreaktoren, GRS-A-3719, Köln, Germany, May 2014, published as “Ressortforschungsbericht zur kerntechnischen Sicherheit und zum Strahlenschutz“, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, December 2016,
https://doris.bfs.de/jspui/bitstream/urn:nbn:de:0221-2016120914218/3/BfS_2016_3610R01370.pdf.

[10] Forell, B., at al.: Technical reliability of active fire protection features - generic database derived from German nuclear power plants, Reliability Engineering & System Safety, Volume 145, pp. 277-286, January 2016.

[11] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Database Version 2017:02, Paris, France, May 2019 (limited to FIRE Database Project member countries only).

Experience in Regard to Qualification of Fire Detection and Control Systems

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ABSTRACT

The qualification requirements to fire detection and control systems have increased over the last decades. In nuclear projects started in the last century fire detection and control systems were manufactured mostly based on industrial standards. The only nuclear specific qualification concerns are to seismic events where functional operation during and after a seismic event was required.

Later on, the equipment qualification was extended to radiation resistance. This was however just to define the lifetime of specific equipment when exposed to radiation.

In 2014, IT security was questioned for fire detection and control systems. How can the fire detection and control system be protected against influences from any interface, and could the system affect other interfacing systems?

With new nuclear projects in 2018/19 to the engineering of fire detection and control systems additional requirements were introduced:

- Software qualification according to the international standard IEC 62138;
- EMC (*e*lectro*m*agnetic *c*ompatibility) qualification according to IEC 62003;
- Lifetime management for 30 years.

This paper shows the increase of qualification for fire detection and control systems in nuclear power plants over the last decades as well as the qualification approach today.

INTRODUCTION

This document presents the requirements for qualification of fire detection and control systems and the relevant process implemented at Framatome GmbH.

The fire detection and alarm systems approved according to industrial standards and regulations are highly qualified regarding high MTBF (*m*ean *t*ime *b*etween *f*ailures) and high resistance to false alarms. Currently, for new nuclear power plants (NPPs) as well as for the modernization of ones several additional qualifications are required by owners and authorities.

The latest requirements for the qualification of a fire detection and control system concern the defined safety function. The relevant qualifications are accordingly:

- Software qualification;
- Cyber security;
- EMC (electromagnetic compatibility) qualification;
- Thermal and ionizing radiation ageing;
- Seismic qualification;
- Lifetime management.

Framatome GmbH is not a manufacturer of fire detection and alarm systems. However, Framatome GmbH is acting as an engineering entity and system integrator for fire detection and control systems for NPPs. Framatome GmbH supports manufacturer regarding the qualification of the entire system or of single equipment. In some cases, Framatome GmbH also provides fire control systems, e.g. for the control of fire dampers or of smoke removal systems.

SAFETY FUNCTION

The safety function refers to nuclear safety, not to safety of personnel or investment. Even if personnel and investment safety is the main objective of any fire detection and control system, the nuclear safety function is not a standard function of such system. The requirement for nuclear safety functions originates from the safety I&C (*i*nstrumentation and *c*ontrol) via safety HVAC (*h*eating, *v*entilation and *a*ir *c*onditioning) to the fire detection and control system.

Figure 1 shows a type test process as of today, stipulated by KTA 3503 [1] and KTA 3505 [2] for safety I&C with the different steps concerning the fire detection and control system.

For fire detection and control systems three of the steps presented in Figure 1 are not applicable. The reason is that the relevant equipment for the safety functions is not located inside the reactor containment. The necessary safety equipment of the fire detection and control system are located inside the safety trains but not in places of accidental ambient conditions.

It is highly recommended to use existing type tests, if available. The type test shall be verified by the comparison between plant requirements and the existing type test certificates. Due to the short product lifecycle of fire detection and control systems and the open bidder process for the manufacturer the use of existing type test will be really difficult. Depending on the manufacturer of the fire detection and control system, a major release of the software takes place each 12 to 18 month with additional minor updates for the major release in between. And even if the hardware had a product lifecycle of 10 to 20 years, small updates of the hardware will lead to different versions over the years. Regarding the times needed to realize a new built of a nuclear power plant, five to ten years or even longer, and especially to the long time period of the system in operation, up to 30 years, the mentioned short lifecycle of the equipment leads to spare parts, which must be re-qualified as long as they are not of the same version of the original tested version. So a dedicated type test of the equipment and/or system for each project (new built of a nuclear power plant or system replacement in an existing plant) is necessary.

In a first step, the resistance to the normal ambient conditions shall be verified based on the data sheets. Only if the normal ambient conditions of the plant exceed the specified data of the equipment an additional test is necessary.

In a second step, the equipment shall be pre-stressed by thermal and ionizing radiation ageing (if applicable). The pre-stressing due to radiation will only be done for equipment exposed to radiation inside the plant. This test will also be used for the evaluation of the equipment lifetime exposed to radiation in normal plant operation.

The pre-stressed equipment shall then be tested regarding its resistance to induced vibrations in a third step. The EMC and software qualification will be done in parallel to the steps outlined before.

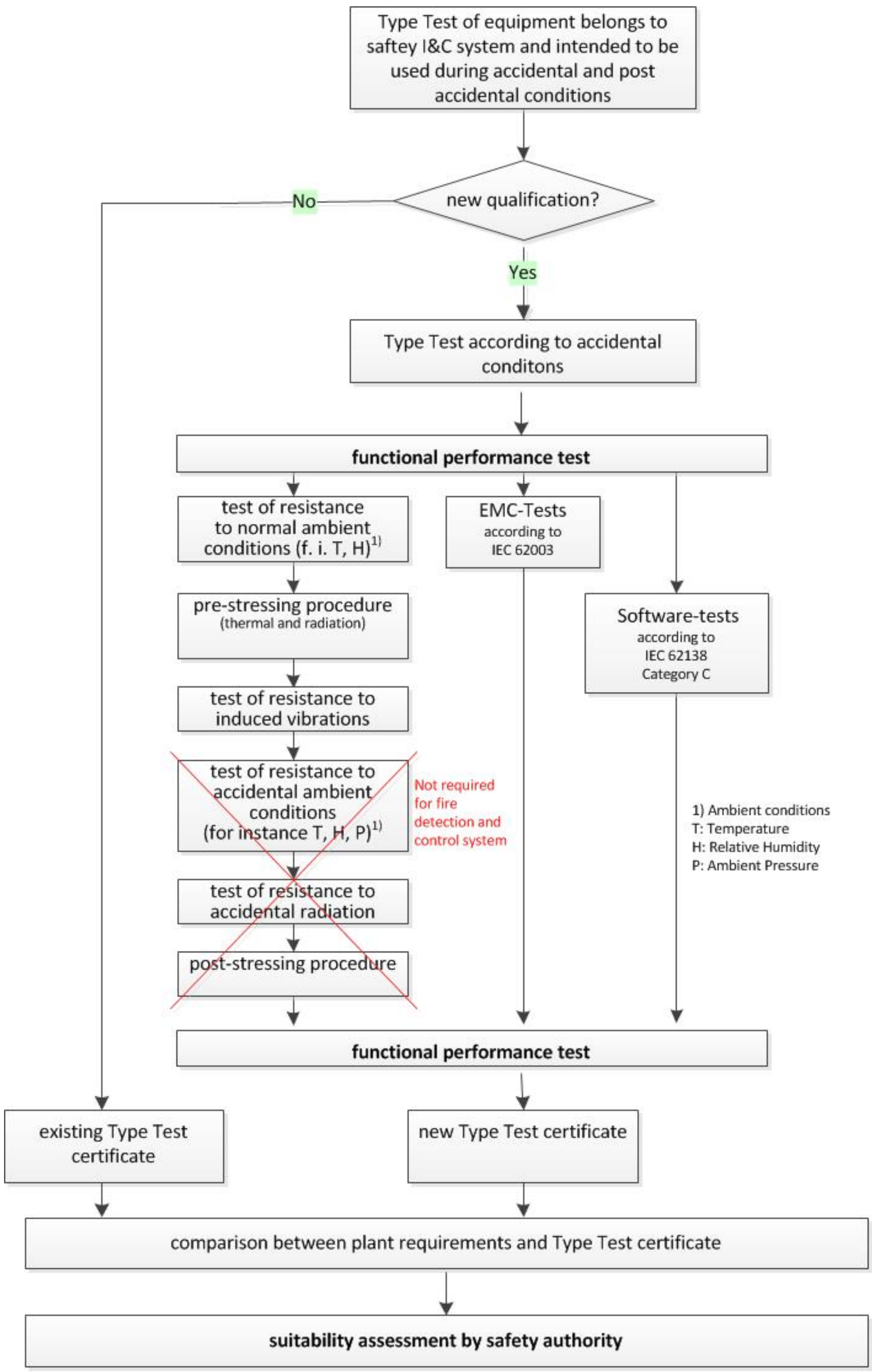

Figure 1 Type test process related to safety I&C

Software Qualification

According to IEC 61226 [3], the fire detection and control system shall be assigned as Category C function *"… functions to monitor and take mitigating action following internal hazards within the NPP design basis (e.g., fire, flood)*".

Even though a software qualification for an already developed industrial product is challenging, Category C is in general the easiest way to perform it. The software qualification for Category C equipment shall be done in accordance with IEC 62138 [4]. It includes the basic programming of the processing units, the firmware on each equipment type (e.g., logic inside of detectors) and the individual, project specific programming of the system.

The software qualification could be divided into three phases:

- Phase 1 – Defining the qualification status of system and requirements;
- Phase 2 – Qualification of the system;
- Phase 3 – Approval of the qualification files.

Phase 1 defines the current qualification status of the system as well as the requirements of the project for the system. In this phase all necessary data of the fire detection and control system are collected, and, in addition, it is evaluated what was already done. The requirements of the project are also described in detail for the fire detection and control system. In the frame of the risk assessment the gap between existing software qualification and the project requirements shall be evaluated.

Within Phase 2 the qualification process itself takes place. The participation of the manufacturer of the fire detection and control system is essential in this phase. All findings identified within the risk assessment of Phase 1 shall be clarified or requalified during this phase. New findings during Phase 2 shall be directly implemented during the qualification process.

Phase 3 starts with the software evaluation report for the fire detection and control system. This report is subject for approval by the customer and the safety authorities.

In Figure 2 the principal sequence of the software qualification process as used for fire detection and control systems is shown.

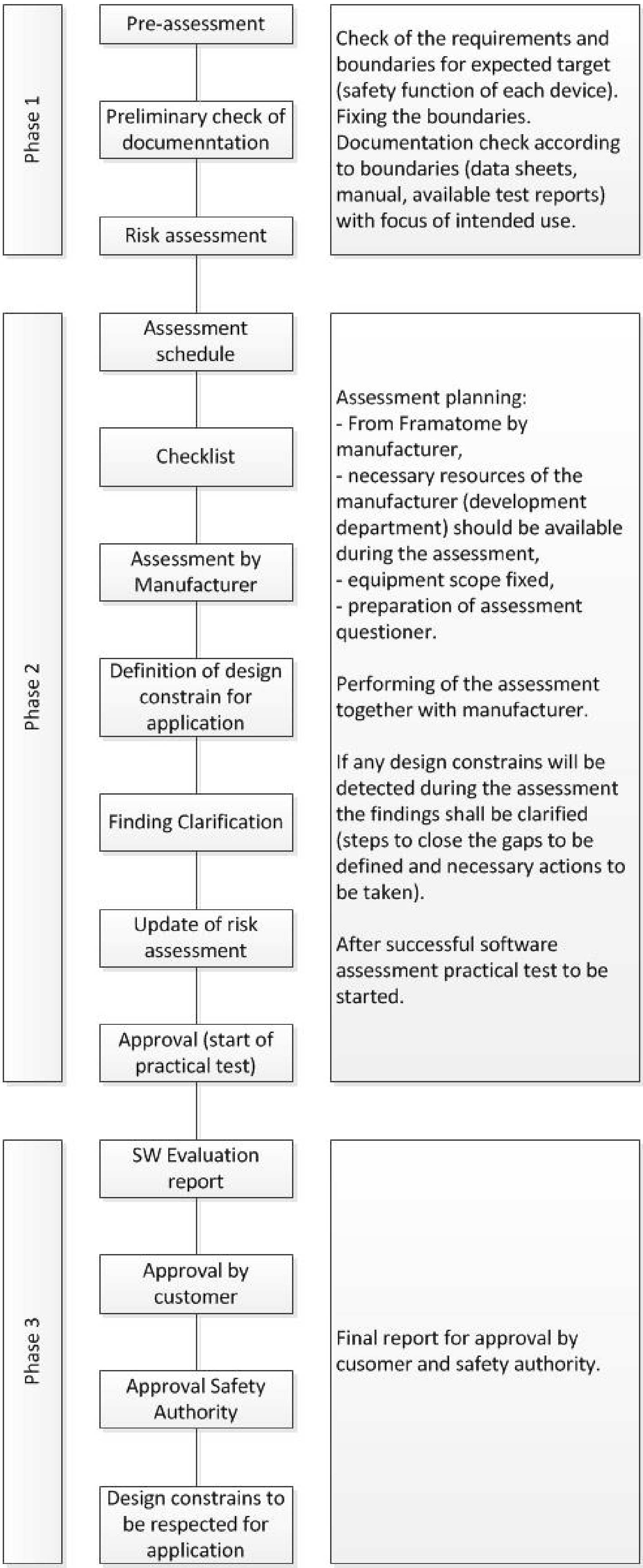

Figure 2 Sequence for software qualification

Cyber Security

In contrary to the software qualification, which ensures the quality of the software used inside the system and each equipment, the cyber security assures the integrity of the system at the interfaces. It shall neither be possible to inadmissibly impair the fire detection and control system via any kind of interface nor other systems of the NPP via an interface.

The fire detection and control system represents an important system with respect to cyber attacks, since a fire alarm could be a potential threat regarding plant security as evacuation of personnel may overrule access control checks and normally closed doors will be opened.

Typical interfaces are plant internal hardwired interfaces to adjacent systems such as operational I&C or the clock system. Plant external interfaces such as a direct connection to the public fire brigade are also possible. Open interfaces like an unused USB (*u*niversal *s*erial *b*us) port need to be secured against unauthorized use as well.

EMC Qualification

Any fire detection and control system should already be approved by the manufacturer regarding EMC. Such approval is normally only done in accordance with the valid industrial standards (IEC 61000-4-3 [5] and IEC 61000-4-4 [6]) and European Directives [7], but not specifically for nuclear power plants.

The standard IEC 62003 “Nuclear power plants - Instrumentation, control and electrical systems important to safety - Requirements for electromagnetic compatibility testing (IEC 45A/1052/CD:2015)” [8] defines the EMC requirements for I&C systems important to safety. The IEC 62003 is significantly more restrictive with respect to the requirements for electromagnetic compatibility than the industrial standards. A new test shall be performed for the equipment of the fire detection and control system to approve the EMC requirements.

Thermal and Ionizing Radiation Ageing

For the ageing qualification two similar processes will be used, thermal and ionizing radiation ageing. The thermal ageing test must always be done as part of qualification. This test is necessary for pre-stressing the equipment before the seismic test will be done.

The ionizing radiation ageing test is only necessary in case of using the equipment in areas exposed to radiation (e.g., reactor building).

Three material groups utilized in equipment of the fire detection and control system are sensitive with respect to the ageing process:

- metals (mainly corrosion),
- polymers, and
- semiconductors.

Metals represent the materials group with less sensitivity to the ageing process. Only thermal ageing including humidity inflicts corrosion effects on metal. Corrosion at electrical conductors has a negative effect on the electrical function of components and equipment.

Polymers are used as a standard carrier plate and as isolation for the electrical components in the equipment of the fire detection and control system. Thus, the stability of the polymers used is important for the functionality of the equipment as well as for electrical safety. In general, polymers are sensitive to thermal and radiation ageing. This however depends on the type of polymer. Some polymers are highly resistant to ionizing radiations but are more sensitive to thermal ageing or vice versa. Particularly with respect to the stability of the polymers, the ageing test should be conducted before the seismic qualification.

Most semiconductors are sensitive to thermal and to ionizing radiation effects. Particularly with respect to the sensitivity of semiconductors to ionizing radiation, any detector type shall be tested if it should be used inside reactor building or other areas exposed to ionizing radiation. Together with the radiation ageing the lifetime of the detectors inside such areas could be defined as well.

Seismic Qualification

For the seismic qualification it is important to understand that the fire detection and control system is one of the few systems in a NPP which could be found in nearly each room. This results in an enveloping seismic spectrum, which covers the seismic spectra of any room. Such covering spectra are considerably higher than a spectrum of an electrical room inside the safeguard buildings.

The seismic test shall use the pre-aged equipment from the thermal ageing qualification. For each component/equipment used in the project at least one piece shall be tested, three pieces are highly recommended in the test.

The functionality of the system shall be verified before the seismic test in order to determine the correct initial point. During the seismic test the system shall be controlled with respect to any failures which could occur. After the seismic test, each function shall be tested again, and the results shall be documented by means of a protocol. Any electrical or mechanical defects shall be evaluated.

Lifetime Management

Modern NPPs are designed for a lifetime of 60 years. Integrated systems such as fire detection and control systems shall be designed for a lifetime of at least 30 years (this does not include wear parts like detectors). Therefore, the fire detection and control system shall be in operation up to the general overhaul of the plant in the middle of its lifetime. This must be taken into account in projects starting five years prior to the start of commercial plant operation. For electronic systems with a typical product lifecycle of four to six years, full system and equipment qualification support is necessary.

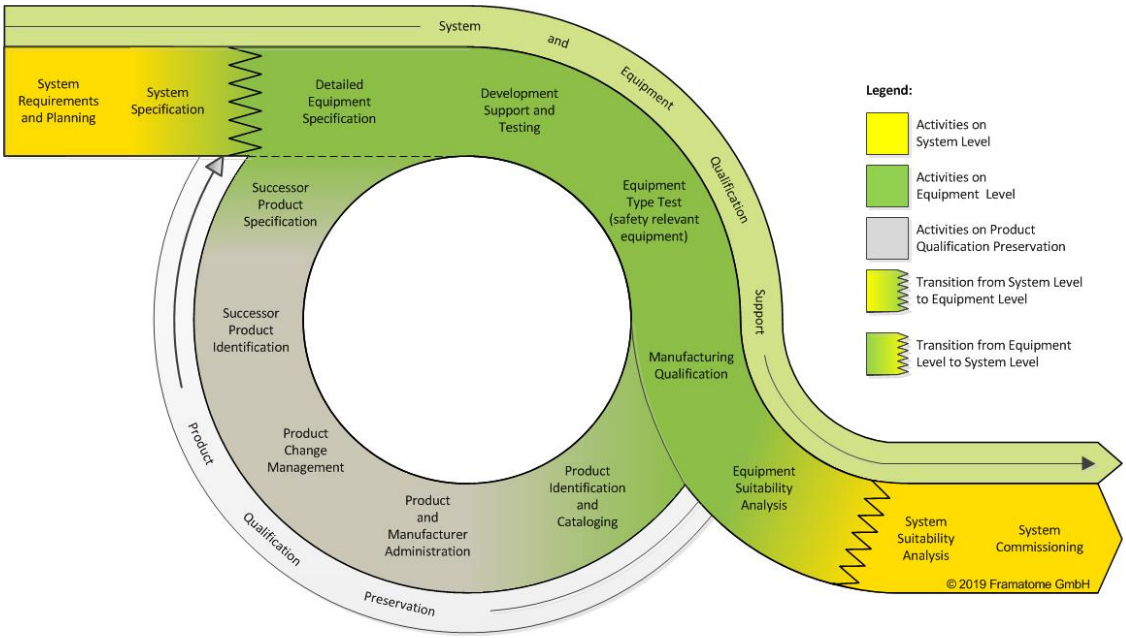

Figure 3 System and equipment qualification support

The lifetime management system for a fire detection and control system includes a well-defined spare part management. This is necessary for approximately 5,000 fire detectors in each power plant unit.

Any equipment which will be replaced by the manufacturer during the system lifetime shall be evaluated for possible successors. The selected successor shall be qualified in the same way as any original equipment of the system before it could be implemented in the system.

As an alternative to the qualification of new equipment the possibilities of equipment refurbishing should be taken into account together with the manufacturer of the fire detection and control system.

CONCLUSIONS

Even a system which like a fire detection and control system is highly qualified by conventional non-nuclear safety standards shall be treated in a nuclear power plant like a safety I&C system regarding its qualification.

The qualification approaches have increased from project to project over the last decades. This has also increased the costs of fire detection and control systems inside nuclear power plants.

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REFERENCES

[1] Nuclear Safety Standards Commission (KTA, German for: Kerntechnischer Ausschuss): KTA 3503 (2015-11), "Type Testing of Electrical Modules for the Instrumentation and Control System Important to Safety", November 2015, http://www.kta-gs.de/e/standards/3500/3503_engl_2015_11.pdf.

[2] Nuclear Safety Standards Commission (KTA, German for: Kerntechnischer Ausschuss): KTA 3505 (2015-11), "Type Testing of Measuring Sensors and Transducers of the Instrumentation and Control System Important to Safety", November 2015, http://www.kta-gs.de/e/standards/3500/3505_engl_2015_11.pdf.

[3] International Electrotechnical Commission (IEC): IEC 61226: Nuclear power plants - Instrumentation and control important to safety - Classification of instrumentation and control functions, 2009.

[4] International Electrotechnical Commission (IEC): IEC 62138: Nuclear power plants - Instrumentation and control systems important to safety - Software aspects for computer-based systems performing category B or C functions, 2018.

[5] International Electrotechnical Commission (IEC): IEC 61000-4-3 and AMD1 and AMD2 CSV Consolidated version, "Electromagnetic compatibility (EMC) - Part 4-3: Testing and measurement techniques - Radiated, radio-frequency, electromagnetic field immunity test", 2006, 2007 and 2010.

[6] International Electrotechnical Commission (IEC):, IEC 61000-4-4 RLV Redline version, Electromagnetic compatibility (EMC) – Part 4-4: Testing and measurement techniques – Electrical fast transient/burst immunity test, 2012:

[7] European Commission (EC): Electromagnetic Compatibility (EMC) Directive 2014/30/EU, "Directive 2014/30/EU of European Parliament and of the Council of 26 February 2014 on the harmonisation of the laws of the Member States relating to electromagnetic compatibility (recast), Brussels, Belgium, 2014.

[8] International Electrotechnical Commission (IEC): IEC 62003: "Nuclear power plants - Instrumentation and control important to safety - Requirements for electromagnetic compatibility testing, 2009.

3.3 Session on “Special Topic: High Energy Arcing Faults (HEAF)”

In the more recent past, the risk significance of high energy arcing fault (HEAF) events with the potential of ensuing fires and/or deterioration of fire protection means, has become an important topic of interest in several countries resulting in various national and international activities for evaluating the corresponding operating experience and performing experimental research and analysis to gain more insights on this type of events for improving nuclear safety. This was the reason to hold one session specifically addressing HEAF induced fires. The three presentations highlighted the most recent insights from the operating experience as well as from national and international research activities which will result in regulatory actions and guidance depending on the risk potential of HEAF events in nuclear installations.

The first two contributions prepared by U.S. NRC and experimental sub-contractors outlined the first key experimental series of HEAF tests carried out within an OECD/NEA experimental program for a better understanding of the phenomena involved and the effects of the high energetic, very rapid pressure and temperature increase causing damage to the components directly involved and potential targets in the near vicinity. One key result was that Aluminum – either implemented in the component itself or as part of the enclosure strongly contributes to the severity of this type of events due to creating conductive compounds resulting in electrical failures. The second presentation focussed on the second phase of the U.S. and international testing program meanwhile initiated based on results from a Phenomena Identification and Ranking Table (PIRT) of the test results having identified a list of important phenomena needing further in-depth investigations.

The third contribution to this session presented by the Dutch regulatory body ANVS provided a short overview on the international experience with HEAF induced fires and the resulting participation of the Netherlands and Germany in the HEAF, Phase 2 experimental program of the OECD/NEA by electrical breakers in cabinets as typically installed in European nuclear power plants.

The corresponding Seminar contributions are provided hereafter.

U.S. NRC Pre-Generic Issue (GI) 0-18 High Energy Arcing Faults (HEAF) Involving Aluminum

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ABSTRACT

The first phase of the Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) High Energy Arcing Faults (HEAF) test program identified a potential issue exists for plants having electrical equipment containing components made of aluminum in areas subject to HEAF conditions. A HEAF event involving aluminum may cause more damage to structures, systems, and components (SSCs) than previous analyses indicated. The insights from this testing were documented in the Nuclear Safety Report NEA/CSNI/R(2017)7 "*Report on the Testing Phase (2014-2016) of the High Energy Arcing Fault Events (HEAF) Project*".

The United States Nuclear Regulatory Commission (NRC) performed an informal survey of the member countries sponsoring the first phase of testing and discovered that U.S. nuclear power plants (NPPs) may be unique in the amount of aluminum used in electrical components. In the spring of 2016 the NRC entered the HEAF involving Aluminum into the agency's Generic Safety Issue (GI) Program. It was determined that there was no immediate safety concern to the operating NPP; however, additional work was needed to determine if there was reasonable assurance of adequate protection from HEAFs involving aluminum at the operating NPP units. The NRC assigned this issue as Pre-GI-018.

To address this issue, the NRC developed a series of tests to better understand aluminum materials involved in HEAFs. This testing is separate from, but complementary to the OECD/NEA HEAF testing programs. The NRC also formed an expert working group with the Electric Power and Research Institute (EPRI) to develop the experiments and evaluate the hazard.

This paper will status the issue in the GI process, highlight the key experiments of the testing and discuss the interactions of the working group.

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. These types of high-energy electrical devices are subject to high-energy arcing faults (HEAFs). This fault mode leads to the rapid release of electrical energy in the form of heat, vaporized copper, and mechanical force. While these events are infrequent, the hazard they pose and the potential consequences to plant safety are not well understood. In the early 2000s, keystone documents such as an operating experience assessment [1] and risk modeling of HEAFs [2] helped improving the state-of-knowledge. However, both documents were limited to the past and drew minimal conclusions based on the physics.

Working under the OECD/NEA FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) Database Project, international experts were able to exchange and review plant-operating experience from ten member countries [3]. From this review, the experts concluded that a non-negligible number of events involved HEAFs. Based on these results, the group recommended a testing program to further evaluate and understand the hazards of HEAFs.

From 2014 to 2016, a series of 26 tests were performed by the United States NRC Office of Nuclear Regulatory Research (RES) under an OECD/NEA agreement. The results are presented in a report issued in 2017 [4]. The report concluded that

"…experiments where aluminum was consumed during the HEAF resulted in more severe physical damage to equipment than those involving only copper and steel at any voltage level. In both experiments where aluminum was consumed during the HEAF, measurement devices were damaged, or the maximum measuring range was exceeded. These instruments were unable to measure the actual maximum temperature and heat flux." …

"*HEAF events involving aluminum were seen to produce a conductive aluminum compound that coated the test facility causing short circuits and unintended current paths in electrical systems.*"

These results prompted the U.S. NRC to consider the impacts on plant safety for those components containing aluminum. Based on information available to the staff, the issue was proposed as a potential safety issue as a Generic Issue in May 2016 [5]. In July 2017, an NRC Generic Issue Review Panel (GIRP) provided its screening evaluation of the issue [6], concluded that the issue met all seven criteria, and recommended that the proposed issue proceed into the Assessment Stage. Under the Assessment Stage, the NRC developed an assessment plan [7] and a subsequent update [8] to identify and describe the steps necessary for the staff to assess the risk resulting from the influence of aluminum HEAF. At the completion of the assessment stage, the assessment team will recommend that the issue either be transferred to the regulatory office for implementation or be removed from the Generic Issues Program.

GENERIC ISSUE PROGRAM OVERVIEW

U.S. NRC Management Directive (MD) 6.4 "*Generic Issues Program*" delineates the process for handling GIs) The GI Program was born in the late 1970s out of Commission and congressional (Public Law 95-209) direction. The program has evolved throughout the years and supports the agency objectives through timely and effective treatment of GIs.

A GI is defined as a well-defined, discrete, radiological safety, security, or environmental matter of which risk significance can be adequately determined and which

(1) applies to two or more facilities and/or licensees/certificate holders, or holders of other regulatory approvals (including design certification rules),

(2) may affect public health and safety, the common defense and security, or the environment,

(3) is not already being processed under an existing program or process,

(4) cannot be readily addressed through other regulatory programs and processes, existing regulations, policies, guidance, or voluntary industry initiatives, and

(5) can be resolved by new or revised regulation, policy, or guidance or voluntary industry initiatives.

A GI may lead to regulatory changes [9].

The program itself consists of three stages: Screening, Assessment, and Regulatory Office Implementation. During the first two stages, the staff determines if more information is needed, if the issue should proceed to the next stage, or if the issue should exit the GI Program. When an issue exits the GI Program, the possible outcomes include no action, further study, or referral to the appropriate regulatory program.

Screening Stage

The purpose of this stage is to evaluate the proposed GI against the seven screening criteria to determine if the issue should proceed to the assessment stage or if the issue should exit the GI Program. A GIRP was formed on July 6, 2019, and its responsibility was to evaluate whether the proposed GI met the seven screening criteria in MD 6.4. The following summarizes the screening criteria, the GIRP's conclusion that all seven criteria were met, and its recommendation that the issue proceed to the assessment stage.

Criterion 1:
The issue affects public health and safety, the common defense and security, or the environment. For issues that are not amenable to quantification using risk assessment, qualitative factors may be developed and applied as necessary to assess safety/risk significance.

Criterion 1 Evaluation:
Based on HEAF test results, existing analytical models supporting plant- specific fire safety analyses may be non-conservative for those plants having aluminum components where HEAFs are postulated to occur. Based on test results, the staff has concluded that plants having aluminum components in areas where HEAFs are postulated to occur could experience potentially larger damage areas than the currently analyzed zone of influence (ZOI). Therefore, the GIRP found that sufficient evidence exists to substantiate a potential increase in risk for those plants having aluminum components in electrical equipment in areas where HEAFs are postulated to occur.

Criterion 2:
The issue applies to two or more facilities and/or licensees/certificate holders, or holders of other regulatory approvals.

Criterion 2 Evaluation:
Based on a literature and operating experience review by the staff and an informal survey provided by the Nuclear Energy Institute (NEI) [10], the staff concluded that multiple plants have aluminum material used in electrical distribution equipment.

Criterion 3:
The issue is not being addressed using other regulatory programs and processes.

existing regulations, policies, or guidance.

Criterion 3 Evaluation:
No NRC activities specifically address the aluminum aspect of the issue. While a follow-on Phase 2 of the international HEAF testing program exists, evaluation of aluminum is not a specific objective of that program.

Criterion 4:
The issue can be resolved by new or revised regulation, policy, or guidance.

Criterion 4 Evaluation:
Because the presence of aluminum in and around electrical equipment, where HEAFs are postulated to occur can increase the ZOI, the staff must review and possibly change existing regulations or guidance to ensure adequate safety is being maintained. However, additional information will be required to support a basis for any such revision.

Criterion 5:
The issue's risk or safety significance can be adequately determined in a timely manner (i.e., it does not involve phenomena or other uncertainties that would require long-term study and/or experimental research to establish the risk or safety significance).

Criterion 5 Evaluation:
The staff believes the issue can be resolved in a timely manner and identified a step-by-

step process for resolution. These steps include determine an extent of condition, develop and use empirical data to understand how the ZOI changes, and then estimate the change in risk due to the presence of aluminum.

Criterion 6:
The issue is well defined, discrete, and technical.

Criterion 6 Evaluation:
HEAF events are defined as highly energetic or explosive electrical faults characterized by a rapid release of energy in the form of heat, light, and pressure due to high current arcs between energized electrical conductors or between energized electrical components and neutral or ground. The high temperatures associated with the HEAF vaporize the aluminum particulates, causing a significantly larger release of energy and materials. The pressure wave produced during HEAF events may cause ejection of plasma, vaporized metal, and projectiles from the electrical component of origin and cause additional fires and equipment damage in both the originating electrical equipment and in any nearby external exposed combustibles.

Criterion 7:
Resolution of the issue may involve review, analysis, or action by the affected licensees, certificate holders, or holders of other regulatory approvals.

Criterion 7 Evaluation:
The location and amount of aluminum components being used in electrical equipment in existing nuclear facilities needs to be known to assess whether a HEAF event with aluminum could adversely impact current plants. Due to the increased ZOI, affected licensees would need to re-evaluate their current plant configurations to determine if any additional critical components will be damaged during a HEAF event and the significance of this damage. The NRC staff would need to review the licensee's evaluations of any affected areas of the plants to ensure adequate safety margin is being maintained. In addition, the NRC staff would evaluate whether changes to existing regulations and guidance are necessary.

Assessment Stage

The purpose of this stage is to develop an assessment of the proposed GI to determine if it merits further regulatory action. The assessment of the issue in the assessment stage includes an evaluation of risk significance, safety significance, security significance, regulatory compliance, a limited regulatory analysis, and a proposed regulatory path forward.

Following entry into the assessment stage, the assessment team devised an assessment plan [7]. The assessment plan describes the steps necessary for the staff to assess the risk resulting from the influence of aluminum on a HEAF inside a nuclear power plant. Based on the assessment results, the staff should be able to recommend whether the GI should proceed to the Regulatory Office Implementation (ROI) stage of the GI process.

The heart of the assessment plan is the steps and milestones that support the various risk, safety, and regulatory assessments. Table 1 and Figure 1 present the status of these milestones and the timeline, respectively.

Table 1 Assessment timeline and milestones

Activity	Status
Published report on results from Phase 1 testing of HEAF	Complete
Published Phase 2 test plan for public comment	Complete
Published results of the international phenomena identification and ranking table (PIRT) in NUREG-2218 "*An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arcing Faults (HEAFs)*"	Complete
The NRC hosted a 2-day public meeting to discuss the proposed large-scale and small-scale HEAF test plans.	Complete
Conduct small-scale testing at Sandia National Laboratories (SNL)	Complete
Large-scale testing for medium voltage enclosures at KEMA test laboratory	Complete
Initiate EPRI/NRC working group to develop an interim ZOI model (if necessary) based on preliminary test information. This effort does not include developing refined frequencies but should be coordinated with that effort led by EPRI to ensure that the ZOI models correlate to the frequency binning. This interim ZOI will be used for the purposes of the GIRP risk assessment only.	Started Fall 2018
SNL will be developing the hazard modeling, model development, target fragility, and model refinement and validation.	Started Spring 2019
Large-scale testing for low voltage enclosures at KEMA test laboratory	Started Fall 2019
Large-scale testing for medium voltage bus ducts at KEMA test laboratory	Currently on hold
Generator decay curve testing and confirmatory power supply configuration testing (based on public stakeholder interaction; Public Workshop; April 18, 2018, ML18108A210, Public Meeting January 23, ML19046A388, Public Meeting; March 20, 2019, ML19108A420) project activities will be delayed to include data associated with these additional tests.	Planned 2020
NRC staff will begin to work with a voluntary pilot plant(s) to examine the impact on the plant's risk using recommendations from the EPRI/NRC working group.	Planned 2020
Complete the remainder of the large-scale testing at the KEMA facility.	Estimated 2022

In addition, the NRC staff is working very closely with experts from, including consultants with EPRI. From this collaboration, new information and expertise has highlighted important aspects of plant configurations and failure mechanisms that could impact the severity of HEAFs. As such, the NRC is making change to the way and methods used to develop empirical data. Examples include, the testing scenarios representative of generator fed faults and failures on the supply bus of switchgear. This collaboration has also focused on improving HEAF fire ignition frequencies and identified new HEAF operating experience information previously not available for the staff. All of this world class collaboration will undoubtably lead to a more realistic and reliable assessment of plant risk.

Regulatory Office Implementation Stage

The purpose of this stage is to develop and perform the appropriate regulatory actions to implement the resolution of a GI in a timely manner. A determination has yet to be made as to whether Pre-GI-018 will reach this stage.

CONCLUSIONS

The U.S. NRC had identified a potential safety issue and entered the issue into the NRC Generic Issues Program. Following that program, the issue has met all seven screening criteria and proceeded into the assessment phase. In that assessment phase, the staff is diligently working to develop the technical basis to support revisions (if necessary) to the methods in which HEAFs are modeled in risk assessments.

Based on these results, the staff with the support of pilot plants will assess any risk increase. If the risk, safety, and regulatory assessments warrant further action, the Pre-GI-018 may proceed to the regulatory implementation phase as a full generic issue. In that phase, the regulatory office will engage the affected facilities to ensure that the hazard is adequately understood and addressed to ensure the safety of the public.

REFERENCES

[1] United States Nuclear Regulatory Commission (NRC): Operating Experience Assessment Energetic Faults in 4.16 kV to 13.8 kV Switchgear and Bus Ducts that Caused Fires in Nuclear Power Plants 1986-2001, NRC Accession No. ML021290358, Washington, DC, USA, 2002, https://www.nrc.gov/docs/ML0212/ML021290358.pdf.

[2] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 (EPRI 10191989), Washington, DC, and Palo Alto, CA, USA, 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/.

[3] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Project - Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R(2013)6, Paris, France, June 2013, http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf.

[4] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Experimental Results from the International High Energy Arcing Fault (HEAF) Research Program Testing Phase 2014 to 2016, NEA/CSNI/R(2017)7, Paris, France, 2017, http://www.oecd-nea.org/documents/2016/sin/csni-r2017-7.pdf.

[5] United States Nuclear Regulatory Commission (NRC): Memorandum "Submittal of Possible Generic Issue Concerning the Damage Caused by High Energy Arc Faults in Electrical Equipment Containing Aluminum Components", NRC Accession No. ML16126A096, Washington, DC, USA, May 06, 2016, https://www.nrc.gov/docs/ML1612/ML16126A096.pdf.

[6] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Memorandum "Results of Generic Issue Review Panel Screening Evaluation for Proposed Generic Issue Pre-Gi-018, High-Energy Arc Faults Involving Aluminum", NRC Accession No. ML16349A032, Washington DC, USA, July 15, 2017, https://www.nrc.gov/docs/ML1634/ML16349A032.pdf.

[7] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Memorandum "Assessment Plan for Pre-Gi-018, Proposed Generic Issue on High Energy Arc Faults Involving Aluminum", NRC Accession No. ML18172A189, Washington, DC, USA, August 22, 2018, https://www.nrc.gov/docs/ML1817/ML18172A189.pdf.

[8] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Memorandum "Update to The Assessment Plan for Pre-GI-018, Proposed Generic Issue on High Energy Arc Faults Involving Aluminum", NRC Accession No. ML19127A202, Washington, DC, USA, August 2018, https://www.nrc.gov/docs/ML1912/ML19127A202.

[9] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Procedure for Processing Generic Issues, TEC-002, Revision 2, NRC Accession No. ML11242A033, Washington, DC, USA, September 2011, https://www.nrc.gov/docs/ML1124/ML11242A033.pdf.

[10] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Nuclear Energy Institute (NEI),: HEAF Aluminum Survey Summary, NRC Accession No. ML17165A140, Washington, DC, USA, 2017, https://www.nrc.gov/docs/ML1716/ML17165A140.pdf.

OECD/NEA High Energy Arcing Faults (HEAF) Research – Second Phase of Testing

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ABSTRACT

The first phase of Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) High Energy Arcing Faults (HEAF) testing produced a significant amount of data and insights to the HEAF phenomena. The results were documented in the Nuclear Safety Report NEA/CSNI/R(2017)7 "*Report on the Testing Phase (2014-2016) of the High Energy Arching Fault Events (HEAF) Project*" [1]. A key finding from the testing was the energetic failures of electrical components that contained aluminum.

In early 2017, the United States Nuclear Regulatory Commission (NRC) sponsored an International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation (NUREG-2218). The purpose of this PIRT was to develop a comprehensive list of HEAF phenomena and rank their importance. The PIRT exercise aids as a road map for future research. A second phase of OECD/NEA HEAF testing has been proposed and approved by the member countries.

This paper highlights the key experiments of the second phase of testing and discuss the instrumentation measurements and challenges involved in the testing.

INTRODUCTION

International nuclear power plant (NPP) operating experience data clearly show that a significant number of high energy arcing fault (HEAF) events have occurred worldwide in operating plants. A report published by the Organisation for Economic Co-operation and Development, Nuclear Energy Agency (OECD/NEA) in June 2013 [2] documents 48 different HEAF fire events reported by the at that time twelve member countries of the OECD/NEA Fire Incidents Records Exchange (FIRE) Project. This number, which has further increased in recent years, represents approximately 10 % of the significant fire events reported to the FIRE Database.

Although much of the fire physics and fire dynamics is readily understood for the typical NPP fire event, the same cannot be said about the HEAF phenomena. In 2009, an OECD/NEA IAGE Task Group developed a working definition of a "High Energy Arcing Fault (HEAF)" [2].

> *"High Energy Arc Faults (HEAF) are energetic or explosive electrical equipment faults characterized by a rapid release of energy in the form of heat, light, vaporized metal and pressure increase due to high current arcs between energized electrical conductors or between energized electrical components and neutral or ground. HEAF events may also result in projectiles being ejected from the electrical component or enclosure of origin and result in fire.*
> *The energetic fault scenario consists of two distinct phases, each with its own damage characteristics and detection/suppression response and effectiveness.*

- *First phase: short, rapid release of electrical energy which that may result in 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 this energetic phase.*
- *Second phase (i.e., the ensuing fire[s]): is treated similar to other postulated fires within the zone of influence.*
  An arc is a very intense abnormal discharge of electrons between two electrodes that are carrying an electrical current. Since arcing is not usually a desirable occurrence, it is described as an "arcing fault." The arc is created by the flow of electrons through charged particles of gas ions that exist as a result of a vaporization of the conductive material."

Another factor that becomes readily apparent about HEAF events with respect to safe NPP operation is that the HEAF events tend to create challenges that complicate the plant's ability to safely shut down the reactor and maintain it in a safe condition. The electrical disturbance initiating the HEAF often causes loss of essential electrical power and physical damage, while the products of combustion pose significant challenges to the operators and fire brigade members handling the emergency. In the United States, for example, internal fire risk is one of the most dominant hazard contributors for many plants. A preliminary examination of the risk assessment information from ten U.S. NFPA 805 nuclear power plants found that HEAF-initiated scenarios were significant contributors to the overall fire risk. The range of fire risk contributed by HEAF initiated fire scenarios ranged from 1 % to 27 % on a unit basis. The average per unit risk contribution was approximately 15 % [3].

Two full-scale HEAF research programs related to the hazards posed by HEAF events in NPP electrical equipment have been recently completed. One sought to understand the HEAF events that occurred at the Onagawa NPP in Onagawa, Miyagi, Japan during the earthquake of March11, 2011 [4]. The second recently completed HEAF research program is Phase 1 of the OECD/NEA CSNI (*C*ommittee on the *S*afety of *N*uclear *I*nstallations) HEAF experimental research program [5]. Both research programs illustrated that more severe physical damage occurred to equipment where aluminum was consumed during the HEAF.

LESSONS LEARNED FROM THE HEAF, PHASE 1 PROJECT

One of the major outcomes of the OECD/NEA HEAF, Phase 1 Project testing was the unexpected impact of aluminum conductors on the amount of energy released from the faulted equipment. Specifically, test 23 and 26 resulted in large releases of thermal energy that overwhelmed the instrumentation and damaged test facility equipment. These tests also generated particulate matter that is suspected to have caused errant conduction paths in the test facility's equipment.

During the post-test analysis, the NRC and its contractors identified several modifications and improvements that would enhance future testing and data collection. One such modification was the addition of tungsten slug calorimeters to the instrumentation array. The thermal energy released in test 23 damaged the instrument racks, melted Inconel thermocouples, and exceeded the range of the calorimetry equipment. To adequately capture the high heat fluxes generated when aluminum is present, the National Institute of Standards and Technology (NIST) developed and validated a tungsten slug calorimeter that can withstand that thermal environment.

The particulate matter that was present after tests 23 and 26 offered another opportunity to collect valuable data that was not considered in Phase 1. Sandia National Laboratories (SNL) provided carbon tape and aerogel collection instruments that could capture the particulate matter for spectroscopic analysis. This analysis would provide information about particle composition, size distribution, and conductivity.

Another lesson learned from HEAF, Phase 1 testing concerned the instrumentation rack and cable routing. Plasma, ionized gases, and high heat fluxes resulted in melted instruments, severed cables, and data loss. To prevent this, the instrumentation racks were redesigned with steel channels through which the cables were routed. A glass-reinforced thermoset polymer (GPO3) was also added to protect the cable runs to the data acquisition equipment.

During the NRC's review of the Phase 1 data, it became apparent that videographic data were extremely valuable in understanding the behavior of the arc and response of the test equipment. During Phase 1, videographic data were collected only with commercially available cameras. To improve the type and quality of data, the NRC contracted with SNL to provide several advanced videographic data collection techniques. An array of high-speed, high-dynamic-range cameras and infrared cameras were added to the instrumentation array for HEAF, Phase 2 testing. In addition, SNL provided digital-image-processing capabilities including particle trajectory and velocity tracking.

ALUMINUM HEAF NRC GENERIC ISSUE PROCESS

The staff in the Office of Nuclear Regulatory Research (RES) proposed the potential safety concern regarding HEAFs involving aluminum as a Generic Issue (GI) in a letter dated May 6, 2016 (ML16126A096) [6]. A GI is a well-defined, discrete, technical or security issue, the risk/or safety significance of which can be adequately determined. A GI (1) applies to two or more facilities and/or licensees/certificate holders, or holders of other regulatory approvals (including design certification rules), (2) affects public health and safety, the common defense and security, or the environment, (3) is not already being processed under an existing program or process, and (4) can be resolved by new or revised regulation, policy, or guidance or voluntary industry initiatives. A GI issue may lead to regulatory changes that either enhance safety or reduce unnecessary regulatory burden [7].

The NRC's process for resolving GIs is described in Management Directive (MD) 6.4, "*Generic Issues Program*". It includes five distinct stages that may be exercised: Identification, Acceptance Review, Screening, Safety/Risk Assessment, and Regulatory Assessment. During each stage, the staff determines if the issue needs more information, if it proceeds to the next stage, or if it should exit the GI process. When issues exit the GI process, the possible outcomes include: no action, further research, transfer to appropriate regulatory programs, or possible industry initiative. In any case, the GI process provides feedback to the person proposing the GI of the outcome at each stage. Issues that proceed through all five stages result in regulatory solutions being provided to program offices for implementation and verification.

The Generic Issue Review Panel (GIRP) completed its screening evaluation for proposed Generic Issue Pre-GI-018 "*High Energy Arc Faults (HEAFs) Involving Aluminum*" and concluded that the proposed issue met all seven screening criteria outlined in Management Directive (MD) 6.4, "*Generic Issues Program*". Therefore, the GIRP recommended that this issue continue into the Assessment Stage of the GI program. The GIRP has completed an assessment plan issued August 23, 2018 (ML18172A185) [8], and a revised assessment plan was issued on July 10, 2019 (ML19127A205) [9]. The assessment plan contains the major milestones and timelines identified for resolving this potential GI. This assessment plan will be updated as needed based on the ongoing testing as well as identified needs of the assessment team.

HEAF PHENOMENA IDENTIFICATION RANKING TABLE

In February 2017, the NRC hosted an international Phenomena Identification and Ranking Table (PIRT) exercise [10]. A PIRT is a process designed to poll subject matter experts on phenomena and parameters of importance on a given topic. This information can be used in the development of a "roadmap" for future research and allows for an informed focusing of resources for research and regulatory entities.

The expert panel was presented with a series of specific HEAF scenarios, each based on the types of scenarios typically encountered in nuclear power plant (NPP) applications. For each scenario, a specific figure of merit was defined (i.e., a specific goal to be achieved in analyzing the scenario). The panel identifies all those related phenomena that are of potential interest to an assessment of the scenario via probabilistic risk assessment (PRA) tools and methods. The phenomena are then ranked relative to their importance in predicting the figure of merit. Each phenomenon is then further ranked for the existing state of knowledge with respect to the ability of existing tools and methods to predict that phenomena, the underlying base of knowledge associated with the phenomena, and the potential for developing new data to support improvements to the existing tools. The PIRT panel covered three distinct HEAF scenarios.

- **Scenario 1** – HEAF occurring in an electrical enclosure with a cable tray passing over the enclosure

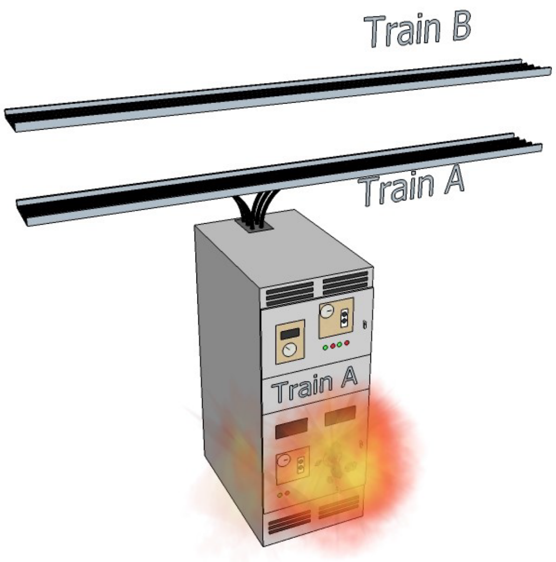


Figure 1 HEAF PIRT scenario 1

* **Scenario 2** – HEAF occurring in a bus duct passing over an electrical enclosure

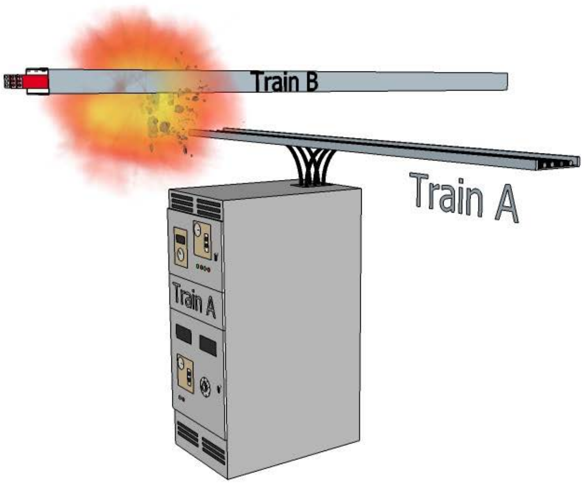


Figure 4 HEAF PIRT scenario 2

* **Scenario 3** – HEAF occurring in an electrical enclosure situated in a bank of similar enclosures´

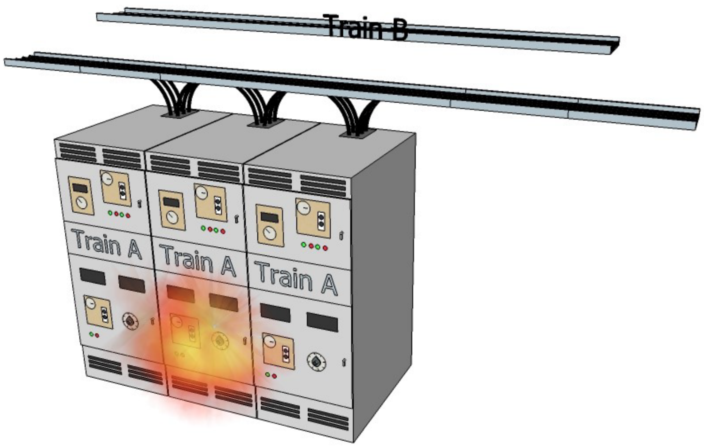


Figure 3 HEAF PIRT scenario 3

As a result of the PIRT process, "level one" phenomena were identified. The level one phenomena are those that were ranked with high importance and low state of knowledge. These would nominally represent potential research priorities. The level one phenomena identified by the panel included the following:

* Electrical arc characterization: thermal and magnetic effects of the arc, arc ejecta (smoke, ionized gas, conductive particulate), arc location, and migration;

- Pressure effects: mechanical shock, projectile impact, and degradation of the compartment pressure boundary;
- Arc mitigation: the use of HEAF-resistant equipment, thermal insulation, or "HEAF shields" to minimize damage incurred as a result of a HEAF;
- Target characterization: establishing the sensitivity of target equipment to various failure mechanisms and associated damage criteria;
- Internal ensuing fire: the likelihood, impact, and phenomenology of an enclosure fire ignited by a HEAF event.

DEVELOPING THE HEAF, PHASE 2 TEST MATRIX

The HEAF, Phase 2 test matrix was designed to explore the impact of important parameters on the behavior of a HEAF with consideration given to the types of information available to probabilistic risk assessment (PRA) practitioners. From the available literature, PIRT exercise, and public feedback, the major variables expected to have a first order influence on the energy output of a HEAF are voltage, current, duration, and material (aluminum).

Accordingly, the test matrix is designed to isolate the impact of these experimental variables. Each parameter is varied for a subset of tests providing a number of direct comparisons from which the sensitivity of the energy output to each experimental variable can be discerned.

The specific voltage, current, and durations specified in the test matrix were determined from a survey of United States and international operating experience and the capabilities of the test laboratory's short circuit generator. In April 2018, the NRC held a public workshop to solicit feedback from industry stakeholders on the types of equipment and test parameters selected for testing [11]. As a result of this workshop, the NRC revised the test plan to more accurately reflect the equipment and configurations in operating plants. The final test matrix is shown in Figures 4 and 5.

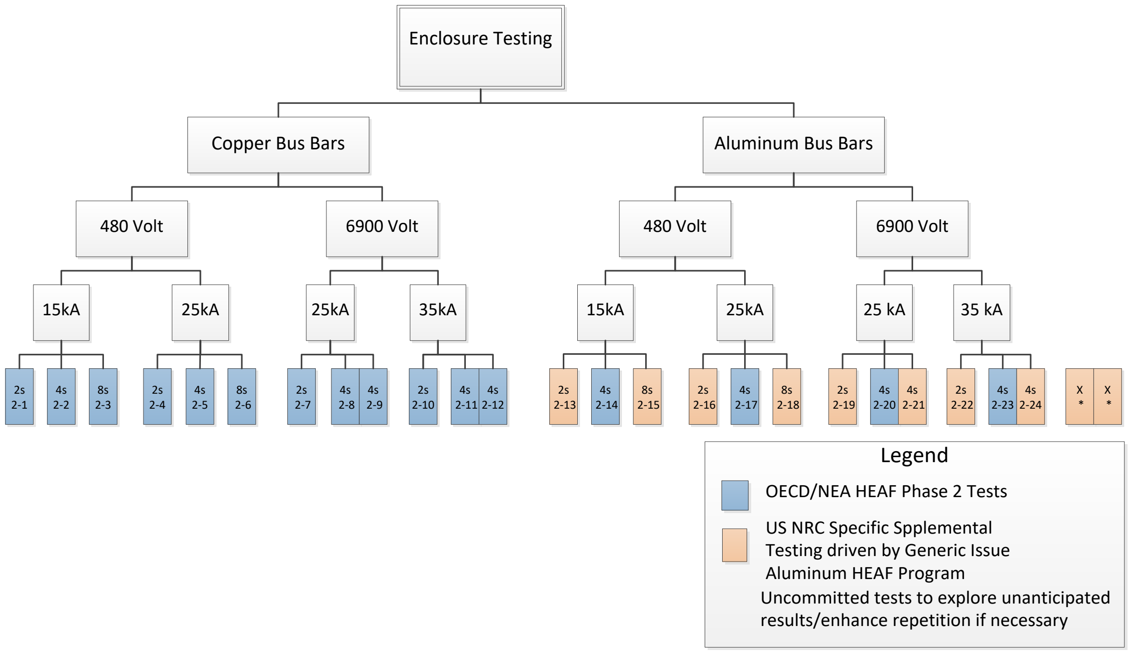


Figure 4 HEAF, Phase 2 electrical enclosure test matrix

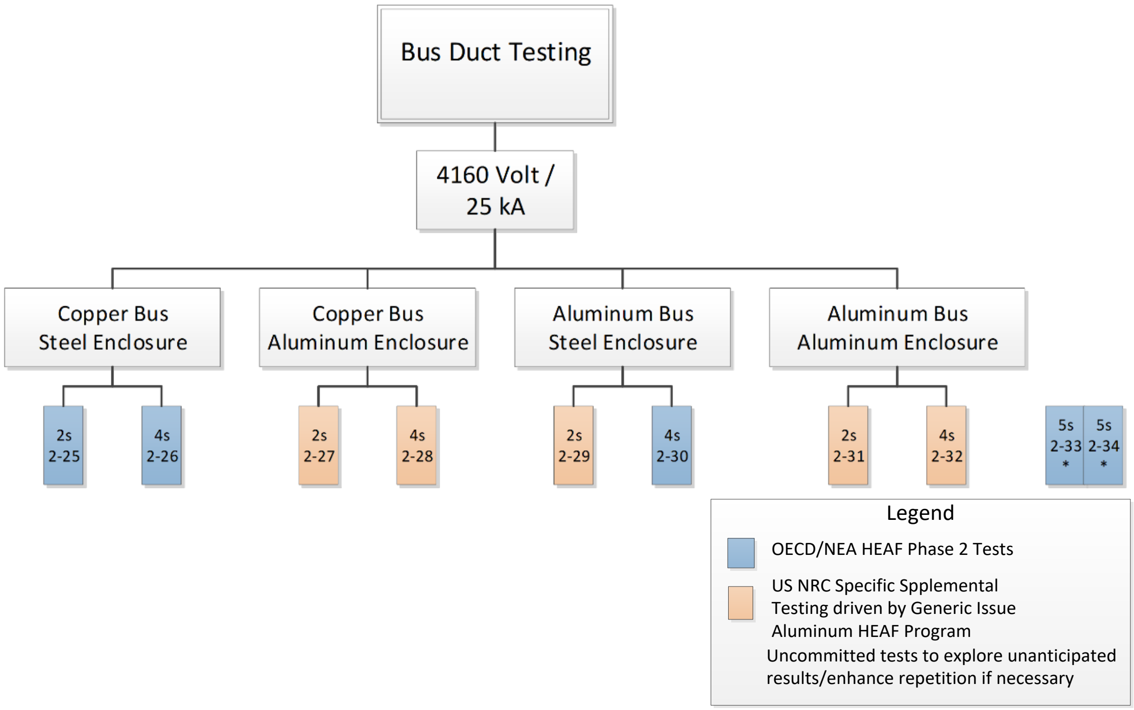


Figure 5 HEAF, Phase 2 bus duct test matrix

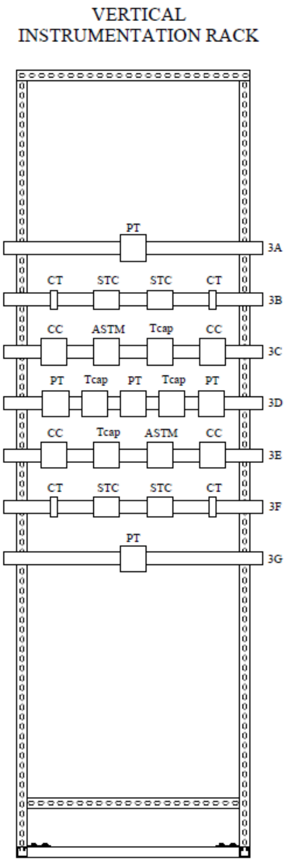
NEW INSTRUMENT ARRANGEMENT FOR HEAF, PHASE 2 TESTING

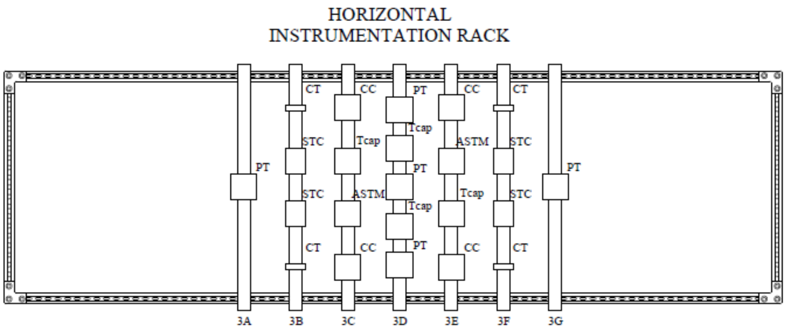
Based on lessons learned from Phase 1 testing, updated data needs, and public feedback, NRC RES staff have updated the instrumentation arrangement for the Phase 2 testing (cf. Figure 6).

Unistrut was used for the instrumentation rack frame to allow for easy placement and adjustment of the instrumentation cluster. The Unistrut also provided a steel channel through which the instrumentation cables were routed. This extra protection served the purpose of insulating the thermocouple wire from the thermal effects of the HEAF event to protect against data loss. In areas where cables were exposed, Kaowool ceramic insulation and glass-reinforced thermoset polymer sheets were used for additional protection.

Each rack consisted of the following instruments:

- Five plate thermocouples to capture heat flux and temperature data;
- Two copper ASTM F1959 slug calorimeters;
- Four tungsten thermal capacitance slug calorimeters to capture high-range heat flux and temperature data;
- Four cable coupon samples for qualitative damage assessment;
- Four carbon tape samples for particle collection and analysis;
- Four silica aerogel samples for particle collection and analysis.





SWGR VERTICAL RACK 1

LOCATION	SENSOR	DAQ CHANNEL
3A	PT	NIST DAQ -1
3B	CT	N/A
3B	STC	N/A
3B	STC	N/A
3B	CT	N/A
3C	CC	N/A
3C	ASTM	KEMA - A
3C	Tcap	NIST DAQ - 2
3C	CC	N/A
3D	PT	NIST DAQ - 3
3D	Tcap	NIST DAQ - 4
3D	PT	NIST DAQ - 5
3D	Tcap	NIST DAQ - 6
3D	PT	NIST DAQ - 7
3E	CC	N/A
3E	Tcap	NIST DAQ - 8
3E	ASTM	KEMA - B
3E	CC	N/A
3F	CT	N/A
3F	STC	N/A
3F	STC	N/A
3F	CT	N/A
3G	PT	NIST DAQ - 9

SWGR VERTICAL RACK 2

LOCATION	SENSOR	DAQ CHANNEL
3A	PT	NIST DAQ -10
3B	CT	N/A
3B	STC	N/A
3B	STC	N/A
3B	CT	N/A
3C	CC	N/A
3C	ASTM	KEMA - C
3C	Tcap	NIST DAQ - 11
3C	CC	N/A
3D	PT	NIST DAQ - 12
3D	Tcap	NIST DAQ - 13
3D	PT	NIST DAQ - 14
3D	Tcap	NIST DAQ - 15
3D	PT	NIST DAQ - 16
3E	CC	N/A
3E	Tcap	NIST DAQ - 17
3E	ASTM	KEMA - D
3E	CC	N/A
3F	CT	N/A
3F	STC	N/A
3F	STC	N/A
3F	CT	N/A
3G	PT	NIST DAQ - 18

SWGR VERTICAL RACK 3

LOCATION	SENSOR	DAQ CHANNEL
3A	PT	NIST DAQ -19
3B	CT	N/A
3B	STC	N/A
3B	STC	N/A
3B	CT	N/A
3C	CC	N/A
3C	ASTM	KEMA - E
3C	Tcap	NIST DAQ - 20
3C	CC	N/A
3D	PT	NIST DAQ - 21
3D	Tcap	NIST DAQ - 22
3D	PT	NIST DAQ - 23
3D	Tcap	NIST DAQ - 24
3D	PT	NIST DAQ - 25
3E	CC	N/A
3E	Tcap	NIST DAQ - 26
3E	ASTM	KEMA - F
3E	CC	N/A
3F	CT	N/A
3F	STC	N/A
3F	STC	N/A
3F	CT	N/A
3G	PT	NIST DAQ - 27

Figure 6 Instrumentation design

CURRENT STATUS OF HEAF, PHASE 2

Phase 2 of the HEAF research program comprises two different parts of the test program: (1) an NRC test program with a focus on aluminum to resolve the proposed GI and (2) an OECD/NEA test program to study the broader phenomenology of HEAFs.

The NRC test program with a focus on aluminum comprises several test campaigns. The first campaign, which concerns medium-voltage switchgear enclosures with aluminum conductors, was completed in September 2018. This testing (see also Figure 7) consisted of the tests 2-19, 2-21, 2-22, and 2-24 from the electrical enclosure test matrix in Figure 4.

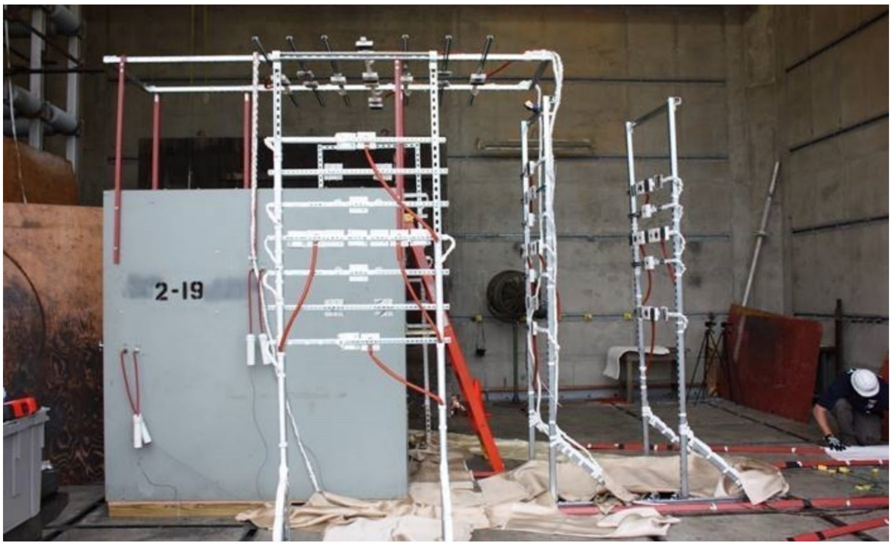


Figure 7 Medium voltage electrical enclosure series

The second campaign, which concerns low voltage switchgear (480 V – 600 V) enclosures (see Figure 8) and medium voltage bus ducts (4160 V), is scheduled for fall 2019.

Figure 8 Low voltage enclosure test series

A third campaign in the NRC's test program is planned for spring 2020. This test campaign investigates the behavior of a generator-fed HEAF as the generator coasts down as well as the differences between switchgear supply and load configurations. These tests have been added to the test matrix as a part of industry and stakeholder comments during the process of public meetings for the GI program.

The broader OECD/NEA test program, which contains the remainder of the tests, is scheduled for fall 2020. This OECD Program will utilize lessons learned from the NRC Generic Issues testing and expand the knowledge base for future HEAF modeling for both copper and aluminum materials.

The OECD tests will explore cabinet to cabinet effects (cf. Figure 9) to investigate propagation between electrical enclosures. Adjacent enclosures will not be electrically energized but will be monitored for thermal and pressure effects. The power supply configuration and arc initiation locations will be configured to best represent operating experience from the OECD/NEA FIRE Database.

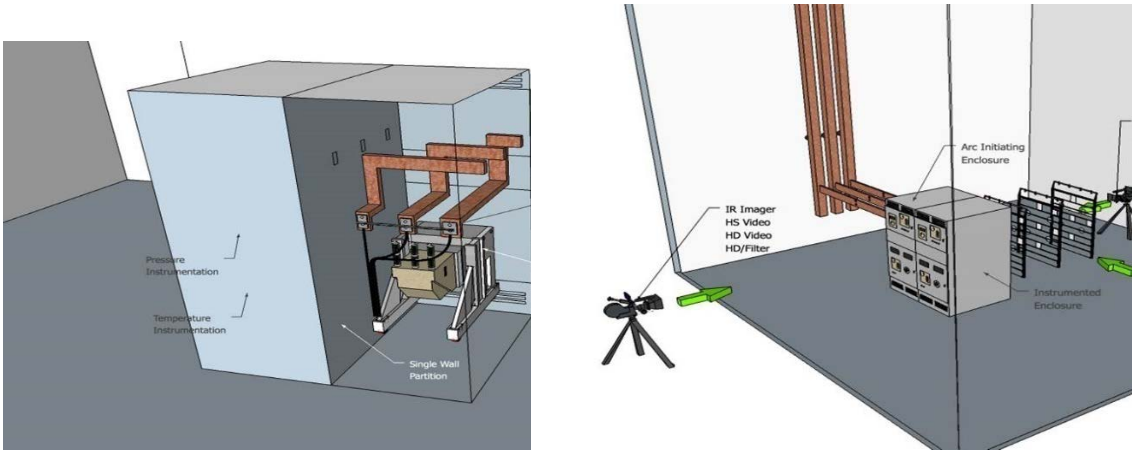


Figure 9 Medium voltage OECD test enclosure to enclosure configuration

Item	2018	2019	2020	2021	2022*
Medium Voltage Enclosures NRC Tests	**4 (completed)**				
Low Voltage Enclosure NRC Tests		**4 (in process)**			
Medium Voltage Bus Ducts NRC Tests		**5 (delayed)**			
NRC Decrement Curve Testing (generator fed faults)			**4**		
OECD Tests Medium Voltage Bus Ducts Tests- 2-25, 2-26, 2-30, 2-33*, 2-34*			**3 - 5**		
OECD Tests Medium Voltage Enclosures Tests- 2-13, 2-16, 2-20, 2-23 Tests-2-7, 2-8, 2-9, 2-10, 2-11, 2-12				**10**	
Low Voltage Enclosures OECD/NEA HEAF 2 Tests Tests- 2-1, 2-2, 2-3, 2-4, 2-5, 2-6					**6**

* Testing delays and/or laboratory time unavailability may affect the proposed timelines. This test program is being coordinated with multiple member countries as well as taking place with robust stakeholder interaction to ensure successful results and useful outcomes

REFERENCES

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Experimental Results from the International High Energy Arcing Fault (HEAF) Research Program Testing Phase 2014 to 2016, NEA/CSNI/R(2017)7, Paris, France, 2017,

http://www.oecd-nea.org/documents/2016/sin/csni-r2017-7.pdf.

[2] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Project - Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R(2013)6, Paris, France, June 2013, http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf.

[3] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Memorandum "Results of Generic Issue Review Panel Screening Evaluation for Proposed Generic Issue Pre-Gi-018, High-Energy Arc Faults Involving Aluminum", NRC Accession No. ML16349A032, Washington DC, USA, July 15, 2017, https://www.nrc.gov/docs/ML1634/ML16349A032.pdf.

[4] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Nuclear Regulatory Authority Experimental Program to Characterize and Understand High Energy Arcing Fault (HEAF) Phenomena, International Agreement Report, prepared by S. Tsuchino, H. Kabashima, Regulatory Standard and Development Department. Secretariat of Nuclear Regulatory Authority (S/NRA/R), Tokyo, Japan 106-8450, S. Turner, Leidos, Inc. 301 Laboratory Road, Oak Ridge, TN 37830, S. Mehta, D. Stroup, N. Melly, G. Taylor, F. Gonzalez, M. H. Salley, NRC Project Manager Division of Risk Analysis Office of Nuclear Regulatory Research, NUREG/IA-0470, Volume 1, Washington, DC, USA, August 2016, https://www.nrc.gov/docs/ML1623/ML16235A163.pdf.

[5] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI), Report on the Testing Phase (2014-2016) of the High Energy Arcing Fault Events (HEAF) Project: Experimental Results from the International Energy Arcing Fault Research Programme, NEA/CSNI/R(2017)7, Paris, France, May 2017, https://www.oecd-nea.org/nsd/docs/2017/csni-r2017-7.pdf.

[6] United States Nuclear Regulatory Commission (NRC): Memorandum "Submittal of Possible Generic Issue Concerning the Damage Caused by High Energy Arc Faults in Electrical Equipment Containing Aluminum Components", NRC Accession No. ML16126A096, Washington DC, USA, May 06, 2016, https://www.nrc.gov/docs/ML1612/ML16126A096.pdf.

[7] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Frequently Asked Questions About Generic Issues Program, October 9, 2018, https://www.nrc.gov/about-nrc/regulatory/gen-issues/faqs.html.

[8] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Memorandum "Assessment Plan for Pre-Gi-018, Proposed Generic Issue on High Energy Arc Faults Involving Aluminum", NRC Accession No. ML18172A189, Washington, DC, USA, August 22, 2018, https://www.nrc.gov/docs/ML1817/ML18172A189.pdf.

[9] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Memorandum “Update to The Assessment Plan for Pre-GI-018, Proposed Generic Issue on High Energy Arc Faults Involving Aluminum”, NRC Accession No. ML19127A202, Washington DC, USA, August 2018, https://adamsweb-search2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML19127A202.

[10] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arc Faults (HEAFs), NUREG-2218, Washington, DC, USA, 2018, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2218/ .

[11] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Proceedings of the Information-Sharing Workshop on High Energy Arcing Faults (HEAFS), NUREG/CP-0311, Washington, DC, USA, July 2019, https://www.nrc.gov/docs/ML1921/ML19212A150.pdf.

Medium Voltage Breakers from Nuclear Power Plants to be Tested Within the OECD Nuclear Energy Agency Experimental Project HEAF 2

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ABSTRACT

Operating experience from nuclear installations has indicated that high energy arcing faults (HEAF) of medium and high voltage (0.4 to 12 kV) electric breakers can induce ensuing fires and may deteriorate fire barrier elements. Under the auspices of the OECD Nuclear Energy Agency (NEA) a first HEAF experimental series has taken place in the more recent past underpinning the observations from the operating experience. In conclusion, fire hazard analysis as well as Fire PSA (*P*robabilistic *S*afety *A*ssessment) should therefore adequately consider such events in assessing the safety of NPPs. Moreover, regulatory guidance addressing HEAF events including induced fires is needed.

For gaining further insights in more realistic HEAF events and consequential fire scenarios in NPPs, a follow-on project, HEAF Phase 2 (HEAF 2), has been recently started. The Dutch Regulatory Body (The Authority for Nuclear Safety and Radiation Protection ("Autoriteit Nucleaire Veiligheid en Stralingsbescherming", ANVS)) is also interested in these experiments since some comparable HEAF endangered breakers are in operation. In the OECD/NEA FIRE Database one HEAF induced fire event is recorded from the Borssele nuclear power plant (NPP), the only operating NPP in the Netherlands. The Netherlands therefore participate in the OECD/NEA HEAF 2, providing together with Germany medium voltage breaker cabinets already used in NPPs for HEAF testing with as far as practicable realistic geometries of two or more cabinets in row.

The paper provides insights from the operating experience, the medium voltage breaker cabinet selection, delivery and use for the intended HEAF testing.

INTRODUCTION

Operating experience from nuclear installations has demonstrated that switchgears, breakers, load centers and bus bars/ducts (typically with nominal voltages of typically 380 V and above) can be subject to a unique failure mode that causes extensive damage, which may affect nuclear safety. In particular, these types of high energy electrical devices are subject to a failure mode known as high energy arcing fault (HEAF). This failure mode causes a rapid release of electrical energy in the form of heat, vaporized metals (e.g., copper and aluminum), plasma, and explosive mechanical force (see also Figure 1).

Observations from the international operating experience of nuclear installations collected in the OECD/NEA Database FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) have indicated that HEAF of medium and high voltage (0.4 to 12 kV) electric breakers can induce ensuing fires and may deteriorate fire barrier elements. In the FIRE Database one HEAF induced fire event is recorded from the Borssele nuclear power plant (NPP).

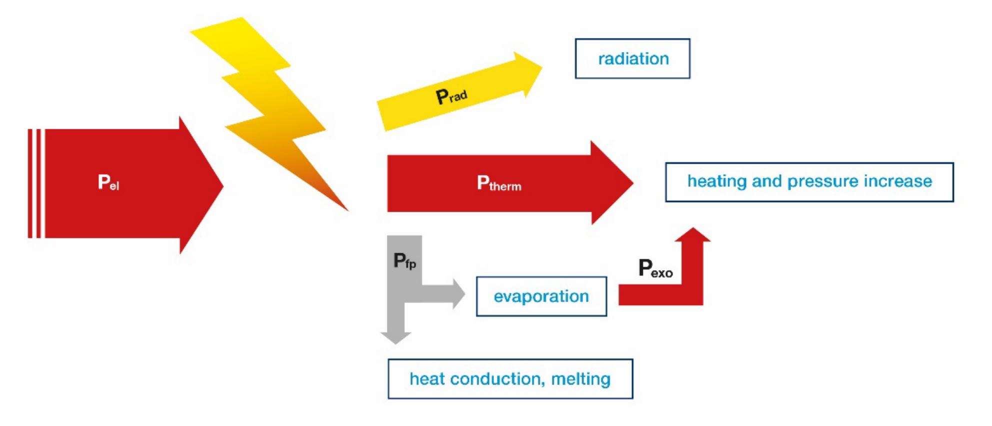


Figure 1 Scheme of the HEAF process with power balance

Under the auspices of the OECD/NEA, a first HEAF experimental series has taken place in the more recent past underpinning the observations from the operating experience. In conclusion, fire hazard analysis (FHA) as well as Fire PSA should therefore adequately consider such events in assessing the safety of NPPs. Moreover, regulatory guidance addressing HEAF and HEAF induced fires is needed.

For gaining further insights in more realistic HEAF events and consequential fire scenarios in NPPs, a follow-on project, HEAF Phase 2 has been recently started. The Dutch Regulatory Body is also interested in these experiments since some comparable HEAF endangered breakers are in operation in the Dutch operating NPP. The Netherlands therefore participate in the OECD/NEA HEAF 2 Project providing, together with Germany, medium voltage breaker cabinets for HEAF testing in an as far as practicable realistic geometry of two or more cabinets in row.

OPERATING EXPERIENCE WITH HEAF EVENTS IN NUCLEAR INSTALLATIONS

HEAF in electrical equipment are principally initiated in one of three ways: poor physical connection between the switchgear and the holding rack, environmental conditions or the introduction of a conductive foreign object (e.g., a tool used during maintenance). A high energy fault scenario typically consists of two distinct phases, each with its own damage characteristics. The first phase is characterized by the short, a rapid release of electrical energy from the arc of typically 100 cal/m^2, which may result in a catastrophic failure of the electrical enclosure, the ejection of hot projectiles from damaged electrical components or housing and/or fire(s) due to the extremely high temperatures of approx. 10,000 °C. Such fires may only involve the electrical device itself or any external combustibles exposed. The second phase, i.e., the ensuing fire typically includes ignition of combustible material within the HEAF zone of influence (ZOI).

As a result of first indications from the FIRE Database Project on the significance of HEAF induced fire events, in 2009 the OECD/NEA initiated an international task on the operating experience with HEAF events in order to provide an in-depth investigation with the main objective to determine HEAF damage mechanisms, extent of areas affected, methods of protecting systems, structures and components (SSC) important to safety, and possible calculation methods for modeling HEAF events as applicable to fire protection in NPPs [1].

This task as well as a task carried out by the FIRE Database Project on HEAF induced fires – as part of the HEAF event or as an event combination of a HEAF and consequential fire event (see the corresponding FIRE Topical Report [2]) – clearly demonstrated that HEAF fire events represent a non-negligible number of events recorded in the FIRE Database. Up to the time being, 62 of the in total 492 statistically countable events representing a share of nearly 13 % are HEAF induced fires. While 28 HEAF fires were limited to the component where the HEAF occurred, 34 HEAF events also contributed remarkably to event combinations with fires (cf. Figure 2).

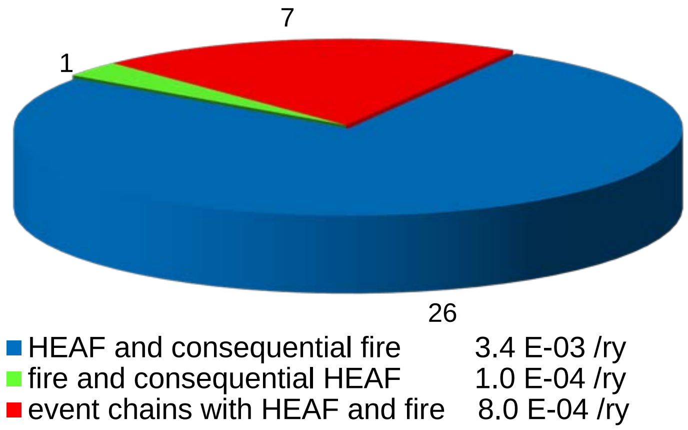


Figure 2 Contributions of different types of event combinations of HEAF and fire events from the most recent version of the OECD/NEA FIRE Database [3]

Five HEAF fire events were induced by another hazard (e.g. one by a hydrological hazard with the water ingress resulting in a longer duration arc, two by seismic events, and another two by plant internal fires), 32 HEAF events caused consequential events, a majority of these (26) internal fires.

Examples can be found in [2] and [4]. Moreover, several HEAF events, as reported from different countries, e.g. Germany (cf. [5]) resulted in a non-negligible deterioration of fire protection features as demonstrated in Figure 3.

Figure 3 Fire door damaged according to a HEAF fire event; photograph from outside the fire compartment, from [2]

The operating experience has also indicated that HEAF fire events do not only significantly contribute to the fire risk at NPPs during power operation, but also during low power and shutdown phases. 44 events recorded in [3] occurred during power operation representing a fire frequency of 6.6 E-03 /ry, 18 events representing a fire frequency of 9.2 E-03 /ry were observed during low power and shutdown phases. The HEAF induced fire frequency during low power and shutdown states is by a factor 1.4 higher than that for reactor units in power operation.

In order to prepare as far as possible realistic HEAF fire experiments, it is also valuable to know from which components the highest contributions to HEAF fire events result. The following Figures 4 and 5 show the shares of HEAF fire events according to the different types of components, where these occurred for reactors during power operation as well as during low power and shutdown phases. The results show that ~ 36 % of the HEAF fires during power operation occur at transformers, ~ 32 % in electrical cabinets, and ~ 12 % at electrical breakers representing the major components to be analyzed.

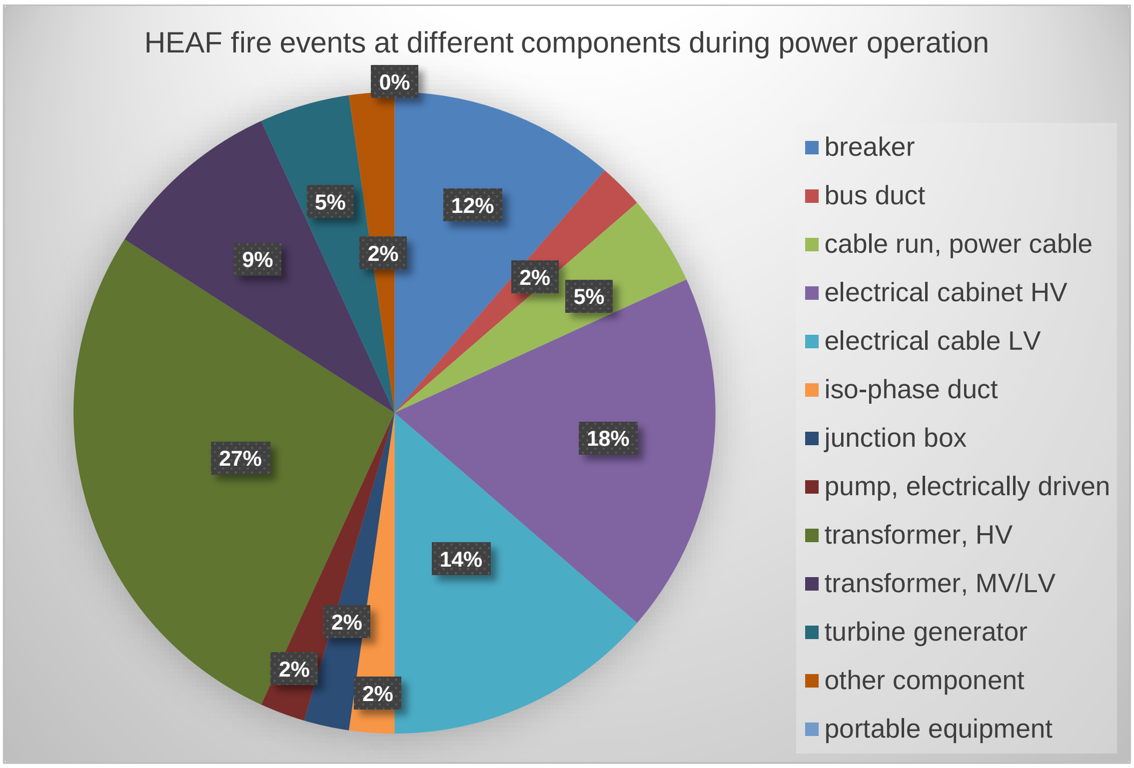


Figure 4 Shares of HEAF fire events for different types of components for reactors during power operation, from [3]

During low power and shutdown states the major components of HEAF initiation are electrical cabinets with ~ 38 %, transformers with ~ 37 %, bus ducts with ~ 22 %, and again breakers with ~ 17 %.

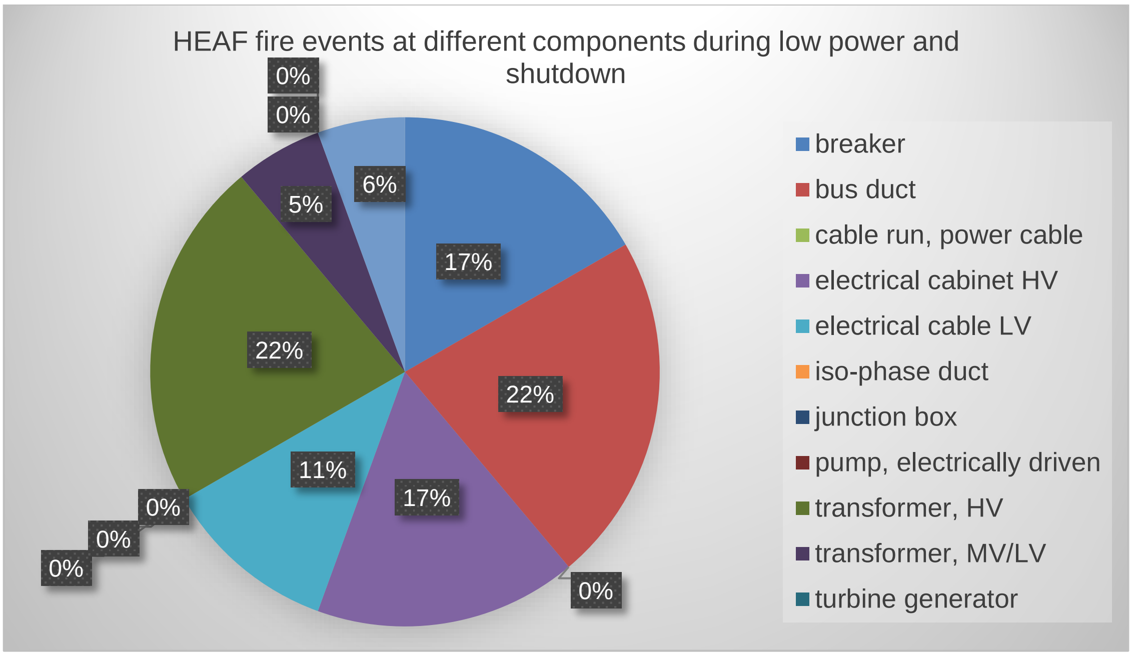


Figure 5 Shares of HEAF fire events for different types of components for reactors during low power and shutdown states, from [3]

The total HEAF fire frequency of 7 E-03 /ry supports the experts´ opinion that HEAF events represent an important contributor to core and fuel damage, and thus more and more realistic full-scale experiments at these components are needed for improving the safety with respect to the internal hazard HEAF.

As a result of the findings from the operating experience with HEAF in nuclear installations worldwide, at least in the United States, various activities are ongoing by the regulatory body NRC (*N*uclear *R*egulatory *C*ommission) preparing Information Notices and further regulatory Generic Letters in order to better consider such events in NPPs. As of August 2017, the Japanese regulation also covers amended HEAF requirements as stated in [6].

Moreover, the potential risk from HEAF has also been taken into account in the update of the IAEA Safety Guide SSG-64 on internal hazards [7].

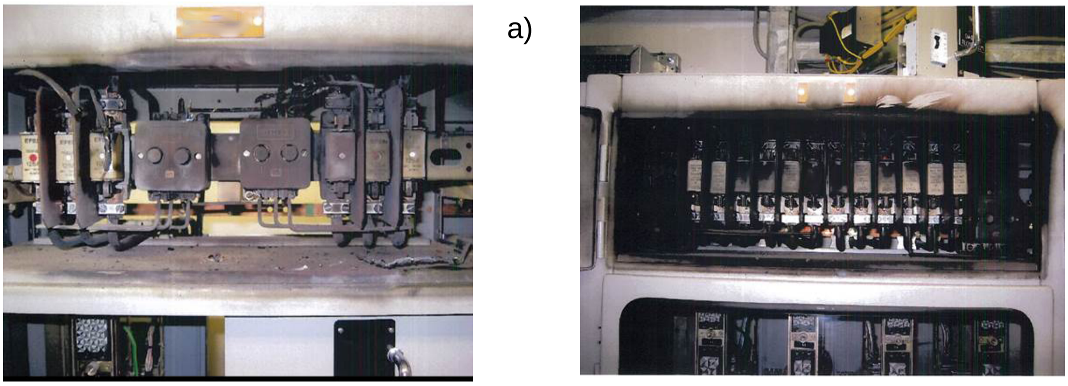
Specific Experience from Germany

12 of the fire events recorded in the OECD/NEA FIRE Database up to the end of 2017 were HEAF induced fires, one of them was an event combination of a HEAF at a transformer with consequential fire, the other one was a HEAF evaporating cables routed underground on a length of more than 1 m. This together with the collection of more than 40 HEAF events from German NPPs (cf. [5]) having not resulted in fires (not recorded in the FIRE Database as these were HEAF only, not HEAF fires) but having the potential of either deterioration of fire protection features or – under other circumstances – causing ensuing fires clearly demonstrate that HEAF events need to be considered in fire and explosion protection of NPPs. This was the main reason that event combinations with HEAF events are meanwhile considered in the German nuclear standards KTA 2101.1 on fire protection [8] and KTA 2103 on explosion protection [9].

Specific Experience from the Netherlands

One Dutch fire event recorded in the OECD/NEA FIRE Database [3] is also a HEAF fire event. On January 29, 2013 during operation at full power an arc flash occurred while performing maintenance activities (including short circuit measurements) at the pump motor of the reactor boron and water makeup system. During the measurement a short to ground occurred. In one compartment a fuse was blown. This should have been detected by a miniature circuit breaker switching off, which triggers an alarm. However, the circuit breaker malfunctioned and did not switch off. Approximately 30 min later, another electric motor in the same compartment was started. Even the starting current of this motor was not enough to switch off the miniature circuit breaker. It failed resulting in a short circuit.

An arc flash damaged the compartment of the cabinet and this arc ran over the rails on the top of the cabinet and damaged also other, non-neighboring compartments (see Figures 6 a and b). The entire cabinet was automatically switched off from the supply side and therefore became unavailable. With this board being unavailable, the technical specifications require the reactor to be in the cold subcritical state. Therefore, the reactor was shut down. The event had no direct consequences for the safety functions.



a) b)

Figure 6 HEAF event at the Borssele NPP, from [10];
a) Failing fuse detection: short circuit (left)
b) secondary damage at the end of the rail (right)

OECD/NEA HEAF EXPERIMENTS

As part of the above mentioned CSNI task on HEAF, an experimental program was carried out by U.S. NRC and their sub-contractors from 2014 to 2016 for investigating HEAF fire phenomena to inform future deterministic and probabilistic assessment methods. This experimental program covered 26 full-scale HEAF experiments. The program involved several challenges, such as suitable and efficient measurement science and techniques or the collection and analysis of the data recorded during the experiments. The experimental project provided "the following major insights (cf. [11]):

- *"All enclosures with components of 4.16 kV and above maintained the arc for a time periods of more than 2 s.*
- *Increased duration arcing events were more likely to create an ensuing fire, experiments ensuing fire were not observed for arc durations less than 2 s.*
- *The most severe electrical enclosure damage was observed as a result of a low voltage (480 V) HEAF in an enclosure with aluminum bus bars.*

- *HEAF events involving aluminum were seen to produce a conductive aluminum compound that coated the test facility causing short circuits and unintended current paths in electrical systems. Moreover, experiments with aluminum being consumed during the HEAF resulted in more severe physical damage to equipment than those involving only copper and steel at any voltage level.*
- *In 480 V enclosures, there was large variability in the ability to create a sustained arcing event. In some instances, the arc would immediately extinguish or not sustain for the full intended duration."*

Based on the results of the OECD/NEA HEAF Project, a phenomena identification and ranking table (PIRT) exercise for NPP HEAF analysis applications was performed by the U.S. NRC Office of Nuclear Regulatory Research by means of a facilitated expert elicitation process in early 2017 (cf. [12]). Main objective of the PIRT was to (1) identify key phenomena for a series of specific HEAF scenarios associated with the intended application, and (2) to rank the current state of knowledge relative to each identified phenomenon. Each scenario included a figure of merit, i.e., a specific goal to be achieved in analyzing the scenario using HEAF analysis or modeling tools. The panel identified phenomena that are of potential interest to an assessment based on the figures of merit. The phenomena identified were then ranked relative to their significance in predicting the figure of merit and further for the existing state of knowledge and the adequacy of existing modeling tools to predict that phenomenon. The PIRT panel covered three HEAF scenarios and identified a number of areas potentially needing further analysis and model development.

The PIRT exercise [12] has been used as basis for the second OECD/NEA HEAF Project phase (HEAF 2), which officially started in October 2018 with the main objective to develop experimental and scientific fire data related to the HEAF phenomenon applicable to NPPs, since the significant energy released during a HEAF event can act as an ignition source for secondary fires involving other components (cf. [13]). The results from this second experimental series will provide more insights on the thermal and mechanical damage posed by HEAF events. The data collected from the experiments shall support updating and advancing methods in order to assess and characterize the risk from HEAF events.

Participation of the Netherlands and Germany with Medium Voltage Breakers in the OECD/NEA HEAF 2 Project

In total nine countries participate in the HEAF 2 Project with the U.S. NRC Office of Research together with their sub-contractors for carrying out the experiments and performing all the measurements acting as Operating Agent (OA). The other eight countries participate either via monetary contributions, or equipment donations (medium voltage electrical enclosures), or a combination of the two. The project officially started in October 2018 and will finish by the end of 2021. A total of more than thirty real scale experiments is foreseen, with a minimum of 24 experiments of busbars in enclosures and 8 experiments at bus ducts.

The Netherlands and Germany participate in the HEAF 2 Project by providing in total 11 breakers in enclosures and the additional monetary value for the removal of these components and their enclosures from the NPP where they were installed and their transport to the test facility in the United States. The components to be tested in the HEAF 2 Project from Germany and the Netherlands are (see also Figures 7 and 8):

- Two high voltage circuit breakers in the original cabinets, type BBC-BA2, nominal design voltage 12 kV, operated at 10 kV nominal voltage (Germany),

– Nine medium voltage circuit breakers in the original cabinets, type BBC-MNS, nominal design voltage 660 V, operated at 480 to 500 V (six from Germany and three from the Netherlands).

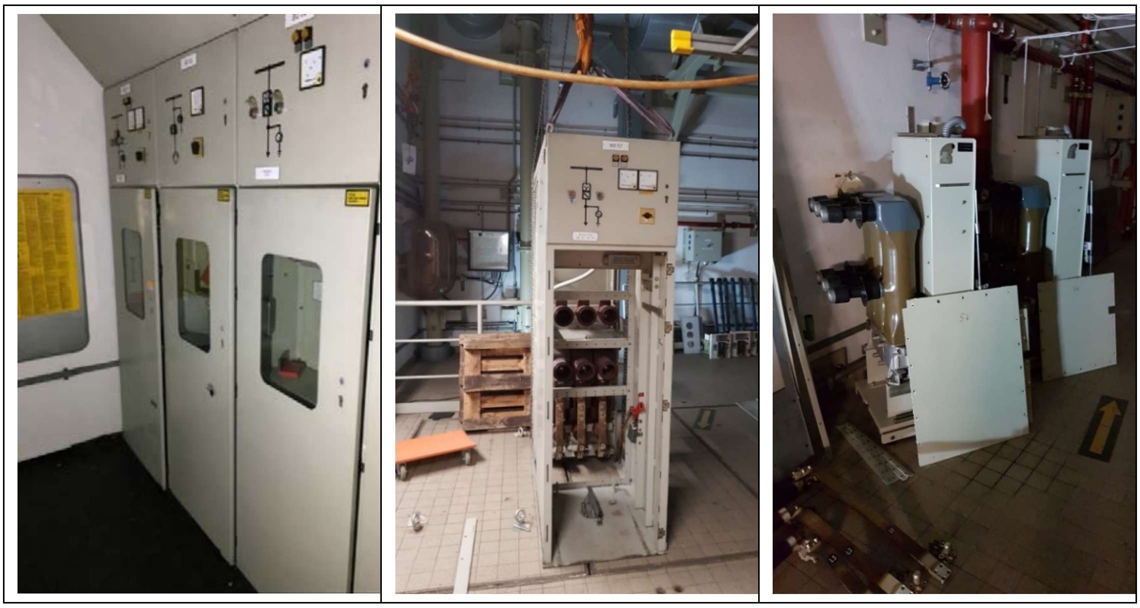


Figure 7 High voltage breaker cabinets; view from the outside (left), open cabinet enclosure (middle), and breaker itself (right)

Figure 8 Medium voltage breaker cabinets; view from the outside (left), and breaker inside (right)

The breakers were donated by PreussenElektra from a NPP in the North of Germany under decommissioning for several years. Removing the seismically proven mounted breakers and their enclosures took nearly three weeks, within another three weeks the breakers arrived in Chalfont, PA (USA). Figures 9a and 9b give an impression on the corresponding activities for equipment removal and transport.

Figure 9 Transport activities for the electrical breaker cabinets to be donated by Germany and the Netherlands to the HEAF 2 Project; a) cabinets prepared for transport (left), b) cabinets being loaded on a truck (right)

HEAF 2 Breaker Tests for European Breakers

Major goal of the HEAF 2 experiments is to simulate more realistic scenarios typical for NPPs, which are also representative for the operating experience. Therefore, there are two parts of the testing: In a first step, supplementary tests being representative for electrical breakers/bus bars and bus ducts installed in U.S. NPPs, partly involving aluminum, are being tested. These tests are highlighted with orange background in Figure 10 below.

In a second step, the OECD/NEA specific experiments, both with bus ducts and with bus bars/breakers in cabinet enclosures, will be tested. These, in total 16 tests marked in blue as Phase 2 tests in the lower part of Figure 10, also cover tests with the electrical breaker cabinets from Europe. These tests will be carried out as far as practicable under typical conditions with cabinets in row. Arc durations will be varied taking at least two of the three values of 2 s, 4 s, and 8 s, two different currents will be chosen from 15 kA, 25 kA, and 35 kA.

More details on the HEAF 2 experimental setup can be found in a paper by the Operating Agent from the U.S. NRC [13].

Figure 10 OECD/NEA HEAF 2 experiments and supplementary U.S. tests to be carried out, from [14]

CONCLUSIONS

The ongoing international experimental project under the auspices of the OECD/NEA, HEAF 2, is aiming on further insights in more realistic HEAF events and consequential fire scenarios in NPPs. The corresponding test plan of the OECD/NEA experiments and the supplementary tests which specifically cover the situation in U.S. nuclear power plants, reflects the HEAF efforts as a whole in two distinct parts. The OECD/NEA driven tests and the additional testing are being performed by the NRC with an enhanced focus on the phenomena associated with HEAF events in the presence of aluminum either in the component itself or as part of the enclosure. These experiments are intended to characterize thermal and pressure conditions as well as by-product deposits on surfaces created by HEAF occurring in electrical cabinets and bus ducts. The experimental results will provide qualitative information of the impact of HEAFs on typical targets in electrical rooms such as electrical cables and nearby equipment.

For the participating countries it is essential that – after the first HEAF tests series focusing on the phenomena and the possibility of generating ensuing fires by HEAF – the second phase of this international project will gain more insights on HEAF with arrangements of electrical cabinets being representative for existing NPPs.

Some regulatory bodies, such as the Dutch Authority for Nuclear Safety and Radiation Protection or the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety ("Bundesministerium für Umwelt, Naturschutz und nukleare Sicherheit", BMU) are highly interested in these experiments since some comparable breakers endangered by HEAF are in operation in their existing NPPs in operation, or even under post-commercial permanent safe shutdown or decommissioning. Both countries therefore participate in the OECD/NEA HEAF 2 by providing medium and high voltage breaker cabinets for HEAF testing in an as far as practicable realistic geometry of two or more cabinets in row.

ACKNOWLEDGEMENTS

The authors want to acknowledge the outstanding support provided by the HEAF 2 Project Review Group and, in particular, by the U.S. NRC Office of Research as Operating Agent as well as by the OECD/NEA Secretariat for making this project a successful one.

Furthermore, they are very thankful for the support by PreussenElektra having donated breakers for the HEAF 2 tests from a NPP under decommissioning.

Moreover, the authors want to acknowledge the support provided by the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) for funding of the German participation in the HEAF 2 Project.

REFERENCES

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): A Review of Current Calculation Methods Used to Predict Damage from High Energy Arcing Fault (HEAF) Events, NEA/CSNI/R(2015)10, Paris, France, 2015, https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-10.pdf.

[2] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Project – Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R(2013)6, Paris, France, June 2013, http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf.

[3] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Database Version 2017:02, Paris, France, May 2019 (limited to FIRE Database Project member countries only).

[4] Türschmann, M., M. Röwekamp, W. Werner: The OECD FIRE Database – Analysis of Fire Events in Nuclear Power Plants and Assessment of Event Sequences, GRS-A-3913, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2017(limited to FIRE member countries only).

[5] Röwekamp, M., et al.: Weiterführende Untersuchungen zum hochenergetischen Versagen elektrischer Einrichtungen mit sicherheitstechnischer Bedeutung in Kernkraftwerken, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-A-3630, Rev. 1, Köln, Germany, September 2011, http://www.grs.de/publikation/grs-A-3630.

[6] Shirai, K., et al.: Proposal of an Evaluation Method for Prevention of High Energy Arcing Fault (HEAF) Induced Fires at Low and High Voltage Electrical Cabinets, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

[7] International Atomic Energy Agency (IAEA): Protection against Internal Hazards in the Design of Nuclear Power Plants", IAEA Safety Standards Series No. SSG-64, Vienna, Austria, in publication, 2020.

[8] Nuclear Safety Standards Commission (KTA, German for Kerntechnischer Ausschuss): KTA 2101.1 (2015-11), Fire Protection in Nuclear Power Plants; Part 1: Basic Requirements, November 2015, http://www.kta-gs.de/e/standards/2100/2101_1_engl_2015_11.pdf.

[9] Nuclear Safety Standards Commission (KTA, German for Kerntechnischer Ausschuss): KTA 2103 (2015-11), Explosion Protection in Nuclear Power Plants with Light Water Reactors (General and Case-Specific Requirements), November 2015, http://www.kta-gs.de/e/standards/2100/2103_engl_2015_11.pdf.

[10] Kuriene, L.: Fire Protection at Nuclear Power Plants in the Netherlands, presentation at Technical Meeting on Fire Protection Approaches and Experiences of Nuclear Power Plants, International Atomic Energy Agency (IAEA), Vienna, Austria, 2015.

[11] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Experimental Results from the International High Energy Arcing Fault (HEAF) Research Program Testing Phase 2014 to 2016, NEA/CSNI/R(2017)7, Paris, France, 2017, http://www.oecd-nea.org/documents/2016/sin/csni-r2017-7.pdf.

[12] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research: An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arc Faults (HEAFs), NUREG-2218, Washington, DC, USA, 2018, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2218/.

[13] Melly, N., and M. H. Salley: OECD/NEA High Energy Arcing Faults (HEAF) Research – Second Phase of Testing, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

[14] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA): Agreement on the OECD Nuclear Energy Agency (NEA) High Energy Arcing Fault Events (HEAF) Phase 2 Project to characterize high-energy arcing-fault events, Paris, France, 2019.

3.4 Session on "Experimental Fire Research and Modelling"

The session on experimental fire research and modelling covered in total three presentations.

The first two contributions were prepared by experts from IRSN, France. The first one provided an overview on the intended cable fire benchmark exercise which will be conducted jointly by the OECD/NEA FIRE Database Project and the experimental project PRISME on fires in complex nuclear installation geometries with the intention to assess and compare the capabilities of different fire simulation codes for regulatory activities.

In the second presentation a decision support tool for fire safety analysis was discussed. This tool shall help to overcome the issue of extremely large numbers of simulation runs needed for statistically meaningful calculation results by the IRSN software tool SYLVIA for fire simulations.

The last presentation by KAERI from the Republic of Korea discussed the results of implementation strategies for a semi-empirical cable fire propagation model in the fire simulation code FDS. Such a model maybe helpful to achieve more accurate and detailed predictions under real conditions.

The Seminar Contributions prepared for this session are provided hereafter.

Common Cable Fire Benchmark Activity of the OECD Nuclear Energy Agency Projects PRISME 3 and FIRE

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ABSTRACT

Based on a request from the OECD Nuclear Energy Agency (NEA) Database Project FIRE in 2016, the recommendation was given by the PRISME 2 Program Review Group (PRG) that there should be a common Benchmark Exercise on a realistic cable fire scenario in an electrical system being as far as reasonably practicable representative of a real cable fire event recorded in the FIRE Database, using information on electrical cable fires from the OECD/NEA PRISME (French acronym for "Fire Propagation in Elementary Multi-Room Scenarios") Project.

The FIRE Database has clearly demonstrated the significance of fire events involving cables and shows that a majority of these events were either safety related or had the potential to impair nuclear safety. Both PRISME and PRISME 2 cable fire experiments have significantly increased the knowledge on cable fire behavior and investigated various types of cables implemented in nuclear power plants (NPPs) in member states. The major goal of this Benchmark Exercise is to simulate a real cable fire scenario in order to assess and compare the capabilities of different types of fire simulation codes to model such a complex and realistic fire scenario.

The strong interest of experts from regulators, technical safety organizations (TSOs) and licensees in predicting cable fires shows that such a Benchmark Exercise is a unique opportunity for cross-cutting work between experts from the OECD/NEA FIRE and PRISME Projects. Due to the high expert interest a decision was made to open the common PRISME 3 and FIRE Benchmark Exercise to other OECD/NEA CSNI (Committee on the Safety of Nuclear Installations) member countries.

The real fire event selected for the Benchmark Exercise from the FIRE Database covering more than 500 fire events occurred in a heater bay of a NPP and involved two electrical cable trays loaded with PVC insulated cables. Since a numerical Benchmark Exercise on a real fire event is quite challenging, the following three-step methodology for conducting this Benchmark Exercise has been proposed: (1) a calibration phase, (2) a blind simulation of a PRISME cable fire experiment, and (3) the real fire event simulation. This methodology is based on the fact that a similar behavior is expected between the steps 2 and 3 making it possible to extrapolate the error estimation.

INTRODUCTION

Based on a request from the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Fire Incidents Records Exchange (FIRE) Database Project in 2016, the 10th PRISME 2 Program Review Group (PRG) provided the recommendation that there should be a common Benchmark Exercise on a realistic cable fire scenario in an electrical system being as far as reasonably practicable representative for a real cable fire event included in the FIRE Database, using information on electrical cable fires from the OECD/NEA PRISME (French acronym for “Fire Propagation in Elementary Multi-Room Scenarios”) Projects.

Observations from the FIRE Database ([1] and [2]) have clearly demonstrated the significance of fire events involving cables and shown that a majority of these events were either safety related or had the potential to impair nuclear safety. On this topic, both PRISME and PRISME 2 [3] cable fire experiments have significantly increased the knowledge regarding cable fire behavior and investigated various types of cables implemented in NPPs in member states.

The major goal of this Benchmark Exercise (BE) is to simulate a real cable fire scenario in order to assess the behavior of fire models for such a complex and real fire scenario from the knowledge available so far.

The strong interest of experts from regulators, technical safety organizations (TSOs) and licensees in predicting cable fires shows that such a Benchmark Exercise is a unique opportunity for cross-cutting work between experts from the OECD/NEA FIRE and PRISME Projects. Due to the high interest, a decision was made to open the common OECD PRISME 3 and FIRE Benchmark exercise also to other CSNI member countries.

The real fire event selected for the Benchmark Exercise from the FIRE Database, covering meanwhile more than 500 fire events, occurred in a heater bay of a NPP and involved two electrical cable trays loaded with PVC insulated cables. Since a numerical benchmark on a real fire event is quite challenging, the following three-step methodology for conducting this Benchmark Exercise has been proposed:

- Step 1: a calibration phase,
- Step 2: a blind simulation of a PRISME cable fire experiment, and
- Step 3: the real fire event simulation.

This methodology is based on the fact that a similar behavior is expected between the steps 2 and 3 making it possible to extrapolate the error estimation.

The Fire Event Simulation Exercise (FREESE) dedicated website for this Benchmark Exercise activity was created and is now available on the IRSN gforge website (https://gforge.irsn.fr/gf/project/freese/).

Benchmark participants come from different institutions in Belgium, Canada, Finland, France, Germany, Japan, Korea, Spain, Sweden, the United Kingdom and the United States of America.

The Benchmark methodology, as well as the tools used to quantify the differences between simulation results and experimental data, are presented in the next section. The third section briefly characterizes the real fire event selected from the OECD FIRE Database and who will perform fire simulations during the third step of the BE. Results for predictive simulations of the first step are presented in a further section of this paper. Finally, the overall conclusions on the work done so far and an outlook are presented.

PRESENTATION OF THE CABLE BENCHMARK EXERCISE

Benchmark Methodology

In contrary to a well-controlled experiment, a real fire event does not occur under laboratory conditions and thus inputs and outputs are weakly under control. Assessing the quality of numerical results simulating such an event is therefore highly challenging. Based on the fact that a code-to-code comparison of simulation results is still possible, a three-step methodology has been proposed.

The first step (Step 1) consists of a calibration phase, i.e. in simulating a well-controlled fire scenario (a cable fire test from the PRISME 2 Project) for which the entire experimental data and uncertainties are available and provided to the participants. The second step (Step 2) will consist of a blind simulation of a cable fire experiment from the PRISME 3 Project. Only the input data on the experimental boundary conditions will be accessible to the participants, the experimental output data will not be provided to them. Finally, the last step (Step 3) will consist of performing a real fire event simulation based on the basis of the available information recorded in the FIRE Database. This methodology is based on the fact that a similar behavior is expected between Step 2 and Step 3 making it possible to extrapolate the error estimation. The three phases are outlined below:

- **Step 1:** Calibration phase on a cable fire experiment
  - The goal is to calibrate the fire models of each participant using a cable fire experiment from the OECD PRISME 2 Project (CFS (Cable Fire Spreading) campaign [4], [5]). The choice of the fire experiment is based on the cable type. The ventilation renewal rate should be as close as possible to the real fire event characteristics.
  - Features: open calculation, experimental data available, assessment of numerical versus experimental results (error estimation), assessment of the relative behavior of numerical results (behavior estimation).
- **Step 2:** Blind simulation phase of a cable fire experiment
  - The goal is to simulate a cable fire scenario from the PRISME 3 Project under so-called “blind” conditions for the simulation without knowing the experimental results. Such an experiment will be part of the CFP (Cable Fire Propagation) campaign. The fire test selection also depends on the cable type and the ventilation renewal rate that should be as close as possible to the real fire event characteristics.
  - Features: blind calculation, experimental data available after completion of the test, assessment of numerical versus experimental results (error estimation), assessment of the relative behavior of numerical results (behavior estimation).
- **Step 3:** Simulation phase of the real fire event
  - The goal is to simulate a real cable tray fire event from the OECD/NEA FIRE Database.
  - Features: calculation, not all fire event data available, limited assessment of numerical versus event results, assessment of the relative behavior of numerical results (behavior estimation).

Error Estimation

The error estimation aims at quantitatively comparing the difference between a simulation result (the output) and an experimental result as well as between several simulation results.

The output quantities selected for comparison are chosen from the usual fire quantities: the fire heat release rate (HRR), the fuel mass loss rate (MLR), gas temperatures, oxygen and carbon dioxide concentrations, relative pressure, mass flowrates at the inlet and outlet branches of the ventilation network, and wall temperatures.

An orientation chosen at the beginning of the exercise was that the quantities must be calculated by each fire simulation code participating in the BE. Computational fluid dynamics (CFD) codes and zone models can be distinguished on some quantities but common values, usually the average in space, should make the link.

A previous work on quantifying the differences between simulation results and experimental results was carried out in the frame of the PRISME SOURCE test series

[6]. It was devoted to a pool fire scenario in a confined and mechanically ventilated compartment. Authors of this document detailed many ways for quantifying differences between numerical and experimental results. Quantifying the capabilities of fire models can be made by using metric operators as suggested by the ASTM E1355-12(2018) [7], which may depend on the characteristics of the data (single point comparison, steady-state regime or time-dependent values).

The mathematical methods first depend on the characteristics of the data studied, and a distinction is made between single point comparison, steady-state comparison, and time-dependent values.

For a single point comparison, the data do not depend on time and space. It is typically a peak value, either a minimum or a maximum value, e.g. a temperature peak, a critical oxygen concentration, or a pressure peak, etc. In this case, the quantitative comparison can be made using an absolute or relative difference. Based on the work presented in [6], the normalized relative difference seems to be well appropriate for this study. It enables to take into account the initial state of the calculation as a reference state and to avoid any unit troubles. It is called the local error and expressed as follows:

$$\epsilon_{local} = \frac{(y-y_0)-(x-x_0)}{x-x_0} \quad (1),$$

where “x” is the experimental value and “y” is the numerical one, and “x_0” and “y_0” are the initial values.

For a steady-state regime, in the case of stationary systems or low-fluctuation quantities, it is recommended to compare the average difference.

For time-dependent values, the numerical result is compared with the experimental values all over the fire scenario duration. These time-dependant quantities can be either averaged in space (important for comparison with zone code results) or measured at a specific point (comparison with CFD codes). This approach introduces the concept of vector norms. In that case, the difference is called the global error and is defined as the normalized Euclidean distance between two vectors and expressed as follows:

$$\epsilon_{global} = \frac{\|\vec{y}-\vec{x}\|}{\|\vec{x}\|} = \sqrt{\frac{\sum_{i=1}^{n}(y_i-x_i)^2}{\sum_{i=1}^{n}(x_i)^2}} \quad (2).$$

To perform such a calculation, it is required to interpolate the numerical and experimental data to a common time discretization with a constant time-step [6].

Obviously, metrics can be used to conduct both code-to-experiment and code-to-code comparisons.

Simulation results will be compared to the experimental results and also to the mean simulation results, obtained by calculating the average value from all the simulations. The difference between the mean simulation and the experimental results will also be assessed providing the link between the simulation trends and the experiment in the manner of an expert panel review. This last point is particularly important for the last step of the Benchmark on the simulation of the real fire event whose output results for comparison are limited. Consequently, the mean simulation should be foreseen to act as a reference base for extrapolating the error estimations.

FIRE EVENT FROM THE OECD/NEA FIRE DATABASE

Event Selection Criteria

In May 2019, the latest version 2017:02 of the OECD/NEA FIRE Database was released gathering more than 500 fire events recorded up to the end of 2017 [8]. The criteria for the cable tray fire event selection, prescribed by the PRISME 2 PRG members, are the following:

- a cable fire scenario with flames and smoke;
- a quite recent event (from about the last ten years);
- a sufficiently well-documented event;
- the significance of being able to receive additional information;
- the closeness to PRISME cable fire experimental scenarios;
- preferably only one compartment involved in the fire;
- a fire duration between 15 min and 60 min.

A shortlist of four potential candidates was presented at the third PRISME 3 meeting in April 2018 providing a quite detailed description of the ignition phase and the sequences of the events from the available information so far. Further investigations were carried out to gather as many data as possible. To do so, licensees concerned by these events were contacted by the FIRE National Coordinators of the corresponding countries. Based on these investigations, the ability of gathering enough data on the event from licensees for supporting the activity was the decisive criterion for the final event selection.

An agreement on the selection of the event presented below was reached during the first Benchmark meeting held in Aix-en-Provence, France, in November 2018.

Characterization of the Selected Event

The authors emphasize that specific details on the fire event, in particular information from the NPP where the fire occurred could be identified, remain undisclosed due to confidentiality matters.

The finally selected fire event occurred in the heater bay of the turbine building of a nuclear power plant. The fire involved two 0.90 m wide horizontal cable trays loaded with PVC insulated cables and was caused by a non-conforming cable routing. The fire started by the self-ignition of 120 V power cables due to an arc fault. It was initiated when the high humidity and condensate from a steam leak provided the environment necessary for the existing flaw in an electrical cable to short to ground. The source of the cable flaw was identified as routing inconsistent with the current standard for minimum static bend radius for this type and size of cable, i.e. resting across rungs on a horizontal cable tray and exiting at a sharp angle downward into a 12 m vertical run, as illustrated in Figure 1.

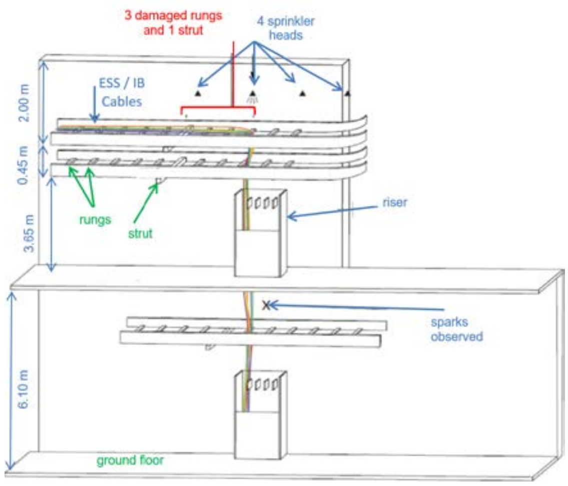


Figure 1 Front view scheme of the heater bay and the cable routing

The fire duration between ignition and successful extinguishing by wet pipe sprinkler was estimated to be approximately 20 min (± 2 min). Investigations by the licensee provided a detailed description of the fire behavior and the sequence of the event detailed hereafter.

The non-conforming cable routing concerned three instrument bus cables and three essential service system (ESS) bus cables. The arc fault from one of the six cables to the rung at the exit point damaged the insulation of nearby cables and heated the rung, leading to the severing of five (2 IB and 3 ESS) of the six cables. This conclusion is based on in-situ examination recording the remaining riser portion of each of the five cables, of equal length, with severed ends at the rung, and the remaining horizontal portions of two of the ESS bus cables also aligned with the rung. When they were severed, the two-line cables for the IB feed arc faulted together. These cables continued to arc until approximately 0.61 m of copper had been consumed from each of the IB line cables.

The arc fault between the IB line cables ended at a strut support, as evidenced by the ends of the IB cables being found aligning with and in the molten portion of the strut. It is likely that the breaker for the IB feed tripped at this time, as evidenced by the remaining intact copper. The total time from the initial arc fault to the IB feed breaker trip is estimated to be approximately 1 min corresponding to the time when sparks were observed outside the heater bay (see Figure 1), and the time of the swap of the IB from its main feed to a backup feed.

The fire on the bottom tray was initiated by debris falling from the fire on the top tray. This is evidenced by the concentration of the damaged cable in the upper levels of the bottom tray, with the lower levels undamaged or much less damaged. Additionally, there is no example of molten copper conductor in the bottom tray. The fire in the bottom tray appears to have continued to burn until extinguished by the fixed fire extinguishing system (wet sprinklers).

Four sprinkler heads were located in the vicinity of the fire, at about 1.20 m to 1.50 m above the top tray. The sprinkler heads actuate by thermal (fusible) links set to break at 100 °C (standard response time. The heat generated by the fire caused flow from only one of the four sprinkler heads in the area. Laboratory testing determined that the three sprinkler heads not actuated had no existing flaws and were physically capable of responding if required to actuate. The activated sprinkler head successfully extinguished the cable tray fire.

Available Results from the Selected Fire Event

This paragraph presents additional results of post-event analyses. At the exit point from the top tray (see rung location in Figure 2), cables were severed, and the applicable jacket/insulation were removed due to excessive heat. An initial inspection showed localized fire damage in the shape of a semicircle 0.76 m long and 0.51 m wide (cf. Figure 2). Charring was present throughout the entire depth of cables in that section (approximately 0.15 m deep).

The extent of cable damage in this area included cables severed, cable jacket/insulation damage, cable jacket/insulation completely removed and sections of the cables missing. The most extensive charring and damage was observed on the cables located at the bottom of the top tray with signs of an arc flash. The initial inspection of the bottom tray showed localized fire damage in the shape of a semicircle approximately 1.02 m long and 0.66 m wide (again cf. Figure 2**Fehler! Verweisquelle konnte nicht gefunden werden.**). Charring was present throughout the top 0.10 m of cables in that area. Molten drip could be seen on top of numerous cables on the bottom tray (tests subsequently verified that cuprous oxide was present in large quantities on the cables of the bottom tray).

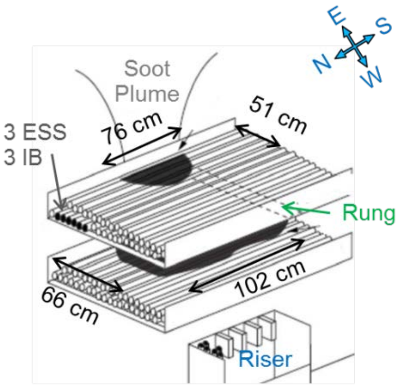


Figure 2 Damage pattern of the cable trays

Signs of minor concrete spalling were observed on the diagonal concrete overhang located above the affected cable trays. The smoke damage to the wall formed a v-pattern with a wide shape that extended to the ceiling, indicating a slowly burning fire.

Severe degradation was observed on a 0.09 m thick strut located just north of the semicircle fire damaged area (cf. Figure 2). More damage was present on the south side of the strut than on the north side and the top of the strut was completely molten between the north and south sides. Three rungs located south of the affected strut and north of the riser had sections completely melted through.

Additional degradation was observed on the east side of the top tray, which consisted of the east cable tray side wall bowing away from the fire. This bowing was locally centered in the semicircle fire damage previously mentioned.

CURRENT STATUS AND PROGRESS OF THE BENCHMARK EXERCISE

Investigations by the respective National Coordinator with the licensee of the affected NPP indicated that several cable types where involved in the fire, all of them contained chlorine. During the fire, there was no forced ventilation in the fire compartment which has a volume of about w x d x h = 30 m x 5 m x 6 m = 900 m^3. Considering the large dimensions of the fire compartment, the fire location inside the room and the limited fire propagation (cf. Figure 2), the oxygen limitation on the fire is supposed to be small. The fire is therefore supposed to be fire load controlled until extinguished. This information resulted in the selection of the PRISME 2 CFS-2 experiment [5] characterized by the highest ventilation renewal rate (15 h^{-1}) and a stack of five cable trays loaded with PVC cables for the open simulation in Step 1.

The first simulation results for Step 1, were presented during the second Benchmark Exercise meeting held in May 2019. These results consisted in the prescribed simulation of CFS-2, i.e. using the experimental heat release rate measured during the experiment as input data. Nine simulations with five different fire models have been performed.

A second set of simulation results concerned predictive simulations of the CFS-2 experiment. Seven predictive simulations were performed, as indicated in Table 1.

These simulation results for the HRR and the MLR are presented in Figure 3 and Figure 5. The mean simulation values as well as the minimum and maximum values are given in Figure 4 and Figure 6. Error values from the comparison with the experiment are indicated in Figure 7 and Figure 8 for the simulation results, as well as for the mean-simulation:

– the relative difference compared during the first 500 s "ε_local_1 vs EXP",
– the relative difference compared during the first 2000 s "ε_local_2 vs EXP"[4],
– the normalized Euclidean distance during the first 500 s "ε_global_1 vs EXP",
– the normalized Euclidean distance during the first 2000 s "ε_global_2 vs EXP".

Table 1 Participation in the predictive phase of Step 1 of the Benchmark Exercise

Simulation	Fire Simulation Software	Institution
CFD 2	ISIS	IRSN
CFD 4	COCOSYS[5]	GRS
CFD 5	FDS	NRC
CFD 6	FDS	IBMB
CFD 7	FDS	VTT
ZC 1	SYLVIA	IRSN
ZC 2	BRI2002	CRIEPI

Note: CFD – computational fluid dynamics code, ZC – zone code

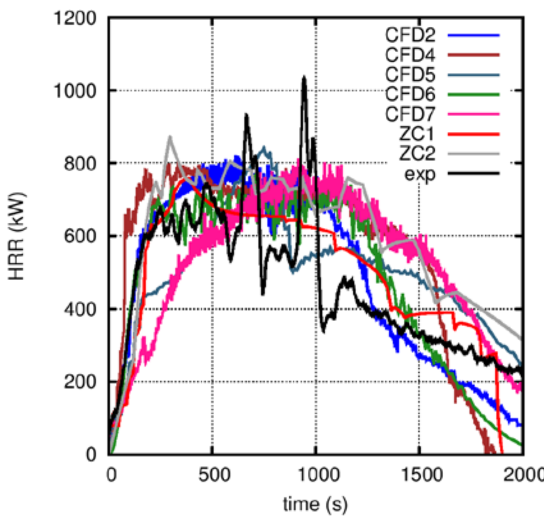


Figure 3 Predictive simulation results of the HRR for Benchmark Exercise Step1 compared to

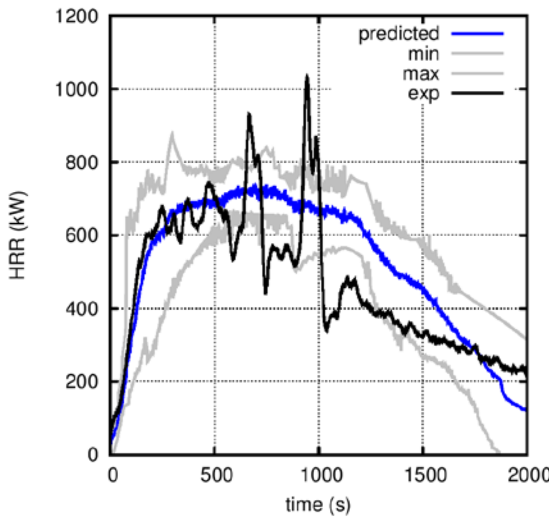
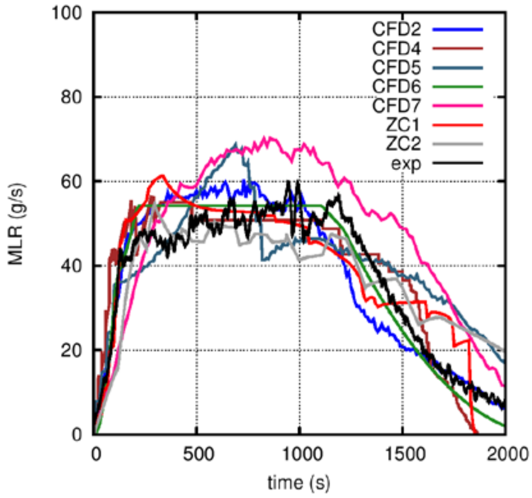


Figure 4 Minimum, maximum and average simulation values of the HRR of Benchmark Exercise Step 1

4 available for the MLR only in order to avoid considering peaks due to the fast combustions under the ceiling (550 s – 1200 s) which are unpredictable (or at least not in the scope of the BE).

5 COCOSYS is a lumped parameter code but it is considered as a CFD code in the post-processing of the results since all the "CFD-required" quantities were provided (e.g. temperatures at several elevations).

the CFS-2 experimental results



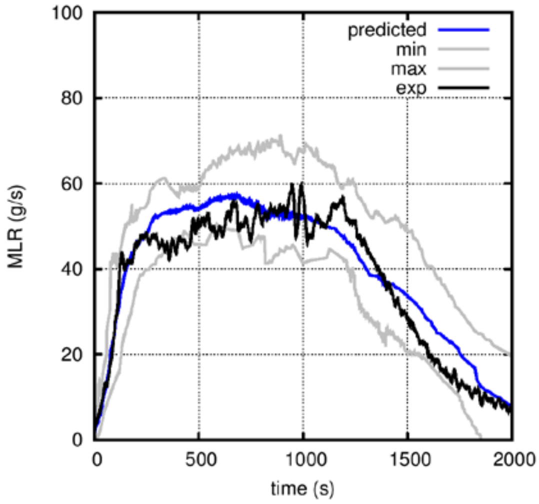
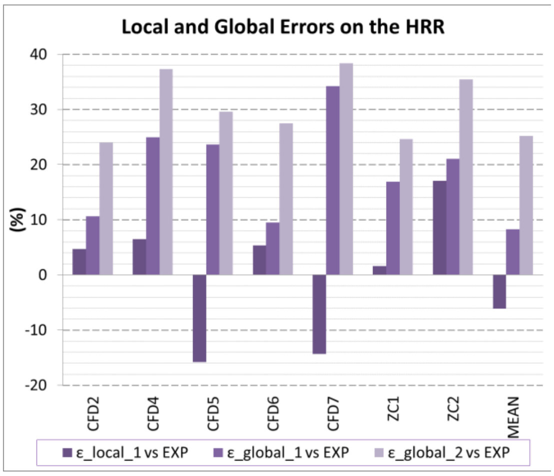


Figure 5 Predictive simulation results of the MLR for Benchmark Exercise Step1 compared to the CFS-2 experimental results

Figure 6 Minimum, maximum and average simulation values of the MLR of Benchmark Exercise Step 1



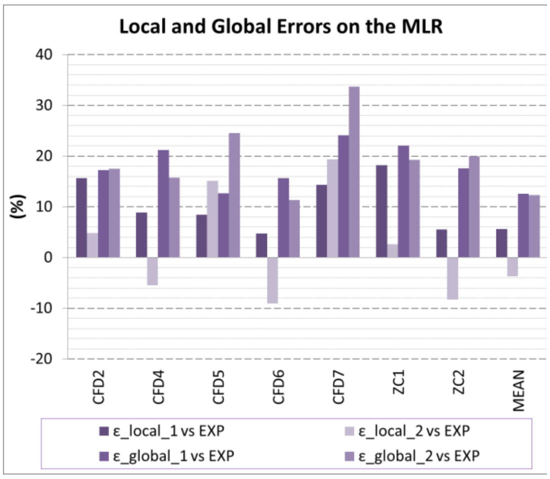


Figure 7 Local and global errors on the HRR for the predictive phase of Benchmark Exercise Step 1

Figure 8 Local and global errors on the MLR for the predictive phase of Benchmark Exercise Step 1

As illustrated in Figure 3, participants performing predictive simulations of the CFS-2 experiment obtained results in good agreement with the experimental results, with some under- or over-estimations or time shift.

The HRR peaks during the first 500 s of the transient are predicted within a range of – 15 % up to + 17 % (see ε_local_1 in Figure 7). The mean simulation underestimates the HRR peak by about 6 %. Figure 4 indicates that the predicted development over time is smoother as the experimental one characterized by some oscillations where the experimental peak value is reached. Concerning how close curves are to the experiment, it becomes obvious that the mean simulation has a global error of about 8 % over the first 500 s and 25 % all over the first 2000 s of the transient.

Global values calculated over the period when fast combustion occurred under the ceiling get large due to sharp unpredictable evolutions leading to large distances between the curves.

The MLR peaks during the first 500 s of the transient are predicted within a range of + 5 % up to + 18 % and between – 9 % and + 20 % during the first 2000 s (see ε_local_1 and ε_local_2, respectively, in Figure 8). The mean simulation overestimates the MLR peak by about 5 % over the first 500 s and it underestimates the global peak by about 4 %. Figure 6 demonstrates that the mean simulation value is very close to the experimental one and that the latter is encompassed in the minimum and maximum values of the predictions. The mean simulation has a global error of about 12 % over the first 500 s as well as over the first 2000 s of the transient.

The simulation results are highly satisfying, particularly keeping in mind that predictively simulating cable tray fires in confined conditions is a quite difficult task. Resorting to a mean simulation also seems to be relevant to be considered as an expert opinion as raised above.

CONCLUSION AND DISCUSSION

This paper presents the methodology proposed for assessing the capability of different types of simulation software to model a real fire scenario such as a fire event recorded in the OECD/NEA FIRE Database making use of the knowledge brought by the OECD/NEA PRISME experimental projects.

The proposed methodology features three steps. It consists of a calibration phase on the CFS-2 experiment from the PRISME 2 Project, a blind simulation of a cable fire experiment to be performed during the PRISME 3 Project, and the simulation of a real fire event from the FIRE Database.

Metrics operators, namely the local error on peak values and the global error defined as the normalized Euclidean distance between two vectors, are used to quantify the differences between the simulation results and experimental ones. The three-step methodology is based on the fact that a similar behavior is expected between Step 2 and Step 3 enabling the analysts to extrapolate the error estimation of Step 3 for which only few comparison points are available.

The real fire scenario is an event having occurred in the heater bay of a nuclear power plant. The fire was initiated by an arc flash and involved two electrical cable trays loaded with PVC insulated cable. The duration was approximately 20 min before being extinguished by a sprinkler. The extent of the damages is quite limited with a length of cable tray burnt of about 1 m.

Simulation results are presented for the first step of the Benchmark with predictive simulations of the CFS-2 experiment. Local errors on peak values as well as the global errors on different portions of the fire scenario are calculated. These error values quantitatively illustrate that the simulation results are satisfying, in particular for predictive simulations of cable tray fires. The mean simulation is calculated as a post-processing of the simulation results in order to be used as a reference in the manner of an expert panel opinion. The mean simulation is foreseen to take on importance during the third stage when the availability of comparison points is limited.

In the next, second step of the Benchmark Exercise blind simulations of an experiment to be performed during PRISME 3 in a corridor room configuration with the same type of fire source as in the CFS-2 experiment are intended once the corresponding experimental results are available. Local and global errors between the simulation results and the mean-simulation values can be calculated whenever needed throughout the Benchmark Exercise.

A continuous task in the frame of this cable Benchmark Exercise is to improve the understanding of the real fire event, its ignition mechanism and the circumstances that lead to the post-event observations as preparatory work for Step 3.

ACKNOWLEDGMENTS

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REFERENCES

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI), FIRE Project Report: “Collection and Analysis of Fire Events (2010-2013) - Extensions in the Database and Applications”, NEA/CSNI/R(2015)14, Paris, France, August 2015,
https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-14.pdf.

[2] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Combinations of Fires and Other Events - The Fire Incidents Records Exchange Project Topical Report No. 3, NEA/CSNI/R(2016)7, Paris, France, July 2016,
http://www.oecd-nea.org/documents/2016/sin/csni-r2016-7.pdf.

[3] Audouin, L., et al., 2013, OECD PRISME 2 Fire Research Project (2011-2016) – Current Status and Perspectives, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 22, 13th International Seminar on Fire Safety in Nuclear Power Plants and Installations, September 18-20, 2013, Columbia, SC, USA, GRS-A-3731, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany, December 2013,
http://www.grs.de/sites/default/files/pdf/grs-a-3731.pdf.

[4] Zavaleta, P., and L. Audouin: Cable tray fire tests in a confined and mechanically ventilated facility, Fire and Materials, Vol. 42, Issue 1, pp. 28-43, Wiley & Sons, Hoboken, NJ, USA, 2018,
https://doi.org/10.1002/fam.2454.

[5] Zavaleta, P., S. Suard, and L. Audouin: Cable tray fire tests with halogenated electric cables in a confined and mechanically-ventilated facility, Fire and Materials, Vol. 43, Issue 5, pp. 431-609, Wiley & Sons, Hoboken, NJ, USA, 2019,
https://doi.org/10.1002/fam.2717.

[6] Audouin, L., et al.: Quantifying differences between computational results and measurements in the case of a large-scale well-confined fire scenario, Nuclear Engineering and Design, Vol. 241, Issue 1, pp. 18-31, Elsevier, Amsterdam, the Netherlands, 2011,
https://www.sciencedirect.com/science/article/pii/S0029549310007065.

[7] American Society for Testing and Materials (ASTM): ASTM E1355-12(2018), Standard Guide for Evaluating the Predictive Capability of Deterministic Fire Models, ASTM International, West Conshohocken, PA, USA, 2018.

[8] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Database Version 2017:02, Paris, France, May 2019 (limited to FIRE member countries only).

SEVEN Expert System: A Decision Support Tool for Fire Safety Analysis in the Nuclear Area

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ABSTRACT

For fire safety studies in nuclear facilities, IRSN uses the SYLVIA software to simulate fire scenarios in highly confined and mechanically ventilated compartments and airborne contamination transfers inside nuclear facilities. In order to take into account the different sources of uncertainty resulting from initial and boundary conditions as well as from model parameters, the SYLVIA software is associated with the SUNSET statistical software. However, such a use of SYLVIA requires a large number of runs and a significant statistical analysis what is not always compatible to the requirements of safety assessments in terms of deadlines. To overcome this difficulty, IRSN is currently developing expert systems based on SYLVIA result databases. This approach allows deriving the most likely diagnosis or prognosis in a very short time, but also deriving a more complex form of reasoning intertwining prognostic and diagnostic inferences.

These expert systems are based on the Bayesian Belief Network methodology and consist in two steps: first, a large database obtained from SYLVIA runs allows the estimation of conditional probability tables; then, a message passing algorithm is used to dynamically exploit this database. The illustrating example is based on the study of the behavior of the final level of aerosol filtration in nuclear facilities, in a fire situation. The holding of the final level of filtration is conditioned by the thermal and mechanical stresses experienced by high-efficiency particulate air filters. The database of the expert system SEVEN is built from the results of ten million calculations performed with the SYLVIA software.

This example illustrates how an expert system can be used as a decision support tool for fire safety analysis in the nuclear area. Expert systems represent a new generation of calculation tools in the field of probabilistic fire simulation and contribute to building the enhanced expertise of tomorrow.

INTRODUCTION

The SYLVIA software system [1] has been developed by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) to simulate a complete ventilation network, fire scenarios in a highly confined and mechanically ventilated facility, and airborne contamination transfers inside nuclear facilities. This software is based on a two-zone approach and is used by IRSN for fire safety studies.

To evaluate the impact of uncertainties, the SYLVIA software is coupled to the SUNSET software [2], one of IRSN's statistical tools, used in support of risk analysis studies. This coupling makes it possible to directly carry out a set of parametric studies and then measure the impact on some selected responses. A typical use of the SYLVIA/SUNSET coupling is to perform a Monte Carlo simulation in which a set of variables, known as study parameters, is modeled by random variables. The results obtained from a Monte Carlo simulation constitute a database linking parametric configurations determined by the set of values assigned to the study parameters and uncertainties to the corresponding results. However, the direct use of this database in the context of a safety assessment encounters two main difficulties:

- The database is necessarily very limited considering the possible configurations. The SYLVIA simulations constituting the database represent a small percentage of the possible parametric configurations. This is due to the combinatorial explosion of the configurations as a function of the possible values taken for each parameter and the number of parameters considered. For instance, if we consider 16 parameters and each of them can take only three values, the number of combinations of values is 3^{16}, i.e. approximately 43 million configurations.
- The database is not specific to the characteristics of a safety assessment. It is necessary to extract from the database the information compatible with the specificities of the case of interest. For example, a safety assessment can focus more specifically on large volumes, low air renewal rates, etc. and seek to discriminate configurations compatible with safety issues, such as maximum pressure difference through High-Efficiency Particulate Air (HEPA) filters of the final level of aerosol filtration.

To meet this dual challenge, it is necessary to be able to correctly update the information contained in the database by integrating the characteristics of each safety assessment. One solution is to use an expert system [3]. This approach allows deriving in a negligible time prognostic and diagnostic like inferences, but also more complex forms of reasoning intertwining prognostic and diagnostic inferences. To achieve this goal, a large SYLVIA result database has to be built.

The illustrating example is based on the study of the behavior of the final level of aerosol filtration in nuclear facilities in a fire situation. The holding of the final level of filtration is conditioned by the thermal and mechanical stresses experienced by HEPA filters. The database of this expert system, named SEVEN, is built from the results of ten million calculations performed with the SYLVIA software. This example illustrates how an expert system can be used as a decision support tool for fire safety analysis in the nuclear area.

THE EXPERT SYSTEM

An expert system is a tool that aims to simulate the cognitive mechanisms of an expert in a particular field. This is one of paths leading to artificial intelligence. More precisely, an expert system in artificial intelligence is defined as a computer program that has the ability to represent and reason from observations and generic knowledge. In fire safety, it is useful to be able to quickly discern the configurations of a facility at risk. The idea behind the expert system approach is to make the most benefit of the SYLVIA software to build a database covering a wide range of configurations, and then to use the expert system reasoning abilities to discern configurations of this database useful to one specific case of interest.

An expert system can be divided into three separated components [4], as demonstrated in Figure 1:

- **The knowledge base** that contains all the generic information in which the expert system will operate. This information will be encoded by means of conditional probability tables (CPTs, cf. the green rectangle in Figure 1).
- **The observation base** that gathers all the contingent or specific information from which inferences can be performed. This information has to be provided by the users in terms of likelihood or probability.
- **The inference engine**, a set of algorithms (the yellow arrows in Figure 1) that propels the information coming from the observation base through the knowledge base. Contrary to “physical” computer codes that intertwine the numerical data coming

from initial and boundary conditions with the solving algorithms, the expert systems algorithms are designed to be independent and separated from the data.

The general principle is to update in a real-time process the knowledge related to the variables defined in the expert system. More precisely, the expert system objective is to have a numerical tool able to perform three types of inferences:

1. In a forward chaining (prognostic inference), to determine for a configuration of input data the possible responses;
2. In a backward chaining (diagnostic inference), to identify for a given configuration of the responses the compatible input data;
3. In a mixed chaining, an inference that intertwines prognostic and diagnostic inferences.

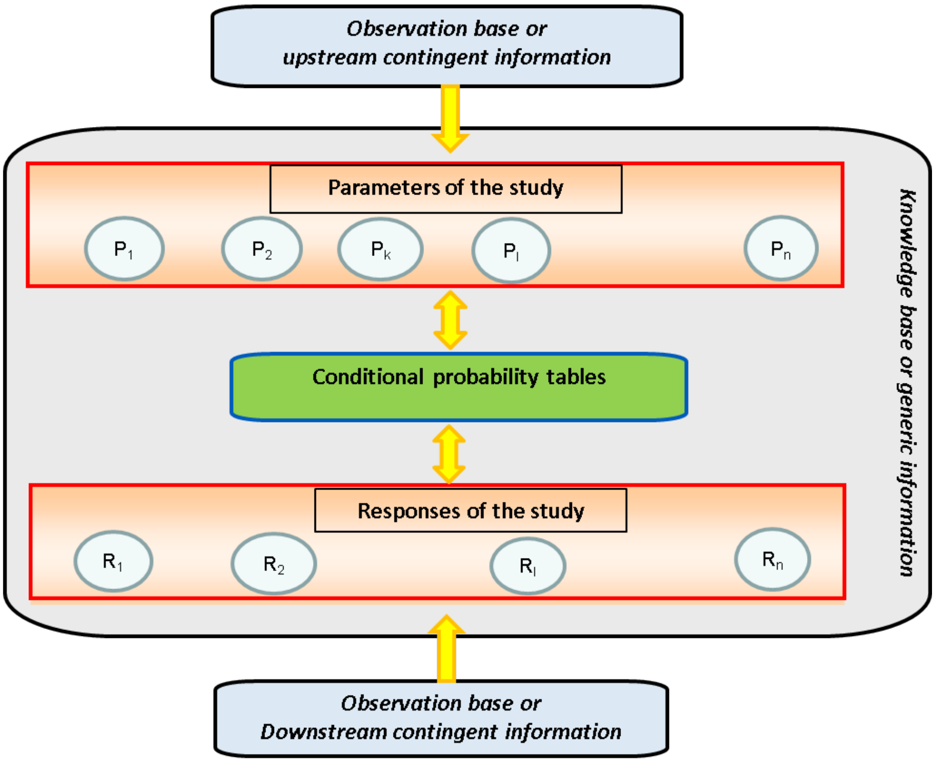

Figure 1 Flow chart of the expert system

The Knowledge Base

Since the computation time is short enough to perform many calculations, our approach consists in building a database, which relates to the case studied – such as the behavior of the final level of aerosol filtration in nuclear facilities, in a fire situation – by performing a stratified Monte Carlo study by a Latin Hypercube Sampling (LHS) method [5]. This Monte Carlo study is carried out by varying the input parameters of the calculation code in the study area under consideration. Thus, if we want the expert system to be able to answer to queries for compartment volume ranges between 100 and 500 m^3, we have to model this parameter by a random variable between 100 and 500 m^3 in the Monte-Carlo simulation. This way, we can build a large database made of SYLVIA calculations. This database is made up of all the data corresponding to both parameters and outputs. Then, this database can be interpreted as a numerical transcription of the generic knowledge carried by the SYLVIA software.

In a formal way, the SYLVIA software can be seen as a mapping of the parameters´ domain to the responses´ domain (see Figure 2). This can be written as:

$$R_i = S(P_1, \dots , P_N) \tag{1}$$

where R_i is any response of interest, P_j, the parameters and S, the SYLVIA software acting as a transfer function.

With this formalism, a SYLVIA computation is defined by fixing values p_j to each parameter P_j and by calculating the values r_i of any code output R_i. It should be noted that the independent variables P_j and the response R_i of the equation (1) can be either continuous or discrete.

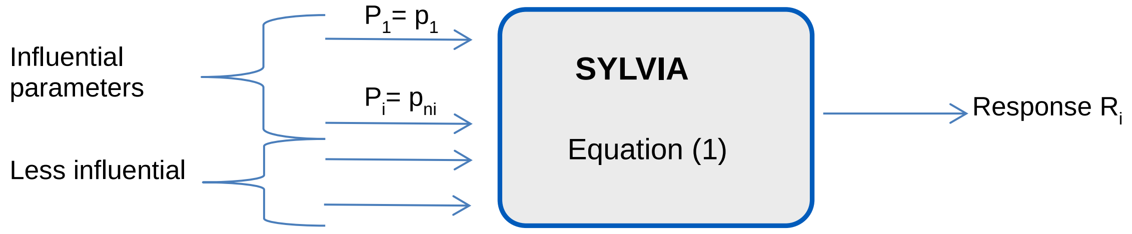

Figure 2 The formal model of SYLVIA

The principle followed to establish the SYLVIA knowledge base consists in transcribing the transfer function S into numerical tables (one for each response). In order to carry out this transcription of SYLVIA into numerical tables (see Figure 3), two simplifications are necessary. The first one consists in discretizing all the continuous variables of the equation (1) as:

$$R_i^* = S(P_1^*, \dots, P_N^*) \tag{2}$$

where R_i^* and P_j^* can only take discrete values.

The second simplification [6] concerns the identification of influential parameters for each response to limit the combinatorial aspect induced by the numerical transcription of the equation (2). Therefore, a preliminary step before making the knowledge base is the identification, for each response R_i, of its most n_i influential parameters. It has been done with a covariance analysis. Thus, equation (2) becomes:

$$R_i = S(P_1, \dots , P_{ni}, U_i) \tag{3}$$

where U_i is a random variable modeling the loss of information induced by the discretization step and by neglecting the less influential variables of the response R_i. It is worth noting that this model is stochastic, since for a given parametric configuration of $P_1, \dots, P_{ni}$, R_i may have different values.

From these simplifications, each SYLVIA calculation is replaced by a set of discrete values: the levels of the parameters and of the responses. Then, the whole set of SYLVIA simulations is used to calculate the conditional probability of each response knowing the combination of its influential parameters.

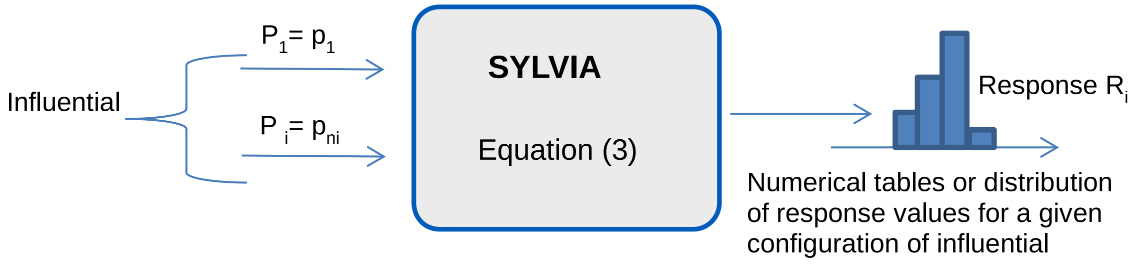

Figure 3 The structural model of SYLVIA

The Observation Base

In the expert system based on a SYLVIA database, the variables of the observation base are identical to the variables of the knowledge base. Unlike the knowledge base that encodes the generic information (i.e. the information carried out through the SYLVIA code), the observation base encodes the contingent knowledge for which we wish to solicit the expert system.

In a Bayesian network, each variable receives two kinds of information: an upstream information and a downstream information [4]. This distinction is essential to correctly perform the information propagation in a network. We will come back to this notion in the next section, as for now, it is sufficient to know that upstream information is required for the parameters and downstream information for the responses. This information is given by means of probability or likelihood. For example, if a variable V (associated to either a parameter or a response) is discretized into four levels (very low, low, high, very high), a (2, 1, 1, 0) u-plet is equivalent to the (0.5, 0.25, 0.25, 0) u-plet and means that the very low level is twice likely as the low or high level and the very high level is either impossible or not considered. More generally, the observational database consists in providing for each parameter P_i and for each response R_j some information that specifies (by means of vectors π_{Pi} and λ_{Rj}) the domain in which the expert system will operate.

The Inference Engine

A Bayesian network is not merely a passive tool storing factual knowledge, but also a computational architecture reasoning on that knowledge. This means that the links in the network have to be seen as mechanisms that propel information in order to update it. The CPTs (Conditional Probability Tables) attached to the nodes (cf. the left-hand variables of the equations in the system 3) act as single processors so that the inference engine is the set of processors (as many single processors as equations in the system 3).

To propagate and update information, the inference engine distinguishes upstream and downstream information. For a parameter P_i, the upstream information is the information provided by the vector π_{Pi} defined in the observation base and the downstream information is the vector λ_{Pi}which will be calculated by the inference engine. In a similar way, for a response R_j, the downstream information is the information provided by the vector λ_{Rj} defined in the observation base and the upstream information is the vector π_{Rj}, which will be calculated by the inference engine.

From this distinction between upstream and downstream information, each single processor is able to perform three kinds of local computation independently of other things happening in the network:

– **A forward propagation mechanism**. It consists of gathering all the upstream information coming from the right-hand variables and transforming it into upstream information of the left-hand variables.
– **A backward propagation mechanism**. It consists of gathering all the downstream information coming from the left-hand variables and transforming it into downstream information of the right-hand variables.
– **An updating mechanism**. It consists for a variable X of combining all the downstream information coming from the equations, where X is on the left-hand side, with all the upstream information, where X is on the right-hand side.

As each processor is connected to another in a Bayesian network, the local information can circulate through the whole network. These propagation mechanisms proposed by J. Pearl [7] are called the "message passing" algorithms: they act as information propellers from one variable to its neighbors.

BUILDING THE KNOWLEDGE BASE OF THE EXPERT SYSTEM SEVEN

The knowledge base collects all the generic information from which the expert system will perform inferences. It determines the application domain of the expert system. Thus, a first step consists of delimiting the general framework of the study and in defining the parameters and responses of the study and their variation ranges. In a second step, the SYLVIA database is built, and the conditional probability tables are computed.

Delimitation of the General Framework

In the presence of radioactive materials a fire can become a vector of resuspension and dissemination of these materials, and thus, can generate uncontrolled radio exposure of workers or even a release of radioactive materials into the environment. The estimation of the fire source term (total activity corresponding to the release of radioactive materials into the environment) allows IRSN to assess the sufficiency of the risk control measures taken by the operator of a nuclear facility and to apprehend the decisions to be taken in a crisis situation as the delimitation of a security perimeter. The study of the behavior of the final level of aerosol filtration in nuclear facilities in a fire situation is a major step in the assessment of the fire source term by simulation tools.

The perimeter of the knowledge base determines the scope of the expert system. It is delimited by the general framework of the study. This one is presented in Figure 4.

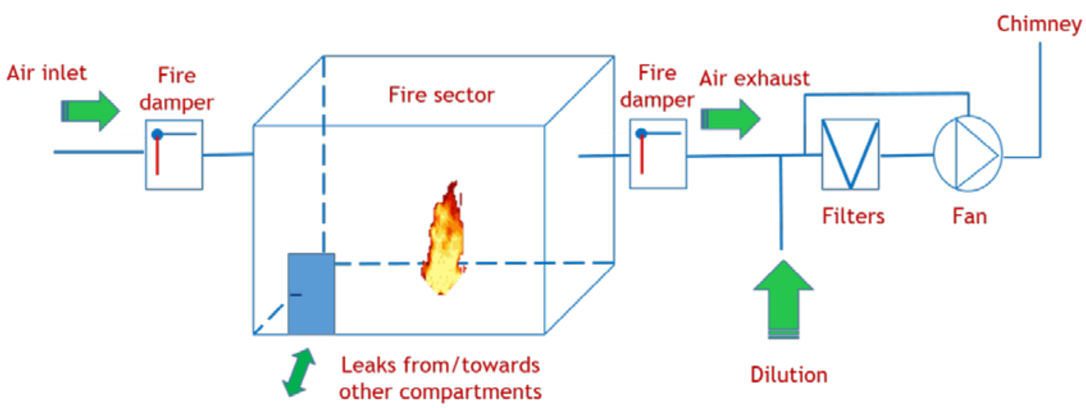

Figure 4 Topology scheme of the general framework of the study

It consists in a fire sector, represented by a 4 m high compartment, whatever the considered volume. This one is provided with fire dampers at the inlet and the exhaust air vents. A leak representative of all the leaks of the compartment (including leaks through doors) is modeled. The ventilation network is composed of an air inlet line and an air exhaust line with a dilution line. The air flow is ensured by two fans located, for one, at the entrance of the inlet line and modeled by a boundary condition node (80 Pa, characteristic value of what is usually observed), and for the other, at the end of the exhaust line, upstream of which a battery of HEPA filters (final level of aerosol filtration) is connected. A regulation of the exhaust fan is taken into account in the study. This one is equipped with a rotation speed control device allowing to assign the pressure upstream of the filters to a set value (- 2000 Pa, characteristic value of what is usually observed). Nevertheless, cases without regulation are also taken into account in order to cover non-regulated historical facilities.

In the nominal state, the compartment pressure is set at – 100 Pa relative to the atmospheric pressure. This value was chosen in coherence with the under-pressure value recommended in ISO 17873 standard (criteria for the design and operation of ventilation systems of nuclear facilities other than nuclear power plants, 2004) for C2 confinement class rooms [8].

Assumptions of the Study

The expert system SEVEN is based on the following assumptions:

- The modeling of the ventilation network adopted in the study is based on the methodology for the simplification of the ventilation networks, developed and validated at IRSN for fire scenarios using the SYLVIA software. This methodology is very conservative because it does neither take into account thermal losses nor the deposition of combustion aerosols occurring in the ventilation ducts between the fire room and the final level of aerosol filtration. Since these phenomena are highly dependent on the geometry of the ventilation network (duct diameter, number of bends, duct length, etc.), the deposition rate of soot particles upstream of the filters is therefore a parameter of the study. Similarly, thermal losses in the ventilation ducts are not taken into account, only the cooling of the gases of a dilution located downstream of the fire room is taken into account.
- The deposition of soot particles in the fire room is in particular due to the thermophoresis phenomenon (deposition related to a thermal gradient at the vicinity of the deposition surface). To take into account this phenomenon, a fine description of equipment in terms of surface and materials is required. These aspects are not taken into account in this version of the expert system. Thus, the parameter characterizing the deposition rate of soot particles upstream of the filters also integrates the soot particle deposition in the fire room.
- The modeling of the ventilation network of the study does not allow to study the effect of the shift to the half-regime of ventilation on the behavior of the filters of the final level of aerosol filtration, in a fire situation. Indeed, the study of this ventilation management strategy requires the modeling of two inlet fans and two exhaust fans, which considerably increases the number of parameters to take into account. Since this strategy is not part of the strategies commonly used by operators in their safety analysis, it is not integrated in the knowledge base of the expert system SEVEN. However, the knowledge base could be completed later, if necessary.
- The fire source is modelled by a design fire [9] in order to cover all the kinetics of fire growth in the study. A design fire is characterized by its maximum heat release rate in open atmosphere and by its fire growth factor:

$$HRR = \alpha t^2 \tag{4}$$

where HRR [kW] is the heat release rate, α [kWs^{-2}] the fire growth factor and t [s] the time.

– The filter clogging model [10] was developed at IRSN from data acquired during the combustion of different fuels studied at IRSN. This model only retains direct parameters known to have an influence on the filter clogging, in order to be easily usable. The empirical law elaborated is in the form:

$$\begin{cases} R(t=0) = R_0 \\ \frac{d}{dt}(R) = \left(\frac{a(1-FC)}{d_p} + 2b\left(\frac{1-FC}{\max(v_f, v_0/9)d_p}\right)^2 m_{ae}\right)\frac{dm_{ae}}{dt} \end{cases} \quad (5)$$

with:

- R: aeraulic resistance of the filter [kg m^{-4} s^{-1}];
- R_0: initial aeraulic resistance of the filter [kg m^{-4} s^{-1}];
- FC: Mass fraction of condensate contained in aerosols and deposited on filters [-], set to 0 in the study;
- m_{ae}: mass of aerosols deposited on filters per unit surface area [kg^{-2}];
- d_p: characteristic diameter of soot particles: diameter of the monomers constitutive of the aggregates for particles of fractal morphology, or volume-equivalent diameter for particles of compact morphology [m];
- v_f: filtration velocity [m s^{-1}];
- v_0: nominal filtration velocity [m s^{-1}];
- a, b: empirical clogging constants (a=2.8 10^{-5} <m^3 kg^{-1}; b=5.5 10^{-15} m^8 kg^{-2} s^{-2}).

– Fire scenarios with rupture of a fire break door are not taken into account in the study. They require the modeling of adjacent compartments as well as their ventilation system in order to take into account a recovery of a part of the soot inventory by the ventilation. However, these scenarios could enrich the knowledge base of the expert system later.

– In order to cover the various ventilation regimes of the study, a parametrization of the characteristic curve of the exhaust fan is achieved. This curve links the volume flow rate of the gas passing through the fan (Qv) to the total pressure difference at its edges (manometric height H) and is in the form of a polynomial of the second degree:

$$H = C_2Q_v^2 + C_1Q_v + H_0 \quad (6)$$

Coefficients H_0, C_1 and C_2 were deduced from the observation of 39 exhaust fan curves studied at IRSN (see Figure 5).

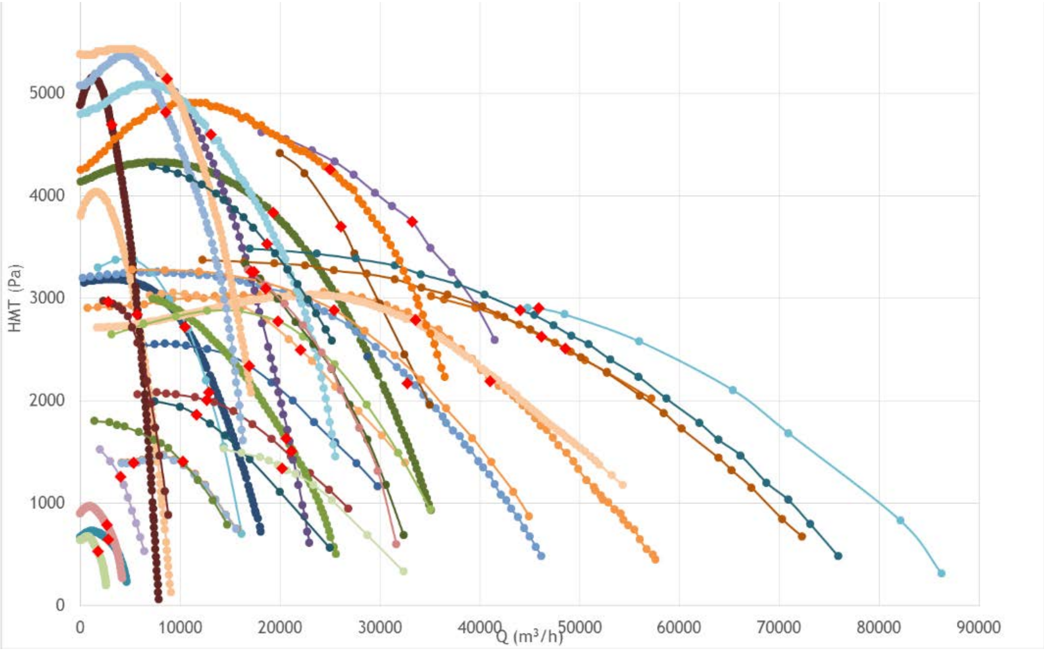

Figure 5 Characteristic curves of exhaust fans studied at IRSN

Parameters and Responses of the Study

According to the issue addressed in this study, three responses were retained: the maximum pressure difference and the maximum gas temperature through HEPA filters of the final level of aerosol filtration as well as the initial dilution rate of the gas upstream of the filters. The latter is defined as the ratio of the air volume flowrate in the dilution line to the gas volume flowrate at the exhaust of the fire room. The discretization of these three responses is reported in Table 1. The classes of response constitute the columns of the conditional probability tables.

Table 1 Discretization of the responses of the study

Responses	Discretization
Maximum pressure difference through filters [Pa]	< 500 [500; 1000] [1000; 1500] [1500; 2000] > 2000
Maximum gas temperature through filters [°C]	[20; 50] [50; 100] [100; 150] [150; 200] > 200
Initial dilution rate of gas [-]	< 5 [5; 50] [50; 500] [500; 1000] > 1000

According to the responses of the study, 20 parameters have been identified as influencing these responses. They are split into three categories, as reported in Table 2 to Table 4. The discretization of the variables is also specified in these tables.

Table 2 Parameters related to the fire, D {discrete values}, C [continuous values]

Parameters	Type	Discretization
Mass of fuel [kg]	C	[100; 400] [400; 700] [700; 1000] [1000; 1500] > 1500
Fire growth factor [kWs^{-2}]	D	{3 10^{-3}; 0.012; 0.047; 0.19}
Maximum HRR in open atmosphere [kW]	C	[200; 800] [800; 1500] [1500; 3000] [3000; 5000]
Heat of combustion [$MJkg^{-1}$]	C	[15; 25] [25; 35] [35; 50]
Fire extinction on O_2 criterion [v/v %]	D	{8; 12}
Soot production rate [%]	C	[1; 5] [5; 10] [10; 15] [15; 20]
Soot particle diameters [µm]	C	[0.005; 0.02] [0.02; 0.06] [0.06; 0.1] [0.1; 0.3] [0.3; 1]

Table 3 Parameters related to filters

Parameters	Type	Discretization
Initial filtration velocity [cm s^{-1}]	C	[1.7; 1.9] [1.9; 2.1]
Initial pressure difference through filters [Pa]	C	[250; 500] [500; 1000]

Table 4 Parameters related to the ventilation network

Parameters	Type	Discretization
Compartment volume [m^3]	C	[50; 300] [300; 700] [700; 1000] [1000; 1500]
Compartment air renewal rate [vol h^{-1}]	C	[1; 3] [3; 7] [7; 10]
Location of the inlet air vent [-]	D	{low; high}
Location of the exhaust air vent	D	{low; high}
Compartment leak rate [vol h^{-1}]	C	[0.1; 0.4] [0.4; 0.7] [0.7; 1]
Fire dampers closing times inlet/exhaust [s; s]	D	{150; 0} {150; 1800} {150; 3600} {1200; 0} {1200; 0} {1200; 1800} {1200; 3600} {∞ ; ∞}
Fire dampers aeraulic resistance in closed position [m^{-4}]	D	{10^4; 10^6}
Soot deposition rate [%]	C	[0; 20] [20; 40] [40; 60]
Dilution volume flow rate [m^3h^{-1}]	C	[4.5 10^3; 2 10^4] [2 10^4; 4 10^4] [4 10^4; 6 10^4] [6 10^4; 8 10^4] [8 10^4; 10^5]
Slope of the fan curve at functional point [-]	C	[-0.6; -0.2] [-0.2; -0.1] [-0.1; -0.01]
Regulation index [-]	D	{0; 1}

Size of the Database

To study a set of configurations, the SYLVIA software is coupled to the SUNSET software. This coupling directly allows Monte Carlo simulations. For this, a set of variables, known as study parameters, is modeled by random variables. Thus, each study parameter is associated to a variation domain and a distribution function (a uniform distribution function is used in the study). For each study parameter, a value is randomly drawn in its range of variation (or imposed as for the closing time of the fire dampers), creating a set of values for the parameters characterizing the SYLVIA calculation to be performed. By performing this simulation, one obtains the values of the responses associated with this parametric configuration. The Monte Carlo method consists in reiterating this procedure a large number of times. The storage of the values taken by the study parameters and by the responses for all the runs constitutes the SYLVIA database. The Monte Carlo simulation allows to explore the whole variation range of the study parameters and to estimate the impact of these variations on responses of interest.

The size of the database depends on the number of influential parameters and on the level of discretization of these parameters. The minimum number of SYLVIA calculations to be performed is given by the product of the highest value of the number of classes of individuals among the responses by a number of runs to have a sufficient statistic for each combination of classes. For a given response, the number of classes of individuals is the sum of all the class combinations of its influential parameters and corresponds to the number of rows of the conditional probability table associated with this response.

The identification of the influential parameters of a response is based on its correlations with the parameters determined from the Monte Carlo simulation. It was obtained from the results of a Monte-Carlo simulation performed on a sampling of 100,000 runs, by a covariance calculation. Results are reported in Table 5. Less influential parameters do not explicitly appear in the knowledge base. Nevertheless, the variability induced by these parameters is taken into account in the conditional probability tables.

Table 5 Correlations of the responses [%] according to the parameters of the study; most influential parameters are highlighted with orange background

Parameters	ΔP Filters	T° Filters	Gas Dilution Rate
Mass of fuel	0.00	0.00	0.00
Fire growth factor	0.02	0.14	0.00
Maximal HRR in open atmosphere	0.05	0.28	0.00
Heat of combustion	-0.12	-0.01	0.00
Fire extinction on O_2 criterion	-0.07	-0.02	0.00
Soot production rate	0.26	-0.01	0.00
Soot particle diameters	-0.44	0.00	0.00
Compartment air renewal rate	0.05	0.16	-0.31
Compartment volume	0.10	0.12	-0.46
Compartment leak rate	0.05	0.01	0.05
Location of the inlet air vent	0.00	0.00	0.00
Location of the exhaust air vent	0.12	0.28	0.00

Parameters	ΔP Filters	T° Filters	Gas Dilution Rate
Fire dampers closing times	0.26	0.14	0.00
Fire damper resistance	-0.17	-0.01	0.00
Soot deposition rate	-0.10	0.00	0.00
Dilution volume flow rate of gas	-0.25	-0.43	0.31
Slope of the fan curve at functional point	-0.01	0.01	0.00
Initial filtration velocity	0.01	0.00	0.00
Initial pressure difference through filters	0.38	0.00	0.00
Pressure regulation at exhaust	0.03	-0.03	0.00

For this study, based on the ten most influential parameters of the response "maximum pressure difference through the filters", the conditional probability table associated with this response contains 3 × 4 × 5 × 4 × 2 × 7 × 2 × 3 × 5 × 2 = 201,600 configurations. A base of simulations of ten million calculations guarantees over 99 % that each parametric configuration will be observed in the database. Thus, ten million runs were performed with the SYLVIA software to build the knowledge base of the expert system SEVEN. For information, this number of runs required seven full days of CPU time distributed on 144 cores.

APPLICATION OF THE EXPERT SYSTEM SEVEN

The Graph of the Knowledge Base

The graphical user interface is composed of two main sheets: the graph of the knowledge base, as shown in Figure 6, that gathers data entered for the analysis and the associated results and a sheet to visualize a priori and a posteriori likelihood in the form of histograms. The graph of the knowledge base is divided into three zones: at the top, the fourteen influential parameters of the study on which the expert system can make inferences; at the bottom, the three responses of interest on which the expert system can also make inferences; and in the center, a button to launch a query. Three columns are associated to each influential parameter and each response of interest reported on the graph of the knowledge base. The first column corresponds to the discretization of the variable, the second column to the a priori likelihoods taken by the variable and the third column to a posteriori likelihood of the variable (results of the query). The computing time required for a query is negligible.

Application

To illustrate the potential interest of an expert system as an aid tool for safety assessment, we consider the following issue: What are the configurations leading to the loss of the final level of aerosol filtration due to filter clogging during an in-cell solvent fire in a reprocessing facility? The organic phase considered in this example is composed of a solvent mixture of 30 % in volume of tributyl phosphate (TBP) and 70 % in volume of HTP (hydrogenated tetrapropylene).

If the expert system is used as a prognostic tool or in a forward chaining, the result of the expert system is rather like a direct exploitation of the database. In this case, only the knowledge relative to the parameters can be used: a fast kinetics of fire growth, a heat of combustion in the range of 25 to 35 MJ/kg, a soot production rate in the range of 10 to 15 %, soot particle sizes in the range of 0.1 to 0.3 µm according to [11] and a soot deposition rate in the range of 0 to 20 %]. The lowest class retained for the particle deposition rate upstream of the filters is justified by the size of the soot particles that corresponds to the minimum efficiency of particle deposition [12].

If the expert system is now used as a diagnostic tool or in a backward chaining, only the knowledge relative to the responses is used: a pressure difference through the filters greater than 2000 Pa, corresponding to the loss of filters.

To fully answer the question, it appears necessary to combine the forward and the backward reasoning. Three cases are here illustrated: (1) a conventional closing time of the fire dampers at 2 min and 30 s, corresponding to a servo control of the fire dampers to the automatic fire detection (cf. Figure 6); (2) a manual closing of the fire dampers by the shift personal at 20 min, to consider the case of an aleatory failure of the automatic closing of the fire dampers (cf. Figure 7) and (3) a fire damper closing time at the inlet air vent at 2 min and 30 s and a fire damper closing delay at the exhaust air vent of 30 min, to consider one of the ventilation management strategies, in a fire situation, used by operators in their safety analysis (cf. Figure 8).

The crossing of the upstream information (kinetics of fire growth, heat of combustion, soot production rate, soot particle sizes, soot deposition rate and fire dampers closing times) and downstream information (pressure difference through filters) indicates that, for a closing time of the fire dampers at 2 min and 30 s (see Figure 6), the compartment volume has low effect on filter clogging due to an early closing of the fire dampers, that 60 % of cases leading to a loss of filters are predicted for an exhaust air vent in the upper part of the fire room, that 76 % of cases are given for pre-clogged filters (initial pressure difference through filters in the range of 500 to 1000 Pa) and that 100 % of cases are predicted for fire dampers of low aeraulic resistance in a closed position and dilution volume flow rates lower than 60000 m^3h^{-1}, with 76 % of cases in the class of 4500 to 20000 m^3h^{-1}. Leaks in the compartment and through fire dampers of low aeraulic resistance contribute to maintain the fire and allow soot transfer in the ventilation network. Since the number of filters depends on the value of the dilution flow rate, low dilution flow rates lead to filter clogging in this configuration.

Fire growth factor		
slow	☐	0%
medium	☐	0%
fast	☑	100%
ultrafast	☐	0%

Maximum HRR in open atmosphere (kW)		
200-800	☑	25%
800-1500	☑	25%
1500-3000	☑	25%
3000-5000	☑	25%

Heat of combustion ($MJ\ kg^{-1}$)		
15-25	☐	0%
25-35	☑	100%
35-50	☐	0%

Fire dampers closing times Inlet (s)/Exhaust (s)		
150 / +0	☑	100%
150 / +1800	☐	0%
150 / +3600	☐	0%
1200 / +0	☐	0%
1200 / +1800	☐	0%
1200 / +3600	☐	0%
∞ / ∞	☐	0%

Fire dampers aeraulic resistance (m^{-4})		
10^4	☑	100%
10^6	☑	0%

Soot particle diameters (µm)		
0.005-0.02	☐	0%
0.02-0.06	☐	0%
0.06-0.1	☐	0%
0.1-0.3	☑	100%
0.3-1	☐	0%

Soot production rate (%)		
1-5	☐	0%
5-10	☐	0%
10-15	☑	100%
15-20	☐	0%

Soot deposition rate (%)		
0-20	☑	100%
20-40	☐	0%
40-60	☐	0%

Initial pressure difference through filters (Pa)		
250-500	☑	24%
500-1000	☑	76%

Compartment volume (m^3)		
50-300	☑	30%
300-700	☑	25%
700-1000	☑	24%
1000-1500	☑	22%

Compartment renewal rate ($vol\ h^{-1}$)		
1-3	☑	33%
3-7	☑	33%
7-10	☑	33%

Compartment leak rate ($vol\ h^{-1}$)		
0.1 - 0.4	☑	33%
0.4 - 0.7	☑	33%
0.7 - 1	☑	33%

Dilution volume flow rate ($m^3\ h^{-1}$)		
$4.5\ 10^3 - 2\ 10^4$	☑	76%
$2\ 10^4 - 4\ 10^4$	☑	19%
$4\ 10^4 - 6\ 10^4$	☑	4%
$6\ 10^4 - 8\ 10^4$	☑	0%
$8\ 10^4 - 10^5$	☑	0%

Exhaust vent location		
Low	☑	40%
High	☑	60%

Maximum pressure difference through filters (Pa)		
500	☐	0%
500-1000	☐	0%
1000-1500	☐	0%
1500-2000	☐	0%
>2000	☑	100%

Maximum gas temperature through filters (°C)		
20-50	☑	93%
50-100	☑	7%
100-150	☑	0%
150-200	☑	0%
>200	☑	0%

Initial dilution rate (-)		
5	☑	18%
5-50	☑	65%
50-500	☑	17%
500-1000	☑	0%
>1000	☑	0%

Figure 6 Results for a closing time of the fire dampers at 2 min and 30 s

If we now consider a manual closing of the fire dampers by the shift personnel at 20 min (case of an aleatory failure of the automatic closing of the fire dampers, cf. Figure 7), the expert system indicates that the size of the fire room has more effect than in the previous case (only 18 % of the cases in the range of 50 to 300 m^3 against 30 % for an early closing of the fire dampers) that 71 % of cases are predicted with a high position of the exhaust air vent, due to higher soot concentrations in the upper part of the fire room, that the percentage due to pre-clogged filters is equivalent to the previous case (78 %), that 29 % of cases are found with a high aeraulic resistance of the fire dampers, due to their late closing and that all levels of dilution of the study are involved in filter clogging, with 67 % of cases in the lowest class ranging from 4,500 to 20,000 m^3h^{-1} against 2 % of cases in the highest class ranging from 80,000 to 10,0000 m^3h^{-1}.

Consider now the case a fire damper closing time at the inlet air vent at 2 min and 30 s and a fire damper closing delay at the exhaust air vent of 30 min (see Figure 8). In this case, the expert system informs us that the compartment volume has still low effect on filter clogging in this configuration, that an early closing of the fire damper at inlet does not change significantly the percentage of cases leading to a loss of filters with an exhaust air vent in the upper part of the fire room (66 % against 60 % in case 1 and 71 % in case 2) that 80 % of cases are predicted for pre-clogged filters, a higher percentage compared to case 2 due to a longer delay of the fire damper closing at exhaust, that 81 % of cases are given for fire dampers of low aeraulic resistance in a closed position and that all levels of dilution are involved in filter clogging, with a slightly higher percentage in the lowest classes compared to case 2 (72 % of cases in the lowest class (from 4,500 to 20,000 m^3h^{-1}) against 1 % of cases in the highest class (from 80,000 to 100,000 m^3h^{-1}).

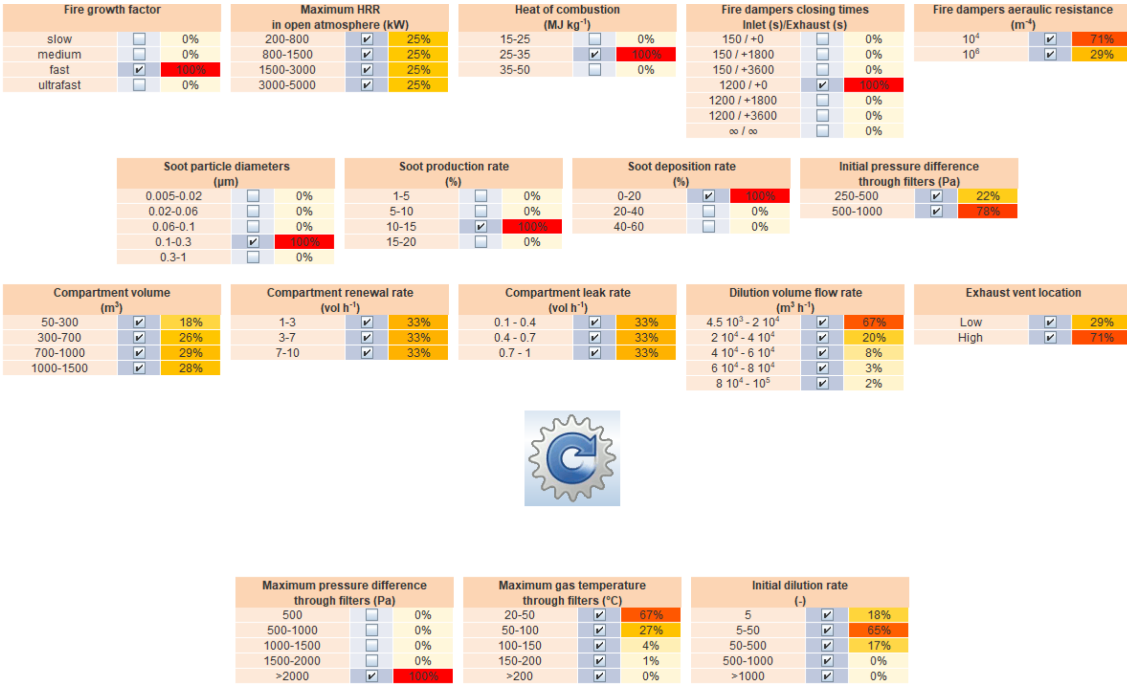

Figure 7 Results for a manual closing of the fire dampers at 20 min

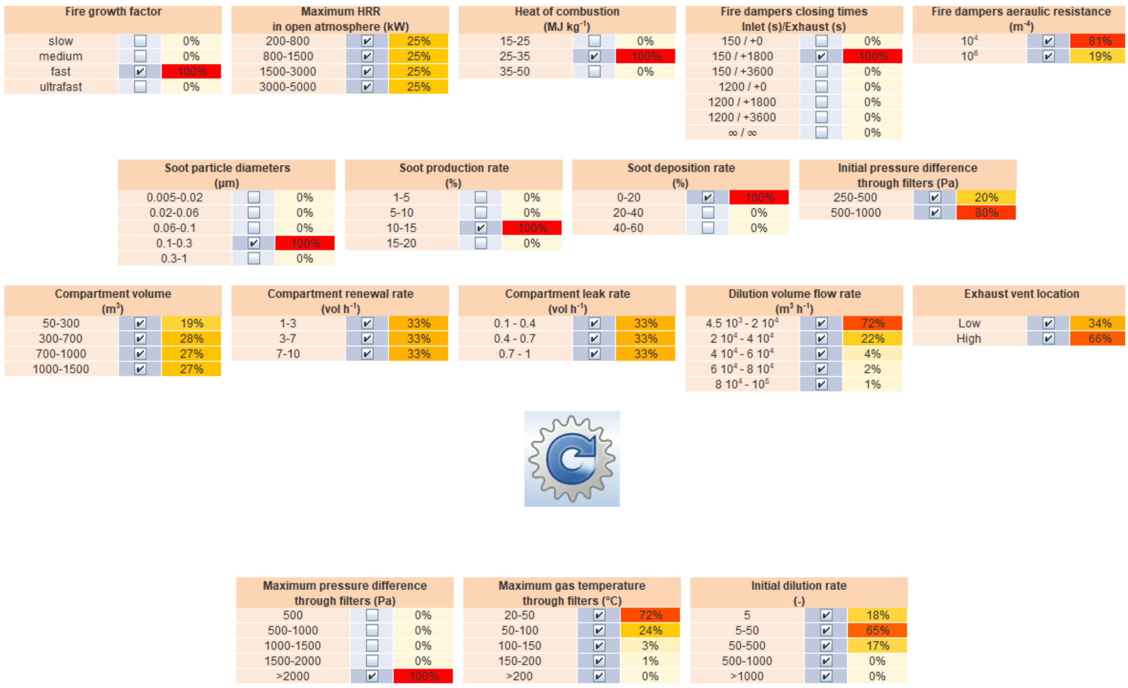

Figure 8 Results for a fire damper closing time at the inlet air vent at 2 min and 30 s and a fire damper closing delay at the exhaust air vent of 30 min

CONCLUSIONS

In fire safety assessment, it is essential to be able to quickly discern configurations at risk in a nuclear facility. For that purpose, an expert system approach, based on the Bayesian Belief Network methodology, was undertaken to take advantage of the SYLVIA software. A knowledge base including the results of ten million runs performed with the SYLVIA software was built to study the behavior of the final level of aerosol filtration in nuclear facilities, in a fire situation. The first results confirm the interest of the expert system approach in order to dynamically use large databases as part of a fire safety analysis. Indeed, it can help the identification of configurations increasing the risk for a particular scenario from the exploitation of a large database of SYLVIA runs.

The perimeter of the knowledge base determines the scope of the expert system. If the framework of the study were to change, it would then be necessary to integrate the new generic knowledge (enrichment of the knowledge base). For any other safety assessment needs the database, the identification of the responses of interest and their influential parameters as well as the characterization of the conditional probability tables are likely to be different. However, the general framework is generic. Thus, for new issues coming from some fire safety analysis, another study done with SYLVIA / SUNSET software may be necessary. Nevertheless, the algorithmic part of the expert system (the inference engine) is, in principle, unchanged, but may, however, need to be adapted in order to take into account the characteristics specific to the new question of interest in terms of parameters and responses.

In the approach of the assessment of the fire source term by simulation tools, the next step would be to develop "satellite" expert systems for the fire room and the ventilation network that would be plugged to the SEVEN expert system. These expert systems would integrate the geometry and the specificities of the installation to be studied. This step would make it possible to calculate the soot deposition in the fire room and the ventilation network as well as thermal exchanges. The ultimate step would consist in introducing radionuclides in the SYLVIA simulations in order to calculate the fire source term.

To achieve the coupling of different expert systems, a possible solution would consist in introducing intermediate responses to aggregate the effect of subsets of upstream parameters. This approach has the advantage of increasing the number of modelled parametric configurations while maintaining the computational efficiency but requires identifying and validating appropriate intermediate responses. A difficulty lies in taking into account the feedback of the filter clogging effects on the ventilation and the fire, such as the rise in pressure.

The development of Bayesian Belief Network tools based on large simulation database can be considered as a complementary way to take advantage of the SYLVIA software allowing an expert to quickly target the configurations at risk in a specific safety assessment. Moreover, the computing time required by such expert systems being negligible, this kind of tools can be highly profitable for training. To the authors´ opinion, expert systems represent a new generation of computational tools in the field of probabilistic fire simulation.

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REFERENCES

[1] Plumecocq W., et al.: Numerical method for determining water droplets size distributions of spray nozzles using a two-zone model, Nuclear Engineering and Design 324, pp. 67-77, 2017.

[2] Chojnacki, E., and A. Ounsy: Description of the IPSN method for the uncertainty and sensitivity analysis and the associated software: SUNSET, in: Proceedings of ASME/JSME ICONE, Volume 3, pp. 545-550, New Orleans, LA, USA, 1996.

[3] Chojnacki, E., W. Plumecocq, and L. Audouin: An expert system based on a Bayesian network for fire safety analysis in nuclear area, Fire Safety Journal 105, pp. 28-40, 2019.

[4] Jackson, P.: Introduction to expert systems (International Computer Science Series), ISBN: 9780201876864, Addison-Wesley Pub. Co., Wokington, United Kingdom, 1999.

[5] Helton, J. C., and F. J. Davis: Latin Hypercube Sampling and the Propagation of Uncertainty in Analyses of Complex Systems, Sandia National Laboratories (SNL), Albuquerque, NM, USA, 2002.

[6] Pearl, J.: Causality, Cambridge University Press, Cambridge, MA, USA, 2013.

[7] Pearl, J.: Probabilistic Reasoning in Intelligent Systems, Morgan Kaufmann Publishers Inc., Burlington, MA, USA, 1988.

[8] International Organization for Standardization (ISO): ISO 17873:2004, Nuclear facilities – Criteria for the design and operation of ventilation systems for nuclear installations other than nuclear reactors, Geneva, Switzerland, 2004, https://www.iso.org/standard/37257.html.

[9] Quintiere, J. G.: Principles of fire behavior, ISBN-13: 978-1498735629, CRC Press; 2nd Ed., August 2016.

[10] Mocho, V. M., and F. X. Ouf: Clogging of Industrial Pleated High Efficiency Particulate Air (HEPA) Filters in the Event of Fire, Nuclear Engineering and Design, 241, 5, pp. 1785-1794, 2011.

[11] Ohno, T., et al.: Rapid Clogging of High-Efficiency Particulate Air Filters During In-Cell Solvent Fires at Reprocessing Facilities, Nuclear Technology, 2019, DOI: https://doi.org/10.1080/00295450.2019.1620057.

[12] Hinds, W. C.: Aerosol Technology, 2nd Ed., ISBN: 978-0-471-19410-1, John Wiley & Sons, Hoboken, NJ, USA, January 1999.

Implementation Strategies of a Semi-Empirical Cable Fire Model in the FDS Fire Simulation Code

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ABSTRACT

This study focuses on implementation strategies of a semi-empirical model for fire spread and propagation on a vertical stack of multiple horizontal cable trays in the computational flu-id dynamics (CFD) based fire modeling tool FDS (*F*ire *D*ynamics *S*imulator). The semi-empirical model is based on the FLASH-CAT (*fla*me spread over *h*orizontal *ca*ble *t*rays) model. In contrary to the FLASH-CAT model, the semi-empirical model can appropriately reflect the effects of the heat transfer deterioration due to the local oxygen depletion and the heat transfer enhancement due to the presence of structures such as a wall and/or ceiling. A key strategy for the implementation of this semi-empirical model within FDS is to divide both outer parts of each cable tray used for horizontal spread areas into multiple discrete areas with variable spreading rates and peak heat release rate per unit area evaluated within the model.

The results of this study could maximize the advantages of the semi-empirical model in respect that the model is implemented in FDS, which can provide more accurate and detailed predictions of fire environments, particularly under more complex and real conditions such as a fire of a vertical stack of multiple horizontal cable trays near a wall and/or ceiling in a non-rectangular compartment with no or a wide range of ventilation.

INTRODUCTION: FLASH-CAT MODEL

The FLASH-CAT model (cf. [1], [2] and [3]) is a simple flame spread model for horizontal tray configurations. It has been developed based on the basic approaches in Appendix R of NUREG/CR-6850 [1] and the small and intermediate-scale experimental data summarized in NUREG/CR-7010, Vol. 1 [2]. The FLASH-CAT model was compared to the results of 26 multiple tray experiments [2] and 16 vertical tray and corridor experiments [3], and it was observed that the heat release rates (HRRs) predicted by the FLASH-CAT model are similar to or larger than the measured HRRs. Figure 1 shows an example of the FLASH-CAT model application.

One of the most notable assumptions made for the FLASH-CAT model is that the cable trays are positioned in an open environment, which means they are not installed directly below a ceiling or in front of a wall; or confined within a relatively narrow corridor or shaft. The results of the corridor experiments indicate that the ceiling caused the formation of a hot gas layer (HGL) near the horizontal cable trays; the HGL preheated the cables in front of the spreading fire; and consequently, spreading rates of up to ten times larger than those observed in the open environmental experiments were reached. Based on this experimental observation, the FLASH-CAT model simply applies ten times enhanced fire spreading rates for fire scenarios where the HGL temperatures are expected to exceed 200 °C considering the cable trays near a ceiling or a wall. Because use of these simple enhanced fire spreading rates having a considerable amount of uncertainty, NUREG/CR-7010 Vol. 2 [3] suggests that the analysist should consider a range of spreading rates to appropriately determine the HRR profiles of a cable tray fire.

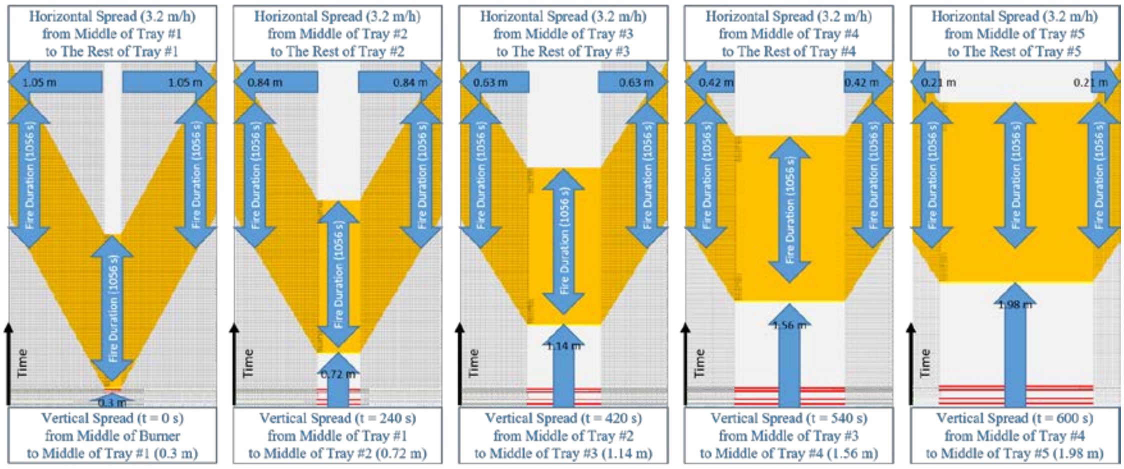

Figure 1 An example of the FLASH-CAT model application

SEMI-EMPIRICAL MODEL

The semi-empirical cable fire model proposed and implemented in the two-zone based fire modeling tool SYLVIA by W. Plumecocq, et al. [4] is used as a reference model of this study. The basic approaches of the model employed in this study are the same as those of the reference model. The presence of structures such as walls and/or ceilings may enhance the heat transfer from the hot gas plume to the unburnt cables, which in turn increases the local temperatures near the corresponding cable trays. This consequently increases the spreading rates. Fires under the confined and ventilated conditions may undergo local oxygen depletion, which in turn deteriorates the heat transfer from the flame to the fuel surfaces. This consequently decreases the fire spreading rates as well as the fuel mass loss rates (MLRs), and thus also the heat release rates (HRRs). The main advantage of the semi-empirical model is that these effects are appropriately reflected in the model.

The semi-empirical model basically utilizes experimental data and empirical approaches used in the FLASH-CAT model. On the other hand, the model also makes full use of analytical approaches and additional experimental observations, especially in evaluating time evolutions of the fire spreading and propagation and the heat release rate per unit area (HRRPUA) as well. This enables the model to appropriately reflect the effects of the heat transfer deterioration due to the local oxygen depletion and the heat transfer enhancement due to the presence of structures (walls and/or ceilings), which is a distinct advantage over the FLASH-CAT model.

The horizontal outward spread ingrates along the length of each cable tray were evaluated based on the spread law proposed by J. Quintiere [5] by additionally taking into account the effects of changes in local oxygen concentrations and temperatures near the corresponding cable trays instead of using the constant values recommended in the FLASH-CAT model depending on the cable materials (0.3 mm/s for thermoset cables and 0.9 mm/s for thermoplastic cables) [2]. The ignition times of each cable tray (for vertical upward fire propagation) were evaluated by directly calculating heat transfer and the resulting temperatures at the internal sheath surfaces of each cable tray and comparing them to the ignition temperature of the cable instead of using the “minute rule” recommended in the FLASH-CAT model [1], [2]. The peak HRRPUA of the cable tray was evaluated based on the bench-to-full-scale empirical correlation proposed by Lee [6] by additionally taking into account the effects of changes in local oxygen concentrations near the corresponding cable trays instead of using the constant values recommended in the FLASH-CAT model depending on the cable materials (150 kW/m² for thermoset cables and 250 kW/m² for thermoplastic cables) [2]. The effects of local oxygen concentrations, or more precisely, the effects of local oxygen depletion, were considered in the formulas in the form of a correction factor represented as a linear function of the dimensionless fuel MLR ranging from 0 to 1 according to the oxygen volume fraction available for combustion near the burning surface ranging from 11 % to 21 % based on the oxygen-limiting law proposed by Peatross and Beyler [7].

FDS IMPLEMENTATION STRATERIES

As one might expect, the implementation of this semi-empirical model in the FDS requires more complex and difficult techniques than that of the FLASH-CAT model within FDS does. Figure 2 summarizes some major differences between two implementation models. Among differences between two implementation models, the most notable one is about setting up the both outer parts of each cable tray as fire spreading areas. In the FLASH-CAT implementation model, each of these parts is set up as a single continuous area with constant spreading rate and peak HRRPUA values recommended in the FLASH-CAT model depending on the cable materials. In the semi-empirical implementation model, on the other hand, it is divided into multiple discrete areas with variable spreading rate and peak HRRPUA values evaluated within the model. This is to effectively and efficiently extract the location and time dependent information, i.e., the local oxygen concentrations and temperatures at each time step, and reflect that information to evaluating the spreading rates and peak HRRPUAs.

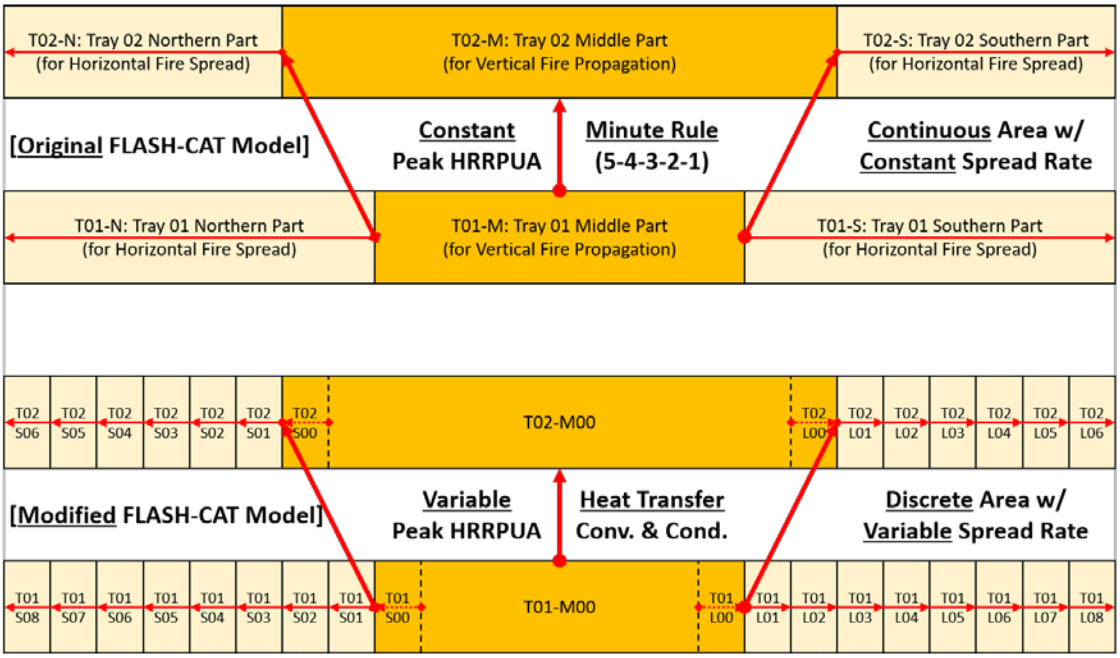

Figure 2 Key implementation strategies of the FLASH-CAT model and semi-empirical model

The model simulates vertical fire propagation and extinguishing, i.e., the model activates and deactivates the middle part of each cable tray as a fire area by repeating the following simplified steps:

(1) The model activates fire in T01-M00 (T01-M00 “ON”).
(2) The model activates fire in T01-S00 and L00 (T01-S00 and L00 “ON”) at the same time.
(3) The model calculates the heat transferred (A) to T02-M00.
(4) The model calculates the surface / inside wall temperature (B) of T02-M00 using (A).
(5) The model compares (B) to the ignition temperature (C) of T02-M00.
(6) If (B) is larger than and equal to (C) (B ≥ C), then the model activates fire in T02-M00 (T02-M00 “ON”).
(7) The model calculates the oxygen volume fraction (D) near T02-M00.
(8) The model calculates the heat release rate of T02-M00 and its time integration (total amount of heat release) (E) of T02-M00 using (D).
(9) The model compares (E) to the available amount of heat release (F) of T02-M00.
(10) If (E) is larger than and equal to (F) (E ≥ F), then the model deactivates fire in T02-M00 (T02-M00 “OFF”).
(11) The model repeats step (3) – (10) for T03-M00, T04-M00, … and Tn-M00.

The model simulates horizontal fire spreading and extinguishing, i.e., the model activates and deactivates the outer parts of each cable tray as fire areas by repeating the following simplified steps:

(1) The model activates fire in T01-M00 (T01-M00 “ON”).
(2) The model activates fire in T01-N00 and S00 (T01-N00 and S00 “ON”) at the same time.
(3) The model calculates the oxygen volume fractions and gas temperatures (A) near T01-N00 and S00.
(4) The model calculates the fire spread rates and their time-integration (total fire spread length) (B) of T01-N00 and S00 using (A).
(5) The model compares (B) to the segment lengths (C) of T01-N00 and S00.
(6) If (B) is greater than and equal to (C) (B ≥ C), then the model activates fire in T01-N01 and S01 (T01-N01 and S01 “ON”).
(7) The model calculates the oxygen volume fractions (D) near T01-N01 and S01.
(8) The model calculates the heat release rates of T01-N01 and S01 and their time-integration (total amounts of heat release) (E) of T01-N01 and S01 using (D).
(9) The model compares (E) to the available amounts of heat release (F) of T01-N01 and S01.
(10) If (E) is greater than and equal to (F) (E ≥ F), then the model deactivates fire in T01-N01 and S01 (T01-N01 and S01 “OFF”).
(11) The model repeats step (3) – (10) for T01-N02 and S02, T01-N03 and S03, … and T01-Nn and Sn.

CONCLUSIONS

This study provides and describes implementation strategies of a semi-empirical model for fire spreading and propagation on a vertical stack of multiple horizontal cable trays in the CFD-based fire modeling tool FDS. The key implementation strategy presented in this study is to divide both outer parts of each cable tray used to simulate horizontal fire spread areas into multiple discrete areas with variable spread rate and peak HRRPUA values evaluated within the model. The FDS semi-empirical model simulates fire spreading and propagation on a vertical stack of multiple horizontal cable trays as follows:

- The model simulates vertical fire propagation by calculating the heat transferred to and surface / inside wall temperature of the next above non-activated middle part; comparing the latter one with the ignition temperature of the cable in the next above middle part; and then activating fire in the next above middle part.
- The model simulates horizontal fire spreading by calculating the fire spreading rate and time integral of it, i.e., total fire spreading length for the current activated segment; comparing the latter one with a length of the current segment; and then activating fire in the next adjacent segment.
- The model simulates fire extinction (or fire duration) by calculating the heat release rate and time integral of it, i.e., total amount of heat release for the current activated middle part or segment; comparing the latter one with available amount of heat release of the current middle part or segment; and then deactivating fire in the current middle part or segment.

The results of this study could maximize the advantages of the semi-empirical model in respect that the model is implemented in the CFD-based fire modeling tool FDS which can provide a more accurate and detailed prediction of fire environments, especially under more complex and real conditions such as a fire of a vertical stack of multiple horizontal cable trays near a wall and/or ceiling in a non-rectangular compartment with no or a wide range of ventilation.

ACKNOWLEDGEMENTS

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REFERENCES

[1] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 (EPRI 10191989), Washington, DC, and Palo Alto, CA, USA, 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/.

[2] McGrattan., et al.: Cable Heat Release Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE) Phase 1: Horizontal Trays, NUREG/CR-7010, Vol. 1, United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC, USA, July 2012, https://tsapps.nist.gov/publication/get_pdf.cfm?pub_id=905788.

[3] McGrattan., et al.: Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Phase 2: Vertical Shafts and Corridors, NUREG/CR--7010, Vol. 2, United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC, USA, March 2013, https://tsapps.nist.gov/publication/get_pdf.cfm?pub_id=913000.

[4] Plumecocq, W., L. Audouin, and P. Zavaleta: Horizontal cable tray fire in a well-confined and mechanically ventilated enclosure using a two-zone model, Fire and Materials, Vol. 43, Issue 5, pp. 1-13, Wiley and Sons, Hoboken, NJ, USA, 2019, https://doi:10.1002/fam.2698.

[5] Quintiere, J., A Simulation Model for Fire Growth on Materials Subject to Room Corner Test, Fire Safety Journal, 20, pp 313-339, 1993.

[6] Lee, B. T.: Heat Release Rate Characteristics of Some Combustible Fuel Sources in Nuclear Power Plants, NSBSIR 85–3195, National Bureau of Standards, Gaithersburg, MD, USA, July 1985, https://tsapps.nist.gov/publication/get_pdf.cfm?pub_id=100178.

[7] Peatross, M. J., and C. L. Beyler: Ventilation effect on compartment fire characterization, Fire Safety Science 5, pp. 403-414, doi:10.3801/IAFSS.FSS.5-403, 1997, http://www.iafss.org/publications/fss/4/205.

3.5 Session on “Fire Safety Analysis and Modelling”

The session on fire safety analysis and modelling was devoted to one of the major topics of this seminar, providing insights on fire safety assessment of nuclear installations supported by probabilistic analyses.

The first contribution by Appendix R Solutions, USA focussed on the underlying elements of a successful fire probabilistic safety assessment (so-called Fire PSA).

The need of in-depth investigations for improving the development of probabilistic fire event analyses and their quantification within a Fire PSA model and the corresponding enhancements in the near past were presented by KAERI, Republic of Korea. The insights from this study will reduce the level of conservatism within probabilistic analyses of fires. The third contribution by U.S. NRC also pointed out the importance of improving the level of realism in Fire PSAs in order to further increase the level of confidence in this tool for recent nuclear power plant applications, which still is lower for hazards than for plant internal events. An important step forward in the United States is the guidance meanwhile available for fire growth and suppression to be modelled within PSA.

The last presentation by Carleton University in Canada provided a proposal for a Fire PSA model specifically developed for CANDU type reactors including the first results of its application to a corresponding case study for a selected fire zone.

The Seminar contributions of this session are provided hereafter.

Foundations for A Successful Fire PRA Project

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ABSTRACT

A common topic in nearly every nuclear market today is Fire PRA (*P*robabilistic *R*isk *A*ssessment) realism and whether the chosen methodology accurately characterizes plant risk. What is often overlooked is that perhaps the greatest risk to the Fire PRA project is the quality of the supporting foundation. Because the Fire PRA follows other work such as the internal events PRA model, Appendix R Safe Shutdown Analysis, re-start safety evaluation, etc. it is natural for it to be built on these foundations.

A review of the quality of the building blocks and the way they are applied can not only improve project schedule and reduce costs but improve the accuracy of the Fire PRA results. PRAs are inherently iterative in nature, through the implementation of the following items the number of iterations are expected to be reduced; Fire PRA Readiness Assessment / Gap Analysis, Gap Analysis Resolution Plan, Implementation of the Gap Analysis Resolution Plan. The reduction in the number of iterations will have a direct impact on improving the project schedule and thereby reducing costs.

This paper will provide insights into why a strong foundation is critical to a successful Fire PRA.

INTRODUCTION

This paper provides a summary of insights gained from conducting Fire PRAs (*P*robabilistic *R*isk *A*ssessments) for many U.S. plants. About half of the U.S. fleet has transitioned to an NFPA 805-based licensing basis [1] – of which a key element is development of a Fire PRA. Nuclear Energy Institute document NEI 04-02 [2] provides guidance for implementing a risk-informed, performance-based fire protection program under 10 CFR 50.48 [3]. The United Stated Regulatory Commission (U.S. NRC) published Regulatory Guide 1.205 [4] to provide a regulatory framework for risk-informed, performance-based fire protection for existing light-water nuclear power plants. License Amendment Requests (LARs) submitted to the U.S. NRC) for the transitioning plants document that 100 % of the transitioning plants made risk significant improvements based on the insights from the Fire PRA. Around the globe, nuclear utilities and regulatory bodies are setting expectations for Fire PRAs to identify risk insights for plant improvements and to risk-informed decision making. In addition, many U.S. plants are now pursuing Fire PRAs as part of the 10 CFR 50.69 [5] initiative to implement the benefits of risk-informed engineering programs (10 CFR 50.69) and right-sizing regulatory requirements on the structures, systems, and components (SSCs) based on their impact on the plant risk.

Fire PRAs, typically, build on the work that has already been completed, such as the internal events (IEPRA) model, Appendix R Safe Shutdown Analysis, Restart Safety Evaluation, Fire Hazard Analysis (FHA), fire characteristics table, operations procedures, Manual action feasibility studies, and plant modifications to resolve separation issues. To fully realize the substantial benefit of using the legacy/existing data, the data should be carefully examined and adapted for use in a Fire PRA to ensure technical adequacy/appropriateness for such usage. It is important to note that Fire PRA is an analytical tool that provides an integrated assessment of the fire hazard based on the physical characteristics and configuration of a plant in combination with its operational, emergency, and maintenance practices and procedures. Therefore, like any other analytical tool, the sophistication of the Fire PRA depends on its intended use, which impacts the needed output, and which in turn dictates the complexity of the Fire PRA. A strategic decision needs to be made with regards to its short- and long-term intended use, which will enable a cost-effective approach to be used for timely completion of the tool.

The following figure from NUREG/CR-6850 [6] shows the highly dependent nature of the work. The basic methodology is meant to apply increasingly detailed analysis, resulting in a gradual cost increase only where necessary.

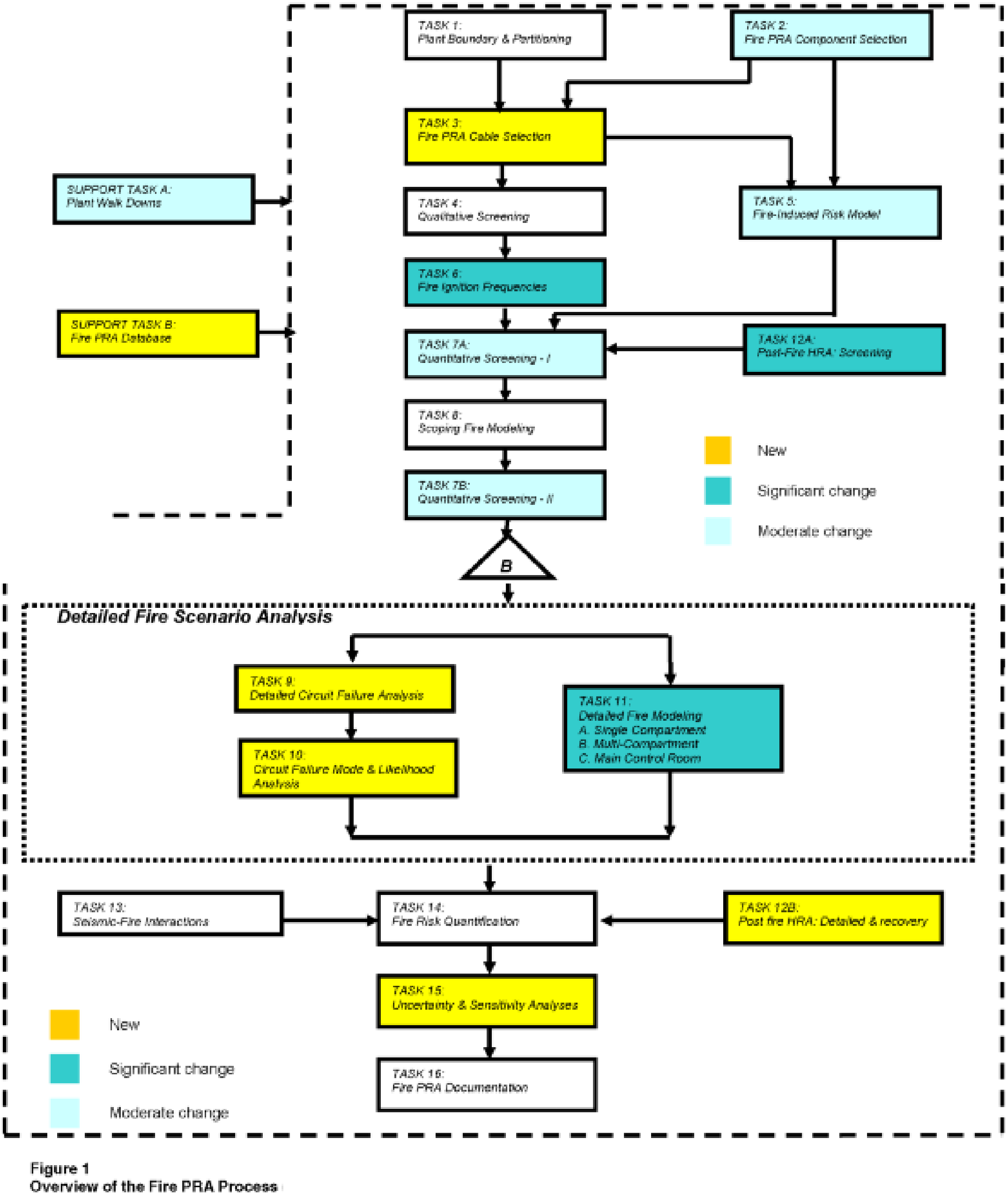

Figure 1 Overview over the Fire PRA process

REVIEW OF THE BUILDING BLOCKS FOR FIRE PRA

The following is a brief review of several of the early Fire PRA tasks (NUREG/ CR-6850 [6]) and a review of benefits/risk of using legacy data (where applicable) for Fire PRA tasks. The discussion presented here is based on the challenges that have been faced by many utilities within the United States and abroad that have developed Fire PRAs in support of new regulatory expectations, transitioning to the NFPA 805-based fire protection program or in support of taking advantage of other advanced risk-informed performance-based programs (such as transitioning to the 50.69 program, detailed information and guidance can be found in [7] to [11]). These advanced risk-informed performance-based programs require a highly-sophisticated Fire PRA. Therefore, the input information quality and the base model (internal events model) have to be of high technical adequacy and have to undergo under significant adaptation. Finally, the insights and recommendations in this paper will have different levels of deployment impact between those organizations that have the same starting point and those organizations that have a different starting point and intended use.

Task 1 "Plant Boundary Analysis and Partitioning"

"For the purposes of a Fire Probabilistic Risk Assessment (PRA), the plant is divided into a number of fire compartments. The analysis then considers the impact of fires in a given compartment, and fires that might impact multiple compartments. This procedure establishes the process for defining the global plant analysis boundary and partitioning of the plant into fire compartments. The product of this task will be a list of plant fire compartments in the nuclear power plant under analysis." [6]

The work in Task 1 is typically based on the existing plant Fire Hazards Analysis (FHA) and fire area drawings, but the methodology also allows the analyst to subdivide an area based on additional criteria that indicate the "*heat and products of combustion from a fire within the enclosure will be substantially confined*". Experience has shown that taking credit for these additional criteria, which substantially reduce the likelihood of whole-room burnup, will be very significant in the final risk calculations. An experienced analyst will use the legacy information and make appropriate revisions to achieve the benefits from the legacy data without unnecessary rework.

Key insights for this task that were not present in the original legacy documents include:

- Identification of non-rated or less-than-3-hour-rated barriers that can be shown to substantially contain a fire;
- Plant walkdowns to confirm compartment boundaries.

Potential Project Impact – Due to Deficiencies in Task 1 Without Credit for Additional Non-rated Barriers

The need to revise Task 1 would most likely be identified in Task 7 "Quantitative Screening", or in Task 14 "Fire Risk Quantification". The defined need would be based on an unacceptably high core damage frequency (CDF) / conditional core damage probability (CCDP), which would be driven by the failure of too many Fire PRA significant cables and/or components. A review of Task 1 in more detail may identify where additional partitions could be credited as part of the foundation for the Fire PRA.

If Task 1 changes are identified during Task 7, the following tasks would require rework:

- Task 1 "Plant Boundary Analysis and Partitioning"
  - Task 1 would have to be reworked to take credit for additional divisions within a fire area.
- Task 3 "Cable Selection"
  - As part of Task 3, once the cable selection is completed, the analyst identifies the route of the cables and maps the cables to the associated fire compartments. If the fire compartment definitions are later changed, rework of Task 3 cable routing will be required to map the cables. In some cases, this could require repeating costly plant walkdowns.
- Task 6 "Ignition Frequencies"
  - Similar to the cable selection in Task 3, a change in fire compartments will require a revision to the ignition frequency calculation to map the ignition sources to the new fire compartments. This would likely also require repeating plant walkdowns.
- Task 8 and Task 11 "Fire Modeling"
  - These tasks are examples of the more detailed (and expensive) work that typically needs to be performed for only a selection of components. Generally, the results of Task 7 are used to determine where to spend the additional resources to help refine the model. If Task 1 does not adequately subdivide fire areas to take credit for barriers that "substantially contain" the fire, it is likely that additional work performed in Task 8, Task 9, Task 10, and Task 11 may not have been necessary.

Task 2 "Fire PRA Components Selection"

"This list serves as the basis for those components modeled in the Fire PRA, and it is the key source of information for which corresponding cables need to be identified and located for the Fire PRA. As such, the Fire PRA Component List, Fire PRA Model, and corresponding cable identification are iterated upon to ensure an appropriate correspondence among these three items. The product of this task is a list of the equipment to be included in the Fire PRA and for which corresponding cables need to be identified and located for the nuclear power plant under analysis." [6]

As described in NUREG/CR-6850, this activity includes a systematic review to identify the following:

- Equipment whose fire induced failure will cause an initiating event to be modeled in the Fire PRA Model,
- Equipment to support the success of mitigating safety functions credited in the Fire PRA,
- Equipment to support the success of operator actions credited in the Fire PRA,
- Equipment whose spurious actuation or other fire-induced failure modes could have an adverse effect on the success of the mitigating safety functions credited in the Fire PRA, and
- Equipment whose spurious operation or other fire induced failure modes could likely induce inappropriate or otherwise unsafe actions by the plant operators during a fire damage sequence.

The importance of a systematic review of Task 2 "Components Selection", and initial HRA component selection (Task 12) is that it identifies the components that will be analyzed in subsequent tasks. Early identification of components allows for efficient performance of subsequent tasks such as Task 3 “Cable Selection and Routing”. At the same time, an experienced analyst will be able to screen out components that have low-risk benefit, which has an immediate benefit of scope reduction. Failure to screen out the extra (not needed) components has the potential to significantly increase scope.

Potential Project Impact – Due to Deficiencies in Task 2, Components Selection

The need to revise Task 2 would most likely be identified in Task 7 “Quantitative Screening”, or in Task 14 “Fire Risk Quantification”. Deficiencies in Task 2 would result in project schedule/cost in the following areas:

- Task 3 “Cable Selection”
  - If Task 2 components are not identified early in the project, it creates inefficiency in Task 3 “Cable Selection” work. The cable selection task is most efficient if it can be performed over a relatively short period of time by experienced analysts. Additional inefficiencies are also introduced if the components are identified after completion of cable selection, which then requires additional site-specific training of the analysts, additional supervision and management, and document revision costs.
  - If extra, not needed components are included in Task 2, the result is an immediate increase in cable selection scope. An example of this is instrumentation. A relatively small set of instrumentation is needed to support successful operator actions (e.g., carrying out the emergency operating procedures (EOPs), following specific fire emergency procedures (FEPs), etc.). The addition of “extra” instruments has an immediate impact on Task 3 cable selection scope. This has the potential to add hundreds or more “instruments” that are not needed to support the Fire PRA.
- Task 5 “PRA Model” could also be affected if the component is not previously included (IEPRA shortcuts typically omit several components).
- Task 8 and Task 11 “Fire Modeling”
  - These tasks are examples of some of the more detailed (and expensive) work that typically needs to be performed for only a selection of components. Generally, the results of Task 7 are used to determine where to spend the additional resources to help refine the model. Depending upon how “extra” components are tied to basic events, the result could be to create a significant number of additional failures (depending upon whether they are screened out through detailed reviews before or after fire modeling). This could significantly increase the scope and cost of fire modeling. In addition, failure to include all the required components needed could result in a higher indicated risk resulting in additional detailed fire modeling.
- Task 9, Detailed Circuit Analysis, and Task 10 “Circuit Failure Mode Likelihood Analysis”
  - These tasks are generally performed to help refine the Fire PRA and to reduce risk for circuits that are not separated by 3-hour fire barriers. As discussed above, the addition of unnecessary components, depending upon the Fire PRA logic, could be to create a significant number of additional failures (depending upon whether they are screened out through detailed reviews before or after Task 9 and 10). This could significantly increase the scope and cost of Tasks 9 and 10.
- Task 7 “Quantitative Screening”, and Task 14 “Fire Risk Quantification”
  - As discussed above, the addition of unnecessary components, depending upon the Fire PRA logic, could be to create a significant number of additional failures.

This could significantly increase the scope and cost of cutset reviews for Tasks 7 and 14.

Task 3 "Fire PRA Cable Selection"

"The Fire PRA Cable List identifies the circuits/cables needed to support proper operation of equipment contained in the Fire PRA Equipment List. Essential electrical power supplies are also identified during this task. The Fire PRA Cable List might also include Associated Circuits. Associated Circuits are cables that are not necessarily directly linked to a component but have the potential to cause improper operation of a component as a result of certain failure modes associated with fire-induced cable damage. The Fire PRA Cable List is not simply a list of cables. It also establishes, for each cable, a link to the associated Fire PRA component and to the cable's routing and location. These relationships provide the basis for identifying potential equipment functional failures at a fire area, fire compartment, or raceway level." [6]

The legacy data related to Task 3 "Cable Selection" is perhaps one of the greatest schedule and cost savings opportunities in the Fire PRA. It has been estimated that up to 70 % of cable selection and cable routing tasks may have already been completed for plants with a quality Safe Shutdown Analysis or Restart Safety Evaluation. The quality of this work is a key determining factor in whether it can be used for Fire PRA. The legacy cable selection / cable routing work may have been performed by a number of different organizations, over a long period of time, and using different methodologies. Due to aging plant staff, it is also possible that sufficient knowledge transfer has not occurred within organizations, which may leave newer less experienced staff thinking that this data needs to be developed from scratch versus relying on previous plant reports.

The cable selection and routing information is one of the fundamental building blocks of the Fire PRA. The following are examples of issues with legacy cable selection and cable routing data that can significantly affect subsequent tasks in the Fire PRA:

- Methodology – In order to meet the requirements of ASME/ANS RA-Sa-2009, the Fire PRA must document the cable selection and location process and results. In addition to being a key element to assure quality, a documented methodology is essential to meeting the ASME/ANS requirements.
- Missing circuits – This item is related to the methodology item above. In some cases, circuits that could adversely affect the Fire PRA components or basic events, such as circuits related to interlocks, may have been omitted from the cable selection in the legacy data.
- Missing or erroneous cable routes – In some cases, legacy routing data may include incomplete cable routes.
- Extra or unnecessary circuits – In some cases, the legacy cable data may include associated cables that do not adversely affect the Fire PRA component or basic event.
- Errors in cable associations – Errors in cable associations to components or basic events can adversely affect the Fire PRA result.

Potential Project Impact – Due to Deficiencies in Task 3 "Cable Selection"

If the cable selection methodology and documentation is not reviewed as part of Task 3, the likely phase of the project where the errors would be identified is in Task 7, Quantitative Screening, or Task 14 "Fire Risk Quantification". Deficiencies in Task 3 would result in increased project schedule/cost in the following areas:

- Task 3 “Cable Selection”

  This would cause rework of cable selection / cable routing
- Task 8 and Task 11 “Fire Modeling”
  - These tasks are examples of some of the more detailed (and expensive) work that typically needs to be performed for only a selection of components. Typically, high-risk areas or risk-significant sequences are identified for additional refinements, including fire modeling. Because the cable selection tasks identifies the targets of concern, changes in the Task 3 results could invalidate the work performed for a given fire model and require a complete rework of associated fire models.
- Task 9 “Detailed Circuit Analysis” and Task 10 “Circuit failure Mode Likelihood Analysis”
  - These tasks are generally performed to help refine the Fire PRA and to reduce risk for circuits that are not separated by 3-hour fire barriers. Similar to the fire modeling discussion above, changes in the Task 3 results could invalidate the Task 9 and 10 work performed and require significant rework.
- Task 7 “Quantitative Screening” and Task 14 “Fire Risk Quantification”
  - Because the Task 3 results are fundamental building blocks which drive which components are potentially fire affected for each fire compartment, changes in the Task 3 results can result in significant rework to Task 7 “Quantitative Screening” and Task 14 “Fire Risk Quantification”, including any reviews of risk significant sequences (such as cutset reviews).

Task 5 “Fire Induced Risk Model”

“The primary objective of this task is to provide an approach that allows the user to configure or modify the Internal Events PRA model to quantify fire-induced CDF, LERF, CCDP, and CLERP.” [6]

The fire induced risk model is constructed largely from the Internal Events PRA and is a key building block of the Fire PRA. As such, it is very important to have an experienced Fire PRA analyst review the risk model early in the project to make sure the model structure is adequate for Fire PRA. Some examples of issues identified in Fire PRA Models (possibly carryover from IEPRA) include:

- An example of oversized basic event is the RCIC pump start failure. This pump’s start failure can result from the mispositioning of five (5) motor-operated valves or the governor valve. The model would benefit from this larger basic event being made up of smaller basic events in the Fire PRA model. These smaller basic events would focus on each valve that could cause the pump to fail to start, for example, “valve MVxxx-22 fails to open”.
- An example of a basic event whose analysis is too focused is the “MVxxx-5A Limit Switch 5A - not activated”. Typically, limit switches themselves would not be cable selected or assigned a basic event.
- In this example, the initial review showed a single cable was captured for this basic event in the internal events database. The cable that was selected relays the position of the valve to another circuit. The failure of the originally selected cable has no impact on the valve MVxxx-5A itself. If the position of MVxxx-5A affects another component in the Fire PRA model, then that relationship should be made in the model via the valve analysis rather than be limited to the limit switch. In that case, all cables that could cause the mispositioning of the valve will be captured in a way that

conveys the failure of the interlocked component. That cable would then be mapped to the other component.

Potential Project Impact – Due to Deficiencies in Task 5 "Fire Induced Risk Model"

If significant changes in the model are necessary later – after Task 7, there is a significant impact on other tasks.

- Task 8 and Task 11 "Fire Modeling"
  - These tasks are examples of some of the more detailed (and expensive) work that typically needs to be performed for only a selection of components. Typically, high risk areas or risk significant sequences are identified for additional refinements, including fire modeling. Any significant changes in the Fire PRA model could invalidate the work performed for a given fire model and require a complete rework of associated fire models.
- Task 9 "Detailed Circuit Analysis" and Task 10 "Circuit Failure Mode Likelihood Analysis"
  - These tasks are generally performed to help refine the Fire PRA and to reduce risk for circuits that are not separated by 3-hour fire barriers. Similar to the fire modeling discussion above, changes in the Fire PRA model Task 5 could invalidate the Task 9 and Task 10 work performed and require significant rework.
- Task 7 "Quantitative Screening" and Task 14 "Fire Risk Quantification"
  - Because the Fire PRA Model is a fundamental building block that drives the calculation of risk, changes in the Fire PRA model can result in significant rework to Task 7 "Quantitative Screening" and Task 14 "Fire Risk Quantification", including any reviews of risk significant sequences (such as cutset reviews)

Task 6 " Fire Ignition Frequencies"

"This section describes the procedure for estimating the fire-ignition frequencies associated with fire ignition sources. Generic ignition frequencies that can be specialized to plant conditions in terms of plant characteristics and plant fire event experience are provided. Uncertainties in the generic frequencies are also provided in terms of 5th, 50th, and 95th percentiles." [1]

"Although Task 6, Fire Ignition Frequencies, is not legacy information, it is a key building block of the Fire PRA and affects the risk results for each fire compartment. As such, it is very important to have an experienced Fire PRA analyst review the Task 6, Fire Ignition Frequencies early in the project to make sure the ignition frequency calculation and documentation meets ASME/ANS standards for Fire PRA. It is particularly important that Task 6 is supported by walkdowns and that the ignition sources are correctly binned to support the calculation of ignition frequencies." [6]

Potential Project Impact – Due to Deficiencies in Task 6 "Fire Ignition Frequencies"

If significant changes in the model are necessary later (i.e., after Task 7) there is a significant impact on other tasks.

- Task 8 and Task 11 "Fire Modeling"
  - These tasks are examples of some of the more detailed (and expensive) work that typically needs to be performed for only a selection of components. Typically, high-risk areas or risk-significant sequences are identified for additional

refinements, including fire modeling. Any significant changes in the Fire PRA model could invalidate the work performed for a given fire model and require a complete rework of associated fire models.

- Task 9 “Detailed Circuit Analysis” and Task 10 “Circuit Failure Mode Likelihood Analysis”
  - These tasks are generally performed to help refine the Fire PRA and to reduce risk for circuits that are not separated by 3-hour fire barriers. Similar to the fire modeling discussion above, changes in the Fire PRA model Task 5 could invalidate the Task 9 and 10 work performed and require significant rework.
- Task 7 “Quantitative Screening” and Task 14 “Fire Risk Quantification”
  - Because the Fire PRA Model is a fundamental building block that drives the calculation of risk, changes in the Fire PRA model can result in significant rework to Task 7 “Quantitative Screening” and Task 14 “Fire Risk Quantification”, including any reviews of risk significant sequences (such as cutset reviews).

BUILDING A SOLID FOUNDATION

As discussed above, use of legacy information can lead to significant project efficiencies and may even be a key to making the business case for performance of the Fire PRA. Similarly, other priorities, such as the need to build a team that can use and maintain the Fire PRA going forward, may necessitate the use of less experienced staff for the project. How then do we balance the benefits of using legacy information with the potential vulnerabilities discussed above?

Project Planning

As part of the project planning process, key assumptions should be identified for later scrutiny. Early identification of key project assumptions, for example an assumption that the Appendix R Safe Shutdown Analysis Cable Selection and Routing can be used for 70 % of the Task 3 “Cable Selection and Routing”, is vital to the project success. By identifying these key assumptions early in the project planning, they can be verified to prevent adverse project impact.

Verification of Key Assumptions

A Fire PRA Readiness Review is one way to verify key project assumptions. Although the Fire PRA Readiness review should be customized based on the project plan, the following is an example for Fire PRA Readiness Review Scope:

NUREG/CR-6850 Task	Readiness Assessment	Desired Effect / End State
1 - Plant Boundary Definition and Partitioning	Review the definition and basis for the safe shutdown fire areas and fire zones. Identify the barrier ratings and whether any issues exist (e.g., generic issues or recent operating experience).	Ensure the plant boundaries have a referenceable basis with no issues. Decide whether walkdowns are needed to support Task 1. Optional – Establish a computer-based approach that allows sorts/queries of the data

NUREG/CR-6850 Task	Readiness Assessment	Desired Effect / End State
		and results. Data in this format facilitates the Fire PRA (FPRA) quantification and reporting.
2 - Component Selection (including MSO Expert Panel)	Review the Safe Shutdown Equipment List. Review the existing spurious failure evaluation and MSO Expert Panel Report (if available).	Ensure the plant component information has a referenceable basis with no issues. Optional – Establish a computer-based approach that allows sorts/queries of the plant systems, structures, and components (SSCs) in the Fire PRA. Data in this format facilitates the Fire PRA (FPRA) quantification and reporting.
3 - Cable Selection and Cable Location	Review cable selection documentation, including documentation of methodology to support ASME/ANS requirements. • Cable type such as Thermoset or Thermoplastic (the amount of each). • Cable raceway and cable routing information. This includes the format of the data (preferably in a database). • Modifications performed to achieve compliance with regulatory requirements (e.g., NRA 2.3.1 or 10 CFR 50, Appendix R). • Dates of the last update and last verification. • Are there any known issues with the cable data? Review the basis for circuit-breaker coordination, to ensure selective tripping.	As described in Tasks 1 and 2, having the cable information in a database is preferred. Systems/components evaluated as sensitivities in Task 2 may be traced initially, or the tracing deferred (e.g., pending initial Fire PRA results to re-check sensitivity results).
4 - Qualitative Screening	Later – as part of Fire PRA	For long-term modeling, this approach eliminates the need to justify screening (initially and then later on as the model is refined).
5 - Fire-Induced Risk Model	Because the Fire PRA model is built on the foundation of the internal events model, it is very important to assess the existing internal events model for readi-	Same as Task 1, this but extends the computer-based approach to include the PRA. Use existing circuit failure analyses to eliminate modeling of

NUREG/CR-6850 Task	Readiness Assessment	Desired Effect / End State
	ness for the Fire PRA. The following are a few of the key factors that should be assessed in the Readiness Assessment: • Capability category of Internal Events PRA • Modeling changes needed to convert Internal Events model to Fire PRA such as: – Differences between EOPs and FEPs – Adverse effects of fire induced spurious operations – Alternate shutdown strategies (control room evacuation) – Level of detail in the Internal Events Model	components that are not susceptible to spurious operation. Similarly, use fire-adjusted human error probabilities based on lessons learned at similar plants. Use more refined failure probabilities and circuit analysis as early in the quantification process as practical in order to provide a better focus on the important areas in the detailed fire analysis of Task 11.
6 - Fire Ignition Frequencies	Assess availability of fire event data. Review if existing plant walkdown data has been used to develop fixed ignition sources. Review plant historical data for fire events. Check if the plant controls hot work and transient combustibles in high risk areas.	Same as Tasks 1 and 5.
7 - First Quantification	Later – as part of Fire PRA	The results of the first quantification are used as foundation for the subsequent quantifications, allowing the focus of resources on the dominant contributors.
8 - Scoping Fire Model	Later – as part of Fire PRA	Using the automated quantification process, make data and/or model changes to successively quantify the Fire PRA and reduce conservatism. Rather than focus on these changes in a serial fashion, make the first detailed quantification with as many updates (or partial updates) as available from fire modeling (scoping or detailed fire modeling), circuit failures,

NUREG/CR-6850 Task	Readiness Assessment	Desired Effect / End State
		operator actions, detection/ suppression, and/or likelihood of leading to reactor trip.
9 - Detailed Circuit Failure Analysis	Review documentation of detailed circuit failure analysis performed for compliance with Safe Shutdown Analysis.	Same as Task 5.
10 - Circuit Failure Likelihood Analysis	Review the ability to adjust spurious basic event probabilities based on known cable types (e.g., reduce unavailabilities for double-break design cables, if applicable, and account for timely conductor grounding or open circuit reversing the spurious actuation). Evaluate existing detailed circuit analysis in order to maximize input into initial quantification phases and minimize need for detailed fire modeling.	Same as Task 5.
11 - Detailed Fire Modeling	Review existing plant walkdown information performed to support Fire Hazards Analysis or Fire Characteristics Table. Review plant information on detection and suppression, including performance data. Review existing fire modeling if available. Determine if certain types of area geometries and certain types of fires (e.g., oil fires) were modeled using 75 % heat release rate (HRR) at the same time as the 98 % HRR. Determine if fire incident data analysis was used to develop severity factors for major oil fire sources (e.g., all pumps and EDGs). Evaluate the potential impact of using the latest HRR information.	Same as Task 8.
11b - Main Control Room Evaluation	Review analysis of MCR for compliance with fire protection rule. Review documentation of actual separation of targets within control boards (i.e., do not just rely on	

NUREG/CR-6850 Task	Readiness Assessment	Desired Effect / End State
	separation of switches and instruments on the board facia). Check for cables in the overhead. Check the cable types, especially for thermoplastic cables and optical cables.	
11c - Multi Compartment Analysis	Evaluate the potential to use plant-specific records to assess barrier failure probabilities as some values are often lower than generic. Evaluate the potential for fire compartments that have not undergone detailed analysis to perform bounding HGL assessments in order to screen scenarios early.	
12 - Post-Fire Human Reliability Analysis	Review Operator Actions and evaluate/credit a minimal set agreed upon with the plant, such as AFW for transient decay heat removal, cool down / injection / recirculation for LOCA response, and support system recoveries or alternates such as DC power or fire water cooling. Check the availability of plant-specific thermal-hydraulic information for potential use during Fire PRA quantification such as to add recovery to operator actions.	Same as Task 5.
15 - Uncertainty Analysis	Later – as part of Fire PRA	Document major assumptions and potential sources of modeling uncertainty in each individual task analyses in preparation for the overall uncertainty analysis in this task.

Use of Peer Reviews

Although many U.S. Fire PRAs are required to undergo a Peer Review prior to being used for risk-informed applications, informed use of focused scope Peer Reviews are a useful tool to verify quality of individual tasks to prevent impact on subsequent tasks as described above. The use of focused scope peer reviews can be a cost-effective tool to ensure quality and to manage project costs. This can be particularly beneficial when tasks are performed with less experienced staff.

CONCLUSIONS

A significant risk to a timely and cost-effective development of a Fire PRA is the quality and technical appropriateness of the supporting foundation for developing it.

Experience in the United States indicates that deficiencies in foundational Tasks (Fire PRA Tasks 1 - 6) have been responsible for significant inefficiencies and cost overruns in U.S. Fire PRAs. While legacy data provides an opportunity to reduce project costs by taking advantage of work that was already performed, the use of legacy data comes with additional project risk if the quality of the work and the technical appropriateness of the information for use in Fire PRA are not efficiently verified before use.

Significant unnecessary work can be avoided by:

- Establishment of a Project Plan
  - Identifying key assumptions in the planning stage of the Fire PRA,
- Verifying key assumptions, and
- Using Peer Reviews or experienced mentors to review key tasks before quantification is performed.

REFERENCES

[1] National Fire Protection Association (NFPA): NFPA-805, Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, Quincy, MA, USA, 2001.

[2] Nuclear Energy Institute (NEI): Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), NEI 04-02, Revision E, Washington, DC, USA, April 2004, https://www.nrc.gov/docs/ML0409/ML040970017.pdf.

[3] United States Nuclear Regulatory Commission (NRC): Alternate Fire Protection Rule [10 CFR 50.48c, NFPA 805], Washington, DC, USA, latest update: May 2017, https://www.nrc.gov/reactors/operating/ops-experience/fire-protection/protection-rule.html.

[4] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, Washington, DC, USA, December 2009, https://www.nrc.gov/docs/ML0927/ML092730314.pdf.

[5] United States Nuclear Regulatory Commission (NRC): 10 CFR 50.69: Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors, Washington, DC, USA, latest update: August 2017, https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0069.html.

[6] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 (EPRI 10191989), Washington, DC, and Palo Alto, CA, USA, 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/.

[7] United States Nuclear Regulatory Commission (NRC) and Electric Power Research Institute(EPRI): Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, NUREG-1855, Vol. 1, Washington, DC and Palo Alto, CA, USA, March 2009, https://www.nrc.gov/docs/ML0909/ML090970525.pdf.

[8] Electric Power Research Institute (EPRI): Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, EPRI-1016737, Palo Alto, CA, USA, December 2008,
https://www.epri.com/#/pages/product/1016737/?lang=en-US.

[9] Nuclear Energy Institute (NEI): NEI FAQs, Frequently Asked Questions in NFPA 805 Transition Process, NEI 805 Task Force, Washington, DC, USA,
https://www.nei.org.

[10] Electric Power Research Institute (EPRI): Fire PRA Methods Enhancements, Additions, Clarifications and Refinements to EPRI 1019189, EPRI Interim Report 1016735, Palo Alto, CA, USA, December 2008,
https://www.nrc.gov/docs/ML0902/ML090290195.pdf.

[11] United States Nuclear Regulatory Commission (NRC): USNRC Fact Sheet on Fire Protection for Nuclear Plants, Washington, DC, USA, 2013,
http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fire-protection-fs.html.

KAERI's Research and Development Activities for the Construction and Quantification of Fire Event PSA Model

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ABSTRACT

This paper introduces the research and development (R&D) activities of KAERI (*K*orea *A*tomic *E*nergy *R*esearch *I*nstitute) on fire Probabilistic Safety Assessment (PSA), particularly for the construction and quantification of the PSA model. Main R&D activities of KAERI covered in this paper are (1) development of modification rules for the construction of a one-top Fire PSA model, (2) development of computerization tools for fire event PSA, and (3) validation study for the construction of fire induced support system initiating event fault trees (SSIE FTs).

Fire PSA (PSA for fire events) models are generally developed by modifying internal events PSA models (PSA for plant internal events). As some fire induced accident impacts may not be adequately covered by the internal events PSA model, additional fire scenario logics are often developed and incorporated into the Fire PSA model. KAERI developed modification rules for the construction of a one-top PSA model for fire events by using a one-top PSA model for internal events. A one-top fault tree is a large single fault tree representing the PSA logics including all the event trees and fault trees for core damage frequency (CDF) or large early release frequency (LERF) quantifications. The modification rules can be applied to the entire fire events, for fire induced equipment failure events as well as for spurious operation events.

KAERI has developed computerization tools Fire PSA to facilitate Fire PSA works for identifying and modeling fire induced component failure modes, and to construct a one-top Fire PSA model based on the modification rules mentioned above. KAERI developed the IPRO-ZONE (*i*nterface *pro*gram for constructing *zone* effect table) for the construction of a one-top Fire PSA model with its output, the AIMS (*a*dvanced *i*nformation *m*anagement *s*ystem for PSA) -PSA and a one-top internal events PSA model. The IPRO-ZONE developed, however, has some limitations in the use of cable data and the determination of a target set damaged by fire. In an effort to overcome these limitations, KAERI is currently developing an improved Fire PSA program named ProFire-PSA (*Pro*gram for *Fire* PSA).

For support systems with redundant trains, SSIE FTs are generally developed using initiators (frequency) and enabling events (probability). In Fire PSA, fire induced SSIE FTs can be appropriately constructed solely with initiators, i.e., with no consideration for enabling events. However, there were no detailed validation processes on that approach. KAERI conducted a validation study for fire induced loss of component cooling water system (LOCCWS) accident sequences by constructing and quantifying two different LOCCWS IE FTs: one with initiators only, and the other with both, initiators and enabling events. Fire induced LOCCWS accident sequences with two LOCCWS IE FT models were quantified to compare their risk results. The results of this study show that the LOCCWS IE FT model with initiators only is similar to that of a model with initiators and enabling events in terms of risk quantification results.

INTRODUCTION

An internal fire event probabilistic safety assessment (PSA) model has been generally quantified by modifications of a pre-developed internal events PSA model. New fire induced accident sequence logics not covered in the internal events PSA model are separately developed to incorporate them into the Fire PSA model. Jung et al. [1] have developed the JSTAR (Jung's Single Top And Run) method to build a one fire event PSA model for a one-shot calculation of all the fire event scenarios. A one-top fault tree is a one fault tree representing the PSA logics including all the event trees and fault trees for the core damage frequency (CDF) and large early release frequency (LERF) quantifications. Previous work [1] has been limited to the equipment failures whose probability is one. However, as discussed in NUREG/CR-6850 [2], for the cases of the spurious operations of the equipment impacted by a fire, their probabilities might not be estimated as one. Therefore, new modification rules [3], [4] were developed to construct the one-top PSA model for fire events by using the pre-developed one-top internal events PSA model.

KAERI has developed computerization tools for Fire PSA to facilitate Fire PSA works for identifying and modeling fire induced component failure modes, and to construct a one-top Fire PSA model based on the modification rules mentioned above. KAERI developed the IPRO-ZONE (*i*nterface *pro*gram for constructing *zone* effect table) [5] for the construction of a one-top Fire PSA model with its output, the AIMS-PSA (*a*dvanced *i*nformation *m*anagement *s*ystem) for PSA) [6] and a one-top internal events PSA model. The IPRO-ZONE developed, however, has some limitations in the use of cable data and the determination of a target set damaged by fire. In an effort to overcome these limitations, KAERI is currently developing an improved Fire PSA program named ProFire-PSA (*Pro*gram for *Fire PSA*) [7].

It has been recognized that the SSIE FT for a fire event PSA could be constructed by considering only initiators [4]. Initiators are events that cause reactor shutdown and pose challenges to safety functions. The modeling approaches for developing SSIE FTs with only initiators were not validated. In this study, we confirmed that the quantification results of the fire induced loss of component cooling water system (LOCCWS) accident sequences with LOCCWS IE FT model considering only initiators are the same as to those with LOCCWS IE FT model considering both initiators and enabling events. Enabling events are events that put the system in a critical state for IE [4].

The remainder of this paper is organized as follows. The second section presents the modification rules to construct a one-top PSA model for fire events by using the pre-developed one-top internal events PSA model. The third section provides the introduction of the ProFire-PSA. The fourth section presents the validation study results on LOCCWS IE FT with initiators. Some conclusions are given.

MODIFICATION RULES

CDF Equation and Modification Rules

The fire induced core damage frequency (CDF) of a nuclear power plant (NPP) can be represented by equation (1) [2], [3]:

$$\mathrm{CDF}=\sum_{k=1}^{m} \mathrm{CDF}_k \tag{1}$$

In equation (1), CDF_k represents the CDF of each fire compartment or fire scenario *k*. The CDF_k can be further represented as

$$CDF_k = \%R_k \times S\%R_k \times CCDP_k \tag{2),}$$

where:

– $\%R_k$ = fire frequency of compartment or scenario *k*,
– $S\%R_k$ = severity factor of the compartment or the scenario *k* including non-suppression probability,
– $CCDP_k$ = conditional core damage probability (CCDP) of compartment or scenario *k.*

The modification algorithm of an internal events PSA model into a fire event PSA model is as follows [3], [4]:

- Initiating event:

$$\%I \Rightarrow \%I + \Sigma\, \%R_k \times S\%R_k \tag{3}$$

- Basic event for the component failure:

$$a \Rightarrow a + \sum \%R_k \times S\%R_k \times P(a)\%R_k \tag{4}$$

where:

– $\Rightarrow$: change from the left event to the right event,
– $\%I$: initiating event for an internal events PSA model,
– *a*: basic event for the component failure,
– $P(a)\%R_k$: fire induced component failure event related to the basic event "*a*".

Equation (3) represents that an internal IE resulting from a fire is changed by an "OR" logic combination of the internal IE itself and the fire events including the severity factor. In the real applications of equation (3) to Fire PSA quantifications, the internal IEs (%I) are set to false. equation (4) represents that an internal basic event for a component failure is changed by an "OR" logic combination of the internal basic event itself and the fire events including the severity factor, and fire induced failure events for the component.

Demonstration of the Appropriateness of Modification Rules

The hypothetical plant, as shown in Figure 1, has two fire compartments R_1 and R_2. It is assumed that if the fire event $\%R_1$ occurs, then an internal initiating event IE_1 occurs, and that if the fire event $\%R_2$ occurs, then an internal initiating event IE_1 and IE_2 occur. The fire damage events for the basic events relating to the equipment A, B, C, and D are assumed to be 'a_f', ' b_f', 'c_f', and 'd_f', respectively. The definitions of the other events for the application of the modification rules are presented in Table1. Let us assume that the CDF for an internal events PSA is expressed as equation (5):

$$CDF = IE_1abce + IE_2acdg \tag{5}$$

In equation (5), 'IE_1' and 'IE_2' represent internal initiating events, 'a', 'b', 'c', and 'd' represent the basic events for the equipment random failures. 'e' and 'g' represent the basic events for the equipment random failures not shown in Figure 1.

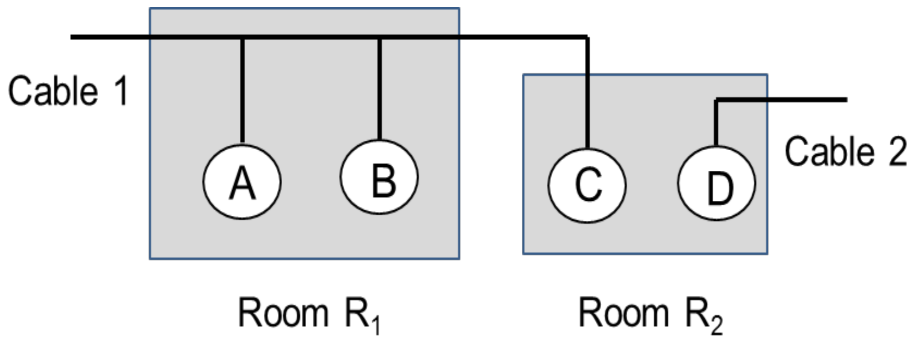

Figure 1 Components located at fire rooms of hypothetical plant

Table 1 Event descriptions

Event name	Event descriptions
$S\%R_1$	Severity factor for $\%R_1$
$S\%R_2$	Severity factor for $\%R_2$
a_f	Component A failure event or probability due to a fire in room 1
b_f	Component B failure event or probability due to a fire in room 1
$c_{f\text{-}1}$	Component C failure event or probability due to a fire in room 1
$c_{f\text{-}2}$	Component C failure event or probability due to a fire in room 2
d_f	Component D failure event or probability due to a fire in room 2

With the information in Table 1, the proposed modification rules were applied to equation (5). The events 'IE_1', 'IE_2', 'a', 'b', 'c', and 'd' in equation (5) were replaced by the right-hand Boolean formulas as follows:

$$\begin{aligned} &IE_1 \Rightarrow \%R_1S\%R_1 + \%R_2S\%R_2 \\ &IE_2 \Rightarrow \%R_2S\%R_2 \\ &a \Rightarrow a + \%R_1S\%R_1a_f \\ &b \Rightarrow b + \%R_1S\%R_1b_f \\ &c \Rightarrow c + \%R_1S\%R_1c_{f-1} + \%R_2S\%R_2c_{f-2} \\ &d \Rightarrow d + \%R_2S\%R_2d_f \end{aligned} \tag{6}$$

For the simplification of the changing process, the probabilities of all events in Table 1 are assumed to be one except '$\%R_1$', '$\%R_2$', and 'c_{f-1}'. Substituting equation (6) into equation (5), CDF equation of hypothetical plant X is represented as:

$$\begin{aligned} CDF &= IE_1abce + IE_2acdg = (\%R_1 + \%R_2)(a + \%R_1)(b + \%R_1)(c + \%R_1c_{f-1} \\ &+ \%R_2)e + (\%R_2)(a + \%R_1)(c + \%R_1c_{f-1} + \%R_2)(d + \%R_2)g = \\ &\%R_1ce + \%R_1c_{f-1}e + \%R_2abe + \%R_2ag \end{aligned} \tag{7}$$

The appropriateness of the proposed modification rules can be demonstrated easily by using the previous Fire PSA method, so-called damage term insertion method [1]. According to [1], if the fire induced failure events associated with equipment failures are added to the corresponding basic events, equation (5) can be represented as:

$$IE_1(a + a_f)(b + b_f)(c + c_{f\text{-}1} + c_{f\text{-}2})e + IE_2(a + a_f)(c + c_{f\text{-}1} + c_{f\text{-}2})(d + d_f)g \qquad (8)$$

In equation (8), if the fire event $\%R_1$ occurs, internal initiating event IE_1 occurs and internal initiating event IE_2 does not occur. Thus, IE_1 is changed to 'TRUE (Ω)', and IE_2, $c_{f\text{-}2}$, and d_f are changed to 'FALSE ($\varnothing$)'. The CCDP of the fire event $\%R_1$ can be represented as "$(a + \Omega)(b + \Omega)(c + c_{f\text{-}1} + \text{FALSE})e$". The same approach can be applied to the other fire events as follows:

$$\%R_1 \Rightarrow CCDP(\%R_1): \quad IE_1 = a_f = b_f = \Omega,\ IE_2 = c_{f\text{-}2} = d_f = \varnothing \quad \text{--->}\ ce + c_{f\text{-}1}e \qquad (9)$$

$$\%R_2 \Rightarrow CCDP(\%R_2): \quad IE_1 = IE_2 = c_{f\text{-}2} = d_f = \Omega,\ a_f = b_f = c_{f\text{-}1} = \varnothing \ \text{--->}\ abe + ag$$

In equation (9), for the case where fire damage events were estimated one, they are set to 'TRUE'. From equation (2) and equation (9), equation (5) can be represented as follows:

$$\%R_1CCDP(\%R_1) + \%R_2CCDP(\%R_2) = \%R_1ce + \%R_1c_{f\text{-}1}e + \%R_2abe + \%R_2ag \qquad (10)$$

Consequently, we can see that equation (7) is the same as the equation (10). Thus, it can be concluded that the developed modification rules can be used for the construction of the Fire PSA model.

FIRE PSA TOOL [7]

Main Function and Modules of the ProFire-PSA

The main function of the ProFire-PSA is to produce the SIMA (*S*cript *I*nterpreter for *M*apping *A*lgorithms) [6] or the RID [8] file to be read in the domestic PSA programs. Figure 2 below shows the relationship between the ProFire-PSA and AIMS-PSA [6] / SAREX [8]. The ProFire-PSA produces the SIMA (AIMS: KAERI) and the RID (SAREX: Industry) files to insert fire scenario-related input data into internal events PSA models built with the AIMS-PSA and the SAREX, which are PSA tools for building and quantifying internal events PSA models. Using the SIMA or the RID file, the pre-built internal events PSA model is changed into a Fire PSA model.

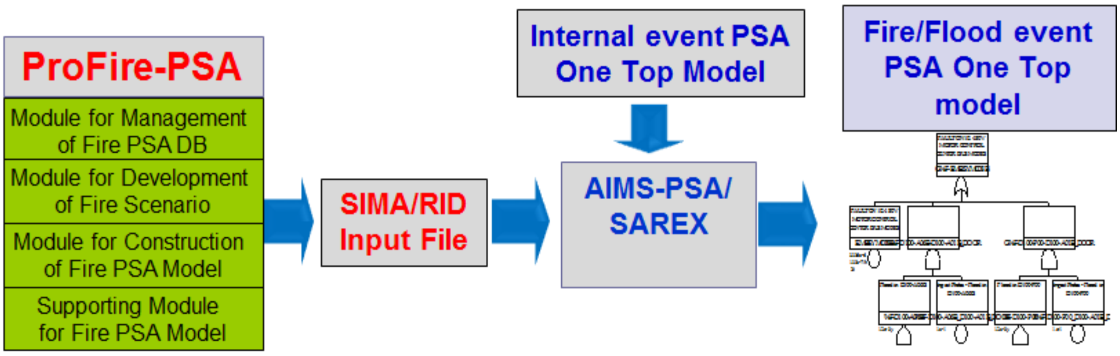

Figure 2 Relation between the ProFire-PSA and AIMS-PSA/SAREX

The other functions of the ProFire-PSA are as follows:

- Determination of failure modes for equipment affected by a fire;
- Direct use of cable data for fire scenarios;
- Easy creation of fire scenarios.

The ProFire-PSA consist of the following four modules:

- Module for Management of Fire PSA DB: DB module;
- Module for Development of Fire Scenario: Scenario module;
- Module for Construction of Fire PSA Model: PSA module;
- Supporting Module for Fire PSA Model: Supporting module.

Each module relationship of the ProFire-PSA is shown in Figure 3 below. In the DB module, Access data such as zones and raceways are read and structured so that these data are available in Scenario and PSA modules. The Scenario module identifies the equipment and cables to be included in the fire scenario. The PSA module generates the SIMA or the RID file to be used as input to the AIMS-PSA or the SAREX. The Supporting module creates fire scenarios with room information and fire ignition analysis results.

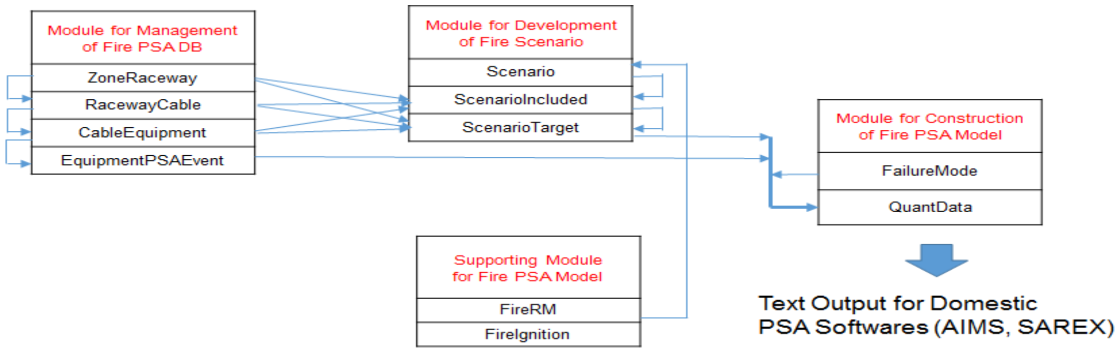

Figure 3 Relation between the modules for the ProFire-PSA

Performing ProFire-PSA

The program module above is implemented as shown in Figure 4 below. As demonstrated in that figure, the ProFire-PSA is performed in three steps. When determining the failure modes of equipment affected by a fire, there are two options, default and realistic. If the default option is selected, the fire induced equipment failure probability is one. If the realistic option is selected, the fire induced equipment failure probability is estimated differently depending on the cable type, equipment type, desired and failed states, etc.

When creating the SIMA or the RID file, the analyst can determine the fire event types (three events (ignition, severity, and non-suppression) or one event including three events) and modeling types (addition or replacement of fire induced failure events to the pre-existing internal events). An example for performing each step is presented in Figure 5. The SIMA file generated from the ProFire-PSA will be applied to the construction of Fire PSA model for the reference NPP.

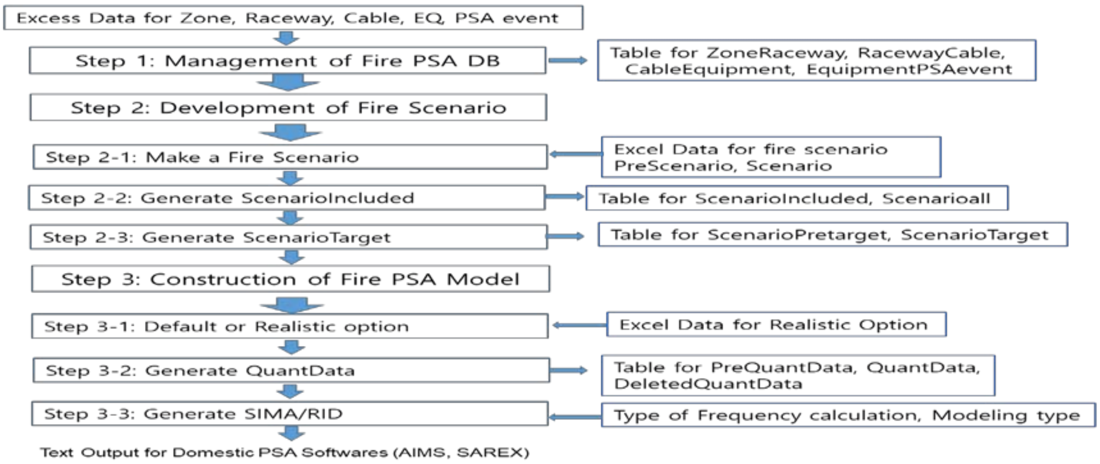

Figure 4 Process for performing ProFire-PSA

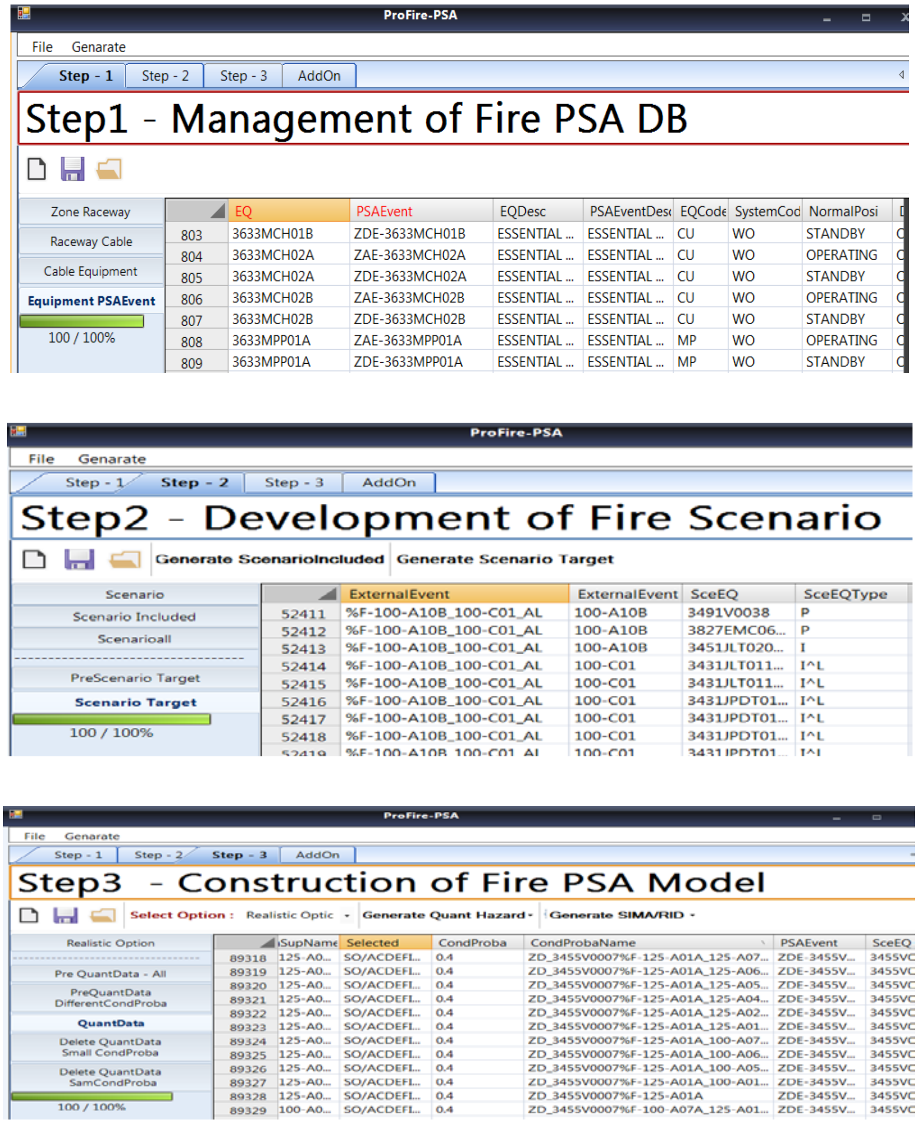

Figure 5 Example for performing each step of the ProFire-PSA

VALIDATION STUDY ON THE CONSTRUCTION OF FIRE INDUCED LOCCWS INITIATING EVENTS FAULT TREES [4]

Construction of LOCCWS Initiating Events Fault Trees

A LOCCWS IE is defined as the loss of component cooling water system (CCWS) train A. The CCWS consists of two independent closed-loop trains. Each train consists of two pumps, two heat exchangers, and other components. During normal power operation, one pump and one heat exchanger in each train are in service to supply cooling water to safety-related and non-safety-related components in the train required for normal power operation. In addition to the electrical system, the essential service water system (ESWS) and the essential chilled water system (ECWS) are required to support the heat exchangers and pumps of the CCWS.

Because the LOCCWS IE for the internal events PSA of the reference NPP was modeled as a single event, the fire induced LOCCWS IE FT with the initiators was developed by referring to the mitigating system FT of the CCWS for the internal events PSA. The IEFT has composed of AND/OR logic gates and zero fire damage events (dummy events), the probabilities of which are zero and remain unchanged. The previous basic events for the equipment affected by a fire were replaced with the zero fire damage events. The other basic events for the equipment not affected by a fire were deleted. The multiple fire scenario AND gates composed of fire ignition frequency (IF), severity factor (SF) including non-suppression probability, and fire induced conditional component failure probability are connected to these zero fire damage events using a computerization tool to assist Fire PSA such as the ProFire-PSA. According to the Fire PSA practices, any fire included in the fire event PSA is assumed to lead to reactor shutdown. Thus, it was further assumed that any fire results in general transient (GT) IE.

A similar approach mentioned above for the fire induced LOCCWS IEFT with the initiators was used for the construction of the fire induced LOCCWS IE FT with initiators and enabling events. The zero fire damage events were added to the equipment affected by the fire. All previous basic events of the CCWS modeled for the mitigating system FT for internal events PSA remained. However, the mission time for the running failure events was changed from 24 hours to 72 hours and the events related to other IEs were eliminated. In the calculation of failure probability for the standby component, enabling event, 72 hour limiting condition for operation for the CCWS was adopted as mission time for the estimation of failure probability.

Quantification of LOCCWS Accident Sequences

Fire induced LOCCWS accident sequences with different LOCCWS IE FT models were quantified to compare their quantification results. Small event tree and large FT approach were used for the construction of an internal events PSA model of the reference NPP. The SSIEs considered in the internal events PSA are a loss of off-site power (LOOP) including station blackout (SBO), a loss of a 4.16 kV A train, a loss of a 125 V DC A, a loss of CCWS A, and a total loss of CCWS. The quantification result of LOCCWS IE FT with only initiators is almost the same as those with initiators and enabling events. The minor quantification difference may result from the inherent PSA quantification approach such as rare event approximation. Based on the quantification results, it could be confirmed that the LOCCWS IE FT for an actual fire event PSA model could be constructed by considering only IE initiators.

CONCLUSIONS

KAERI has developed modification rules for the construction of a one-top PSA model for fire events by using a one-top PSA model for internal events. A one-top fault tree is a large single fault tree representing the PSA logics including all the event trees and fault trees for core damage frequency (CDF) or large early release frequency (LERF) quantifications. The modification rules can be applied to the entire fire events, for fire induced equipment failure events as well as for spurious operation events.

KAERI is currently developing an improved Fire PSA program named ProFire-PSA to save working hours for a Fire PSA in identifying fire induced component failures and modeling them and to construct Fire PSA model. In this paper, the ProFire-PSA is introduced, and its application result is presented. In the near future, full applications of the ProFire-PSA to reference NPP will be performed for finding the items to be improved.

In Fire PSA, fire induced SSIE FTs can be appropriately constructed solely with initiators, i.e., with no consideration for enabling events. However, there were no detailed validation processes on that approach. KAERI conducted a validation study for fire induced loss of component cooling water system (LOCCWS) accident sequences by constructing and quantifying two different LOCCWS IE FTs: one with initiators only, and the other with both, initiators and enabling events. The results of this study show that the LOCCWS IE FT model with initiators only is similar to that of a model with initiators and enabling events in terms of risk quantification results.

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REFERENCES

[1] Jung, W. S., Y. H. Lee, J. Ë. Yang: Development of an automated fast fire PSA method, in: Proceedings (CD) of Topical Meeting PSA'08, American Nuclear Society (ANS), Knoxville, TN, USA, 2008.

[2] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 (EPRI 10191989), Washington, DC, and Palo Alto, CA, USA, 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/.

[3] Kang, D. I., S. H. Han, and K. Y. Kim: An approach to the construction of a one-top fire event PSA model, Nuclear Engineering and Design, Vol.2 39, pp. 2514–2520, 2009.

[4] Kang, D. I: Study on Construction of Loss of Component Cooling Water System Initiating Event Fault Tree for Fire Event PSA, Journal of Nuclear Science and Technology, in preparation, 2019.

[5] Kang, D. I., S. H. Han, and S. Y. Yoo: Development of the IPRO-ZONE for internal fire probabilistic safety assessment, Nuclear Engineering and Design, Vol. 257, pp. 72–78, 2013.

[6] Han, S. H., et al.: AIMS-PSA: A Software for Integrated PSA, in: Proceedings of 13th International Probabilistic Safety Assessment and Management Conference (PSAM13), Seoul, Republic of Korea, October 2016.

[7] Kang, D. I, and Y. H. Yung: Introduction of Fire PSA tool: ProFire-PSA, Transactions of the Korean Nuclear Society Autumn Meeting, Goyang, Republic of Korea, October 24-25, 2019.

[8] KEPCO-E&C: SAREX User's Manual, Version 1.2, Gimcheon-si, Republic of Korea, 2011.

Improving Realism in Fire PRAs for Nuclear Power Plant Applications

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and the Electric Power Research Institute (EPRI) are working jointly under a memorandum of understanding (MOU) on several projects to address improving realism in fire probabilistic risk assessment (PRA). Specifically, these projects include improving methods for modeling electrical enclosure fires; revising heat release rate (HRR) distributions for pumps, motors, and dry transformers; modeling fires in the main control board; analyzing detection and suppression; and revising HRR distributions for transient fuel packages. This paper highlights some of the ongoing activities.

INTRODUCTION

In 2005, EPRI and NRC RES issued a joint technical report titled, "*EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*" NUREG/CR-6850 (EPRI 1011989) [1]. This publication documented state-of-the-art methods, tools, and data for conducting a PRA for a commercial nuclear power plant (NPP) application. Following this publication, many utilities developed Fire PRAs using, among others, the guidance in NUREG/CR-6850 to support risk-informed applications including the transition to National Fire Protection Association (NFPA) Standard 805 [2]. The results obtained from the Fire PRA models have suggested specific elements in the fire scenario analysis where improved methods and/or guidance could reduce conservatism and increase realism in the risk estimates. Consequently, over the past 15 years, Fire PRA research covering the areas of fire ignition frequencies (e.g., NUREG-2169 (EPRI 3002002936) [3]), fire modeling (e.g., NUREG-2178 (EPRI 3002005578) [4], and NUREG/CR-7197 [5]), human reliability analysis (NUREG-1921 (EPRI 1023001) [6]), and electrical cable spurious operations (e.g., NUREG/CR-7150 [7]) have been published and made available to the industry.

Most recently, a working group composed of experienced fire protection and Fire PRA researchers and practitioners from NRC and industry supported by other technical experts completed three reports to support improvements in Fire PRA realism. A fourth report is currently being finalized as a draft for public comment. The first report is the second volume of NUREG-2178 (EPRI 3002016052), "*Refining and Characterizing Heat Release Rates from Electrical Enclosures during Fire (RACHELLE-FIRE), Volume 2: Fire Modeling Guidance for Electrical Cabinets, Electric Motors, Indoor Dry Transformers, and the Main Control Board*" [8] that includes: (1) a modified approach for computing flame radiation and a detailed method for determining the thermal radiation impact from fires inside electrical cabinets, (2) a detailed approach for modeling fire propagation between vertical sections in a bank of electrical cabinets, (3) revised HRR probability distributions for electric motors and dry transformers, (4) revised guidance for modeling fires located adjacent to walls or in corners, (5) a lower non-suppression probability floor value for use in the main control room, and (6) a comprehensive event tree based approach for characterizing the fire scenario progression in the main control board.

The second report, NUREG-2230 (EPRI 3002016051), "*Methodology for Modeling Fire Growth and Suppression for Electrical Cabinet Fires in Nuclear Power Plants*" [9] documents efforts to develop an approach that more closely models the types of fire progressions and response activities observed in operating experience. The third report, NUREG-2232 (EPRI 3002015997), "*Heat Release Rate and Fire Characteristics of Fuels Representative of Typical Transient Fire Events in Nuclear Power Plants*" [10] documents the results from a series of fire tests of transient fuel packages found in NPPs. The final report, NUREG-2233 (EPRI 3002016054), "*Methodology for Modeling Transient Fires in Nuclear Power Plant Fire Probabilistic Risk Assessments*" [11] provides methods for using the new transient fuel package test data in Fire PRAs and is being released for public comment in October 2019. These reports will be discussed in more detail in the following sections.

MODELING OF FIRES IN ELECTRICAL CABINETS AND OTHER EQUIPMENT

The first volume of NUREG-2178 [4] was published in April 2016. This document included methods focused on refining the modeling of fires in electrical cabinets including new HRR probability distributions and an obstructed fire plume model. While preparing Volume 1, the working group identified additional methods to further refine the modeling of selected ignition sources within the Fire PRA that could be developed and published in a second volume.

As in the case of Volume 1 of NUREG-2178, this second volume would describe improved methods for achieving realism by reducing some of the conservatisms present in current methods. As such, the guidance and methods would not replace or invalidate existing methods or guidance but rather would provide more realistic methods and data than previously published. This second volume of NUREG-2178 [8] includes the following methods that can be used for refining the modeling of selected ignition sources including:

- Flame radiation and obstructed radiation. The report describes and reviews existing methods for calculating flame radiation. From that discussion, the document offers a modified approach for computing flame radiation. The document then develops a method for determining the thermal radiation impact from fires inside electrical cabinets. To develop the detailed guidance, modeling of electrical cabinet fires was performed using the Fire Dynamics Simulator (FDS) [12]. As such, this approach extends the research documented in NUREG-2178, Volume 1 [4] associated with modeling plume temperatures generated by fires inside electrical cabinets (i.e., the obstructed plume temperature model) by developing guidance on predicting thermal radiation that is obstructed by vented or unvented cabinet walls (cf. Figure 1).

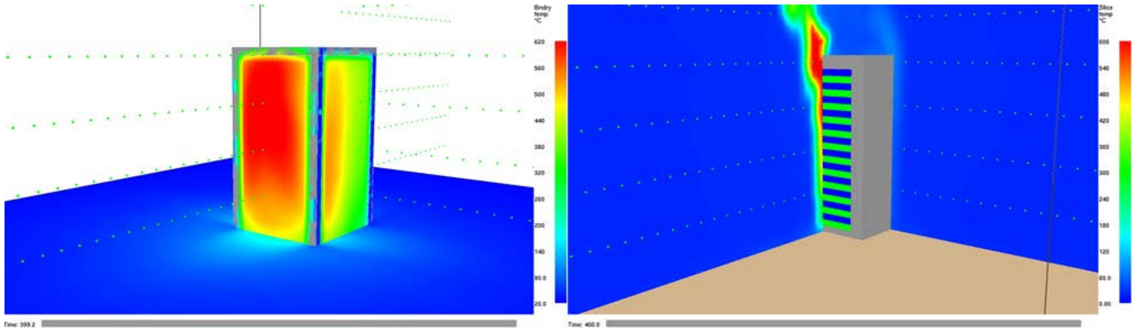

Figure 1 FDS [12] - Predicted wall temperature (left) and gas temperature (right) for a large cabinet with a full-face vent and a 98th percentile fire.

- Fire propagation between adjacent electrical cabinets. The report describes a detailed approach for modeling fire propagation between vertical sections in a bank of electrical cabinets. This method expands upon the guidance provided in Appendix S of NUREG/CR-6850 [1] that referred to this scenario as “enclosure-to-enclosure fire spread”.
- HRRs for electric motors and dry transformers. Appendix G of NUREG/CR-6850 [1] recommended bounding/conservative values for HRRs associated with electric motors and dry transformers based on the values assessed for electrical cabinet fires. However, electric motors and dry transformers are different in terms of ignition sources, modes of ignition, and combustible configuration in comparison to electrical cabinets. Consequently, this document recommends revised HRRs for electric motors (including those motors associated with pumps) and dry transformers based on the size (horsepower or kVA, respectively) of the equipment.
- Fire location factor. Existing guidance [1], [13] suggests that fires adjacent to walls or in corners of a room may generate elevated plume temperatures when compared to fires away from these surfaces (sometimes referred to as the wall/corner plume correction factors). Using data obtained from tests conducted by the National Institute of Standards and Technology (NIST) [14], this report provides new guidance for estimating plume temperatures from fires located along walls or in corners (cf. Figure 2). The guidance is applicable to both fixed and transient ignition sources.

Figure 2 Photographs of burners used for wall and corner tests showing burners located against wall (left), in corner (middle), and moved 0.3 m (1 ft) away from corner (right)

- Non-suppression floor value. Appendix P of NUREG/CR-6850 [1] recommends that the non-suppression probability versus time curves be used subject to a floor (minimum) value of 0.001 for all cases. This assumption means that, in effect, 1 fire in 1,000 is never suppressed, which is not entirely supported by the available data. This document discusses the basis and development of a lower non-suppression probability floor value specifically for use in the main control room (MCR).
- Main control board fire model. Appendix L of NUREG/CR-6850 [1] described a simplified model for determining the severity factor and non-suppression probability for fire scenarios associated with the main control board based on a predefined zone of influence (i.e., a defined set of damage target components). Although easy to apply, this model limits the ability to integrate the main control board scenarios with other elements associated with the risk quantification of fire scenarios inside the MCR. This document describes a comprehensive event-tree-based approach for characterizing the fire scenario progression following ignition of a component in the main control board.

A draft of NUREG-2178, Volume 2 was issued for public comment [15] on June 28, 2019, with a closing date of August 27, 2019. Four sets of comments were received during that period. All of the comments have been resolved, and the final report is being published by EPRI in October 2019. Due to additional publishing requirements, the NRC version will be published at a later date.

MODELING OF DETECTION AND SUPPRESSION BY PLANT PERSONNEL

Over the last decade, there has been significant experience applying the Fire PRA methodology published in NUREG/CR-6850 [1]. Through this experience, certain aspects of the methodology were identified as candidates for additional research and development. One aspect of the Fire PRA methodology that had not undergone revision is the fire scenario progression and interaction between the fire ignition, growth, and suppression models.

NUREG/CR-6850 [1] provided a simplified framework for calculating fire ignition frequency, the fire hazard, and the suppression effectiveness. This model captured actual U.S. NPP experience to develop the fire ignition frequencies and manual non-suppression rates. The fire hazard, on the other hand, was derived from experimental fire tests to predict a distribution of HRRs. In addition to the HRR, fire testing informed the timing of the fire, specifically the rate at which the fire grows to its peak HRR, steady-state burning, and decay phases. When applied, the combination of operating experience and experimental testing has resulted in a high percentage of electrical cabinet fire scenarios damaging external targets. This does not align with the insights in the EPRI Fire Events Database (FEDB), which suggests that most fires are contained and limited to the ignition source [16].

Around 2010, the EPRI FEDB underwent an extensive upgrade to improve the data quality (including timing, event descriptions, and so on) and source document traceability and to add more recent U.S. NPP operating experience. This update marked a significant improvement over previous versions that provided minimal details, and it allowed for the revision of ignition frequencies and non-suppression probabilities through 2009 [3]. Although NUREG-2169 (EPRI 3002002936) "*Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009*" updated the data [3], it was recognized that further research would be needed to more realistically model the fire progressions observed in actual experience.

NUREG-2230 [9] provides an approach that more closely models the types of fire progressions and response activities observed in operating experience. Specifically, the report provides an updated framework for treatment of the fire scenario progression starting from ignition through suppression. The detection-suppression event tree described in Appendix P of NUREG/CR-6850 [1] is revised to include additional fire sequences commonly observed in NPP fire events (see Figures 3 and 4).

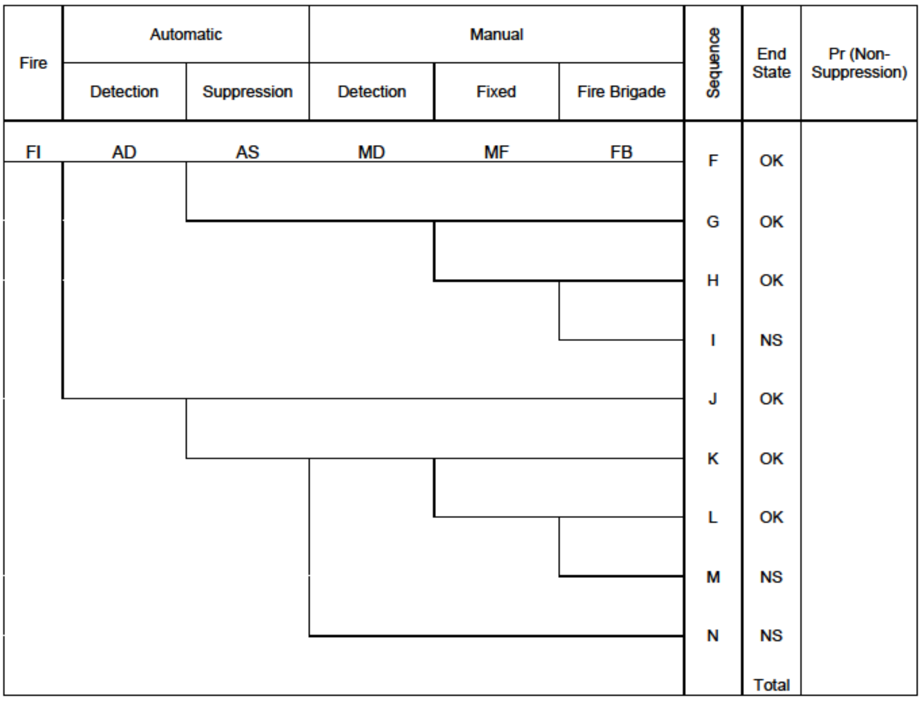

Figure 3 Original NUREG/CR-6850 [1] event tree

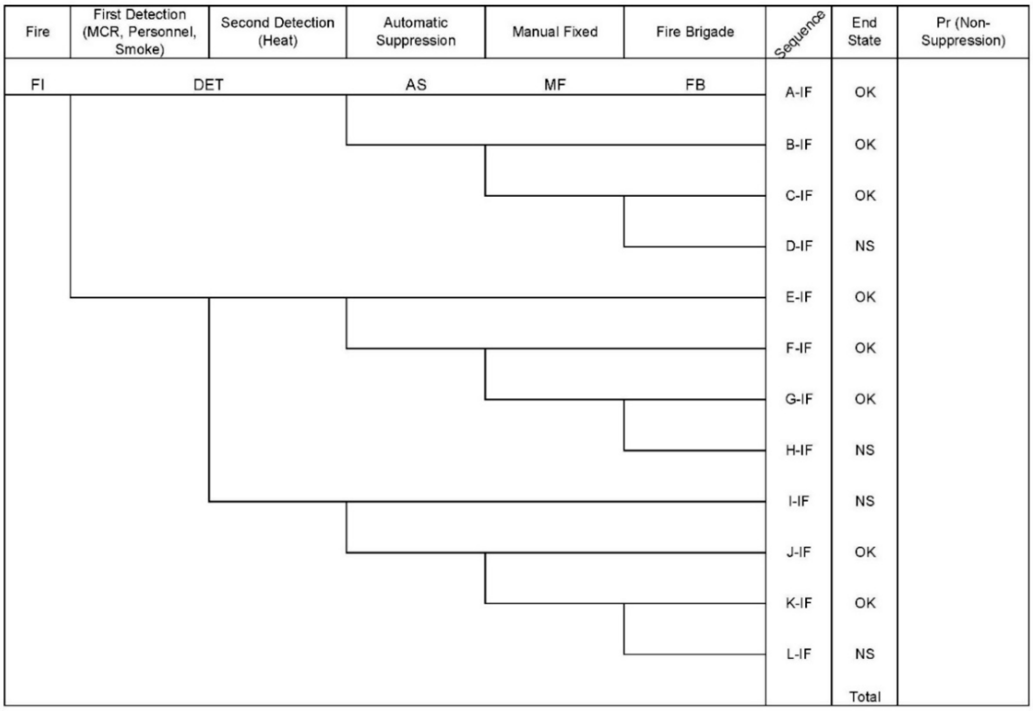

Figure 4 Improved event tree developed in NUREG-2230 [9]

The methodology described in NUREG-2230 [9] provides the following:

- An updated fire frequency for electrical cabinets (NUREG/CR-6850, Bin 15) that makes use of the most recent fire event data classified in the study "*Fire Events Database Update for the Period 2010–2014: Revision 1*" (EPRI 3002005302) [17].
- Split fractions for interruptible and growing fires were developed for use in the revised detection-suppression event tree. *Interruptible Fires* are characterized as events in which plant personnel could detect and perform early suppression activities. These are fires that progress at a rate such that plant personnel may discover and suppress them prior to experiencing external target damage. *Growing Fires* are characterized as events where the fire grows in a manner that may not allow for plant personnel suppression in the early stages of the fire development.
- A conceptual fire event tree progression model developed through an event review of insights from the FEDB. A procedure and rule set have been developed to allow for consistent classification of fire events into the two different growth profiles: interruptible and growing.
- A revised electrical cabinet HRR profile developed for use in the detailed fire modeling of interruptible fires. This revised profile includes a pre-growth period of up to 8 min of negligible HRR. The treatment for the HRR profile for growing fires was not updated in this research.
- Revisions to the detection-suppression event tree to include paths for crediting early intervention by plant personnel as well as developing new parameters to facilitate these revisions. These new parameters include an opportunity to credit detection by general plant personnel.
- An opportunity for MCR indications as a means for fire detection when applicable in the detection-suppression event tree.
- The fires observed in electrical cabinets do not always produce conditions significant enough to actuate fixed detection systems, therefore introducing the probability of automatic smoke detection effectiveness parameter to characterize the ability of spot- type smoke detection devices to operate in a range of geometric conditions and HRRs.
- The scope of the methodology described in NUREG-2230 [9] was limited to electrical cabinet sources (NUREG/CR-6850, Bin 15: electrical cabinets). However, due to the legacy treatment of manual suppression curves, the research also produced new manual suppression curves for the main control room and electrical equipment other than electrical cabinets (i.e., motors and dry transformers).

A draft of NUREG-2230 was issued for public comment [18] on July 1, 2019, with a closing date of July 31, 2019. Four sets of comments were received during that period. All of the comments have been resolved, and the final report is being published by EPRI in October 2019. Due to additional publishing requirements, the NRC version will be published later.

MODELING OF TRANSIENT FUEL PACKAGES

The current recommended characterization of transient fires, which are fires that result from maintenance, hot work, or storage of combustible materials, for a Fire PRA consists of a specified fire growth rate and a specified distribution of HRRs. No current recommendation exists on duration or HRR decay. The HRR distribution is specified in NUREG/CR-6850 [1] as a gamma distribution with a 75th percentile HRR of 142 kW and a 98th percentile HRR of 317 kW. Growth rates are specified in NUREG/CR-6850 Supplement 1 [19] as 2 min for unconfined transient fuel packages, 8 min for confined fuel packages (i.e., in a container like a trash can), and instantaneous for spills of flammable liquids.

The HRR recommendation in NUREG/CR-6850 [1] is based on 27 experiments. HRRs in the selected tests range from 12 kW to 351 kW. One third of the experiments included a significant amount of flammable liquid as part of the fuel package. One third of the tests were large trash bags or plastic trash containers, with most of those involving multiple co-located bags or containers. Three of the test experiments were wood cribs, a fuel package specifically designed as a repeatable fire source with rapid growth. The frequency of flammable liquid presence, the fact that most trash fires involved multiple containers, and the inclusion of wood cribs suggests that these fuel packages are not representative of actual transient fire events occurring in NPPs.

Actual industry experience with transient fires [16] is not adequately reflected in the experiments used to develop the guidance in NUREG/CR-6850. A much lower incidence of trash containers is seen in the EPRI FEDB than in the tests selected for NUREG/CR-6850 [1]. Flammable liquids are rarely seen in the FEDB, and fires with multiple fuel containers (trash bags, trash cans) are extremely rare events. One significant consequence of the HRR distribution provided in NUREG/CR-6850 is that an expansive zone of influence exists around a fire where damage to cables or equipment might occur.

A two-phase approach was used to address improved realism in the modeling of transient fires in a Fire PRA. The first phase was a program developed to test a wide range of fuel packages (cf. Table 1) selected to reflect industry operating experience. The experimental program involved 99 transient fuel packages with repeat tests for a total of 290 tests. Table 1 lists the general categories of fuel packages. In addition to selecting representative fuel packages, the ignition methods in Table 2 used during testing were selected to represent relatively low-energy ignition sources seen in industry experience. In addition to HRR, the test program collected other data (effective heat of combustion, soot yield, and carbon monoxide (CO) yield) to support improved realism in fire modeling of transient fires. NUREG-2232 [10] documents the selection of test items, the experimental setup, and a summary of results for the tests. The test matrix of items was developed from a review of operating experience, inspection findings, and professional judgement.

Table 1 Typical transient fuel packages found in NPPs

Tape	Trash
Rags	Tool bag
Clothing / Personal Protective Equipment (PPE)	Welding machine
Fire/Welding blanket	Cardboard
Plastic	Foam
Roper	Insulation

Debris	Mop
Cloth	Oily Rag
Oxy-acetylene hose	Duct
Power cord	PPE bag
Paper	Absorbent pad
Tarp	Acetylene
Wood	Paint
Filter	Cable
Flammable liquid	Chair
Plastic bag	Hose

Table 2 List of ignition methods

Method	Description
Lighter	Butane fireplace lighter. Used for easily ignited items where the lighter can be safely held to the side of the test item.
Wick	A 7.6 cm (3 in.) piece of 1.2 cm (0.5 in.) diameter nylon rope. The ends of the rope have a 6 mm (0.25 in.) wide double wrap of duct tape to prevent unravelling of the rope. Immediately before placement on the test item, 5 mL of heptane was applied to the wick. The wick was ignited with the lighter immediately after placement.
Flame	A small continuous propane flame about 7 to 10 cm (3 to 4 in.) in size from a piece of 6 mm (0.25 in.) metal tubing.
Panel	A flat radiant panel with dimensions approximating a typical halogen work lamp. The panel to test item distance was established to provide a 30 kW/m^2 incident heat flux at the fuel surface.

The data obtained from a “Medium Empty Box” test are shown in Figure 5.

The second phase was the development of recommended approaches for incorporating transient fires in a Fire PRA. This Fire PRA guidance report, NUREG-2233 (EPRI 3002016054) [11] consolidates existing methods and recommendations on the modeling of transient fires; provides new probabilistic distributions for peak HRR, total energy release (TER), and zones of influence (ZOIs) for various types of targets; and provides a method and recommendations for the detailed modeling of transient fires including fire growth and decay, yields of minor combustion products, and the physical size and location of the fire.

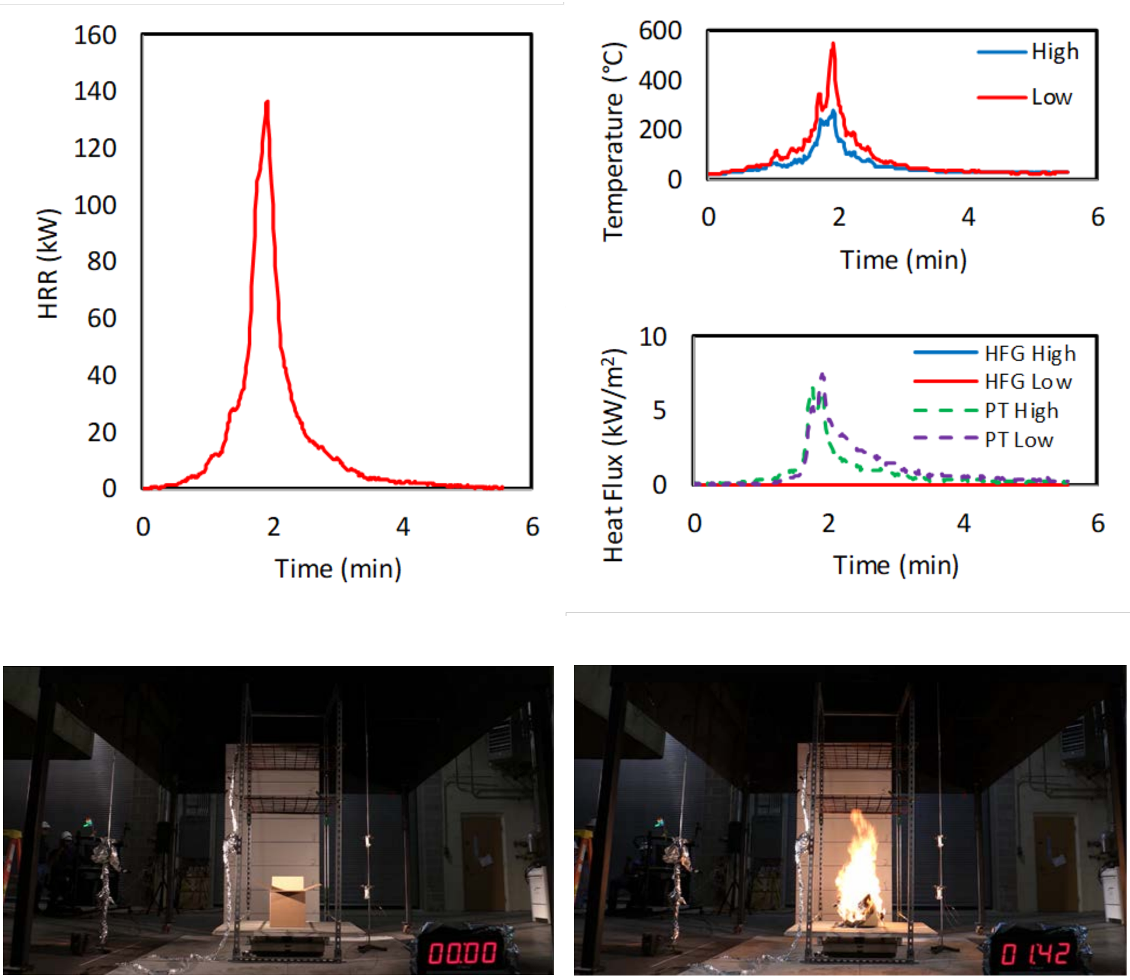

Figure 4 Summary of data obtained from a "Medium Empty Box" test

This report is being released for public comment in October 2019 with expected publication of the final report in early 2020.

CONCLUSIONS

The NRC is working jointly with EPRI under a memorandum of understanding to improve the tools, methods, and data used to improve the realism in Fire PRAs for NPPs. Over the past year, this joint effort has resulted in four reports to support these improvements. These reports address modeling and heat release rates for electric cabinets; rules for considering cabinet to cabinet fire propagation; heat release rate distributions for pumps, motors, and dry transforms; main control board fire modeling, detection, and suppression scenarios more representative of actual operating experience; and heat release rate data for typical NPP transient fuel packages.

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REFERENCES

[1] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 (EPRI 10191989), Washington, DC, and Palo Alto, CA, USA, 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/.

[2] National Fire Protection Association (NFPA): NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, Quincy, MA, USA, 2001.

[3] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, NUREG-2169 (EPRI 3002002936),, Washington, DC, and Palo Alto, CA, USA, 2015, https://www.nrc.gov/docs/ML1501/ML15016A069.pdf.

[4] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, NUREG-2178 (EPRI 3002005578), Washington, DC, and Palo Alto, CA, USA, April 2016,https://www.nrc.gov/docs/ML1611/ML16110A140.pdf.

[5] United States Nuclear Regulatory Commission (NRC): Heat Release Rates of Electrical Enclosure Fires (HELEN-FIRE), Final Report, NUREG/CR-7197, Washington, DC, USA, February 2016, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7197/.

[6] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire Human Reliability Analysis Guidelines NUREG-1921 (EPRI 1023001), Washington, DC, and Palo Alto, CA, USA, 2012. https://www.nrc.gov/docs/ML1221/ML12216A104.pdf.

[7] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), NUREG/CR-7150 (EPRI 1026424), Washington, DC, and Palo Alto, CA, USA, 2012, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7150/.

[8] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 2: Fire modeling guidance for electric cabinets, electric motors, indoor dry transformers, and the main control board, NUREG-2178, Volume 2 (EPRI 3002016052)," Washington, DC, and Palo Alto, CA, USA, in preparation, 2019.

[9] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Methodology for Modeling Fire Growth and Suppression Response of Electrical Cabinet Fires in Nuclear Power Plants, NUREG-2230 (EPRI 3002016051), Washington, DC, and Palo Alto, CA, USA, in preparation, 2019.

[10] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Heat Release Rate and Fire Characteristics of Fuels Representative of Typical Transient Fire Events in Nuclear Power Plants, NUREG-2232 (EPRI 3002015997), Washington, DC, and Palo Alto, CA, USA, in preparation, 2019.

[11] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Methodology for Modeling Transient Fires in Nuclear Power Plant Fire Probabilistic Risk Assessments Draft NUREG-2233 (EPRI 3002016054), Washington, DC, and Palo Alto, CA, USA, in preparation, 2019.

[12] McGrattan, K., et al.: Fire Dynamics Simulator User's Guide, NIST SP 1019-1, National Institute of Standards and Technology (NIST), Gaithersburg, MD, USA, 2017.

[13] United States Nuclear Regulatory Commission (NRC): Technical Basis for Fire Protection Significance Determination Process (IMC 0609, Appendix F) at Power Operations, Washington, DC, USA, 2005.

[14] McGrattan, K. B., M: J. Selepak, and E: J. Hnetkovsky: The Influence of Walls, Corners and Enclosures on Fire Plumes, NIST Technical Note 1984, National Institute of Standards and Technology (NIST), Gaithersburg, MD, USA, 2018.

[15] Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire, 84 Fed. Reg. 31125, June 28, 2019.

[16] Electric Power Research Institute (EPRI): The Updated Fire Events Database: Description of Content and Fire Event Classification Guidance, EPRI 1025284, Palo Alto, CA, USA, 2013.

[17] Electric Power Research Institute (EPRI): Fire Events Database Update for the Period 2010–2014: Revision 1, EPRI 3002005302, Palo Alto, CA, USA, 2016.

[18] Methodology for Modeling Fire Growth and Suppression for Electrical Cabinet Fires in Nuclear Power Plants, 84 Fed. Reg. 31354-311355, July 1, 2019.

[19] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Supplement 1: Fire Probabilistic Risk Assessment Methods Enhancements, NUREG/CR-6850 Supplement 1 (EPRI 1019259), Washington, DC, and Palo Alto, CA, USA, 2010, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/s1/.

Case Study for a Proposed CANDU Fire Probabilistic Risk Assessment Model

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ABSTRACT

This paper examines two different Canadian CANDU fire zones in different nuclear facilities by applying the CANDU Fire PRA methodology and CANDU Fire Database. A qualitative screening methodology for Canadian CANDU reactors was recently developed to assist in performing CANDU Fire Probabilistic Risk Assessments (PRA).

Two fire zones were selected for this study using the screening methodology. The first selected fire zone was screened out, while, the second selected fire zone was identified as a zone that requires to be fully analyzed quantitatively. In order to quantify the FSSA cable damage probability for this fire zone, two different FDS simulations were carried out to help construct the event tree needed for the assessment. The first FDS model was done using two heat detectors and assuming that the zone is fully sprinklered. The second FDS simulation was done assuming no fire protection systems in the zone and no manual intervention. All combustibles were considered to be fully consumed by the fire. The selected fire zone had unqualified FSSA cables, and the total FSSA cable damage probability of this fire zone was 2.2 E-07. If the FSSA unqualified cable was switched to a qualified FSSA cable, the FSSA cable damage probability would be reduced to zero.

INTRODUCTION

CANDU Fire Probabilistic Risk Assessment (PRA) is a systematic methodology for risk quantification of Canadian nuclear facilities and for the identification of the major accident scenarios due to fires. Fire PRA is a beneficial source of information in identifying and quantifying risks and their impacts. Recently, a Fire PRA for CANDU reactors in Canada has been developed [1] as well as a CANDU Fire Database [2]. This case study will implement the developed CANDU Fire PRA to two selected fire zones in two different nuclear facilities in Canada: monitoring room and process room. Both rooms were selected from the CANDU fire survey [3] as they contain fire safe shutdown analysis (FSSA) cables.

The CANDU Fire PRA [1] steps are shown below:

Task 1 – Plant Boundary Definition and Partitioning

Task 2 – Selection of Fire PRA Cables and Components: The selection of target components includes components vulnerable to fire that may cause thermal effects, smoke, or water spray damage to safe shutdown (SSD) appliances and/or equipment. There are some incidents where the fire is at a far distance from the physical location of SSD appliances or equipment but can still damage cables / circuits / wiring associated with SSD components.

Task 3 – Qualitative Screening: The qualitative screening identifies fire analysis compartments that can be screened out from further assessment. A qualitative screening matrix was developed and outlined [4].

Task 4 – Frequency Ignition Frequency and Fire Modeling: The ignition frequency forecasting value for all CANDU reactors in Canada is 0.5 fires per reactor year [2]. The plant operation mode due and prior to fire, the building where fire started, the room (compartment) where fire started, ignition mechanism, detection performance, fire extinguisher system performance, and manual fire performance (weighting factor) are all defined in [2].

Task 5 – Circuit Failure Analysis and Fire Modeling: The purpose of this task is to determine the response of the components in each compartment fire as a way to filter out cables that have no impact on the operation of the component.

Task 6 – Post-Fire Human Reliability Analysis (HRA): This task is divided into two parts; the playback of events and the quantification of human failure events (HFEs).

Task 7 – Quantification of the Fire Risk: Development of fire event and fault trees applies to all fire induced failures as independent events. However, for a given initiating fire, many components of the SSD could be the subject to these failures.

FIRE DYNAMICS SIMULATOR (FDS)

Computational fluid dynamics (CFD) models are one of the most complex tools available to fire safety engineers. These models are established on the basis of physics laws rather than empirical correlation. FDS is a computer program that solves the governing equations of fluid dynamics with a particular emphasis on fire and smoke transport. SMOKEVIEW is the program producing images and animations of the FDS calculations [5].

FDS has been used in this case study for its ability to analyze very detailed scenarios, which require the high precision for implementing the fundamental equations, while taking into consideration aspects such as turbulence description, reaction kinetics, radiation transport, and pyrolysis. FDS also permits users to model the equations (with a finite volume elements approach) modeling the combustion reaction and the transportation phenomenon taking dynamically into account the mutual interactions between the two processes. FDS has been subject to a series of validation tests over different configurations, thus it is considered appropriate for complex geometries. This validation and verification (V&V) project [6] started using FDS, version 4.05. As part of the V&V process, several improvements were made, and a minor bug was corrected in this version. The final version of FDS used in this study is FDS, version 4.06 which includes the changes described above. The mathematical and numerical robustness of FDS has been verified in accordance with the general criteria listed in ASTM E 1355 on “Analytical Tests, Code Checking and Numerical Tests” as mentioned in [6]. NUREG/CR-6850 uses FDS as its main fire-modeling tool [5]. FDS can reliably predict gas temperatures, major gas species concentrations, compartment pressures up to approx. 15 %, as well as heat fluxes and surface temperatures up to approximately 25 % [5].

CANDU FIRE DATABASE

Table 1 below shows the number of fires reported from CANDU reactor facilities in Canada. The three sequential period between 1997 and 2008 show a higher number of reported fires than the rest of the data. After 2008, the number of events per period decreased to a lower level of approximately two fires/ per four-years period. There is no obvious reason or scientific explanation for the spike in number of fires between 1997 and 2008.

Table 5 Number of fires reported from NPPs in Canada

Period	Years	Number of Fires Reported from NPPs in Canada
1	1981–1984	1
2	1985–1988	2
3	1989–1992	2
4	1993–1996	1
5	1997–2000	20
6	2001–2004	39
7	2005–2008	6
8	2009–2012	0
9	2013–2016	2
10	2017–2020	?

However, it is worth mentioning that there were different fire reporting criteria during the period from 1981 up to date. From 1981 to 1996, the fire reporting criteria for each CANDU licensee were listed in their license condition handbook. From 1996 to 2003, the fire reporting criteria followed R-99: Reporting Requirements for Operating Nuclear Power Facilities [7]. From 2003 to 2014, the fire reporting criteria followed S-99: Reporting Requirements for Operating Nuclear Power Plants [8], and from 2014 up to the time being, the fire reporting criteria followed REGDOC-3.1.1, Reporting Requirements for Nuclear Power Plants (NPPs) [9]. There is a clear relation between the increased number of fire reporting since the transition from listing the fire reporting criteria in regulatory document from the license condition handbook in 1996.

There is an increase of reported fires from 1997 to 2004. The first CSA-N293 standard, Fire Protection for CANDU NPPs was developed and approved for publication in February 1997 [10]. The CSA-N293 standard was then adopted in the Canadian NPPs' license conditions handbooks. CSA-N293 provides the minimum fire protection requirements for the design, construction, commissioning, operation, and decommissioning of CANDU NPPs, including structures, systems, and components (SSCs) that directly support the plant and the protected areas. CSA-N293 also provides fire protection requirements for Canadian NPPs.

Some of these requirements have a fire protection program, performing fire protection assessments (Code Compliance Review, Fire Hazard Analysis and FSSA). CSA-N293 also defines the design and installation requirements for fire protection systems. CSA-

N293 implementation requires a lot of effort, workforce, expertise and time to adopt and mature. This can be an explanation for the decrease in the number of fires after 2004.

Therefore, the predicated number of fires from 2017 to 2020 can be assumed to be two fires per period (four years) or 0.5 fires per year, similar to the last reported period of 2013 to 2016.

MONITORING ROOM DESCRIPITION AND ANALYSIS

The monitoring room was selected from the CANDU fire load survey [3]. The monitoring room dimensions are length (L) x width (W) x height (H) = 6 m x 4.5 m x 4.5 m. The combustible load is 478.8 MJ consisting mainly of electric cables. The FSSA cable is 4 m away from the combustibles and located at a height of about 4 m. All ceilings, walls and floors are made of concrete. The monitoring room is not sprinklered and does not have any fire detection. The main cause of ignition in the room is due to electrical faults.

Applying CANDU Fire PRA to the Monitoring Room

Applying the first three tasks of the CANDU Fire PRA to the monitoring room gives the following:

Task 1 – Plant Boundary Definition and Partitioning: The monitoring room has concrete walls/ceilings/floors that are considered fire barriers. They all have a fire resistance rating of more than one hour. All fire barrier elements, such as fire door sand penetrations, exceed the minimum rating of one hour.

Task 2 – Selection of Fire PRA cables and components: The monitoring room contained a FSSA cable. Therefore this room was identified as a FSSA cable credited fire zone that can impact safety systems during a fire event.

Task 3 – Qualitative screening [4]: The monitoring room has a combustible load < 500 MJ with a floor area between 26 m^2 and 571 m^2. Therefore it considered a low hazard level room and will be screened out from further analysis, see Table 2 below [4].

Table 2 Hazard levels risk table

Low	Medium	High
Target < 205 ºC and/or < 6 kW/m^2	> 205 ºC and < 330 ºC and/or > 6 kW/m^2 and < 11 kW/m^2	Target > 330 ºC and/or > 11 kW/m^2

Low Hazard Level: Total combustibles in fire zone that is < 500 MJ and with an area between 26 m^2 and 571 m^2. The potential fire is at a distance of more than 5 meters away from the FSSA cable (where the FSSA cable is not IEEE-383 qualified) can be screened out from further analysis.

Medium Hazard Level: If the FSSA cable is IEEE-383 qualified, then the following fire zones can be screened out from further analysis, and if the FSSA cables are not IEEE-383 qualified, then the following fire zones should be screened in for further analysis. Total combustibles in fire zone that is < 500 MJ and occurs in areas between 26 m^2 and 571 m^2, and the potential fire is assumed in a distance of 2 m to 5 m for the FSSA cable. The total amount of combustibles in fire zone provides a fire loaf of < 5000 MJ in areas of 571 m^2 and more, and the potential fire is assumed in a distance of 7 m or more to the FSSA cable.

High Hazard Level: Any other fire that is not listed under the Low Hazard Level and/or Medium Hazard Level should be screened in for further analysis.

The monitoring room has a combustible load of < 500 MJ distributed on a floor area between 26 m^2 and 571 m^2 (representing a fire load density of 0.9 to 19 MJ/m^2) therefore, the monitoring room is considered a low hazard level room and will be screened out from further analysis.

PROCESS ROOM DESCRIPITION AND ANALYSIS

The process room was selected from the CANDU fire load survey [3]. The process room dimensions are L x W x H = 25 m x 23 m x m. The combustible load is approx. 4,400 MJ; the combustibles are cable trays. The distance between the FSSA cable the combustibles is 7 m and at a height of 6 m with the FSSA cable being a non-qualified cable. All ceilings, walls and floors are made of concrete. The process room is sprinklered by 16 sprinkler heads and has two heat detectors. The main cause of ignition in the room is an electrical fault.

The damage threshold temperature of IEEE-383 qualified cables is 330 ºC, for non-qualified cables it is 205 ºC. The damage threshold heat flux of IEEE-383 qualified cables is 11 kW/m^2, for non-qualified cables 6 kW/m^2 [12].

Applying CANDU Fire PRA to the Process Room

Task 1 – Plant Boundary Definition and Partitioning: The process room has concrete walls/ceilings/floors that are considered fire barriers, which provide fire resistance rating of more than one hour and can be credited in partitioning and all elements of the barrier, such as fire door and penetrations, exceeds the minimum rating of one hour.

Task 2 – Selection of Fire PRA cables and components: A FSSA cable inside the fire zone (process room) was identified as FSSA component credited in this analysis that can impair safety systems during fire event.

Task 3 – Qualitative screening [4]: The monitoring room has a combustible load of approx. 4,400 MJ. The distance between the FSSA cable the combustibles is 7 m, and the FSSA cable is a non-qualified cable. This is considered a medium hazard level room and will be screened in for further analysis.

Task 4 – Ignition frequency and fire modeling: The ignition frequency for all CANDU reactors in Canada is 2 fire events between 2017 and 2020 (four year) and hence the expected annual ignition frequency is 0.5 fires per reactor year [ry] [2].

There are 19 operating CANDU reactors in Canada.

Ignition frequency per reactor =
number of fires year and number of operating reactors in Canada (1).

Therefore, the ignition frequency per reactor is (0.5 fires / year / 19 operating reactors) providing a frequency of approx. 2.6 E-0^2 fires /yr.

Task 5 – Circuit failure analysis and fire modeling: The FSSA cable in the process room have an impact on the reactor's safety systems and on the operation of the component. For this step, two FDS simulations for the process room will be conducted. The first FDS model will include the fire detection system as well as the fixed sprinkler extinguishing system. The second simulation will neither include a detection system nor a sprinkler system and will therefore simulate the fire until the entire combustibles available are completely burnt. The outcome of these two FDS simulations is essential to identify the event tree analysis consequences.

Both FDS models will simulate a fire of a power cable tray due to an electric fault. The fire will occur around midnight when the room is not occupied. There are no windows or opening in this room, except for the fire door. For maximum ventilation, the door was propped open in both simulations.

- Concrete properties [14, [15]:
  - Thermal conductivity: 0.001 kW/mK,
  - Density: 2000 kg/m^3,
  - Specific heat; 0.88 kJ/kg,
  - Wall thickness: 0.3 m;
- Cable Tray (Nylon/PVC) properties [14]:
  - HRRPUA for cable: 231 kW/m^2

Combustible Loads for the Two Simulations

For miscellaneous power cable components of the fuel load, the power cable was modelled as a complete cable neglecting the small amount of additional plastics. The approximation of the cable structure as a planar surface is illustrated in Figure 1. The first and third layers consist of cable sheath material. The inner layer is a homogenous mixture of the insulation and the filler materials. Conductors are non-combustible and thus neglected in the model for simplicity and to save computational time in large-scale simulations [14], [15]. The fuel load survey performed [3] has identified different diameters for power cables in Canadian NPPs. The most common diameter of power cables in Canadian NPPs is 6 cm [3].

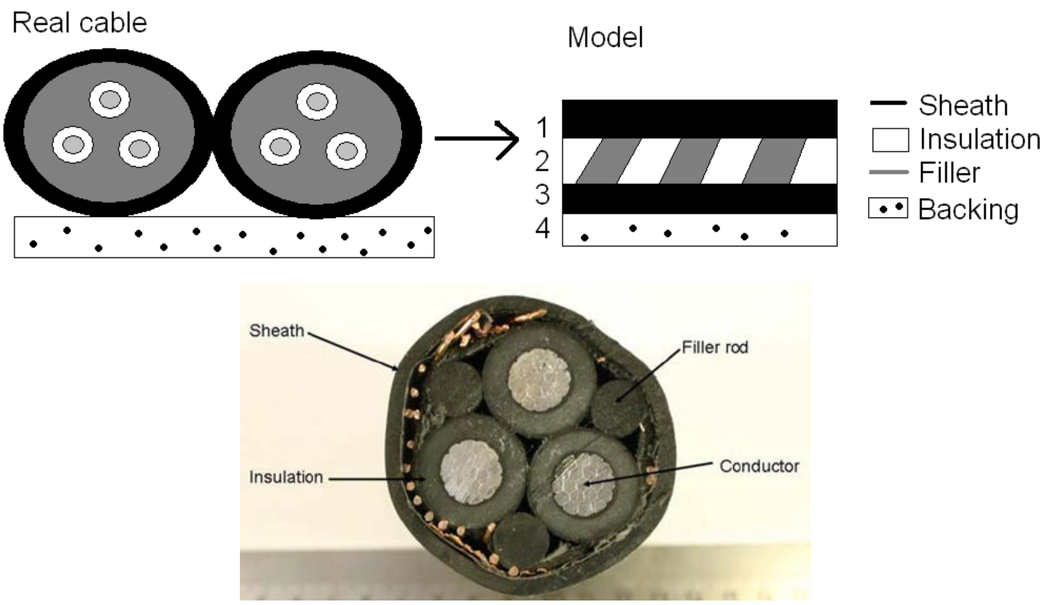

Figure 6 Real cable versus model [14], [15]

- FDS input file:

```
&MATL ID='Sheath',
     SPECIFIC_HEAT=1.0,
     CONDUCTIVITY=0.05,
     DENSITY=1501.0/
&MATL ID='Filler',
     SPECIFIC_HEAT=3.0,
     CONDUCTIVITY=0.15,
     DENSITY=950.0/
&MATL ID='Insulation',
     SPECIFIC_HEAT=3.5,
     CONDUCTIVITY=0.2,
     DENSITY=1039.0/
&SURF ID='Cable',
     COLOR='GRAY 20',
     HRRPUA=589.0,
     RAMP_Q='Cable_RAMP_Q',
     IGNITION_TEMPERATURE=205.0,
     BURN_AWAY=.TRUE.,
     MATL_ID(1,1)='Sheath',
     MATL_ID(2,1:2)='Filler','Insulation',
     MATL_ID(3,1)='Sheath',
     MATL_MASS_FRACTION(1,1)=1.0,
     MATL_MASS_FRACTION(2,1:2)=0.5,0.5,
     MATL_MASS_FRACTION(3,1)=1.0,
     THICKNESS(1:3)=0.025,0.01,0.025/
```

- Cable tray:

  There are two cable trays, each tray has the following volume:
  LWH = 2 m x 1 m x 0.3 m = 0.6 $m^{2.}$

  One cable tray area = 2 (L x W) + 2 (L x H) + 2 (W x H) (2)

  Each cable tray area = 2 (2 m x 1 m) + 2 (2 m x 0.3 m) + 2 (1 m x 0.3 m) = 5.8 m^2

  Total Area = 5.8 m^2 x 2 = 11.6 m^2

  The t-squared parabolic growth equation is given by $Q = \alpha\, t^2$ (3),

  where Q is the HRR [kW], α is the fire growth coefficient (kW/s^2), and t is time (s).

  HRR = HRRPUA x total area (4),

  HRR = 231 kW/m^2 x 11.6 m^2 = 2679.6 kW, α = ultra-fast ~ 0.1876 kW/s^2; therefore, approx. 2679.6 kW will be attained within 120 s.

  Total MJ = HRRPUA x AREA x time x F value (5)

  Total MJ = {231 x 11.6 x (1635 - 127) x 1} + {(2) 231 x 11.6 x 127 x 0.5} = 4,040,836.8 + 340,309.2 = 4,381.2 MJ

- FDS input:

  &RAMP ID='Cable_RAMP_Q', T=0.0, F=0.0/

  &RAMP ID='Cable_RAMP_Q', T=120.0, F=1.0/

  &RAMP ID='Cable_RAMP_Q', T=1628.0, F=1.0/

  &RAMP ID='Cable_RAMP_Q', T=1755.0, F=0.0/

- Simulations conditions:
  - Domain, room dimensions XYZ: 25.0 m x 23.0 m x 7.0 m;
  - Thermocouple and heat flux sensor XYZ: 0.6, 20.3, 6.0;
  - Location of the cable (First Tray) XYZ: 5.0, 13.8, 0.9 to XYZ: 7.0, 14.8, 1.2
    (Second Tray) XYZ: 5.0, 13.8, 1.5 to XYZ: 7.0, 14.8, 1.8;
  - Distance between thermocouple/heat flux sensor and fire:
    $\{(5.0 - 0.6)^2 + (14.8 - 20.3)^2\}^{1/2} = 7.0$ m;
  - Door open, vent XYZ: 5.0, 13.8, 0.1 to XYZ: 6.7, 13.8, 2.6;
  - Mesh size no. of elements 250 x 230 x 70 = 4,025,000; and
  - Grid size: XYZ: 0.1 m x 0.1 m x 0.1 m.
- Boundary conditions:
  - Burner: HRRPUA 233.3 kW/m^2;
  - Burner location XYZ: 6.0, 14.4, 0.1;
- Sprinkler system conditions:
  - The sprinkler system (wet pipe) has 16 sprinkler heads, RTI = 148.0, and activation temperature = 74.0 ºC.
- FDS input: (Simulation 1):

  &DEVC ID='SPRK01', PROP_ID='Default_Water Spray', XYZ=2.3, 22.5,6.89/

  &DEVC ID='SPRK02', PROP_ID='Default_Water Spray', XYZ=2.3,16,6.89/

  &DEVC ID='SPRK03', PROP_ID='Default_Water Spray', XYZ=2.3,9,6.89/

  &DEVC ID='SPRK04', PROP_ID='Default_Water Spray', XYZ=2.3,2.5,6.89/

  &DEVC ID='SPRK05', PROP_ID='Default_Water Spray', XYZ=9,22.5,6.89/

  &DEVC ID='SPRK06', PROP_ID='Default_Water Spray', XYZ=9, 16, 6.89/

  &DEVC ID='SPRK07', PROP_ID='Default_Water Spray', XYZ=9, 9, 6.89/

  &DEVC ID='SPRK08', PROP_ID='Default_Water Spray', XYZ=9,2.5,6.89/

  &DEVC ID='SPRK09', PROP_ID='Default_Water Spray', XYZ=16, 22.5,6.89/

```
&DEVC ID='SPRK10', PROP_ID='Default_Water Spray', XYZ=16, 16, 6.89/
&DEVC ID='SPRK11', PROP_ID='Default_Water Spray', XYZ=16, 9, 6.89/
&DEVC ID='SPRK12', PROP_ID='Default_Water Spray', XYZ=16, 2.5,6.89/
&DEVC ID='SPRK13', PROP_ID='Default_Water Spray', XYZ=23,22.5,6.89/
&DEVC ID='SPRK14', PROP_ID='Default_Water Spray', XYZ=23,-16, 6.89/
&DEVC ID='SPRK15', PROP_ID='Default_Water Spray', XYZ=23, 9, 6.89/
&DEVC ID='SPRK16', PROP_ID='Default_Water Spray', XYZ=23,22.5,6.89/
```

Simulation 1 Output

Figure 2 below shows that maximum temperature measured at the thermocouple 7 m away from the fire is 129.8 ºC and was reached 28.3 min after the start of the simulation time. Figure 7 also shows that the maximum heat flux measured at the heat flux sensor 7 meters away from the fire is 0.5 kW/m^2 and was reached at 32.8 minutes, and the maximum HRR reached is 2104.5 kW after 31.2 min.

The first sprinkler head SPRK08 was actuated after 19 min, when the temperature reached 74 ºC. A few seconds later, SPRK07 and SPRK03 were actuated. At 20 min, SPRK04 was actuated and 5 min later, SPRK06 was actuated. The last sprinkler head to be actuated was SPRK02 after around 30 min in the model. However, the fire continued to grow from the time of the first actuation of sprinkler head SPRK08"to the last actuation of sprinkler head SPRK02, and the fire was not fully suppressed. However, the actuated fire sprinkler heads managed to control the fire and prevented the rest of the sprinkler heads from actuating. The remaining sprinkler heads were not actuated, as they never reached the threshold temperature of 74 ºC. The sprinkler performance illustrates that the capacity, response time and reliability of the sprinkler system designed for the process room is sufficient to control the worst-case scenario fire. The first heat detection took place at 71 ºC after 15.5 min simulation time and the second heat detection never reached 71 ºC and therefore was not actuated.

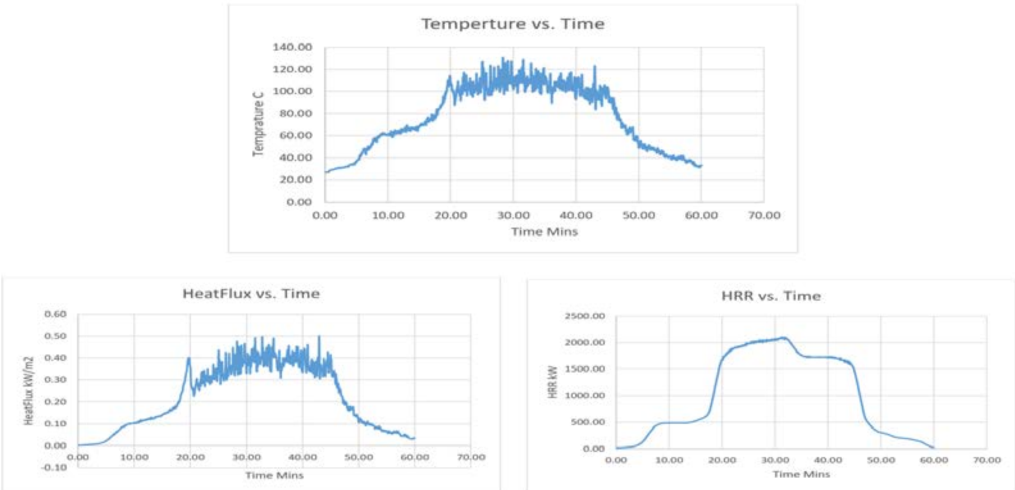

Figure 2 Simulation 1 output: a) temperature vs time, b) heat flux vs time: c) HRR vs time

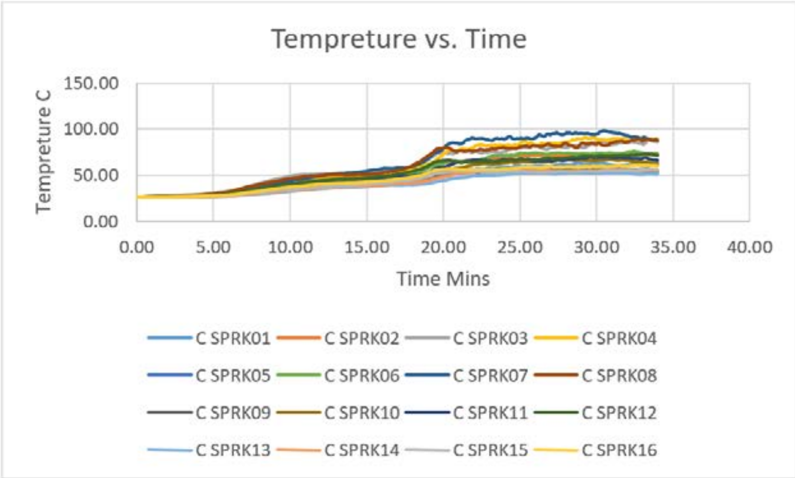

Figure 3 Simulation 1 sprinkler system: temperature vs. time

Simulation 2 Output

This fire development curves below shows a fuel limited fire. More fuel became involved in the fire (up to 20), the energy level continues to increase until all of the fuel available is burning (fully developed from 10 min until around 49 min). Then as the fuel is burned away, the energy level begins to decay (49 min to 60 min). Figure 3 below shows that maximum temperature measured at the thermocouple 7 m away from the fire is 220.7 ºC and was reached 33.5 min from the start of the simulation time. Figure 3 also shows that the maximum heat flux measured at the heat flux sensor 7 m away from the fire is 1.6 kW/m^2 and was reached after 40.7 min, and the maximum HRR reached is 2447 kW at 31 min.

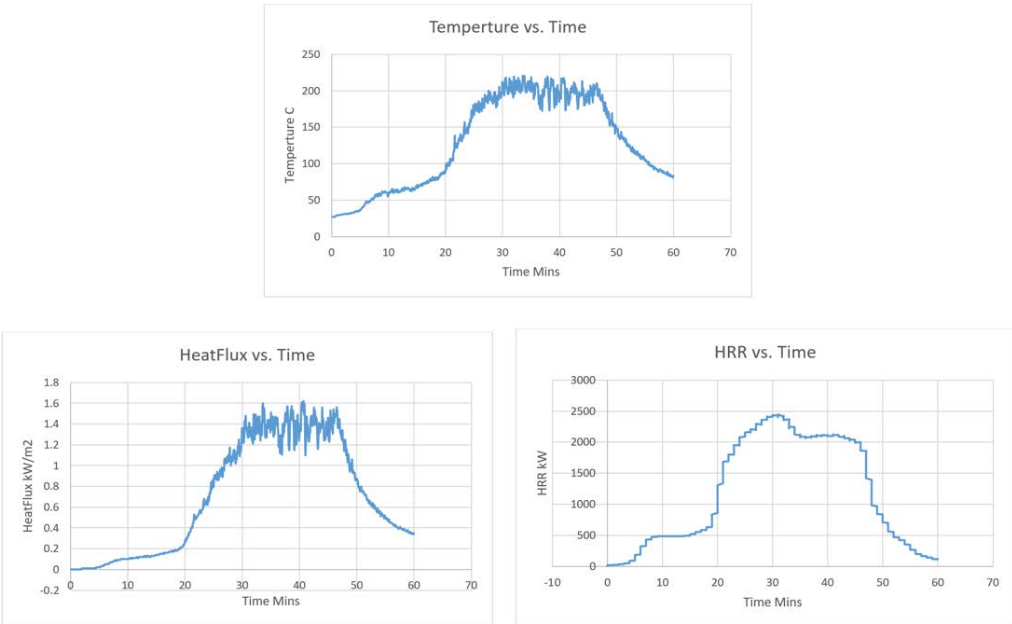

Figure 4 Simulation 4 output: a) temperature vs time, b) heat flux vs time, c) HRR vs time

Task 6 – Post-Fire Human Reliability Analysis (HRA): For this fire scenario, there is no HRA analysis since the fire occurred at midnight while the room was not occupied, and no credit was given to manual intervention from any plant worker.

Task 7 – Quantification of the fire risk: Event tree analysis (ETA) provides a process that combines likelihoods, fire protection system (FPS) success probability and consequences into risk and risk reduction effects. ETA steps are divided into:

1. Initiating fire sources events
2. Fire protection systems performance
3. Fire outcomes
4. Consequences at the target

Risk level = ∑ initiating event likelihood x FPS success probability x consequences

1. Initiating fire sources events:

The frequency of a fire occurring is 0.5 (fires per year) in all Canadian CANDU nuclear stations. The CANDU fire database provides percentages for rooms where the fire started and percentages for causes of ignition [2]. For the process room this value is 25.33 %, and for electrical fire causes it is 41.33 %.

The frequency of a fire to develop in a process room due to an eclectic fault is:

frequency of a fire in all Canadian CANDU nuclear stations x
process room frequency x electrical fire cause frequency (6).

The number of fires per year in the process room caused by an electric fault = 0.026 (fires per year) x 0.2533 x 0.4133 = 0.0027 fires per year

2. Fire protection system performance:

In fixed wet-pipe sprinklers suppression system is being credited, it is appropriate to identify the type failure rate to be 2 % [16]. Heat detectors have a failure rate of 0.3 2% [18]. This includes all smoke detectors, restorable heat detectors, and non-restorable heat detectors. The on-site manual fire department is credited for fire suppression in virtually all areas of the plant. A review of plant training records and past fire drills were found appropriate to determine the anticipated response time of less from 6 to 8 min to the process room. All past fire drills and actual fires on this site were 100 % successfully extinguished. For the purpose of this analysis, it was assumed that the success rate will be 99.9 %.

3. Fire outcomes:

As illustrated in fire scenario development and results part of this paper, there were two FDS models. Both simulation 1 and simulation 2 had the same area/height and same combustible load. Simulation 1 included 2 heat detectors and was sprinklered. Simulation 1 maximum temperature and heat flux output did not exceed the threshold of the non-qualified FSSA cable (205 ºC and/or 6 kW/m^2). Simulation 2, which did not assume any fire protection system being present, exceeded the temperature threshold of 205 ºC and can damage the FSSA cable.

Table 2 Simulation 1 and 2 output results with FDS errors

Simulation Number	Maximum Temperature [ºC]	Maximum Heat Flux [kW/m^2]
Simulation 1	129.8 ± 32.4	0.5 ± 0.1
Simulation 2	220.7 ± 55	1.6 ± 0.4

4. Consequences at the target:

Simulation 1 heat detector reached its actuation temperature (71 ºC) 15.5 min after the start of the simulation. Simulation 2 shows that the damage temperature of 205 ºC ± 25 % was reached after 24.5 min (164 ºC). From past fire drills it was found that the anticipated response time is between 6 and 8 min for the process room. Therefore, the fire department will have sufficient time to extinguish the fire.

Fire Protection System (FPS) Controls

1. Sprinkler system success rate 98 % [16],
2. Detection system success rate 99.68 % [17], and
3. Fire department success rate 99.9 % (plant specific).

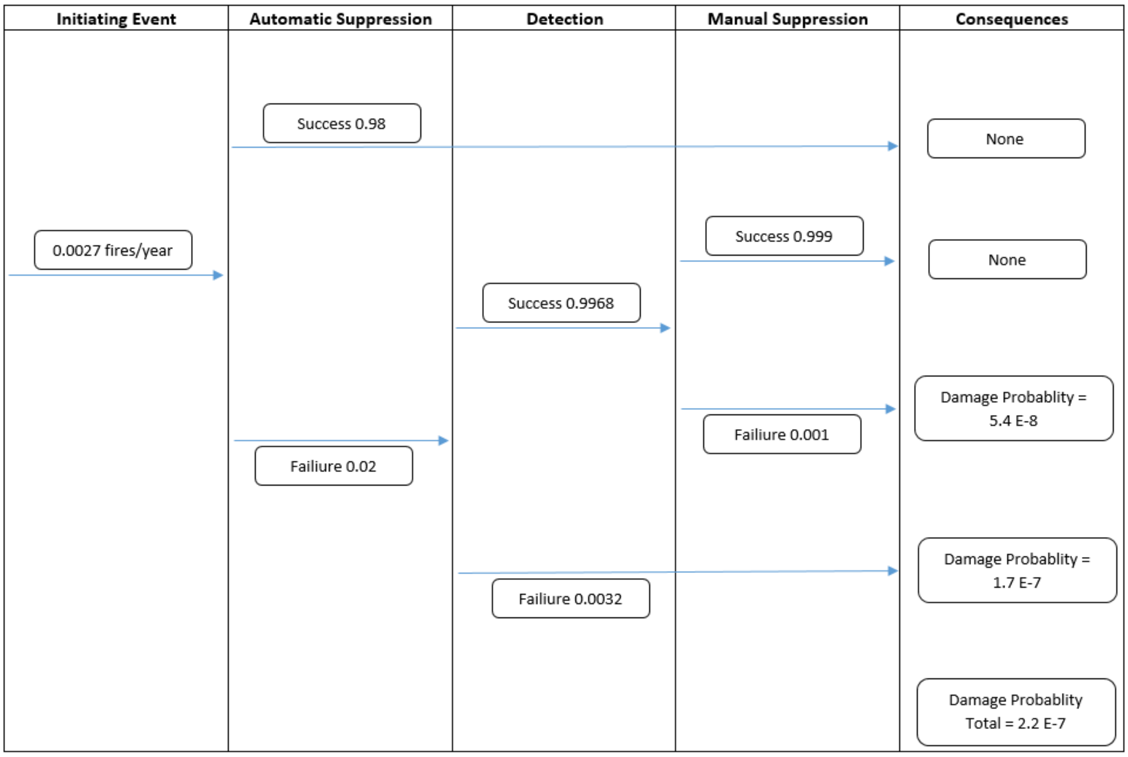

Figure 5 Simulation 2 event tree

The total FSSA cable damage probability of the process room is 2.2 x E-07. If the FSSA nonqualified cable was replaced by a qualified FSSA cable, this will exclude the process room for any FSSA cable damage probability.

CONCLUSION

This case study has demonstrated the implementation of the CANDU Fire PRA model on two different fire zones in two different Canadian nuclear facilities and determined the FSSA cable damage probability. The CANDU Fire PRA was carried out using Canadian past data, CANDU fuel survey, CANDU qualitative , and CANDU Fire PRA.

REFERENCES

[1] Shalabi, H., and G. Hadjisophocleous: CANDU Fire Probabilistic Risk Assessment (PRA) Model, CNL Nuclear Review, 18 December 2018, https://doi.org/10.12943/CNR.2017.00020.

[2] Shalabi, H., and G. Hadjisophocleous: CANDU Fire Database, CNL Nuclear Review, 18 December 2018, https://doi.org/10.12943/CNR.2017.00019.

[3] Shalabi, H., and G. Hadjisophocleous: CANDU Fire Load Densities in Canadian Nuclear Plants, CNL Nuclear Review, 6 August 2019, https://doi.org/10.12943/CNR.2019.00004.

[4] Shalabi, H., and G. Hadjisophocleous: CANDU Fire PRA Qualitative Analysis Screening Step, CNL Nuclear Review, Chalk River, ONT, Canada, under consideration 2019.

[5] McGrattan, K. (Ed.): Fire Dynamics Simulator (Version 4), Technical Reference Guide, NIST Special Publication 1018, National Institute of Standards and Technology (NIST), Gaithersburg, MD, USA, March 2006, https://www.nist.gov/publications/fire-dynamics-simulator-version-4-technical-reference-guide.

[6] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research and Electric Power Research Institute (EPRI): Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Volume 7: Fire Dynamics Simulator (FDS), NUREG-1824 and EPRI 1011999 Final Report, Washington, DC and Palo Alto, CA, USA, May 2007, https://www.nrc.gov/docs/ML0717/ML071730543.pdf.

[7] Atomic Energy Control Board (AECB): Regulatory Document R-99: Reporting Requirements for Operating Nuclear Power Facilities, Ottawa, ONT, Canada, January 1995.

[8] Canadian Nuclear Safety Commission (CNSC): Regulatory Standard Reporting Requirements for Operating Nuclear Power Plants, S-99, Ottawa, ONT, Canada, March 2003, https://nuclearsafety.gc.ca/pubs_catalogue/uploads/S99en.pdf.

[9] Canadian Nuclear Safety Commission (CNSC): REGDOC-3.1.1, Version 2, Reporting Requirements Reporting Requirements for Nuclear Power Plants, Ottawa, ONT, Canada, April 2016, https://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/published/html/regdoc3-1-1-v2/index.cfm.

[10] Standards Council of Canada (CSA): CAN/CSA-N293-95 (R2006), Fire Protection for CANDU Nuclear Power Plants November 1995, reaffirmed February 2006, superseded by N-293-07 (R2012), https://www.scc.ca/en/standardsdb/standards/23327.

[11] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA

Methodology for Nuclear Power Facilities, NUREG/CR-6850 and EPRI 1011989 Final Report, Washington, DC and Palo Alto, CA, USA, September 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/.

[12] IEEE Standards Organization: IEEE 383-2015 - IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities, October 2015.

[13] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research and Electric Power Research Institute (EPRI): Fire Probabilistic Risk Assessment Methods Enhancements, NUREG/CR-6850 Supplement 1, and EPRI 1019259 Technical Report, Washington, DC and Palo Alto, CA, USA, September 2010,
https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/s1/.

[14] Matala, A., and S. Hostikka: Probabilistic simulation of cable performance and water based protection in cable tunnel fires, Nuclear Engineering and Design, 241-12, pp. 5263-5274, 2011.

[15] Matala, A., and S. Hostikka: Pyrolysis Modelling of PVC Cable Materials, Fire Safety Science 10, pp. 917-930. 10.3801/IAFSS.FSS.10-917, 2011.

[16] Parkinson, W., et al.: Automatic and Manual Suppression Reliability Data for Nuclear Power Plant Fire Risk Analyses, in: Evaluation of Fire Suppression System Reliability; NSAC-179L, Electric Power Research Institute (EPRI), Palo Alto, CA, USA,1992.

[17] Carter, S.: Inspection, Testing and Maintenance Records: A Window into System Reliability, Orr Protection Systems, Inc., March 2008,
http://www.nfpa.org/~/media/files/news-and-research/resources/research-foundation/foundation-proceedings/carter.pdf?la=en.

3.6 Joint FSEP 2019 and SMiRT Fire Seminar Plenary Session

A joint plenary session of the SMiRT Post-conference Fire Seminar and the FSEP 2019 conference took place after the Seminar at the last conference day. The in total four high level presentations covered topics of interest for both groups of experts involved. Three of the presentations were given by Seminar Chairs providing a broad spectrum of topics important also for future expert exchanges.

The first of these three presentations titled “Fire Safety at Nuclear Sites: Challenges for the Future – An International Perspective” was given by M. Roewekamp, GRS (Germany) as Chair of various international expert groups on fire safety in nuclear installations. This presentation provided an overview on already resolved issues related to fire safety, indications on aspects to be investigated in more detail and challenges the nuclear fire experts´ community will face in the near future. Abstract and slide presentation are provided at the end of this section.

Another presentation was given by S. Suard, IRSN (France) on the international experimental research project by the OECD/NEA on fires in complex geometries of nuclear installations called PRISME. After an overview of the experiments in the first two project phases carried out in the past, the actually ongoing experimental series of the third phase (PRISME 3) focussing on cable fires and fires in electrical cabinets was presented. Again, abstract as well as slide presentation are provided at the end of this section.

The third presentation from the viewpoint of the SMiRT Fire Seminar was given by K. Shirai, CRIEI-NRRC (Japan). A proposal of a methodological approach for preventing HEAF(*H*igh *E*nergy *A*rcing *F*ault) induced fires as developed in Japan for electrical cabinets of different voltage levels based on the operating experience with such events and experimental and analytical research activities carried out on a national as well as international basis was presented and discussed. The respective paper is included to this section hereafter.

Fire Safety at Nuclear Sites: Challenges for the Future – An International Perspective

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ABSTRACT

Nuclear industry worldwide actually faces new challenges with the nuclear phase-out in some European countries on the one hand, but an increase of new built power stations, in particular Gen. III+ and VI light water reactors, but in future also advanced and/or small modular reactors (AMR/SMR).

Fire safety plays an important role over the whole lifetime of any nuclear installation and does not only affect one individual reactor unit. Fires have the potential to affect a whole nuclear site and to occur during construction phase, the entire plant operational phases including the post-commercial safe shutdown phase of a power reactor, as well as the decommissioning and deconstruction phase.

Assessing fire safety of nuclear site as a whole with all its reactor units and radioactive sources will in several countries cover new challenges arising from new types of reactors being constructed and operated on a site, where other reactors and sources such as dry fuel storage facilities are being operated and probably also older installations are under decommissioning.

The International Atomic Energy Agency (IAEA) as well as the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) committees CNRA (*C*ommittee on *N*uclear Regulatory *A*ctivities) and CSNI (*C*ommittee on the *S*afety of *N*uclear *I*nstallations) with their working groups are on the way to take care of these developments and changes in nuclear industry trying to be well-prepared for the future.

The IAEA has recently developed an updated Safety Guide on "*Protection against Internal Hazards in the Design of Nuclear Power Plants*" [1] merging the existing former Guides on "*Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants*" [2] und "*Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants*" [3] providing guidance on how to address the protection against fires for power reactors being designed considering the whole site with all the already existing nuclear facilities. An update of the IAEA PSA Guide [4] is also intended to be started. This update will also cover internal hazards such as fires including hazard combinations for risk aggregation from all nuclear sources (reactors, spent fuel pools (SFPs), nuclear waste storage and treatment facilities, etc.), various types of risk (from plant internal events and hazards), and different plant operational states (POS).

The OECD/NEA also covers different activities regarding fire safety at nuclear installations.

Particularly, the CSNI Working Group on Risk Assessment has conducted several tasks related to the fire risk at nuclear sites. An international workshop on Fire Probabilistic Risk Assessment (PRA) conducted in 2014 revealed valuable insights on the still existing gaps in Fire PSA methods and data, but also demonstrating that this tool is mature

enough to support risk-informed decision making [5]. An update of the WGRISK Technical Opinion Paper (TOP) on Fire Probabilistic Safety Assessment for Nuclear Power Plants was published in early 2019 [6] providing the recent international experts´ view on the state of Fire PSA as performed in support of NPP design and operation covering e.g. the following key messages:

- For many WGRISK member countries, at the industry level, fire continues to be an important risk contributor.
- The risk due to fires at a particular plant site is strongly dependent on plant-specific factors in design and operation.
- Fire PSA insights regarding major contributors to the total fire risk are generally aligned with operating experience and appear to be largely representative of the expected plant responses. However, some aspects of the quantitative results of modern Fire PSA can be conservative.
- Methods, models, tools and data continue to improve, and the practice of Fire PSA continues to mature.
  - The PSA expert community generally agrees on the overall approach for performing Fire PSA.
  - The level of realism in Fire PSA for a plant often represents a trade-off between the level of modelling effort and the needs of the plant's Fire PSA application.
  - Key sources of uncertainty, including potential non-conservatisms as well as conservatisms, have been identified. Many are being addressed through ongoing research and development (R&D).
- Fire PSA is a valuable tool that provides useful results and insights in support of risk-informed decision making. As with all risk-informed decisions, it is important that the decision makers are informed of potential biases and other uncertainties associated with these results and insights.

The experts involved in the TOP expect ongoing and future improvement activities will improve the broad acceptance of Fire PSA for a wider set of applications and recommend that organizations continue supporting R&D on key topics identified, and carrying out activities to collect, manage, and provide access to quality data, facilitating the sharing of challenges, solutions, uncertainties, uses, and good practices; and to develop practical guidance based on the above.

One of the successfully ongoing activities under the auspices of the CSNI is the Database Project FIRE (*F*ire *I*ncidents *R*ecords *E*xchange). This Project is amongst others also supporting Fire PSA. The need for this Database emerged in the late 1990s when it became evident that the only international recording of fire events by the joint IAEA/NEA International Reporting System for Operating Experience (IRS) was not suitable for specific analysis and use in risk assessment. The purpose of the FIRE Database Project actually in its fifth phase (2016 – 2019) is therefore to provide a platform for meanwhile fourteen countries (two from Asia, ten from Europe and another two from North America) to collaborate and exchange fire data and thereby to enhance the knowledge of fire phenomena and in turn improve the quality of risk assessments that require fire related data and knowledge.

The objectives of the FIRE Project as defined according to the Terms and Conditions are as follows:

- Collect fire event experience (by international exchange) in an appropriate format in a quality-assured and consistent database;
- Collect and analyze fire events over the long term to better understand such events and their causes, and to encourage their prevention;

- Generate qualitative insights into the apparent and root causes of fire events in order to derive approaches or mechanisms for their prevention and to mitigate their consequences;
- Establish a mechanism for efficient operation feedback on fire event experience including the development of policies of prevention, such as indicators for risk-informed and performance-based inspections;
- Record characteristics of fire events in order to facilitate fire risk analysis, including quantification of fire frequencies.

Thus, the FIRE Database Project does not only cover qualitative aspects but also serves as a data source for Fire PSA.

Actually, the Database covers more than 520 fire events having occurred in nuclear power plants (NPPs) of FIRE member countries with 501 of them being statistical evident events representing more than 9415 years of observation for different phases of the operational plant lifetime.

The FIRE Database can be used for

- identifying all types of events and scenarios to be included in PSA models ensuring that all mechanisms are accounted for,
- as support to Fire PSA by real data from NPP operating experience, particularly to evaluate fire occurrence frequencies, and
- for comparing national fire event data from member countries with the accumulated international data.

Recent applications focus on generic fire occurrence frequencies estimations relevant for Fire PSA (compartment specific occurrence frequencies by buildings as well as component specific ones), the significance of event combinations including fire events, HEAF (*h*igh *e*nergy *a*rcing *f*ault) induced fire events as non-negligible PSA contributor, and apparent causes of fire events. A detailed overview on the FIRE Database is provided in the contribution titled "*Operating Experience with Fires in Nuclear Installations – The OECD/NEA FIRE Database*" presented at the 16th SMiRT 25 Post Conference Seminar on Fire Safety in Nuclear Power Plants and Installations [7] in conjunction with FSEP 2019.

With respect to generic fire occurrence frequencies relevant mainly for Fire PSA, the findings revealed valuable products from the FIRE Database, but also indicated a few challenges. The most important of these findings and challenges are as follows:

- Well-defined observation times sub-divided into the different POS (full power operation as well as low power and shutdown states including the post-commercial operation safe shutdown phase), but also covering construction and decommissioning phases, are meanwhile available for all reactors in the Database.
- Generic compartment (or room) specific fire frequencies for various buildings based on average numbers of rooms of different types can be principally generated. These are statistically relevant for reactors from those countries reporting all events, not only those ones being reportable to given national criteria. One still remaining challenge is that the amount of statistically relevant data is partly not sufficiently high.
- Generic component specific ignition frequencies can already be provided for selected components, a comparison with national data from Fire PSA is in principle possible. Main challenge in this context are fire frequencies for cables, which are still difficult to derive. For cable fire frequencies, either information on cables by segments or by length is needed. Only limited information is available in the Database so far. Moreover, for simulations of cable fire scenarios within Fire PSA, specific data (e.g., on materials, burning rate, ventilation conditions, etc.) are needed. For enabling analysts to model such, partly more frequent scenarios it is intended to collect

for cable fire events recorded in the Database in future, as far as available information on cable characteristics and layout.

- Event combinations of fires and other anticipated events (covering correlated as well as uncorrelated events) represent nearly 8 % of the events stored in the Database. Analyses of these event combinations have clearly indicated that fires induced by HEAF represent an important contributor to the overall risk and need to be investigated for risk aggregation purposes. HEAF induced fires represent more than 10 % of the fire events in the Database and result in a non-negligible generic fire frequency of 7 E-03 /ry underpinning that such events need to be addressed in Fire PSA.
- Generic frequency data derived from the FIRE Database can be applied at least as a priori information in Bayesian approaches if the site specifically available data are insufficient for a nuclear site under investigation.
- Although detailed information from the operating experience is being collected in the Database as far as available, for modelling fire detection and suppression within the fire specific event trees more details on the time sequences are needed, such as fire detection and alarm times or information on start and end of fire suppression.
- The analysis of apparent causes of fire events in the Database demonstrated that it is possible to identify potential precursor events. Moreover, insights on potential backfitting measures and their effect on the fire specific core and/or fuel damage frequencies (CDF/FDF) can be gained. However, one finding is that more detailed investigations of the root causes of fire events are needed. In this context, it is challenging that the root cause codings are not yet complete and exhaustive, requiring improved analyses for reducing uncertainties and increasing the level of confidence with respect to the analytical results.

Moreover, one recently started activity concerns the on-site plant fire brigades, their organization, responsibilities and challenges, which are quite different in member states. The activity shall help members to find good practices which may be implemented in their countries. Details can be found in the corresponding contribution to the 16th SMiRT 25 Post Conference Fire Seminar [8].

In addition, a project report titled “*Survey of Member Countries’ Nuclear Power Plant Fire Protection Regulations by the OECD Nuclear Energy Agency (NEA) Fire Incidents Records Exchange (FIRE) Database Project – Topical Report No. 2*” is being prepared as NUREG/IA Report published by the U.S. Nuclear Regulatory Commission (U.S. NRC) [9]. This report presents a valuable comparison of the nationally applied regulations with respect to deterministic as well as probabilistic assessment of NPP fire safety in FIRE member countries.

Two OECD/NEA experimental projects need to be mentioned as highly important for providing valuable results for assessing fires in nuclear installations:

- PRISME (*P*ropagation d’un *i*ncendie pour des *s*cenarios *m*ulti-locaux *é*lémentaires, French acronym for “*Fire Propagation in Elementary Multi-Room Scenarios*”), and
- HEAF (*H*igh *E*nergy *A*rcing *F*aults).

<u>PRISME</u>

The PRISME Project was initiated in 2006 and is meanwhile in its third phase (PRISME 3) being carried out by the French IRSN (*I*nstitut de *R*adioprotection et de *S*ûreté *N*ucléaire) in Cadarache. Major research areas addressed within the first PRISME Project phase were the propagation of heat and smoke from a fire compartment to adjacent rooms as well as the impact of heat and smoke on items important to safety and on the ventilation network. In total, five experimental campaigns with more than 35 fire tests in real scale geometries were carried out. Details can be found in [10].

The second phase of PRISME (PRISME 2) was carried out from 2011 to 2016 supported by nine member countries from Asia, Europe and North America. Major goal of the in total four experimental campaigns with more than twenty fire tests again in large-scale room geometries was to advance the knowledge regarding smoke and hot gas propagation through a horizontal opening between two superposed compartments, fire propagation on real fire sources such as cable trays or electrical cabinets, and fire suppression by fixed water-based fire extinguishing systems. In addition to the large-scale fire tests, support tests for the characterization of fire sources were carried out in open atmosphere in order to provide additional data for validation purposes. Details on the PRISME 2 Project can be found in [11].

The PRISME Project is actually in its third phase focussing on smoke stratification and propagation, fire propagation between electrical cabinets, and electrical cable tray fires in confined and ventilated conditions in order to provide answers to various issues of interest for nuclear fire safety. Details on the actually ongoing PRISME 3 Project are provided in a separate presentation by Suard et al. [12]. In conjunction with the FIRE Database Project, an international cable fire Benchmark Exercise based on a fire event stored in the Database and a corresponding PRISME 3 fire test is being performed. Details can be found in the SMiRT 25 Post-Conference Fire Seminar contribution titled "*Common Cable Fire Benchmark Activity of the OECD Nuclear Energy Agency Projects PRISME 3 and FIRE*" by Bascou et al. [13].

HEAF

Nuclear installations´ operating experience has clearly shown that switchgears, load centers and bus bars/ducts (with nominal voltage of typically 380 V and above) can be subject to a unique failure mode that causes extensive damage. In particular, these types of high energy electrical devices are subject to a failure mode known as high energy arcing fault (HEAF). This failure mode leads to an extremely rapid release of electrical energy in the form of heat, vaporized metals (e.g., copper and aluminum), plasma, and explosive mechanical force.

In general, HEAF in electrical equipment are initiated in one of three ways: poor physical connection between the switchgear and the holding rack, environmental conditions, or the introduction of a conductive foreign object (e.g., a metal wrench or screwdriver used during maintenance). A high energy fault scenario typically consists of two distinct phases, each with its own damage characteristics. The first phase is characterized by the short, rapid release of electrical energy which may result in a catastrophic failure of the electrical enclosure, ejection of hot projectiles from damaged electrical components or housing and/or fire(s). Such fires may only involve the electrical device itself or any external combustibles exposed. The second phase, i.e., the ensuing fire typically includes ignition of combustible material within the HEAF zone of influence.

As a result of indications in the FIRE Database Project on the significance of HEAF induced fire events [14] the OECD/NEA initiated an international task on the operating experience with HEAF events in 2009 for providing an in-depth investigation with the main objective to determine damage mechanisms, extent of areas affected, methods of protecting systems, structures and components (SSC) and possible calculation methods for modeling HEAF events as applicable to fire protection in NPPs [15]. As part of this effort, an experimental program was carried out from 2014 to 2016 in order to investigate HEAF fire phenomena to inform future deterministic and probabilistic methods. This experimental program covered 26 full-scale HEAF experiments. The program involved several challenges, such as suitable and efficient measurement science and techniques or the collection and analysis of the data recorded during the experiments. The experimental project revealed the following major insights (cf. [16]):

– All enclosures with components of 4.16 kV and above maintained the arc for a time periods of more than 2 s.

– Increased duration arcing events were more likely to create an ensuing fire, experiments ensuing fire were not observed for arc durations less than 2 s.
– The most severe electrical enclosure damage was observed as a result of a low voltage (480 V) HEAF in an enclosure with aluminum bus bars.
– HEAF events involving aluminum were seen to produce a conductive aluminum compound that coated the test facility causing short circuits and unintended current paths in electrical systems. Moreover, experiments with aluminum being consumed during the HEAF resulted in more severe physical damage to equipment than those involving only copper and steel at any voltage level.
– In 480 V enclosures, there was large variability in the ability to create a sustained arcing event. In some instances, the arc would immediately extinguish or not sustain for the full intended duration.

Based on the results of the HEAF experimental project, on behalf of the U.S. NRC's Office of Nuclear Regulatory Research a phenomena identification and ranking table (PIRT) exercise for NPP HEAF analysis applications was performed by means of a facilitated expert elicitation process in early 2017. The objective of the PIRT was to identify key phenomena for a series of specific HEAF scenarios associated with the intended application and to then rank the current state of knowledge relative to each identified phenomenon. Each scenario included a figure of merit, i.e., a specific goal to be achieved in analyzing the scenario using HEAF analysis or modeling tools. The panel identified phenomena that are of potential interest to an assessment based on the figures of merit. The phenomena identified were then ranked relative to their significance in predicting the figure of merit and further for the existing state of knowledge and the adequacy of existing modeling tools to predict that phenomenon. The PIRT panel covered three HEAF scenarios and identified a number of areas potentially needing further analysis and model development.

The PIRT exercise [17] was the basis for a second HEAF Project phase (HEAF 2), which officially started in October 2018 with the primary objective to develop experimental and scientific fire data related to the HEAF phenomenon applicable to NPPs, since the significant energy released during a HEAF event can act as an ignition source for secondary fires involving other components. The results from this second experimental series will report on the thermal and mechanical damage posed by HEAF events. The data collected from the experiments shall support updating and advancing methods for assessing and characterizing the risk of HEAF events. One session of the SMiRT Post-Conference Fire Seminar and a presentation on a "Proposal of an Evaluation Method for Prevention of High Energy Arcing Fault (HEAF) Induced Fires at Low and High Voltage Electrical Cabinets" by Shirai et al. from CRIEPI, Japan at the FSEP 2019 [18], [19] are specifically devoted to HEAF induced fire events.

REFERNCES

[1] International Atomic Energy Agency (IAEA): Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, Vienna, Austria, in publication, 2020.

[2] International Atomic Energy Agency (IAEA): Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants, Safety Guide No. NS-G-1.7, Vienna, Austria, 2004, http://www-pub.iaea.org/MTCD/publications/PDF/Pub1186_web.pdf.

[3] International Atomic Energy Agency (IAEA): Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants, Safety Guide NS-G-1.11, IAEA Safety Standards Series, Vienna, Austria, September 2004,
https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1191_web.pdf.

[4] International Atomic Energy Agency (IAEA): Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, STI/PUB/1430, ISBN 978-92-0-114509-3, Vienna, Austria, April 2010,
http://www-pub.iaea.org/MTCD/publications/PDF/Pub1430_web.pdf.

[5] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment (WGRISK): Proceedings of International Workshop on Fire PRA, NEA/CSNI/R(2015)12, Paris, France, August 2015,
https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-12.pdf.

[6] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA): CSNI Technical Opinion Paper No. 17, Fire Probabilistic Safety Assessments for Nuclear Power Plants: 2019 Update, OECD 2019, NEA No. 7417, Paris, France, 2019,
http://www.oecd-nea.org/nsd/pubs/2019/7417-csni-top17.pdf.

[7] Beilmann, M., et al.: Operating Experience with Fires in Nuclear Installations – The OECD/NEA FIRE Database, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, in preparation, December 2019,
https://www.grs.de/en/publication/grs-3963.

[8] McGrath, K., et al.: Nuclear Power Stations Fire Brigade Organization, Responsibilities and Challenges in Different OECD/NEA FIRE Member Countries, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019,
https://www.grs.de/en/publication/grs-3963.

[9] Melly, N. B., at al.: Survey of Member Countries' Nuclear Power Plant Fire Protection Regulations by the OECD Nuclear Energy Agency (NEA) Fire Incidents Records Exchange (FIRE) Database Project – Topical Report No. 2, NUREG/IA-0519, United States. Nuclear Regulatory Commission (NRC) Office of Nuclear Research, Washington, DC, USA, in publication, 2020.

[10] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): PRISME Project Application Report, NEA/CSNI/R(2012)14, Paris, France, July 2012, https://www.oecd-nea.org/nsd/docs/2012/csni-r2012-14.pdf.

[11] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Investigating Heat and Smoke Propagation Mechanisms in Multi-Compartment Fire Scenarios, Final Report of the PRISME Project, NEA/CSNI/R(2017)14, Paris, France, January 2018,
https://www.oecd-nea.org/nsd/docs/2017/csni-r2017-14.pdf.

[12] Suard, S., P. Zavaleta, and H. Prétrel: Overview of the OECD PRISME 3 Project, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

[13] Bascou, S., L. Audouin, and S. Suard: Common Cable Fire Benchmark Activity of the OECD Nuclear Energy Agency Projects PRISME 3 and FIRE, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

[14] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Project - Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R(2013)6, Paris, France, June 2013, http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf.

[15] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): A Review of Current Calculation Methods Used to Predict Damage from High Energy Arcing Fault (HEAF) Events, NEA/CSNI/R(2015)10, Paris, France, 2015, https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-10.pdf.

[16] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Experimental Results from the International High Energy Arcing Fault (HEAF) Research Program Testing Phase 2014 to 2016, NEA/CSNI/R(2017)7, Paris, France, 2017, http://www.oecd-nea.org/documents/2016/sin/csni-r2017-7.pdf.

[17] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research: An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arc Faults (HEAFs), NUREG-2218, Washington, DC, USA, 2018, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2218/.

[18] Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

[19] Shirai, K.: Proposal of an Evaluation Method for Prevention of High Energy Arcing Fault (HEAF) Induced Fires at Low and High Voltage Electrical Cabinets, in: Röwekamp, M., H.-P. Berg (Eds.): Proceedings of SMiRT 25, 16th International Seminar on Fire Safety in Nuclear Power Plants and Installations, October 28-30, 2019, Ottawa, ONT, Canada, GRS-A-3963, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Köln, Germany, December 2019, https://www.grs.de/en/publication/grs-3963.

SLIDE PRESENRATION

Fire Safety at Nuclear Sites: Challenges for the Future – An International Perspective

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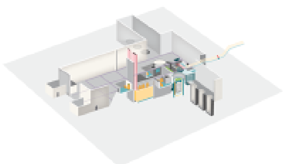
FSEP 2019 and SMiRT *16th International Post-Conference Seminar on "FIRE SAFETY IN NUCLEAR POWER PLANTS AND INSTALLATIONS"*

IASMiRT

Ottawa, ONT, Canada; October 27-30, 2019

Challenges for Nuclear Installations With Respect to Fire Safety

- New challenges for nuclear industry worldwide resulting from
  - Nuclear phase-out, at least in some European countries
  - Increase of new built power stations
    - Gen. III+ and VI light water reactors
    - Advanced and/or small modular reactors (AMR/SMR)

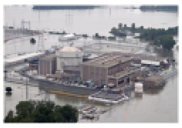


IMSR facility view, from www.terrestrialenergy.com

- Importance of fire safety over the whole lifetime of any nuclear installation
  - Fires may affect a whole nuclear site with
    - Multiple reactor units of different age and same and/or different design
    - Multiple radioactive sources (reactors, SFP, dry nuclear storage facilities, nuclear waste treatment facilities
  - Fires can occur during
    - Construction phase
    - All plant operational phases including the post-commercial safe shutdown
    - Decommissioning and deconstruction phase

IAEA Fire Safety Activities

- New **IAEA** Guide **DS494** "*Protection against Internal Hazards in the Design of Nuclear Power Plants*" merging former Guides (in preparation)
  - NS-G-1.7 "*Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants*" and
  - NS-G-1.11 "*Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants*"
  - ➢ Guidance how to address protection against fires for power reactors being designed considering whole site with all existing nuclear facilities
  - ➢ Better defining and considering combinations of hazards, in particular fires of all types
- Recently started update of IAEA SSG-3 "*Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants*"
  - Covering internal hazards (e.g. fires) and hazard combinations for risk aggregation from
    - All nuclear sources
    - Various types of risk (from plant internal events and hazards)
    - Different plant operational states (POS)



OECD Nuclear Energy Agency Tasks (1)

- Various OECD Nuclear Energy Agency (NEA) activities, mainly under the auspices of the CSNI (Committee on the Safety of Nuclear Installations)
- CSNI tasks related to fire risk at nuclear site by the Working Group on Risk Assessment (WGRISK)
  - International Workshop on Fire Probabilistic Risk Assessment (PRA) in 2014 https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-12.pdf
    - Fire PSA has achieved a reasonable level of maturity
    - On international level, Fire PSAs are performed within a consistent framework
    - Many Fire PSAs apply common methods, tools, and guidance
    - Common understanding of weaknesses, activities for reducing gaps are ongoing
    - Results of Fire PRA are being used to support to some extent risk-informed decision making
    - **Challenges:** multiple events and/or combination of fires with other events, multi-unit impacts, component based versus compartment based fire frequency estimations

OECD Nuclear Energy Agency Tasks (2)

- OECD/NEA CSNI WGRISK activities (contd.)
  - Update of the WGRISK Technical Opinion Paper (TOP) on Fire Probabilistic Safety Assessment for Nuclear Power Plants published in early 2019
    http://www.oecd-nea.org/nsd/pubs/2019/7417-csni-top17.pdf
    - International experts´ view on the state of Fire PSA as performed in support of NPP design and operation covering e.g. the following key messages:
      - Fire continues to be important risk contributor for nuclear installations
      - Fire PSA methods, models, tools, and data continue to improve
      - Fire PSA practice continues to mature
      - Knowledge of uncertainties and potential biases in Fire PSA results can and should be addressed
    - Technical opinions consider operating experience regarding fires in NPPs (e.g. from OECD/NEA FIRE Database)
    - Demonstrating that this tool is mature enough to support risk-informed decision making related to fire safety assessment

OECD/NEA Projects – FIRE Database (1)

FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) Database Project Overview

- This Project is amongst others also supporting Fire PSA
- In the late 1990s need for this Database emerged
- It became evident that the only international recording of fire events by the joint IAEA/NEA International Reporting System for Operating Experience (IRS) was not suitable for specific analysis and use in risk assessment
- Project is close to the end of its fifth phase (2016–2019)
- Actually 14 member countries: 10 from Europe, 2 from Asia, and 2 from North America
- Purpose: to provide a platform for member countries to collaborate and exchange fire data and thereby to enhance the knowledge of fire phenomena and in turn improve the quality of risk assessments that require fire related data and knowledge

photo from FIRE Database

OECD/NEA Projects – FIRE Database (2)

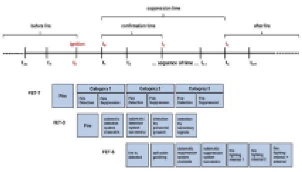
FIRE Database Project Scope and Objectives

- Objectives as defined according to the Terms and Conditions
  - Collect fire event experience (by international exchange) in an appropriate format in a quality-assured and consistent database
  - Collect and analyze fire events over the long term to better understand such events and their causes, and to encourage their prevention
  - Generate qualitative insights into the apparent and root causes of fire events in order to derive approaches or mechanisms for their prevention and to mitigate their consequences
  - Establish a mechanism for efficient operation feedback on fire event experience including the development of policies of prevention, such as indicators for risk-informed and performance-based inspections
  - Record characteristics of fire events in order to facilitate fire risk analysis, including quantification of fire frequencies

OECD/NEA Projects – FIRE Database (3)

FIRE Database Project Use and Applications (1)

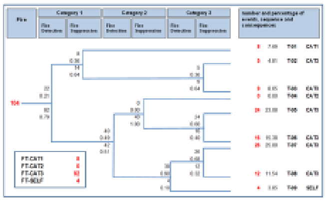
- Actually, Database covers **> 520 fire events** from NPPs of FIRE member countries
  - 501 of these being statistical evident events
  - Representing **operating experience** from **> 9415 years** of observation for different phases of the operational plant lifetime
- FIRE Database can be used, e.g.
  - for identifying all types of events and scenarios to be included in PSA models ensuring that all mechanisms are accounted for
  - as support to Fire PSA by real data from NPP operating experience, particularly to evaluate fire occurrence frequencies
  - for comparing member countries´ national fire event data with accumulated international data

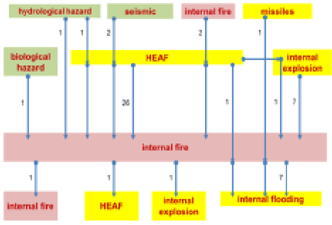


OECD/NEA Projects – FIRE Database (4)

FIRE Database Project Use and Applications (2)

- Focus of actual applications on:
  - Generic fire occurrence frequencies estimations for Fire PSA (compartment and component type based)
  - Significance of event combinations of fires and other events
    - Causally related (consequential, subsequent) events (including event chains)
    - Correlated events: (by a common cause initiator)
    - Unrelated events occurring independently from, but simultaneously to each other
  - HEAF (*h*igh energy *a*rcing *f*ault) induced fire events as non-negligible PSA contributor
  - Apparent and root causes of fire events recorded





OECD/NEA Projects – FIRE Database (5)

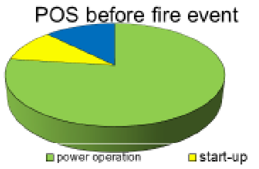
FIRE Database Project Use and Applications (3)

- FIRE Database can be used, e.g.
  - for identifying all types of events and scenarios to be included in PSA models ensuring that all mechanisms are accounted for
  - as support to Fire PSA by real data from NPP operating experience, particularly to evaluate fire occurrence frequencies
  - for comparing member countries´ national fire event data with accumulated international data

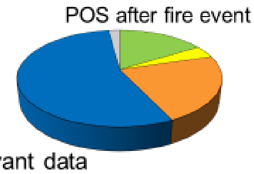
photos from FIRE Database

OECD/NEA Projects – FIRE Database (6)

Important FIRE Findings and Challenges (1)



- Well-defined observation times are available for
  - different POS (full power operation as well as low power and shutdown states including post-commercial operation safe shutdown phase)
  - construction and decommissioning phases



- Generic compartment (or room) specific fire frequencies for various buildings based on average numbers of rooms of different types
  - Statistically relevant for reactors from those countries reporting all events
  - Remaining challenge: partly insufficient amount of statistically relevant data
- Generic component specific ignition frequencies available for selected components
  - Comparison with national data from Fire PSA is in principle possible.
  - Main challenge: fire frequencies for cables, which are still difficult to derive

OECD/NEA Projects – FIRE Database (7)

Important FIRE Findings and Challenges (2)

- **~ 8 %** of the fire records represent **event combinations** of fires and other anticipated events
  - Fires induced by HEAF are an important contributor to the overall risk and need to be investigated for risk aggregation purposes
  - **HEAF induced fires**: > 10 % of the fire event records with non-negligible generic fire frequency of **7 E-03 /ry**
- ➢ Generic frequency data from FIRE Database can be applied at least as prior information in Bayesian approaches in case of insufficient date for nuclear site under investigation
- For modelling fire detection and suppression within the fire specific event trees need for more details on the time sequences
- Analysis of apparent causes of fire events in the Database demonstrated the possibility of identifying potential precursor events

OECD/NEA Projects – FIRE Database (8)

Important FIRE Findings and Challenges (3)

- Possibility to gain insights on potential backfitting measures and their effect on the fire specific CDF/FDF from the FIRE Database
- Need for more detailed investigations of the fire event root causes
  - Challenge: root cause codings are not yet complete and exhaustive, requiring improved analyses for reducing uncertainties and increasing level of confidence
- Recently started activity on on-site plant fire brigades, their organization, responsibilities and challenges, which are quite different in member states
  - Shall help members to find good practices for implementation in their countries



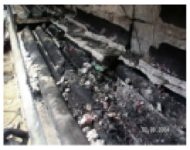


photo from FIRE Database

OECD/NEA Projects – PRISME Experimental Project (1)

PRISME (French for "*Fire Propagation in Elementary Multi-Room Scenarios*")

- PRISME was initiated in 2006 and is meanwhile in its third phase (PRISME 3)
- PRISME Project is conducted by the French IRSN in Cadarache
- Major research areas within first PRISME phase:
  - Propagation of heat and smoke from fire compartment to adjacent rooms
  - Impact of heat and smoke on items important to safety and on the ventilation network
  - 5 experimental campaigns with > 35 fire tests in real scale geometries
- Details see https://www.oecd-nea.org/nsd/docs/2012/csni-r2012-14.pdf

OECD/NEA Projects – PRISME Experimental Project (2)

PRISME (contd.)

- PRISME 2 (2011–2016) supported by 9 member countries from Asia, Europe and North America
- Major goals of the 4 PRISME 2 experimental campaigns with > 20 large-scale fire tests:
  - Advance knowledge regarding smoke and hot gas propagation through horizontal opening between 2 superposed compartments
  - Fire propagation on real fire sources (e.g., cable trays, electrical cabinets)
  - Fire suppression by fixed water-based fire extinguishing systems
  - In addition to large-scale fire tests, support tests for characterizing fire in open atmosphere for providing additional data for validation purposes

➢ Details see https://www.oecd-nea.org/nsd/docs/2017/csni-r2017-14.pdf

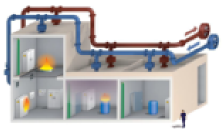
OECD/NEA Projects – PRISME Experimental Project (3)

PRISME (contd.)

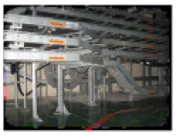
- Actually, third PRISME phase, PRISME 3 (2017–2022) is ongoing focussing on:
  - Smoke stratification and propagation
  - Fire propagation between electrical cabinets
  - Electrical cable tray fires in confined and ventilated conditions

  in order to provide answers to various issues of interest for nuclear fire safety

  ➢ Details see presentation by ***Sylvain Suard***



- In conjunction with FIRE Database Project: international cable fire benchmark exercise based on a fire event in the Database and a corresponding PRISME 3 fire test



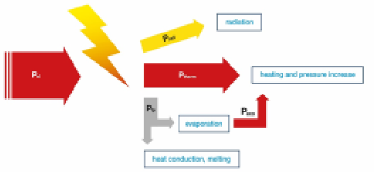
photos from PRISME Project

GRS

OECD/NEA Projects – HEAF Experimental Project (1)

HEAF (*H*igh *E*nergy *A*rcing *F*ault)

- Operating experience from nuclear installations has shown.
  - Switchgears, load centers and bus bars/ducts (nominal voltage typically $\geq$ 380 V) can be subject to a unique failure mode causing extensive damage
  - These types of high energy electrical devices are subject to HEAF failure mode HEAF leading to extremely rapid release of electrical energy (heat, vaporized metals, e.g. Cu and Al, plasma, and explosive mechanical force



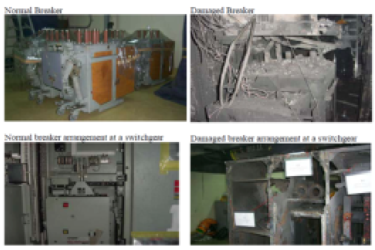
- Generally, HEAF are initiated in one of three ways:
  - Poor physical connection between the switchgear and the holding rack,
  - Environmental conditions, or
  - Introduction of a conductive foreign object, e.g. metal wrench or screwdriver used during maintenance

GRS

OECD/NEA Projects – HEAF Experimental Project (2)

HEAF (contd.)

- HEAF scenarios typically consist of two distinct phases, each with its own damage characteristics
- First phase: short, rapid release of electrical energy, which may result in a catastrophic failure of the electrical enclosure, ejection of hot projectiles from damaged electrical components or housing and/or fire(s)
  - HEAF fires may only involve the electrical device itself or any external combustibles exposed
- Second phase (ensuing fire): typically includes ignition of combustible material within the HEAF zone of influence



OECD/NEA Projects – HEAF Experimental Project (3)

HEAF (contd.)

- In 2009, OECD/NEA initiated an international task on the operating experience with HEAF events for providing an in-depth investigation
  - Main objective: to determine damage mechanisms, extent of areas affected, methods of protecting systems, structures and components (SSC) and possible calculation methods for modeling HEAF events as applicable to fire protection in NPPs
  - Experimental program (2014–2016) to investigate HEAF fire phenomena to inform future deterministic and probabilistic methods
    - 26 full-scale HEAF experiments
    - Challenges: suitable and efficient measurement science and techniques, collection and analysis of the data recorded during the experiments
    - ➢ Details see https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-10.pdf

OECD/NEA Projects – HEAF Experimental Project (4)

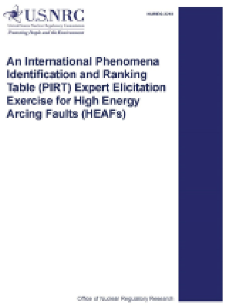
HEAF (contd.)

- Important insights
  - All enclosures with components of 4.16 kV and above maintained the arc for a time periods of > 2 s
  - Increased duration arcing events were more likely to create an ensuing fire, ensuing fires were not observed for arc durations < 2 s
  - Most severe electrical enclosure damage was observed resulting from low voltage (480 V) HEAF in an enclosure with Al bus bars
  - HEAF events involving Al produced a conductive aluminum compound that coating the test facility causing short circuits and unintended current paths in electrical systems
  - Experiments with Al being consumed during the HEAF resulted in more severe physical damage to equipment than those involving only Cu and steel
  - In 480 V enclosures, there was large variability in creating a sustained arcing event

OECD/NEA Projects – HEAF Experimental Project (5)

HEAF (contd.)

- PIRT (phenomena identification and *r*anking *t*able) by U.S. NRC based on HEAF results in early 2017
  - Objectives:
    - identify key phenomena for a series of specific HEAF scenarios associated with intended application
    - rank the current state of knowledge relative to each identified phenomenon
  - Each scenario included a figure of merit, i.e., a specific goal to be achieved in analyzing the scenario using HEAF analysis or modeling tools
  - Panel identified phenomena of potential interest to an assessment
  - PIRT panel covered 3 HEAF scenarios and identified a number of areas potentially needing further analysis and model development.
  - ➢ Details see https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2218/


U.S.NRC

An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arcing Faults (HEAFs)

OECD/NEA Projects – HEAF Experimental Project (5)

HEAF (contd.)

- PIRT was the basis for second HEAF Project phase (HEAF 2), which officially started in October 2018
  - Primary objective, since the significant energy released during a HEAF event can act as an ignition source for secondary fires involving other components:
    to develop experimental and scientific fire data related to the HEAF phenomenon applicable to NPPs
  - Results from HEAF 2 test series will provide insights on thermal and mechanical damage posed by HEAF events
  - Data collected from the experiments shall support updating and advancing methods for assessing and characterizing the risk
  - ➢ More details see presentations by ***Shirai*** and ***Melly***

Challenges Internationally Arising – Fire Brigade Issues (1)

- Fighting fires in nuclear facilities presents many unique challenges
  - Facilities are often located in remote locations
  - Plant and equipment are often located in complex and congested buildings
  - Additional management of radiological hazards requires a firefighting capability appropriately prepared and trained to deal with the complexities of each individual nuclear facility
  - Management of a successful firefighting capability requires appropriate organisational structure, staffing, specialist training and infrastructure as well as investment in specialist equipment, to enable all reasonably foreseeable fire event sequences can be managed and ultimately limit the effects to nuclear and life safety

Challenges Internationally Arising – Fire Brigade Issues (2)

- For investigating and comparing various capabilities and with the aim of identifying good practices, a short questionnaire has been initiated in FIRE member countries to obtain an overview of each national firefighting capability covering:
  - Organisations
  - Staffing and skills
  - Equipment and member nation regulatory expectations
- Analysis of questionnaire findings has shown that the firefighting capabilities across the member countries are not equivalent
- Questionnaire also identified a number of good practices across various responses
- ➢ More detailed investigations are intended in the near future providing support for
  - Comparing different strategies and concepts
  - Improving firefighting capabilities in different countries

Challenges Internationally Arising – Standards on Passive Fire protection Means (3)

- Questions on regulatory requirements and standards for fire barriers and their elements
  - Separation (fire compartments) and segregation, particularly of redundant items important to safety
  - Protection of specific equipment and of personnel (including fire fighters)
- Answers partly included in "*Survey of Member Countries' Nuclear Power Plant Fire Protection Regulations by the OECD Nuclear Energy Agency (NEA) Fire Incidents Records Exchange (FIRE) Database Project – Topical Report No. 2*" by U.S. NRC Office of Nuclear Research in preparation
- Detailed exchange of information on the differing approaches is foreseen on an international basis in the near future

➢ Comparison of benefits and challenges of different national approaches is possible and may result in improvements in some countries

Conclusions and Outlook (1)

- Observations and feedback from the operating experience help
  - Taking corrective actions and backfitting measures in operating nuclear installations
  - Improving regulatory requirements and guidelines for fire safety assessment
  - Application of best practices in the design of new reactors
- Exchange of fire related data and information, particularly on an international basis, provides an essential means for
  - Learning from experiences for own improvements and from good practices by others to either implement or adapt these corresponding to the own situation and boundary conditions
  - Developing enhanced and extended approaches for more risk-informed, performance-based decision making

Conclusions and Outlook (2)

- Still existing fire safety related knowledge gaps need further research as well as extension of data on an international level
  - Phenomena and scenarios with respect to electrical fires non-negligibly contributing to fire risk, not only for commercially operating reactors units but also in installations under decommissioning still need to be investigated in more detail
  - Data need to be further extended
    - More and more detailed fire event data for statistical use in PSA
    - Extension of data from construction and decommissioning phases for covering nuclear sites as a whole, even in phase-out countries or sites with new built reactors
    - Covering not only commercially operated rectors but also research, demonstration and pilot reactors
    - Collecting data also for fire protection features, particularly for operating experience feedback and reducing uncertainties within analyses

➢ **There is still a lot to do for further improving nuclear fire safety!**

Thank you for your attention!

for questions please contact
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IASMiRT

Overview of the OECD PRISME 3 Project

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ABSTRACT

Fire hazard analyses and probabilistic fire safety analyses have demonstrated that fires may cause significant damages in nuclear power plants (NPPs). Fire modelling is nowadays applied by licensees or technical safety organisations to assess fire consequences in NPPs. Thereby, one important aspect is the availability of verified and validated fire models for such fire scenarios.

Several members of the Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) expressed their interest in participating in a joint international research project on the topic of fire events to be carried out under the auspices of the NEA. The PRISME (French acronym for "Fire Propagation in Elementary Multi-Room Scenarios") Project was realized from 2006 to 2010, by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN, France) in its facilities specifically designed for large-scale fire tests in confined environments. In the continuity of the PRISME Project, PRISME 2 was launched in July 2011 ending in 2016. The main experimental results of these two projects have been summarized in OECD/NEA reports 00. In parallel to the experimental campaigns, PRISME partners evaluated the capabilities of various fire simulation codes for modelling fire scenarios based on the PRISME results. Both PRISME 1 and PRISME 2 Projects highlighted the strong interaction between the fire dynamics and the mechanical ventilation. Indeed, the analysis of the tests greatly contributed to enhance the knowledge of under-ventilated fires including realistic and complex fires. An improvement in the validation process of different fire models was also noticed during these two experimental programs.

From these experimental findings and modelling considerations, some grey zones have been highlighted and allowed to define the outlines of the PRISME 3 Project. Various recommendations have been provided for addressing some further phenomena not studied in the past Projects. These phenomena are smoke stratification and spread, fire propagation between electrical cabinets, and electrical cable tray fires in confined and ventilated conditions for new configurations. The ongoing PRISME 3 Project aims at addressing the above mentioned three phenomena and at providing answers to various issues of interest for nuclear safety. A total of eight countries have joined the PRISME 3 Project: Belgium (Bel V and Tractebel-ENGIE), Finland (Technical Research Centre VTT), France (IRSN as Operating Agent and Électricité de France – EDF), Germany (Gesellschaft für Anlagen- und Reaktorsicherheit – GRS), Japan (Nuclear Regulation Authority – NRA and Central Research Institute of Electric Power Industry – CRIEPI), Korea (Korea Institute for Nuclear Safety – KINS and Korea Atomic Energy Research Institute – KAERI), United Kingdom (Office for Nuclear Regulation – ONR) and the United States of America (United States Nuclear Regulatory Commission – U.S. NRC).

The objective of the first campaign, named S3 for *S*moke *S*tratification and *S*pread, is to study new configurations of interest for smoke propagation in a mechanically ventilated multi-room facility with simple fire sources. This choice is relevant for a complete validation of fire models on smoke propagation. The first topic of interest is to combine vertical

and horizontal smoke propagation coupled with a mechanical ventilation system. This experimental configuration allows highlighting multiple interaction mechanisms of propagation during the fire scenario. The second topic of interest concerns the issue of multiple fire sources, simulating for example a seismically induced fire incident and its consequences on smoke propagation. The fire scenario involves two fire sources ignited simultaneously and located in two adjacent rooms or in two rooms separated by another one. The distance between fires is a key parameter determining different combustion regimes with or without interaction. The third topic of interest is smoke propagation induced by an elevated fire source. This configuration leads to a complex situation for the fire dynamics, since it evolves in a hot and vitiated environment. For the S3 campaign, six fire tests have been defined in the multi-compartments facility of IRSN, named DIVA, which is composed of a mechanical ventilation system. For this campaign, three or four rooms with volumes of 120 m^3 and 170 m^3, have been implemented.

The second campaign, named ECFS for *E*lectrical *C*abinet *F*ire *S*pread, aims to better understand the fire spread from an open-door cabinet to other adjacent or opposite cabinets, connected via cable trays. Four tests in the IRSN DIVA facility have been defined including the two configurations of interest: the adjacent one and the opposite one. For each configuration two tests involving HFR (*h*alogenated *f*lame retardant) or HFFR (*h*alogen *f*ree *f*lame *r*etardant) insulated cables will be conducted. Two cable trays have been positioned above the cabinets and inside a false floor to diversify the potential paths of fire propagation. Furthermore, for the adjacent configuration, the fire propagation through the walls and an air gap potentially separating the cabinets is also considered. In addition to these confined fire tests, four additional tests have been specified in open atmosphere. These tests are needed for characterizing the fire source or the fire spread from one electrical cabinet to another one in a reference configuration. The comparison of the two configurations will highlight the effect of the confinement on the fire spreading.

The purpose of the third campaign, named CFP for *C*able *F*ire *P*ropagation, is twofold: in a first step, the effect of the compartment geometry will be studied by conducting three cable tray fires in a corridor. The fire dynamics on a long cable tray will then be compared to those obtained in previous PRISME Project for shorter cable tray configurations. In a final step, additional scenarios involving the effects of under-ventilated conditions and of the cable tray configuration on cable fires will be investigated. In addition, an assessment of cable fire models used within simple or complex fire numerical tools will be conducted for these specific configurations. Cable type as well as the air renewal rate of the compartment will be considered. Consequences of such fire scenarios on the facility will be investigated through time sequences of pressure, temperature and gas concentrations inside the facility and in the ventilation network. The campaign is composed of six tests in the DIVA facility and two tests under the SATURNE calorimeter of IRSN which is composed of an extraction hood in an open domain of 20,000 m^3.

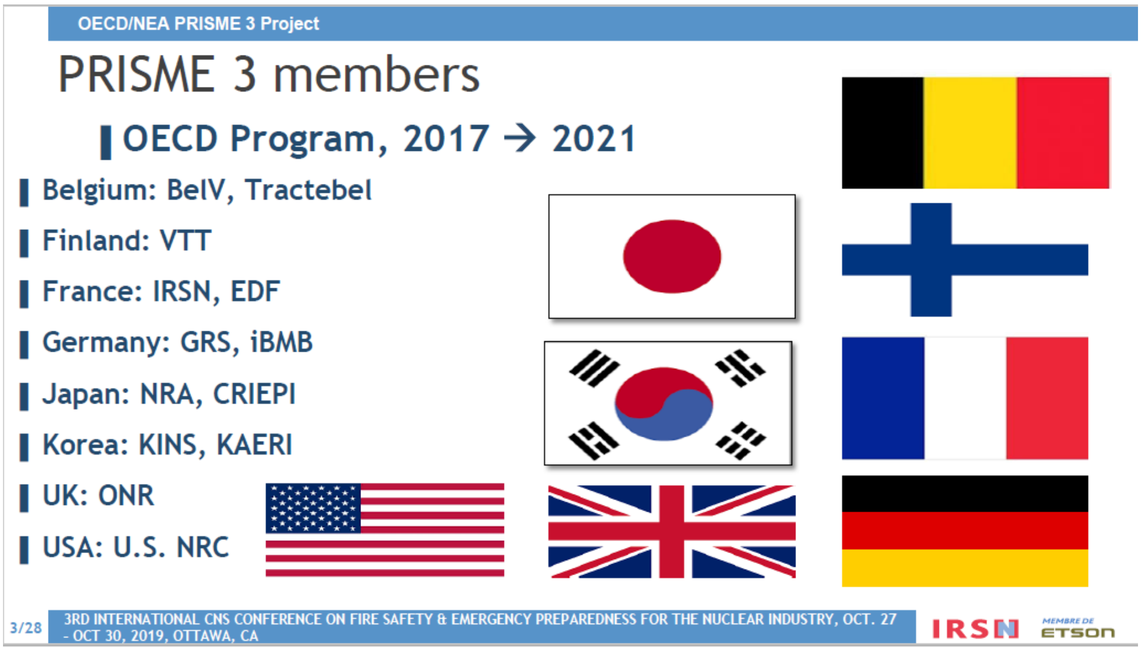
The main advances of PRISME 3 will make it possible, in a first step, to increase the predictive ability of models on smoke propagation problems for substantially complex situations. With regard to fires of electrical components, particularly electrical cable fires, the PRISME Projects will also provide a fairly complete database and can therefore be used for model improvements. All the contributions of the Project will allow the community to position itself on the scenarios of interest to study in future years.

REFERNCES

[1] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): PRISME Project Application Report, NEA/CSNI/R(2012)14, Paris, France, July 2012, https://www.oecd-nea.org/nsd/docs/2012/csni-r2012-14.pdf.

[2] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): Investigating Heat and Smoke Propagation Mechanisms in Multi-Compartment Fire Scenarios, Final Report of the PRISME Project, NEA/CSNI/R(2017)14, Paris, France, January 2018, https://www.oecd-nea.org/nsd/docs/2017/csni-r2017-14.pdf.

SLIDE PRESENTATION


OECD/NEA PRISME 3 Project
PRISME 3 members
OECD Program, 2017 → 2021
Belgium: BelV, Tractebel
Finland: VTT
France: IRSN, EDF
Germany: GRS, iBMB
Japan: NRA, CRIEPI
Korea: KINS, KAERI
UK: ONR
USA: U.S. NRC
3/28
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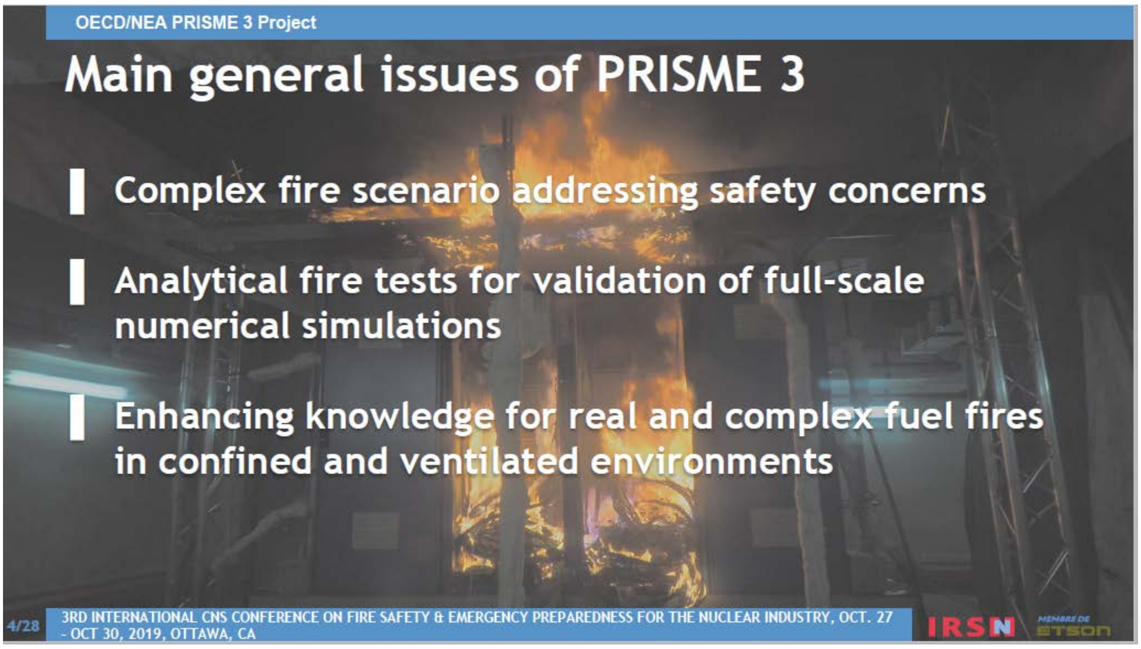
IRSN
MEMBRE DE ETSON


OECD/NEA PRISME 3 Project
Main general issues of PRISME 3
Complex fire scenario addressing safety concerns
Analytical fire tests for validation of full-scale numerical simulations
Enhancing knowledge for real and complex fuel fires in confined and ventilated environments
4/28
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OECD/NEA PRISME 3 Project

General Overview

3 campaigns – 3 different fuels



Smoke Stratification Spread (S3)
Smoke propagation in a facility

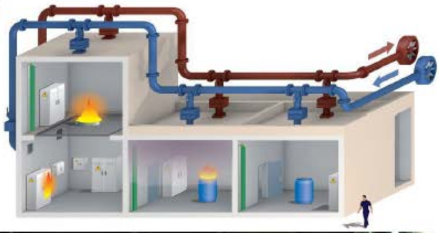
Electrical Cabinet Fire Spread (ECFS)
Fire spread from an EC to another one

Cable Fire Propagation (CFP)
Effect of renewal rate and geometry

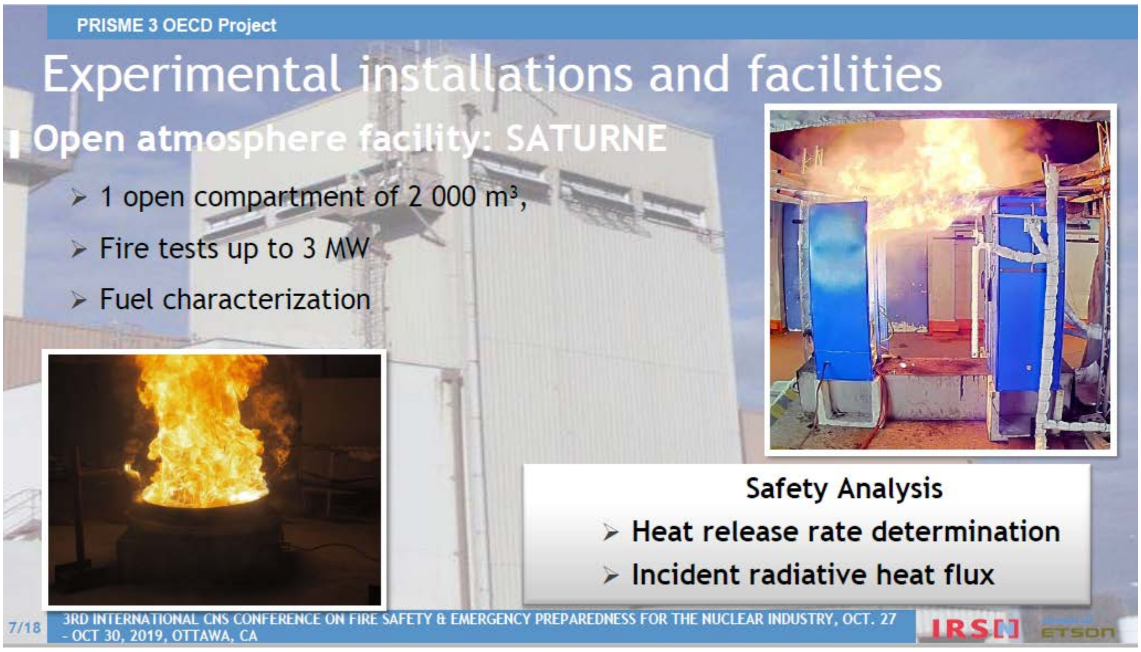
OECD/NEA PRISME 3 Project

Experimental installations and facilities

Mechanically ventilated facility: DIVA



- 3 rooms of $120 m^3$,
- 1 corridor of $150 m^3$,
- 1 room of $170 m^3$
- Overpressure of ≈ 500 hPa
- Under pressure of -100 hPa
- Reinforced concrete walls
- open doorways
- ventilation network


PRISME 3 OECD Project
Experimental installations and facilities
Open atmosphere facility: SATURNE
1 open compartment of 2 000 m³,
Fire tests up to 3 MW
Fuel characterization
Safety Analysis
Heat release rate determination
Incident radiative heat flux
7/18
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PRISME 3 OECD Project
Experimental Campaigns
8/18
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OECD/NEA PRISME 3 Project

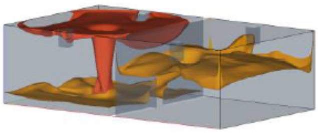
Smoke Stratification and Spread (S3)

Main safety Issue

- Smoke consequences: malfunction of electrical materials
- Validation of fire field and zone models for a full-scale compartment

➧ **Need for a good estimation of smoke flows in fire compartments**

➧ **Analytical tests to validate vent flow models in different complex configurations**



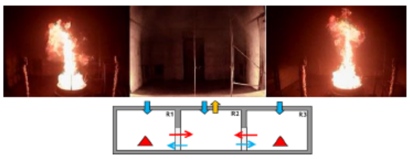
9/28 3RD INTERNATIONAL CNS CONFERENCE ON FIRE SAFETY & EMERGENCY PREPAREDNESS FOR THE NUCLEAR INDUSTRY, OCT. 27 - OCT 30, 2019, OTTAWA, CA IRSN MEMBRE DE ETSON

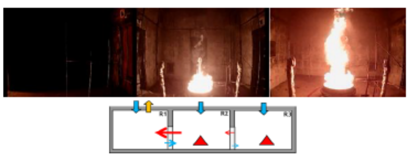
OECD/NEA PRISME 3 Project

Smoke Stratification and Spread (S3)

Objectives of the campaign

- Effect of the ventilation configuration on smoke propagation through 4 compartments:
  - Upper room with a horizontal opening
  - 3 lower rooms connected by open doorways
- Effect of smoke stratification and propagation for multiple fires
- Effect of smoke stratification for an elevated fire source





10/28 3RD INTERNATIONAL CNS CONFERENCE ON FIRE SAFETY & EMERGENCY PREPAREDNESS FOR THE NUCLEAR INDUSTRY, OCT. 27 - OCT 30, 2019, OTTAWA, CA IRSN MEMBRE DE ETSON

OECD/NEA PRISME 3 Project

Smoke Stratification and Spread (S3)

Test matrix: 6 large scale fire tests in the DIVA facility

- Reference test: 1
- Elevated fire source: 1
- Smoke stratification and propagation from multiple fires: 2
- Smoke propagation through openings: 2

OECD/NEA PRISME 3 Project

Electrical Cabinet Fire Spread

Safety Issues

- OECD FIRE Database: Electrical cabinets are a significant contributor

Need for an evaluation of the relevance of guidelines commonly used in nuclear safety (NUREG/CR-6850)

Background

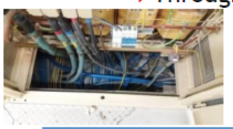
- Only few experimental tests performed by SNL, VTT and IRSN
  - Fire propagation through double wall
  - Tests in confined environment: only PRISME 2 experiments

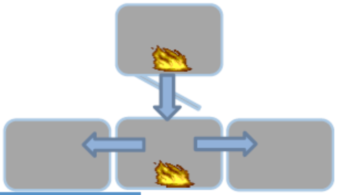
Investigation of specific configurations with fire spreading between electrical cabinets

Electrical Cabinet Fire Spread

Objectives of the ECFS campaign

- Study the fire spread from an electrical cabinet (EC) to another one
  - Adjacent cabinet configuration
  - Opposite cabinet configuration
- Different fire spread path
  - Through a double wall with an air gap
  - Through a raised floor
  - Through overhead cable tray





Electrical Cabinet Fire Spread

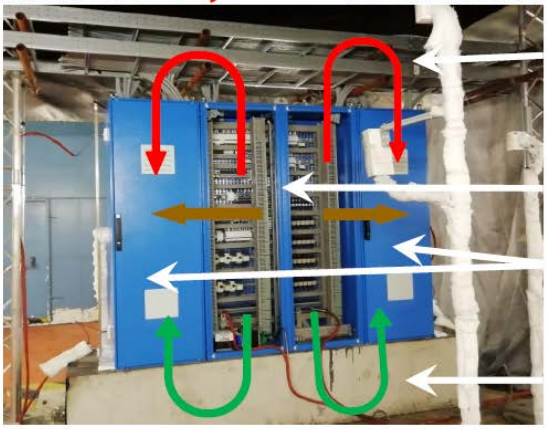
Test matrix: 8 large scale fire tests in open and confined environments

	Tests in open environment (SATURNE)	*Tests in confined and ventilated environment (DIVA)*
Adjacent configuration	*2*	*2*
Opposite configuration	*2*	*2*
Total	*4*	*4*

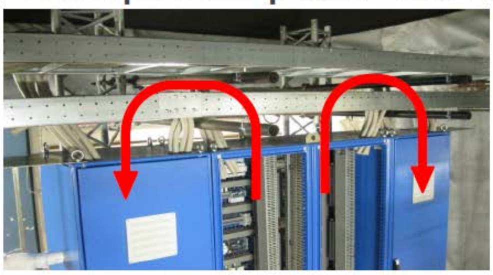
ECFS-S1 and ECFS-S2 tests (adjacent cabinets)

Three separated fire spread paths between the cabinets

- Via overhead cable trays
- Via the raised floor
- Via the double walls



Fire spread path via the overhead cable trays



Test ID	Cable	Loading
ECFS-S1	G (PVC)	20
ECFS-S2	A (HFFR)	27

CORE-1 (PRS2)

CFSS-2,CFS-3-4, CORE2-4 (PRS2)

- Two overhead cable trays
- Spacing (s) = 20 cm
- One layer of cables per tray arranged tightly

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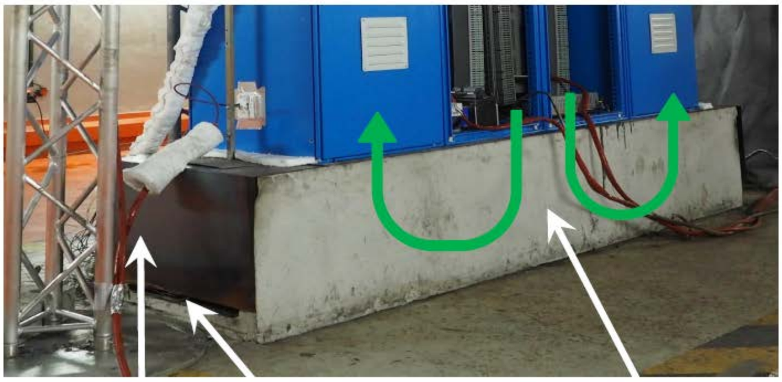
Fire spread path via the raised floor

- Two cable trays
- Spacing (s) = 20 cm
- One layer per tray of cables arranged tightly



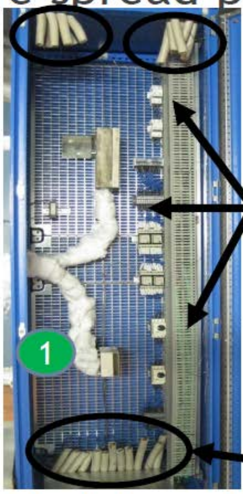
Corner-shape metallic plates

Natural ventilation (passage of 0.03 m^2)

Raised floor

OECD/NEA PRISME 3 Project

Fire spread path via the double walls

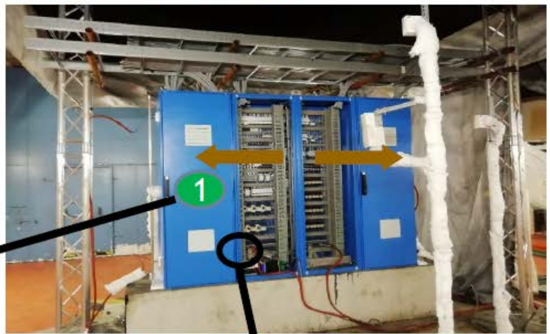


Cables from the overhead cable trays

Targets along the side walls

Cables from the raised floor

Left-hand adjacent cabinet



OECD/NEA PRISME 3 Project

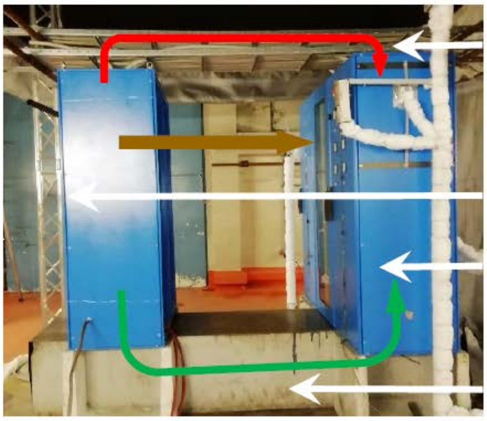
Fire scenario ECFS-S1

OECD/NEA PRISME 3 Project

ECFS-S3 and ECFS-S4 tests (opposite cabinets)

Three fire spread paths between the opposite cabinets

- Via overhead cable trays
- Via the raised floor
- Via the doors



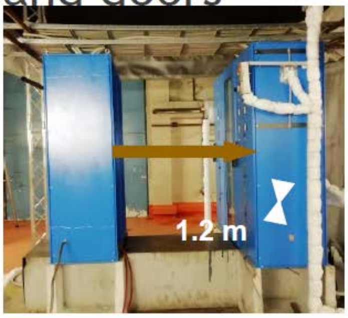
Two overhead cable trays

Open-doors cabinet

Three closed-door opposite cabinets

Raised floor (two cable trays)

Fire spread path via cables and doors



RH — Righ-hand cabinet with a monitored metallic door with monitored

C — Central cabinet with a glazed door

LH — Left-hand cabinet with a metallic door

Fire scenario ECFS-S3

Cable Fire Propagation

Safety Issues

- OECD FIRE Database: a large number of fire events are due to the burning of electrical cables
- Multiple cable trays configurations and types of cables

Background

- Several tests performed in open atmosphere (type of cable, cable tray configuration,…)
- Important results obtained in confined and ventilated conditions, during the PRISME and PRISME 2 projects

Additional data : effects of under ventilated environments and of the cable tray configuration

Cable Fire Propagation

Objectives of the CFP campaign

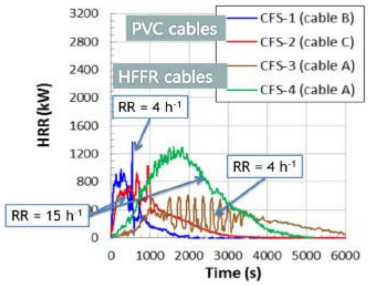
- Study of under-ventilated configuration
  - ➡Complementary data to link the oxygen concentration to the cable burning rate
- Study of cable tray in a service gallery with different location of ventilation inlet and outlet
  - ➡Influence of the geometry and scale effect
- Study of cable tray configuration on fire spread
  - ➡To complete the experimental database in open atmosphere in order to study the influence of the cable tray configuration

Cable Fire Propagation

Test Matrix: 8 large scale fire tests

- 3 tests in DIVA (under-ventilated issue)
- 3 tests in the DIVA corridor (service gallery issue)
- 2 tests in SATURNE (open environment, configuration issue)

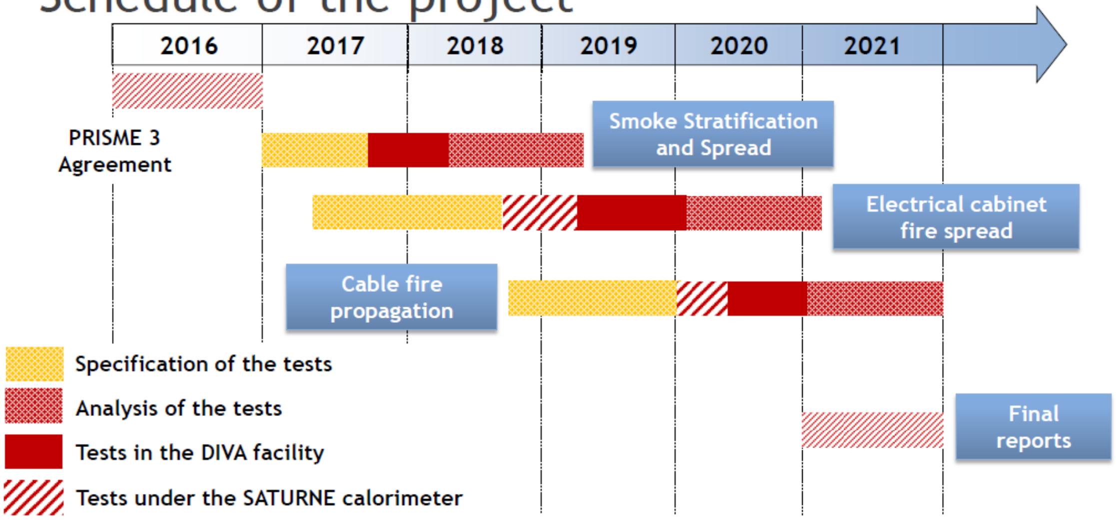
test configuration (exemple)



General PRISME 3 test matrix

Campaigns	SATURNE	DIVA	Fire source	DIVA Configuration
Smoke Stratification and Spread (S3)	0	6	Liquid pool	4 compartments
Electrical Cabinet Fire Spread (ECFS)	4	4	Electrical cabinets and cables	2/3 compartments
Cable Fire Propagation (CFP)	2	6	Cable trays	2 compartments, 1 corridor - 2/3 compartments

Schedule of the project



Thank you for your attention

Proposal of an Evaluation Method for Prevention of High Energy Arcing Fault (HEAF) Induced Fires at Low and High Voltage Electrical Cabinets

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ABSTRACT

In Japan, the 2011 off the Pacific coast of Tohoku Earthquake occurred on March 11th, 2011, caused high energy arcing faults (HEAF) in two of ten sectors of the non-emergency high voltage electrical cabinets and resulted in a seismic induced HEAF fire event in the Onagawa nuclear power plant (NPP). In response, in August 2017, the Nuclear Regulation Authority (NRA) of Japan partially amended the safety requirement concerning technical standards for practical power generation nuclear reactors and their attached facilities, especially for the power supply to mitigate the influence of a HEAF induced fire event.

In order to propose the design criteria for preventing the ensuing fire occurrence of non-arc proof electrical cabinets due to HEAF, CRIEPI conducted a series of three-phase HEAF tests with full-scale high (6.9 kV class) and low (420 – 480 V class) voltage electrical cabinets (power centers and motor control centers) in a three-phase short-circuit current with 5 – 45 kA (measured arcing energy over 40 MJ). Arcing energy, thermal heat flux, mechanical damage of the cabinet due to the impulsive pressure and thermal damage of the surrounding equipment due to the arc ejecta released from the test cabinet were obtained. According to the test results, a threshold value for each cabinet type to prevent HEAF induced fire occurrence was proposed.

Moreover, CRIEPI performed exposure tests of flame-retardant cables subjected to the arc ejecta released from the arc flash produced in the atmospheric condition with 5 – 20 MJ arcing energy. According to the test results, damage criteria for the flame-retardant cables under the extremely high heat flux in a short time duration were also proposed to estimate the zone of influence (ZOI) which defines the regions where HEAF fire conditions will cause thermal damage of the target cable.

INTRODUCTION

Large electrical discharges, referred to as HEAF fires, have occurred in NPPs switching components throughout the world [1]. In general, HEAF fires in electrical equipment are initiated in one of three ways: poor physical connection between the switchgear and the holding rack, environmental conditions, or the introduction of a conductive foreign object (e.g., a metal wrench or screwdriver used during maintenance) [2]. According to the Topical Report on HEAF fire events by the OECD Nuclear Energy Agency (NEA) FIRE (Fire Incidents Records Exchange) Database Project [3], more than 11 % of the fire events collected in this international fire events database represented fires in conjunction with HEAF. HEAF fire events were observed at only a limited number of different components with the potential of such type of event. The dominant contributions to HEAF events with consequential fires came from high voltage transformers (29 %) and from medium and low voltage transformers (8 %). As most of transformers are located outside of buildings or plant areas relevant to safety, safety significant consequences have not been observed in the events collected for high voltage transformers.

On the other hand, HEAF events in high or medium voltage electrical cabinets have a high potential for impairing nuclear safety. Particularly in Japan, the 2011 off the Pacific coast of Tohoku Earthquake which occurred on March 11, 2011 caused HEAF in two of ten sectors of the non-emergency high voltage metal-enclosed electrical cabinets and resulted in a seismically induced HEAF fire event in the Onagawa NPP. In response, in August 2017, the NRA partially amended the safety requirement concerning technical standards for practical power generation nuclear reactors and their attached facilities, especially for the power supply to consider the influence of HEAF fire events as shown in Table 1 [4]. Therefore, it is urgently necessary to establish the design criteria to prevent the HEAF fire event and enhance the experiment data of the HEAF event.

Table 1 Current status of the amended requirement in Japan (as of August 2017), from [4]

Technical standards for practical power generation nuclear reactors and their attached facilities	
Article 45	Safety power supply facility
Term 3	For safety power supply facilities (facilities for supplying electricity to safety facilities) including generators and emergency power facilities that are always used the power lines designated para. 1, and the power generation reactor facilities, the following measures shall be taken so that the supply of electric power to the equipment necessary for ensuring the safety of the facility will not be suspended.
Item 1	Measures necessary to prevent the spread of damage of the electrical cabinets due to HEAF
Item 2	In addition to what is listed in the preceding item, measures necessary to detect equipment damage, failure or other abnormality and prevent its expansion

In this paper, in order to propose the design criteria for preventing the ensuing fire occurrence of non-arc proof electrical cabinets due to HEAF, we executed several series of three-phase HEAF tests with high (6.9 kV) and low (420 – 480 V) voltage electrical cabinets. Moreover, discussions took place about the damage criteria for flame-retardant cables under the extremely high heat flux within a short time period based on the exposure tests.

HEAF TESTS

Test Facility

The Electric Power Engineering Research Laboratory of the CRIEPI, located about 65 km in the south from the center of Tokyo, was established in 1963 to make an important contribution to the progress of power transportation technology and conducted research and tests on the short circuit performance of power equipment and materials using its high-power short circuit testing facility.

The High Power Testing Laboratory shown in Figure 1 was established in 2001, and laboratory accreditation was granted by the Japan Accreditation Board for Conformity Assessment (JAB) in compliance with ISO/IEC17025 [5]. As a laboratory meets international standards, there is a variety of test activities that include publishing test reports and issuing certificates. In this test facility, short-time withstand current and peak withstand current tests for circuit breakers, disconnectors, earthing switches, load break switches, metal-enclosed switchgears and gas insulated switchgears can be conducted with the test capacity of currents up to 60 kA and durations up to 2 s.

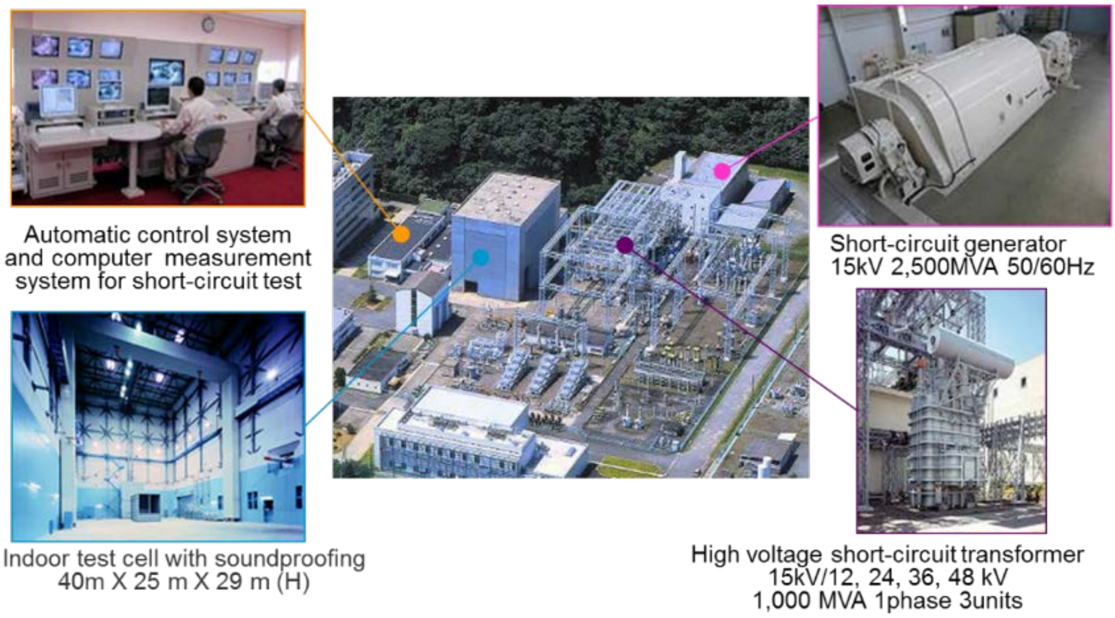


Figure 1 High Power Testing Laboratory (Yokosuka-shi, Kanagawa-ken, Japan)

Test Program

The HEAF tests using high and low voltage electrical cabinets were conducted at the high-power testing laboratory of the CRIEPI (cf. [6] to [8]). The HEAF test program consisted of seven phases as follows.

- Phase 1: Ten HEAF tests using non-seismic 7.2 kV electrical cabinets;
- Phase 2: Three HEAF tests using seismic 7.2 kV electrical cabinets;
- Phase 3: Four HEAF tests using seismic 480 V power centers (PCs);
- Phase 4: Five HEAF tests using non-seismic 480 V PCs;
- Phase 5: Three HEAF tests using non-seismic 420 V PCs;

– Phase 6: Seven HEAF tests using non-seismic 460 V motor control centers (MCCs);
– Phase 7: Three HEAF tests using non-seismic 6.9 kV electrical cabinets to energize a diesel generator (DG).

The equipment considered in our study consists of high or low voltage non-arc-proof electrical cabinets as shown in Table 2.

Table 2 HEAF test program for high and low voltage electrical cabinets

<table>
<tr><th colspan="5">High Voltage Tests</th></tr>
<tr><th>Test Items</th><th>Phase 1</th><th colspan="2">Phase 2</th><th>Phase 7</th></tr>
<tr><td>Cabinet type</td><td>Non-seismic 7.2 kV,
max. 4 cabinets
</td><td colspan="2">Seismic 7.2 kV,
2 cabinets
</td><td>Non-Seismic 6.9 kV,
2 cabinets
</td></tr>
<tr><td>Phase number</td><td colspan="4">Three-phase</td></tr>
<tr><td>Frequency</td><td colspan="4">50 Hz</td></tr>
<tr><td>Voltage</td><td>6.9 kV</td><td colspan="2">8.0 kV</td><td>6.9 kV</td></tr>
<tr><td>Current</td><td>18.9 kA</td><td colspan="2">40.0 kA</td><td>5.0 kA</td></tr>
<tr><td>Duration</td><td>0.10 - 2.00 s</td><td colspan="2">0.20 - 0.60 s</td><td>2.65 - 6.10 s</td></tr>
<tr><td>Arc Discharge Point</td><td colspan="3">cable room
circuit breaker room</td><td>circuit breaker room</td></tr>
<tr><th colspan="5">Low Voltage Tests</th></tr>
<tr><th>Test Items</th><th>Phase 3</th><th>Phase 4</th><th>Phase 5</th><th>Phase 6</th></tr>
<tr><td>Cabinet type</td><td>Seismic
480V, 2 PCs
</td><td>Non-seismic
480 V, 2 PCs
</td><td>Non-seismic
420 V, 2 PCs
</td><td>Non-seismic
460 V, 1 MCC
</td></tr>
<tr><td>Phase</td><td colspan="4">Three-phase</td></tr>
<tr><td>Frequency</td><td colspan="4">50 Hz</td></tr>
<tr><td>Voltage</td><td colspan="4">504 V</td></tr>
<tr><td>Current</td><td colspan="4">45.0 kA</td></tr>
<tr><td>Duration</td><td>0.2 - 1.5 s</td><td>1.3 - 1.4 s</td><td>1.3 - 1.4 s</td><td>0.1 - 0.5 s</td></tr>
<tr><td>Arc Discharge Point</td><td colspan="4">circuit breaker room</td></tr>
</table>

Test Matrix

The total number of HEAF tests was 35, varying the type of electrical cabinets, rating voltage, current and the arc discharge locations as shown in Table 3. The Test matrix was setup referring “JEM1425-2011, Appendix A – Internal Fault –” [9]. In these tests, seismic/non-seismic and non-arc-proof 6.9 – 7.2 kV electrical cabinets, 420 – 480 V PCs and 460 V MCCs were tested in the unloaded condition (no primary load attached to them). The arcing current was set to 5 – 45 kA considering a maximum three-phase short circuit current from the designated power system. Referring the standard of JEM1425-2011 [9], the arc duration was set to the range from 0.1 s to 1.5 s. Moreover, from the safety point of view, a longer arc duration of 2.0 s was also considered. In case of fire, fire extinguishing activities were performed by portable water spray fire extinguishers.

Table 3 HEAF test matrix and test results for high and low voltage electrical cabinets

Test Case	Arc Discharge Location*	Voltage [kV]	Current [kA]	Duration [s]	Arc Energy [MJ]	Fire
Phase 1: 10 tests with 8 non-seismic-proof, Non-arc-proof 7.2 kV electrical cabinets						
1-1	Upper C.R.at secondary bus	6.9	18.9	0.103	3.09	no
1-2	Upper C.R.at secondary bus			0.302	8.17	no
2-1	Upper C.R.at secondary bus			0.527	12.9	no
2-2	Upper C.B.R. at VCB terminal			0.526	10.4	no
3-1	Upper C.R.at secondary bus			1.23	24.7	no
3-2	Upper C.B.R. at VCB terminal			1.23	20.3	no
3-3	Lower C.B.R. at VCB terminal			1.23	27.6	yes**
3-4	Lower C.B.R. at VCB terminal			2.18	41.8	yes
4-1	Lower C.B.R. at VCB terminal			2.39	44.6	yes
4-2	Lower C.B.R. at VCB terminal			1.23	17.7	no
Phase 2: 3 tests with 2 seismic-proof, non-arc-proof 7.2 V electrical cabinets						
5-1	Upper C.R.at secondary bus	8.0	40.0	0.22	12.8	no
5-2	Lower C.B.R. at VCB terminal			0.21	8.68	no
5-3	Lower C.B.R. at- VCB terminal			0.63	25.3	no
Phase 7: 10 tests with 3 non-seismic-proof, non-arc-proof 6.9 kV electrical cabinets						
9-1	Upper C.B.R. at VCB terminal	6.9	5.0	2.69	14.7	no
9-2	Upper C.B.R. at VCB terminal			3.05	16.6	no
9-3	Lower C.B.R. at VCB terminal			6.27	32.3	yes
Phase 3: 4 tests with 3 seismic-proof, non-arc-proof 480 V PCs						
6-1	Upper C.B.R. at ACB terminal	0.504	45.0	0.20	2.49	no
6-2	Middle C.B.R. at ACB terminal			0.51	6.34	no

<table>
<tr><th>Test Case</th><th colspan="2">Arc Discharge Location*</th><th>Voltage [kV]</th><th>Current [kA]</th><th>Duration [s]</th><th>Arc Energy [MJ]</th><th>Fire</th></tr>
<tr><td>6-3</td><td colspan="2">Lower C.B.R. at ACB terminal</td><td rowspan="2"></td><td rowspan="2"></td><td>1.53</td><td>19.8</td><td>yes</td></tr>
<tr><td>6-4</td><td colspan="2">Middle C.B.R. at ACB terminal</td><td>0.18</td><td>2.91</td><td>no</td></tr>
<tr><td colspan="8">Phase 4: 5 tests with 4 non-seismic-proof, non-arc-proof 480 V PCs</td></tr>
<tr><td>7-1</td><td colspan="2">Upper C.B.R. at ACB terminal</td><td rowspan="5">0.504</td><td rowspan="5">45.0</td><td>0.43</td><td>5.76</td><td>no</td></tr>
<tr><td>7-2</td><td colspan="2">Upper C.B.R. at ACB terminal</td><td>0.05</td><td>0.88</td><td>no</td></tr>
<tr><td>7-3</td><td colspan="2">Middle C.B.R. at ACB terminal</td><td>0.02</td><td>0.34</td><td>no</td></tr>
<tr><td>7-4</td><td colspan="2">Lower C.B.R. at ACB terminal</td><td>1.32</td><td>18.5</td><td>no</td></tr>
<tr><td>7-5</td><td colspan="2">Upper C.B.R. at ACB terminal</td><td>1.43</td><td>18.9</td><td>no</td></tr>
<tr><td colspan="8">Phase 5: 3 tests with 4 non-seismic-proof, non-arc-proof 420 V PCs</td></tr>
<tr><td>8-1</td><td colspan="2">Middle C.B.R. at ACB terminal</td><td rowspan="3">0.504</td><td rowspan="3">45.0</td><td>1.32</td><td>17.4</td><td>no</td></tr>
<tr><td>8-2</td><td colspan="2">Middle C.B.R. at ACB terminal</td><td>1.32</td><td>17.3</td><td>no</td></tr>
<tr><td>8-3</td><td colspan="2">Middle C.B.R. at ACB terminal</td><td>1.43</td><td>18.7</td><td>no</td></tr>
<tr><td colspan="8">Phase 6: 7 tests with 5e non-seismic-proof, non-arc-proof 460 V MCCs</td></tr>
<tr><td>10-1</td><td colspan="2">Upper MCCB unit</td><td rowspan="7">0.504</td><td rowspan="7">45.0</td><td>0.06</td><td>0.90</td><td>no</td></tr>
<tr><td>10-2</td><td>Lower</td><td rowspan="6">Contact point between MCCB unit and bus bar</td><td>0.52</td><td>7.56</td><td>yes</td></tr>
<tr><td>10-3</td><td>Middle</td><td>0.32</td><td>4.49</td><td>no</td></tr>
<tr><td>11-1</td><td>Lower</td><td>0.07</td><td>1.02</td><td>no</td></tr>
<tr><td>11-2</td><td>Lower</td><td>0.15</td><td>2.24</td><td>no</td></tr>
<tr><td>11-3</td><td>Lower</td><td>0.05</td><td>0.80</td><td>no</td></tr>
<tr><td>11-4</td><td>Middle</td><td>0.28</td><td>3.94</td><td>no</td></tr>
<tr><td colspan="8">* C.R.: cable room, C.B.R.: circuit breaker room, MCCB: molded case circuit breaker
** Fire occurred, but self-extinguished</td></tr>
</table>

The test cabinets included the associated electrical equipment to provide a representative configuration. Moreover, secondary combustibles such as cables were also included in the test setup to verify the occurrence of the fire.

Test Measurements

During the tests, rated current and voltage, arc duration, inner pressure inside the cabinet by pressure sensors, surface temperature of the cabinets by thermography and temperature within ZOI were measured. The total arc energy is estimated by integrating the product of measured arc current and arc voltage over time. Additional instrumentation included the high-speed video cameras. In case of series5, the instrumented cable trays were placed in the vicinity of the test cabinet to investigate the thermal damage or the potential fire resulting from the arc event.

Moreover, in case of Phases 4, 5, 6 and 7, the hood calorimeter with the scrubber to mitigate smoke effects on the surrounding environment was placed above the test cabinets to measure the heat release rate and judge the occurrence of the ensuing fire resulting from the arc event as shown in Figure 2. Additionally, in case of test series 8, a smoke detector was installed to monitor the smoke concentration at a distance of 1.5 m from the top of the test cabinet during the tests.

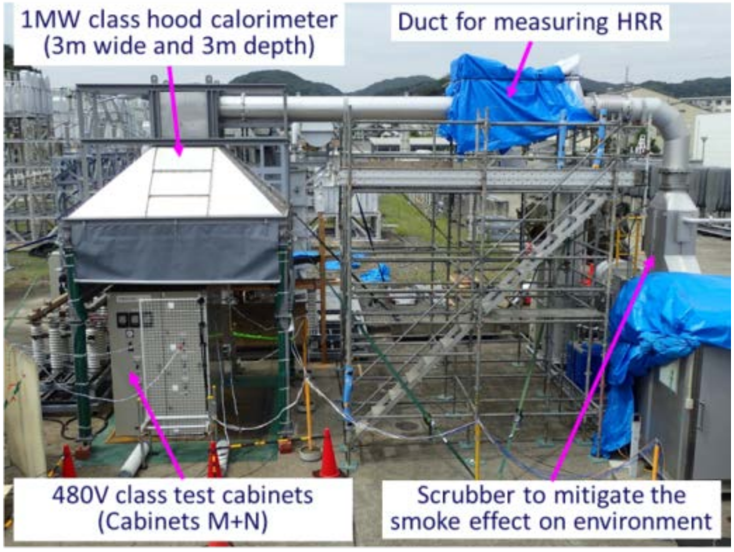


Figure 2 HEAF test equipment layouts in Phase 4

A secondary bus in the cable room and VCB (*v*acuum *c*ircuit *b*reaker) / ACB (*a*ir *c*ircuit *b*reaker) terminal in the circuit breaker room were selected as arc discharge points, and the arc was initiated by means of a copper wire 0.5 mm in diameter as shown in Figure 3. The current flows through the copper wire, and the wire explodes igniting an arc. The cabinet doors were closed to represent events that may occur under normal operation.

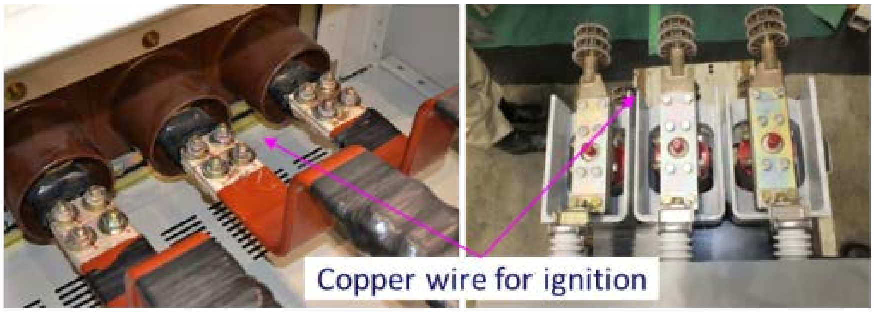
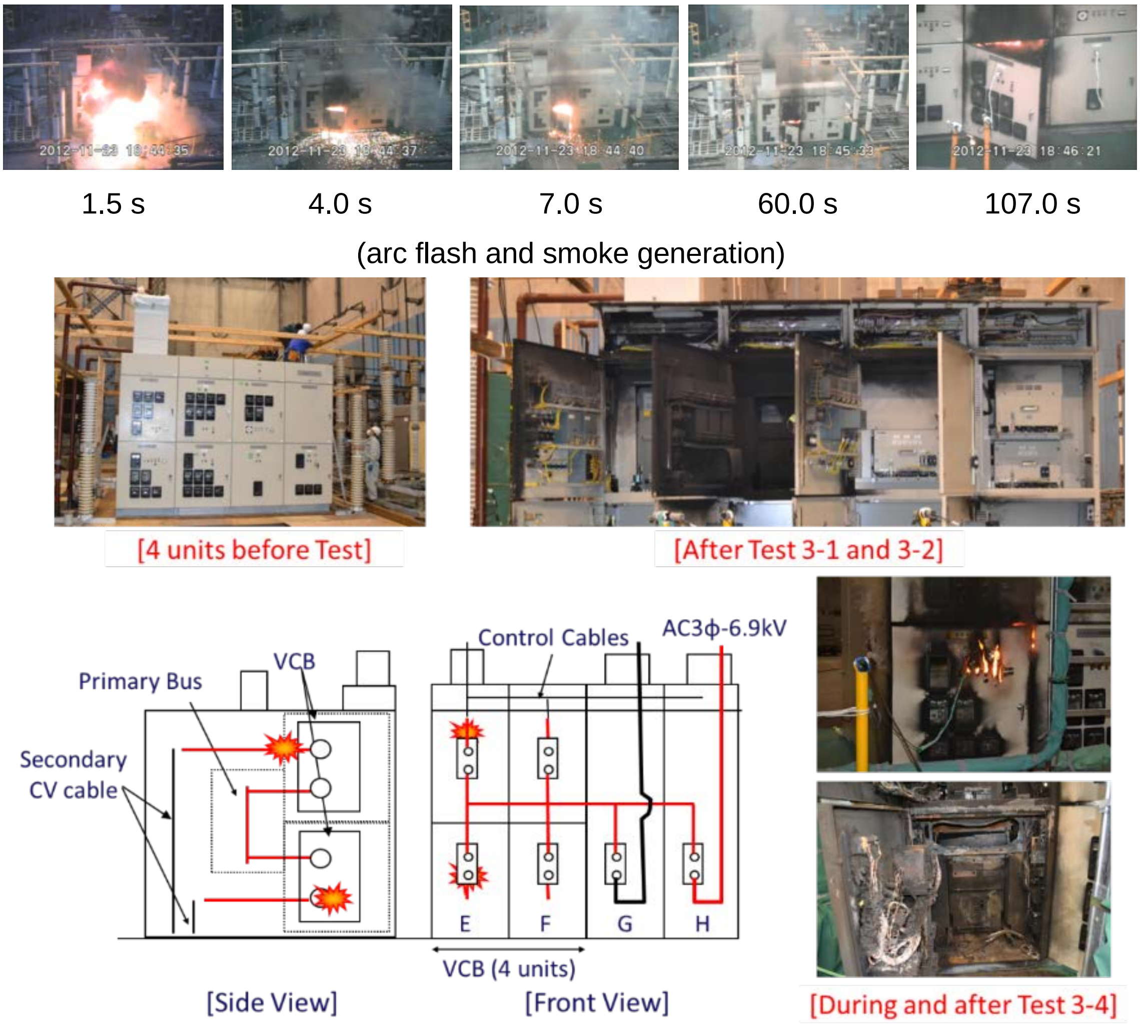


Figure 3 Arc discharge point at the cable room and the VCB terminal in Phase 1

HEAF TEST RESULTS

Arc Flash and Damage of the Cabinets

Figure 4 shows the arc flash and smoke generation, the damage of the test cabinet and temperature profile observed in the test 3-4 carried out in Phase 1. In the test 3-4, as the arc was ignited at the VCB terminal located in the lower circuit breaker room with an arc duration of 2.0 s (arc energy of more than 40 MJ), the door opening due to the high pressure and the melting of the front panel of the VCB were observed. A rapid increase of the cabinet surface temperature was observed after 180 s, and as the maximum temperature exceeded 450 °C after 450 s, this ensuing fire was actively extinguished by portable water spray extinguishers according to the safety procedure of the test facilities.



damage of the test cabinet

Figure 4 Test results from test 3-4 (arc energy 41.8 MJ)

Figure 5 shows the arc flash and smoke generation as well as the damage of the surrounding components observed in the test 5-1 also carried out in Phase 1. As the arc was ignited in the upper cable room with a duration of 0.2 s (arc energy of approximately 13 MJ), the roof and rear panels came off and the cable trays were deformed remarkably due to the impact of the fallen down roof panel. However, there was no remarkable thermal damage of the cables on the tray even if being exposed to the highly hot arc ejecta gas.

0.02 s 0.04 s 0.10 s 0.20 s 10.00 s

arc flash and smoke generation)

damage of the surrounding components

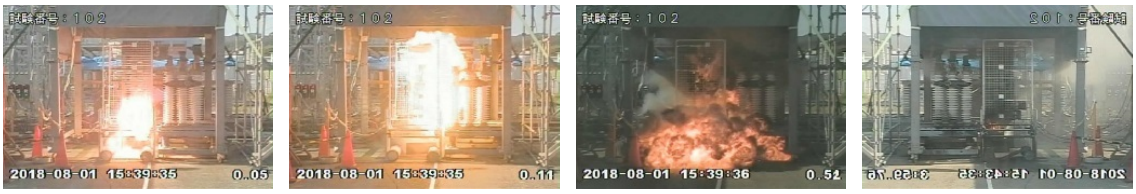
Figure 5 Test results from test 5-1 (type: high voltage, arc energy 12.8 MJ)

From Phase 3 up to Phase 5, the arc energy was measured using 420 V or 480 V PCs (three-phase, three-wire system) under a condition with short circuit current of around 45 kA and volume lower than high-voltage electrical cabinet, an ensuing fire was identified as soon as the arcing energy exceeded 19 MJ. Figure 6 shows the arc flash and smoke generation, and the ensuing fire observed in the test 6-3 executed in Phase 3. The ensuing fire was induced in the lower circuit breaker room and immediately suppressed by portable water spray according to the safety procedure of the test facilities.

0.2 s 0.6 s 1.6 s HEAF fire

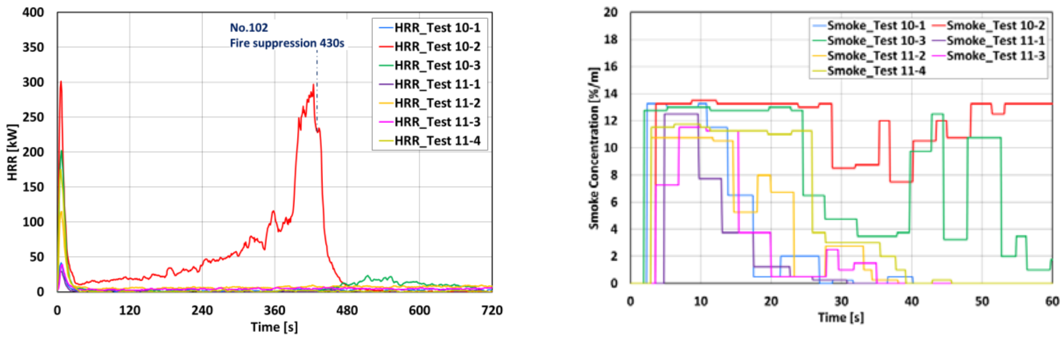
Figure 6 Test results from test 6-3 (type: PC, arc energy 19.8 MJ)

In the Phase 6, the arc energy was measured using 460 V MCCs under a condition with short circuit current around 45 kA and durations from 0.2 to 0.5 s. Figure 7 shows the arc flash and smoke generation as well as the ensuing fire observed in the test 10-2 in the lower cable room with a significant influence on the MCCB units within the cabinet. Figure 8 shows the heat release rate (HRR) and smoke concentration measured in the Phase 6. The smoke detector immediately provided an alarm signal in all tests within Phase 6. Particularly in the test 10-2, as the signal from the smoke detector continued, it seems that burning of some of the cables inside the cabinet continued and induced the smoke.



0.05 s 0.11 s 0.52 s HEAF fire

Figure 7 Test results from test 10-2 (type: MCC, arc energy 7.56 MJ)



HRR smoke concentration

Figure 8 Heat release rates and smoke concentrations measured in Phase 6 using MCCs

The thermal transfer coefficient (TTC) can be defined as a value obtained by dividing the integral value of the HRR by the arc energy, which is well-known as k_p value [10] representing that fraction of the energy going into increasing the gas pressure. According to the test results, the measured mean TTC value was 0.52 and seems to be in good agreement with values from the literature (e.g., k_p = 0.53 for a copper electrode [10]).

Inner Arc Pressure in the Cabinet

Neglecting all transient and hydrodynamic effects, the discharge of an arc in a cabinet can be treated as an ideal gas within a constant volume system. If an amount of energy *ΔQ* is injected into the volume, the change in pressure *Δp* is expressed as follows (cf. [11]):

$$\Delta p = (\gamma - 1) \cdot k_p \Delta Q / V \quad (1)$$

with volume *V*, and the adiabatic coefficient γ = 1.4, which is often used in high voltage electrical cabinet application [11].

For example, for a cabinet of 1 m^3 volume and an arc energy of 1 MJ, an inner pressure of 212 kPa can be obtained. Figure 9 shows the relationship between arc power derived from half-cycle of the arc discharge and measured inner pressure. The dotted line shows the estimated arc pressure per unit volume calculated by equation (1). As a result, it is found that the measured values of the inner pressure are below the dotted line regardless of the shape or type of the electrical cabinet. Therefore, it seems that the electric cabinet structure will easily release the arc pressure just after the first arrival of the shock wave due to the arc discharge. Nevertheless, it should be noted that in some cases, as the several pieces of the broken bolts were scattered away from cabinet, they should be potentially considered as a projectile which might induce local damage to the surrounding equipment.

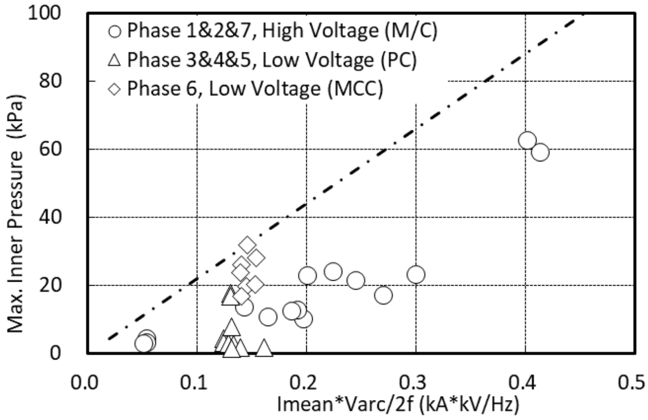


Figure 9 Relationship between arc power during half-cycle and measured inner pressure

Arc Energy

The arc energy E_{arc} for three-phase inner arc occurrence can be calculated by equation (2).

$$\mathrm{E}_{3\emptyset} = \int V_R \cdot I_R dt + \int V_S \cdot I_S dt + \int V_T \cdot I_T dt \tag{2}$$

where: $\mathrm{E}_{3\emptyset}$ the three-phase arc energy, V_R the arc voltage for the R-phase, V_S the arc voltage for the S-phase, V_T the arc voltage for the T-phase, I_R the arc current for the R-phase, I_S the arc current for the S-phase, and I_T the arc current for the T-phase.

Here, it is assumed that the arc current of each phase is almost identical to equation (3),

$$I = I_R = I_S = I_T \tag{3}$$

As the arc voltage V_{arc} (sum of , V_R, V_S, and V_T) may be constant, namely not depending on the bias of the arc current, $\mathrm{E}_{3\emptyset}$ can be derived by equation (4).

$$\mathrm{E}_{3\emptyset} = (V_R + V_S + V_T) \int I\, dt = V_{arc} \int I\, dt \tag{4}$$

As the integrated value of the arc current $\int I\, dt$ represents the product of the average current value and the arc duration, $\mathrm{E}_{3\emptyset}$ can be rewritten by equation (5) as follows:

$$\mathrm{E}_{3\emptyset} = V_{arc} \times I_{average} \times t_{arc} \tag{5}$$

where: $I_{average}$ the mean value of the current I, and t_{arc}. the arc duration.

If the current waveform is assumed to be a sinusoidal wave, $I_{average}$ can be derived as follows:

$$I_{average} = {}^{2}/_{\pi} \cdot I_p = 0.637 \cdot I_p \tag{6}$$

$$I_{rms} = I_p/\sqrt{2} = 0.707 \cdot I_p \tag{7}$$

$$I_{average}/I_{rms} = 0.637/0.707 = 0.9 \tag{8}$$

$$I_{average} = 0.9 \cdot I_{rms} \tag{9}$$

where: I_p the symmetrical component peak value, and I_{rms} the current effective value.

Therefore, $\mathrm{E}_{3\emptyset}$ can be rewritten using I_{rms} by equation (9):

$$\mathrm{E}_{3\emptyset} = V_{arc} \times (I_{rms} \times 0.9) \times \tag{10}$$

Figure 10 shows the measured arc voltage for high and low voltage electrical cabinets. It is found that the measured arc voltage seems to be constant regardless of the arc duration, and their mean values for high voltage electrical cabinets, low voltage PCs and MCCs were 1.344 kV, 467 V and 675 V, respectively.

Accordingly, applying the measured arc voltages values to equation (9), the estimation formula for the arc energy can be proposed as follows.

$$\mathrm{E}_{3\emptyset,HV\ switchgears} = 1.344\ kV \times (I_{rms} \times 0.9) \times t_{arc} \tag{11}$$

$$\mathrm{E}_{3\emptyset,LV\ PCs} = 0.467\ kV \times (I_{rms} \times 0.9) \times t_{arc} \tag{12}$$

$$\mathrm{E}_{3\emptyset,LV\ MCCs} = 0.675\ kV \times (I_{rms} \times 0.9) \times t_{arc} \tag{13}$$

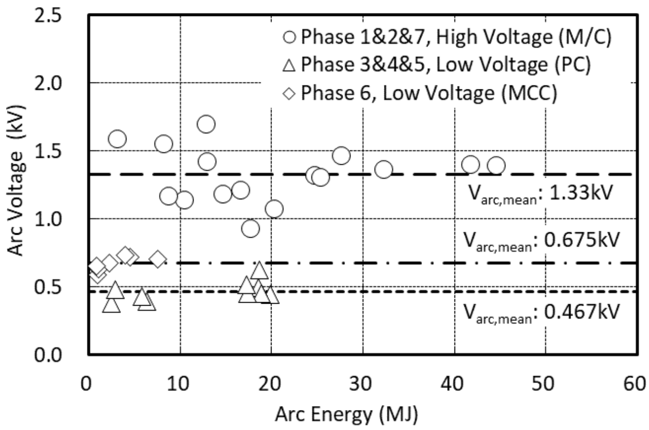


Figure 10 Measured arc voltage for high/low voltage electrical cabinets

Figure 11 shows the relationship between E_{arc} and t_{arc} for high and low voltage electrical cabinets. It seems that as the scattering of the experimental data seems to be not so high and linear approximation for each Phase can be applied. As a result, the upper limit values to prevent the ensuing fire after HEAF event for high and low voltage electrical cabinets were summarized as shown in Table 4.

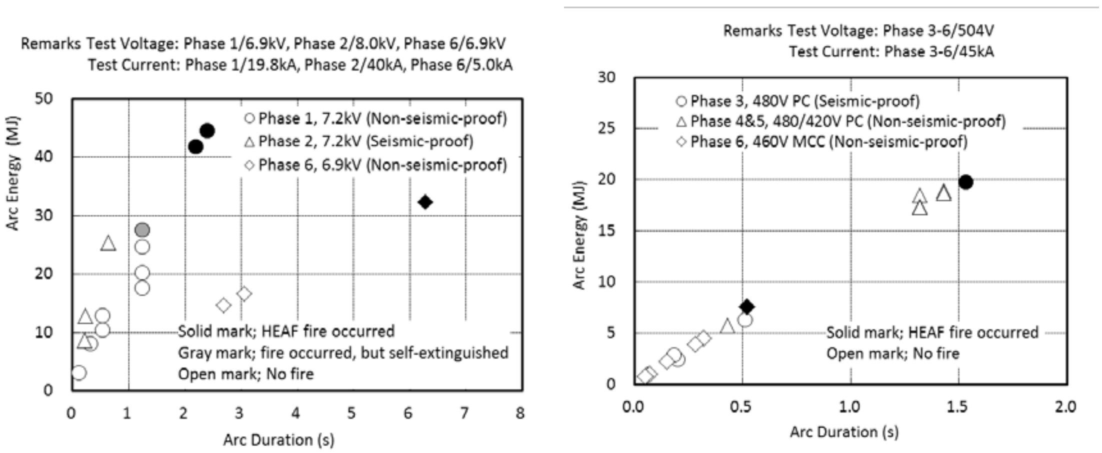


Figure 11 Relationship between E_{arc} and t_{arc} for high and low voltage electrical cabinets

Table 4 Upper limit values to prevent ensuing fire after a HEAF event

Cabinet Type		Upper limit value
High voltage electrical cabinets	High Current	25.3 MJ
	Low Current	16.6 MJ
Low voltage electrical cabinets	Power Center	18.9 MJ
	Motor Control Center	4.49 MJ

CABLE EXPOSURE TEST SUBJECTED TO ARC EJECTA GAS

Experimental Setup and Conditions

Figure 12 shows the test apparatus for the cable exposure test. The experimental conditions for the cable exposure test are summarized in Table 5. A short circuit generator (50 Hz, 15 kV, and 2,500 MVA) of CRIEPI as shown in Figure 1 was used as power source. The arc was ignited between two electrodes in an open condition. The copper electrodes were 20 mm in diameter with a flat tip. The arc was ignited by fusing of a fine copper wire of 0.5 mm diameter, which connected two electrodes. Electrical operating conditions including arc voltage and arc current, and incident energies were measured during the experiments. Incident energies were measured with slug calorimeters installed at every sixty degrees location around the electrodes. The experiments were documented with normal and high-speed videography, infrared imaging, and photography. The arc current was 35 kA (single phase, 50 Hz) and the arcing duration ranged from 0.05 s to 1.0 s. The total arc energy (from 1 MJ to 20 MJ) was estimated by integrating the product of measured arc current and arc voltage over time. As a target, the flame-retardant cable specimen of 150 mm length was used, and the isolation distance was set to 0.3, 0.6, 0.9, and 1.5 m. After the exposure test, each cable specimen was subjected to a withstand voltage test in order to detect the thermal damage to the dielectric strength of the cable specimen.

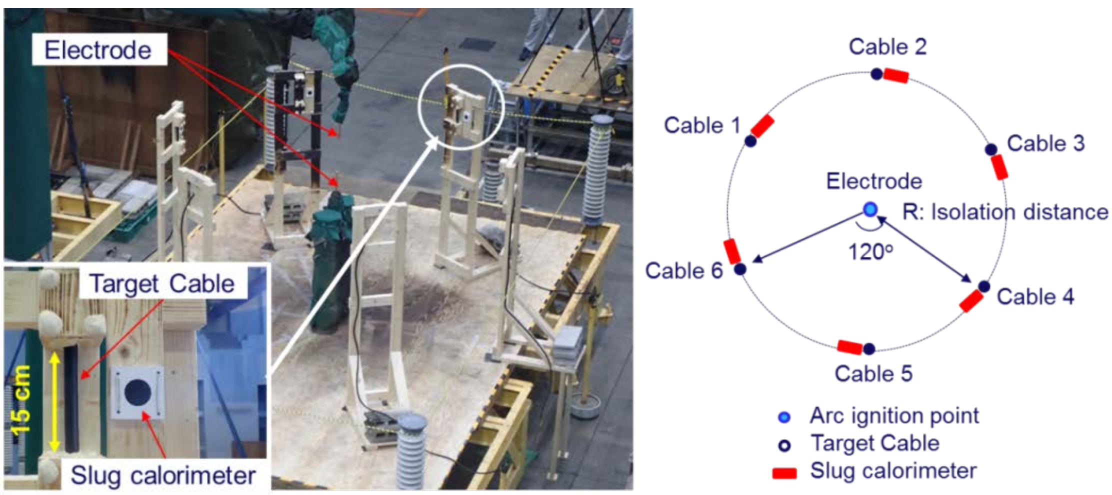


Figure 12 Test apparatus for the cable exposure test

Table 5 Experimental conditions for cable exposure test

Test Items	Conditions
Electrode equipment	Vertical arrangement of two opposed rod electrodes
Rod electrodes	Copper with 20 mm diameter and 300 mm gap length
Frequency	50 Hz
Voltage, current	9.2 kV, 35 kA
Arc duration (arc energy)	0.05, 0.25, 0.5, 1.0 s (1, 5, 10, 20 MJ)
Target cable	Flame-retardant (600 V/CV/3C, 22sq) with 150 mm length
Target distance	0.3, 0.6, 0.9, 1.5 m

Exposure Test Results

Figure 13 shows the occurrence of the arc flash and of cable damage. Figure 14 shows the test results of the cable exposure test subjected to hot arc ejecta, and the damage criteria provided in NUREG/CR-6850 [12] were also plotted.

According to the investigation of the damaged cables, the damage status could be distinguished as follows:

- Status 1 – E_{ins}: 100 – 300 kJ/m^2:
  Non-challenging damage: Partial melting and carbonization of the cable surface
- Status 2 – E_{ins}: 300 – 500 kJ/m^2:
  Potentially challenging damage: Carbonization of the cable heating surface
- Status 3 –- E_{ins}: 500 – 3000 kJ/m$^{2:}$
  Challenging damage: Remarkable carbonization and deposits all around the cable

Nevertheless, as there was no degradation of the dielectric strength of the cable specimen ranked in status 3 during the withstand voltage test, the specified cable damage criteria subjected to impulsive thermal load such as hot arc ejecta should be formulated.

[Before test] [E_{arc} 5MJ] [E_{ins} : 2822 kJ/m^2] (All of the surface: Carbonization)

Figure 13 Occurrence of the arc flash and cable damage

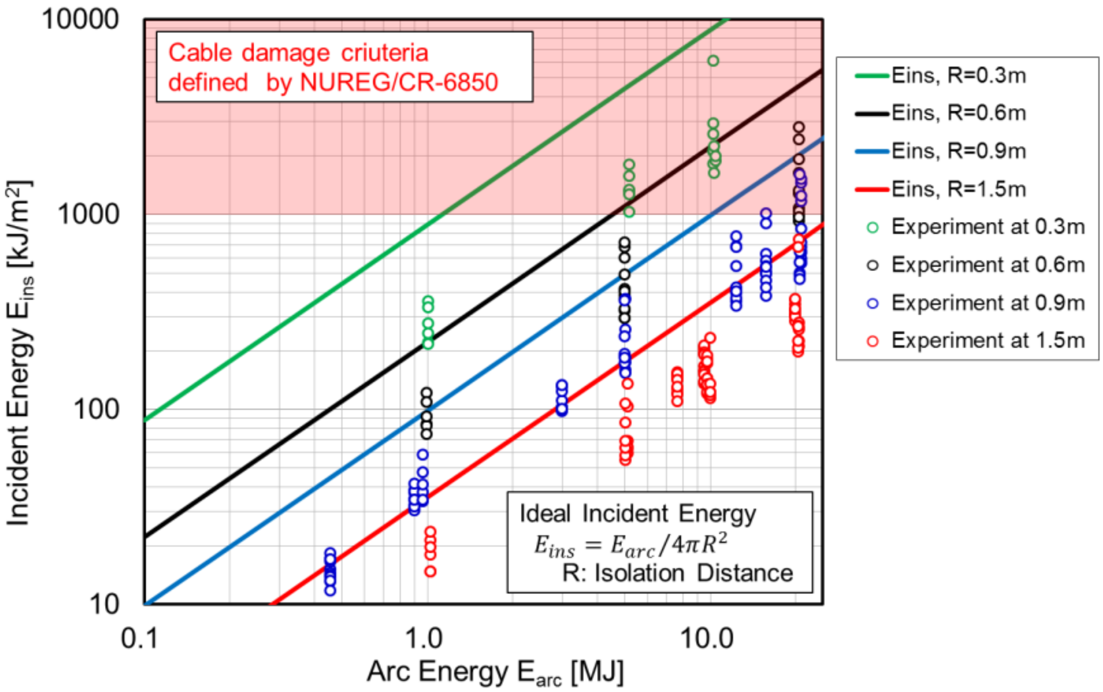


Figure 14 Cable exposure test results using flame-retardant cables

CONCLUSIONS

The ensuing fire due to the HEAF event in a high voltage metal-clad switchgear was identified at the Onagawa NPP after the 2011 off the Pacific coast of Tohoku Earthquake in March 2011. During the HEAF event, hot gas heated in the metal enclosure and, due to the arc flash were emitted out of the enclosure or to the adjacent enclosure and had a potential to damage the surrounding equipment.

In light of this, internal arc tests were performed by CRIEPI with full-scale high and low voltage electrical cabinets (non-arc proof type, copper bus conductor), and the arc energy, the mechanical damage of the cabinets and the surrounding equipment due to the impulsive pressure and the possibility of ensuing fire occurrence were evaluated. As a result, an empirical formula to estimate the total arc energy emitted during HEAF event from high and low voltage electrical cabinets including power centers and motor control centers was proposed. On addition, the upper limit values to prevent the HEAF fire event were proposed.

Moreover, cable exposure tests using flame-retardant cables subjected to the hot arc ejecta gas in an open condition were performed. According to the test results, the thermal damage to the dielectric strength of the cable specimen was not detected even under the high incident energy of ore than 2000 kJ/m^2.

ACKNOWLEDGMENTS

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REFERENCES

[1] United States Nuclear Regulatory Commission (NRC) Office of Research: Joint Analysis of Arc Faults (Joan of ARC) – OECD International Testing Program for High Energy Arc Faults (HEAF), NEI Fire Protection Information Forum, Austin, TX, USA, September 2012.

[2] Roewekamp, M., et al.: Experiences from the OECD FIRE Database and Intended Future Extensions, Paper in: Proceedings of 13th International Probabilistic Safety Assessment and Management Conference (PSAM13), Seoul, Republic of Korea, October 2016, https://publons.com/journal/325745/proceedings-of-the-international-conference-on-pro.

[3] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI): OECD FIRE Project – Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R(2013)6, Paris, France, June 2013, http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf.

[4] Quoted from volume 7077 of the Japanese government newsletter, dated August 8, 2017.

[5] International Organization for Standardization (ISO) / International Electrotechnical Commission (IEC): International Standard ISO/IEC17025, General requirements for the competence of testing and calibration laboratories, Third edition 2017-11, Reference number ISO/IEC 17025:2017(E), Vernier, Geneva, Switzerland, 2017, http://nbsm.gov.np/uploads/files/ISO_IEC_17025_2017(E)-Character_PDF_document.pdf.

[6] Shirai, K., et al.: Experimental Studies of the HEAF (High Energy Arcing Fault) Fire Event for High Voltage Switchgears, CRIEPI Report No. O16001, Tokyo, Japan, March 2017 (in Japanese).

[7] Miyagi, T., et al.: Experimental Studies of the HEAF (High Energy Arcing Fault) Fire Event for a MCC (Motor Control Center), CRIEPI Report No. O18002, Tokyo, Japan, December 2018 (in Japanese).

[8] Shirai, K., et al.: Experimental Studies of the HEAF (High Energy Arcing Fault) Fire Event for a High Voltage Cabinet to Energize a Diesel Generator, CRIEPI Report No. O18003, Tokyo, Japan, March 2019 (in Japanese).

[9] Japan Electrical Manufacturers' Association (JEMA): A.C. metal-enclosed switchgear and control-gear for rated voltages above 1 kV and up to and including 36 kV, Standards of the JEMA, JEM1425-2011, 2012.

[10] Iwata, M., et al.: Influence of Current and Electrode Material on Fraction k_p of Electric Arc Energy leading to Pressure Rise in a Closed Container during Internal Arcing, IEEE Trans. on Power Delivery, Vol. 25, No.3, July 2010.

[11] Babrauskas, V.: Electric Arc Explosions, in: Interflam 2010 – Proceedings of 12th International Conference, pp. 1283-1296, Interscience Communication Ltd, London, United Kingdom, 2010.

[12] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research, and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Final Report, NUREG/CR-6850, EPRI 101198), Washington, DC, USA and Palo Alto, CA, USA, September 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/.

3.7 SMiRT Fire Seminar Contributions Presented During FSEP 2019 Sessions

Four contributions prepared for the SMiRT Post-conference Fire Seminar could not be accepted to be presented in the frame of the limited Seminar time schedule. As these contributions were also of interest for FSEP 2019 participants, the presentations were given there. The papers are nevertheless published as part of the Seminar Proceedings in alphabetical order.

The first of these presentations was prepared by P. Boulden Jr., who outlined the implementation process of a fire protection program in order to effectively manage the risk from fires in nuclear power plants for improving nuclear safety.

J. Eduful from CNSC (Canada) presented an overview of fires of liquid sodium representing an important safety issue also for modern sodium-cooled fast reactors.

Another presentation was given by E. Elewini, Candu Energy Inc. (Canada) addressing investigations regarding the habitability of the main control room of Canadian nuclear power plants in case of fire representing an important safety related issue.

Last but not least, the contribution by A. Niggemeyer, Framatome (Germany) dealt with the issue of concepts for access and escape routes in nuclear power plants focussing on possible challenges arising from the design of the facility being analyzed and proposals for resolving this issue.

The four contributions are provided hereafter.

Implementation of an Effective Fire Protection Program to Manage Fire Risk and Improve Nuclear Safety

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ABSTRACT

Meeting the basic (required) separation criteria of various regulations such as U.S. NRC 10 CFR 50, Appendix R or Japan NRA Fire Protection Rule is an important first step to managing fire risk; yet experience on several projects around the world demonstrates that nuclear industry continues to struggle with full integration of site-wide fire protection programs.

Fire protection for nuclear power plants uses the concept of defense-in-depth to achieve the required degree of reactor safety. This concept entails the use of echelons of administrative controls, fire protection systems and features, and safe shutdown capability to achieve the following goals:

- Prevent fires from starting;
- Detect rapidly, control, and extinguish promptly those fires that do occur;
- Protect structures, systems and components (SSCs) important to safety, so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

Each of these elements may be further reviewed in order to provide illustration for how these may successfully be integrated into a plant-wide program.

This paper provides an overview of how a well-coordinated fire protection program significantly reduces fire risk in the plant and therefore improves nuclear safety.

INTRODUCTION

Fires can be a major contributor to both core damage frequency (CDF) and large early release frequency (LERF). In comparison to other internal and external events, many of which are limited to a single component or system failure, fire can affect multiple components or systems. When a fire occurs, it has the potential to grow, spread very quickly, affect many components, and cascade through multiple systems. An event like this is typically a result of inadequate fire prevention and protection programs. In contrast, an effective fire protection program minimizes the potential of fire ignition and growth; will lower the probabilistic risk of fire damage within your plant; and limit the deterministic threshold level of damage should a fire initiate.

Under “Fire Protection Program Goals and Objectives” United States Nuclear Regulatory Commission (U.S. NRC) RG 1.189 [1] includes:

- “Defense-in-Depth

  Fire protection for nuclear power plants uses the concept of defense in depth to achieve the required degree of reactor safety. This concept entails the use of *echelons of administrative controls, fire protection systems and features, and safe-shutdown capability to achieve the following objectives:*

- *Prevent fires from starting.*
- *Detect rapidly, control, and extinguish promptly those fires that do occur.*
- *Protect SSCs important to safety, so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant."*

Other goals cited by RG 1.189 [1] include:

- Fire protection program performance goals to protect systems, structures, and components (SSCs) important to safety from the effects of fire;
- Post-fire safe shutdown performance objectives related to safe shutdown after a fire;
- Prevention of radiological releases.

Beyond the regulatory goals listed in RG 1.189 [1], other goals of an effective fire protection program include life safety goals and the protection of the business unit and assets.

The primary goals of an effective and strong fire protection program are the following:

- Recognize the risk and potential consequence of a fire in any location of the power plant;
- Minimize the potential of occurrences and consequences of a fire;
- Minimize the migration of a fire and its effects from the fire area or fire compartment of origin;
- Rapidly detect a fire initiation and ensure the capability to rapidly deploy effective suppression strategies;
- Ensure plant safety, reliability, and continued electrical generation;
- Ensure that operations have the capability to effectively shutdown the reactor and maintain it in a safe and stable mode for a single fire occurring in any fire area or fire compartment by maintaining at least one train free of fire damage;
- Support and exceed the local, regional, and national regulatory expectations;
- Identifies the various positions of responsibilities and authority within the organization to implement an effect program;
- Verifies that the firefighting systems are designed to assure their rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components;
- Minimize the potential that a fire will release radiological and contaminates outside of the plant boundary.

This paper introduces the insight and provides the background for demonstrating why other fire protection features and administrative processes other than those constructed for maintaining fire separation are important parts of a successful fire protection program. These other fire protection features and administrative processes, which will be discussed, support the Fire PRA (*P*robabilistic *R*isk *A*ssessment) and contribute to significantly reduce the risk contribution from a postulated fire event. U.S. NRC Reg. Guide 1.205 "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" [2] and NUREG/ CR-6850 [3] provide additional information regarding the implementation of Fire PRA for compliance with U.S. fire protection regulation and methodology.

PAPER OVERVIEW

- What are the elements of an effective fire protection program?
- What is the basis for the fire protection program elements?
- Why an effective fire protection requires attention, review, and update?
- How do you measure fire protection program effectiveness?
- What is an effective fire protection maintenance, inspection, and testing program?
- What are the economic benefits of an effective fire protection program?
- Conclusion

ELEMENTS OF AN EFFECTIVE FIRE PROTECTION PROGRAM

Fire protection programs can be divided into three (3) main elements:

- Classical fire protection: control of fire loading, control of potential ignition sources, early warning fire detection, and automatic and manual fire suppression;
- Safe shutdown analysis (SSA): ensures the plant can maintain the core in a safe and stable condition in the event of fire in any fire location;
- Risk analysis: determines what specific fires have the potential to happen, the postulated frequency of occurrence, and their potential consequences as quantified in CDF and the LERF results.

Within the most successful and effective fire protection programs, the responsibilities associated with these three main elements are directed by the individual responsible for the program's day to day operation, while also taking ownership of the program and having the authority to make changes where applicable. Other shared responsibilities are extended throughout the plant organization into the uppermost levels of operations and maintenance, which also contributes to the success and effectiveness of the fire protection program. These responsibilities of the fire protection program require support from other departments, such as electrical, civil, PRA, mechanical engineering, and housekeeping.

Effective fire protection programs must also perform the following:

- Document a clear and concise interpretation of all of the current governing regulations as they were adopted, negotiated, interpreted, and superseded because they form the facility's licensing basis. Each plant's application of the regulation is unique; therefore, the more concise on how the regulation was licensed and implemented into the fire protection program provides the ability for all those involved, the regulator as well as the utility and its employees, to understand the basis, maintain consistent configuration control, and recognize the level of adherence that the fire protection program is measured against.
- Define its primary goals and then builds program layers of fire prevention and fire protection around it, such as the following: fire confinement, fire brigade readiness, fire suppression mitigation strategies, fire shutdown operational procedures, monitoring of electrical equipment and rotating machinery, controls of transient and fixed combustibles, and control of "hot work" activities. The effective fire protection program is also always evaluating if the correct fire prevention administration programs are in place for the conditions and if the available fire mitigation equipment and strategies sufficient.
- Develop an understanding that fire protection program requirements are specific to your site and its regulatory licensing; however, the key elements necessary for an effective program are common to all effective fire protection program elements. As

can be demonstrated in Figure 1, using a simplified “wheel and spoke” analogy, there are many attributes that make up the effective fire protection program to prevent or mitigate fires in a commercial nuclear plant.

Basis for Fire Protection Program Elements

The integral fire protection programs and features, at a very high level, were shown in Figure 1. Each of the fire protection features and programmatic processes should be addressed to result in an effective fire protection program. Below are some considerations for the functional areas that are included in the fire protection program.

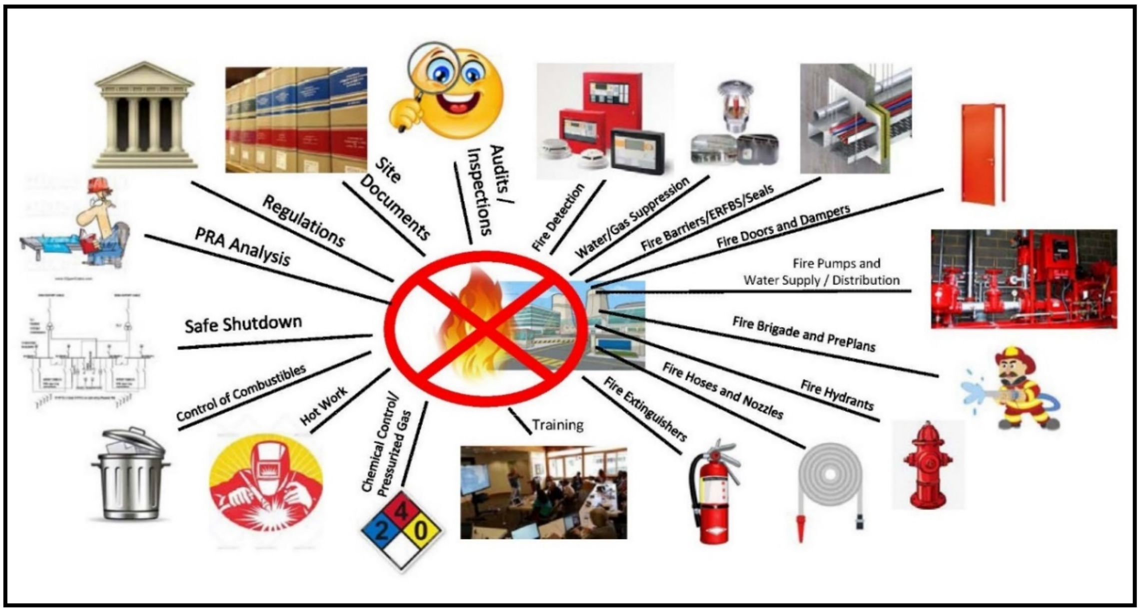

Figure 1 Inputs to a fire protection program

1. Detailed information regarding U.S. NRC inspection criteria and guidance for implementation may be found in [4] to [11]. For classical functional features and programmatic processes, an effective fire protection program performs the following:
   - Fixed and transient combustibles controls:
     - Help maintain a reduction in the number of additional fuels that could be exposed to fire ignition sources;
     - Aid in the reduction of combustible loading and ability to spread should a fire occur;
     - Provide an understanding of the high consequence of additional plant combustibles to plant managers, operators, and workers;
   - Administrative controls and limitations in the material and equipment procurement specifications:
     - Ensure that SSCs are procured and manufactured with appropriate material to help minimize overall fire heat release, smoke development, and plant risk;
     - Ensure the quality of manufacturing and materials for SSCs are procured for use in fire prevention applications and exposures to fire;

- Ensure the SSCs have been certified by the appropriate testing agencies, such as Underwriter Laboratories (UL), Factory Mutual (FM), American Petroleum Institute (API), and Steel Door Institute (SDI);

- Hot work and welding administration programs:
  - Control and manages activities that produce elevated temperatures and sparks, i.e. potential transient fire ignition sources;
  - Provide guidance on how to minimize the potential exposure to and ignition of surrounding materials;
  - Provide guidance on the expectations for the people performing the work and those providing the fire prevention observations to minimize the potential of a fire and how to rapidly extinguish any fire that may occur;
- Fire protection water supply systems:
  - Full flow testing of the supply piping ensures the overall water delivery system can provide the expected and required flow and pressures to each suppression system as the plant ages;

    Note: Typical codes recommend only testing the buried water delivery yard piping, but it is good practice to also test the internal suppression piping to ensure substantial degradation has not occurred in the inner piping supply standpipes and the fire protection mains.
- Fixed suppression systems:
  - Testing verifies the functionality, availability, and reliability of the systems to ensure they remain within the PRA analysis assumptions.
  - Maintenance and surveillance inspections verify construction materials and sub-components are not degraded such that functionality is assured till the next inspection.
- Fire Detection Systems:
  - Testing verifies the functionality, availability, and reliability of the systems to ensure they remain within the PRA analysis assumptions;
  - Ensure that the main control room (MCR) has the quickest possible fire development notification in order to take safe shutdown actions and to notify the fire brigade to respond.
- Fire Barrier Separations:
  - These include: fire doors, HVAC dampers, cable and pipe penetrations, electrical raceway fire barriers systems, radiant heat shields, and separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards;
  - Periodic surveillance of the required building separation features ensures that the features will perform the fire, smoke, and heat separation functions as well as inhibiting the potential of fire migration;

  Note: The inspection process does not need to involve 100 % of the components due to the slow aging degradation process of these elements. Inspections are typically performed on a 10 % population selection annually, whereas 100 % of the separation components are inspected within a 10-year cycle. Failures to meet acceptance criteria usually results in an immediate increase in the inspection population.
- Inspections for controls spills and leakage of combustible and flammable fluids:
  - Inspections ensure the fluids are confined and away from contact with potential ignition sources;

- Identification of leakages should be documented in the corrective action or maintenance program to establish a higher repair priority;

- Monitoring for electrical "hot spots" and rotating and electrical equipment vibration:
  - Electrical hot spots are potential precursors to electrical component failures that can lead to a wide range of fire scenarios, from small fires to large high energy arcing fault (HEAF) fires;
  - High vibration in electrical equipment can lead to relay chatter and initiate fires;
  - High vibration in rotating equipment can lead to equipment failure that releases lubrication fluids in their liquid/vapor state and expose them to surfaces close to their ignition temperatures. In addition, high vibration could cause a mechanical failure that results in a failure of the electrical driver (motor), which could be a fire ignition source;
- Inspection of electrical cabinet access panels and penetrations:
  - Inspections ensure secured access panels are in the closed position, and seal penetrations are intact to prevent potential electrical arc faults. These two preventative measures are also used to inhibit internal fires from coming in contact with combustibles outside the cabinet of origin;
- Test and surveillance frequencies:
  - Monitoring of the testing and surveillance frequencies is performed to determine if adjustments based on the results are required. Adjustments include extensions or increases in periodicity when results are negatively trending;
  - Ensure that equipment testing and surveillance is maintained with the frequency of the industry;
  - As an example, if a site's diesel fire pump is being tested weekly, but the rest of the industry tests theirs quarterly, then the site is either over testing their diesel fire pump or their maintenance and test results are not supportive of extended testing periods.
- Maintenance activities, temporary plant changes, and modifications:
  - Administrative programs or processes should in place to review each of these activities to ensure the existing fire strategies and safe shutdown analyses are not compromised;
  - In some instances, maintenance activities or the installation of new equipment can exceed the limitations on the local fixed or portable fire mitigation equipment or impede access to the location or other capabilities of the fire response team and safe shutdown recovery actions;
  - In other situations, maintenance in the area or modifications can cause obstructions that would adversely affect the actuation of a fire detection or suppression system or the expected fire suppression discharge pattern;
- Conducts process and programmatic reviews:
  - Identify and evaluate changes to SSCs for the risk of fire ignition and growth;
  - Continually evaluates if the fire prevention and mitigation strategies are the most effective and still are within the anticipated time margins;
- Fire brigade and/or fire department effectiveness:
  - Drills are performed in training facilities to demonstrate general fire mitigation capabilities;
  - Drills are performed in risk significant areas when possible to help familiarize the brigade with the layout and firefighting strategies used in those areas;

- Drills ensure a minimum time for activation is achieved, plant awareness, and knowledge of available plant shutdown functions and procedures consistent with the PRA assumptions;
    - Drills confirm that the fire brigade is knowledgeable and familiar with the hazards in the area, trained on the judicious use of suppressants to minimize damage to non-affected equipment, and trained to conduct the fire response suppression activities to minimize migration to other locations.
    - Frequent drills ensure that the fire brigade and local firefighting agencies can readily access any potential location within the facility's ownership and perform the strategies for that location. This ensures prompt response and mitigation.
  - Fire brigade strategy plans (i.e. fire pre-plans):
    - Periodic validation ensures the plans remain relevant to current plant configurations;
    - Periodic validation ensures the response can be highly effective during the fire mitigation and suppression process, considers water discharge runoff, and the implements effective smoke control, and monitors radioactive contamination release based on the latest advanced training techniques and equipment available.
2. Within the safe shutdown functional area, an effective fire protection program performs the following:
   - Uses fire and internal events PRA to identify the high-risk areas;
     - Even for those plants that meet the regulatory requirements for equipment and cable separation, Fire PRA can locate where there are potential "blind spots" in areas where additional mitigation might be appropriate to manage risk.
   - Ensures the plant is separated of into fire areas/zones/compartments;
     - These separations are designated in such a way that most effectively provide one train of shutdown equipment free of fire damage.
     - In the event that one train of shutdown equipment cannot be maintained free of fire damage for a fire area, the safe shutdown analysis should validate that redundant or alternate systems are available for use to maintain a safe and stable core geometry and negative reactivity.
   - Validation of credited structures, systems, equipment, components, and instrumentation:
     - The safe shutdown analysis validates that the credited structures, systems, equipment, components, and instrumentation to support safe shutdown remain reliable and available through periodic testing and, thereby, demonstrating operation and capabilities as analyzed for the fire event.
   - Verifies periodically that the selection of electrical power delivery systems for the equipment, components, and instrumentation:
     - Used to ensure the electrical delivery is properly evaluated in the analysis to remain available and reliable in the post fire event;
     - Used to ensure the circuits are "coordinated" to prevent the loss of the power supply buses under certain faulted conditions;
   - Evaluates systems, equipment, components, and instrumentation for new failure modes in each fire area of consideration using the latest industry operating experience (OE).

   - Replace or perform additional analysis on instrumentation and components that have been determined to fail as-is or high when requiring a low condition, or valves that have been analyzed to fail closed instead of open;
- Evaluates the risk potential for a fire causing damage to a complete or partial fire location and develop safe shutdown operating procedures utilizing the credited safe shutdown systems, components, and instrumentation that are free of fire damage;
- Evaluates the feasibility of the operator recovery and manual actions given potential adverse environmental conditions attributable to changes in accessible pathways to achieve access to systems, equipment, components, and instrumentation, or changes due to plant modifications or capabilities of the workforce;
- Evaluates the adequacy of lighting and communication during the fire event to evacuate personnel, operate equipment, or provide essential communication;
- Ensures operating personnel are periodically trained on safe shutdown procedures and performs routine simulator drills or Main Control Room exercises during the fire brigade drills.

3. Within the Probabilistic Safety Analysis (PSA) functional area, an effective fire protection program performs the following:
   - Evaluates and identifies through risk analysis computations, those targets most likely to be affected in any given fire location, such that the safe shutdown analysis can be further refined to ensure the most efficient and effective safe shutdown strategies are in place;
   - Evaluates and identifies through risk analysis computations, where modifications may or may not be cost effective and that crediting fire protection features and administrative programs for additional defense and depth demonstrate the most gain in reducing CDF and LERF to achieve the safe shutdown goals for a train free of fire damage;
   - Revises the fire modelling analyses based on the latest industry information to ensure the more realistic fire growth and damage in the fire locations are postulated;
   - Demonstrates and communicates to the fire support organization, management and operations those systems, equipment, components, and instrumentation that are most important to ensure nuclear safety and the expectations of functionality, availability, and reliability.
   - The Fire PRA can provide key insights on risk mitigation strategies that focuses key fire protection program enhancements in areas which provide the most effective risk reductions to core damage frequency (CDF) and the large early release frequency (LERF).
   - The Fire PRA results when reviewed in comparison to the internal events PRA demonstrates the importance of maintaining an effective fire protection program to all levels of management that attention and support since its insights indicate that Fire is one of the greatest – if not the single greatest – risk for an induvial nuclear power plant (NPP).
4. Not illustrated in Figure 1, are the ancillary processes that support each one of the inputs, such as:
   - Reliance on globally recognized consensus codes (such as the International Building Code (IBC), National Fire Protection Association (NFPA), and American Society of Mechanical Engineers (ASME), etc.) and manufacturer information for installation, maintenance, and performance expectations. Fire protection standards are recognized and suggested by governmental regulation in the form

of national, regional, and local regulations and ordinances. Insurance companies and their underwriters also impose requirements that influence the interpretation and implementation of the fire protection standards. National consensus codes, such as NFPA, use operating experience and the latest technology and research to enhance the design of fire protection systems and shutdown analysis in such a way to further improve the safety of a building occupants and/or property assets (i.e. a nuclear plant's reactor).

- The performance of regular maintenance programs that test and inspect the credited fire protection features and SSCs used for power production and safe shutdown; such as testing of the fire pumps, flow testing the fire water supply headers, operating fire hydrant, inspecting fire barrier configurations, checking the function of the fire detection cabinets and actuating devices, etc. Examples of maintenance on components used for power production include; circuit breakers, electrical cabinets, pumps, motors, other rotary equipment, etc. Two very strong fire protection program preventive maintenance initiatives are; the use of thermal imaging cameras to determine "hot spots" in operating equipment that would otherwise go undetected and monitoring rotating equipment for vibration. Without intervention and maintenance, these thermal anomalies can lead to fire ignition.
- The periodic operational functional testing to ensure the SSC is meeting its design performance criteria and that any design margin has not decreased and monitoring/trending the results to anticipate required maintenance before failure. Some equipment due to the low risk consequences may actual be allowed to trend to failure.
- The development of procedures and administration programs; such as for combustible control, hot work, fire impairments, fire brigade training, and fire brigade drills, testing and acceptance criteria for fire detection and suppression systems, etc.
- The continuous review and update of analysis (calculations), databases, and documentation that support the current configuration of all the fire protection program elements; such as the hydraulics of the fire protection piping and the effects of aging, trending the performance results of equipment testing, review plant incident reports to determine if they could actual also be precursor events to fire, etc.
- Organization management support – All levels of the organization have responsibilities to the success of an effective fire protection. From the lowest level identifying concerns to the highest level making sure the appropriate emphasis, scheduling, and support is allotted to ensure the fire protection program remain in a functional configuration, and
- Performing periodic program self-assessments to ensure the goals of functionality, availability, and reliability are maintained in accordance with the license bases and the credited national codes and standards.

Fire Protection Program Attention, Review, and Update

Once originated, does an effective fire protection program require attention, review, and update? *Yes,* in order for the fire protection program to remain effective. Science and technology continue to advance in addition to continuous industry information for improvement from operating experience; therefore, the knowledge to prevent and protect continues change, as well as, the scope of maintenance and testing, and the tools available (thermal imaging).

Does this mean that the fire protection program has to continually change? *No*. What it does mean is that an effective fire protection program not only continues to monitor the change of conditions at each location, but fire protection is continually evaluated to meet all prior commitments as well as consideration to apply both the newer advances in fire science technology and also the insight provided by probabilistic analysis. This ensures the fire protection program remains effective and uses weighted emphasis on the site locations of greatest concern to determine if fire protection features and/or program changes are warranted.

The results from the various fire protection maintenance, testing, and trending are constantly reviewed for negative trends. Operating experience (OE) is on a case by case occurrence. These type of reviews and oversite process are typically performed by the fire protection and safe shutdown engineers. Periodically, the fire protection program should be assessed by external reviewers to perform an independent critique and develop a status of the fire protection program as compared to the metric of adherence to its license bases. The frequency of the independent assessment is dependent on industry OE and how well the program performed in its last review. It is suggested that low performing programs be assessed annually and that higher performing programs be reviewed on a three-year cycle. The three-year review period aligns with the requirements for the U.S. NRC requirements.

There is a vulnerability for a plant's fire protection program effectiveness to diminish if it does not continually evaluate its programs. Complacency in believing that once established a fire protection does not require review and changes will allow for missed opportunities to remove the potential for fire initiating events, improved fire detection response, and fire suppression advancements. These improvements to the fire protection and prevention programs are constantly being identified by the standards, research, and operating events and could result in less precursors to fires and actual fires with decreased magnitude.

Fire Protection Program Effectiveness

An effective fire protection is established by initially setting up metrics, and then periodically reviewing the program against them. This process measures the effectiveness of the overall fire protection program. Some of the key program review metrics are as follows:

- Reported fires;
- Call outs of fire brigade or fire department;
- Continuity of electrical production;
- Loss of life due to fire;
- Audit violations for deviations to the program implementation processes or to license bases adherence;
- The PRA continues to demonstrate the fire event analysis results maintain stable or reduced CDF and LERF;
- Competency of the fire brigade during drills;
- The activities associated with "hot work" do not result in a fire;
- The activities of plant personnel do not violate the staging of combustible or flammable materials beyond that approved for the location;
- Manual fire suppression equipment remains readily available
- Automatic suppression and detection systems avoid multiple maintenance outages and continue to demonstrate functionality;

- Operations continues demonstrate competency in drill execution of safe shutdown procedures and emergency notifications;
- Fire impairments for maintenance and testing are kept to a minimum;
- Compensatory measures performed during periods are commensurate with the risk significance of fire equipment removed form availability.

The metrics are typically captured in a report format. The data sources are the site's logs, test procedure's recorded results, field observations, equipment incident reports, and the reports of fire.

The reporting of fire (e. g. smoke, visual flaming, use of manual suppression) is not the only classification to determine fire reporting. Precursors of fire are located in site incident reports using key word searches (such as; fire brigade drills failures, open, short, arc, flash, burn, hot, fire, flood, smoke, door, detector, hose, brigade, Appendix R, fire water system, welding, soldering, brazing, control of combustible, transient, hot work, surveillance, impairment, fire watch, etc.) are also important factors that inform and define the effectiveness of your fire prevention and mitigation programs.

Fire Protection Program Maintenance, Inspection, and Testing

Is power production and safety related equipment maintenance part of a good fire protection program? - *Yes*

Adhering to routine maintenance frequencies, manufacturer recommended inspections, and industry operating experience (OE) will lead to fewer functional failures which can result in a fire. However, as our industry is unique in the way we pay particular attention to our fire protection equipment as compared to most general industries. Therefore, regulatory authorities; agencies like Electric Power Research Institute (EPRI); and the nuclear utility insurance carrier Nuclear Electric Insurance Limited (NEIL) have developed additional suggested guidance that suggests prolonged frequencies for testing and maintenance as the results of maintenance and testing demonstrate successful performance. In contrary to demonstrating successful performance these additional guidelines also suggest accelerated testing frequencies until adequate performance is obtained.

The Fire PRA provide key insights to aid the fire protection program owner and management to efficiently allocate resources (i.e. proportion the monitoring effort in alignment with those fire protection features and programs that are significant risk reduction contributors). In using a more focused approach, the fire protection program becomes more effective, reduces the potential damage from fire, and consequently reduces the overall program implementation cost.

Ensuring that the SSCs of fire protection and safe shutdown equipment is always in a state of readiness and knowing the functionality, availability, and reliability provides the operators with a true sense of security that any event can be responded to quickly and the obtain the desired safe shutdown results. In addition, the benefits of fewer reports of SSCs´ failures results in greater public confidence that the utility has a safety conscious culture.

Economic Benefits of an Effective Fire Protection Program

An effective fire protection program manages the fire risk to lower frequency and the potential consequences of the fire event to challenge nuclear safety and loss of electrical production or interruption. The costs of an effective fire protection are reduced because there is no longer an equivalent application of resources to all the elements within the fire protection program. Instead, fire protection program cost are proportioned such that the rigorous maintenance and inspections are performed only to those elements that have been demonstrated to significantly lower risk. The remaining fire protection elements that provide minimal risk offsets or are there for non-nuclear protection are allocated reduced resources; therefore, the restructuring the fire protection program is a more cost effective. An unrealized cost of an effective fire protection program is that associated with fire recovery after the event since the fire event is now less likely to occur.

As stated above an effective fire protection program does this by emphasizing in fire locations of most significance a rigorous maintenance, testing, and inspection program with focus on the fire ignition notification and suppression capabilities; the safe shutdown selected set of systems, equipment, components, instrumentation; and administrative programs. This also has the effect to programmatically reduce the maintenance and inspections on equipment that are not as risk significant. Therefore, those fire protection program structures, systems, equipment, components, instrumentation, and administrative programs with lower risk incur reduced cost because they can be having less oversight and reduced frequencies of inspection, maintenance, and testing.

CONCLUSION

An effective fire protection program is one of the most powerful tactics a plant can utilize to reduce its overall fire risk and improve its overall level of safety. This is performed by developing and effectively executing the many fire protection program attributes. In addition, the program owner and the many individuals indirectly associated the fire protection program are ultimately responsible for effectively managing the integration into the plant's daily operation, recognize when it is appropriate to upgrade the processes and equipment and to make sure it occurs in a timely manner, and periodic assess the strength of the program as compared to its license basis and the general industry.

The development and integration of the following goals provide the backbone of a successful and effective fire protection program that reduces risk and elevates overall plant safety:

- Prevent fires from starting;
- Control combustibles effectively;
- Detect rapidly, control, and extinguish promptly those fires that do occur;
- Protect SSCs important to safety so a fire that goes undetected or unextinguished by fire suppression activities will not inhibit safe shutdown of the plant;
- Emphasize efforts of maintenance and inspection to SSCs that provide the most risk reduction benefit in lowering CDF and LERF.

REFERENCES

[1] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Regulatory Guide 1.189, Fire Protection for Nuclear Power Plants, Revision 3, Washington, DC, USA, https://www.nrc.gov/docs/ML0925/ML092580550.pdf.

[2] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES): Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, Washington, DC, USA, December 2009, https://www.nrc.gov/docs/ML0927/ML092730314.pdf.

[3] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory

Research and Electric Power Research Institute (EPRI): Fire Probabilistic Risk Assessment Methods Enhancements, NUREG/CR-6850 Supplement 1, and EPRI 1019259 Technical Report, Washington, DC and Palo Alto, CA, USA, September 2010, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/s1/.

[4] Nuclear Electric Insurance Limited (NEIL): Loss Control Manual, September 2010 Edition, Wilmington, DE, USA, 2010.

[5] National Fire Protection Association (NFPA): NFPA-805, Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, Quincy, MA, USA, 2001.

[6] Nuclear Energy Institute (NEI): Fire Protection Benchmarking Report, LP006, Washington, DC, USA, August 2002

[7] United States Nuclear Regulatory Commission (NRC): Inspection Procedure (IP) 71111.05, Fire Protection (Annual,) Washington, DC, USA, January 2019, https://www.nrc.gov/docs/ML1808/ML18085A040.pdf.

[8] United States Nuclear Regulatory Commission (NRC): Inspection Procedure (IP) 71111.05T, Fire Protection (Triennial), Washington, DC, USA, January 2012, https://www.nrc.gov/docs/ML1120/ML11201A170.pdf.

[9] United States Nuclear Regulatory Commission (NRC): Inspection Procedure (IP) 71111.05XT, Fire Protection-NFPA 805 (Triennial), Draft Version – available for public release, Washington, DC, USA, https://www.nrc.gov/docs/ML1105/ML110550560.pdf.

[10] Electric Power Research Institute (EPRI): Reliability and Preventative Maintenance: Balancing Risk and Reliability, Technical Report TR-1002936, Palo Alto, CA, USA, December 2002, https://www.epri.com/#/pages/product/1002936/?lang=en-US.

[11] Electric Power Research Institute (EPRI): Fire Protection Equipment Surveillance, Optimization and Maintenance Guide, Technical Report TR-1006756USA, Palo Alto, CA, USA, July 2003, https://www.epri.com/#/pages/product/1006756/?lang=en-US.

[12] Electric Power Research Institute (EPRI): Optimization of Fire Protection Impairments at Nuclear Power Plants, EPRI TR-1013211, Palo Alto, CA, USA, August 2006, https://www.epri.com/#/pages/product/1013211/?lang=en-US.

[13] National Fire Protection Association (NFPA): NFPA 25, Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems, Edition 2017, Quincy, MA, USA, 2017.

Overview of Liquid Sodium Fires: A Case of Sodium-Cooled Fast Reactors

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ABSTRACT

Six reactor technologies were selected and believed to represent the future shape of nuclear energy in the Generation IV reactor program. These are: gas-cooled fast reactor (GFR), lead-cooled fast reactor (LFR), molten salt reactor (MSR), sodium-cooled fast reactor (SFR), supercritical water-cooled reactor (SCWR), and very high-temperature gas reactor (VHTR). SFRs have the largest experience based among the six reactor technologies selected by the Generation IV International Forum (GIF). That is approximately 400 reactor years experienced over five decades and in over eight countries. The SFR has been the main technology of interest in the GIF.

The SFR uses liquid sodium as reactor coolant, allowing high power density with low coolant volume at low pressure close to atmospheric and a good safety margin to coolant boiling point. As a result, coolant leaks or pipe breaks do not result in the same loss-of-coolant (LOCA) accident effect such as the possible depressurisation, coolant boiling, and loss of cooling capability often postulated with light water or heavy water reactors. However, sodium has a high reactivity with air and water. Specifically, sodium leaks in air could lead to the production of toxic sodium-oxide aerosols caused by sodium fires. In addition, sodium's fast and exothermic reaction with water produces sodium hydroxide and hydrogen that could cause hydrogen explosions. To meet fire protection regulatory objectives, adequate evaluation and verification is required to be performed. This ensures that fires and explosions that may be associated with sodium's reactivity are prevented or adequately controlled not to compromise nuclear safety, damage structures, systems and components (SSCs) important to safety and put personnel at risk.

This paper presents an overview and discussion of liquid sodium fires associated with SFRs. It further discusses potential fire protection challenges and concerns that require careful evaluation to ensure fire safety of the next generation of nuclear reactors. A better understanding of fire and explosion hazards associated with liquid sodium handling is key to developing appropriates fire protection measures for future SFR design and enhancing regulatory requirements for SFRs.

INTRODUCTION

The Generation IV International Forum (GIF) selected six reactor technologies which are believed to represent the future shape of nuclear energy. These were selected on the basis of being clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks. These are: Gas-cooled fast reactor (GFR), lead-cooled fast reactor (LFR), molten salt reactor (MSR), sodium-cooled fast reactor (SFR), supercritical water-cooled reactor (SCWR), and very high-temperature gas reactor (VHTR). Out of the six reactors, three are fast neutron spectrum reactor systems, cooled with gas, sodium or a heavy liquid metal (lead or lead-bismuth).

Fast Neutron Reactor Technology

According to the World Nuclear Association [1], about 20 fast neutron reactors (FNR) have been operating since the 1950s. Over 400 reactor years of operating experience have been accumulated.

Natural uranium is composed primarily of two isotopes: 0.7 % is fissile U-235 and 99.3 % is U-238.This implies that only a very small percentage of the uranium is used in current power reactors that apply the thermal neutron design concept where a moderator is used to slow neutrons to lower energy to sustain fission. The principle of fast neutron reactors is to directly use neutrons arising from fission that have not been thermalized. This requires a far larger amount of fissile material and the amount (or "inventory") will then consist of either highly enriched uranium or plutonium. Fast neutron reactors can convert U-238 into fissile material at a faster rate than it is consumed. This makes it economically feasible to utilize cores with very low uranium concentrations [2], [3]. According to [4], fast neutron reactors are the ideal complement to currently operating water reactors to ensure a sustainable nuclear resource that optimizes the fuel cycle and the management of waste. Specifically, operating in a fully closed fuel cycle, fast neutron reactors have the potential to extract 60 - 70 times more energy from uranium than existing thermal reactors and contribute to a significant reduction in radioactive waste [5].

The core of a fast reactor is relatively smaller compared to current operating nuclear reactors (thermal neutron), and it has a higher power density, requiring very efficient heat transfer [1]. Fast neutron reactors are normally cooled by liquid metal such as sodium, lead or lead-bismuth, with high conductivity and boiling point and no moderating effect [6], [7]. They typically operate at around 500 – 550 °C at or near atmospheric pressure.

Sodium Cooled Fast Reactors

The SFR uses liquid sodium as the reactor coolant. This allows high power density with low coolant volume at low pressure [8]. This implies that coolant leaks or pipe breaks do not typically result in the same loss-of-coolant accident (LOCA) effect such as the possible depressurization, coolant boiling, and loss of cooling capability often postulated with light water or heavy water reactors.

The SFR utilizes depleted uranium as the fuel matrix. The coolant temperature ranges between 500 ºC and 550 ºC with the primary sodium circuit at near atmospheric pressure [9]. The reactor unit can be arranged as a pool layout or a compact loop layout. According to [10], three variants are considered:

i. a small size (50 - 150 MW_{el}) modular type,

ii. an intermediate-to-large size (300 - 1500 MW_{el}) pool-type, and

iii. a large size (600 - 1500 MW_{el}) loop-type.

The small modular type uses uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyro-metallurgical processing in facilities integrated with the reactor. The intermediate type uses oxide or metal fuel. The large loop-type design uses mixed uranium-plutonium oxide fuel and potential minor actinides, supported by fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.

Sodium as a Reactor Coolant: Physical and Chemical Properties

One of the key factors in selecting a nuclear reactor coolant is the efficient removal of heat generated in the reactor's core. In addition, the coolant should have excellent physical and chemical properties and low activation by neutrons.

Physically, sodium has one stable isotope - 23Na with an atomic mass of 22.99 g, and it is a very reactive element that easily loses its outer electron [4], [7]. Sodium has a melting point of approximately 98 ºC and a boiling point of 883 ºC at atmospheric pressure [11]. Sodium has a relatively high critical temperature in the range of 2,573 K to 2,733 K [12]. Sodium has a density of about 850 kg/m^3 at 450 ºC as a liquid and 966 kg/m^3 at 20 ºC as a solid [11]. Liquid sodium density is lower than density in the solid state and the volume expansion during the transition from the solid to the liquid state is about 2.7 % 311. Sodium has a very high thermal conductivity, i.e. 68.8 W/mK and specific heat of about 1.269 kJ/kgK at 450 ºC [11].

Chemically, sodium's reaction with water generates sodium hydroxide (NaOH) and gaseous hydrogen (H). This reaction is strongly exothermic and extremely quick. The reaction depends on temperature and impurities present in sodium [7]. Sodium interacts with dry hydrogen and forms sodium hydride (NaH), which is soluble in sodium. At the temperature of 420 ºC, NaH is decomposed with the release of hydrogen. In the liquid state, only sodium oxide (Na_2O) is stable, while sodium peroxide (Na_2O_2) dissociates [7]. Sodium reacts with air to generate sodium peroxide Na_2O_2 or in the case of low oxygen content of sodium oxide Na_2O, of hydrated sodium hydroxide (NaOH-xH_2O), and of sodium carbonates (Na_2CO_3 and $NaHCO_3$) in variable proportions depending on the humidity or presence of carbon dioxide [4]. Liquid sodium burns above its melting point of 97.8 ºC when it is dispersed in air as droplets or burns above 140 ºC, if it is in the form of a liquid sheet [4].

Sodium has several advantages in its use as a primary coolant. This includes but is not limited to the following: very good thermal properties (low melting point and large margins to the boiling point); a low viscosity; a low density; a low activation by neutrons; a very good compatibility with materials; and a strong availability at low cost. However, sodium's high reactivity with air and water could lead to fires and explosions.

OVERVIEW OF SFR DEVELOPMENT

SFR have the largest experience based among the six reactor technologies selected by the Generation IV International Forum (GIF). That is approximately 390 reactor years experienced over five decades and in over eight countries. The SFR has been the main technology of interest in GIF.

Experimental reactors, industrial prototypes and industrial-sized reactors were built and operated in a number of countries. The first nuclear power reactor of this kind was the EBR-I at the Idaho National Laboratory (INL) in the USA (1.4 MW_{th}, 0.2 MW_{el}). The EBR-I operated between 1951 and 1963. Since then, a number of SFRs have been put into operation in France, the United Kingdom, Germany, Russia, USA, Japan and India.

There are about seven SFRs that are currently in operation or standby: BOR-60, BN-600 and BN-800 in Russia, CEFR in China, FBTR and PFBR in India, Joyo in Japan, see Table 1. In addition, a number of projects are underway with varying degrees of advancement, see Table 2.

Table 1 Current SFRs in operation or standby [1]311

Country	Reactor	Type	Power [MW] thermal/electric	Year
Russia	BOR-60	Loop, Oxide	55/10	1969
Russia	BN-600	Pool, Oxide	1470/600	1980
Russia	BN-800	Pool, Oxide	2100/864	2014
India	FBTR	Pool, Carbide	40/13	1985
India	PFTBR	Pool, Carbide	1250/500	2020[6]
China	CEFR	Pool, Oxide	65/20	2010

Table 2 Proposed SFR projects [1]

Country	Reactor	Type	Power [MW] thermal/electric	Year
Russia	BN-1200	Pool, Oxide nitride	2800/1220	2020
Russia	MBIR	Loop, Oxide	100 - 150 MW_{th}	2020
India	FBR 1 & 2	Oxide	600 MW_{el}	-
China	CDFR-1000	Pool, Oxide	1000 MW_{el}	2023
China	CDFBR-1200	Pool, Metal	1200 MW_{el}	2028
Japan	JSFR	Loop, Oxide	500 MW_{el}	2025
USA	PRISM	Pool, Metal	840/311	2020
USA	ARC-100	Pool, Metal	260/100	-
Republic of Korea	PGSFR	Pool, Metal	392/150	2028
China, with USA	TWR	Metal	600 MW_{el}	2023
France with Japan	Astrid	Pool, Oxide	100 - 200 MW_{el}	2025

SODIUM FIRES IN POWER REACTORS AND TEST FACILITIES

Sodium leak in air could lead to the production of toxic sodium-oxide aerosols caused by sodium fires. In addition, sodium's fast and exothermic reaction with water produces sodium hydroxide and hydrogen that could cause hydrogen explosions.

[6] Physical construction of the reactor started in 2004 and was completed in 2015. However, the reactor has not yet reached first criticality.

A number of incidents have resulted in sodium fires in NPPs and test facilities in the past. The notable accidents in the literature [13] include Monju, BN-600, ILONA test facility, and Almeria solar test facility.

Monju, Japan

Monju is a loop-type 280 MW_{el} sodium cooled reactor with mixed oxide fuel. The inlet sodium temperature for the primary coolant loop was 397 °C and the outlet sodium temperature was 529. The sodium temperature for the secondary loop ranged between 325 °C and 505 °C.

The plant experienced a sodium leak and fire during a scheduled power rating test in 1995. The fire resulted in high sodium temperatures and increased thick smoke layer that interfered with normal plant shutdown procedures. Investigations later confirmed that the sodium leak and fire resulted from a damaged temperature sensor associated with the design of the well tube.

BN-600, Russia

The BN is a pool-type 600 MW_{el} reactor with oxide fuel. The inlet sodium temperature for the primary coolant loop was 337 °C and the outlet sodium temperature was 550 °C. The sodium temperature for the secondary loop ranged between 328 °C and 518 °C. In 1993, BN-600 experienced a major leak in the pipeline for sodium removal from the cold trap that resulted in a spray and pool fire. Investigation of the incident revealed that the leak resulted from two cracks close to a welded joint that might have been caused by fatigue.

ILONA Test Facility, Germany

ILONA was a sodium test facility to investigate natural convection in decay heat removal loops in Germany. In 1992, ILONA experienced a leak that resulted in sodium-concrete interaction and an ensuing fire. The leak was initiated by failure in a pressure regulating valve which caused the control gas to be unavailable during a heating procedure in receiving sodium in storage vessels. The leak caused the concrete floor to be lifted by reaction products and thermal expansion of the reinforcing steel. Concrete temperatures were observed up to 900 °C.

Almeria, Spain

Almeria was an experimental solar plant test facility near Almeria, Spain. Sodium was used as a means to store and transport heat to power conversion components of the plant. In 1986, a violent sodium spray fire occurred during repair work on a leaking sodium valve. Surrounding materials reached temperatures in the range of 1450 °C. The flames damaged a nearby storage vessel, deformed and raptured steel piping and support structure, and melted aluminium valve drive components.

SODIUM FIRE DYNAMICS

Sodium's high propensity to oxidize at relatively lower temperatures presents safety challenges that require careful evaluation and planning. Liquid sodium burns above its melting point of 97.8 °C when it is dispersed in air as droplets or burns above 140 °C, if it is in the form of a liquid sheet. Ignition of sodium has been observed to be in the range of 120 °C to 320 °C dependent on several factors [14]. However, the IAEA [15] noted that ignition in air may occur at lower temperatures for fine mist. Recent fire experiments conducted by Sandia National Laboratories (SNL) [16] indicated that sodium ignited at temperatures below previously observed ignition temperatures. The ignition temperature of sodium depends on the surface to volume ratio, oxygen molar fraction and humidity, and the degree of sodium purity [17]. It is important to note that sodium in nuclear reactors, i.e. inlet and outlet temperatures are operated in a closed system above the ignition temperature of sodium. This implies that sodium leaks are expected to ignite. As part of the evaluation and planning, reactor designs are required to incorporate a means of preventing such leaks or mitigating fires associated with such leaks.

Sodium fires result in thick smoke layer that can interfered with normal plant shutdown procedures or challenge emergency response activities. This is typically sodium oxide when sodium burns in normal air [18], see equation 1:

$$2\,Na + O_2 \rightarrow Na_2O_2 \qquad (1).$$

Na_2O_2 will then quickly react with air humidity and are transformed into sodium hydroxide (NaOH), a toxic gas due to its corrosive nature [19]. This is further transformed into sodium carbonate (Na_2CO_3) by reaction with carbon dioxide in the atmosphere. Narayanan et al. [20] noted through an experimental study that after an extended period of time (btn 30 – 90 min) the sodium carbonate transforms further into the formation of sodium bicarbonate ($NaHCO_3$) by reaction with CO_2 and water.

The sodium flame has a quite characteristic yellow-orange color and corresponds to the 589 nm wavelength sodium emission line. Temperature within the flame may reach 1,340 °C [4]. Sodium fires increase the temperature and pressure in the containment, producing aggressive compounds, increase the corrosive burden on reactor components and cause the release of large amounts of aerosols consisting of different sodium compounds. To maintain reactor safe state, safety functions such as reactor shutdown, core cooling and containment should be secured. In addition, sodium aerosol to the atmosphere can become a source term and needs to be limited to safeguard the public and the environment. Given the high chemical reactivity, liquid sodium should be handled extremely carefully. For example, consideration should be given to liquid-phase sodium being stored or used in the presence of inert gas (without oxygen or moisture), e.g., with nitrogen or argon. Sodium fires are typically described as spray fires or pool fires.

Sodium Spray Fires

Spray fires are characterized by the dispersion of metal particles (droplets) through the oxidizing atmosphere typically resulting from a sodium leak. The spray fire configuration maximizes the surface-to-volume ratio and this tends to maximize the potential reaction and heat-release rates. The expected droplet size is large enough that the time for droplet to fall to the ground will often be less than the time to completely oxidize the droplet [16]. This can lead to incomplete combustion of the sodium leaving some fraction of the sodium behind to burn as a pool fire (creating a combined spray/pool scenario). The ignition temperature for sodium spray is substantially lower than that of a pool fire.

Sodium Pool Fires

Sodium pool fires exhibit a surface combustion between the time of ignition and a temperature of approximately 420 °C. This is followed by the vapour phase combustion between 600 - 650 °C. The oxides will then sink and burning then proceeds over a flat metal surface [21]. As a result of sodium’s low volatility, sodium flames occurring above a pool fire height are very short, and the heat produced by convection is low, in contrast with a hydrocarbon fire. Results from experimental pool test conducted by Sandia National Laboratories [16] show the significance of a smoldering mode of combustion where the oxidation is well within an oxide-crust layer that eventually forms on the surface of the sodium pool. This means that a sodium pool fire can be smothered by spreading an extinguishing powder on the surface of the burning pool [4]. However, the IAEA [15] noted that further investigations are necessary to develop sodium firefighting applications and procedures for powder systems.

FIRE PROTECTION IN SFR

Structures, systems and components important to safety in a nuclear power plant are to be designed and located to minimize the likelihood and effects of fires and explosions caused by external or internal events. Generally, in reactor design, defence-in-depth concepts, i.e. incorporation of redundant parts, diverse systems or paths, physical separation and barriers are employed to achieve the following fire protection objectives:

- Prevent fires from starting: requires that the design of the plant minimizes the probability of a fire starting;
- Quickly detect and extinguish fires that do start: this relies on active measures by a combination of automatic and/or manual techniques;
- Prevent the spread of fires that have not been extinguished: this relies on the use of passive systems such as fire separations or barriers, physical or spatial barriers.

Specifically for SFR, the safety design against sodium fire should also consider the following:

- Mitigation of pressure and temperature rise inside containment, and
- Limitation of sodium aerosol release in both the primary and secondary sodium heat transport;
- Limitation of sodium-concrete reaction

Typically, a deterministic approach (fire hazard analysis, FHA) is used as the basis in achieving the fire protection objectives. The approach will typically include: postulating a fire for fixed or transient combustible materials; single fire postulated to occur at any given time, and the fire is postulated considering the different states of the power plant, i.e. at power or shutdown.

Fire Prevention by Design

The design of the nuclear power plant should be such that fire loads[7] and ignition sources are minimized as low as reasonably achievable. This often calls for the use of non-combustible materials or fire-retardant materials in the design. In addition, transient combustible materials that are integral to the operation of the power plant should be properly handled, stored or protected.

In SFR, the sodium coolant presents a significant amount of fire load in the operation of the power plant. It must be noted that in the rare case of failure of a sodium bearing component, sodium can leak out and react with oxygen in air to ignite when oxygen

[7] Fire loads: heat output or total energy content of combustible materials in an area. This is a measure of the potential severity of a fire in a given area.

concentration in the air is more than 5 % and the sodium temperature is more than 200 °C [22]. The plant design should be such that sodium is appropriately protected and sodium leak is minimized to ensure safe operation. Sodium components and pipes should be designed firmly against external and internal events to ensure its integrity during the plant life. The design considerations such as reducing sodium boundary area or pipe length and reducing the junctions of pipes maybe effective in limiting sodium leak [23]. Specific leak-mitigation provisions may include: insulation of sodium components, leak detection at the microscopic level and sodium collection systems.

Sodium components, including piping, is insulated or double jacketed with an annular space (possibly inerted) between the component surface and the inner surface of the insulation. The annular space serves as a means to mitigate spray formation. The design approach is to quickly detect sodium leak, confine, collect and direct the potential sodium leakages out of the annular space through a drainpipe connected to the inner jacket. Drain connections discharge into a piping system that carry leakage away to a hold-up tank that excludes oxygen necessary to support combustion. According to [24], the key features of the concept include:

- an inner jacket capable of withstanding sodium jet impingement loads from the sodium leak,
- a mechanical seal between the inner jacket and penetrations, such as pipe clamps and heater leads, to minimize sodium leakage into the cell environment, and
- a 20 cm diameter drain connection to the annular space between the pipe and the inner jacket, sized to channel sodium into a liquid stream with minimum buildup of backpressure in the annulus.

In general, the leak before break concept is facilitated in SFR due to the following features [20]: the low pressure of the coolant causes less stress on the boundary; materials used have sufficient ductility and penetrating crack is prevented from causing an unstable fracture for main pipes of the loops; and the leakage of chemically active sodium is easily detectable.

Fire Detection and Extinguishment

Means to quickly detect and extinguish fires should be provided to minimize the adverse effects of fire on safety related systems, personnel and the environment.

Sodium fires produce toxic aerosol compounds and increase the corrosive burden on reactor components and can adversely affect life safety. Specific life safety concerns may include but are not limited to respiratory system burn due to sodium aerosol, skin burn due to metallic sodium contact and alkaline burn due to sodium hydro-oxide. Therefore, prompt detection of sodium fires or leak is key to the safety of susceptible reactor component and life safety. Sodium fire detection systems include but are not limited to sodium ionization detectors and photoelectric smoke detections. To ensure prompt detection, Very Early Smoke Detection Apparatus (VESDA) systems should be considered.

Fire extinguishing systems are designed for automatic or manual actuation. However, the systems should be designed and located to ensure that their spurious or accidental operation does not significantly impair the capabilities of safety related structures, systems and components. Several fire protection systems have been developed to mitigate the effects of sodium fires and are generally based on limiting the oxygen available for combustion [23]. This includes sodium leak collecting systems, compartmentalization and space isolation and the use of fire extinguishing agent.

Specific fire extinguishing powders have also been developed. The approach to extinguishing sodium fires is to employ a powder that is thermodynamically stable in the presence of the metal and oxidizer. The powder is employed to smother the fire through inhibition of fuel-air mixing. Some of the commercially available products contain additives to facilitate fluidization and to render them non-hygroscopic. These powders dispersed on the surface will inhibit fuel oxidizer mixing [16]. In addition, they are generally designed either to release inert gases into the vapor-phase void fraction or to partially fuse or cake to reduce porosity. Both of these actions serve to further reduce the ability of the oxidizer to penetrate the porous mixture.

The operations of manual firefighting activities should be appropriately supported with systems that are designed for sodium aerosol environments. This may include but is not limited to suitable breathing apparatus, emergency lighting and communication equipment.

Fire Confinement and Control

Fire compartments, fire barriers or physical separations are often used to separate redundant safety related systems in nuclear power plants. This ensures that a fire affecting one division of a safety system would not prevent the safe execution of the safety function by another division [25].

In SFR, passive fire protection systems are used to mitigate the consequences resulting from a sodium spill [15]. These passive systems include sodium drainage from a fire area using a catch-pan and directing the leak by gravity and piping system to a hold-tank that excludes oxygen to support combustion. This system is also used to prevent chemical reactions of sodium pool and the concrete floor. The system typically consists of an insulated steel container (hold-tank) covered with leak collection trays equipped with drain and vent pipes. The design of the leak collection trays is based on immediate channeling of burning liquid sodium on to the funnel shaped sloping cover tray to a hold-up tank where sodium fire extinguishes due to the lack of oxygen [22]. Sodium spills on the leak collection trays will be collected through the drainpipes into the catch pans. The pipes will remain filled with sodium and the bulk of the liquid pool is isolated from the cell atmosphere. As a result, the burning pool surface area is effectively reduced to the surface area of the drainpipes where fire suppression can be effectively directed [24].

Fire compartmentation or space isolation and the use of inert gases can mitigate the effects of sodium fires. The principle of space isolation is to prevent oxygen entry into the fire area and to allow the oxygen concentration to decrease by the sodium-oxygen reaction to below levels that can sustain combustion. This requires secured compartment boundaries with minimal leaks consisting of adequate fire separations, barriers, fire/smoke dampers, fire doors etc. to be effective. Inert gas, such as nitrogen gas, is introduced to slightly pressurize the cell to prevent the in-flow of air. Adequate research and testing is required to assess barrier leak rates and the overall effectiveness of this approach.

Mitigation of Pressure and Temperature Rise in the Containment due to Fire

Sodium fires on the primary sodium circuit in the containment vessel can cause a significant increase in pressure and temperature inside the containment vessel. Therefore, the design should ensure the integrity of containment to withstand thermal and mechanical loads generated during a sodium fire. Pressure relief systems should be considered to mitigate the pressure rise. Concrete walls are insulated and provided with steel covers to mitigate temperature rise effect on concrete.

The number of penetrations through the containment should be kept to a practical minimum and all penetrations should meet the design requirements of the containment structure itself.

Limitation of Sodium Aerosol Release

Sodium aerosol compounds generated during a fire such as sodium monoxide and sodium peroxide are corrodible and can impair the safety functions of susceptible reactor components. In addition, in the case of sodium fires on the primary circuit, aerosols generated through combustion of the radio-active sodium could be a source term.

According to [23], the following measures may be effective in limiting the consequences of sodium aerosol release:

- Ventilation and dampers designed to restrict the movement of the aerosol in the building and release into the environment;
- Installation of aerosol filter installed at the outlet of the air convection path; and
- Separation of safety systems such that aerosol generated in the accident circuit will not affect other circuits.

Sodium-Concrete Reaction

Liquid sodium's reaction or interaction with concrete generates heat and hydrogen. This increases the temperature and pressure and presents an explosion or deflagration hazard in the containment. In addition, the chemical reaction can cause erosion of the concrete leading to a breach in the containment.

A series of experiments conducted at the Japan Atomic Energy Research Institute observed that the structural strength of the reacted concrete depends on the reaction temperature [26]. It was observed that the concrete was not destroyed when reacted with sodium at 400 °C. However, at 550 °C, the concrete matrix was almost completely destroyed. This was due to the decomposition of calcium hydroxide according to the study. Sandia National Laboratories and Hanford Engineering Development Laboratories [27] performed intermediate-scale tests and observed that there are major differences among the chemical interaction mechanisms and response of different types of concrete. It is further noted by the same study that the chemical reaction that occurs between the various types of concrete and sodium depends on the reacting species.

Limited experimental work has been done on finding concrete materials that do not adversary react to liquid sodium. In SFR, concrete walls and floors are typically insulated and provided with steel covers to avoid sodium-concrete contact.

CONCLUSIONS

An overview of liquid sodium fires, potential fire protection challenges, and fire protection measures associated with SFRs have been presented in this paper.

A number of notable incidents that resulted in sodium fires in NPPs and test facilities in the past have been reviewed and presented. Sodium leaks are attributed to several factors, including but not limited to the following: component design and manufacturing defects, weld and joints failure, maintenance and personnel errors, valve failures, pipe cracks, steam generator and sodium valve failure. This resulted either in a sodium spray fire, pool fire or both with intense heat, and strong aerosol production with increased smoke. It has also been noted that sodium fires increase the temperature and pressure in the containment, produce aggressive compounds, increase the corrosive burden on reactor components and cause the release of large amounts of aerosols. The fast neutrons in the core will activate sodium, causing the sodium in the primary heat transport loop to become radioactive.

This study observed that sodium's high reactivity in air and water could lead to sodium fires and explosions or deflagration that can challenge nuclear safety in SFR. As part of the design of new SFRs, a comprehensive assessment of fire and explosive hazards associated with sodium's reactivity should be performed. This will ensure that the design of the SFR employs adequate preventative or controlled measures so as to not compromise nuclear safety, damage safety related structures, systems and components and put personnel at risk.

The design of SFR should employ defence-in-depth fire protection concepts where redundant parts, diverse systems or paths, physical separation and barriers are employed to achieve a high degree of nuclear fire safety. The design should focus on preventing fire from starting, promptly detecting and extinguishing fires that start, and prevent the spread of fires that are not promptly extinguished. In addition, the design of SFR should incorporate measures that mitigate pressure and temperature rising inside containment as a result of fire, limit sodium aerosol releases and sodium-concrete interaction.

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DISCLAIMER

This paper is considered a research in progress and does not claim to be exhaustive in the area of sodium fires in SFRs. This paper represents the opinions of the authors and is the product of professional research. It is neither meant to represent the position or opinions of the CNSC nor the official position of any staff members.

REFERENCES

[1] World Nuclear Association: Fast Neutron Reactors – Updated April 2019, https://www.world-nuclear.org/information-library/current-and-future-generation/fast-neutron-reactors.aspx, retrieved July 10, 2019.

[2] Cohen, B. L.: Breeder Reactors: A Renewable Energy Source, American Journal of Physics, 51, 1, January 1983.

[3] Weinberg, A.: The Second Fifty Years of Nuclear Fission, in: Proceedings of Special Symposium: 50 Years of Nuclear Fission in Review,ONT, Canada.

[4] Berthoud, G., et. al.: Sodium-cooled nuclear reactors, Editions Le Moniteur, France, 2014.

[5] Hore-Lacy, I.: Nuclear Energy in the 21st Century, World Nuclear University Press, Elsevier Inc., London, United Kingdom, 2006.

[6] World Nuclear Association: Nuclear Power Reactors – Updated October 2018, https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/nuclear-power-reactors.aspx, retrieved July 2018.

[7] International Atomic Energy Agency (IAEA): Liquid Metal Coolants for Fast Reactors Cooled by Sodium, Lead, and Lead-Bismuth Eutectic, Nuclear Energy Series,

No. NP-T-1.6, Vienna, Austria, 2012, https://www-pub.iaea.org/MTCD/Publications/PDF/P1567_web.pdf.

[8] Generation IV International Forum (GIF) R&D: Outlook for Generation IV Nuclear Energy Systems, Generation IV International Forum, August 2009.

[9] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI): Experimental facilities for sodium fast reactor safety studies, NEA No. 6908, Paris, France, 2015.

[10] Zohuri, B., and P. McDaniel: Thermodynamics in Nuclear Power Plant Systems, Springer, New York, NY, USA, 2015.

[11] Kirillov, P. L., and N. B. Deniskina: Thermophysical properties liquid metals coolants, TsNIIAtominform, Review IPPE-0291, Moscow, Russia, 2000.

[12] Petiot, P., J. M. Seiler, and G. Cen Physical properties of sodium: a contribution to the estimation of critical coordinates, High Temperatures High Pressures, Vol. 16, pp. 289-293, Grenoble, France, 1984.

[13] Tara, J., et al.: Metal Fire Implications for Advanced Reactors, Part 1: Literature Review, Sandia Report, Sandia National Laboratories (SNL), Albuquerque, NM, USA, October 2007.

[14] Newman, R. N., and J. F. B. Payne: Burning Rates of Sodium Pool Fires, Combust Flame, 33(3), pp. 291-297, 1978.

[15] International Atomic Energy Agency (IAEA): Sodium Fires, Design and Testing, Specialists' Meeting Summary Report of the International Working Group on Fast Reactors, Richland, WA, USA, 24-28 May 1982.

[16] Tara, J., et al.: Metal Fires and Their Implications for Advanced Reactors, Part 3: Experimental and Modeling Results, Sandia Report, Sandia National Laboratories (SNL), Albuquerque, NM, USA, October 2010.

[17] Hilliard, R. K., R. P. Johnson, and D. A. Powers: United States Position Paper on Sodium Fires, Design and Testing", Specialists' Meeting Summary Report of the International Working Group on Fast Reactors, Richland, WA, USA, 24-28 May 1982.

[18] Bunz, H., S. Jordan, and W. Lindner: The behaviour of sodium fire aerosols (IWGFR--28), International Atomic Energy Agency (IAEA), Vienna, Austria,1979.

[19] Plantamp, A., et al.: Modeling of the chemical behavior of sodium fire aerosols during atmospheric dispersion, Aerosol Science and Technology, 50:7, pp. 705-716, 2016.

[20] Narayanan Anantha, R., et al.: Experimental Investigations on Carbonation of Sodium Aerosol Generated from Sodium Fire in the Context of Fast Reactor Safety, Ann. Nucl. Energy, 80, pp 188–194, 2015.

[21] Newman, R. N., G. J. Vaughan, and K. Hargreaves: Liquid Metal Fires – A Review of Work Status in the United Kingdom, Specialists' Meeting Summary Report of the International Working Group on Fast Reactors, Richland, WA, USA, 24-28 May 1982.

[22] Parida, F. C., et al.: Experimental Evaluation of Pool Fire Suppression Performance of Sodium Leak Collection Tray in Open Air, in: Proceedings of ASME International Conference on Nuclear Engineering, Volume 5: Safety and Security; Low Level Waste Management, Decontamination and Decommissioning; Nuclear Industry Forum, pp. 301-307, Miami, FL, USA, July 2006.

[23] Kubo, S., Y. Hashiguchi, and A. Okabe: R & D needs for evaluation of sodium fire consequences and aerosol behavior for DFBR (IWGFR--92). International Atomic Energy Agency (IAEA), Vienna, Austria, 1996.

[24] Foster, K W., C. Boasso, and N. N. Kaushal: Clinch River breeder reactor sodium fire protection system design and development, in: Proceedings of the Third International Conference on Liquid Metal Engineering and Technology, Oxford, United Kingdom, 1984.

[25] International Atomic Energy Agency (IAEA): Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants, Safety Guide No. NS-G-1.7, Vienna, Austria, 2004, http://www-pub.iaea.org/MTCD/publications/PDF/Pub1186_web.pdf.

[26] Mitsutsuka, N., and N. Yoshida: Japanese Studies on Sodium Fires, Design and Testing, Specialists' Meeting Summary Report of the International Working Group on Fast Reactors, Richland, WA, USA, 24-28 May 1982.

[27] Randich, E., and R. U. Acton: Intermediate-Scale Test of Sodium Interaction With Calcite and Dolomite Aggregate Concrete, Sandia National Laboratories (SNL), Albuquerque, NM, USA, September 1983.

Habitability of the Nuclear Power Plant's Main Control Room During Fire

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ABSTRACT

Habitability of the main control room (MCR) within nuclear power plants (NPPs) during fire events can significantly impact the plant operation and safety. The importance of MCR is due to the enclosure of the control and instrumentation circuit of one set of redundant safety trains that are required to safely control and shutdown the plant. Habitability analysis assesses how long the operators can remain in the MCR and proceed with actions to operate the plant safely and maintain it in safe condition under abnormal conditions including fire events. During fire events in the MCR, the longer the operators can remain in the MCR, the lower is the risk contribution to the plant from fire. The contribution to the risk of fire events in the MCR is assessed as part of the fire probabilistic safety assessment (PSA) for the plant.

This paper introduces design factors, such as room size and boundaries, type of room ceiling, fire protection systems, HVAC system, etc., that can impact the habitability of the MCR within the Canada Deuterium Uranium®[8] (CANDU) 6 NPPs during fire events. The paper presents fire modelling results, using a two-zone fire modelling tool, of the smoke layer for different design aspects of CANDU 6 MCRs and the time available for operators before abandonment of the MCR to a secondary control area (SCA). This paper also describes the challenges of operators due to environment created by the fire event. Potential design change recommendations are provided based on the sensitivity of the design factors considered.

INTRODUCTION

The CANDU 6 design consists of two separate and diverse control areas: the MCR and the SCA. Typically, the MCR contains the Group 1 equipment performing the operation and control of the plant, primary control of process and safety systems for all design basis events (DBE), testing of safety systems and carrying out the on-line refuelling. The SCA encloses the Group 2 equipment that is required when the MCR becomes uninhabitable. The controls from the SCA are limited to safe shutdown of the plant, control of heat sinks, containment and monitoring of essential safety functions. The SCA is always separated from the MCR and a safe path is ensured to be available for operators to transfer from MCR to SCA if the MCR becomes unavailable as a result of any unsafe circumstances including fire.

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[8] CANDU® (*Can*ada *D*euterium *U*ranium®) is a registered trademark of Atomic Energy of Canada Limited (AECL).

Generally, because of the unique characteristics of the MCR, a separate fire risk analysis is performed for the MCR complex fire compartment. The MCR complex fire compartment within CANDU 6 plants typically consist of the MCR and two control equipment rooms (CERs) that are sharing fire barriers and doors with the MCR. The MCR also encloses small offices. Figure 1 shows the control room simulator at the Embalse NPP in Argentina [1], one of the CANDU 6 plants.

The risk assessment of the MCR during fire events considers either loss of a control feature due to loss of a control panel (small fire) or it could propagate to induce loss of all control features and consequently loss of Group 1 due to operator evacuation as the MCR becomes uninhabitable. This paper discusses the design factors impacting the habitability of the CANDU 6 MCR during fire events.

Figure 1 Control room simulator at Embalse [1]

DESIGN FACTORS

Several design factors can impact the abandonment of the MCR. Some of these factors can be considered for new built and others can be implemented to enhance existing plants. The main factors that can impact the MCR abandonment time are:

- Size of MCR;
- Fire source and size;
- Smoke control system.

Most of the CANDU 6 MCRs are protected by smoke detectors. Some plants have installed very early warning smoke detectors such as the aspirating smoke detector (ASD) VESDA while other plants have installed smoke detectors within the main control panels to faster detect fires within the panels. In addition to these automatic detection systems, the MCR is continuously occupied, providing the capability of prompt detection (cf. [2], pp. 11-32). As such, detection of fires is considered always successful within the MCR and thus the probability of successful detection can be assumed to have a value of 1 (see [2], pp. P-6).

Installation of automatic fire suppression systems, however, is not recommended in the MCR as their auction can lead to MCR abandonment immediately or in very short time.

Delay associated with manual fire suppression can impact the fire propagation within the MCR. Manual suppression is assessed based on the capabilities of the emergency response team (ERT), available manual suppression equipment and the plant emergency procedure. The manual suppression is assessed during the fire PSA task through the non-suppression probability for the purpose of fire risk quantification. The probabilities are calculated for a specific time when the fire propagation can induce additional damage within an area.

Size of the MCR

The typical size of existing CANDU 6 MCR is approximately 15 m width and length and 4 m height. As a means to improve the time to evacuate the MCR, this size can only be considered for a new build.

The CANDU 6 main control board separates the MCR and CERs and there are no physical barriers between these rooms. As such, the back of the main control board panels can be seen in the CERs. Fires starting in the main control panels can impact both the MCR as well as one of the CERs depending on the configuration of the panels. The concern in this paper is the environment of the MCR during fire.

Fire Source and Size

Fire sources within the MCR are mainly due to ignition of the main control panels. This is in addition to ignition of transient combustibles that can occur at any location on the floor. Transient combustible fire can either be limited to burning of transient combustibles or it can propagate to the main control panels. The focus of this paper is only on the main control panel fire. As per guidance of NUREG-2178 [3], factors that impact the size of the fire are:

- Size of the panel (small, medium or large);
- Panel contents (default, low and very low fuel);
- Enclosure ventilation (open/closed doors);
- Type of Cables (thermoset and thermoplastic); and
- Panel separation (single and double wall).

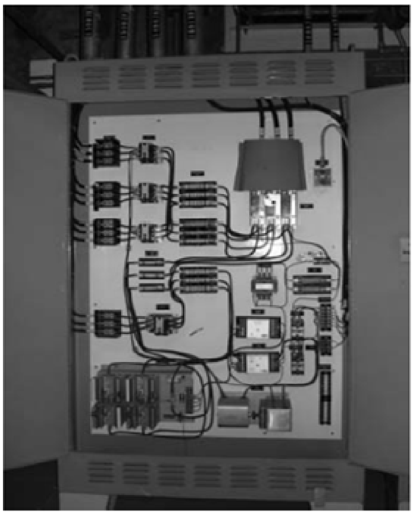
NUREG-2178 [3] includes a general discussion for different types of electrical enclosures. For the main control panels considered in this paper, the size of the CANDU 6 main control panels varies from medium ($\leq 1.42\ m^3$ and $> 0.34\ m^3$) to large ($> 1.42\ m^3$) and the panel content is normally low or very low. An example of low and very low panel content is shown in Figure 2. Panel content discussed in this paper is limited to the low content since very low content generates relatively small fires.

The guidance of [2] also presents different types of cables including thermoset (TS), qualified thermoplastic (QTP), unqualified thermoplastic (TP), and synthetic insulated switchboard wire or XLPE-insulated conductor (SIS). For the purpose of the assessment in this paper, only TS and TP are considered since this represents the type of cables within the CANDU 6 plants. The intensity of the fire source in this paper is represented by the 98th percentile of the heat release rate (HRR) and is presented in Table 1 (from [2], Table 4-2).

Table 1 HRR for large and medium panels with low fuel content

Panel Size	Panel Ventilation (Open/Closed Door)	Type of Cables (TS/TP)	Low Fuel Content
Large	Closed	All	200
	Open	TS	350
	Open	TP	500
Medium	Closed	All	100
	Open	TS	150
	Open	TP	150

As discussed before, the CANDU 6 control board consists of control panels that are separated by either single or double wall. According to the guidance of [2], pp. 11-37. Fire can propagate to directly adjacent panel(s) within 15 min, if the panels are separated by a single wall. As such, after 15 min as many as three (or two for end panels) panels can represent the fire intensity, if the fire suppression activities failed to suppress the fire within 15 min. Cabinets with a double wall separated by an air gap can damage cables within immediately adjacent panel(s), but fire does not propagate (cf. [2], pp. 11-38). Therefore, the fire intensity is limited to a single panel.

a) 

b) 

Figure 2 MCR fire source –
a) Low fuel loading; b) Very low fuel loading (from [2], pp. 3-19 and 3-20)

Smoke Extraction System

The CANDU 6 MCR ventilation system normally operates in two modes, normal and fire modes. Typically, upon receiving fire signals, the ventilation system will shut off. Dampers will close to isolate the room in which the fire originates. Exhaust fans will continue to run to delay the rate of descent of the smoke layer in order to delay the abandonment of the MCR. This rate of smoke extraction is an important design factor.

FIRE SCENARIOS

Scenario Description

Assessment of the design factors mentioned in the previous sections is achieved by postulating potential fire scenarios that can occur in the MCR. The scenarios are then modelled using the computer code CFAST (*C*onsolidated Model of *F*ire Growth *a*nd *S*moke *T*ransport) [4], Version 6.0.10. Fire scenarios within the CERs are not part of the scope of this paper. A CFAST Schematic of the MCR complex is depicted in Figure 3. A Summary of the scenarios modelled is presented in Table 2. The first scenario is the basic case with the MCR size of 15 m x 15 m x 4 m. The fire source is a large (> 1.42 m^3) control panel that is separated from immediate adjacent panels by double wall with air gap. The content of the panel is low. No smoke extraction is considered for the base case.

Fire scenario FS-7 considers smoke extraction. The extraction opening centre height is assumed to be at 3 m above the floor and extraction flow is 0.4 m^3 per second. Boundaries (including floor and ceiling) of the rooms are assumed to be gypsum board. Panel fire growth is modelled as linear for the first 12 min to the peak and then steady for other 8 min. It is likely that the CANDU 6 MCR would have doors to the CERs. These doors are normally open due to operational needs. Therefore, for this study the doors will be considered open at all times.

Table 2 Fire scenarios summary

Fire Scenario	Design Factor	Scenario Description	HRR [kW]	MCR Size [m x m x m]	Smoke Extraction Rate [m^3/s]
FS-1	Base case	Fire is limited to large MCR closed panel with low fuel content (all types of cables).	200	15 x 15 x 4	NA
FS-2	Room size	Fire is limited to large MCR closed panel with low fuel content (all types of cables). Panels are separated by double wall with air gap.	200	15 x 15 x 8	NA
FS-3	Panel size	Fire is limited to medium MCR closed panel with low fuel content (all types of cables).	100	1 5x 15 x 4	NA
FS-4	Panel ventilation (closed/open door) – TS cables	Fire is limited to large MCR open panel with low fuel content and TS cables.	350	15 x 15 x 4	NA

Fire Scenario	Design Factor	Scenario Description	HRR [kW]	MCR Size [m x m x m]	Smoke Extraction Rate [m^3/s]
FS-5	Panel ventilation (closed/open door) – TP cables	Fire is limited to large MCR open panel with low fuel content and TP cables.	500	15 x 15 x 4	NA
FS-6	Panel separation (single wall)	Fire is starting in large MCR closed panel with low fuel content (all types of cables). The panel is separated by single wall. Fire propagates to two directly adjacent panels after 15 min.	200 for the first 15 min and 600 min afterwards	15 x 15 x 4	NA
FS-7	Smoke extraction	Fire is limited to large MCR closed panel with low fuel content.	200	15 x 15 x 4	0.4

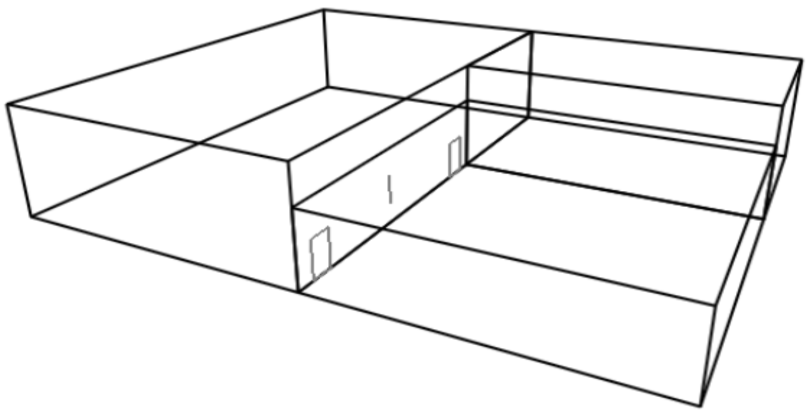


Figure 3 CANDU 6 MCR complex

Abandonment Criteria

Risk assessment of the MCR during fire events varies from other areas within the plant; this is because:

- The control and instrumentation circuits of all redundant trains for almost all Group 1 systems are present in the MCR.
- The MCR is continuously occupied, warranting credit for prompt detection and suppression.
- Habitability of the room is an important factor since the plant safety depends on the well-being of control room operators.

NUREG/CR6850 (see [2], pp. 11-40) has recommended that the abandonment timing estimate of the MCR be based on the habitability conditions of the MCR. At least one of the following criteria shall be met to determine the timing of abandonment of the MCR:

- The heat flux at 1.80 m (6 ft) above the floor exceeds 1 kW/m^2 (relative short exposure). This can be considered as the minimum heat flux for pain to skin. Approximating a smoke layer of around 95 °C (200 °F) could generate such heat flux.
- The smoke layer descends below 1.80 m (6 ft) from the floor, and optical density of the smoke is less than 0.3 m^{-1}. With such optical density, a light-reflecting object would not be seen if it is more than 0.4 m away. A light-emitting object will not be seen if it is more than 1 m away; or
- A fire inside the main control board damages internal targets 2.13 m (7') apart.

The last criterion is not applicable to the CANDU 6 plants since their control board consists of panels that are separated by either single or double panel walls. Considering the first two criteria, the output required for the modelled cases is the temperature of the room, the smoke layer height and the optical density.

In this paper, the abonnement time is when the room temperature of the MCR reaches 95 °C or if the smoke layer/hot gas layer (HGL) reaches 1.80 m above the floor. At this height of the HGL it also assumed that the visibility is less than less than 0.3 m^{-1}.

Fire Modelling Results

The seven fire scenarios discussed in the previous section have been modelled using CFAST version 6.0.10 [4]. The time-temperature relationship due to fire in the MCR is presented in Figure 4. The rate of descend of the HGL above the floor is depicted in Figure 5.

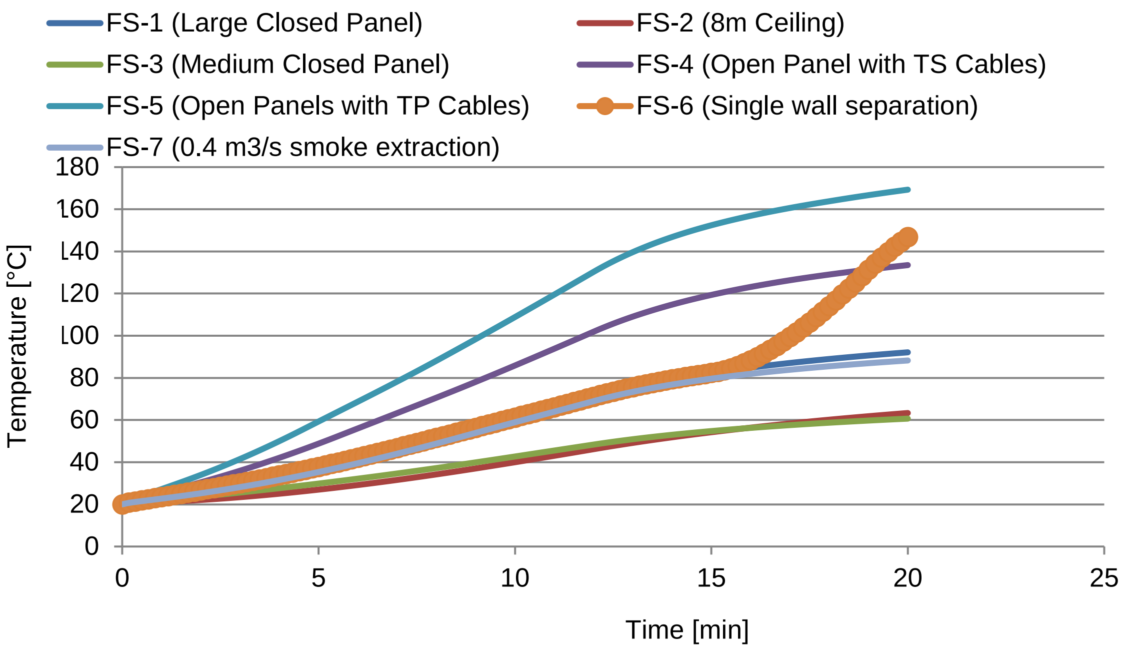


Figure 4 Time-temperature relationship due to fire in the MCR

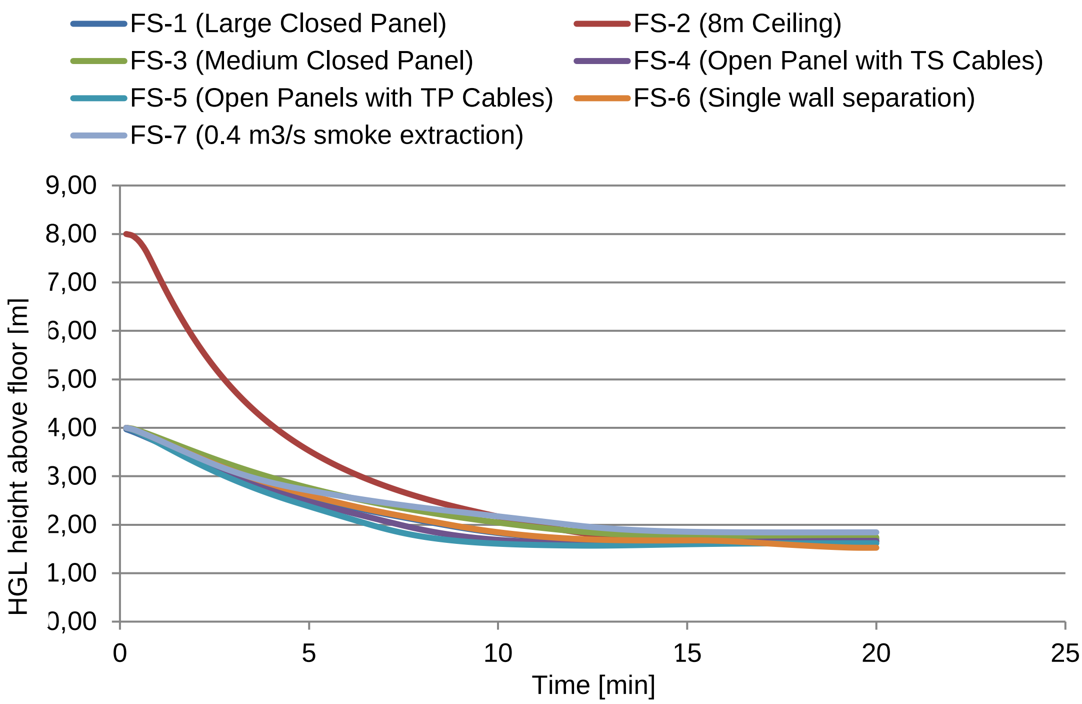


Figure 5 HGL descend rate due to fire in the MCR

Seven fire scenarios have been assessed using CFAST [4]. A summary of the evacuation time as a result of the MCR temperature reaching 95 °C or the HGL reaching 1.8 m is presented in Table 3. As shown in the table, the HGL is the basis for evacuation in all examined cases. That is, evacuation is expected to occur for all scenarios before the temperature can reach the criterion of 95 °C.

The most important design factor that can improve the habitability of the MCR is smoke extraction system. With 0.4 m^3 per second smoke extraction rate, the MCR evacuation can be eliminated if the fire source is limited to 200 kW. The shortest available time to evacuation is associated with the scenario (FS-5) that involves an open back panel with TP cables. It should be noted that the assessment discussed in this paper considers one factor at the time. Abandonment time could be shorter for scenarios in which more than one factor is applied, for example for a scenario using smoke extraction with open back panels that contain TP cables.

Abandonment time for the scenario considering panels separated by single wall (FS-6) does not vary from the base case since the evacuation occurs before the 15 min when the fire size increases. However, the base scenario is less severe than FS-6 in which only one control panel ignites whereas for FS-6 three control panels ignite, meaning that in the FS-6 scenario more control features are lost.

Table 3 MCR abandonment time in minutes

Fire Scenario	Design Factor	Abandonment Time due to Smoke Layer at 1.80 m above Floor	Abandonment Time due to 95 °C Temperature
FS-1	Base case	10	NA
FS-2	Room size	13	NA
FS-3	Panel size	13	NA
FS-4	Panel ventilation (closed/open door) – TS cables	9	11
FS-5	Panel Ventilation (closed/open door) – TP cables	8	9
FS-6	Panel separation (single wall)	10	17
FS-7	Smoke extraction	NA	NA

CONCLUSIONS

This paper addresses the impact of design factors on the habitability of the CANDU 6 MCR during fire. The environment induced by fire can lead to operators evacuating to the SCA if the MCR becomes uninhabitable. Based on the criteria presented, the MCR becomes uninhabitable if the temperature of the room reaches 95 °C or, if the HGL reaches 1.80 m above the floor, whichever occurs first. The assessment has shown that the most significant design factor that can improve the time to evacuation is the smoke extraction system.

General design recommendations for the MCR design to decrease the risk contribution from fires within the MCR are:

1. For new designs, consider MCR size;
2. Include a smoke extraction system;
3. Control panel is the primary fuel within the MCR, this fuel can be reduced by using:
   - Standalone panels that are separated by a double wall with an air gap;
   - Panels with thermoset cables;
   - Using low or very low amount of cables;
   - If possible, using smaller panel size (medium panels vs large panels); and
   - Using closed back panels;
4. No automatic fire suppression system.

REFERENCES

[1] Web reference, https://www2.l3t.com/mapps/en/media-releases/2015-07-16.html.

[2] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 (EPRI 10191989), Washington, DC, and Palo Alto, CA, USA, 2005, https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/.

[3] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI): Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, NUREG-2178 (EPRI 3002005578), Washington, DC, and Palo Alto, CA, USA, April 2016, https://www.nrc.gov/docs/ML1611/ML16110A140.pdf.

[4] Peacock, R. D., et al.: CFAST – Consolidated Model of Fire Growth and Smoke Transport, Version 6, National Institute of Standards and Technology (NIST), Gaithersburg, MD, USA, August 2005.

Escape Route Concepts in Nuclear Power Plants
– Challenges due to Specific Facility Design –

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ABSTRACT

Although fire safety is one of the safety fundamentals of nuclear facilities, fire safety is not limited to this. This is due to the fire phenomenon's nature capable of jeopardizing human life and the property. As this risk is present in the everyday life, many laws, codes, standards and guides dealing with fire safety do exist. The governing of these laws, codes, standards, guides (norms) however mainly focus on human life safety and the protection of the property rather than to nuclear specific demands. There intrinsic request on safety is similar but huge differences in their execution exist from country to country. This however creates a disaster for plant suppliers facing difficulties when transferring designs between different countries.

Escape routes are one of the major challenges in the design of special facilities such as nuclear power plants. This is due to the different legislative requirements which are generally expressed by design solutions rather than as intrinsic requirements, and furthermore being created for non-nuclear facilities. Nuclear specific aspects are typically not addressed in these common standards (security or radiological aspects) or are contrary to these.

Human life safety jeopardized by fire is generally a function of fire exposure and its aligned effects. A unique limiting value however cannot not be defined, as the fire exposure is some kind of "virtual" due to its dependency of multiple variables and accompanying uncertainties.

Safe or not safe remains the question!

Typically, the level of safety is expressed by an escape route travel length for reaching a "safe" area. It is however a question which maximum travel length is acceptable by considering nuclear specific requirements and designs, which performances a safe area shall meet, and which nuclear specific designs can be credited.

INTRODUCTION

Each individual relies on safety. Safety however is hypothetical and relative. Therefore, it cannot be uniquely characterized by a value. It is rather a feeling of a cost benefit relation of each individual. Since the behavior of an individual may influence the safety of another individual or a larger community, certain safety standards (norms) are specified by legislators. These safety standards however can only be design solutions rather than universal design approaches due to their hypothetical nature.

Socio-politically the presence of conditions presenting large negative consequences or deemed as those are generally not accepted and are expressed as "unsafe".

The safety sentience relies on a risk-informed approach which describes a relation between the occurrence frequency of a certain event and its assigned size of "loss". Therefore, the safety sentience principally constitutes a cost benefit relation. It is a question of a threshold of acceptance whether an event is considered to be safe or unsafe. This is generally based on the personal acceptance of each individual.

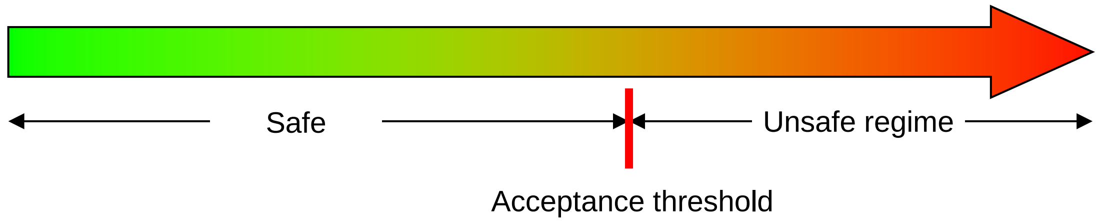

Figure 1 Safety sentience approach

The acceptance threshold is an emotional sentience, therefore, the term "safe" cannot be characterized by a distinctive value.

Fire is a phenomenon jeopardizing personnel and plant safety. That is the reason that it becomes significant, i.e., because of the potentially huge consequences. Fire is a natural phenomenon of large complexity. Therefore, the safety demonstration is subject of uncertainty and thus fire safety is underlined by the probabilistic risk-informed approach.

Principally, fires can occur anywhere and at any time where combustibles are present. As a consequence, the number of fire scenarios must hypothetically be deemed as unlimited. The challenge in fire safety engineering is to identify fire scenarios that are assumed potentially to occur and to analyze if the existing acceptance threshold is exceeded.

However, the determination of the acceptance criteria is no personnel opinion of a single individual, it is the common opinion or understanding of a larger group of individuals and the basis for codes and guidelines. Their fundamental safety objective can roughly be stated as "make it safe".

The Sample Building Code of Germany [1] requires that structural installations shall be arranged, erected, modified and maintained in such a way as to prevent the occurrence of a fire and the spread of fire and smoke (spread of fire) and to enable the rescue of people and animals and effective extinguishing work in the event of a fire. With respect to the safe escape it might be formulated as follows, see [2]: *"It must be possible to evacuate a building safely in case of fire or any other emergency. A building shall be provided with an adequate number of appropriately located exits which are sufficiently spacious and easily passable, so that the time to evacuate the building will not be as long as to cause danger."*

This however doses not describe the required level of protection due to the hypothetical nature of safety, although it describes the essential needs. The required level of protection is therefore characterized by predefined values and design solutions, which however provide only a basic idea as the fundamental safety objectives have to be archived.

The fundamental safety objective specifies the "red line" to which ideally all design solutions or requests are oriented to. It is however noted that a large range of those norms seems to be a collection of self-standing design solutions rather than be based on a close design approach targeting to the fundamental request. This is challenging for designers as most of these detailed design solutions become "chiseled in stone".

Principally there is nothing against applying design solutions; but care has to be taken that the fundamental safety objectives are still met. Applying codes and standards "mechanically" is not the right way for safety engineering. There still exists a potential where situations fully comply with predefined values but fail in achieving the fundamental safety objectives!

As a first approach, however, the application of general rules and standards provides a general basis. This is because they indirectly describe the required level safety. Moreover, it also simplifies the design and approval process.

ESCAPE ROUTE CONCEPTS

Fire may strongly affect human life safety, thus escape routes and other fire and safeguards shall be provided throughout the construction. Because human life safety during fire events is a combination of early warning, appropriate escape routes, and fire and smoke confinement, the infrastructure of the escape routes become significant. Escape routes shall provide proper escape of personnel but also access for firefighting and rescue services.

Therefore, escape routes shall be designed and allocated such that safe scape of persons to the outside (final exit), proper access for firefighting and rescue services as well as safe passage of personnel performing nuclear safety operations (e.g., safe passages between main control room and remote shutdown panel) are provided.

Regular (non-nuclear specific) building codes however do not consider requirements specific to nuclear power plants, such as requirements with respect to security and confinement. The reason simply is that those norms and standards are not made for this specific purpose, they describe only features of “commonly known” buildings. This is challenging as due to security and radioactivity confinement constraints the openings and therefore the means of egress are rather limited following the “as low as reasonably achievable (ALARA)” approach.

Comparing a nuclear power plant to those types of buildings typically characterized in the non-nuclear building codes, a power plant represents high-rise buildings having egress features which are similar to underground storeys.

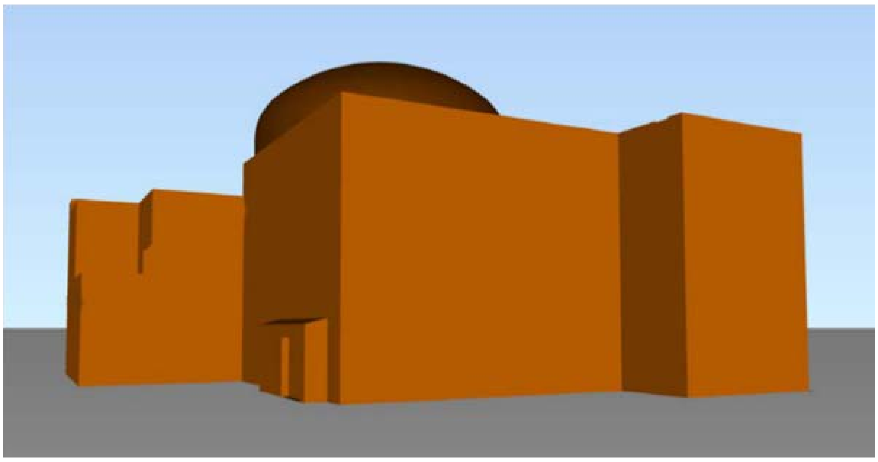

Figure 2 Outer view of a typical Nuclear Island

Therefore, typical escape route designs cannot be applied one by one, they have to be modified accordingly. In order to capture the essential needs aligned to safe egress it is necessary to figure out their basic performances.

Escape routes can generally be subdivided into the following three distinctive parts:

- the access to the exit,
- the exit itself, and
- the exit discharge.

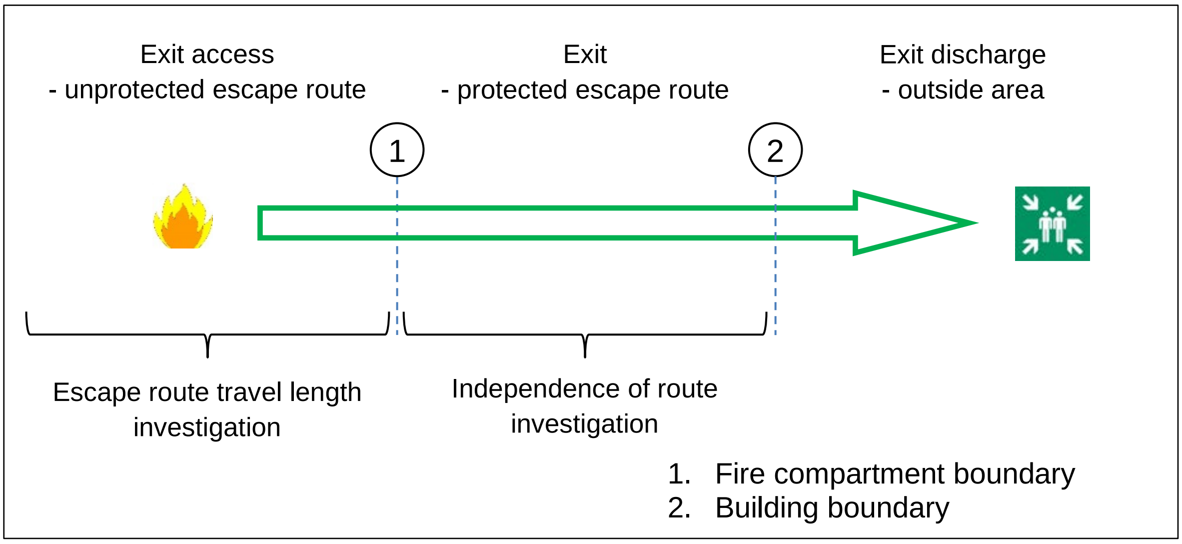

Figure 3 Escape route subdivision

Escape route are generally all passable routes from any point of the construction and/or building to outside where it finally ends. The exit access is determined as that part of an escape route which has to be traveled until the exit access door/gate is reached. It is in principle the route through the area which is affected by a fire in a given situation. Fire fundamentally affects human life safety by its by-products, such as heat and other products of combustion, in particular smoke which is toxic, caustic and obscures visibility. Therefore, the fire exposure to humans has to be limited to provide a means for escaping. The exposure (dose) however is a function of many parameters such as toxicity capability, flowrates, and dwell time, accompanied by huge uncertainties; thus, it can finally only be expressed by a travel distance and its size.

The exit in turn determines the part for the escape route which is not affected by the fire and constitutes therefore the protected part of the escape route. This is also the case for the exit discharge which is outside the construction or building. The exit is that part of an escape route which connects exit access and the exit discharge.

Several building codes explicitly specify staircases and doors to the outside as exits for which the maximum acceptable travel distance is determined. To the outside this is true, but staircases however do not meet the performances of exits by their general function which is providing egress between different storeys. Moreover, these staircases are enclosed by fire resistant barriers preventing the propagation of fire and smoke to the unexposed side.

The staircase is used as synonym of an area protected from the effects of fire. This is however misleading as the protection function is provided by the enclosing fire barriers creating a fire compartment containing the staircase, which becomes critical as it discards the compartmentation from the escape route concept. In consequence, further staircases might need implementation in the plant design.

Whether or not a part of an escape route is assigned to the exit access or to the exit depends on the particular situation. Generally, the interface between exit access and exit is at the fire compartment boundary, because only such boundaries are effectively able to limit fire and smoke propagation to adjacent areas outside the fire compartment. However, fire compartment boundaries cannot per se be determined as the passage towards the “protected area”. It is more a question of the independency from the particular fire. In principle it is determined as that point where alternative routes exist, and people will not be trapped by any fire.

Number and size of the required exits focus on the following three main items:

– the number of people to be confined (in a protected area),
– the travel speed along the escape route, and
– the escape route travel length.

If there is only one exit, there is a possibility that people which are not able to bypass the fire area will be trapped (see Figure 4). Such situations typically exist for single rooms or rooms arranged one after the other, aisles between rows of cabinets, catwalks, etc.

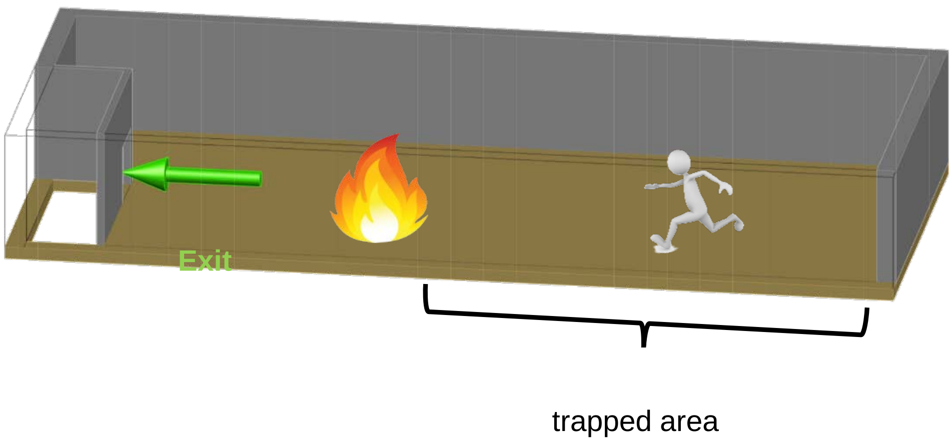

Figure 4 Trapped area at single exit arrangements

In order to limit the risk that a larger number of occupants will probably be trapped in a fire situation, further appropriate located exits are required (cf. Figure 4, according to [3].

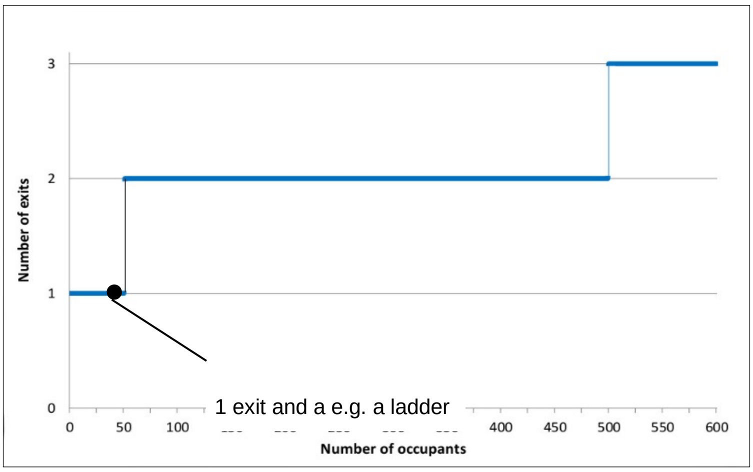

Figure 5 Number of exits (example)

The traveling speed is a function of the occupancy density (occupants per floor area). By trend the traveling speed is reduced the higher the occupancy load is (cf. Figure 6, according to [3]. Therefore, the minimum acceptable size of an escape route is determined based on the number of occupants present inside the evacuation area and is accumulated along the route.

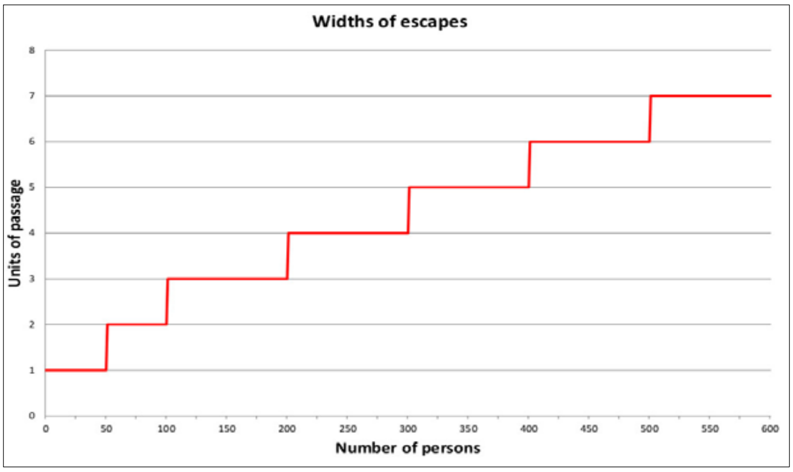

Figure 6 Sizes of exits (example)

For areas having one single exit this is true as all occupants will use this particular exit. Since the areas are typically limited in their size it can be assumed that all occupants will be at the same time at the exit of the particular area (prevention of bottleneck).

Figure 7 Bottleneck of an escape route

For areas having more than one exit the approach includes some type of uncertainty because the occupants are arbitrary distributed to the exits.

ESCAPE ROUTE CONCEPTS OF NUCLEAR POWER PLANTS

Regular building codes do not consider the specifics of nuclear power plants, therefore escaping concepts in nuclear installations deviate from typical non-nuclear industrial designs. The main stakeholders which influence these concepts are security and confinement requirements. This becomes visible by the building exits number of which is extensively limited as e expected for such large buildings. The doors which typically provide exits from the staircases to the outside constitute per definition a "weak point" in the outer building shell. Since the ALARA approach is used the number of such doors is limited to the minimum.

Moreover, the partial or total evacuation is not limited to fire incidents although the escapes are designed for this. Evacuation might also be necessary for security, radiological or other health and safety reasons. Therefore, assembly points might also be provided inside the building complex. The main advantages of this approach are:

- Maintaining radiological confinement;
- Maintaining the buildings security integrity;
- Simplifying completeness checks of personnel;
- Simplifying operational aspects, such as
  - security check of re-entering personnel and
  - radiological entry check of re-entering personnel.

An example of a Nuclear Island (NI) evacuation concept is shown in Figure 8.

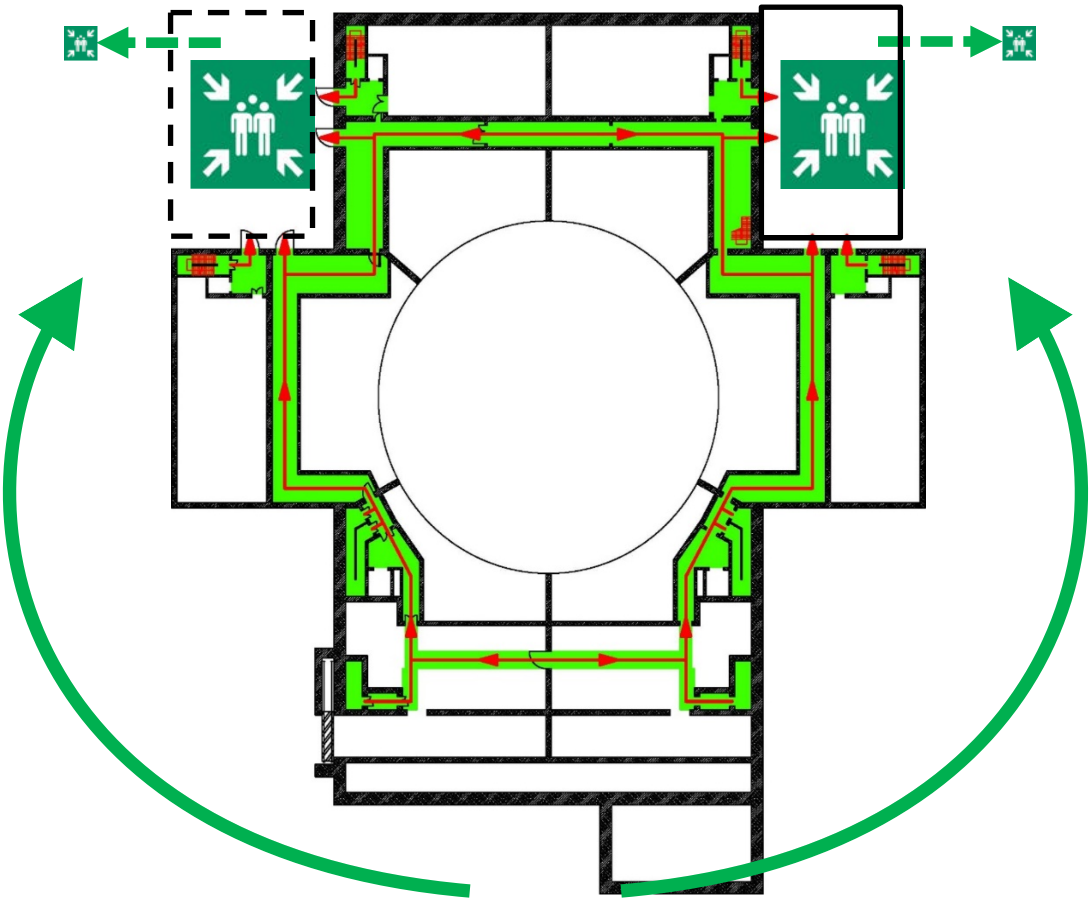

Figure 8 Nuclear Island (NI) evacuation concept

In the concept shown in Figure 8 the evacuees of the exit level and from the staircase escape to the main building exits by the passageway which interconnects the entire buildings. This provides an alternative means to egress to the main building exits; thus a single fire event will not trap all evacuees inside the building complex. One or two assembly point are located in front of the main exits inside the building complex.

As shown in the figure, there are quite long distances to the assembly point. Credit is taken from the compartmentation which provides sufficient protection from the effects of fire. It has however to be checked if an alternative means of escape does exist, which is not affected by fire.

As described above (see chapter “Escape Route Concepts”), some building codes specify staircases and the outside area explicitly as areas protected from the effects of fire. No credit is taken form the compartmentation although the protection function of staircases is based on this. This is problematic for nuclear power plants as a high degree of compartmentation is implemented. It is however noted that some building codes consider compartmentation in the escape route concept (e.g., the German Building Code for Industrial Facilities [4]).

Assessment of Escape Route Exits and Escape Route Travel Distances

As explained in the chapter “Escape Route Concepts”, the exit access is determined as that part of an escape route which has to be traveled within a limited distance, until an area which is not affected by a fire in a given situation (exit) is reached. Therefore, it is necessary to identify those fire doors which meet the criteria of an exit. Since a fire can occur in any location of the plant where combustibles are present certain fire scenarios are possible which have to be analyzed case-by-case and superposed for the final assessment.

As simple example of the analysis is shown in Figure 9.

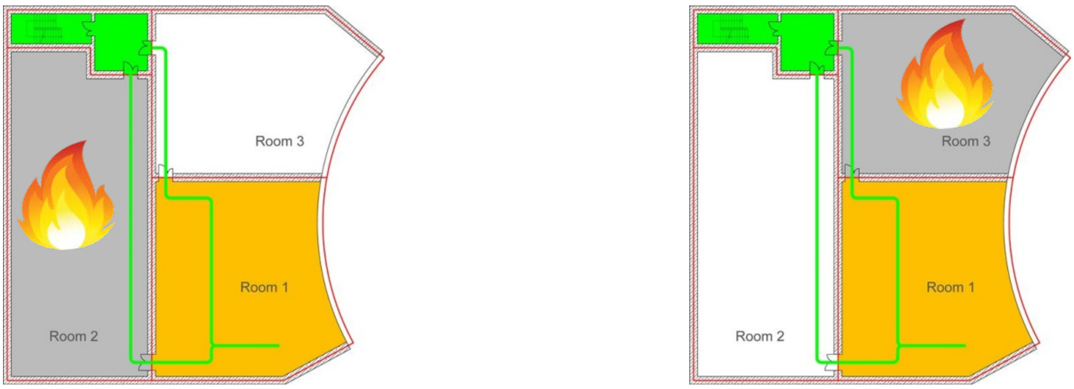

Figure 9 Analysis of exits (example 1)

The example as shown in Figure 9 consists of three rooms and a single staircase. Each of them is forming an own fire compartment. The staircase is assumed to be free of fire loads, thus no fire is considered inside the staircase. As the fire can occur in any of these compartments three fire scenarios result. For the example analysis the escape route travel length from room 1 shall be assessed. Therefore, it is of interest if the fire doors to the adjacent fire compartments meet the criteria of an exit. Two escape routes (green lines) exist to the staircase passing through adjacent fire compartments.

In the left scenario shown in Figure 9 a fire is assumed in room 2. Escape can take place through room 3 (as a separate fire compartment) which not affected by the fire; thus an alternative route exists. The same is valid for a fire inside room 3. Therefore, the fire doors to room 2 and room 3 meet the criteria of an exit. The escape route travel length from room 1 to these exits can be measured.

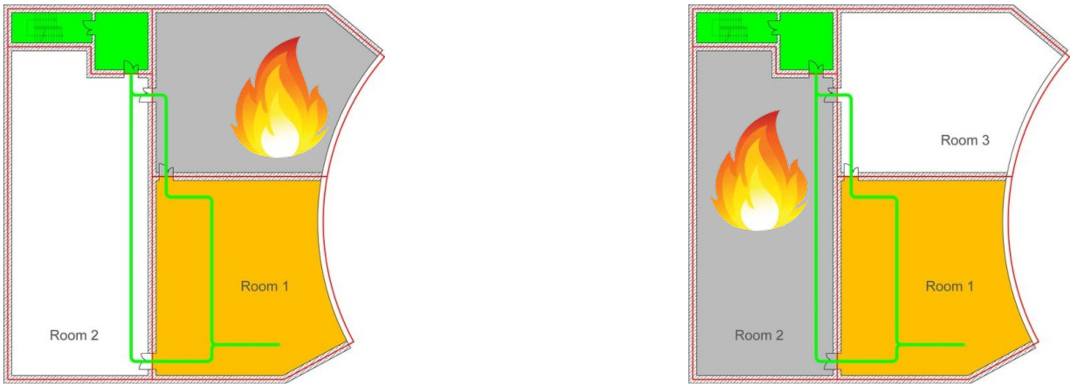

Figure 10 Analysis of exits (example 2)

The scenario shown in Figure 10 is slightly different. The door from room 3 to the staircase has been shifted to room 2. In the event of a fire in room 3 an alternative escape route from room 1 via room 2 to the staircase does exist, but in case of a fire in room 3 no alternative escape route exists. Therefore, the fire doors from room 1 to room 2 or room 3 do not meet the criteria of an exit. Therefore, the escape route travel length to the exits to the staircases has to be measured. The result would be the same if room 2 and room 3 are included in one fire compartment.

This shows that even small modifications result is huge effects on the escape route concept. An identical check has to be performed at the exit level of the staircases in order to provide safe egress to the assembly point of the plant as shown in Figure 11.

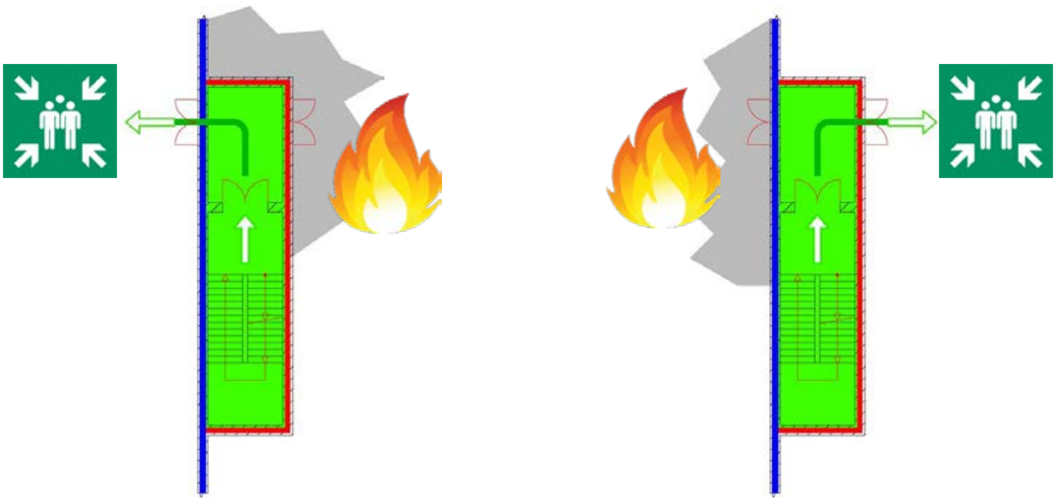

Figure 11 Alternative escape from the exit level

ESCAPE ROUTE ASSESSMENT SCHEME

An assessment scheme for the investigation of an escape route with regard to the exits is described in the following. This scheme describes a method for identifying exits; assessing escape routes travel distances and their independence. The capability of the scheme is:

– identifying escape routes leading to an exit,

– identifying exits,
– identifying those parts of means of egress (escape route) to which a limitation of the escape route travel distance is set.

Boundary Conditions and Assumptions

As boundary condition it is assumed that the fire compartment containing the fire is entirely affected by the fire, and the fire compartment not containing the fire provides sufficient protection from the fire effects.

Methodology

As first step for each room {N} it has to be defined which further room(s) {N+1} can be accessed (N → N+1).

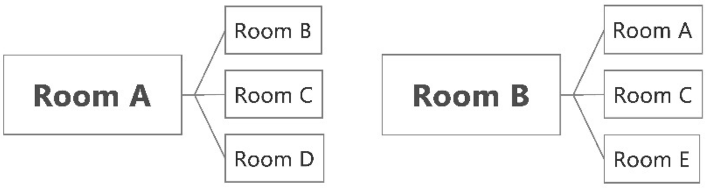

Figure 12 Room connections (example room A and B)

In the second step this approach needs to be continued for all further rooms {N+1 → N+2 → ... → N+x} (cf. Figure 13). Only those routes have to be identified which terminate at the exit of the storey. Those routes which enter a room twice (red colored) along its route must be discarded. Figure 13 shows the escape routes from room A to the exit which is room E (green) in this example. The rooms indicated in red are accessed twice in its route.

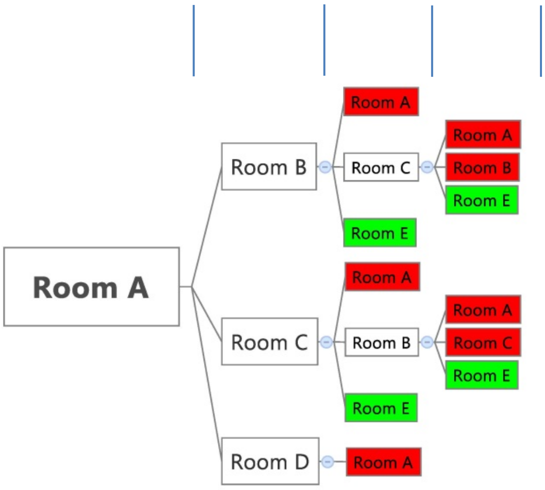

Figure 13 Possible escape routes (example room A)

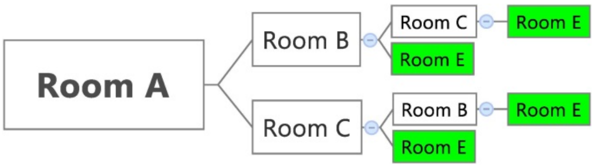

Figure 14 Escape routes leading to an exit (example room A)

Therefore, four routes to the storey exit do exist as shown in Figure 14. Details are provided in the following Table 1.

Table 1 Possible escape rooms from room A

Route 1:	room A	→	room B	→	room C	→	room E
Route 2:	room A	→	room B	→	room E		
Route 3:	room A	→	room C	→	room B	→	room E
Route 4:	room A	→	room C	→	room E	→	

These routes will be used for the further assessment. The next step of the investigation focusses on the travel length until a safe area is reached (exit access travel distance). It is therefore of interest at which location the escape route enters an area where the remaining route becomes independent from the effects of the given fire scenario (exit). Therefore, each of the rooms along the escape routes will be assigned to the fire compartment to which the room is associated to. Then the different escape routes will be checked regarding their availability applying different fire scenarios (i.e., fire inside different fire compartments).

Sequentially, a fire inside each of the fire compartments is assumed (except inside the storey exit). This includes the fire compartment to which the investigated root is assigned as well as the remaining fire compartments. In case of the example it is assumed that each room is a single fire compartment thus three fire scenarios affect the escape routes (fire inside room A, or room B, or room C).

In a next step, each route is investigated regarding the location at which the remaining root did not include a room assigned to a fire compartment where the fire is postulated (fire in a dedicated fire compartment).

As shown in Table 1 in the event of a fire inside room B, route 4 is not affected by the fire. In the event of a fire inside room C, route 2 is not affected by the fire. Therefore, the escape route length is that length from the remotest point inside room A until it’s exits door (scenario fire inside room A).

EXAMPLE INVESTIGATION (SCHEME)

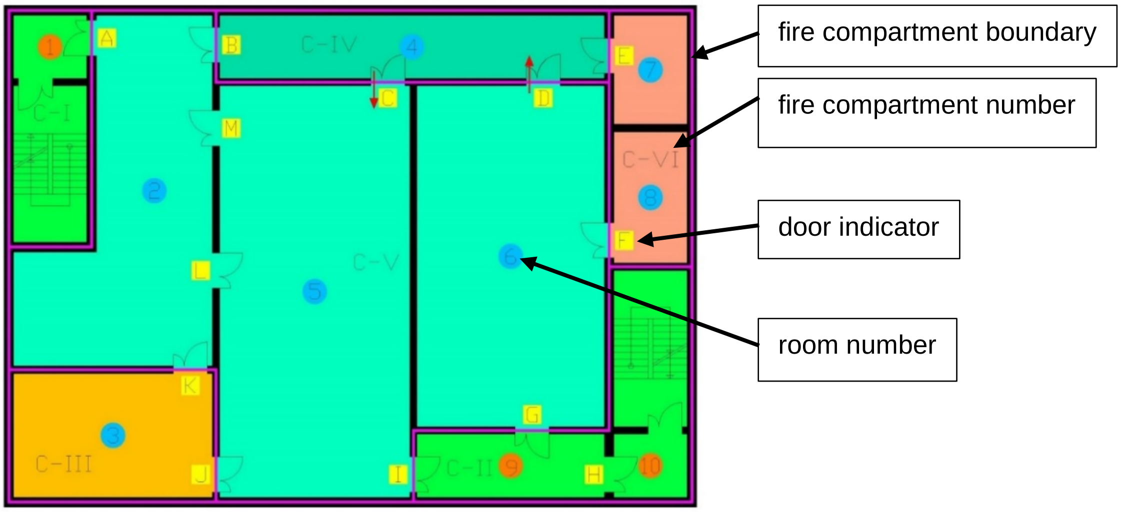

Figure 15 Example configuration

Note: Doors C and D are accessible in one direction only (in the direction of the red arrow).

Determination of the Room Connections (Routes)

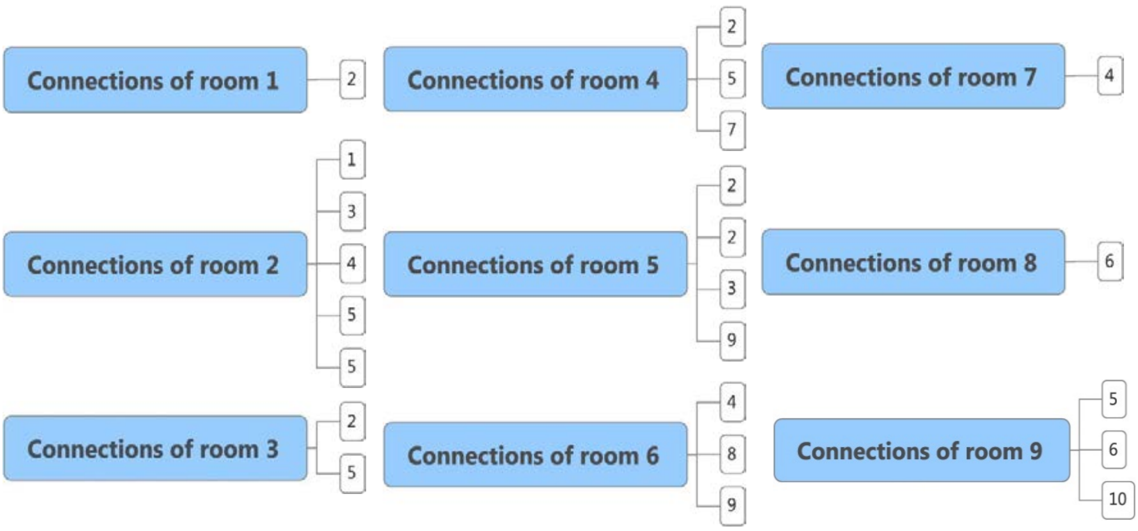

Figure 16 Room connections (routes)

Assignment of the Rooms to the Respective Fire Compartments

Room 1 is assigned to fire compartment C-I which meets the performance requirements of an exit.

Room 3 is assigned to fire compartment C-III.

Room 4 is assigned to fire compartment C-IV.

Room 3, room 5 and room 6 are assigned to fire compartment C-V.

Room 7 and room 8 are assigned to fire compartment C-VI.

Room 9 and room 10 are assigned to fire compartment C-II which meets the performance of an exit.

Example: Escape Routes from Room 7

In the following, the escape route from room 7 is investigated exemplarily. Therefore, all route from room 7 to the story exits (room 1 and room 2) are determined by successively adding the connected rooms {N+1 → N+2 → ... → N+x}. A kind of root diagram results as shown in Figure 17.

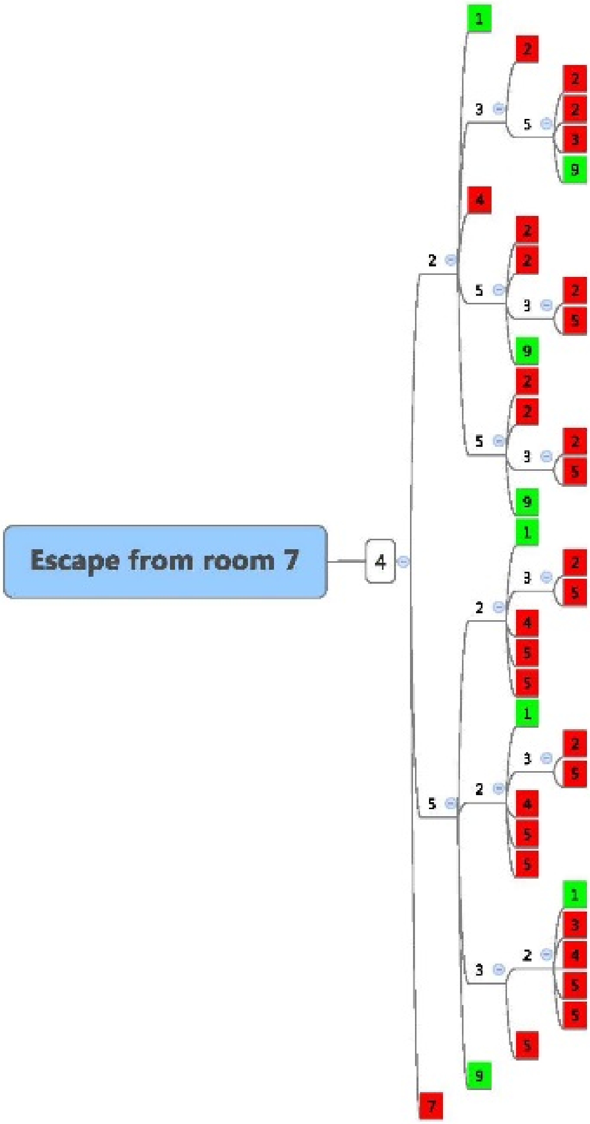

Figure 17 Routes from room 7 (all routes)

Those routes which enter a room twice (red colored) along its route are discarded. The resulting routes which terminate at the story exits room 1 and room 9 are shown in Figure 18.

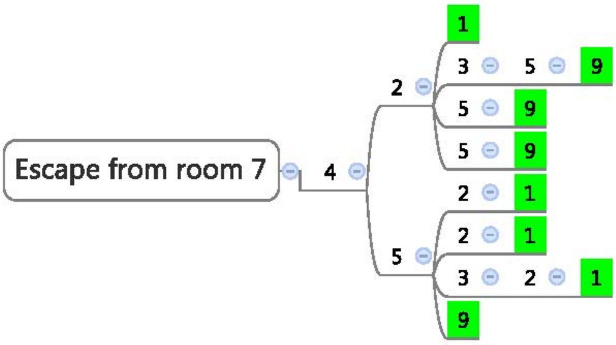

Figure 18 Escape routes from room 7

For a more handy assessment the different routes are displayed in a table format (cf. Table 2).

Table 2 Escape routes from room 7

Route		**Start**	**Rooms along the route***										
1		7	→	4	→	2	→	1					
2		7	→	4	→	2	→	3	→	5	→	9	
3		7	→	4	→	2	→	5	→	9			
4		7	→	4	→	2	→	5	→	9			
5		7	→	4	→	5	→	2	→	1			
6		7	→	4	→	5	→	2	→	1			
7		7	→	4	→	5	→	3	→	2	→	1	
8		7	→	4	→	5	→	9					

* Arrows principally represent doors between the different rooms

Investigation of Fire Scenarios

The next step of the assessment regards the investigation of the different fire scenarios and its influence on the different escape routes. It is assumed that a fire will affect all rooms of the concerned fire compartment. There are four fire compartments in which a fire is postulated thus four fire scenarios result. For each fire scenario one ore more rooms are involved (cf. "Assignment of the Rooms to the Respective Fire Compartments").

At first, for each fire scenario it is checked if at least one route does exist which is not affected by the respective fire.

In the event of fire inside fire compartment C-III only room 3 is involved. As visible in Table 3, only route 3 is affected by the fire, thus six alternative routes exist.

Table 3 Fire scenario 1: fire in fire compartment C-III

Route	Start	Rooms along the route*										
1 C-III	7	→	4	→	2	→	1					
2 C-III	7	→	4	→	2	→	3	→	5	→	9	
3 C-III	7	→	4	→	2	→	5	→	9			
4 C-III	7	→	4	→	2	→	5	→	9			
5 C-III	7	→	4	→	5	→	2	→	1			
6 C-III	7	→	4	→	5	→	2	→	1			
7 C-III	7	→	4	→	5	→	3	→	2	→	1	
8 C-III	7	→	4	→	5	→	9					

* Arrows principally represent doors between the different rooms

The same principle of investigation is applied to all other fire scenarios as shown in Table 4 to Table 6. In the event of fire inside fire compartment C-IV only room 4 is involved. As visible in Table 4, no route exists which is not affected by a fire inside fire compartment C-IV. Therefore, the door between room 7 and room 4 does not meet the performance requirements of an exit.

Table 4 Fire scenario 2: fire in fire compartment C-IV

Route	Start	Rooms along the route*										
1 C-IV	7	→	4	→	2	→	1					
2 C-IV	7	→	4	→	2	→	3	→	5	→	9	
3 C-IV	7	→	4	→	2	→	5	→	9			
4 C-IV	7	→	4	→	2	→	5	→	9			
5 C-IV	7	→	4	→	5	→	2	→	1			
6 C-IV	7	→	4	→	5	→	2	→	1			
7 C-IV	7	→	4	→	5	→	3	→	2	→	1	
8 C-IV	7	→	4	→	5	→	9					

* Arrows principally represent doors between the different rooms

A fire inside fire compartment C-V shows the highest effect on the escape routes from room 7 as several rooms are involved by a single fire.

Table 5 Fire scenario 3: fire in fire compartment C-V

Route	Start	Rooms along the route*									
1 C-V	7	→	4	→	2	→	1				
2 C-V	7	→	4	→	2	→	3	→	5	→	9
3 C-V	7	→	4	→	2	→	5	→	9		
4 C-V	7	→	4	→	2	→	5	→	9		
5 C-V	7	→	4	→	5	→	2	→	1		
6 C-V	7	→	4	→	5	→	2	→	1		
7 C-V	7	→	4	→	5	→	3	→	2	→	1
8 C-V	7	→	4	→	5	→	9				

* Arrows principally represent doors between the different rooms

Table 6 Fire scenario 4: fire in fire compartment C-VI

Route	Start	Rooms along the route*									
1 C-VI	7	→	4	→	2	→	1				
2 C-VI	7	→	4	→	2	→	3	→	5	→	9
3 C-VI	7	→	4	→	2	→	5	→	9		
4 C-VI	7	→	4	→	2	→	5	→	9		
5 C-VI	7	→	4	→	5	→	2	→	1		
6 C-VI	7	→	4	→	5	→	2	→	1		
7 C-VI	7	→	4	→	5	→	3	→	2	→	1
8 C-VI	7	→	4	→	5	→	9				

* Arrows principally represent doors between the different rooms

FINAL ASSESSMENT

In case of fire scenario 1, at least one route exists which is entirely not affected by the fire. Therefore, a totally safe escape from room 7 is possible in case of this scenario. In consequence, this scenario does no longer need to be further investigated.

For all other fire scenarios, the escape route travel length from the remotest location of the assessed room towards the exit has to be measured (where the route become green as indicated in the tables as shown above). A simple conservative approach because the fire scenarios are independent, is the superposing of all different fire scenarios as shown in Table 7.

Table 7 Superposing of fire scenarios

Route	Start	Rooms along the route*									
1 C-VI	7	→	4	→	2	→	1				
2 C-VI	7	→	4	→	2	→	3	→	5	→	9
3 C-VI	7	→	4	→	2	→	5	→	9		
4 C-VI	7	→	4	→	2	→	5	→	9		
5 C-VI	7	→	4	→	5	→	2	→	1		
6 C-VI	7	→	4	→	5	→	2	→	1		
7 C-VI	7	→	4	→	5	→	3	→	2	→	1
8 C-VI	7	→	4	→	5	→	9				

* Arrows principally represent doors between the different rooms

It is obvious that the worst-case scenario is a fire inside fire compartment C-V (see Table 5) which dominates the result. Therefore, the exits are the doors between room 2 and room 1 (door A) and that between room 5 and room 9 (door I)

The escape route length consists of the length from the most distant location of the route inside the room to its particular room exit door (in this case it is only one single door) and the length from the room´s exit door along the route to the exit which is in the shortest distance.

REFERENCES

[1] Jäde, H. (Ed.) Musterbauordnung (MBO 2012): Textsynopse der Fassungen November 2002 und September 2012 mit Begründung (Sample Building Code of Germany), ISBN-13: 9783406651151, Verlag C. H. Beck, June 2013 (in German).

[2] Decree of the Ministry of the Environment on fire safety of buildings: E1 The National Building Code of Finland, Fire safety of buildings, Regulations and guidelines 2002; Helsinki, Finland, 12 March 2002, https://www.edilex.fi/data/rakentamismaaraykset/e1e.pdf.

[3] Code du travail - Article R4216-8 ; Créé par Décret n°2008-244 du 7 mars 2008 - art. (V), 2008 (in French), https://www.legifrance.gouv.fr/affichCodeArticle.do?cidTexte=LEGITEXT000006072050&idArticle=LEGIARTI000018488773&dateTexte=&categorieLien=cid.

[4] Fachkommission Bauaufsicht der Bauministerkonferenz: Muster-Richtlinie über den baulichen Brandschutz im Industriebau (Muster-Industriebau-Richtlinie - MIndBauRL) (German Building Code for Industrial Facilities), Stand Juli 2014, Amtliche Mitteilungen Nr. 1/08.09.2014, Deutsches Institut für Bautechnik (DiBT), Ausgabe 1 September 2014 (in German), https://www.dibt.de/fileadmin/dibt-website/Dokumente/Amtliche_Mitteilungen/2014_1.pdf.

4 Seminar Conclusions and Outlook

The 16th International Seminar on 'Fire Safety in Nuclear Power Plants and Installations' as well as the FSEP 2019 conference both clearly demonstrated ongoing progress in nuclear fire safety covering experimental and analytical research activities as well as enhancements in methods and tools and applications in the frame of safety assessments. However, there are still challenges since the knowledge on some, particular electrical fire and explosion related phenomena is not yet mature enough and the analytical tools applied need to be further advanced and extended.

The presentations given in the frame of this Seminar provided an added value to the state-of-the-art in nuclear fire safety highlighting recent developments, but also describing still unsolved issues in this area in an open manner.

The following conclusions have been drawn from the Seminar sessions and the final panel discussion:

Remarkable progress has been achieved with respect to fire safety in nuclear installations and its assessment. This is also reflected in the recent nuclear regulations, standards and guidance documents, internationally as well as in different countries worldwide. One aspect in this context is an adequate quality management system based on generic high-level criteria to address the necessary elements to ensure that the fire protection programs are implemented correctly and to verify the efficiency of a fire barrier with regard to the potential fire hazard conditions in a given location of the installation.

The Seminar has clearly indicated that the focus of fire safety for nuclear reactor facilities in those countries continuing to produce nuclear energy is currently more or less on the planning, construction and operational phase of nuclear power plants. In other countries with existing nuclear power stations close to the end of their operational lifetime various activities are ongoing on fire safety during the post-commercial safe shutdown and the decommissioning phases.

It has to be pointed out that fire safety is still an important issue, which has to be addressed at all types of nuclear facilities from their construction throughout their operational and post-operational lifetime until decommissioning has been finally completed. The provisions for a nuclear power plant under decommissioning need to include a procedure allowing to take suitable and reliable measures for fire prevention and mitigation.

These should not be limited to the individual reactor units but cover a nuclear site as a whole with all its collocated reactor units and other radioactive sources (spent fuel pools (SFPs) as well as dry storage facilities for fuel elements and/or other nuclear waste, but also nuclear waste treatment facilities. Such a comprehensive approach is called a site-level or multi-unit, multi-source assessment.

In some countries, new or advanced fire protection regulations and standards are being issued for deterministic as well as probabilistic safety assessments of operating nuclear power plants in order to identify significant improvements, e.g., for demonstrating that the capabilities required to safely shut down the reactor, to remove the residual heat and to contain the radioactive material are maintained in case of any single internal fire or to refine post-fire safe shutdown procedures to ensure an adequate fire response.

In case of newly designed nuclear power plants, particularly reactors of the fourth generation, it is expected that the optimization of the plant layout in order to reduce adverse effects from internal hazards and hazard combinations, such as fires or explosions, will take place at an early design stage. This involves the avoidance of large combustible inventories as well as a minimisation of the number of fire barrier penetrations (including doors, ventilation ducts, cabling, and pipework) in plant areas important to safety.

International operating experience from nuclear power plants has clearly indicated that electrical distribution equipment (e.g., breakers, switchgears, bus bars, bus ducts, etc.) can be subject to a failure mode that causes extensive damage known as HEAF. Investigations of event data from the international fire events database OECD FIRE in the recent past have underpinned that the vast majority of event combinations of fires and other anticipated events are HEAF induced fires as consequential events. The risk contribution of HEAF induced fire events and event combinations is non-negligible. One lesson learned is that the damage from HEAF fire events is even more severe in the presence of Aluminum. This underpins the need for more detailed in-depth investigations of HEAF fire events and their consideration in the fire protection regulations.

The already started second phase of the international experimental HEAF research program (HEAF 2) with a systematic test program covering another set of full-scale experiments is intended to provide insights on the thermal and mechanical damage caused by HEAF fire events to different types of targets, such as cables and electrical cabinets, for as much as practicable realistic component configurations representative for nuclear power plants in different countries.

Real scale international experimental series PRISME 3 carried out by IRSN in the frame of an OECD/NEA project on fire at electrical components such as cables and electrical cabinets shall help to close knowledge gaps with respect to fire development and spreading of electrical fires representing the most frequent fires in industrial facilities. More complex computational fluid dynamics (CFD) codes are more and more used in nuclear industry for modelling fire effects. The ability of models to adequately capture the fire behaviour in complex and confined room with mechanical ventilation is of key importance.

The common cable fire benchmark exercise of the PRISME 3 experimental project and the fire events database project FIRE could be an important step forward in modelling of real cable fires, needed not only for deterministic safety analyses but also for probabilistic fire risk assessment. In this context, it should be noted that the routing of cables on cable trays has a strong influence on fire propagation within various industrial buildings and power plants. Cables themselves represent a major fire source within these buildings, with a non-negligible fire load but also acting as an ignition source due to technical malfunction.

In order to comply with the nuclear safety goals, those safety functions, which need to be maintained during and after a fire, have to be identified for performing an adequate fire safety analysis. For that purpose, the safety goals have to be associated with key parameters and performance criteria. One of the key issues of such a fire safety analysis is the assessment of all relevant fire scenarios in the respective plant.

Another aspect again recognized by the Seminar participants to be relevant is the role of fire barrier elements (e.g., fire doors and dampers in the ventilation ducts) and their significance during all phases of the entire life cycle of a nuclear power plant. One important issue results from deviations between the original design and the real situation in the facility due to modifications of the environment during construction, addition of new equipment and materials in the room, improvements of fire protection means, etc. Suitable and not too complex analytical tools may support the analysts in conducting more realistic assessments.

Last but not least, the operating experience from nuclear power plant sites with already started or ongoing handling and treatment (e.g. drying) of partly combustible nuclear waste from reactors under decommissioning has shown new types of fire events which may occur more frequently in the future. These observations have indicated the need to adapt the existing nuclear fire protection for operating nuclear power plants to longer duration shutdown and decommissioning phases.

The participants from Northern America, Asia and Europe, representing the different parties involved in nuclear fire safety – nuclear industry as well regulatory bodies, research institutions and technical expert and support organizations (TSOs), emphasized the added value of and benefits from the information provided in this experts' seminar to be shared inside the nuclear fire community. They strongly expressed their wish of continuing this series of fire safety seminars on a regular basis in time intervals of approximately two years. The next, 17th Seminar of this series is therefore planned to be conducted in late summer 2021 in conjunction with the 26th 'International Conference on Structural Mechanics In Reactor Technology' (SMiRT 26), which will take place in Berlin, Germany in August 2021 (cf. http://www.smirt26.org/).

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